Application of the Technology Neutral Framework to Sodium Cooled Fast Reactors

by

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ABSTRACT

Sodium cooled fast reactors (SFRs) are considered as a novel example to exercise the Technology Neutral Framework (TNF) proposed in NUREG-1860. One reason for considering SFRs is that they have historically had a licensing problem due to postulated core disruptive accidents. Two SFR designs are considered, and both meet the goals of the TNF that LWRs typically would not. Considering these goals have been met, a method for improving economics is proposed where systems of low risk-importance are identified as candidates for removal, simplification, or removal from safety grade. Seismic risk dominates these designs and is found to be a limiting factor when applying the TNF.

The contributions of this thesis are the following:

1. Functional event trees are developed as a tool to allow different designs to be compared on an equal basis. Functional event trees are useful within the TNF as a method for the selection of Licensing Basis Events (LBEs) which take the place of traditional Design Basis Accidents.

2. A new importance measure, Limit Exceedance Factor (LEF), is introduced that measures the margin in system failure probability. It can be used directly with the TNF where standard importance measures cannot. It also reveals that some systems that appear to be of high risk-importance with standard importance measures may have significant margin.

3. The seismic risk dominates these designs. It is shown that even under optimistic assumptions, the goals of the TNF cannot be met by a typical reactor. The effect of seismic isolation to reduce the frequency of seismically initiated large releases is also analyzed and found to be insufficient to reach the goals of the TNF. A limit on initiating event frequency that is consistent with current practices is proposed.

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ACRONYMS

AEC – Atomic energy commission

ALMR – Advanced liquid metal reactor

AOO – Anticipated operational occurrence

CAFTA – Computer aided fault tree analysis

CDA – Core disruptive accident

CDF – Core damage frequency

CFR – Code of federal regulations

CRBR – Clinch river breeder reactor

DBA – Design Basis Accident

DOE – Department of Energy

DRACS – Direct removal air cooling system

EBR-II – Experimental breeder reactor two

EPA – Environmental protection agency

F-C Curve – Frequency consequence curve

FET – Functional Event Trees

FFTF – Fast flux test facility

FV – Fussel-Vesely

GDC – General Design Criteria

GE – General Electric

GEM – Gas expansion module

IHTS – Intermediate heat transport system

LBE – Licensing basis event

LOCA – Loss of coolant accident

LWR – Light water reactor

NRC – Nuclear Regulatory Commission

NGNP – Next generation nuclear plant

OBE – Operating basis earthquake

PCS – Plant control system

PGA – Peak ground acceleration

PRA – Probabilistic risk assessment

PRISM – Power reactor innovative small module

PSER – Pre-application safety evaluation report

PSID – Preliminary safety information document

PWR – Pressurized water reactor

RAW – Risk achievement worth

RPS – Reactor protection system

RRW – Risk reward worth

RVACS – Reactor vessel air cooling system

SAPHIRE - Systems analysis program for hands-on integrated reliability

SFR – Sodium-cooled fast reactor

SSE – Safe shutdown earthquake

TBS – Transition break size

TNF – Technology neutral framework (NUREG-1860)

QHOs – Quantitative health objectives

SUMMARY

I. INTRODUCTION

Sodium-cooled fast reactors (SFRs) are a mature reactor technology. Experimental and prototype SFRs have been constructed and operated in several countries for more than 60 years. They have been considered an option for both running a closed fuel cycle and actinide management for legacy waste in projects such as the Advanced Fuel Cycle Initiative and as part of the goal of Generation IV.

In this thesis, the focus is on pool type SFRs. This type of reactor design contains all of the primary sodium in the reactor vessel. An intermediate heat exchanger transfers heat from the primary sodium to a secondary heat transfer loop. This loop transports heat to the steam generator. Some of the important operating characteristics that set SFRs apart from LWRs are a higher operating temperature (~500 °C compared to ~300 °C), a long thermal response time, a large margin to coolant boiling, a low pressure primary system, no emergency electricity generators, and a positive void coefficient (Gyorey, Hardy, and McGee 1992).

Historically, SFRs have been considered to be more expensive per MWe than traditional light water reactors (LWRs). One of the reasons for this has been the poor capacity factor experience at some sites such as SuperPhenix and Monju. Another reason, and the focus of this thesis, is that the licensing process has been difficult for SFRs in the United States.

I.A Current Regulations

10CFR50, the current set of licensing regulations, is generally designed for use with LWRs. Many of the General Design Criteria (GDC) do not necessarily apply to SFRs or other advanced reactor designs. In NUREG-1368, the Preliminary Safety Evaluation Report (PSER) for the PRISM reactor, the NRC agreed with General Electric (GE) that, for the PRISM SFR, LOCAs are not an accident of concern. There was some disagreement on which of the GDCs might apply, but both parties agreed that many of them would require modification.

The current US SFR licensing knowledge has come about from the Clinch River Breeder Reactor (CRBR) and the Advanced Liquid Metal Reactor (ALMR) program interactions with the NRC. In the 1970s and early 1980s, licensing was initiated for the CRBR but funding was cut before a construction permit was issued. Core Disruptive Accidents (CDAs) have been a licensing issue for SFRs and particularly caused problems in the licensing of the Clinch River Breeder Reactor (CRBR). These accidents were originally postulated as a sudden voiding of the reactor core resulting in the insertion of a large amount of positive reactivity and causing energetic disassembly of the reactor core. This accident was not considered as a design basis but was a driving force causing the designers to change the core layout to reduce the energetics of such postulated events. Other preventative and mitigating measures were also included in the design to address CDAs (Strawbridge and Clare 1985).

Although core disruptive accidents (CDAs) were not considered as part of the design basis for the CRBR, a large amount of regulatory attention was given to these accidents which prolonged the licensing process (Ivans, 2006). The CRBR licensing process did result in a Safety Evaluation Report in 1983, NUREG-0968.

In order to avoid the regulatory delays associated with addressing CDAs, the ALMR design incorporated additional passive safety measures into the PRISM design including gas expansion modules and an ultimate shutdown system. Under the ALMR program, the DOE submitted a Preliminary Safety Information Document (PSID) to the NRC in 1986 and the NRC in turn issued a Pre-application Safety Evaluation Report (PSER) in 1994. As a result of this interaction, the NRC forced changes to the PRISM reactor including the adoption of a containment dome and the addition of an ultimate shutdown system and gas expansion modules (GEMs) to satisfy defense-in-depth concerns (Ivans, 2006).

II. TECHNOLOGY NEUTRAL FRAMEWORK OVERVIEW

The NRC Office of Research has published NUREG-1860, a study on the feasibility of a risk-informed and performance-based licensing framework. This framework is referred to in this thesis as the Technology Neutral Framework (TNF). It is important to note that it is not a regulation and will almost certainly be changed before final implementation, if it actually occurs. PBMR (Pty) Ltd. has proposed a risk-informed and performance-based licensing strategy to the NRC, though not specifically following NUREG-1860. The NRC has deferred any rulemaking until the pre-application reaches a further level of detail (US DOE 2008). Similarly, the NRC plans to test the TNF in parallel with a separate licensing structure for the Next Generation Nuclear Plant (NGNP) (or any other High Temperature Gas Reactor design certification or Combined License application) in which LWR licensing requirements will be modified.

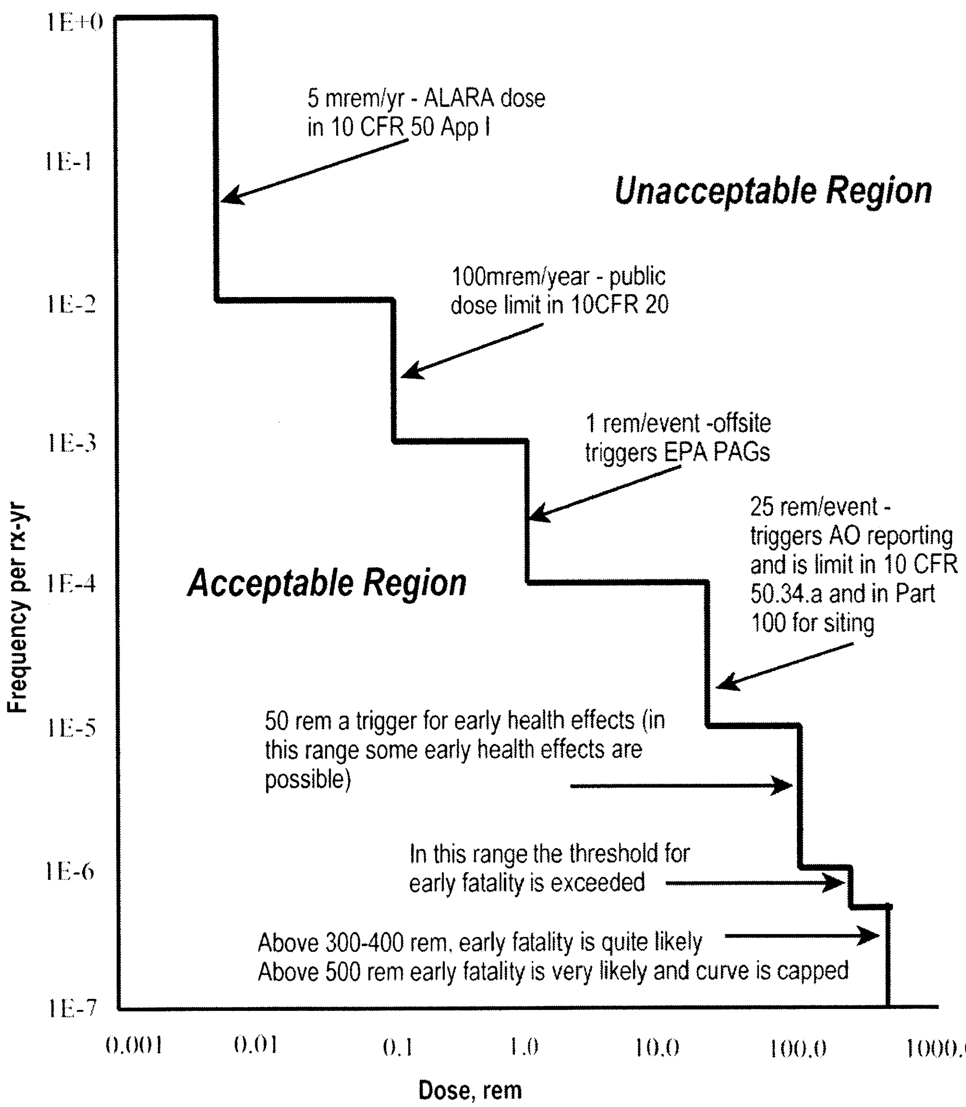


Figure 1. The F-C Curve

In the TNF, PRA methods play a central role in determining licensing basis events (LBEs) which take the place of traditional DBAs. Within the TNF approach, accident sequences are grouped according to similar phenomenology and consequences. The limits set in NUREG-1860 are on a per-LBE basis (Figure 1). LBEs with a high frequency of occurrence must have low consequences to be acceptable. LBEs with a low frequency are allowed to have higher consequences. All LBEs must lie below the frequency-consequence curve (F-C curve). This curve has been developed to be commensurate or more conservative than current NRC and EPA regulations. Appendix H of the TNF describes which portions of 10CFR50 are applicable within the framework.

The LBE representing a group of sequences is assigned the 95th percentile frequency of the most likely sequence in the group and the 95th percentile consequence from the worst (or most challenging) consequence sequence in the group. The systems whose performance is required to keep the LBEs below the F-C curve are categorized as safety related and must conform to the special treatment requirements of safety-related systems.

The lowest frequency considered, as seen in Figure 1, is 10^{-7} per reactor year. NUREG-1860 specifies that LBEs with a mean frequency less than 10^{-7} per reactor year "are screened from the process". In addition, the report advises to "Drop all PRA sequences with point estimate frequency < 1.E-8 per year."[i]

The TNF also includes deterministic requirements for the probabilistically selected LBEs. One example from these requirements is that a certain number of barriers remaining to fission product

[i] The notation 1.E-8 means 1×10^{-8}.

release must remain intact depending on the frequency of the sequence. For more frequent sequences more barriers must stay intact. Another example is that the core must maintain a coolable geometry for all sequences greater than 10^{-5} per year.

A deterministic requirement is that a deterministic LBE is to be negotiated between the licensee and the regulator. This event is to represent "a serious challenge to fission product retention in the fuel and coolant system." Finally, there are defense-in-depth guidelines regarding protective systems, stable operation, barrier integrity, and protective actions. These guidelines should lead to a balanced design with a high level of safety. The specific requirements are detailed in Appendix G of the TNF.

There is a parallel set of requirements within the TNF for security. However, there is not a great deal of guidance as to how it would be demonstrated that the quantitative health objectives (QHOs) are satisfied. Diversion scenarios are included as part of the security threat and are to meet the requirements of 10CFR73.

The primary change from 10CFR50 to the TNF is that LBEs, which are actual sequences determined for each plant design, take the place of postulated DBAs. DBAs are stylized accidents that do not take frequencies into account. A major feature is that there is a stated frequency cutoff for events that are within the licensing basis. This is an important point for SFRs as a designer may argue that CDAs lie far below this cutoff. This could potentially prevent excessive regulatory attention to these very rare events.

II.A. SFR PRA REVIEW

Using the TNF requires the availability of a PRA. The available PRAs for sodium reactors are for PRISM (Hackford 1986), EBR-II (Hill 1991), and ALMR (El-Sheikh 1994). ALMR and PRISM are both pool type reactors that are less than a megawatt thermal. The PRAs for PRISM and ALMR were performed by GE staff. The lack of detail of these documents reflects the fact that these designs are in the conceptual phase. The PRISM PRA is Level 3 while the ALMR PRA is only Level 1. The PRA for EBR-II is thorough and similar to other modern Level 1 PRAs. All of the PRAs include seismic initiators with only EBR-II including fire initiators. There are some features that all of the PRAs share in common. All of the reactors are assessed to have internally initiated core damage frequencies below typical GEN-II PWRs. Additionally, the risk for each of these designs is dominated by seismic initiators. This information is summarized in Table 2. Finally all of the PRAs include only point estimates for all events.

Table 2. Core Damage Frequency Summary

Design	Internal CDF (per year)	Seismic Contribution to CDF
EBR-II	$2x10^{-6}$	$2x10^{-5}$
PRISM	$1x10^{-8}$	$5x10^{-8}$
ALMR	$1x10^{-10}$	$3x10^{-6}$

II.A.1. PRISM PRA

Several of the failure probabilities in the PRISM PRA have been scrutinized by the NRC in their preapplication safety evaluation report (NUREG-1368). There are some differences in the PRA reviewed by the NRC and the PRA used in this report. The original 1986 draft is being used in this work; the NRC reviewed a modified version from 1989 that has some substantial design differences. In particular, scram failure frequencies on the order of 10^{-9} per demand are thought to be overly optimistic. Additionally, the pump coastdown failure frequency on the order of 10^{-9} per reactor year, as well as the scram signal failure frequency on the order of 10^{-11} per demand, have been questioned as being overly optimistic. Additionally, the initiating event frequency for reactivity insertions at 10^{-4} per year was observed to be approximately two orders of magnitude lower than typical LWR numbers. The decay heat removal system is a reactor vessel air cooling system (RVACS) that directly removes heat from the reactor vessel. RVACS has a failure probability around 10^{-8} depending on the initiator and the mission time. This failure probability is justified by GE saying that RVACS is continuously operated and monitored. This would allow operators to detect degradation before failure. There are no support systems necessary to operate RVACS.

While the PRISM PRA is the only Level 3 assessment, the core response and containment response trees are quite conservative. This can be attributed to several factors. First, the EBR-II loss-of-flow experiments had not been performed and the estimated likelihood of eutectic penetration in these transients is overestimated (assumed to be unity). In general, the performance characteristics of metal fuel were not well understood. Secondly, the containment/release response was originally done for oxide fuel and was modified in a conservative manner to represent metal fuel. A larger source term and faster release rate were used than for the oxide core.

II.A.2. ALMR PRA

The ALMR PRA shares many similarities with the PRISM PRA as far as optimistic failure frequencies are concerned. The RPS signal and rod insertion reliabilities are both fairly optimistic and are around 10^{-8} per demand. There is a significant difference with respect to seismic isolators and RVACS. The isolators are assumed to never fail in the ALMR PRA, while in PRISM they are assumed to fail at 1.2g PGA. RVACS is assumed to always work with the exception of large seismic initiators, but it has a fragility much greater than a typical PWR containment; given an earthquake >2g a failure probability of 10^{-5} is used for RVACS. This is significantly more reliable than the PRISM PRA which has a failure probability of 10^{-1} for RVACS given an initiating earthquake of 1.2g PGA. Given this original estimate and no reason that the isolators might be more robust, the failure probability used for RVACS in the ALMR PRA seems particularly optimistic.

II.A.3. EBR-II PRA

The most immediate difference in the EBR-II PRA is the level of detail associated with a real reactor. There are also some significant design differences between EBR-II and the GE designs. The EBR-II (65 MWth) reactor is much smaller than the PRISM (~500 MWth) or ALMR (~800 MWth) designs. Another major difference is that the pumps for EBR-II are centrifugal pumps. The PRA identified leakage of pump lubricant as a significant risk contributor. EBR-II is not seismically isolated. The primary failure mode is that the hangers holding up the vessel fail during a large (0.7g) seismic event. EBR-II also has a much more diverse set of internal events that are significant contributors to risk. Loss of flow, loss of offsite power, partial blockage, and large reactivity insertion each contribute between 14% and 32% of the internal CDF. In comparison, internal CDF is dominated in PRISM by loss of shutdown heat removal during a long shutdown transient at over 90% of the internal CDF. For ALMR the internal risk is dominated by a large reactivity insertion. If modified to include RVACS failure for internal initiators using the PRISM reliability estimates, ALMR is also dominated by loss of shutdown heat removal. For comparison to RVACS, the mean fragility of the passive heat removal system, which is the most robust component in EBR-II, is 1.5g PGA. This is somewhere between the PRISM estimate and the ALMR estimate.

III. FUNCTIONAL EVENT TREES

Functional event trees (FETs) have been developed in this work as a useful tool within the TNF. FETs are event trees where the top events are major safety functions rather than particular systems. There are several benefits to using functional trees as a tool. Different reactors can be compared on a similar basis by using the same set of initiating events and the same top events. The response of each design will be different. The frequency and consequences of each similar sequence for two different designs will be dictated by the systems that perform each function. Since a second PRA is to be constructed that contains only the safety grade systems, the FETs present a tool where this selection can be made and the difference between choices can be quickly calculated. LBEs are to be constructed by first binning together sequences with similar phenomenology. The LBE then has the phenomenology of the sequence in the bin that most challenges the system with the 95th percentile consequences (dose) of that sequence. The frequency of the LBE is the 95th percentile of the most frequent sequence in the bin. Due to both the binning of LBEs and the fact that not all systems need to be included in the PRA for determining LBEs, a functional event tree approach becomes quite attractive.

III.A. Functional Top Events

Functional event trees for SFRs consist of four top events plus an initiating event (these events may also be adequate to describe the safety functions of other reactor designs). The top events being considered are (in order they appear as top events):

1. Shutdown
2. Fuel heat removal
3. Primary heat removal
4. Late shutdown

Shutdown consists of the signal to scram as well as the actual function of the scram system. Because the results have similar phenomenology they may be binned for LBE purposes. Given a failure of shutdown, the following probabilities would be weighted by the relative likelihood of failure due to lack of signal and failure of insertion. For each given initiator this weighting is typically very strongly towards one or the other. This action basically adds the probability of both sequences, a conservative calculation. An alternative method is also explored where the failure probability associated with the most common initiator in the group is used to determine the top event failure probabilities.

Fuel heat removal may vary by design, but it is the ability to have enough flow past the fuel elements to remove heat. This may simply be pump coastdown for example. Primary heat removal is the ability to remove the heat from the primary system to the environment. This function may be carried out by a reactor vessel auxiliary cooling system, a direct removal auxiliary cooling system, or any other type of heat removal system. Late shutdown consists of functions designed to mitigate accidents that may not have initially shutdown. This may be something like an operator recovering scram ability later or an ultimate shutdown late in a transient overpower sequence where the first shutdown function failed

III.B. Initiating Events

Several initiators are considered based on the initiating events in the available PRAs:

1. Small reactivity insertion
2. Medium reactivity insertion
3. Large reactivity insertion
4. Local flow blockage
5. Loss of Offsite Power/Major Loss of flow
6. Shutdown Transients
7. Large Steam Generator Leak
8. Small seismic
9. Medium seismic

10. Large seismic

Some other possible initiators are of interest such as a large steam line break. This initiator is a major contributor to core damage in the EBR-II PRA but it is not considered in other reactor SFR PRAs. Additional external initiators such as flood and fire may be considered in future work.

Some initiators such as shutdown transients include all transients that force shutdown. This means that the sum of all frequencies of shutdown initiator should be used. The failure probabilities used for the top events is a weighted average of all the initiators grouped together. The phenomenology is considered to be the most challenging of those initiators, for example: pump trip is more challenging and would be used for phenomenology rather than spurious scram.

In an alternative method that is explored as more commensurate with the TNF, the most frequent initiator of those grouped together is chosen as the initiating frequency and, as described earlier, determines the failure probability of the top events.

In the design phase, FETs allow one to compare the performance of two different systems designed to perform the same safety functions. This system change assessment has been performed for the ALMR design where a DRACS has been put in place of the RVACS. This change turns out to be fairly minor, as both DRACS and RVACS are found to be approximately equal in reliability.

Although a more detailed PRA would be necessary to show compliance with the TNF, the initial results show that the SFRs tend to have few small release sequences. Figure 2 shows a map of where the LBEs for ALMR, PRISM, and an LWR fall on the F-C curve. Sequences on the left axis have no dose but are plotted at 0.001 rem for conveinience.

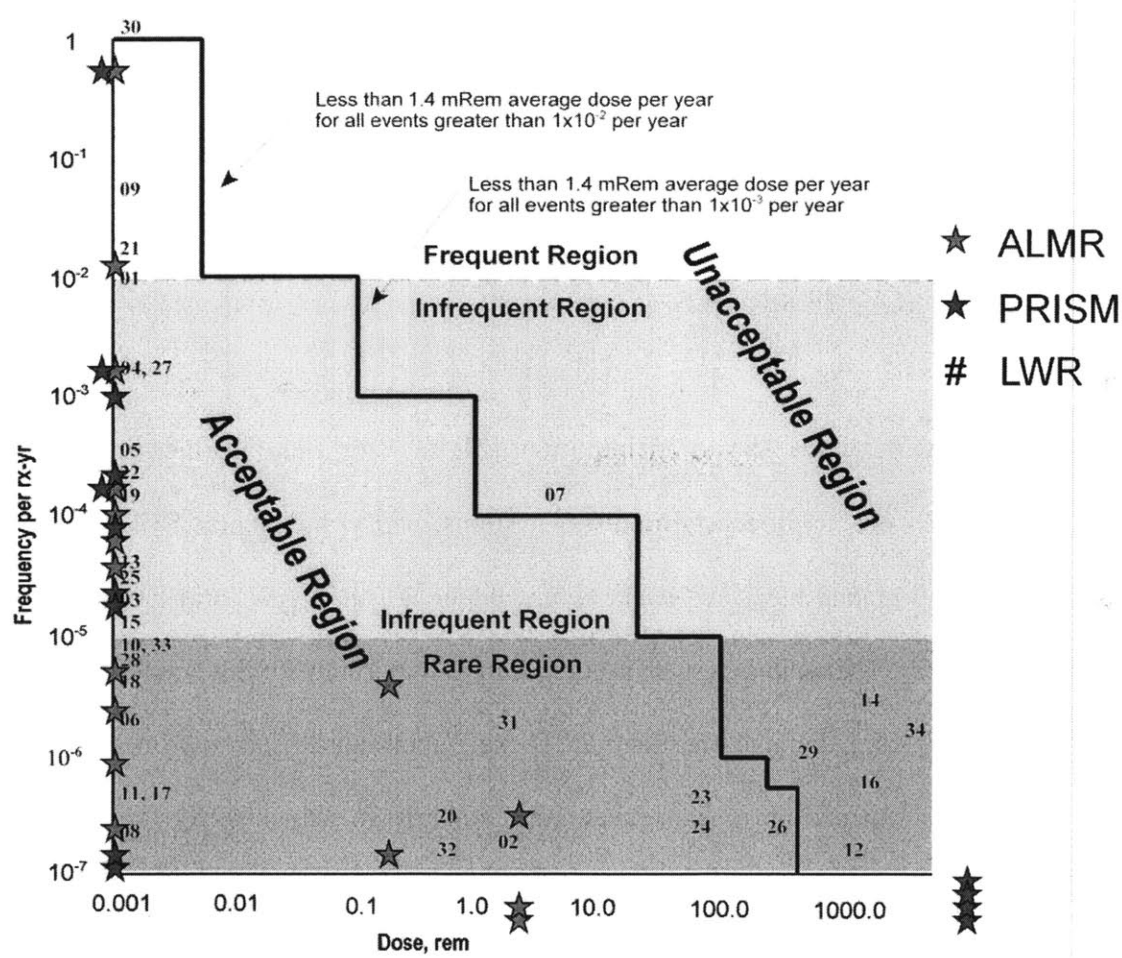


Figure 2. Plot of LBE location on the F-C Curve for several reactors.

When failure leads to a dose, it is rare, and usually very large. This is not very different from the pattern seen for the LWR, though the calculated frequencies are lower for the SFR. The dominant failure mode found in this review of PRAs is failure of decay heat removal during a long transient. Energetic events, which have been the focus of multiple sodium reactor designs, are found to occur only very rarely. The addition of a third scram system, fuel streaming channels, or other features meant to mitigate or prevent these rare events would have no impact with regards to meeting the goals of the TNF.

Seismic events are found to be a dominant contributor to the failure of the decay heat removal system. This is further investigated in Chapter V.

IV. IMPORTANCE MEASURES

Importance measures have played an important role in some risk-informed regulations such as 10CFR50.69. In this approach, structures, systems, and components (SSCs) are categorized based on the importance measures Risk Achievement Worth (RAW) and Fussell-Vesely (F-V). These measures are useful in determining the relative importance of SSCs with an integrated risk measure such as core damage frequency. These importance measures, on their own, do not provide useful information to designers with regards to which SSCs may be suited for simplification or improvement based on the margin between the calculated risk for the system design as is and a safety goal.

$RAW_{threshold}$ was developed by Reinert and Apostolakis to address this issue (Reinert and Apostolakis 2006). This method allows a designer to know the threshold value of RAW for a system such that if it were removed from the design, the safety goal would no longer be met.

Using this method and a surrogate risk metric composed of the sum frequency of all LBEs with a dose greater than 500 rem, it is found that pump coastdown and heat removal via the balance of plant are of low risk-significance to both the PRISM and ALMR designs.

A new importance measure, Limit Exceedance Factor (LEF), has been developed to better address the issue of measuring the margin available for each SSC as it relates to the overall system. LEF is defined as the factor the failure probability of a system can be multiplied by such that the total risk of the system is equal to a limit. LEF can be applied to the same traditional cumulative risk metrics that have been used for standard importance measures. More importantly, though, it has been developed to be compatible with LBEs and the TNF. It provides useful information where traditional importance measures cannot when risk is being measured on a per sequence basis.

The signal to scram and insertion of the rods were both identified as having a good deal of failure probability margin. This is in contrast to the high importance these systems appear to have when ranked using RAW. This comparison can easily be seen in tables 3 and 4. The three systems not shown in table 5 do not have a value for LEF. This is because these systems could have a failure probability of unity and the safety goals of the TNF would still be met.

It is interesting to note that LEF tends to rank major contributors more highly than RAW. This is because of the less extreme nature of the importance measure. To assume that a system will be in the failed state is a very rough way to make importance calculations, especially for systems such as the signal to scram and the insertion of the rods.

Table 3. RAW values for top level systems in the PRISM reactor design. Ordered from most important to least important.

System	RAW (Energetic Release)
Reactor Shutdown System (scram)	5.8E7
Reactor Vessel Air Cooling System (RVACS)	1.8E5
Shutdown Heat Removal through the Intermediate Heat Transfer System	1.4E4
Reactor Protection System /Plant Control System Signal	1.1E3
Pump Costdown	4.0
Nominal Inherent Reactivity Feedback	4.0
Operating Power Heat Removal	1.0

Table 4. LEF values for systems where the RAW value is greater than $RAW_{threshold}$. Ordered from most important to least important.

System	LEF (Energetic Release)
Shutdown Heat Removal through Intermediate Heat Transfer System	240
Reactor Vessel Air Cooling System (RVACS)	250
Reactor Shutdown System (scram)	460
Reactor Protection System /Plant Control System Signal	2.5E8

V. SEISMIC RISK WITHIN THE TNF

Seismic risk dominates the total risk in all of the SFR PRAs reviewed. Seismic requirements for nuclear power plants within the United States have historically been the result of a negotiation between the applicant and the U.S. Nuclear Regulatory Commission (NRC). The NRC and the applicant rely on engineering judgment and knowledge of seismic activity at the site to determine both the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE).

The OBE according to 10CFR50 Appendix S is "the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional. The operating basis earthquake ground motion is only associated with plant shutdown and inspection unless specifically selected by the applicant as a design input."

The SSE is "the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional." The SSE is to be at least 0.1g peak ground acceleration (PGA) and the OBE is typically one third of the SSE acceleration. Typical PGA values of SSEs for plants east of the Rocky Mountains range from 0.1g to 0.25g and the PGA values for plants west of the Rocky Mountains range from 0.25g to 0.75g. For comparison, the Niigataken Chuetsu-Okiearthquake was measured at the Kashiwazaki-Kariwa site in Japan to have a PGA between 0.69g~0.83g.

Under the TNF approach all sequences with a mean frequency of 10^{-7} per year or more must be fully evaluated and must meet the F-C curve. This presents a problem in itself as most reactor sites have not quantified the seismic hazard to such low probabilities. Most SSEs have an annual probability of exceedance of around 10^{-4} to 10^{-5}. With no evidence found that there is a physical upper bound to peak ground accelerations (PGA), an unbounded extrapolation of an existing hazard curve is used to demonstrate the methods and the potential problems associated with this type of treatment.

Some methods for seismic initiating event selection are proposed in this work as there is no specific guidance within the TNF. A conservative method where the initiators are selected from points above the seismic hazard curve and a method where initiators are selected as points on the hazard curve are analyzed. An example of this method is shown in Figure 3. In either method, the exceedance frequency is used as an initiating frequency, and the associated acceleration is used as the initiating acceleration.

Due to the very high level of conservatism involved in choosing points above the hazard curve, points on the curve are used. For example, these points may correspond to where the dotted lines intersect the hazard curve. Additionally this allows us to plot the effect of choosing any given point along the curve as the seismic initiator. Apart from the FETs, a short seismic event tree (Figure 4) is used to demonstrate the implications of including seismic initiators at frequencies as low as 10^{-7} per year.

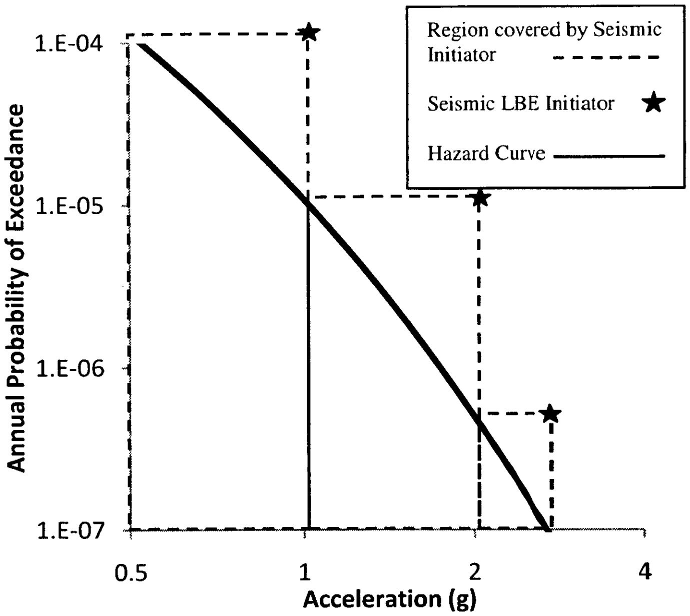


Figure 3. Schematic of LBE initiator selection. The stars show the initiating event frequency and acceleration chosen when conservatively using a point above the hazard curve. The dashed lines represent the limiting frequency and acceleration for the interval the initiating event is supposed to cover. Initiating events may also be selected from the curve itself. For example, points where the dashed line intersects the hazard curve may be used.

In this small event tree model, it is assumed that plant failure (using a overall plant fragility) and failure of containment is sufficient to have a release that exceeds 500 rem as the 95th percentile consequence. The results of this model with a typical LWR fragility are shown in Figure 4. The solid line is the extrapolated hazard curve from the Clinton site. The dashed lines show the large release frequency given either a hardened containment or a confinement building.

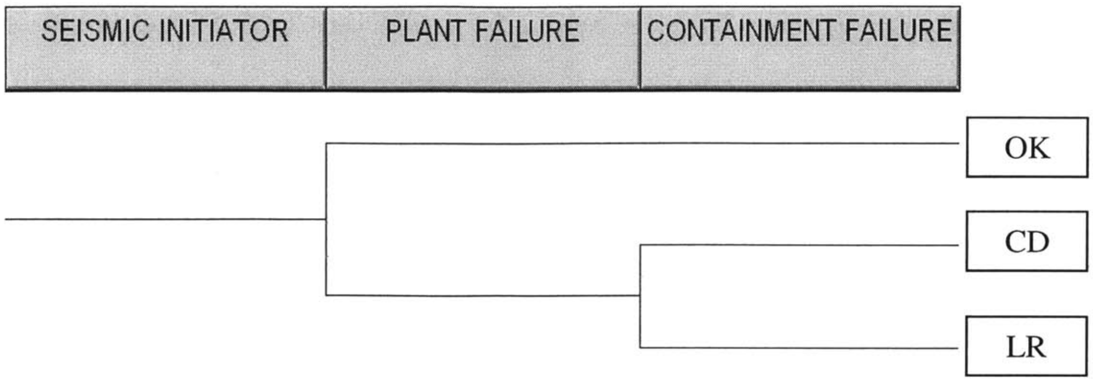


Figure 4. A simple event tree for seismic risk assessment. End states are OK, core damage (CD), and large release (LR).

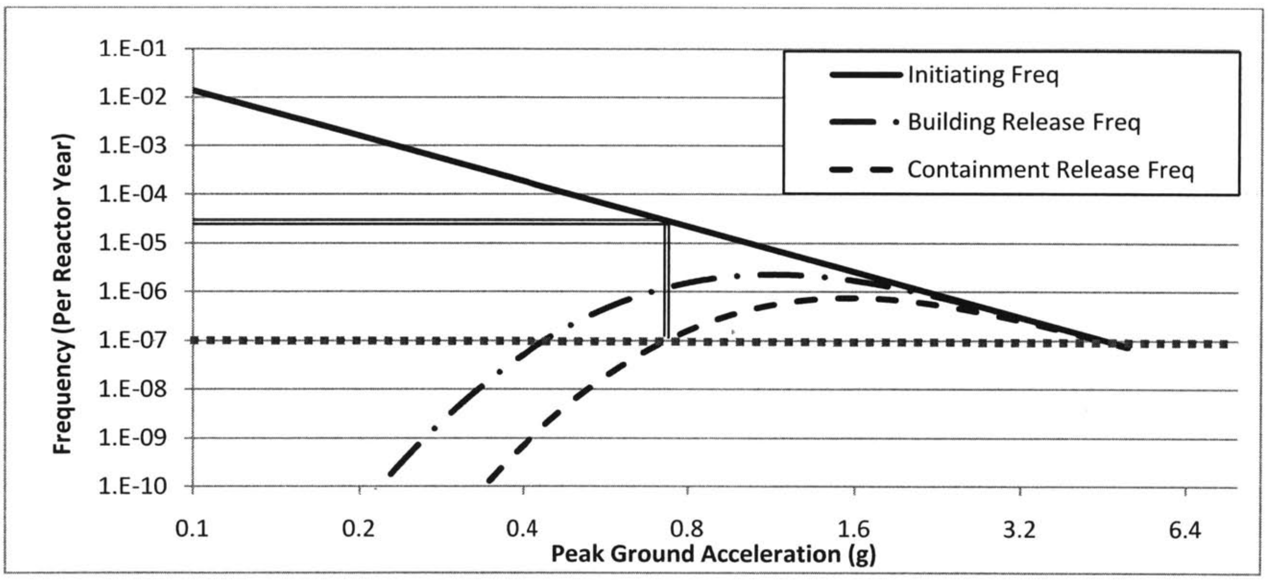


Figure 5. Peak Ground Acceleration vs. Frequency. The dotted horizontal line highlights the 10^{-7} per year cutoff of the TNF. The thin double line shows the frequency cutoff that would allow a plant with containment to be licensed.

The least frequent seismic initiator that can be considered while still meeting the TNF goal is shown by the double thin line. It shows the intersection of the 10^{-7} per year goal and the large release frequency for a reactor with containment. This is then extended to the hazard curve vertically to find the associated initiating frequency. It is observed that this is in the range of typical SSE frequencies of around 10^{-4} to 10^{-5} per year. The optimism used in the ALMR and

PRISM seismic risk analysis has been previously noted. If it is instead assumed that the SFR has a fragility equal to the decay heat removal component in the EBR-II seismic risk analysis, the same short event tree method can be used to show the results for an SFR. It turns out that the decay heat removal system is the least fragile component in the EBR-II PRA. The system has only a slightly higher fragility than that of a typical LWR. Considering this property and that SFRs are also vulnerable to seismic reactivity insertions, an SFR should have about the same level of plant fragility as a typical LWR. Since the analysis done is not of sufficient detail to do a better assessment, it is noted that the analysis done applies to both typical LWRs and SFRs.

One major feature of both the ALMR and PRISM designs is seismic isolation. This system is designed to prevent damage to the nuclear island. To model what the effect might be on the large release frequency, a fragility curve has been assigned to the isolation system based on the maximum capacity of 1.2g peak ground acceleration cited in the PRISM PRA. The basis for this limit is not immediately clear. As such analysis has been done using 1.2g as the 10^{th}, median, and 90^{th} percentile values of a lognormal distribution. Figure 6 illustrates this effect for a reactor with a confinement building and isolation. It is assumed that the isolation must fail for the plant to fail and that failure of the plant, confinement, and isolation are independent events. It is observed that for the most optimistic estimate isolation with confinement has a similar large release frequency as a non-isolated building with a hardened containment.

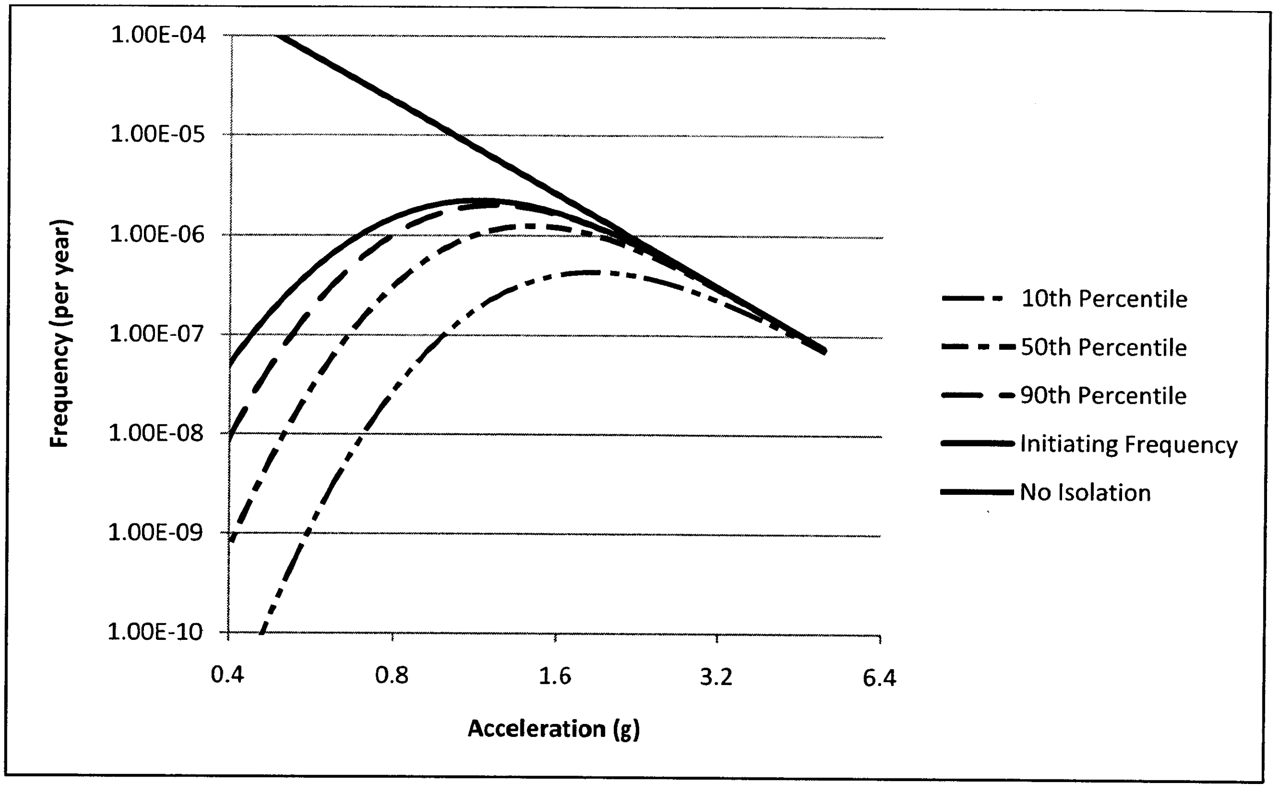


Figure 6. The frequency of large release for a typical reactor with isolation and a confinement building. Each curve represents a different calculation of the isolation system fragility curve. The fragility for the isolation is defined by assigning 1.2g to the percentile failure shown in the legend and β=0.5.

A solution that is not immediately compatible with the TNF is to treat seismic sequences differently than internally initiated sequences. This would be done be using an initiating event frequency limit of 10^{-5} per year. This value is chosen primarily due to the precedence of current practices. It was noted previously, that SSEs have a frequency of about 10^{-5} per year. In addition to this, a previous version of the TNF binned sequences according to their frequency. "Rare" initiators with a frequency of less than 10^{-5} per year were not necessarily required to meet any further mitigation. Finally, the rule being considered on the transition break size (TBS) sets the size based on which pipes fail with a frequency less than 10^{-5} per year. Those breaks rarer than this would be beyond design basis accidents and would not have the same requirements as breaks smaller than the TBS. This cutoff would make typical plants able to meet the TNF goals and would allow for the tradeoff between isolation and hardened containment to be analyzed.

VI. CONCLUSIONS AND RECOMMENDATIONS

In implementing the TNF with existing SFR PRAs, several key conclusions have been reached:

- The review of the ALMR and PRISM PRAs revealed that several of the failure probabilities are quite optimistic, i.e., they are too low.
- As completed, the ALMR and PRISM PRAs show these designs comply with the TNF for internal events (PWRs do not comply). Even if PWR numbers are used for scram and pump seizure, this result is unchanged.
- A more detailed and realistic PRA that includes fire (and other) initiators is necessary before it is attempted to satisfy the TNF.
- A prescriptive approach to LBE construction such as functional event trees is a useful development to prevent applicants from arbitrarily splitting sequences to arrive at apparently lower frequencies.
- Although core disruptive accidents are found to be well below the TNF cutoff, there is still a concern that this accident could be used as the deterministic LBE, thus negating the benefit of the very low frequencies associated with these accidents.
- Traditional risk metrics are not compatible with LBEs.
- Limit exceedance factor is a new measure designed to be used with the TNF or other quantitative risk metrics and reveals that some systems thought to be important using

other importance measures may have a significant amount of margin. Signal to scram and pump coastdown are two examples of systems that would traditionally be considered of high risk-significance that LEF identifies as candidates for simplification.

- Typical designs of SFRs cannot meet the TNF due to the requirement of including sequences initiated by very rare earthquakes. This conclusion also holds for PWRs.
- Seismologists do not quantify seismic risk to very low frequencies. Extrapolation reveals that these rare accelerations seismic events may have huge accelerations.
- It is recommended that a frequency cutoff be established for external events, as these events may pose a significant threat to all systems and may not be practical to design against.

CHAPTER I – INTRODUCTION

Sodium cooled fast reactors (SFRs) are a mature reactor technology. Experimental and prototype SFRs have been constructed and operated in several countries for more than 60 years. Some of the notable reactors that have operated or operating are EBR-II, FFTF, Phenix, Superphenix, BN-350, BN-600, Joyo, and Monju. Several reactors are planned for construction, mostly outside of the United States with the exception of the Toshiba 4S. They have been considered an option for both running a closed fuel cycle and actinide management for legacy waste (AFCI/GEN IV).

Some important operating characteristics set SFRs apart from LWRs. A higher operating temperature (~500 °C compared to ~300 °C) gives the SFR a higher thermal efficiency. A long thermal response time gives operators and safety systems more time to perform corrective actions. A large margin to coolant boiling means that boiling crises are not of importance. A low pressure primary system means less stress on the piping and that leaks may not have an immediate pressurizing effect within the containment structure. No emergency electricity generators are needed as the passive response and decay heat removal are sufficient to survive loss of offsite power transients. A positive void coefficient is one of the main safety issues that SFRs must face that LWRs do not. Postulated accidents, often initiated through some sort of postulated coolant voiding, called core disruptive accidents (CDAs) have been considered as something SFRs should be able to prevent, or respond to (GIF 2002). In a CDA, a large amount of energy may be released which would not only cause major damage to the core but would also pose a serious threat to the primary boundary.

In this thesis, the focus is on pool type SFRs. In the pool type SFR design, the primary sodium is contained within the reactor vessel. An intermediate heat exchanger, which sits in the pool of primary sodium, takes the heat into the secondary sodium loop. This prevents radioactive sodium from interacting with water. The secondary sodium is then sent to the steam generator to produce steam for the power cycle Figure I.1.

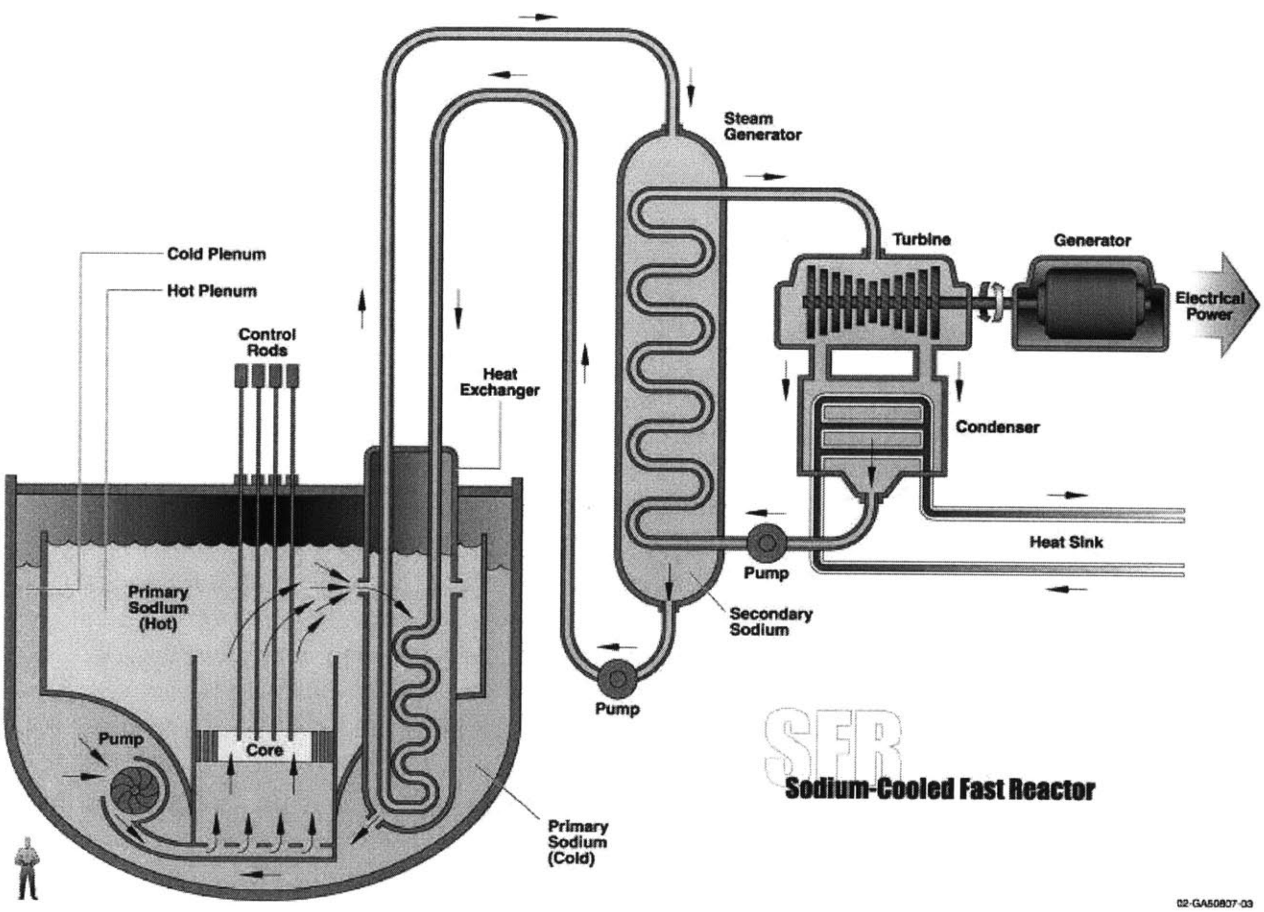


Figure I.1. Pool type SFR (US DOE 2002).

In particular there is a focus on the ALMR and PRISM conceptual designs. General Electric developed both of these designs to a fairly detailed level. They share many characteristics. Both rectors are metallic fueled, operate with a core outlet temperature of 510 °C, utilize electromagnetic pumps, and rely on passive features to respond to most transients. They both use a Reactor Vessel Air Cooling System (RVACS) to remove decay heat. PRISM is a smaller design at less than 500 MWth. ALMR is larger, coming in at about 800 MWth.

Some major differences in the designs considered came from the Nuclear Regulatory Commission (NRC) review of the PRISM Preliminary Safety Information Document. PRISM was faulted for having only a single method to shutdown the reactor as the reactivity feedback that shuts the reactor down as the temperature rises was not considered to be a sufficient secondary shutdown mechanism. Additionally, the PRISM design has no containment, which the NRC deemed unacceptable. As a result, the ALMR includes Gas Expansion Modules (GEMs) which shutdown the reactor when there is a loss of flow by increasing the neutron leakage, a secondary ultimate shutdown system that drops neutron absorbing balls into the reactor, and a small metal containment dome that is attached to the containment vessel. These added safety features have some operational drawbacks. Further discussion is beyond the scope of the thesis.

Historically, SFRs have been considered to be more expensive per MWe than traditional light water reactors (LWRs). This has been primarily due to the poor capacity factor experiences at some sites such as SuperPhenix and Monju. Other reasons for the higher cost include more expensive components and difficult inspection, as sodium is opaque. Experimental reactors, which are known to be more expensive than commercial reactors, are also the primary experience for SFRs. Another major reason in the United States, and the focus of this thesis, is that the licensing process has been difficult for SFRs. This resulted most notably in the huge cost overrun of the Clinch River Breeder Reactor (CRBR) project.

I.A Current Regulations

10CFR50, the current set of licensing regulations, is generally designed for use with LWRs. Many of the General Design Criteria (GDC) do not necessarily apply to SFRs or other advanced reactor designs. In NUREG-1368, the Preliminary Safety Evaluation Report (PSER) for the PRISM reactor, the NRC agreed with General Electric (GE) that, for the PRISM SFR, LOCAs are not an accident of concern. There was some disagreement on which of the GDCs might apply, but both parties agreed that many of them would have required modification. These differences are noted in Table I.A.1. One important takeaway is that many of the GDCs either need to be modified for SFRs or do not apply. The other important takeaway is that for many of the GDCs, the applicant and the regulator do not agree on how they should be modified. This type of uncertainty in regulation makes design a difficult process.

Table I.A.2. GDC Applicability to PRISM Design (USNRC 1994).

GDC Categories	Staff Evaluations by GDC Number	Preapplicant Proposal by GDC Number
GDC directly applicable	1, 2, 3, 5, 10, 11, 12, 13, 14, 16[#], 18, 20, 21, 22, 24, 29, 30, 32, 42, 43, 52, 53, 54, 56, 60, 62, and 63	1, 2, 3, 5, 10, 11, 12, 13, 14, 15, 16, 17, 18, 20, 21, 22, 23, 24, 25, 26, 29, 30, 31, 32, 34, 38, 40, 52, 53, 54, 56, 60, 61, 62, 63, 64
GDC applicable but needing changes	4, 15, 17, 19, 23, 25, 26[#], 27, 28, 31, 34[#], 36, 37, 38, 39, 40, 41, 44, 45, 46, 50, 51, 55, 57, 61, and 64	4, 19, 27, 28, 39, 50, and 51
GDC not applicable	33[*] and 35	33, 35, 36, 37, 41, 42, 43, 44, 45, 46, 55, and 57
Possible additional criteria	Sections 3.2.4.1, 3.2.4.2, and 3.2.4.6	None
GDC for which the NRC staff agrees with the preapplicant	1, 2, 3, 5, 10, 11, 12, 13, 14, 16[#], 18, 20, 21, 22, 24, 29, 30, 32, 35, 39, 51, 52, 53, 54, 56, 60, 62, and 63	
GDC for which the NRC staff requests the preapplicant to address changes to its position on the GDC	4, 15, 17, 19, 23, 25, 26[#], 27, 28, 31, 33[*], 34[#], 36, 37, 38, 40, 41, 42, 43, 44, 45, 46, 50, 55, 57, 61, and 64	

[*] - An alternative to GDC 33 is discussed under that GDC.
[#] - The NRC staff position on GDC 16, 26, and 34 may be changed by the Commission, see Sections on those GDC.

In LWR licensing, best engineering judgment and defense-in-depth play central roles in guiding regulations. There are no specific frequencies mentioned in the regulations but there are some guidelines. For example, anticipated operational occurrences (AOOs) are expected to happen at least once during the life of the plant. These include frequent events such as turbine trip and loss of offsite power. In addition, design basis accidents include an anticipated operational occurrence as well as a single failure of a major safety system. The single failure is also implicitly limited in frequency as failure of the pressure vessel is not considered as part of the design basis and the frequency is typically considered to be less than 10^{-6} and is estimated to be around 10^{-8} per year. On the other hand, a break in the largest pipe in the system is considered a design basis accident and is expected to happen only marginally more frequently than vessel failure.

The current US SFR licensing knowledge has come about from the Clinch River Breeder Reactor (CRBR) and the Advanced Liquid Metal Reactor (ALMR) program interactions with the NRC. In the 1970s and early 1980s, licensing was initiated for the CRBR but funding was cut before a construction permit was issued due to major cost overruns. Core Disruptive Accidents (CDAs) have been a licensing issue for SFRs and particularly caused problems in the licensing of the Clinch River Breeder Reactor (CRBR). These accidents were originally postulated as a sudden voiding of the reactor core. This inserts a large amount of positive reactivity and can cause energetic disassembly of the reactor core. This accident was not considered as a design basis accident but was a driving force causing the designers to change the core layout to reduce the energetics of such postulated events. Other preventative and mitigating measures were also included in the design to address CDAs (Strawbridge and Clare 1985).

Although core disruptive accidents (CDAs) were not considered as part of the design basis for CRBR, a large amount of regulatory attention was given to these accidents which prolonged the licensing process (Ivans, 2006). The CRBR licensing process did result in a Safety Evaluation Report in 1983, NUREG-0968.

In an attempt to avoid the regulatory delays associated with addressing CDAs, the ALMR design incorporated additional passive safety measures in an evolution of the PRISM design including GEMs and an ultimate shutdown system. Under the ALMR program, the DOE submitted a Preliminary Safety Information Document (PSID) to the NRC in 1986 (the PRISM design) and the NRC in turn issued a Preliminary Safety Evaluation Report (PSER) in 1994. As a result of this interaction, the NRC forced changes to the PRISM reactor including the adoption of a containment dome and the addition of an ultimate shutdown system and gas expansion modules (GEMs) to satisfy defense-in-depth concerns (Ivans, 2006).

CHAPTER II – THE TECHNOLOGY NEUTRAL FRAMEWORK

The NRC Office of Research has published NUREG-1860, a study on the feasibility of a risk-informed performance-based licensing framework. This framework is referred to in this thesis as the Technology Neutral Framework (TNF). It is important to note that it is not a regulation and will almost certainly be changed before final implementation. PBMR (Pty) Ltd. has proposed a risk-informed performance based licensing strategy to the NRC, though not specifically NUREG-1860. The NRC has deferred any rulemaking until the pre-application reaches a further level of detail. Similarly, the NRC plans to test the TNF in parallel with a separate licensing structure for the Next Generation Nuclear Plant (NGNP) (or any other High Temperature Gas Reactor design certification or Combined License application) in which LWR licensing requirements will be modified. It is noted that this modification of LWR requirements may be just as difficult and burdensome for NGNP as it was for SFR designs. If the TNF were to be used, there would be more regulatory certainty about what performance would be adequate. This would lead to LWRs having less of an advantage merely due to historical licensing procedures.

In the TNF, PRA methods play a central role in determining licensing basis events (LBEs) which take the place of traditional DBAs. Within the TNF approach, accident sequences are grouped according to similar phenomenology and consequences. The limits set in NUREG-1860 are on a per-LBE basis (Figure II.1).

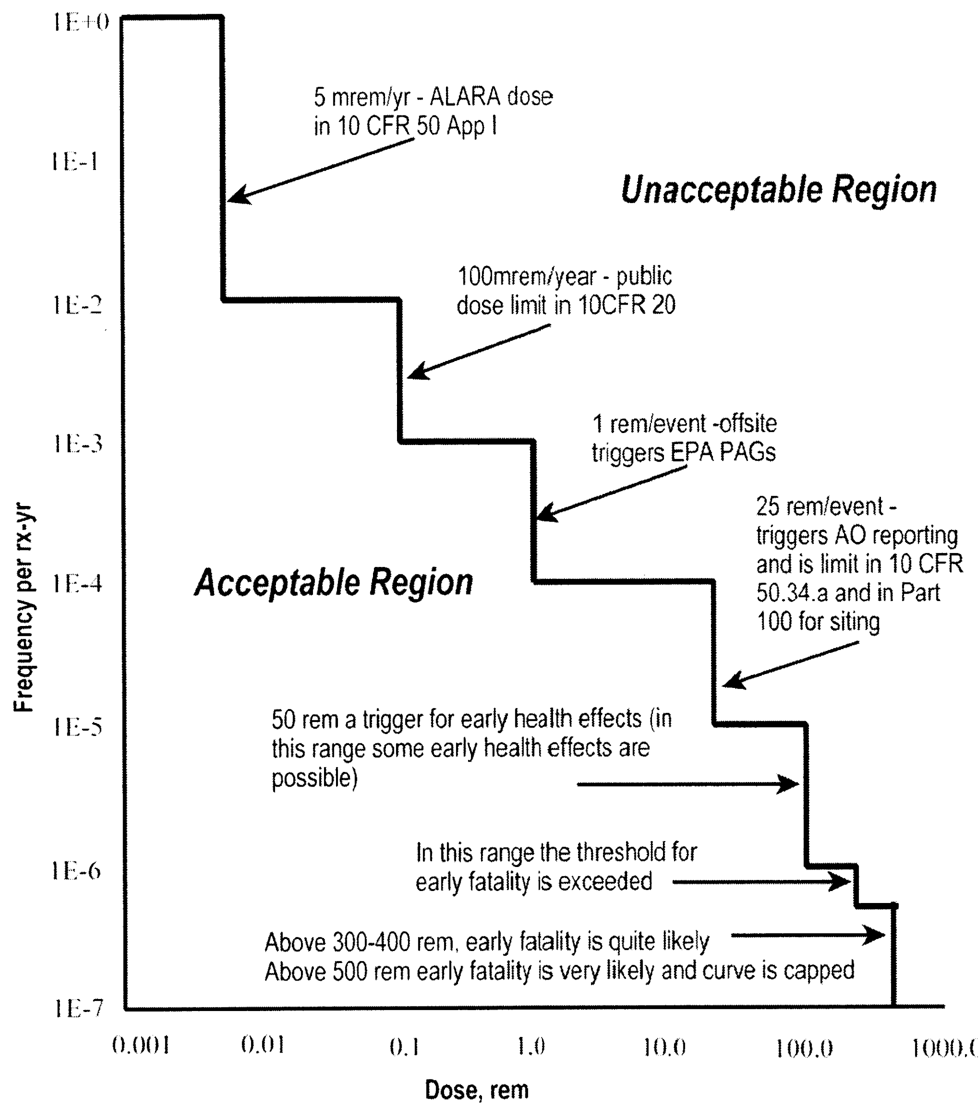


Figure II.1. The F-C Curve (USNRC 2007).

LBEs with a high frequency of occurrence must have low consequences to be acceptable. LBEs with a low frequency are allowed to have higher consequences. All sequences within the PRA and all LBEs must lie below the frequency-consequence curve (F-C curve). This curve has been developed to be commensurate or more conservative than current NRC and EPA regulations.

Appendix H of the TNF describes which portions of 10CFR50 are applicable within the framework.

The process for LBE selection described in Chapter 6 of the TNF as follows:

1. Modify the PRA to credit only those mitigating functions that are considered to be safety significant.

2. Determine the point estimate frequency for each resulting event sequence from the quantification of the modified PRA.

3. For sequences with point estimate frequencies equal to or greater than $1x10^{-8}$ per year, determine the mean and 95th percentile frequency.

4. Identify all PRA event sequences with a [mean or] 95th percentile frequency > $1x10^{-7}$ per year. Event sequences with [mean or][ii] 95th percentile frequencies less than $1x10^{-7}$ per year are excluded from further consideration.

5. Group the PRA event sequences with a 95th frequency percentile > $1x10^{-7}$ per year into event classes.

6. Select an event sequence from the event class that represents the bounding consequence.

7. Establish the LBE's frequency for a given event class.

8. Bin each LBE into one of three frequencies ranges: Frequent, Infrequent or Rare.

9. Determine the total weighted annual frequencies for all events equal to or greater than $1x10^{-2}$ and $1x10^{-3}$.

10. Verify that the selected LBEs meet the deterministic and probabilistic acceptance criteria.

[ii] There is an apparent inconsistency in NUREG-1860. In the main body of the report, a sequence with a mean frequency of less than 10^{-7} per year is not included in the licensing basis. However, in an appendix to the report, sequences are not included in the licensing basis if the 95th percentile frequency is less than 10^{-7} per year. One of the authors of the report, J. Lehner of Brookhaven National Laboratory, was consulted and he confirmed that the 10^{-7} cutoff is to apply to mean values.

The LBE representing a group of sequences is assigned the 95th percentile frequency of the most likely sequence in the group (step 7) and the 95th percentile consequence from the worst (or most challenging) consequence sequence in the group (step 6). The systems whose performance is required to keep the LBEs below the F-C curve are categorized as safety related and must conform to the special treatment requirements of safety-related systems.

The lowest frequency considered, as seen in Figure II.1, is 10^{-7} per reactor year. NUREG-1860 specifies that LBEs with a mean frequency less than 10^{-7} per reactor year "are screened from the process". In addition, the report advises to "Drop all PRA sequences with point estimate frequency < 1.E-8 per year. (USNRC 2007)"[iii]

The TNF also includes deterministic requirements for the probabilistically selected LBEs. One example from these requirements is that a certain number of barriers remaining to fission product release must remain intact depending on the frequency of the sequence. The table for deterministic requirements is reproduced here as Table II.1. Further deterministic criteria are detailed in Appendix G and Appendix J of the TNF. These are with regards to the specific defense-in-depth criteria mentioned in the TNF: physical protection, stable operation, protective systems, barrier integrity, and protective actions.

One notable element of the defense-in-depth requirements is that a traditional LWR containment is not imposed, and that another event may pose the design basis for a controlled low-leakage barrier. Events mentioned in the TNF for SFRs are: flow blockage in the core, large sodium fire,

[iii] The notation 1.E-8 means 1×10^{-8}.

and loss of normal heat removal in conjunction with poor quality fuel. This final event (loss of heat removal) is a major contributor to large release as analyzed in Chapter 3.

Table II.1. Deterministic criteria for probabilistic LBEs (USNRC 2007)

Frequency Category	Additional acceptance criteria for LBEs (demonstrated with calculations at the 95% probability value* with success criteria that meet adequate regulatory margin, as discussed in Section 6.6)
frequent ($\geq 10^{-2}$)	• no barrier failure (beyond the initiating event) • no impact on fuel integrity or lifetime and safety analysis assumptions • redundant means for reactor shutdown and decay heat removal remain functional • annual dose to a receptor at the EAB $\leq$ 5mrem TEDE
infrequent ($< 10^{-2}$ to $\geq 10^{-5}$)	• maintain containment functional capability • a coolable geometry is maintained • at least one means of reactor shutdown and decay heat removal remains functional • for LBEs with frequency > 1E-3 annual dose to a receptor at the EAB $\leq$ 100mrem TEDE • for LBEs with frequency < 1E-3 the worst two-hour dose at the EAB, and the dose from the duration of the accident at the outer boundary of the LPZ, meet the F-C curve
rare ($<10^{-5}$ to $\geq 10^{-7}$)	• the worst two-hour dose at the EAB, and the dose from the duration of the accident at the outer boundary of the LPZ, meet the F-C curve

A final deterministic requirement is that a deterministic LBE is to be negotiated between the licensee and the regulator. This event is to represent "a serious challenge to fission product retention in the fuel and coolant system."

There is a parallel set of requirements within the TNF for security. These goals are set such that different qualitative threats of high, medium, and low are assigned probabilities of occurrence. The applicant is then supposed to show that the conditional threat for these groups meets the quantitative health objectives (QHOs). However, there is not a great deal of guidance as to how this should be accomplished. Diversion scenarios are included as part of the security threat and are to meet the requirements of 10CFR73.

The primary change from 10CFR50 to the TNF that this thesis focuses on is that LBEs, which are actual sequences determined for each plant design, take the place of postulated DBAs. DBAs do not take frequencies into account. A major feature is that there is a stated frequency cutoff for events that are within the licensing basis. This is an important point for SFRs as a designer may argue that CDAs lie far below this cutoff. This could potentially prevent excessive regulatory attention to these very rare events.

II.A. Review of the Available PRAs

Using the TNF requires PRA availability. The available PRAs for pool type SFRs are for PRISM (Hackford 1986), EBR-II (Hill, Ragland, and Roglans-Ribas 1991), and ALMR (El-Sheikh 1994). ALMR and PRISM are both pool type reactors that are less than a gigawatt thermal. The PRAs for PRISM and ALMR were performed by GE staff. The detail of these documents reflects the fact that these designs are in the conceptual phase. The PRISM PRA is Level 3 while the ALMR PRA is only Level 1. The PRA for EBR-II is thorough and similar to other modern Level 1 PRAs in detail. All of the PRAs include seismic initiators with only EBR-II including fire initiators. There are some features that all of the PRAs share in common. All of the reactors are assessed to have internally initiated core damage frequencies below typical GEN-II PWRs. Additionally the risk for each of these designs is dominated by seismic initiators. This information is summarized in Table II.2. Finally all of the PRAs include only point estimates for all events. Error factors of three and five have been applied uniformly to see the impact uncertainties might have on LBE acceptance.

Table II.2. Core Damage Frequency Summary

Design	Internal CDF (per year)	Seismic Initiated CDF
EBR-II	$2x10^{-6}$	$2x10^{-5}$
PRISM	$1x10^{-8}$	$5x10^{-8}$
ALMR	$1x10^{-10}$	$3x10^{-6}$

II.A.1. PRISM PRA

Several of the failure probabilities in the PRISM PRA have been scrutinized by the NRC in their preapplication safety evaluation report (NUREG-1368). These can be seen specifically in the Funtional Event Trees presented in Appendix B. There are some differences in the PRA reviewed by the NRC and the PRA used in this report. The version used in this thesis is the original 1986 draft, the NRC reviewed a modified version from 1989 that has some substantial design differences. In particular, scram failure frequencies on the order of 10^{-9} per demand are thought to be overly optimistic. Additionally, the pump coastdown failure frequency on the order of 10^{-9} per reactor year, as well as the scram signal failure frequency on the order of 10^{-11} per demand, have been questioned as being overly optimistic. The NRC also finds the vessel failure frequency of 10^{-13} per year to be low.

Vessel failure is included in the 1986 draft at 10^{-7} per year, which actually seems pessimistic. LWRs typically use a value of around 10^{-8} per year for vessel failure. Since the SFR has a low pressure primary, this value should be lower. In any event, it is unusual to include vessel failure as an initiating event in a PRA. Justification of any of the failure probabilities is difficult as there is little operating experience with many of the components. Electromagnetic pumps have not been used on the scale of PRISM. Synchronous coastdown motors are not a necessary component for centrifugal pumps, and they have not been used in any existing designs.

On top of some optimistic failure frequencies, the initiating event frequency for reactivity insertions at 10^{-4} per year was observed to be approximately two orders of magnitude lower than typical LWR numbers. RVACS has a failure probability around 10^{-8} depending on the initiator and the mission time. This failure probability is justified by GE saying that RVACS is continuously operated and monitored. This would allow operators to detect degradation before failure. There are no support systems necessary to operate RVACS.

Many of these estimates are likely the result of insufficient detail to put together fault trees for many of the systems. There are only a few fault trees included in the PRA, and they are typically very short. For rod insertion, failure consists of an or gate with the independent failure of six basic events "rod fails to insert," and the basic event "common cause failure of all rods to insert." There is a distinct lack of detail on how the rods fail to insert.

All initiating events in the PRISM PRA have the same Level 1 top event responses. These are covered in more detail in Chapter III. This gives the PRA a largely parallel structure that is not commonly used. Each initiator typically has specific responses to that initiator as the top events.

While the PRISM PRA is the only Level 3 assessment, the core response and containment response trees are quite conservative. This can be attributed to several factors. First, the EBR-II loss-of-flow experiments had not been performed and the estimated likelihood of eutectic penetration in these transients is overestimated (assumed to be unity). Many other probabilities in the core response and containment event trees are 0.5, 0.9, and 0.1. There is not much effort given in the PRA to justify these numbers. This leads to the conclusion that there was not a great deal of knowledge regarding the mechanistic responses that would take place. In general the performance characteristics of metal fuel were not well understood. Secondly, the core response and containment/release response were originally performed for oxide fuel and were then modified in a conservative manner to represent metal fuel. For the release response, a larger source term and faster release rate were used than for the oxide core.

II.A.2. ALMR PRA

The ALMR PRA shares many similarities with the PRISM PRA as far as optimistic failure frequencies. The RPS signal and rod insertion reliabilities are both fairly optimistic and are around 10^{-8} per demand. RVACS is assumed to always succeed for non-seismic initiators. The event trees do not have a parallel format and credit specific combinations of responses for given initiators. For example, the number of rods inserted after an anticipated transient overpower determines the number of pump coastdown motors that must succeed, and thus the failure probability for several branches in the event trees. More fault trees, of greater detail have been used to estimate some of the failure frequencies.

One major issue with the ALMR PRA is that some event trees are apparently missing. Partial flow blockage is listed as an initiating event, but the event tree that should show the response of the plant to this event is missing. Functional event trees have been used to create an appropriate response tree for this initiating event.

There is a significant difference with respect to seismic isolators and RVACS. The isolators are assumed to never fail in the ALMR PRA, where in PRISM they are assumed to fail at 1.2g PGA. RVACS is assumed to always work with the exception of large seismic initiators, where it has a fragility much greater than a typical PWR containment; given an earthquake >2g a failure probability of 10^{-5} is used for RVACS. Compare this to the results of Chapter V and it is clear that this is an optimistic estimate. This is significantly more reliable than PRISM PRA which has a failure probability of 10^{-1} for RVACS given an initiating earthquake of 1.2g PGA.

II.A.3. EBR-II PRA

The most immediate difference from the PRISM and ALMR PRA in the EBR-II PRA is the level of detail associated with a real reactor. Each initiator has specific top events meant to mitigate the effects of that initiator. There are over 700 pages of fault trees. There are many more basic events, and there was an actual operating history that allows the frequency of failures to be updated using the statistical evidence. There is also a fine granularity for the core damage states including possible experimental damage, minor core damage, core damage, and structural damage. While the PRISM and ALMR PRAs have several core damage states, they are differentiated more or less on the percentage of core melt.

There are also some significant design differences between EBR-II and the GE designs. The EBR-II reactor is much smaller than the PRISM or ALMR designs at around sixty megawatts thermal. This leads to a smaller thermal inertia and an added initiator that presents a threat to the reactor. For a steam line break, there is sufficient cooling and fast enough response in the core inlet temperature that an overcooling accident is possible. Another major difference is that the pumps for EBR-II are centrifugal pumps. The PRA identified leakage of pump lubricant as a significant risk contributor. EBR-II is not seismically isolated. The primary failure mode is that the hangers holding up the vessel fail during a large (0.7g) seismic event.

EBR-II also has a much more diverse set of internal events that are significant contributors to risk. Loss of flow, loss of offsite power, partial blockage, and large reactivity insertion each contribute between 14% and 32% of the internal CDF. In comparison, internal CDF is dominated in PRISM by loss of shutdown heat removal during a long shutdown transient at over 90% of the internal CDF. For ALMR the internal risk is dominated by a large reactivity insertion. If modified to include RVACS failure for internal initiators using the PRISM reliability estimates, ALMR is also dominated by loss of shutdown heat removal. For comparison to RVACS, the mean fragility of the passive heat removal system, which is the most robust component in EBR-II, is 1.5g PGA. This is somewhere between the PRISM estimate and the ALMR estimate.

CHAPTER III – FUNCTIONAL EVENT TREES

Functional Event Trees (FETs) have been developed as a tool for several purposes. The first purpose is that it allows one to compare different reactor designs on an equal basis. The second purpose is that it provides a framework in which LBEs can be constructed. This type of framework is important to have within the TNF. As proposed, it is up to the applicant to decide what separates one sequence from another sufficiently that they are two different sequences. This could lead to an applicant splitting certain failures into multiple equally likely branches to reduce the frequency. For example, one could take each one centimeter segment along all the piping and refer to each break as a different initiator. Alternatively one could make many branches out of each attempt to recover a system over a time period. This would make each sequence of a lower frequency, but it would not actually improve the safety of the reactor. Forcing the applicant to use certain functional events as the initiators and top events would prevent this particular method of gaming the system.

FETs also put passive systems on the same footing as complex active systems that perform the same function. The same total failure probability for a function may be due to a single system in a passively safe system, but could be due to the failure of many different systems in an active design. If applicants are able to split up the failure of each different active subsystem into different events, it would be easier for a complex active system with the same functional failure probability to meet the TNF goal than for a passively safe system to meet the TNF goal. For example, consider an active system which has four trains that each have three different failure modes to provide decay heat removal, and a passive system only has two systems for decay heat removal and a single failure mode. The active system could split a total loss of decay heat removal across twelve different sequences, where the passive system can only spread it across two. This can be seen as giving a benefit to defense-in-depth measures but could get out of hand. The TNF recommends there be a limit on the sum frequency of sequences in the "frequent", "infrequent", and "rare" categories. A specific example on implementing this recommendation is not provided.

The functional event trees have been constructed in the computer codes SAPHIRE and CAFTA that allow one to easily group similar events together. This is particularly useful given that most of the event trees arising from different initiating events have the same top events. Initiators have been grouped together that cause similar plant response and have similar phenomenology. The PRISM PRA has twenty-one initiators. These have been reduced to ten using best judgment as to which initiators have the same type of phenomenology. This is not meant to be a definitive list of initiators, but it does cover all of the major contributors of the available PRAs. Certainly in a full application of the TNF, fire initiators would also need to be considered.

A generic initiator is assigned to the frequency of the most common of the initiators in the group and the most challenging phenomenology. This phenomenology determines the failure probability of the top events in the functional event trees. Only point estimates have been used in this analysis. This is primarily because events below 10^{-7} per reactor year are being screened. Actual LBEs selected in this process would require an uncertainty analysis and assurance that the F-C curve is met by the 95th percentile frequency as well as the 95th percentile consequences for sequences with a mean greater than 10^{-7}.

Table III.1 displays the initiator grouping for the generic initiators for the PRISM design. The generic initiator is then given the initiating frequency of the top original initiator and the phenomenology of the bold-faced original initiator. For example, the Shutdown Transient initiator has the phenomenology of IHTS pump failure and a frequency of 0.5 per year. Some PRISM initiating events arise from a conservative extrapolation of a probabilistic fracture mechanics analysis applied to the reactor vessel. These values are not similar to other estimates of vessel failure probability, especially for a low pressure system. These initiating events have not been included in the analysis. These initiating frequencies are found to be much higher than those cited in the ALMR and EBR-II PRAs. This method is consistent with grouping the sequences from the PRISM PRA.

Table III.1. PRISM Initiator Grouping

Generic Initiator	Original Initiators	Frequency (per year)
Shutdown Transient	Normal Shutdown	0.5
	Forced Shutdown	0.1
	Loss of One Primary Pump	0.08
	Loss of Operating Power Heat Removal	0.08
	IHTS Pump Failure	0.05
	Spurious Scram	0.04
	Loss of Shutdown Heat Removal Via BOP	8.2E-3
Local Blockage	**Local Core Coolant Blockage**	1.8E-6
Major Loss of Flow	**Station Black Out**	5E-2
	Loss of Substantial Primary Coolant Flow	3E-5
Large Reactivity Insertion	**Reactivity Insertion >$1.75**	1E-5
Medium Reactivity Insertion	Reactivity Insertion $0.35-**$1.75**	1E-4
Small Reactivity Insertion	Reactivity Insertion $0.11-**$0.35**	1E-4
Steam Generator Leak	Loss of Shutdown Heat Removal via IHTS	1E-2
	Large Na-H_2O Reaction	6E-8
Small Seismic	PGA 0.15g-**.3g**	5E-4
Medium Seismic	PGA 0.3g-**0.6g**	1.1E-5
Large Seismic	PGA 0.6-**1.2g**	8E-8
Vessel Failure and RVAC Failure	Reactor Vessel Leak	1E-6
	Vessel Failure	1E-7
	RVACS Failure	1E-8

Table III.2. PRISM Top Event Grouping

Functional Event	Original Events	Failure Probability Equation
Shutdown	RPS/PCS Signal Reactor Shutdown System	Pr(Signal)+Pr(RSS) (rare event union)
Fuel Heat Removal	Pump Costdown	Pr(PCD)
Primary Heat Removal	Shutdown Heat Removal via IHTS Shutdown Heat Removal via RVACS	Pr(IHTS)Pr(RVACS) (intersection)
Late Shutdown	Nominal Inherent Reactivity Feedback	Pr(NIRF/RSS)

The original event trees of the PRISM PRA are in terms of safety systems. These systems can be grouped according to the safety function they perform. Based on historical safety functions necessary to prevent release from SFRs, four safety functions have been selected (top events in the event tree) for the functional event trees. Table III.2 shows how the original systems are grouped. For shutdown, the RPS/PCS Signal system is grouped with the Reactor Shutdown System because failure of either system causes failure to shutdown.

To calculate the probability of failure for systems grouped together, the rare-event approximation is used for the union of events. For shutdown, the sum of the failure probabilities of the RPS/PCS Signal and Rod Insertion is used as the failure probability. For primary heat removal, the failure probabilities of RVACS failure and Heat Removal via IHTS failure are multiplied together. The failure probability for each event is dependent upon the initiator.

In the case of late shutdown, there is some dependence on the phenomenology of the shutdown mechanism. This is because a lack of signal does not in itself affect the feedback mechanisms. When rods fail to insert, the control rod drive expansion feedback becomes much less reliable.

Failure of rod insertion dominates the failure to shut down for all initiators given in the PRISM PRA. As such, the conditional failure probability of nominal feedback given failure to insert rods is used.

For sequences that do not include the failure of primary heat removal, it is assumed that RVACS is performing the function with failure of shutdown heat removal via IHTS. This assumption groups together nominal response sequences with those sequences that have the same OK end-state yet suffer a failure. This grouping ensures LBEs have the greater frequency of the nominal event with more difficult phenomenology of the single failure event.

Table III.3. and Table III.4 show the initiator and top event groupings for the ALMR design. Some of the major differences are the result of design change. Rod stops are included to reduce the size of reactivity insertions. Station black out, as an initiator, is defined to include both the loss of offsite power as well as a failure to properly go to hot standby. In PRISM this initiator consists of just loss of offsite power.

Although mechanistic transient analyses with the assessment of uncertainties would be required for an applicant to demonstrate compliance with the TNF framework, at the concept-development stage generic guidelines are more helpful to the analyst who is comparing alternatives. Denning et al developed realistic source terms, using best-estimate assumptions, for representative generic release categories for a small metal-fueled design.

Table III.3. ALMR Initiator Grouping

Generic Initiator	Original Initiators	Frequency (per year)
Shutdown Transient	Loss of 1 or 2 pumps Local Minor Fuel Faults Seismic < OBE Reactivity Insertion < 6 cents	0.5 0.1 0.03 0.01 0.05 0.04 8.2E-3
Local Blockage	**Moderate Local Fuel Faults**	1E-3
Major Loss of Flow	**Station Black Out** Loss of Substantial Primary Coolant Flow	7E-3 3E-5
Large Reactivity Insertion	Reactivity Insertion $0.50-**$0.70**	1E-6
Medium Reactivity Insertion	Reactivity Insertion $0.30-**$0.50**	3E-5
Small Reactivity Insertion	Reactivity Insertion $0.06-**$0.30**	1E-3
Steam Generator Leak	Loss of Shutdown Heat Removal via IHTS **Large Na-H_2O Reaction**	1E-2 7.2E-5
Small Seismic	PGA 0.2g-0.4g PGA 0.4g-**0.6g**	3E-4 6E-5
Medium Seismic	PGA 0.6g-**1.0g**	1.7E-5
Large Seismic	PGA 1.0g-2.0g PGA >**2.0g**	2.4E-6 1.3E-7
Incredible Initiators	Reactor Vessel Leak Reactivity Insertion >$0.70 Major Subassembly Blockage	1E-6 1E-7 1E-7

Table III.4. ALMR Top Event Grouping

Functional Event	Original Events	Failure Probability Equation
Shutdown	RPS/PCS Signal Enough Rods Insert Enough GEMs Succeed	[Pr(Signal)+Pr(RSS)]Pr(GEMs) (rare event union of series system intersection with GEMs)
Fuel Heat Removal	Pump Coastdown	Pr(PCD)
Primary Heat Removal	Shutdown Heat Removal via RVACS	Pr(RVACS)
Late Shutdown	Ultimate Shutdown System	Pr(USS)

The release categories were only developed for plant conditions in which either the primary system or containment system is assumed to be in a failed state. Events in which both of these barriers remain intact have extremely small radionuclide releases to the environment. In a system with a confinement system rather than a containment system, such as the PRISM design, it would be necessary to examine the design basis of the system to determine its effectiveness in retaining radionuclides under various accident conditions.

The PRISM PRA identifies twelve core damage categories, with the potential for loss of radionuclide retention capability. For each of these core damage categories, a containment response event tree is examined to determine the likelihoods of various release categories. If confinement is effective, the environmental release is considered negligible. Nine release categories are identified involving failure of confinement. With the exception of the PRISM R3 release category, all PRISM offsite doses exceed 500 rem, this gives an implied maximum acceptable mean frequency of 1×10^{-7} per reactor year. The LBEs that arise from this analysis with a frequency greater than 10^{-13} per reactor year aside from the nominal response to initiating events are displayed in Table IV.

Figure III.1 shows a graphical representation of the PRISM LBEs as well as LBEs calculated in a similar manner for the ALMR PRA and those calculated for an LWR in the TNF. Although our LEF calculations have been done for internal LBEs only, Figure III.1 includes seismic initiated LBEs. Seismic LBEs constitute all of the ALMR LBEs that lead to a non-negligible dose, and three of the four PRISM LBEs that lead to a non-negligible dose. The final PRISM LBE leading to a non-negligible dose is the shutdown transient followed by the failure of primary heat removal. This transient is 600 hours in duration and bulk sodium boiling begins at 20 days (human actions are not included in the PRISM PRA with ALMR only allowing for ultimate shutdown).

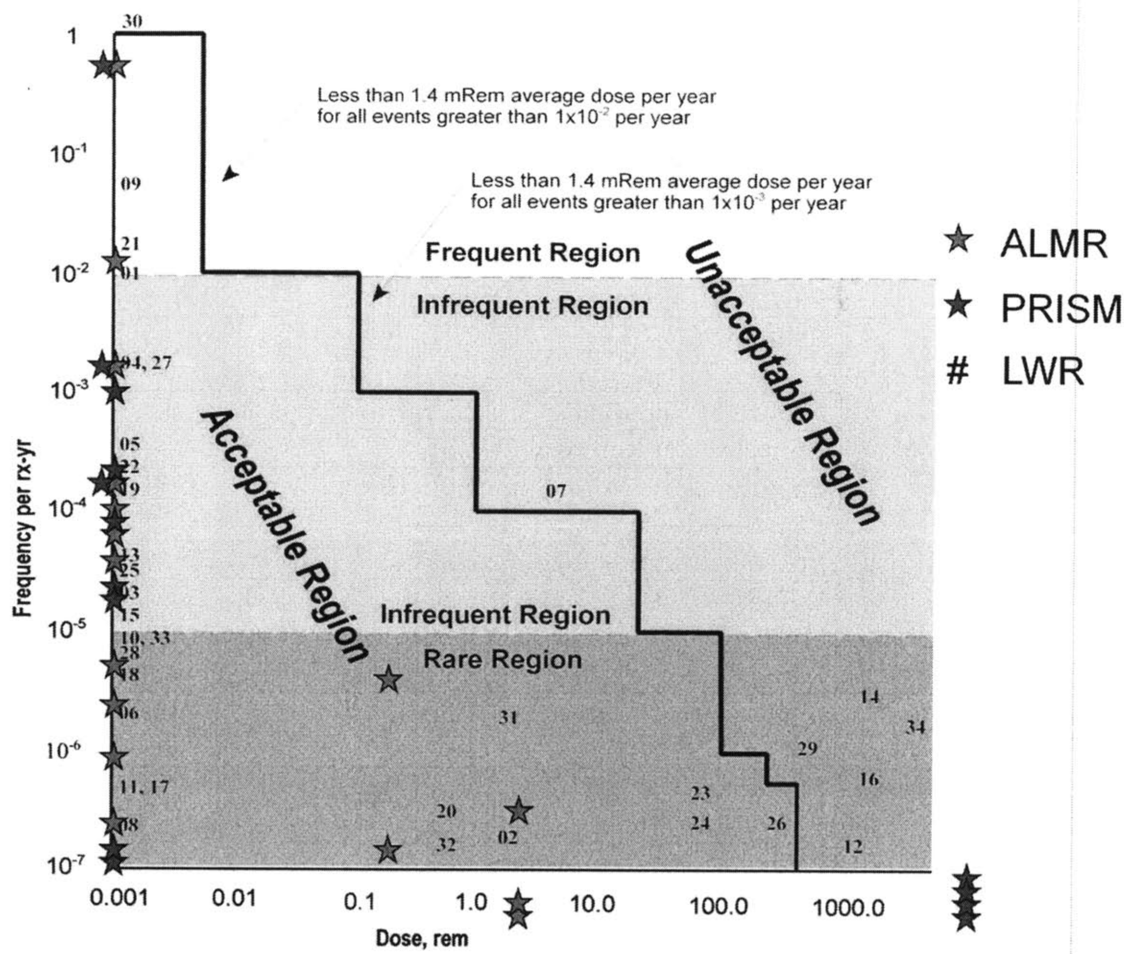


Figure III.1. Plot of LBE location on the F-C Curve for several reactors (Adapted from NUREG-1860).

This figure indicates that ALMR and PRISM would meet the F-C curve if the current PRAs are found to be of sufficient quality. Certainly fire and other external initiators would need to be examined as they are not included in these PRAs. As mentioned before, the seismic portion tends to be optimistic in both PRAs with regards to fragilities of certain systems such as scram and RVACS relative to those values found for even the most hardened systems in PWRs (e.g., containment roof failure). This could be attributed to seismic isolators at lower accelerations, but for very rare large earthquakes, one might expect the isolators to have a considerable probability of failure. However, the isolators are assumed to stay intact in both PRAs.

Table III.5. LBEs Calculated from Functional PRISM PRA

Initiator	Functional Events	Frequency	Dose
Shutdown Transient	Shutdown	2.1E-7	0
	Fuel Heat Removal	2.2E-9	0
	Primary Heat Removal	1.3E-8	>500 rem
	Shutdown, Late Shutdown	1.2E-10	>500 rem
Loss of Flow	Fuel Heat Removal	2.2E-10	0
	Primary Heat Removal	1.3E-9	>500 rem
	Shutdown, Late Shutdown	1.2E-11	>500 rem
Large Reactivity Insertion	Shutdown	1.5E-13	>500 rem
Medium Reactivity Insertion	Shutdown	2.6E-11	0
	Primary Heat Removal	2.6E-12	>500 rem
	Shutdown, Late Shutdown	1.6E-13	>500 rem
Small Reactivity Insertion	Shutdown	2.6E-11	0
	Primary Heat Removal	2.6E-12	>500 rem
	Shutdown, Late Shutdown	1.2E-12	0
Steam Generator Leak	Shutdown	5.2E-11	0
	Fuel Heat Removal	4.4E-11	0
	Primary Heat Removal	2.6E-10	>500 rem
	Shutdown, Late Shutdown	2.4E-12	>500 rem

These frequency values were arrived at by multiplying the conditional probability of release, as predicted by the core response and release event trees of the PRISM PRA, with the LBE frequency predicted in the functional event trees. For example, for the Large Reactivity Insertion scenario in which shutdown has failed, the estimated event frequency is 2.6×10^{-12} per reactor year (this sequence is denoted as G3 in the PRISM PRA). This particular sequence type has a 5.7% chance of resulting in a large radioactive release. This leads to a frequency of release of 1.5×10^{-13}. For the PRISM analysis, those LBEs that have no probability of large release turn out to always end in non-release states, rather than in a smaller release category.

For the ALMR LBEs generic release categories from Denning et al were assigned to match the core damage description given in the original PRA. A second set of generic trees was to be generated to see the effect of using DRACS rather than RVACS. However, the failure probability for all non-seismic initiators was found to be less than 10^{-7} per demand when the failure and repair rates from the PRISM PRA are used. This indicates that there is no particular advantage for either system under the TNF.

It is important to note that the small reactivity insertion with two shutdown failures does not lead to a release. This is because the mission time is much shorter than the other transients where these two failures lead to a high (0.41) conditional probability of release. This does not allow time for bulk sodium boiling.

While a more detailed PRA would be necessary to show compliance with the TNF, the initial results show that the SFR tend to have few to no small release sequences. When failure leads to a dose, it is rare, and usually very large. This is not a lot different than is observed for LWRs though the calculated frequencies are lower for the SFR. The dominant failure mode found in this review of PRAs is failure of decay heat removal during a long transient. Energetic events, which have been the focus of multiple sodium reactor designs, are found to occur only very rarely. The addition of a third scram system, fuel streaming channels, or other features meant to mitigate or prevent these rare events would have no impact with regards to meeting the goals of the TNF. Seismic events are found to be a dominant contributor to the failure of the decay heat removal system. This is further investigated in Chapter V.

CHAPTER IV – IMPORTANCE MEASURES

Importance measures play a significant role in risk-informed regulatory decision making. They serve to categorize structures, systems, and components (SSCs) appearing in a plant's Probabilistic Risk Assessment (PRA) according to their risk significance which, in turn, is one input into the determination of the regulatory requirements, such as special treatment, that are imposed on the SSCs.

There are several standard importance measures in use today. These are the Risk Achievement Worth (RAW), the Risk Reduction Worth (RRW), and Fussell-Vesely (FV) (Cheok, Parry, and Sherry 1998). RAW for an SSC is the conditional frequency of an end state (e.g., core damage or large early release of radioactivity for a light water reactor) given that the SSC fails divided by the baseline frequency of the end state i.e.,

$$\mathrm{RAW} = \frac{\mathrm{Fr(end\ state/\ SSC\ failure)}}{\mathrm{Fr(end\ state)}} \tag{IV.1}$$

RRW for an SSC is the baseline frequency of the end state divided by the conditional frequency of an end state given that the SSC succeeds , i.e.,

$$\mathrm{RRW} = \frac{\mathrm{Fr(end\ state)}}{\mathrm{Fr(end\ state/\ SSC\ success)}} \tag{IV.2}$$

FV is a measure of the fractional contribution of the SSC to the total frequency of the end state. Put another way, FV is the conditional probability that, if the end state is reached, it is reached via a failure mode that involves that SSC. FV is not a third independent importance measure. It is related to RRW as follows:

$$FV = 1 - \frac{1}{RRW} \tag{IV.3}$$

RAW and FV are usually used to place SSCs of nuclear power plants into several safety categories, for example under 10 CFR 50.69. For example, those SSCs with a RAW less than 2.0 and a FV less than 0.005 are considered to be of low safety significance and the regulatory requirements are adjusted accordingly.

IV.A. Motivation for the Development of LEF

Although the importance measures introduced above are useful for ranking the *relative* importance of SSCs, they do not provide information on margin with respect to a limit or goal. For example, suppose that a hypothetical reactor is designed with a core damage frequency (CDF)[iv] a factor of 100 times smaller than a presumed regulatory limit (RL) of 10^{-4} per reactor year, i.e., $CDF = \frac{RL}{100}$. If an SSC with a RAW of 2.5 were to be in the failed state, the new CDF of the hypothetical reactor, CDF*, would still be 40 times lower than the regulatory limit $\left(CDF^* = \frac{RL}{100} x2.5 = \frac{RL}{40} \right)$. This information is not provided by RAW or any of the standard

[iv] In this example, the end state is core damage and CDF is the quantity Fr(end state) in Equations (VI.1) and (VI.2).

importance measures because they deal with relative changes in the risk metric without regard to external limits. As more regulations become risk-informed, importance measures that are related to the regulatory limit could be useful. This becomes even more evident if one considers the following hypothetical reactor:

- The reactor has four completely independent safety systems
- Each system is capable of preventing core damage
- Each system has a failure probability of 10^{-2}
- The initiating event frequency is once per year

For this reactor, the CDF is 10^{-8} per reactor year. Each system has a RAW of 100 $\left(\text{RAW} = \frac{10^{-6}}{10^{-8}} = 100\right)$, a RRW of infinity $\left(\text{RRW} = \frac{10^{-8}}{0} = \infty\right)$, and a FV of unity $\left(\text{FV} = 1 - \frac{1}{\text{RRW}} = 1 - \frac{1}{\infty} = 1\right)$. These values would typically be considered quite high suggesting that all of the systems should be treated as safety significant. However, the actual core damage frequency of the system is 10^{-8} per reactor year. If the reactor were to be in a configuration with two systems down, the reactor would still (barely) meet a regulatory limit of 10^{-4} per reactor year. This example demonstrates the limited use of these importance measures for the purpose of system design within risk-informed constraints.

To address the issue of importance measures in relation to margin, Reinert and Apostolakis developed the concept of a RAW threshold value (Reinert and Apostolakis 2006). Suppose that the frequency of a particular end state (e.g., core damage) has a limit (threshold) Fr(end state)$_{\text{threshold}}$ and that the baseline frequency of that state is Fr(end state)$_{\text{baseline}}$. Then, the $\text{RAW}_{\text{threshold}}$ value is defined as:

$$RAW_{threshold} = \frac{Fr(\text{end state})_{threshold}}{Fr(\text{end state})_{baseline}} \qquad (IV.4)$$

This value is particularly useful when a reactor is still in the design phase. An SSC with a RAW in relation to this end state less than $RAW_{threshold}$ could be a candidate for removal and the regulatory goal would still be met.

Another motivating factor in the development of LEF was to be sure that it would be compatible with the TNF. The standard importance measures require the end states to result from multiple sequences to have a significant meaning. For example, consider a LBE with frequency of 3×10^{-3} per year and a consequence of 10 mrem. Assume that this LBE has an initiating-event frequency of one per reactor year and contains the failure of two SSCs in series. One SSC has a failure probability of 10^{-1} and the other has a failure probability of 10^{-2}. These SSCs would have fairly uninformative importance measures. In particular, RRW for both systems is infinity or undefined (10^{-3} per year/0 per year), and FV is unity for both systems. RAW for these systems is simply the reciprocal of the failure probability of each SSC. This would then rank reliable systems as more important and unreliable systems as less important. It is observed from the F-C curve that the goal is to keep an LBE with a 10 mrem dose below a frequency of 10^{-2} per year. The $RAW_{threshold}$ value is 3.33. Both systems would have a RAW greater than $RAW_{threshold}$. LEF would rank both systems as being equally important. This is because either system could have a failure probability 3.33 times the original value. This is more realistic than the other importance measures. If the systems are one-of-three, then either system would be impacted by the same multiplication if it were changed to one-of-three, assuming either independence or identical common cause failure models.

This advantage is expanded significantly when there are is a different goal for each sequence. LEF would then be the lowest value amongst all LBEs that contain the SSC of interest.

IV.B. LEF Definition and Calculation

LEF is defined as the probability of failure for an SSC that causes the frequency of an end state, e.g., an LBE, to be equal to the frequency limit for that end state divided by the baseline SSC failure probability:

$$\text{LEF} = \frac{\text{Pr(SSC Failure) that makes Fr(end state) equal to a limit}}{\text{Pr(Baseline SSC Failure)}} \qquad \text{(IV.5)}$$

In the case where the baseline end-state frequency is already below the limit, this importance measure is meaningful only for those SSCs with a RAW greater than $RAW_{threshold}$ (see Equation (IV.4)). For SSCs with RAW smaller than $RAW_{threshold}$, the system failure probability does not reach the limit even when their failure probability is set equal to unity. Therefore, LEF does not exist for these SSCs. For SSCs with RAW greater than $RAW_{threshold}$, there exists a failure probability such that Fr(end state) = Limit. In this case, LEF is this limiting probability of failure divided by the original probability of failure. When the baseline end-state frequency is below the limit, all SSCs have a LEF greater than unity. Those SSCs with a large LEF have a great deal of margin. SSCs with a LEF close to unity have little margin. When the baseline end-state frequency is greater than the limit, the meaning of LEF is slightly different. In a similar manner, only those systems with a RRW greater than a certain threshold would have a useful LEF value, i.e.,

$$RRW_{threshold} = \frac{Fr(\text{end state})_{baseline}}{Fr(\text{end state})_{threshold}} \tag{IV.6}$$

If an SSC has a RRW smaller than $RRW_{threshold}$, no amount of improvement to that component alone can move the end-state frequency below the limit; therefore, LEF does not exist for these SSCs. For SSCs with RRW greater than $RRW_{threshold}$, there exists a failure probability less than the baseline probability that will cause the frequency of the end state to equal the limit. Those components with LEF values closer to unity would be good candidates for improvement in reliability for the system failure probability to meet the limit. This is because those SSCs with a LEF closer to unity need less improvement for the system failure probability to be reduced below the limit.

It is expected that LEF should be useful mainly at the system level because design modifications of systems tend to give a multiplicative effect. For example, switching from a one-out-of-three system to a one-out-of-two system for identical independent components decreases the reliability by a factor equal to a single pump's failure probability. Since LEF is applied at the system level, common cause failures are included as part of the baseline system failure probability.

IV.B.1. LEF for LBEs

To calculate LEF for an SSC within the TNF is straightforward. A demonstration on how to find LEF^k for SSC^k follows. Since SSC^k may appear in multiple LBEs (indexed as LBE_i^k), LEF^k is the minimum LEF over all LBE_i^k as shown by Equation (5a), which is a modified version of Equation (IV.5):

$$LEF^k = \min_i \left(\frac{\text{Pr(SSC}^k \text{ Failure) thatsuch that} \left[\text{Fr(LBE}_i^k) = \text{Fr(Limit for LBE}_i^k) \right]}{\text{Pr(Baseline SSC}^k \text{ Failure)}} \right) \quad \text{(IV.5a)}$$

Each LBE has an associated frequency and consequence that can be compared with the corresponding limits of the F-C curve. It is possible that a sequence represented by an LBE could involve different systems than the particular sequence that has the highest frequency within the group. In order to assure compliance with the TNF, it will be necessary to rerun the risk analysis and determine whether the modified LBE falls below the F-C curve and the defense-in-depth requirements, if any, are met. The use of functional event trees allows the relation of each system to an LBE to be defined algebraically. This can eliminate the need to run the risk assessment model.

To find LEF^k for each SSC^k, one must first find the limiting frequency for a given LBE consequence. For example, an LBE with a dose of 0.2 rem (when referring to LBE dose and frequency, the 95th percentile for each is used) would have a frequency limit of 10^{-3} per reactor year (Figure II.1). Next, one finds the algebraic relationship between each LBE_i^k and the failure of SSC^k. This relationship is defined by the model. In the case of typical event tree models, this relationship is linear, i.e.,

$$Fr(LBE_i^k) = a\ Pr(SSC^k \text{ fails}) + b \tag{IV.7}$$

From Equation (IV.5a) one can then arrive at:

$$Fr(\text{Limit for } LBE_i^k) = a\ Pr(SSC^k \text{ fails})\ LEF_i^k + b \tag{IV.8}$$

One can find b by evaluating the conditional frequency of LBE_i^k with SSC^k set to succeed, in which case $Pr(SSC^k \text{ fails}) = 0$. Equation (IV.7) then yields

$$Fr(LBE_i^k \mid SSC^k \text{ succeeds}) = b \tag{IV.9}$$

Finally, an expression for LEF_i^k for each LBE_i^k containing SSC^k is found:

$$LEF_i^k = \frac{Fr(\text{Limit for } LBE_i^k \text{ dose}) - Fr(LBE_i^k / SSC^k \text{succeeds})}{Fr(LBE_i^k) - Fr(LBE_i^k / SSC^k \text{succeeds})} \tag{IV.10}$$

The minimum LEF_i^k calculated in this manner over all LBE_i^k is the value assigned to SSC^k (Eq. (IV.5a). A detailed example is presented below.

In this example the main concerns are the internal events that lead to large release for the PRISM design. LEF does not make sense to apply simultaneously to fragility and random internal failures as a modification to the system design is not likely to have a simultaneous effect on these failure modes. The limiting frequency for large releases is 10^{-7} per year. By using the information in Table III.5 with the equations for predicting top event failure probabilities, LEF^k for each SSC^k can be calculated. These values are presented in Table IV.1. It is observed that the RPS/PCS signal to scram has a large margin. Additionally, the scram system has a relatively large amount of margin. The systems contributing to primary heat removal have little failure probability margin. Nominal inherent reactivity feedback would have a LEF of 590 but because the failure probability is 0.1 the multiplication can only go to 10 (indicated by 10 in parentheses).

Table IV.1. LEF Table for Systems by using LBEs

System	LEF	LBE
Reactor Protection System/Plant Control System Signal	240,000	Shutdown Transient, Late shutdown
Reactor Shutdown System (scram)	590	Shutdown Transient, Late shutdown
Pump Coastdown	-	N/A
Nominal Inherent Reactivity Feedback	590 (10)	Shutdown Transient, Late Shutdown
Operating Power Heat Removal	-	No credit taken
Shutdown Heat Removal through the Intermediate Heat Transfer System	7.7	Shutdown Transient
Reactor Vessel Air Cooling System (RVACS)	7.7	Shutdown Transient

The purpose of this example is to demonstrate the use of LEF in determining candidates for simplification. Although it is beyond the scope of the thesis to suggest specific design alternatives that meet all the design requirements, a possible simplification for the signal to scram will be considered as an example. In the PRISM design, scram success requires at least two-out-of-four cabinets to send a scram signal, which in turn requires two-out-of-four circuit breakers to trip in each cabinet. This could be changed to each cabinet using two-out-of-three breakers, or to using two-out-of-three cabinets. Further analysis would need to be done to show that this design change meets all requirements (including defense in depth).

One may desire to include some additional margin to the regulatory limit to account for uncertainties and possible post-license discoveries. This could be done by setting a LEF limit on a per system basis depending on the uncertainty about the system failure. Additionally, it should not be assumed that simplifying a system will always improve overall economics. This is because more robust systems are likely to have higher reliability and may allow a higher capacity factor.

The difference between the LEF values calculated here and those in Section IV.A.2 can be attributed to the fact that LBEs with a dose limit are used in this section and a sum of sequences that lead to energetic release is used in Section IV.B.2, i.e., different risk metrics are used for the same frequency goal.

IV.B.2. LEF for End States that are a Union of Sequences

Calculating LEF for an SSC where multiple sequences lead to an end state whose sum frequency must remain below the regulatory limit is different than calculating LEF when LBEs, or single sequences, are the limit. For example, the end-state frequency may be the core damage frequency or the frequency of a large early release of radioactivity for a light water reactor. First, some variables are defined:

$$\mathrm{Fr(end\,state)} = \mathrm{R} \tag{IV.11}$$

$$\mathrm{Regulatory\ Limit\ of\ Fr(end\,state)} = \mathrm{R}_{\mathrm{threshold}} \tag{IV.12}$$

$$\mathrm{Pr(SSC}^{k}\ \mathrm{failure\ in\ accident\ sequence\,i)} = q_i^k \tag{IV.13}$$

Using this terminology, an example is presented where an end state consists of the sum of two sequences which each contain SSC^k. This example should be sufficient to communicate the method for calculating LEF^k for an end state that is a union of sequences. For the general case, see Appendix A.

One can take advantage of the fact that the end-state frequency is a linear combination of the frequencies of the accident sequences leading to that end state (rare-event approximation). In this example, SSC^k is involved in two sequences leading to the end-state frequency, i.e.,

$$\mathrm{R} = a_1 q_1^k + a_2 q_2^k + b \tag{IV.14}$$

As an example, take $R=10^{-5}$, $b=4\times10^{-6}$, $a_1=10^{-2}$, $a_2=10^{-5}$, $q_1^k=3\times10^{-4}$, $q_2^k=3\times10^{-1}$. $R_{threshold}$ will be defined later to demonstrate some properties of LEF. The first sequence is a more likely initiator where the system has not been compromised. The second sequence could be a less likely initiator that causes the conditional failure of the system to be high such as losing multiple support systems. It is noted that b is the sum of the frequencies of all accident sequences not containing failure (or explicit success) of SSC^k. One can then write

$$b = R\,|_{q=0} \tag{IV.15}$$

where q=0 means that both q_i^k are set equal to zero.

Within this example, one can easily calculate RAW and RRW for SSC^k.

$$RRW = R/b = 2.5 \tag{IV.16}$$

$$RAW = (a_1 + a_2 + b)/R = 1001.4 \tag{IV.17}$$

It is noted that, on the basis of these traditional importance measures, SSC^k would be of high risk significance. LEF^k is now introduced:

$$R_{threshold} = (a_1\,LEF^k\,q_1^k) + (a_2\,LEF^k\,q_2^k) + b \tag{IV.18}$$

This equation must be modified because the product $LEF^k q_i^k$ may not exceed unity. This leads to:

$$R_{threshold} = (a_1 \min(1, LEF^k q_1^k)) + (a_2 \min(1, LEF^k q_2^k)) + b \tag{IV.19}$$

To solve for LEF^k, subtract Equation (IV.14) from Equation (IV.19)

$$R_{threshold} - R = [(a_1 \min(1, LEF^k q_1^k)) + (a_2 \min(1, LEF^k q_2^k))] - (a_1 q_1^k + a_2 q_2^k) \tag{IV.20}$$

If $R_{threshold}$ is set at a value of $2x10^{-5}$, this will cause $LEF^k q_i^k$ to be less than unity for all *i* and will reduce Equation (IV.20) to

$$\frac{R_{threshold} - R}{a_1 q_1^k + a_2 q_2^k} + 1 = LEF^k = 2.66 \tag{IV.21}$$

This relationship may also be written as Equation (IV.22).

$$\frac{R_{threshold} - R}{R - R/RRW^k} + 1 = LEF^k \tag{IV.22}$$

For more details on the derivation of this expression see Appendix A.

Finally, using Equation (IV.3) one may derive the expression

$$\frac{(R_{threshold}/R)-1}{FV^k}+1=LEF^k \tag{IV.23}$$

If one were to set $R_{threshold}$ at 10^{-4}, this equation would not be valid. This is because $LEF^k q_2^k$ exceeds unity from the solution of Equation (IV.21), i.e., LEF^k=17.5 and $LEF^k q_2^k = 5.25$. For this example, the correct value of LEF^k will be solved for graphically using Equation (IV.19) in Figure IV.1. The dashed line is $R_{threshold}$ and the solid line is the right side of the equation as LEF^k increases. The solution for LEF^k is equal to 28.7. This is higher than the answer predicted from equation (IV.21). A more general method to find LEF^k for a sum of sequences is presented in Appendix A.

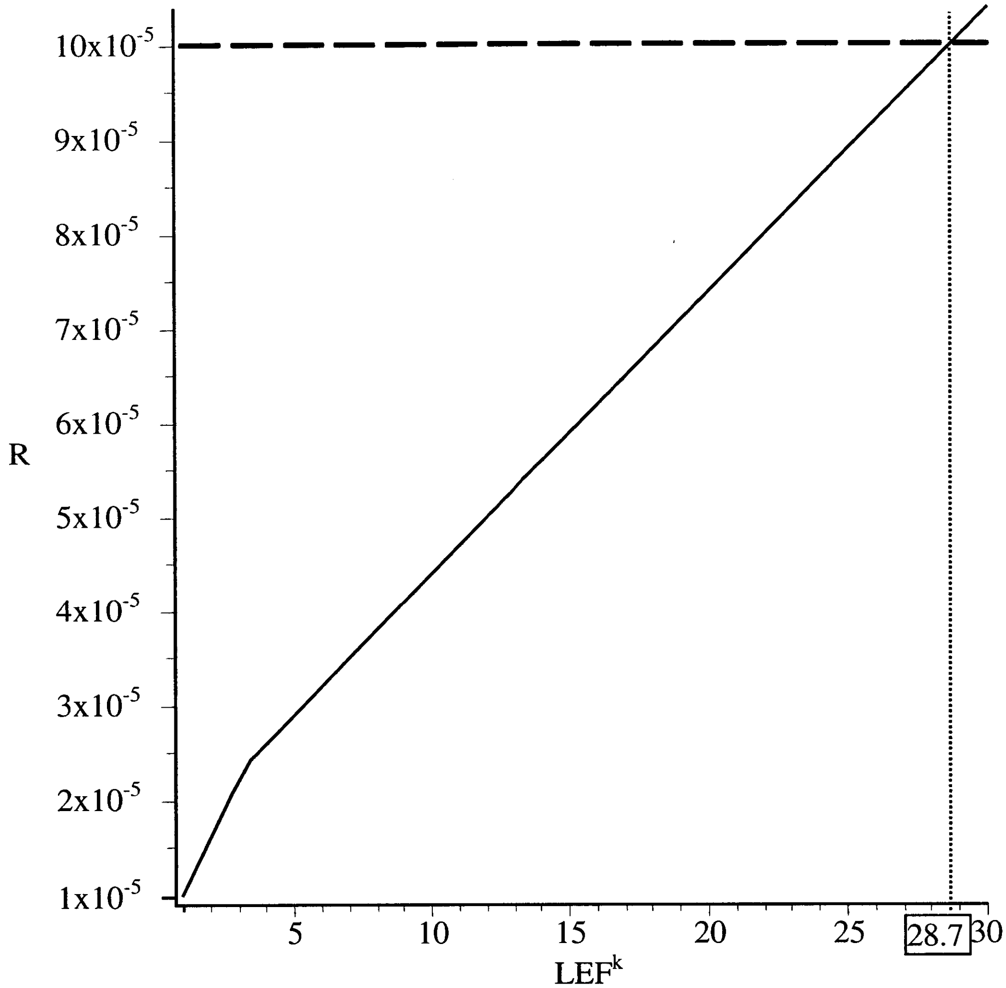


Figure IV.1. A graphical solution of the two-sequence example. The risk increases more steeply until one of the $LEF^k q_i^k$ reaches unity. Risk continues to increase from that point at a lower rate. The solution to this example is 28.7.

In the following example, LEFs for SSCs will be determined for the frequency of an "energetic release"[v] end state (i.e., CDA). This end state has been chosen because it has historically represented a regulatory hurdle for sodium-cooled fast reactors.

[v] Note that the frequency of energetic release here ($6.4x10^{-10}$ per reactor year) plays the role of Pr(System Failure) in Equations (1-3 and 5, 6).

The definition used for an energetic release end state is any end state with early or late recriticality and a dose estimated to be greater than 500 rem[vi] at 1.6 kilometers from the release point. To demonstrate this method, a postulated limit of 10^{-7} per reactor year is used. In this case, though, the sum of the frequencies of all sequences rather than a single dominant sequence is used. The sequences with frequencies below this value are screened from consideration.[vii] It is noted that, although only internal events are included in this example, the dominant contributors to energetic release in the PRISM PRA are rare large earthquakes that compromise several SSCs, predominantly shutdown heat removal. The use of LEF for seismic fragility is not presented here.

In this example, the functional event trees constructed for LBE selection are not used. Instead, the PRISM PRA has been input in the code SAPHIRE. This code is then used to calculate the RAW values with respect to energetic release. Three systems turned out to have low RAW values (RAW << $RAW_{threshold}$). The other four systems turned out to have very high values (RAW > 1000) (Table IV.2).

Table IV.2. RAW values for top level systems in the PRISM reactor design

System	**RAW (Energetic Release)**
Reactor Shutdown System (scram)	5.8E7
Reactor Vessel Air Cooling System (RVACS)	1.8E5
Shutdown Heat Removal through the Intermediate Heat Transfer System	1.4E4
Reactor Protection System /Plant Control System Signal	1.1E3
Pump Coastdown	4.0
Nominal Inherent Reactivity Feedback	4.0
Operating Power Heat Removal	1.0

[vi] These are states R2A, R3, R4A, R8A, R8S, and R8U of the PRISM PRA.[2]
[vii] This practice has also been referred to as "practical elimination."[12]

The frequency of energetic release is equal to 6.4×10^{-10} per reactor year. The same limit of 10^{-7} per reactor year is conservatively used as a limit for energetic release as a sum of sequences. This makes $RAW_{threshold}$ equal to about 150 ($156 = 10^{-7}/6.4 \times 10^{-10}$). This value is significantly larger than two, which is the value used to determine high-risk significant systems in current regulations.

Using SAPHIRE's "change set" function, each of the system failure probabilities were multiplied by a LEF as estimated by Equation (A.13) of Appendix A. The algorithm explained in Appendix A was then implemented manually when iteration was necessary. This was not difficult to do for seven systems. However, if there were many systems with many failure modes, an automated method would save time. The results are given in Table IV.3.

Table IV.3. LEF values for systems where the RAW value is greater than $RAW_{threshold}$

System	**LEF (Energetic Release)**
Reactor Protection System /Plant Control System Signal	2.5E8
Reactor Shutdown System (scram)	460
Shutdown Heat Removal through Intermediate Heat Transfer System	240
Reactor Vessel Air Cooling System (RVACS)	250

Of particular note is the signal to scram (Reactor Protection System/Plant Control System Signal). It is observed that the RAW value is very high indicating high importance according to 10 CFR 50.69. However, LEF is also high, indicating room for simplification while still meeting the safety goals. The scram signal failure probability as given is 1.4×10^{-11} per demand, which is very low compared to typical PWR numbers (USNRC 1994). Even if the probability is set to something more in line with typical PWR reliability, such as 1.4×10^{-5}, LEF would still be 250.

As seen in Section IV.B.1, the scram system itself has substantial margin with a LEF of 38. Both the RVACS and Intermediate Heat Transfer System have relatively low LEF values and due to their passive and simple nature would not be good candidates for simplification regardless of the LEF value. For these systems, both the RAW value and the LEF value indicate high importance.

CHAPTER V – SEISMICALLY INITIATED LICENSING BASIS EVENTS

Seismic requirements for nuclear power plants within the United States have historically been the result of a negotiation between the applicant and the U.S. Nuclear Regulatory Commission (NRC). The NRC and the applicant rely on engineering judgment and knowledge of seismic activity at the site to determine both the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE).

The OBE according to 10CFR50 Appendix S is "the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional. The operating basis earthquake ground motion is only associated with plant shutdown and inspection unless specifically selected by the applicant as a design input" (US Code of Federal Regulations, 2007. Title 10, Part 50, Appendix S to Part 50).

The SSE is "the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional." The SSE is to be at least 0.1g peak ground acceleration (PGA) and the OBE is typically one third of the SSE acceleration. Typical PGA values of SSEs for plants east of the Rocky Mountains range from 0.1g to 0.25g and the PGA values for plants west of the Rocky Mountains range from 0.25g to 0.75g. For comparison, the Niigataken Chuetsu-Oki earthquake was measured at the Kashiwazaki-Kariwa site in Japan to have a PGA between 0.69g~0.83g.

In applying the TNF to SFRs, it was observed that seismically induced failures presented the greatest challenge to the ALMR and PRISM designs. This is because SFRs are susceptible to reactivity insertions, sloshing, and other seismically induced failure modes that LWRs are not. The main failure mode is failure of the decay heat removal system due to a seismic initiator that also causes a reactivity insertion. Both of these designs attempt rely on seismic isolators to help meet the TNF goal. Although isolators should be useful in most scenarios, in very rare large earthquakes, the isolators could fail leading to plant damage states that are difficult to analyze and would likely lead to a large release. The hazard curves used in both PRAs are fairly low, but are intended to cover 90% of all U.S. reactor sites. This is shown later by comparing the initiators used in the PRAs to the hazard curve from the Clinton site which is of low to moderate seismicity (USNRC 2006).

As seen in Figure II.1, the lowest frequency considered is 10^{-7} per reactor year. This means that when selecting seismic LBEs, they should sufficiently cover the range from the OBE frequency down to the cutoff. This means that one must consider seismic hazards all the way down to a mean frequency of 10^{-7}. It is in this very rare region where it is difficult to quantify risk. Even if some assumptions are made that all the risk to be quantified, the problem becomes demonstrating that the goal is met.

When dealing with very rare earthquakes, getting quantitative information from experts or available data is difficult. A question arises as to what the maximum possible peak ground acceleration might be. In a study done to examine rare earthquakes for the Yucca Mountain[3] repository, the model used to determine the largest possible earthquake based on mechanistic analysis finds that there is no maximum for peak ground acceleration (Andrew, Hanks, Whitney 2007):

> *"We will not report peak ground acceleration (PGA). The strength of the material does not limit peak acceleration. In contrast to a lumped mass, a waveform in a continuum can have a step change in velocity (a delta function in acceleration) with finite stress. The physical principles we are applying do not impose a limit on the PGA."*

The authors do report a maximum possible peak ground velocity as well as several maximum spectral velocities. It has also been noted by Bommer *et al* that, although it would be valuable to define an upper limit on PGA, it is difficult to defend any such proposed limit in the light of our relatively short observation period and the failure of previous models to accurately predict a limit (Bommer *et al* 2004). Given that there is no strong evidence for a physical limit to peak ground acceleration, probabilistic methods that have no maximum cutoff should be used.

A review of the available data reveals that most seismic hazard curves do not extend to an annual probability of exceedance less than 10^{-6} per year (Figure V.1) and, more notably, none of the Eastern United States (EUS) sites have a hazard curve that goes beyond 1g PGA.

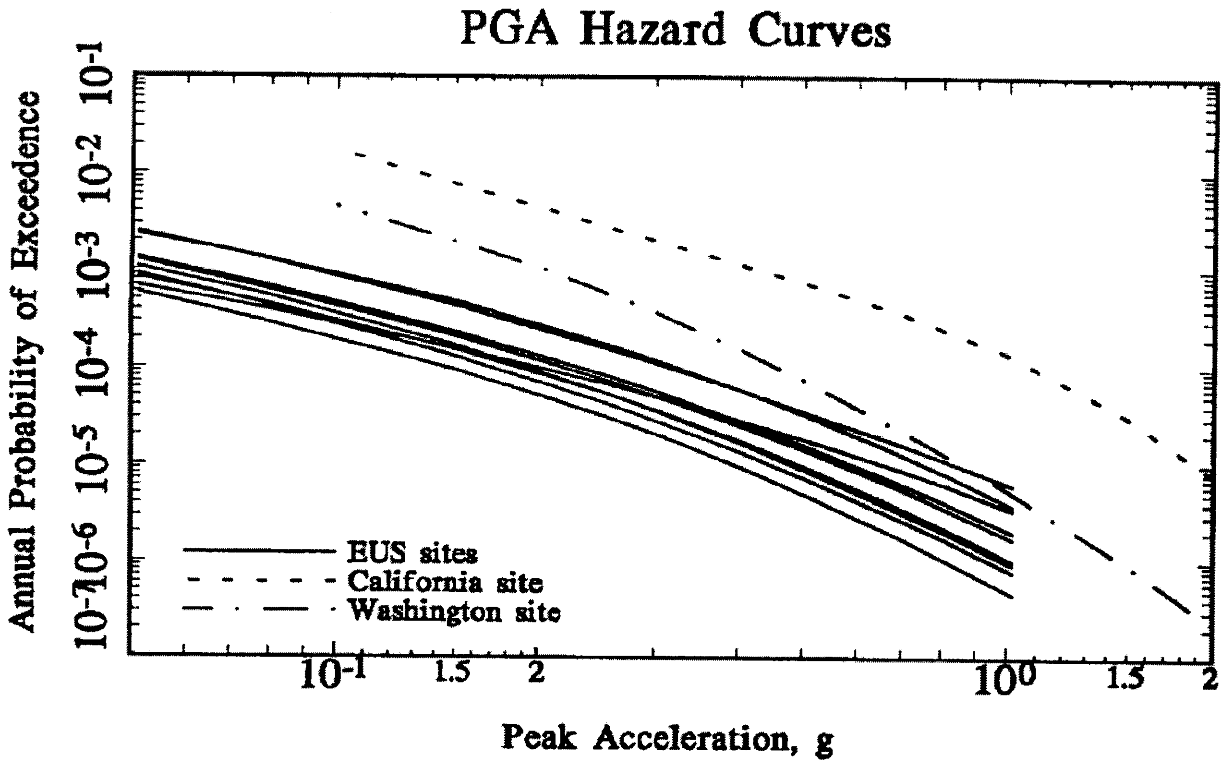


Figure V.1. Multiple seismic hazard curves for a variety of nuclear sites in the US (REI 2001).

Compared with current methods, demanding a radioactivity release of less than 500 rem as the 95th percentile of consequences for an earthquake with a return period of ten million years is quite stringent. For example, the SSE at Diablo Canyon, a site with a high seismic hazard, is 0.75g and the annual probability of exceedance is 2.5×10^{-4}. For a California Site, given the hazard curve in Figure V.1, the magnitude of an earthquake with a frequency of exceedance of 10^{-7} could be higher than 5g.

In this exercise, the Clinton site seismic hazard curves are used. This is due to the fact that the hazard curves of this site are representative of an EUS site with moderate seismicity. Additionally, a performance-based[viii] metric is used in the early site permit (ESP) (U.S. Nuclear Regulatory Commission 2006). This method led to the Clinton ESP containing detailed expressions for the seismic hazard and plant fragility curves. These expressions allow numerical calculations to be performed that illustrate the implications of seismic risk analysis within the TNF and that illustrate the benefits of seismic isolation.

If one is to use the TNF to choose seismic LBEs, this choice could take place in a few different ways. This is because, within the TNF, what constitutes a sequence or initiator is left for the applicant to decide. The end state of each sequence is to be release. The mean and 95th percentile dose of this release at a certain distance from the reactor is the consequence that must meet the F-C curve. A simplified event tree model will be used to demonstrate all the methods explored in this paper. The model consists of the initiating seismic event, a plant failure event, and a containment failure event (Figure V.2). Given plant failure, there is core damage. Given containment failure and plant failure, there is a large release (95th percentile dose > 500 rem).

[viii] In the performance-based approach, the seismic design parameters are chosen so that the total contribution from seismic accident sequences satisfies a goal. For example, in the Clinton ESP, the stated performance goal is 10^{-5} per reactor year for the core damage frequency due to seismic initiated sequences.

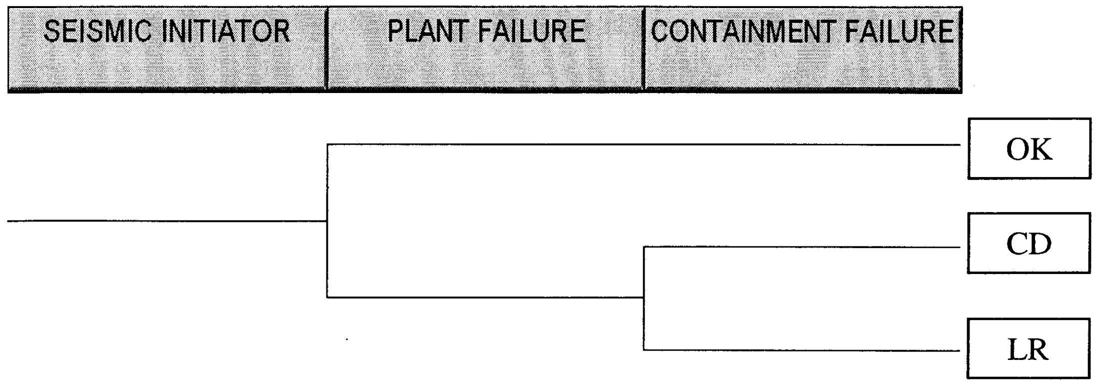


Figure V.2. A simple event tree for seismic risk assessment. End states are OK, core damage, and large release.

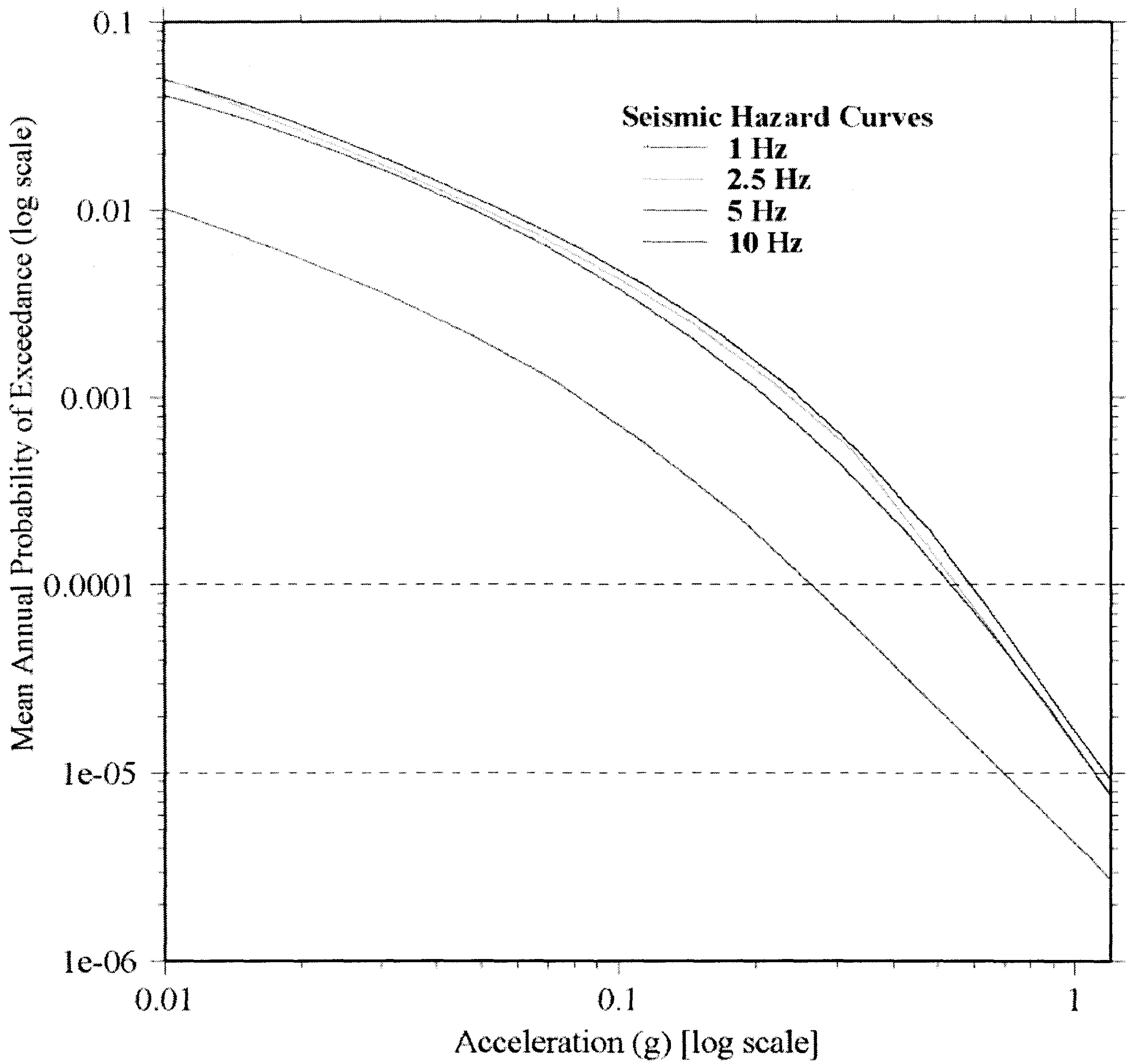


Figure V.3. Clinton site seismic hazard curves (US Nuclear Regulatory Commission 2006).

Before discussing examples of each method, the actual hazard curves (Figure V.3) and fragility curves from the Clinton ESP are introduced, as well as the seismic hazard function and fragility function used within the examples. The mathematical expression of the conservatively extrapolated hazard curves is:

$$H(a) = C\,a^K \tag{V.1}$$

where H(α) is the annual probability of exceeding acceleration α. C and K are constants determined using two points on the curve. In this case, the points used were $H(a)$=10^{-4} and 10^{-5}. This results in a 1 Hz spectral acceleration hazard curve with C=2.7x10^{-6} and K=-2.4. This extrapolates conservatively to 3.9g at an annual exceedance probability of 10^{-7}. Similarly, extrapolation of the 10 Hz spectral acceleration hazard curve results in 4.5g with an annual exceedance probability of 10^{-7}. This 10 Hz spectral acceleration hazard curve will be used to demonstrate the methodology. In addition to this power law extrapolation, a less conservative Weibull distribution is also used that fits the entirety of the hazard curve well. A complete set of data is not available so a goodness of fit test has not been performed. The hazard curve of this particular distribution is given in equation (V.2)

$$H(a) = 0.225e^{-(360a)^{0.39}} \tag{V.2}$$

This is a complementary cumulative Weibull distribution with a shape factor of 0.39 and a scale factor of 360. The front coefficient of 0.225 is necessary as the integration of exceedance frequency does not necessarily integrate to unity. This fit to the 10 Hz spectral acceleration hazard curve results in about 2.6g with an annual exceedance probability of 10^{-7}. This is significantly smaller, but it is still a very large acceleration compared with typical SSEs. The two curves are shown in Figure V.4.

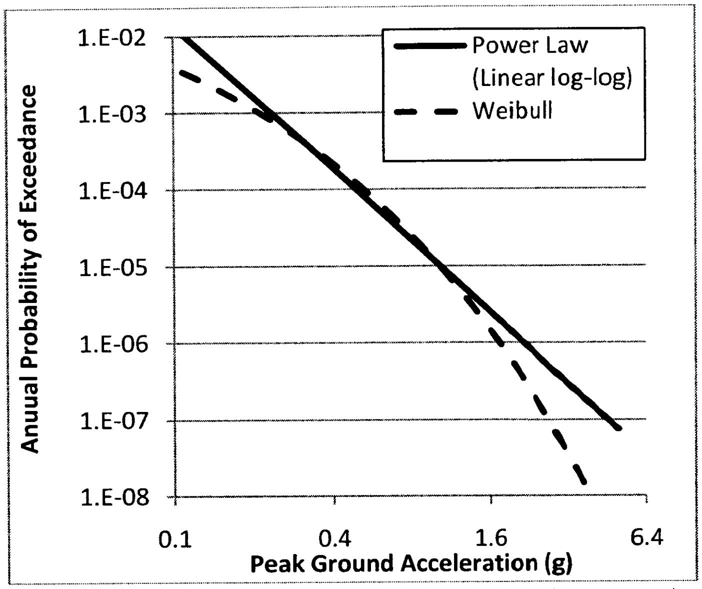


Figure V.4. The power law (linear in log-log space) and Weibull distribution fit hazard curves,

The TNF demands that all seismic events down to 10^{-7} per year be considered. With the high accelerations that are predicted, designing structures that would reduce the large release sequence frequency below 10^{-7} per year is prohibitive.

These results indicate that the TNF, as currently formulated, is impractical. The problem appears to be that there is no limit on initiating event frequency. For example, when one considers earthquakes with frequencies slightly above 10^{-7} per year the accelerations are unknown, but would certainly be very large. Demonstrating that a design meets the F-C curve would be nearly impossible.

To further elaborate on this point, consider a numerical example using the event tree described above. To calculate sequence frequencies, seismic fragilities for the plant failure and containment failure events are needed. In this model, only structural failure of the containment is considered, i.e., bypass is not included. Fragilities usually take the form of a lognormal distribution. The cumulative distribution function is defined as:

$$F(a;\mu,\beta) = \Phi\left(\frac{\ln a - \mu}{\beta}\right) \tag{V.3}$$

where Φ is the cumulative distribution of the normal function, μ=ln(median) and β=0.4 as in the Clinton ESP. For ALMR and PRISM the dominant failure mode for a seismic initiator is failure of decay heat removal. The conditional failure probability for this function is optimistic in both PRAs. It can be assumed that the mean fragility of the decay heat removal system (the most robust system) from the EBR-II PRA of 1.5g is the mean plant fragility. Using the property that the mean of a lognormal distribution is equal to $exp(\mu + \beta^2)$, μ is found to be equal to 0.24g. Alternatively, μ can be determined for a generic plant using the high confidence of low probability of failure (HCLPF) value which is commonly taken as the 1% confidence level. Using the properties of the lognormal distribution:

$$\mu = \ln HCLPF + 2.326\,\beta \tag{V.4}$$

A typical LWR has a HCLPF around 0.5g. This also gives μ=0.24g and fully defines our plant fragility curve. This curve should be approximately valid for both an LWR and an SFR without isolation (as the EBR-II does not feature isolation). It is noted that, when doing the numerical calculations in equations (V.2) and (V.4), a logarithm is taken of a value with the units of g. For example, in equation (V.4) the ln(HCLPF) is equal to ln(0.5).

For the containment failure probability, both a confinement building (for advanced designs that may not need hardened containment) and a typical PWR containment are analyzed. The confinement building will use the same fragility curve as the auxiliary building for a typical PWR with a median fragility of 0.73g (μ=-0.31g) and β=0.56. For the hardened containment a median of 1.8g (μ=0.59g) and β=0.52 are used (Kaplan, Perla, and Bley 1983).

Initiating event frequencies and magnitudes must also be selected. The method that would be the most consistent with the TNF is to choose several ranges on a seismic hazard curve and, then, define initiating events based on that seismic hazard. The initiating events should appropriately cover all events down to a mean frequency of 10^{-7} per year. The most conservative way to define the initiating events would be to take the highest exceedance frequency of the range to be covered and the highest acceleration of the range to be covered and use these values as the initiating event that must be satisfied. For example, one could choose to take the ranges OBE (0.1g)-0.5g, 0.5g-1g, 1g-2g, and 2g-4g. There would then be four initiating events. These events would have the frequency of the hazard at the beginning of the range, and the acceleration of the end of the range. This is commensurate with taking the highest frequency in the bin and the most limiting consequences in the bin.

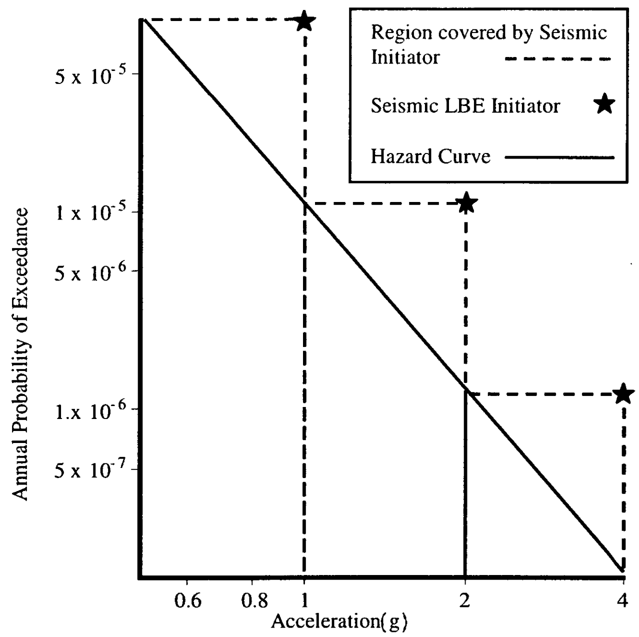
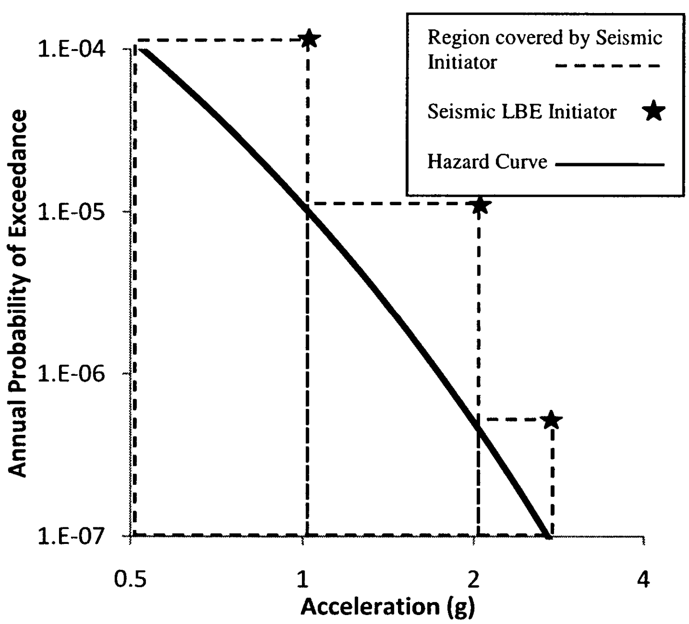


Figure V.4. Schematic of LBE initiator selection using equation (V.1).



FigureV.6. Schematic of LBE initiator selection using equation (V.II)

Figure V.5 shows this graphically for the three larger initiators using the extrapolated hazard curve from the Clinton site for 10 Hz. It is noted that this extrapolation is conservative outside of the range of 10^{-4} and 10^{-5} due to the concavity of the seismic hazard curve. Figure V.5. shows this same selection for the Weibull distribution fit to the hazard curve. This is a better estimate of the actual hazard especially at higher frequencies. It is noted that the first two initiators are more or less the same. This is because the extrapolated hazard curve is accurate between 10^{-4} and 10^{-5} per year. The final initiator covering the range from 2g to the end of the hazard curve is much smaller for the Weibull distribution (2.6g) than for the power law extrapolation (4g).

This is the method used in both the ALMR and PRISM PRAs (Hackford 1986; El-Sheikh 1994). However, the frequency of the initiators and the fragility of structures, systems, and components in these PRAs seem optimistic. For example, the ALMR PRA estimates the annual probability of exceedance of 2g as 1.3×10^{-7}. In comparison to the extrapolated hazard curve, this is an order of magnitude less frequent. In comparison to the Weibull distribution, this is about one fourth as frequent. The seismicity of the hypothetical site proposed in the PRA is supposed to cover 90% of all US sites. An example of optimistic seismic fragility in the ALMR PRA is that RVACs fails with a probability of 10^{-5} given an initiating seismic event with a PGA of 2g. This indicates that the robustness of the building is significantly higher than a typical containment structure for a PWR.

With these optimistic estimates of seismic hazard, the ALMR and PRISM do meet the F-C curve of the TNF as seen in the LBE map of Figure III.1. The most frequent seismically induced sequence resulting in a large release is 2.2×10^{-8} per year in the PRISM PRA and 10^{-13} per year in the ALMR PRA (it is noted here that the ALMR PRA is much more optimistic about decay heat removal fragility than the PRISM PRA).

If the extrapolated hazard curve from the Clinton site is used with the conservative method for initiator selection described above, the most stringent initiator is found to have a frequency of $1.3\mathrm{x}10^{-6}$ and an acceleration of 4g. This is represented by the star furthest right on figure V.4. This initiating event frequency is then multiplied by the failure probability of the plant and the probability of failure of the containment function. This can then be determined for any given seismic initiator with a PGA of a using equation (V.5). This equation arises from the event tree model shown in figure V.2.

$$Large\ Release\ Freq. = Initiating\ Freq.(a) * F_{plant}(a) * F_{containment}(a) \qquad \text{(V.5)}$$

Where F_{plant} and $F_{containment}$ are the conditional failure probabilities determined for the plant and containment function calculated using equation (V.3). When the conditional probability of large release for a given a is mentioned, it refers to the product of F_{plant} and $F_{containment}$. Using the stringent initiator described above and a hardened containment as inputs into equation (V.5) gives a point estimate for large release of $1.2\mathrm{x}10^{-6}$ per year. This means the conditional probability of a large release given a 4g acceleration is 0.94 with hardened containment. This shows that these large earthquakes will almost certainly fail all engineering measures implemented. When the same initiator is used as an input with confinement instead of containment, the conditional failure probability is nearly unity.

These conditional probabilities of large release assume independence between plant failure and containment failure. Only seismically induced failure of the containment is considered without consideration of internal pressure loads. Clearly, with this point estimate, the 95th percentile estimate will surely be above the F-C curve of the TNF. This causes us to explore other methods for choosing the seismic initiators, or to seek an alternative goal for seismically initiated events than for internal events.

Another, perhaps more appropriate, method that would use a similar technique would be to cover a variety of ranges by selecting frequency ranges. For example, one could take 10^{-1} per year to 10^{-3} per year for "frequent" earthquakes, 10^{-3} per year to 10^{-5} per year for "infrequent" earthquakes, and 10^{-5} per year to 10^{-7} per year for "rare" earthquakes. Doing this does not really change the method, but would change the frequency and magnitude of the seismic initiators used as input to the event tree. This change is fairly insignificant and would not change the results of the analysis presented above.

One could use a similar method to the one described in figures V.4 and V.5, but instead of choosing points above the hazard curve to cover sections of the curve, one would choose points on the curve. That is, one would choose an initiator with a certain PGA and use the exceedance frequency as the initiating frequency. This is certainly less conservative than the other method proposed but would give a better estimation of the actual core damage and large release frequencies. Unfortunately, this change in method does not make a typical reactor any more likely to meet the standard. Even if the calculated large release frequency is just above 10^{-7}, this method would not succeed in meeting the goal set forth by the F-C curve.

Seismic initiator selection is shown in Table V.1 using four ranges that each covers an order of magnitude. The higher frequency would be used in the conservative method, and the lower frequency would be used if the initiator frequencies are chosen from the curve. We see that the extrapolated hazard curve has significantly larger initiators than the Weibull fit hazard curve for events rarer than 10^{-5} per year.

Table 1. Seismic LBE Initiators. The range covered by the initiator is in the leftmost column. The high number is the initiating event frequency if the conservative method is used. The low number is initiating event frequency if the hazard curve itself is used.

Initiator Frequency Range	Initiator Peak Ground Acceleration: Power Law Extrapolation	Initiator Peak Ground Acceleration: Weibull Distribution Fit
$10^{-3} - 10^{-4}$ per year	0.5g	0.5g
$10^{-4} - 10^{-5}$ per year	1.0g	1.0g
$10^{-5} - 10^{-6}$ per year	2.2g	1.7g
$10^{-6} - 10^{-7}$ per year	4.5g	2.6g

Once again, using a numerical example, an acceleration of 4.2g is selected as the initiating seismic event and the associated exceedance frequency on the extrapolated Clinton site hazard curve, and thus the initiating frequency used in the event tree, is 1.3×10^{-7} per year. This is used as the input value in equation (V.5). The conditional probability of large release given a hardened containment for this initiator is 0.96. The frequency of the large release for this seismic LBE is then 1.2×10^{-7}. It is observed that this method calculates a frequency approximately an order of magnitude lower than the conservative method where initiators are chosen from above the hazard curve. However, the frequency is still above 10^{-7} per year and would thus be included as an LBE and the TNF goal would not be met.

One would also like to know the impact of a smaller, higher frequency earthquake at 2g. Inputting this value into equation (V.5) gives a conditional probability of large release is 0.51 with traditional containment and 0.84 with a confinement building. This in combination with the annual probability of exceedance for a 2g earthquake of 1.3×10^{-6} retrieved from the Clinton site hazard curve results in a frequency of large release of 6.5×10^{-7} per year with containment and 1.1×10^{-6} per year with confinement. It is noted again that these values will not meet the TNF goals. This means that earthquakes may have a substantial impact under the TNF well before the 10^{-7} cutoff.

The F-C curve is not going to be met by typical reactor designs for very large, rare earthquakes. Trying to design a reactor to meet this standard would be expensive and burdensome. Alternatively, the design would have to have sufficiently small fuel inventory such that a 500 rem release is not possible no matter the damage. This leads us to explore other options with regards to what the goal should be or how rare an external initiating event one should consider when building a plant.

V.A. Earthquake frequency limit

If one limits the external initiator frequency to something less stringent than 10^{-7}, such as 10^{-5}, one arrives at a different, smaller earthquake as the limiting factor in the analysis. If one uses the proposed limit of 10^{-5} per year as the initiating frequency with the Clinton site hazard curve one finds the corresponding acceleration to be 1.0g. Using this value in equation (V.5) yields a conditional probability of large release of 3.5×10^{-2} for a typical reactor with containment and 0.19 for a reactor with a confinement building around the nuclear island. Both of these designs would still not meet the F-C curve because the large release frequency is above the limit at $3.5x10^{-7}$ and $1.9x10^{-6}$ per year respectively (just barely in the case of the reactor with containment). This size of quake is much larger than those typically considered for SSEs, but is significantly smaller than quakes one or two orders of magnitude rarer.

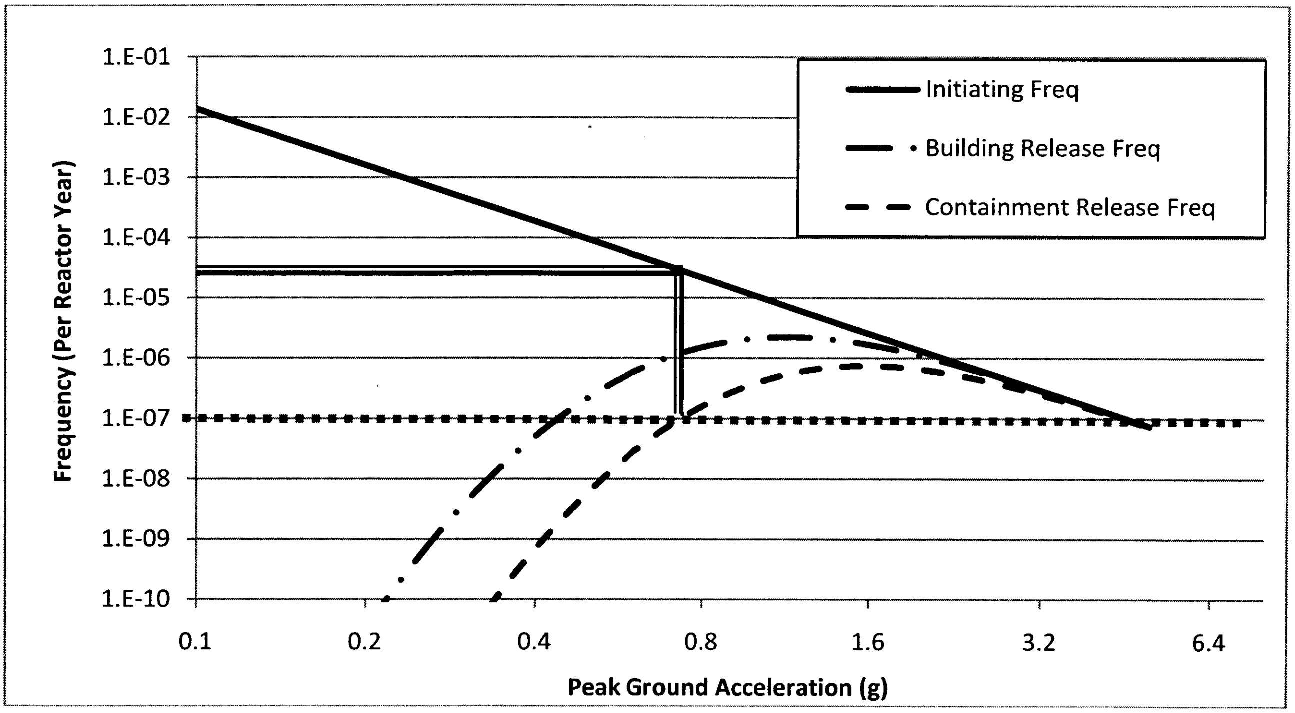


Figure V.6. Peak Ground Acceleration vs. Frequency using the power law extrapolation. The dotted horizontal line highlights the 10^{-7} per year cutoff of the TNF. The dash-dot line shows a plot of equation (V.5) using the extrapolated seismic hazard curve for a plant with confinement. The dashed line shows a plot of equation (V.5) for a plant with a containment building. The thin double line shows the frequency cutoff that would allow a plant with containment to be licensed.

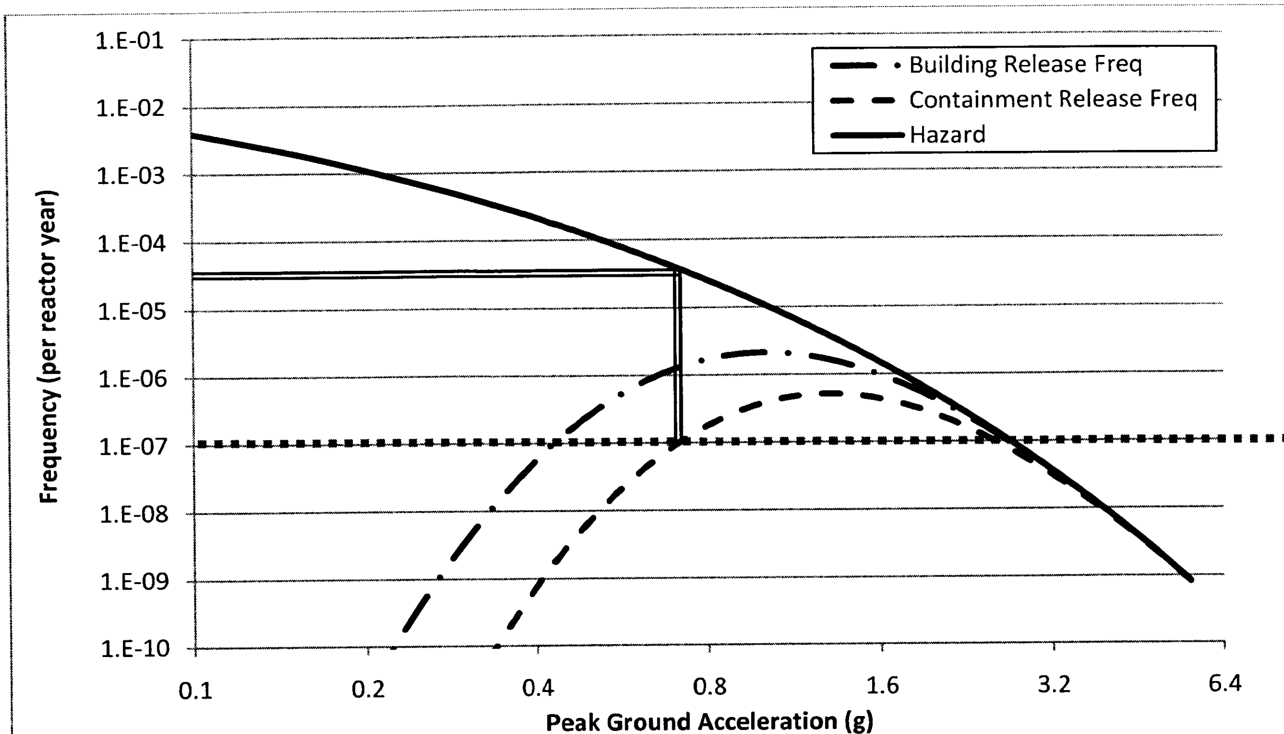


Figure V.7. Peak Ground Acceleration vs. Frequency using the Weibull fit. The dotted horizontal line highlights the 10^{-7} per year cutoff of the TNF. The dash-dot line shows a plot of equation (V.5) using the extrapolated seismic hazard curve for a plant with confinement. The dashed line shows a plot of equation (V.5) for a plant with a containment building. The thin double line shows the frequency cutoff that would allow a plant with containment to be licensed.

Figures V.6 and V.7 show the frequency of the initiating event from the extrapolated hazard curve and the Weibull fit (respectively) as well as the frequency of release for the two different cases considered. From the graph it is found that an initiating event frequency cutoff of $3x10^{-5}$ per year would allow the example design with containment to be licensed. This is true for either method of representing the hazard curve as this is the region where both the extrapolated hazard and the Weibull fit best match the actual hazard curve. This is illustrated using a double thin line that shows where the frequency of large release is below the 10^{-7} cutoff and the associated frequency from the hazard curve. Likewise, a frequency cutoff of $1.5x10^{-4}$ per year would allow the design without containment to be licensed. Compare these frequencies to SSEs which tend to have a frequency around 10^{-4} per year. With proper design features, meeting the TNF goals with an initiating event cutoff at 10^{-5} per year may be possible. This is in contrast a 4g earthquake as the conditional probability of failure for even robust systems is nearly unity and as such adding another system would only slightly reduce the frequency of large release.

A design option that is often explored to reduce seismic risk is seismic isolation. This design feature is used in the PRISM and ALMR design and has a cited maximum capacity of 1.2g according to the PRISM PRA (Hackford 1986). Although a more robust and complete analysis is necessary to show the effect of seismic isolators on plant fragility, a numerical example has been performed. The cited maximum capacity for the isolator has been assigned to the 10^{th}, median, and 90^{th} percentile of lognormal fragility curves with β=0.5 and β=0.3. In this analysis, it is assumed that isolator failure is necessary for core damage. This adds a term to equation (V.5) for the probability of isolation failure leading to equation (V.6).

$$Large\ Release\ Freq. = \\ Initiating\ Freq.(a) * F_{plant}(a) * F_{containment}(a) * F_{isolation}(a) \qquad (V.6)$$

This optimistic assumption may not be a good one for all reactor types. In particular, reactivity insertion due to vertical movement poses a threat to fast reactors. The frequency of large release shown in Figures V.8-V.15 has been calculated by multiplying the conditional failure probability of the plant, isolation, and containment (or confinement) building with the initiating frequency for the given acceleration using equation (V.6). The solid line in all of the figures is the extrapolated hazard curve from the Clinton site using either equation (V.1) or equation (V.2). All failures were assumed to be independent. This is also an optimistic assumption; failure of seismic isolation may lead to failure of the isolated structure. This effect would mean that, if an isolator is less robust than the building it is intended to isolate, it would be a disadvantage to implement.

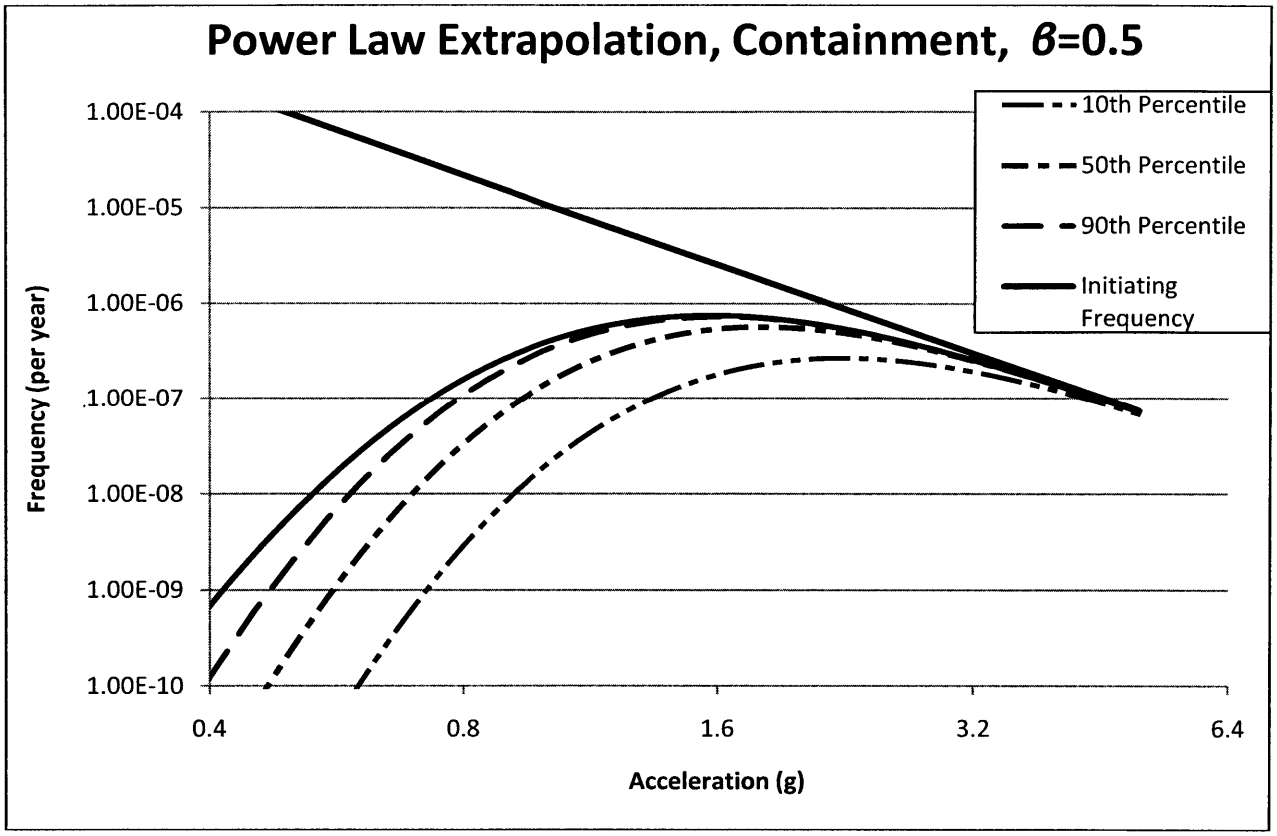


Figure V.8. The frequency of large release for a typical reactor with isolation and a hardened containment dome as calculated in equation (V.6). Each curve represents a different calculation of the isolation system fragility curve. The fragility for the isolation is defined by assigning 1.2g to the percentile failure shown in the legend and β=0.5.

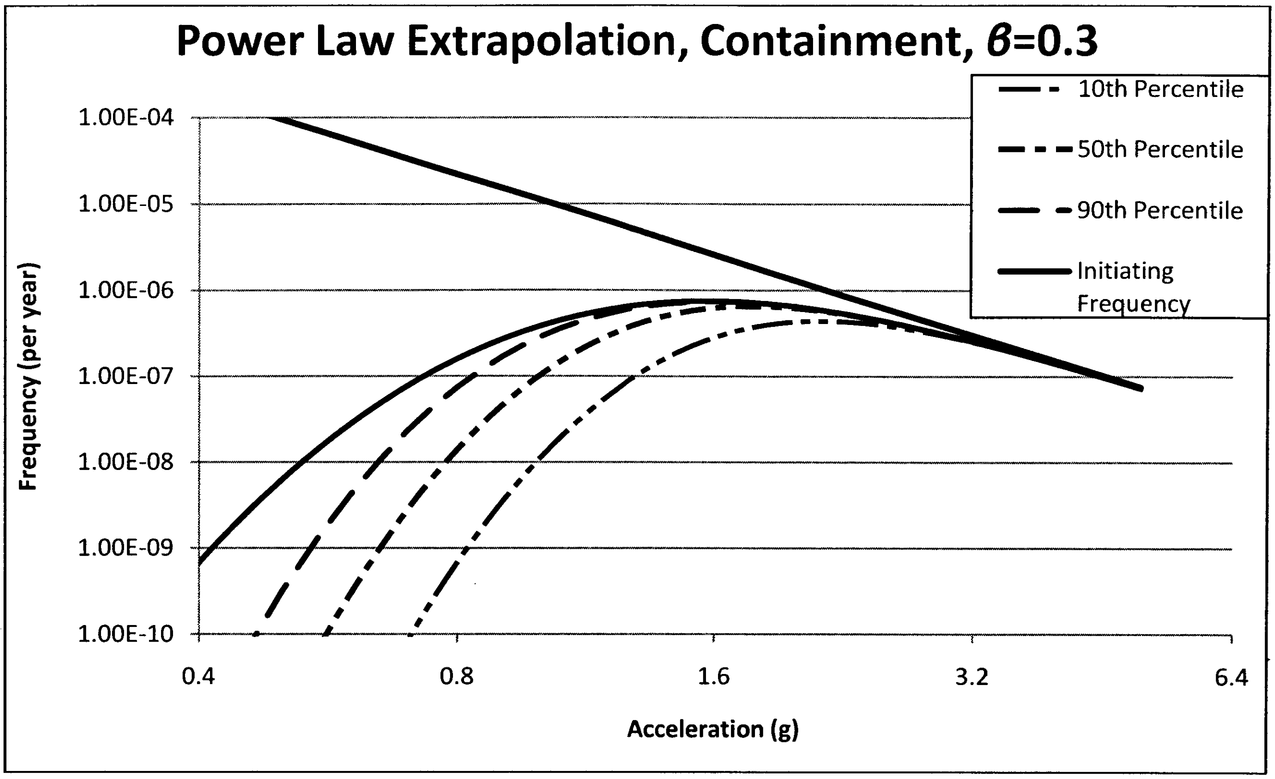


Figure V.9. The frequency of large release for a typical reactor with isolation and a hardened containment dome as calculated in equation (V.6). Each curve represents a different calculation of the isolation system fragility curve. The fragility for the isolation is defined by assigning 1.2g to the percentile failure shown in the legend and β=0.3.

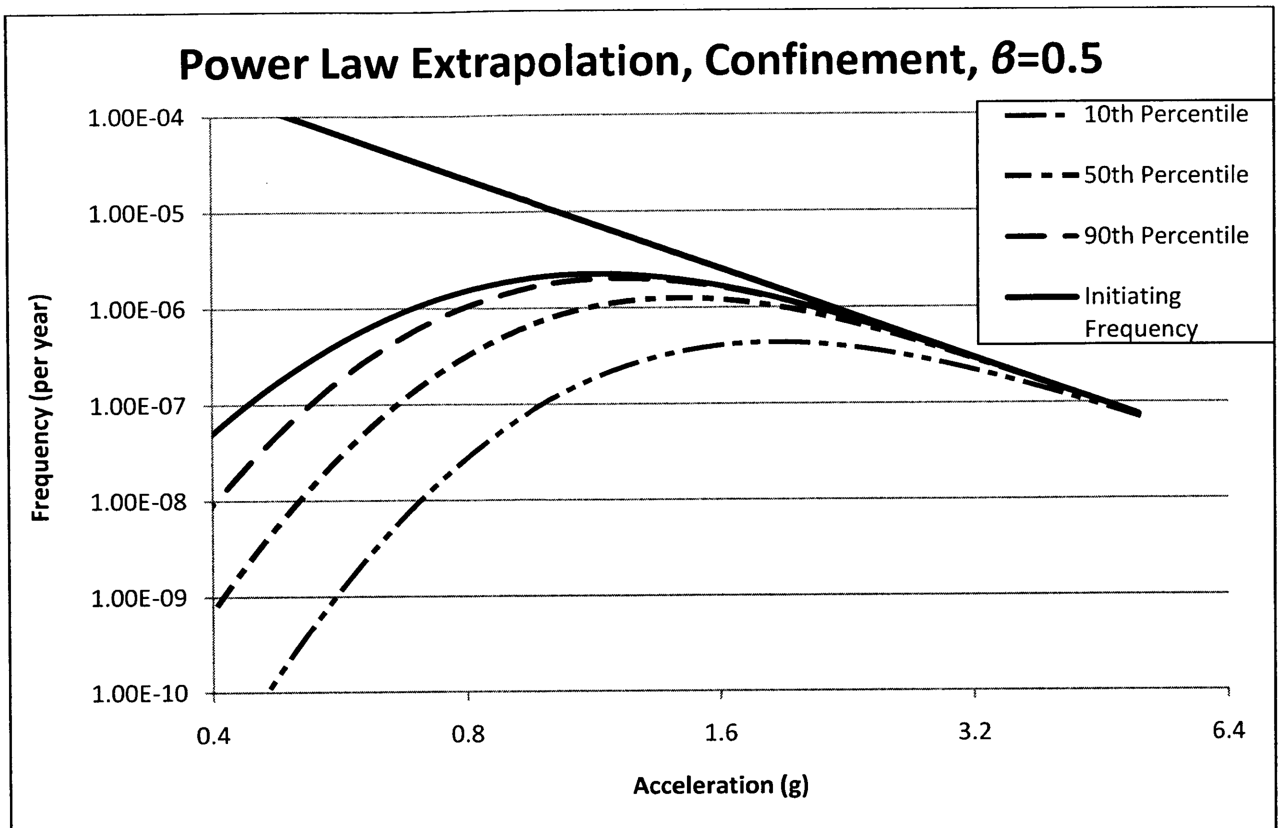


Figure V.10. The frequency of large release for a typical reactor with isolation and a confinement building as calculated in equation (V.6). Each curve represents a different calculation of the isolation system fragility curve. The fragility for the isolation is defined by assigning 1.2g to the percentile failure shown in the legend and β=0.5.

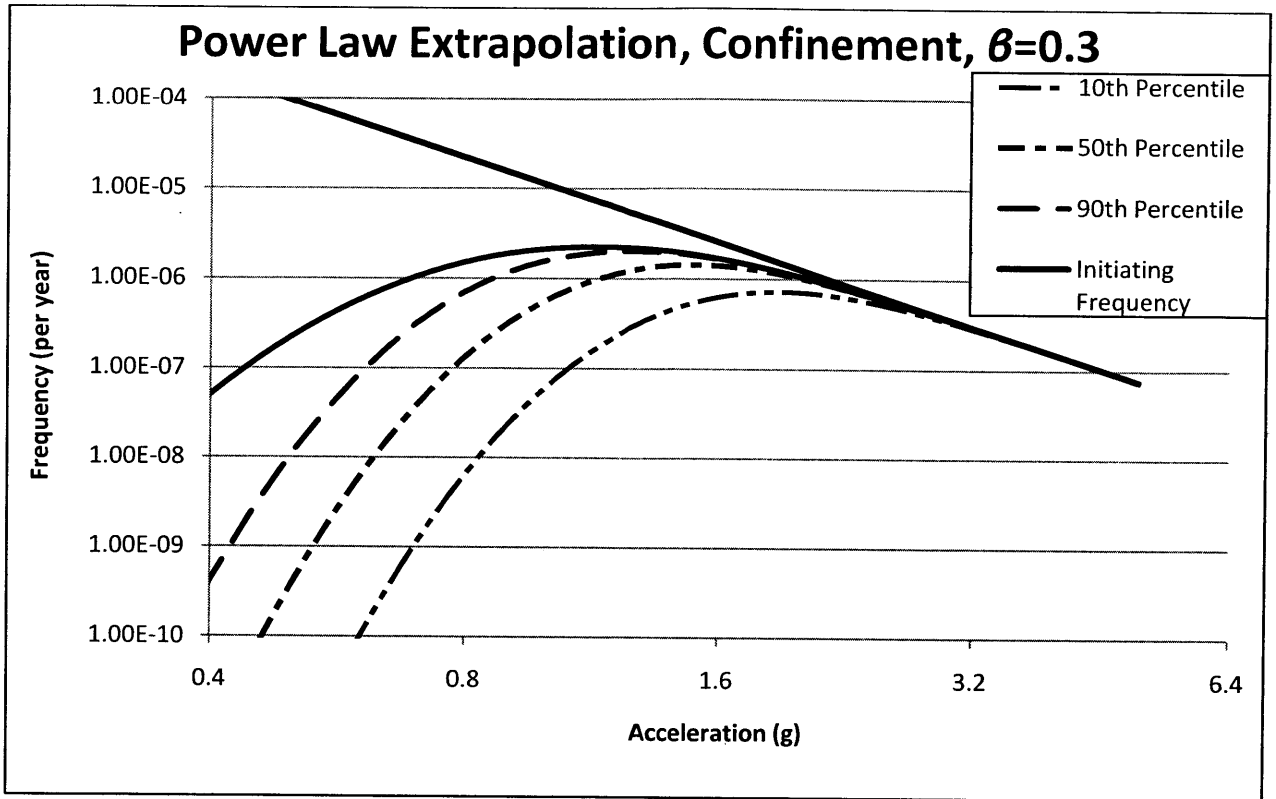


Figure V.11. The frequency of large release for a typical reactor with isolation and a confinement building as calculated in equation (V.6). Each curve represents a different calculation of the isolation system fragility curve. The fragility for the isolation is defined by assigning 1.2g to the percentile failure shown in the legend and β=0.3.

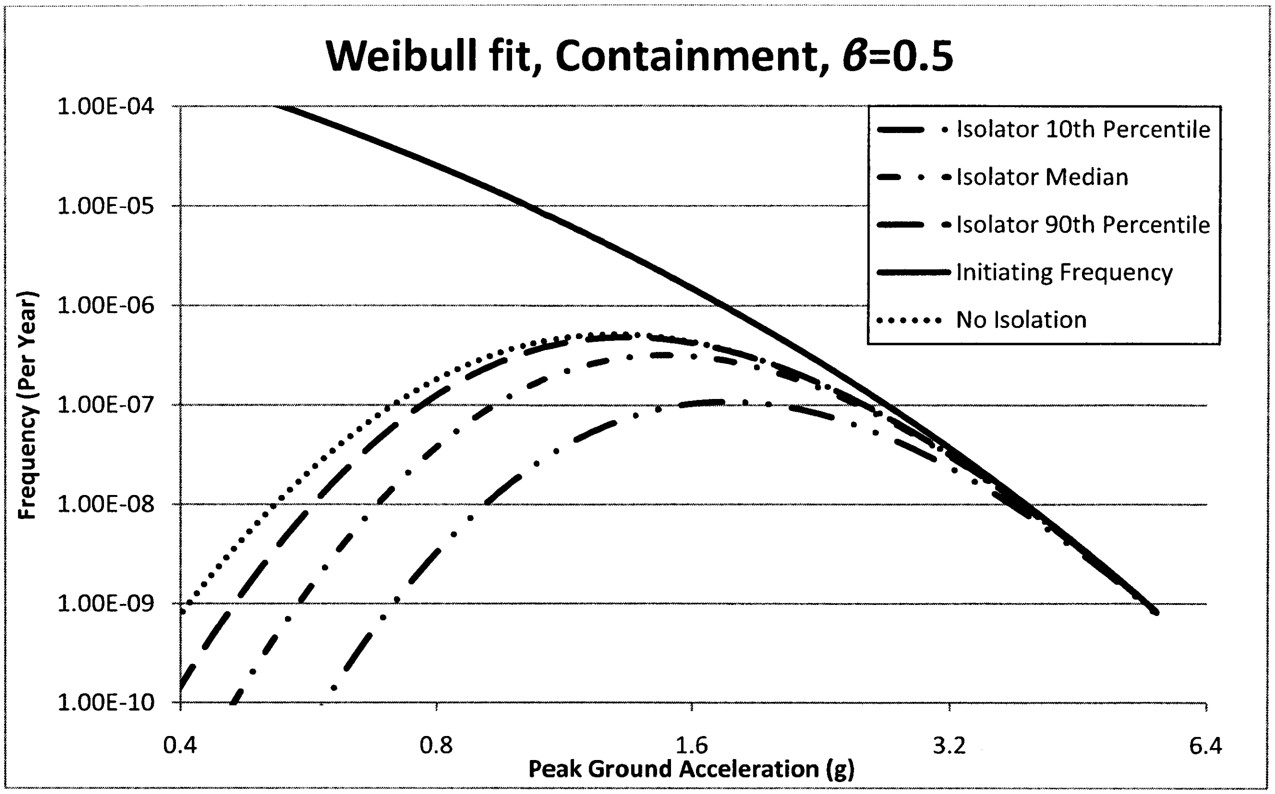


Figure V.12. The frequency of large release for a typical reactor with isolation and a containment building as calculated in equation (V.6). Each curve represents a different calculation of the isolation system fragility curve. The fragility for the isolation is defined by assigning 1.2g to the percentile failure shown in the legend and β=0.5. The Weibull fit of the hazard curve is used.

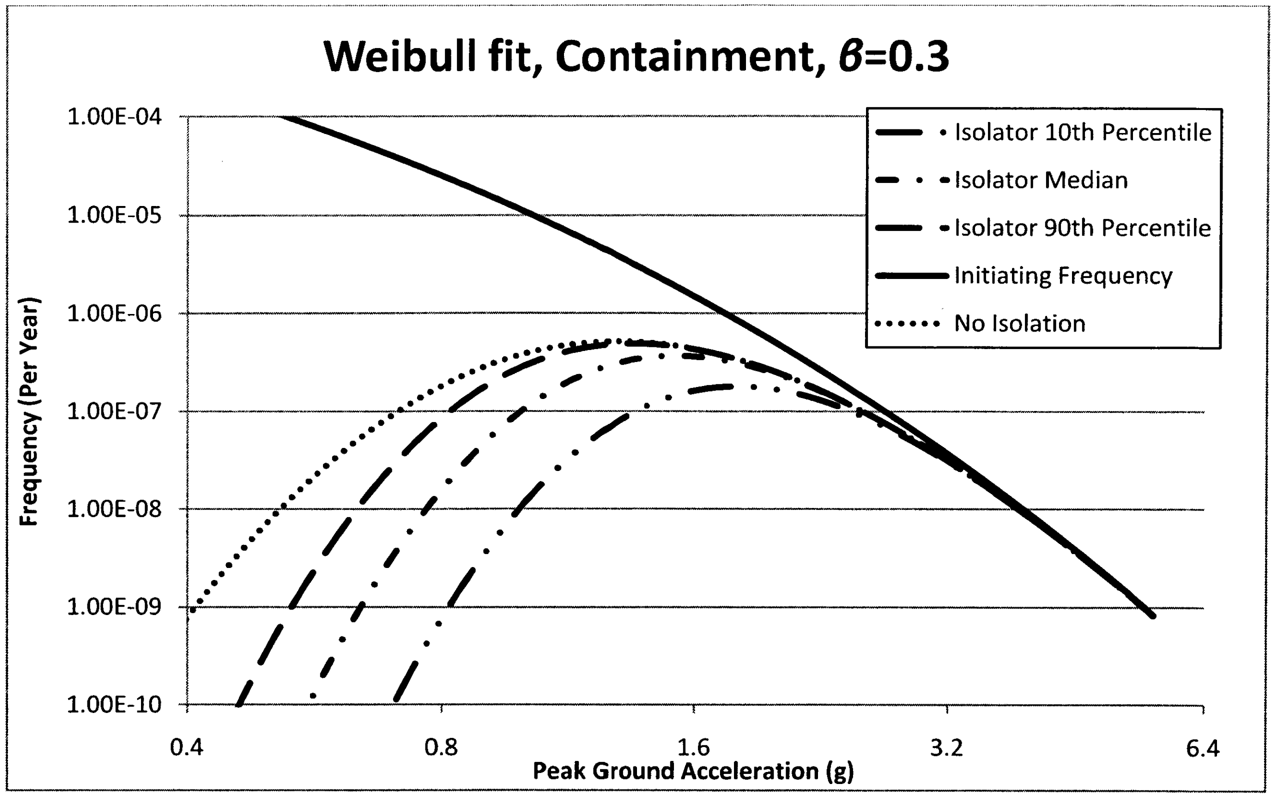


Figure V.13. The frequency of large release for a typical reactor with isolation and a containment building as calculated in equation (V.6). Each curve represents a different calculation of the isolation system fragility curve. The fragility for the isolation is defined by assigning 1.2g to the percentile failure shown in the legend and β=0.3. The Weibull fit of the hazard curve is used.

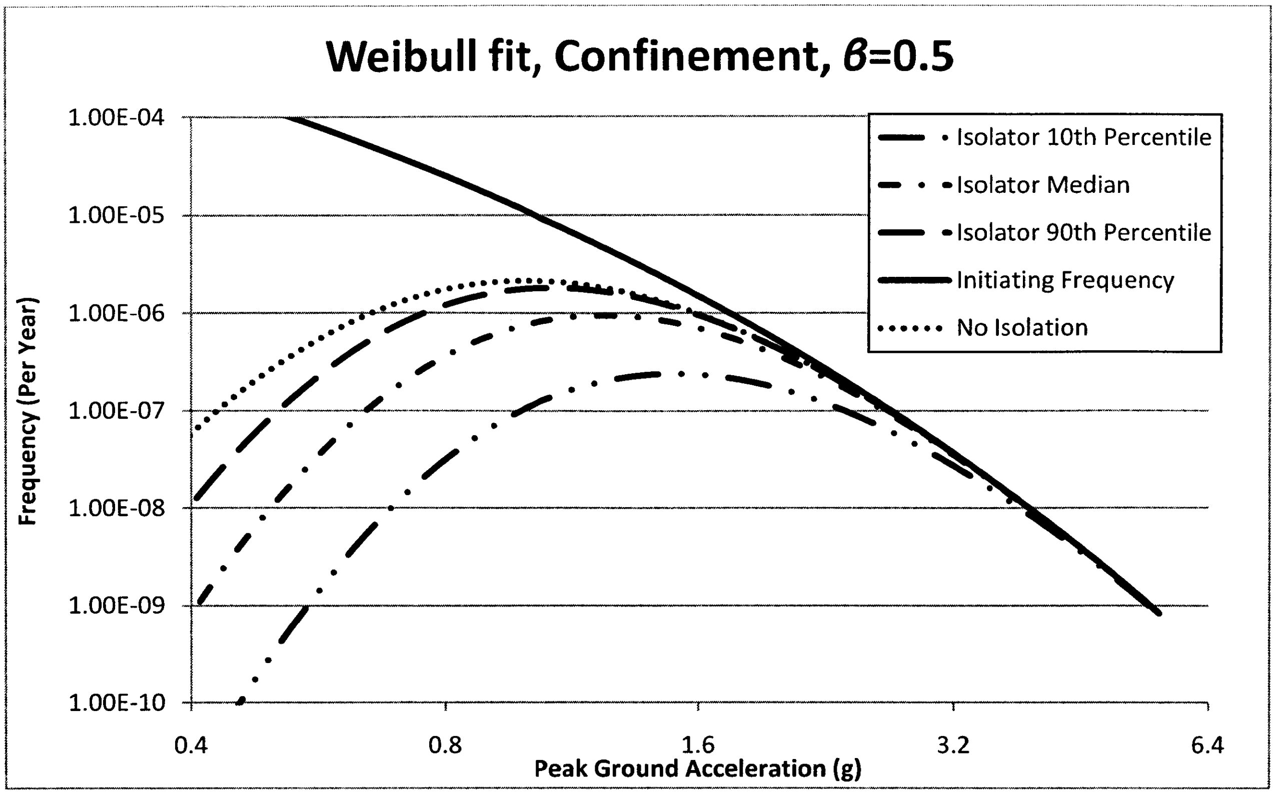


Figure V.14. The frequency of large release for a typical reactor with isolation and a confinement building as calculated in equation (V.6). Each curve represents a different calculation of the isolation system fragility curve. The fragility for the isolation is defined by assigning 1.2g to the percentile failure shown in the legend and β=0.5. The Weibull fit of the hazard curve is used.

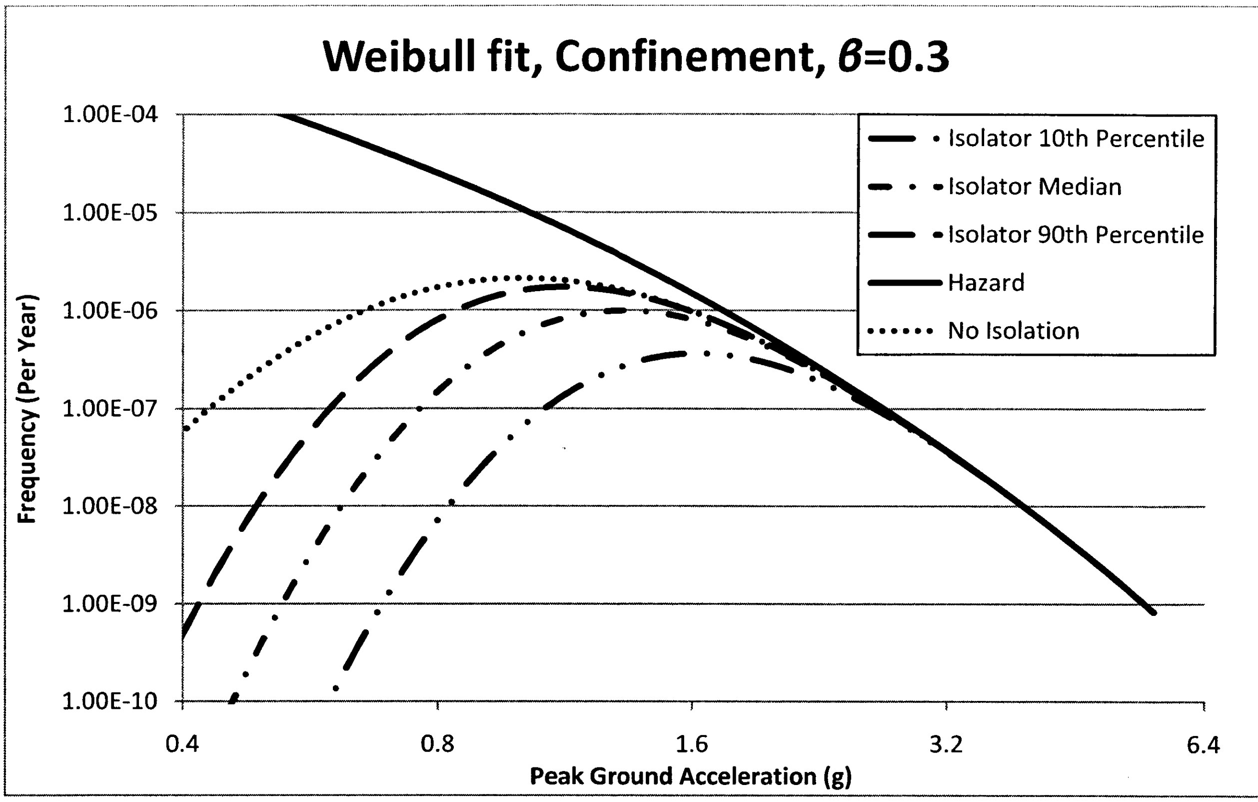


Figure V.15. The frequency of large release for a typical reactor with isolation and a confinement building as calculated in equation (V.6). Each curve represents a different calculation of the isolation system fragility curve. The fragility for the isolation is defined by assigning 1.2g to the percentile failure shown in the legend and β=0.3. The Weibull fit of the hazard curve is used.

As one can see from the figures, seismic isolation would be unlikely to solve issues regarding earthquakes of 10^{-6} per year and rarer. The only scenario that nearly meets the goal is using the Weibull fit hazard curve, containment, and isolation failure of 1.2g as the 10^{th} percentile. This is the most optimistic scenario and as mentioned before is calculated using optimistic assumptions of independent failure of the plant, containment and isolation. It is observed that all of the curves representing the large release frequency pinch together with the hazard curve; usually, this is well before the 10^{-7}per year cutoff. This shows that there is a near unity conditional failure probability given these rare seismic initiating events. Even with containment and the optimistic estimation of 1.2g as the 10^{th} percentile failure probability, our test case does not meet the goal.

There is certainly a benefit to using isolation and that this benefit increases with decreasing frequency of the seismic initiator. This may make implementing a 10^{-5} per year initiating event cutoff more manageable. Figures V.10, V.11, V.14, and V.15 show that with a robust enough isolation system, a confinement building may be enough to meet the TNF goal, if the external initiating event frequency is limited to 10^{-5} per year. It is also noted that this cutoff may allow for a tradeoff analysis to be done between a hardened containment dome and confinement with isolation for some reactor designs.

This type of initiating event cutoff frequency is not consistent with the TNF which specifically states that all external initiators should be considered down to the cutoff. However, limiting the frequency of external initiators gives a little more realism to meeting the goals. When one starts to consider external initiators that occur with a frequency of 10^{-7} per year, one has stopped considering what could be a reasonable threat to safety and has started considering truly incredible events that pose a serious threat to all supporting systems and buildings. By comparison, an internal initiator with a frequency of 10^{-7} per year may simply constitute the failure of two systems, or failure of a single highly reliable passive system. Inclusion of these within the licensing basis makes sense, because, if the probability of failure has been miscalculated, such an event may be significantly more likely than estimated. Additionally, one may design a feature to prevent damage given this rare internal event. Rare seismic events cannot really be designed against for most traditional reactors.

It is worth noting that an earlier proposal by the NRC staff to risk-inform the regulations designated initiating events with frequencies less than 10^{-5} per year as being "rare." For such events, the conditional probabilities of core damage and containment failure are allowed to be as high as unity. In this proposal, there was no consideration for the dose of each sequence (US Nuclear Regulatory Commission 2000).

Additionally, it is stated in WASH-1270 that an "aiming point" of one in a thousand years for the fleet should be the edge of the design envelope. In this document, a fleet of one thousand plants is considered resulting in a proposed design envelope cutoff of 10^{-6} per reactor year. WASH-1270 is careful to note that this is not meant as a hard limit to be demonstrated for a given plant but a design objective (US Atomic Energy Commission 1973).

The NRC staff has been working on risk-informing the large loss-of-coolant-accident (LLOCA) rule. The idea is to propose a "transition break size" (TBS) such that pipe breaks greater than the TBS would be treated as beyond-design-basis events. In this context, the Commission stated: "a frequency of 1 occurrence in 100,000 reactor years is an appropriate mean value for the LOCA frequency guideline for selecting the maximum design-basis LOCA since it is complemented by the requirement that appropriate mitigation capabilities, including effective severe accident mitigation strategies, must be retained for the beyond design-basis LOCA category." (US Nuclear Regulatory Commission 2004; US Code of Federal Regulations 2004). This statement indicates that the Commission considers the frequency of 10^{-5} per reactor year as an appropriate lower bound for the initiating events that should be included in the design basis. The NRC has also assessed the impact of seismic events on the TBS rulemaking and has found that the contribution to large-break LOCAs from earthquakes with an annual exceedance probability of 10^{-5} per year to be negligible and that 10^{-6} per year earthquakes are major contributors to LOCA due to support failure for large piping (US Nuclear Regulatory Commission 2008). They concluded that seismic hazard does not appear to greatly affect the failure frequency of pipes larger than the proposed TBS.

CHAPTER VI – CONCLUSIONS AND RECOMMENDATIONS

Sodium cooled fast reactors (SFRs) are considered as a novel example to exercise the Technology Neutral Framework (TNF) proposed in NUREG-1860. One reason for considering SFRs is that they have historically had a licensing problem due to postulated core disruptive accidents (CDAs). The TNF provides a method to argue that they should not be designed against CDAs to the detriment of reactor operations. In implementing the TNF with existing SFR PRAs, several key conclusions have been reached regarding future implementation:

- The review of the ALMR and PRISM PRAs revealed that several of the failure probabilities are quite optimistic, i.e., they are too low.
- As completed, the ALMR and PRISM PRAs show these designs comply with the TNF for internal events (PWRs do not comply). Even if PWR numbers are used for scram and pump seizure, this result is unchanged.
- A more detailed and realistic PRA that includes fire (and other) initiators is necessary before it is attempted to satisfy the TNF.
- A prescriptive approach to LBE construction such as functional event trees is a useful development to prevent applicants from arbitrarily splitting sequences to arrive at apparently lower frequencies.
- Although core disruptive accidents are found to be well below the TNF cutoff, there is still a concern that this accident could be used as the deterministic LBE, thus negating the benefit of the very low frequencies associated with these accidents.
- Traditional risk metrics are not compatible with LBEs.

- Limit exceedance factor is a new measure designed to be used with the TNF or other quantitative risk metrics and reveals that some systems thought to be important using other importance measures may have a significant amount of margin.
- Typical designs of SFRs cannot meet the TNF due to the requirement of including sequences initiated by very rare earthquakes. This conclusion also holds for PWRs.
- Seismologists do not quantify seismic risk to very low frequencies. Extrapolation reveals that these rare accelerations seismic events may have huge accelerations.
- It is recommended that a frequency cutoff be established for external events, as these events may pose a significant threat to all systems and may not be practical to design against.

Functional event trees are developed as a tool to allow different designs to be compared on an equal basis. Functional event trees are useful within the TNF as a method for the selection of Licensing Basis Events (LBEs) which take the place of traditional Design Basis Accidents. The ALMR and PRISM designs are considered, and for internal events only both are found meet the goals of the TNF that LWRs typically would not. A more thorough analysis would be needed to prove that this is the case.

Considering these goals have been met, a method for improving economics is proposed where systems of low risk-importance are identified as candidates for removal, simplification, or removal from safety grade. Standard importance measures are not directly useful in the TNF for this task. An importance measure that can be used directly with LBEs, Limit Exceedance Factor (LEF), is introduced that measures the margin in system failure probability. Some systems that appear to be of high risk-importance with standard importance measures are revealed to have large margins in their failure probability.

Seismic events are found to dominate risk for the designs considered. The seismic analysis done for these designs is found to be optimistic in comparison to actual seismic hazard and component fragilities. Using the seismic hazard and fragilities from a typical reactor, and from Experimental Breeder Reactor-II, a method for analyzing seismic events in parallel to the method used for internal events is examined. The result of this analysis is that the goals of the TNF cannot be met by typical SFRs or PWRs for the seismically initiated LBEs. The effect of seismic isolation to reduce LBEs that result in large release is analyzed and found to be insufficient to reach the TNF goal. This is because very rare seismic events must be considered that cause the failure probability of engineered systems to be nearly unity. Limiting the initiating seismic event frequency to be considered (e.g. 10^{-5}) is proposed as a solution that is commensurate with current practices.

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APPENDIX A – GENERAL SOLUTION TO LEF FOR AN END STATE THAT IS THE SUM OF SEQUENCES

This is a general case extending from the two sequence case presented in Section IV.B. First, some variable definitions:

$$\mathrm{Fr(end\,state)} = R \tag{A.1}$$

$$\mathrm{Regulatory\ Limit\ of\ Fr(end\,state)} = R_{threshold} \tag{A.2}$$

$$\mathrm{Pr(SSC^{k}\ failure\ in\ accident\ sequence\ i)} = q_i^k \tag{A.3}$$

For any SSC^k, the end-state frequency is a linear function of that SSC's failure probability, i.e.,

$$R = \sum_i (a_i\, q_i^k) + b \tag{A.4}$$

where the sum is over all accident sequences i containing SSC^k. It is important to note that q_i^k may be different in each accident sequence depending on the failure of other components or the initiating event. It is also noted that b is the sum of the frequencies of all accident sequences not containing failure (or explicit success) of SSC^k. One can then write

$$b = R\,|_{q=0} \tag{A.5}$$

where q=0 means that all q_i^k are set equal to zero. LEF is now introduced:

$$R_{threshold} = \sum_i (a_i \, LEF^k \, q_i^k) + b \tag{A.6}$$

This equation must be modified because $LEF^k q_i^k$ in any accident sequence may not exceed unity. We, therefore, write

$$R_{threshold} = \sum_i (a_i \, \min(1, LEF^k \, q_i^k)) + b \tag{A.7}$$

To solve for LEF^k, subtract equation (A.4) from equation (A.7)

$$R_{threshold} - R = \sum_i (a_i \, \min(1, LEF^k q_i^k)) - \sum_i (a_i q_i^k) \tag{A.8}$$

For those SSCs where $LEF^k \, q_i^k$ is always less than unity this simplifies further to

$$\frac{R_{threshold} - R}{\sum_i (a_i\ q_i^k)} + 1 = LEF^k \tag{A.9}$$

The value of the sum in the denominator must be found. This is easily done from (A.4) and (A.5)

$$\sum_i (a_i\ q_i^k) = R - R\,|_{q=0} \tag{A.10}$$

Using Equation (IV.2), this is equivalent to

$$\sum_i (a_i\ q_i^k) = R - R/RRW^k \tag{A.11}$$

Substituting Equation (A.11) into Equation (A.9), one will arrive at

$$\frac{R_{threshold} - R}{R - R/RRW^k} + 1 = LEF^k \tag{A.12}$$

Finally, using Equation (IV.3), the following expression gives LEF in terms of FV, R, and $R_{threshold}$.

$$\frac{(R_{threshold}/R) - 1}{FV^k} + 1 = LEF^k \tag{A.13}$$

For those SSCs where one $LEF^k q_i^k$ reaches unity before the risk limit has been exceeded, Equation (A.13) is not valid. Because there may be many failure modes, finding a closed form solution to Equation (A.8) is a difficult task. A method to find the solution with a small number of model evaluations is preferred. From the form of Equation (A.8), it is observed that $R(q^k)$ is a piecewise linear function where each piece has a lower slope than the last.

In Figure 3, the SSC has three failure modes. The first failure mode may be due to an earthquake initiator. The second might be for a sequence with an initiating event that compromises function of the system. The final failure mode may be the failure probability when the SSC is not compromised.

The shaded region is where Equation (A.13) is a valid solution to find the true LEF^k value. The limit shown is above where the first failure mode has reached unity. The estimate of LEF^k from Equation (A.13), LEF_0^k, will always be an underestimate when this is the case. Finding the true LEF^k is a typical root finding problem.

The piecewise linear nature of the function makes the secant method an ideal way to find the true LEF^k. Using the estimate from Equation (A.13), the new risk, R_1 is found. The first iteration LEF_1^k is:

$$\frac{R_{threshold} - R}{R_1 - R}(LEF_0^k - 1) = LEF^k{}_1 \tag{A.14}$$

The subsequent iterations are slightly different.

$$LEF_n^k = \frac{R_{threshold} - R_n}{R_n - R_{n-1}}(LEF_{n-1}^k - LEF_{n-2}^k) + LEF_{n-1}^k \tag{A.15}$$

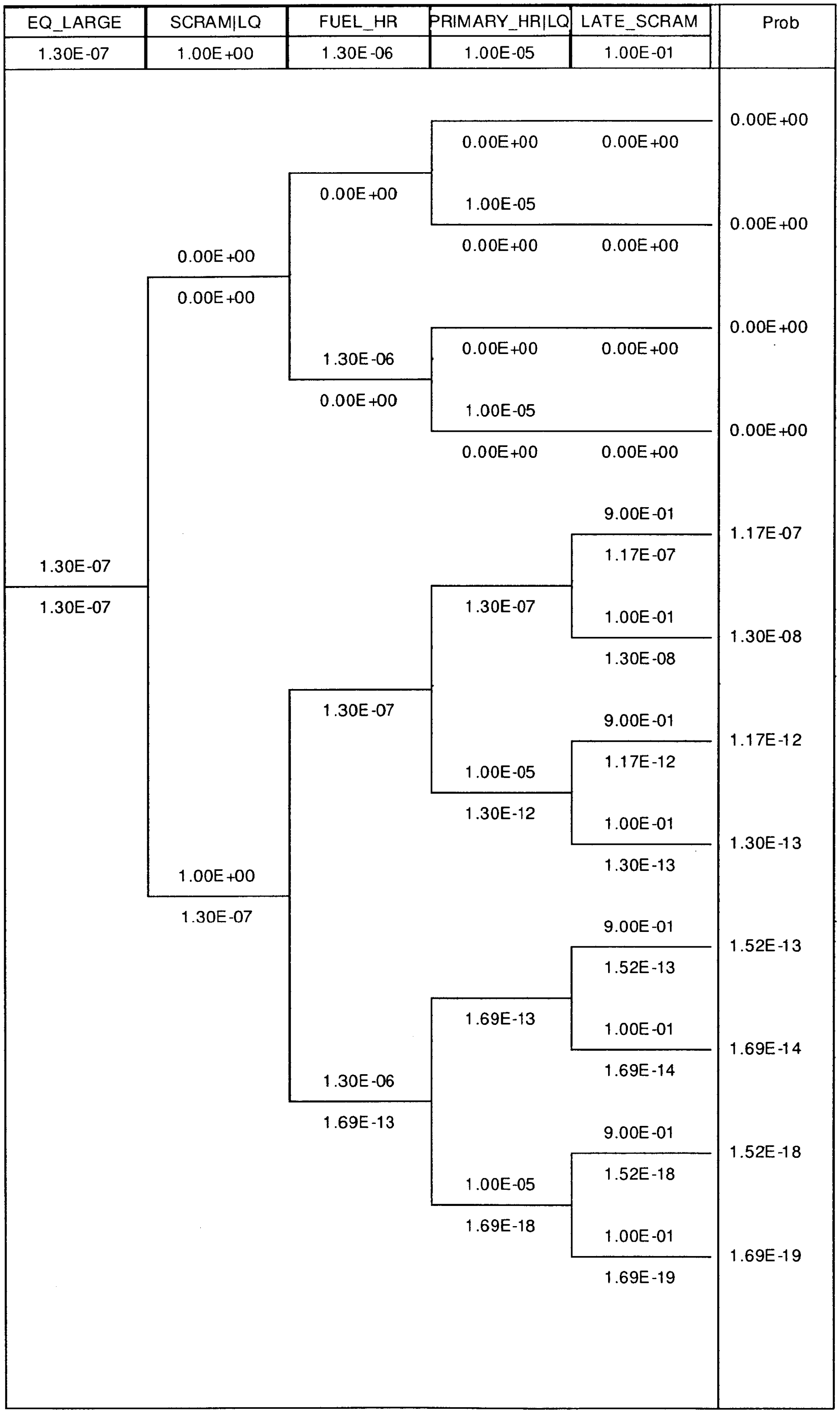
This method in combination with software such as SAPHIRE allows a user to quickly find the exact LEF^k of each SSC^k.

APPENDIX B – FUNCTIONAL EVENT TREES

ALMR FUNCTIONAL TREES

IN THE ORDER OF

Large Seismic
Medium Seismic
Small Seismic
Forced Shutdown
Local Blockage (Adapted from PRISM)
Loss of Flow
Large Reactivity Insertion
Medium Reactivity Insertion
Small Reactivity Insertion


EQ_LARGE
1.30E-07
SCRAM|LQ
1.00E+00
FUEL_HR
1.30E-06
PRIMARY_HR|LQ
1.00E-05
LATE_SCRAM
1.00E-01
Prob
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
1.00E-05
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
0.00E+00
1.30E-06
0.00E+00
0.00E+00
1.00E-05
0.00E+00
0.00E+00
0.00E+00
9.00E-01
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1.17E-07
1.30E-07
1.30E-07
1.30E-07
1.00E-01
1.30E-08
1.30E-08
1.30E-07
9.00E-01
1.17E-12
1.17E-12
1.00E-05
1.30E-12
1.00E-01
1.30E-13
1.30E-13
1.00E+00
1.30E-07
9.00E-01
1.52E-13
1.52E-13
1.69E-13
1.00E-01
1.69E-14
1.69E-14
1.30E-06
1.69E-13
9.00E-01
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1.00E-05
1.69E-18
1.00E-01
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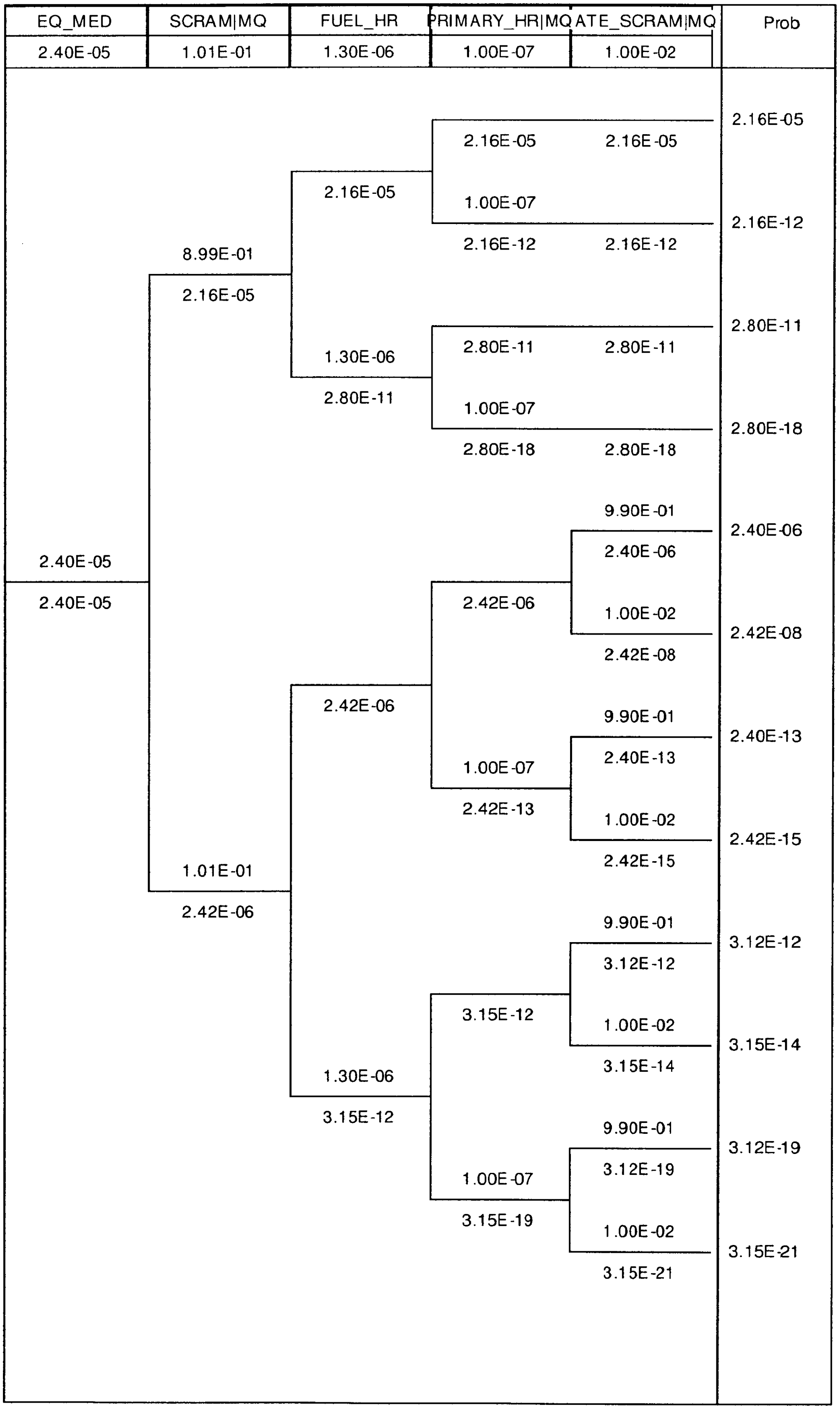
1.69E-19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FUEL_HR
PRIMARY_HR|MQ
ATE_SCRAM|MQ
Prob
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1.00E-07
1.00E-02
2.16E-05
2.16E-05
2.16E-05
2.16E-05
1.00E-07
2.16E-12
8.99E-01
2.16E-12
2.16E-12
2.16E-05
2.80E-11
2.80E-11
2.80E-11
1.30E-06
2.80E-11
1.00E-07
2.80E-18
2.80E-18
2.80E-18
9.90E-01
2.40E-06
2.40E-05
2.40E-06
2.40E-05
2.42E-06
1.00E-02
2.42E-08
2.42E-08
2.42E-06
9.90E-01
2.40E-13
2.40E-13
1.00E-07
2.42E-13
1.00E-02
2.42E-15
1.01E-01
2.42E-15
2.42E-06
9.90E-01
3.12E-12
3.12E-12
3.15E-12
1.00E-02
3.15E-14
1.30E-06
3.15E-14
3.15E-12
9.90E-01
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3.15E-19
1.00E-02
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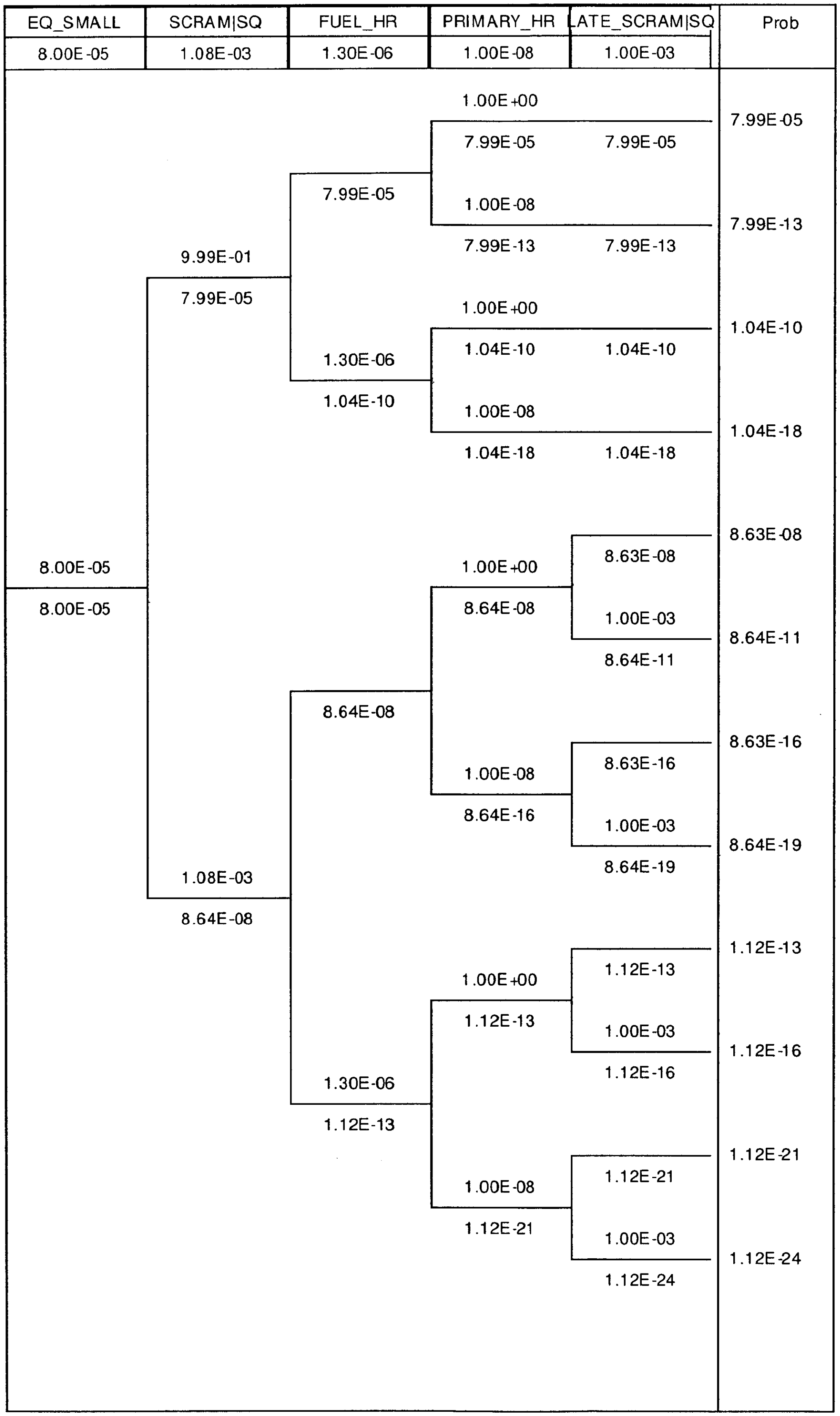
3.15E-21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1.08E-03
FUEL_HR
1.30E-06
PRIMARY_HR
1.00E-08
LATE_SCRAM|SQ
1.00E-03
Prob
1.00E+00
7.99E-05
7.99E-05
7.99E-05
7.99E-05
1.00E-08
7.99E-13
7.99E-13
7.99E-13
9.99E-01
7.99E-05
1.00E+00
1.04E-10
1.04E-10
1.04E-10
1.30E-06
1.04E-10
1.00E-08
1.04E-18
1.04E-18
1.04E-18
8.00E-05
8.00E-05
8.63E-08
8.63E-08
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8.64E-08
1.00E-03
8.64E-11
8.64E-11
8.64E-08
8.63E-16
8.63E-16
1.00E-08
8.64E-16
1.00E-03
8.64E-19
8.64E-19
1.08E-03
8.64E-08
1.12E-13
1.12E-13
1.00E+00
1.12E-13
1.00E-03
1.12E-16
1.12E-16
1.30E-06
1.12E-13
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1.00E-03
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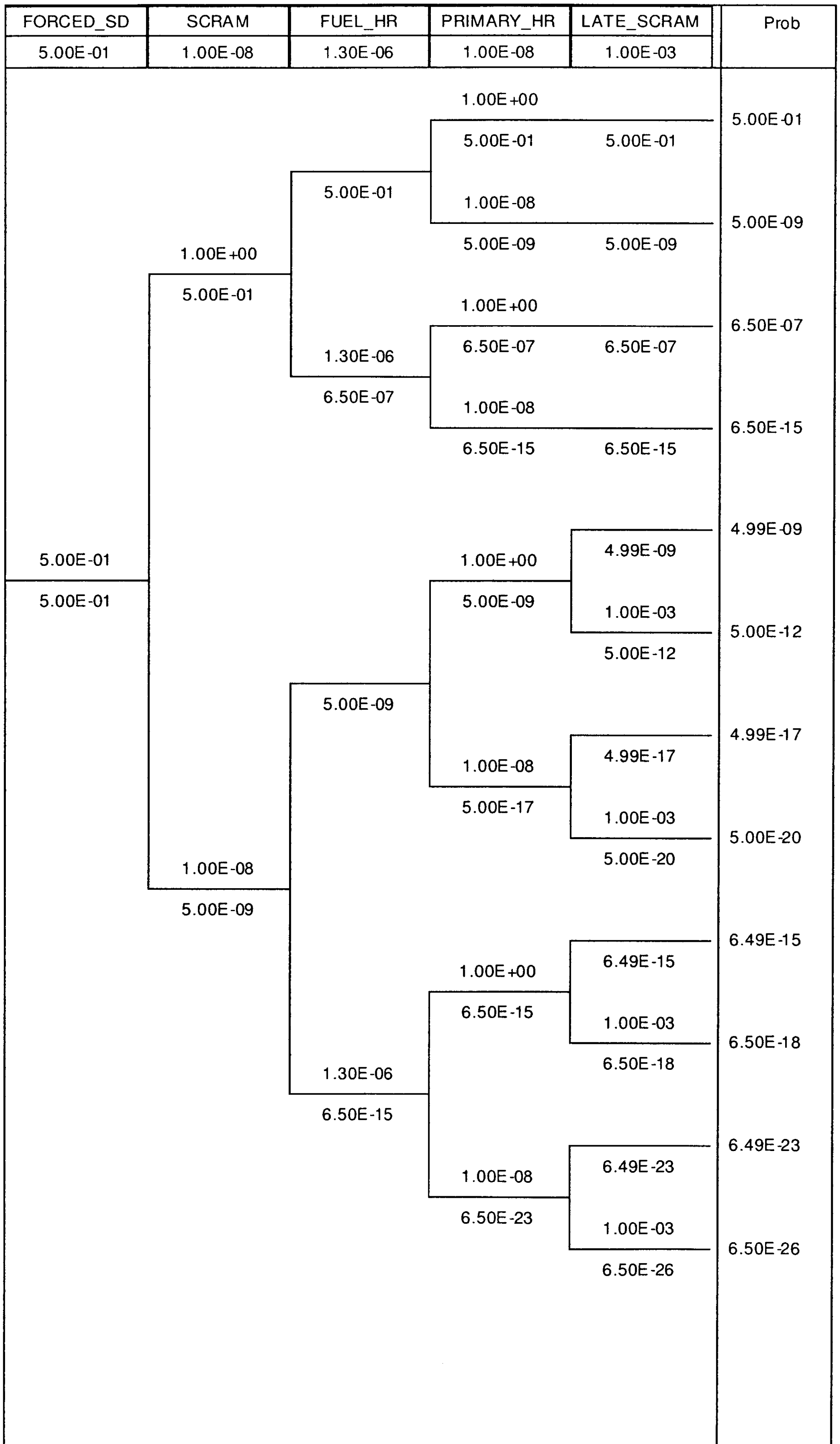
1.12E-24


FORCED_SD
SCRAM
FUEL_HR
PRIMARY_HR
LATE_SCRAM
Prob
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1.00E-08
1.30E-06
1.00E-08
1.00E-03
1.00E+00
5.00E-01
5.00E-01
5.00E-01
1.00E-08
5.00E-09
5.00E-09
5.00E-09
1.00E+00
5.00E-01
1.00E+00
6.50E-07
6.50E-07
6.50E-07
1.30E-06
6.50E-07
1.00E-08
6.50E-15
6.50E-15
6.50E-15
4.99E-09
4.99E-09
5.00E-01
1.00E+00
5.00E-01
5.00E-09
1.00E-03
5.00E-12
5.00E-12
5.00E-09
4.99E-17
4.99E-17
1.00E-08
5.00E-17
1.00E-03
5.00E-20
5.00E-20
1.00E-08
5.00E-09
6.49E-15
6.49E-15
1.00E+00
6.50E-15
1.00E-03
6.50E-18
6.50E-18
1.30E-06
6.50E-15
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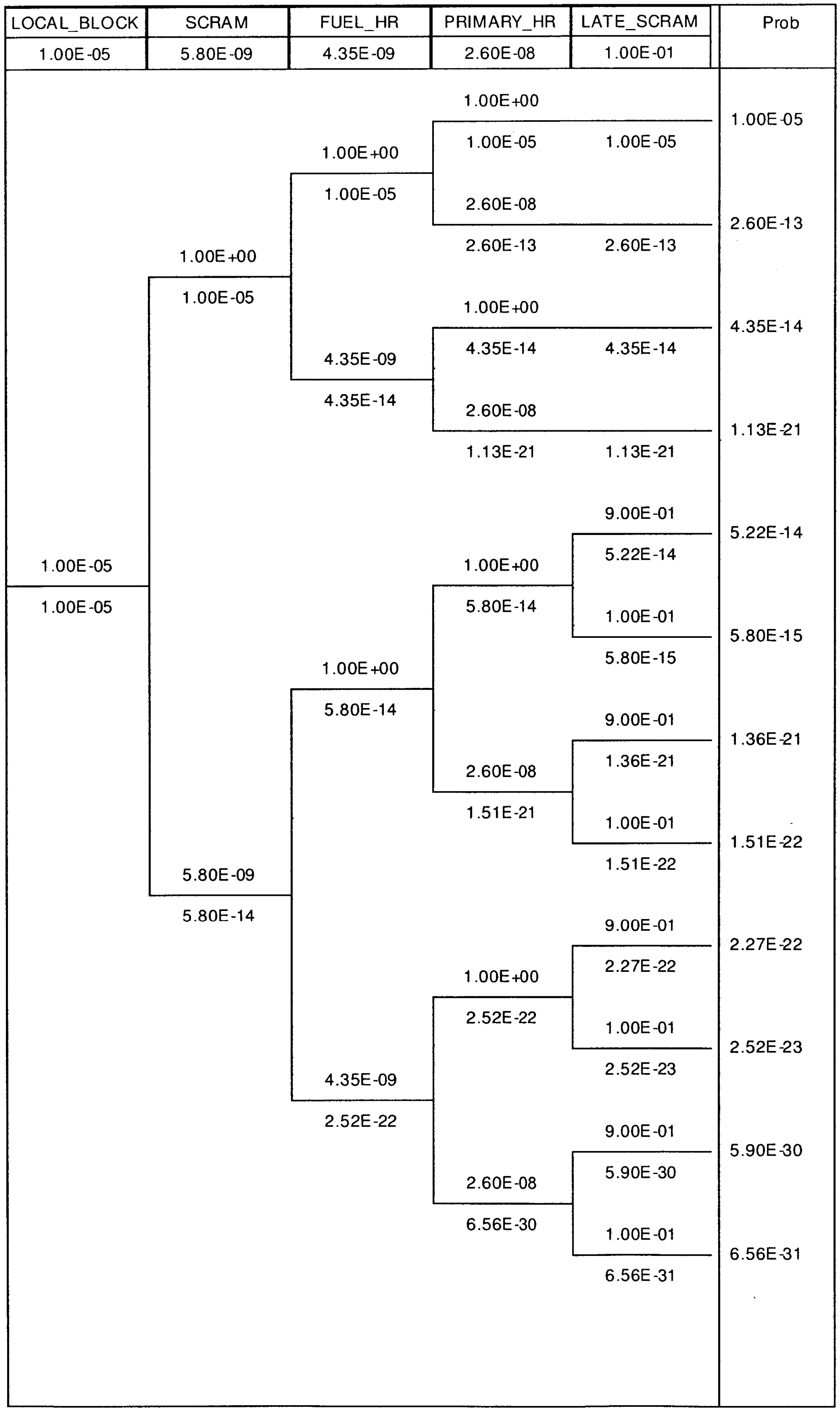
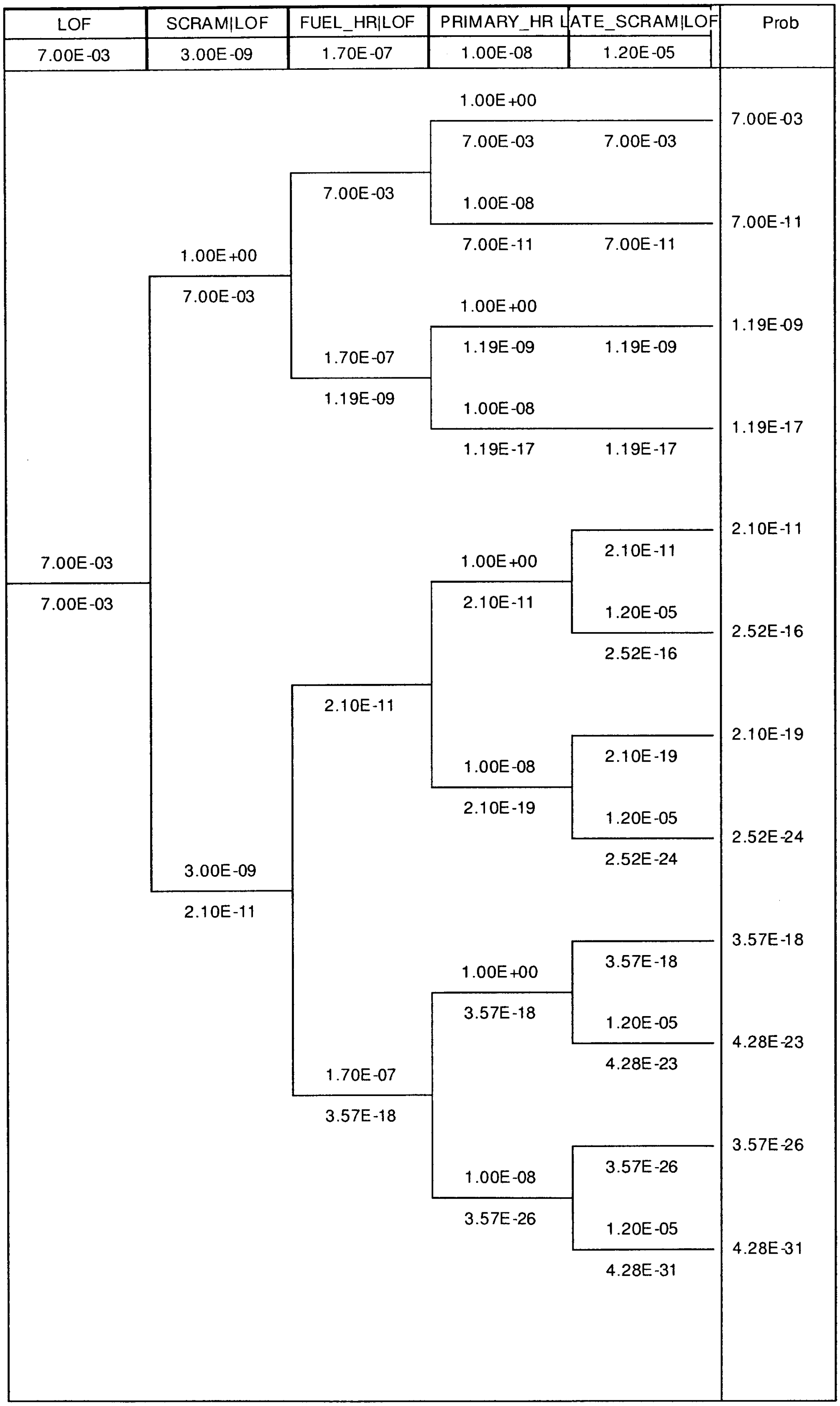
6.49E-23
1.00E-08
6.50E-23
1.00E-03
6.50E-26
6.50E-26


LOCAL_BLOCK
1.00E-05
SCRAM
5.80E-09
FUEL_HR
4.35E-09
PRIMARY_HR
2.60E-08
LATE_SCRAM
1.00E-01
Prob
1.00E-05
2.60E-13
4.35E-14
1.13E-21
5.22E-14
5.80E-15
1.36E-21
1.51E-22
2.27E-22
2.52E-23
5.90E-30
6.56E-31


LOF
SCRAM|LOF
FUEL_HR|LOF
PRIMARY_HR L
ATE_SCRAM|LOF
Prob
7.00E-03
3.00E-09
1.70E-07
1.00E-08
1.20E-05
1.00E+00
7.00E-03
7.00E-03
7.00E-03
7.00E-03
1.00E-08
7.00E-11
7.00E-11
7.00E-11
1.00E+00
7.00E-03
1.00E+00
1.19E-09
1.19E-09
1.19E-09
1.70E-07
1.19E-09
1.00E-08
1.19E-17
1.19E-17
1.19E-17
2.10E-11
7.00E-03
1.00E+00
2.10E-11
7.00E-03
2.10E-11
1.20E-05
2.52E-16
2.52E-16
2.10E-11
2.10E-19
2.10E-19
1.00E-08
2.10E-19
1.20E-05
2.52E-24
2.52E-24
3.00E-09
2.10E-11
3.57E-18
1.00E+00
3.57E-18
3.57E-18
1.20E-05
4.28E-23
4.28E-23
1.70E-07
3.57E-18
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3.57E-26
1.00E-08
3.57E-26
1.20E-05
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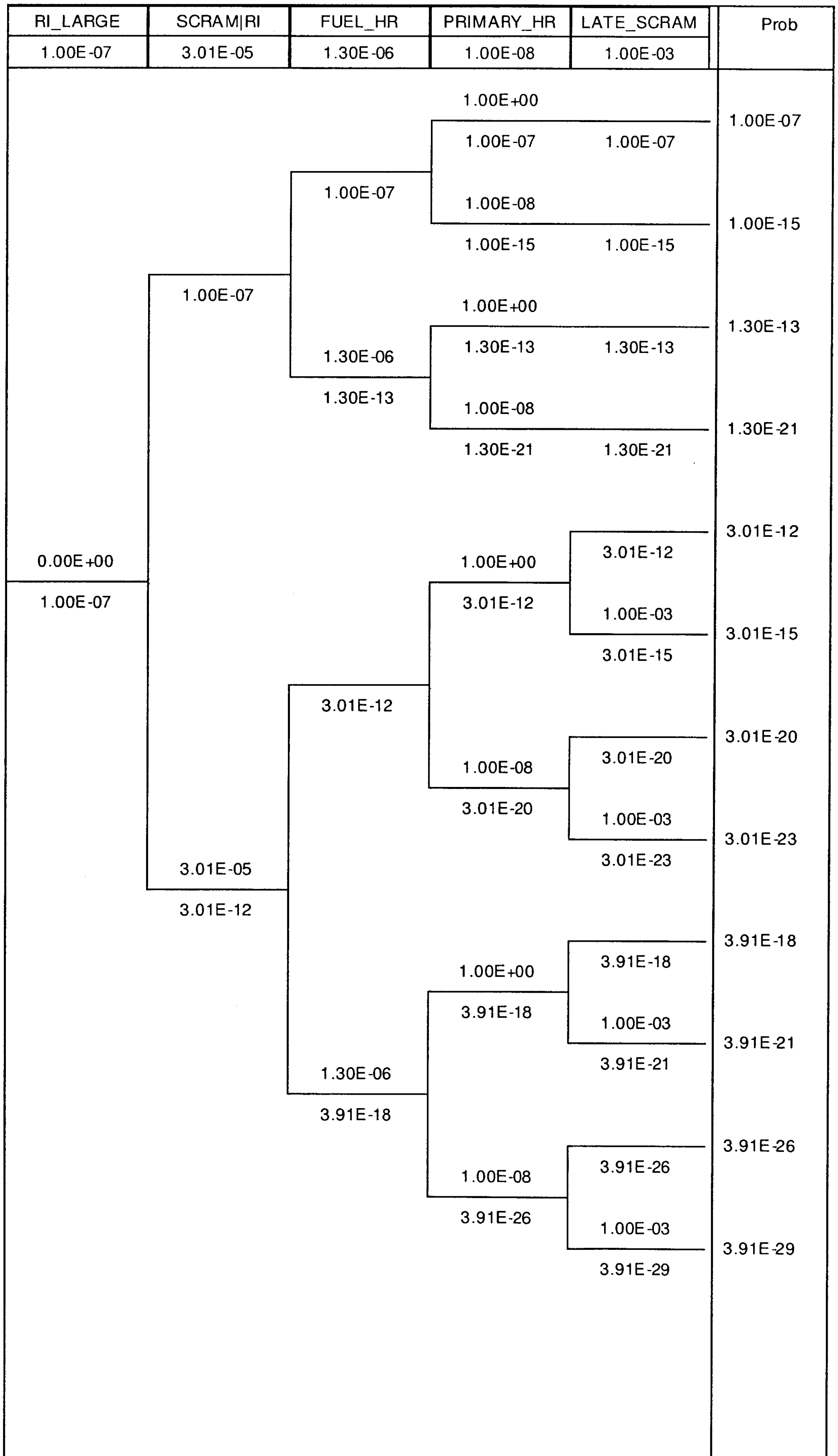
4.28E-31


RI_LARGE
1.00E-07
SCRAM|RI
3.01E-05
FUEL_HR
1.30E-06
PRIMARY_HR
1.00E-08
LATE_SCRAM
1.00E-03
Prob
1.00E+00
1.00E-07
1.00E-07
1.00E-07
1.00E-07
1.00E-08
1.00E-15
1.00E-15
1.00E-15
1.00E-07
1.00E+00
1.30E-13
1.30E-13
1.30E-13
1.30E-06
1.30E-13
1.00E-08
1.30E-21
1.30E-21
1.30E-21
0.00E+00
1.00E-07
3.01E-12
3.01E-12
1.00E+00
3.01E-12
1.00E-03
3.01E-15
3.01E-15
3.01E-12
3.01E-20
3.01E-20
1.00E-08
3.01E-20
1.00E-03
3.01E-23
3.01E-23
3.01E-05
3.01E-12
3.91E-18
3.91E-18
1.00E+00
3.91E-18
1.00E-03
3.91E-21
3.91E-21
1.30E-06
3.91E-18
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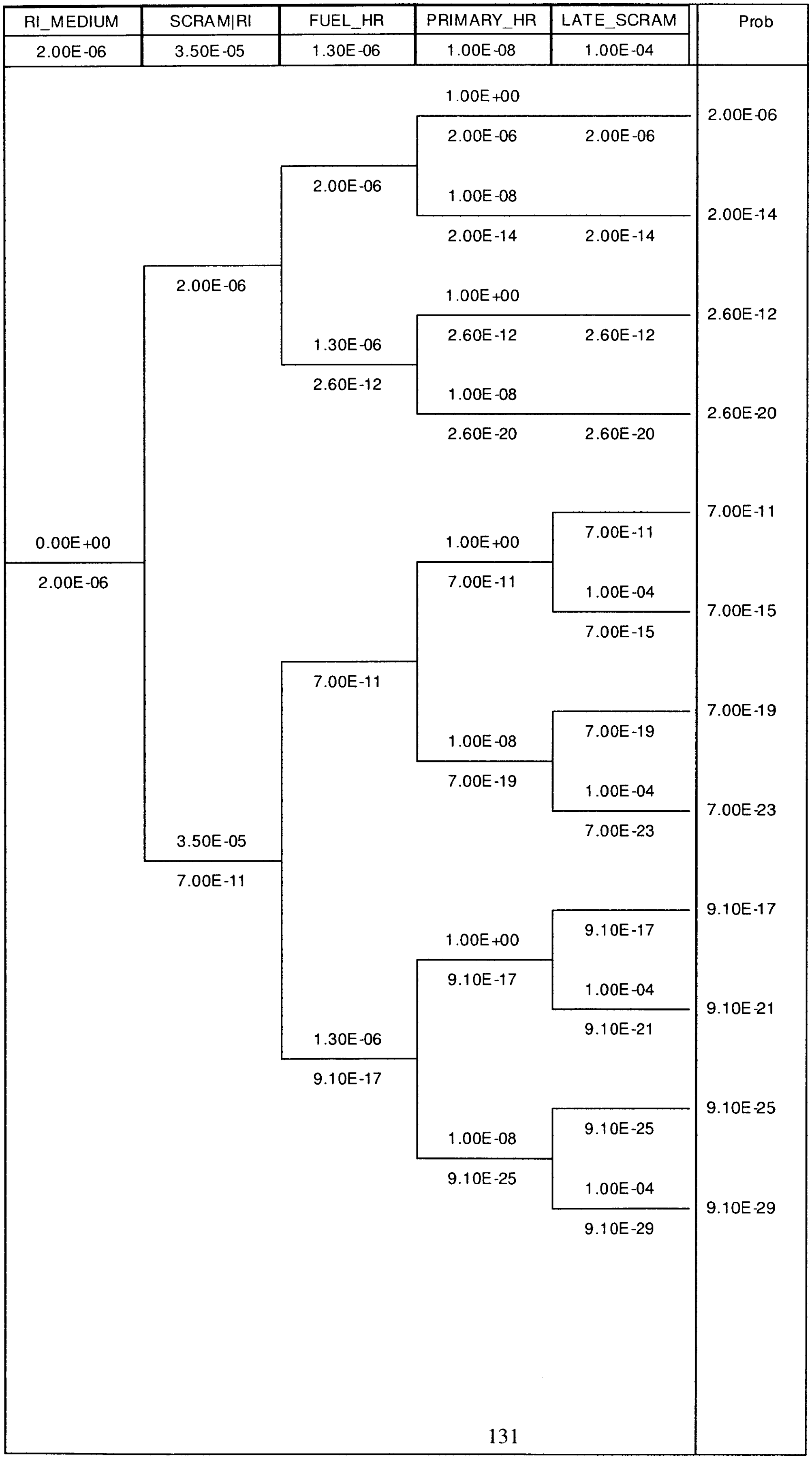
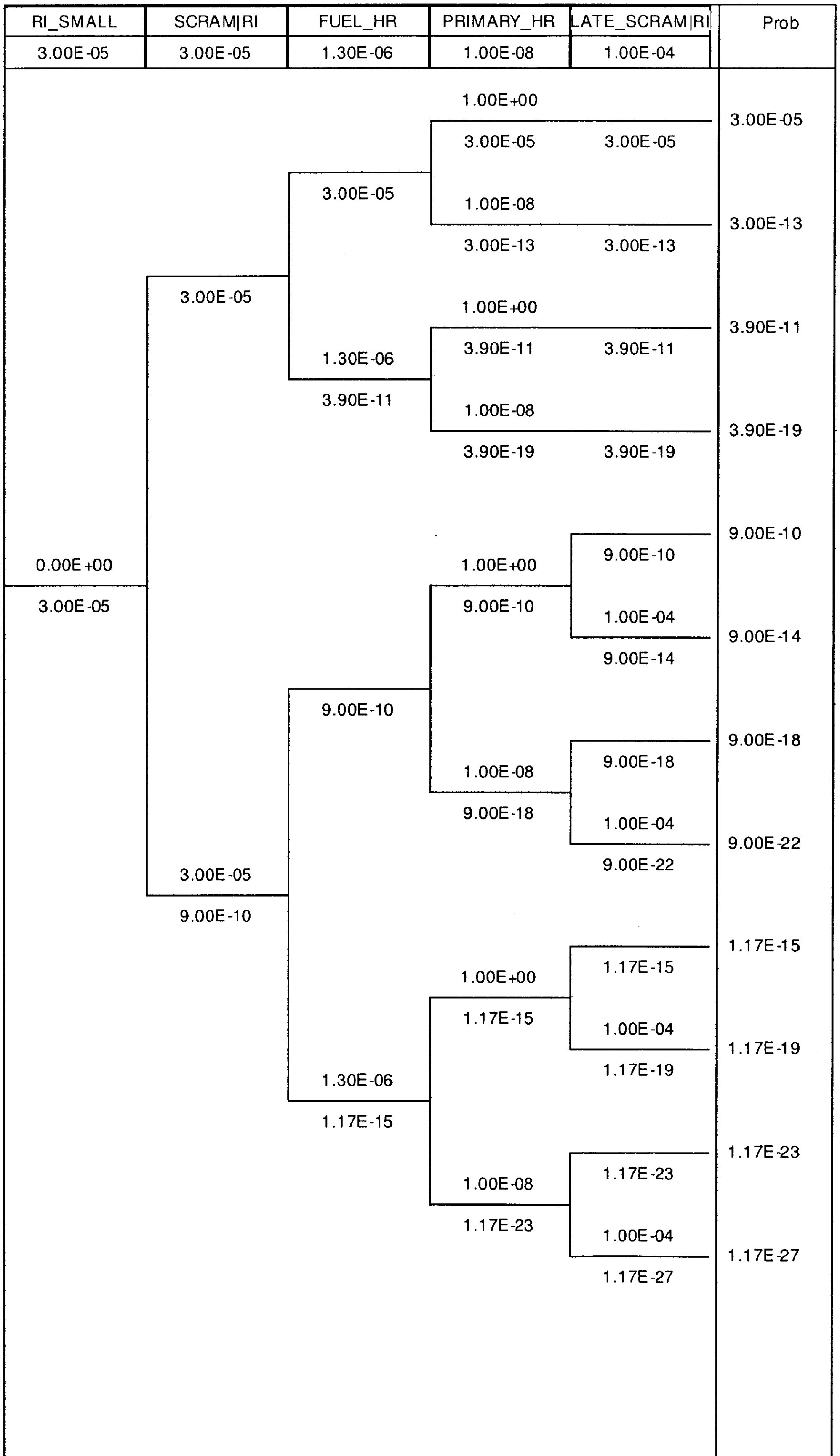
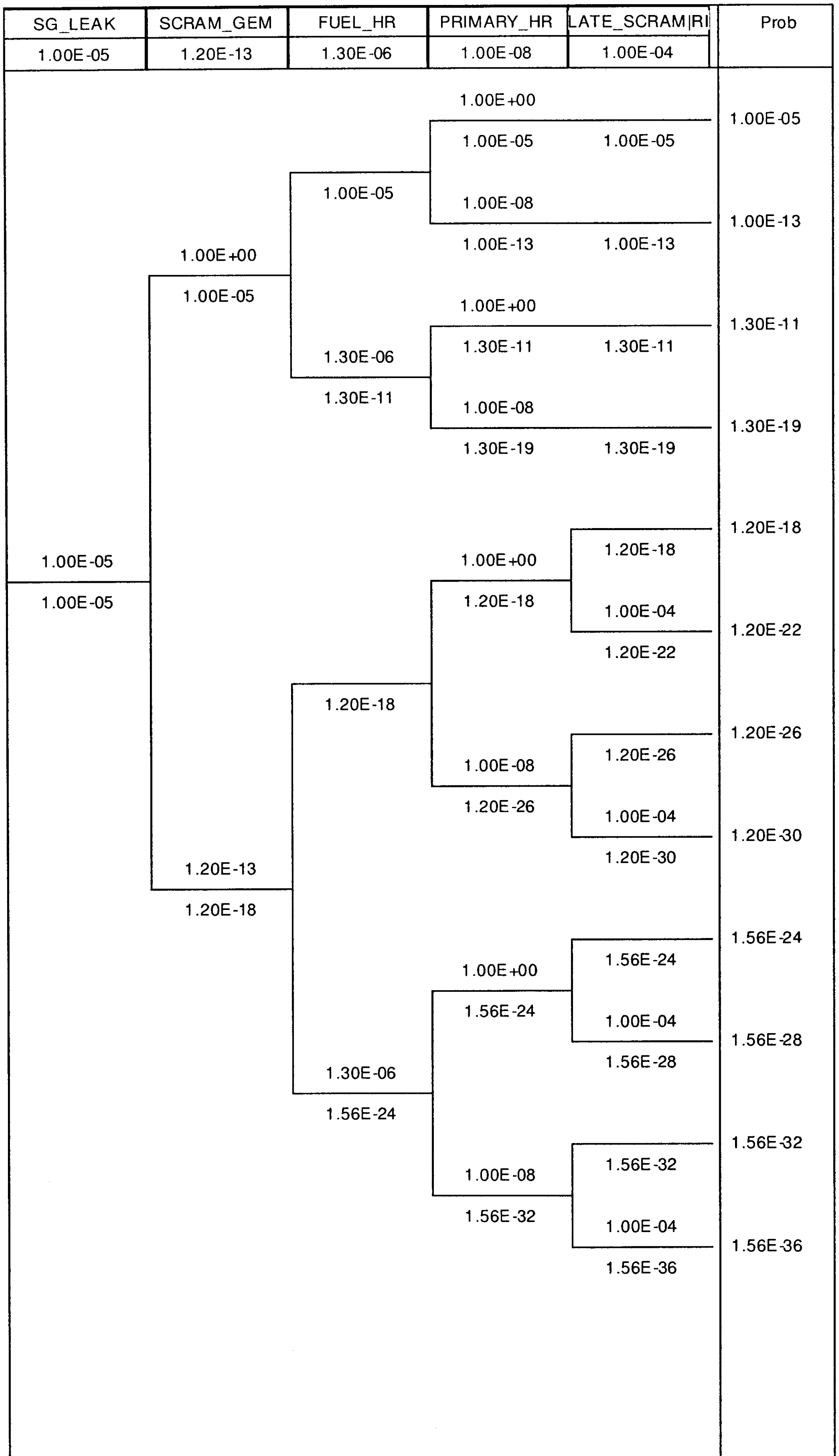
3.91E-26
1.00E-08
3.91E-26
1.00E-03
3.91E-29
3.91E-29


RI_MEDIUM
2.00E-06
SCRAM|RI
3.50E-05
FUEL_HR
1.30E-06
PRIMARY_HR
1.00E-08
LATE_SCRAM
1.00E-04
Prob
1.00E+00
2.00E-06
2.00E-06
2.00E-06
2.00E-06
1.00E-08
2.00E-14
2.00E-14
2.00E-14
2.00E-06
1.00E+00
2.60E-12
2.60E-12
2.60E-12
1.30E-06
2.60E-12
1.00E-08
2.60E-20
2.60E-20
2.60E-20
0.00E+00
2.00E-06
7.00E-11
7.00E-11
1.00E+00
7.00E-11
1.00E-04
7.00E-15
7.00E-15
7.00E-11
7.00E-19
7.00E-19
1.00E-08
7.00E-19
1.00E-04
7.00E-23
7.00E-23
3.50E-05
7.00E-11
9.10E-17
9.10E-17
1.00E+00
9.10E-17
1.00E-04
9.10E-21
9.10E-21
1.30E-06
9.10E-17
9.10E-25
9.10E-25
1.00E-08
9.10E-25
1.00E-04
9.10E-29
9.10E-29


RI_SMALL
SCRAM|RI
FUEL_HR
PRIMARY_HR
LATE_SCRAM|RI
Prob
3.00E-05
3.00E-05
1.30E-06
1.00E-08
1.00E-04
1.00E+00
3.00E-05
3.00E-05
3.00E-05
3.00E-05
1.00E-08
3.00E-13
3.00E-13
3.00E-13
3.00E-05
1.00E+00
3.90E-11
3.90E-11
3.90E-11
1.30E-06
3.90E-11
1.00E-08
3.90E-19
3.90E-19
3.90E-19
9.00E-10
0.00E+00
1.00E+00
9.00E-10
3.00E-05
9.00E-10
1.00E-04
9.00E-14
9.00E-14
9.00E-10
9.00E-18
9.00E-18
1.00E-08
9.00E-18
1.00E-04
9.00E-22
9.00E-22
3.00E-05
9.00E-10
1.17E-15
1.00E+00
1.17E-15
1.17E-15
1.00E-04
1.17E-19
1.30E-06
1.17E-19
1.17E-15
1.17E-23
1.17E-23
1.00E-08
1.17E-23
1.00E-04
1.17E-27
1.17E-27


SG_LEAK
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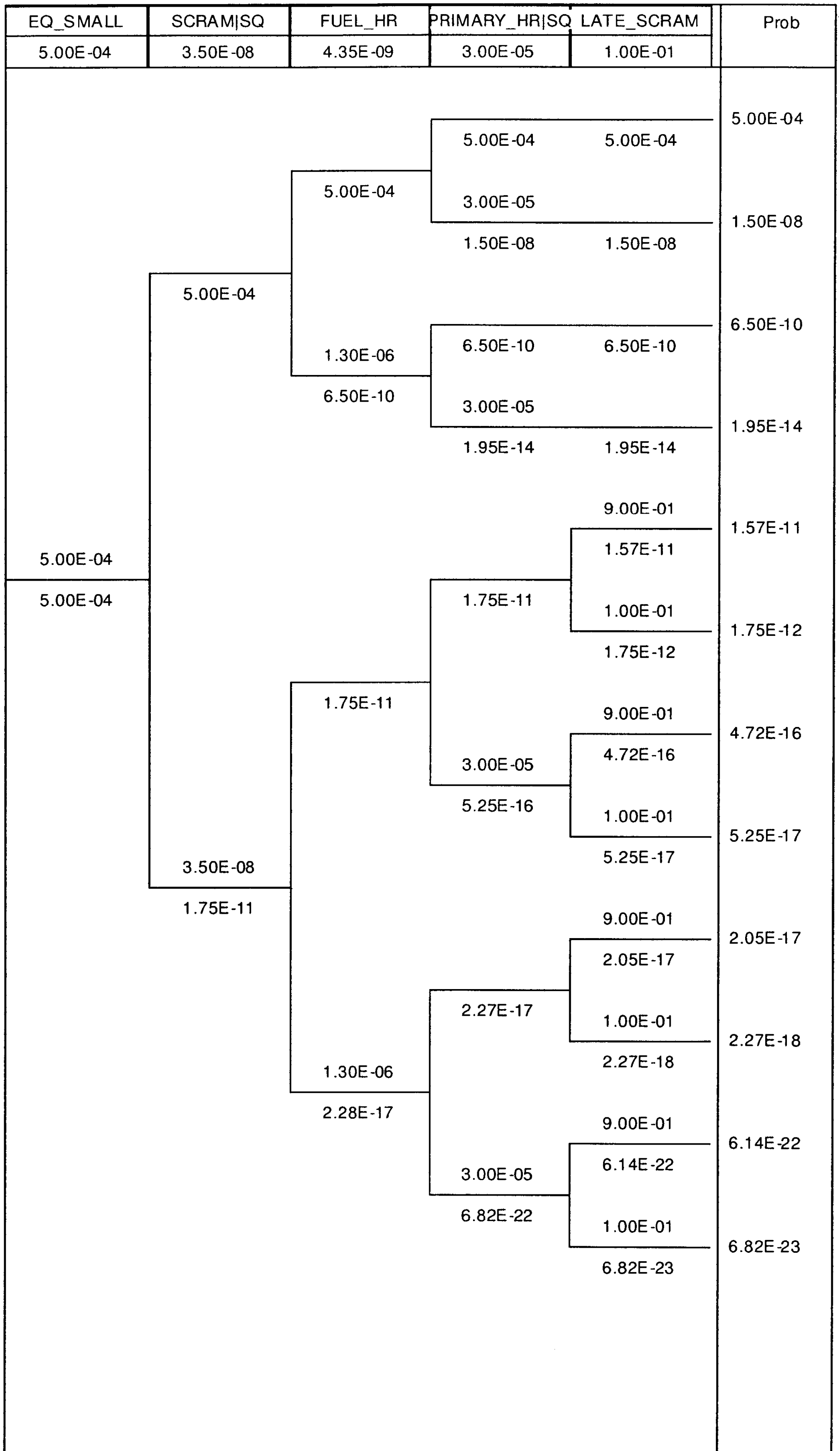
1.20E-13
FUEL_HR
1.30E-06
PRIMARY_HR
1.00E-08
LATE_SCRAM|RI
1.00E-04
Prob
1.00E+00
1.00E-05
1.00E-05
1.00E-05
1.00E-05
1.00E-08
1.00E-13
1.00E-13
1.00E-13
1.00E+00
1.00E-05
1.00E+00
1.30E-11
1.30E-11
1.30E-11
1.30E-06
1.30E-11
1.00E-08
1.30E-19
1.30E-19
1.30E-19
1.20E-18
1.20E-18
1.00E-05
1.00E+00
1.00E-05
1.20E-18
1.00E-04
1.20E-22
1.20E-22
1.20E-18
1.20E-26
1.20E-26
1.00E-08
1.20E-26
1.00E-04
1.20E-30
1.20E-30
1.20E-13
1.20E-18
1.56E-24
1.56E-24
1.00E+00
1.56E-24
1.00E-04
1.56E-28
1.56E-28
1.30E-06
1.56E-24
1.56E-32
1.56E-32
1.00E-08
1.56E-32
1.00E-04
1.56E-36
1.56E-36

PRISM FUNCTIONAL EVENT TREES

Large Seismic
Medium Seismic
Small Seismic
Forced Shutdown
Local Blockage
Loss of Flow
Large Reactivity Insertion
Medium Reactivity Insertion
Small Reactivity Insertion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SCRAM|SQ
3.50E-08
FUEL_HR
4.35E-09
PRIMARY_HR|SQ
3.00E-05
LATE_SCRAM
1.00E-01
Prob
5.00E-04
5.00E-04
5.00E-04
5.00E-04
5.00E-04
3.00E-05
1.50E-08
1.50E-08
1.50E-08
1.30E-06
6.50E-10
6.50E-10
6.50E-10
6.50E-10
3.00E-05
1.95E-14
1.95E-14
1.95E-14
9.00E-01
1.57E-11
1.57E-11
1.75E-11
1.00E-01
1.75E-12
1.75E-12
1.75E-11
9.00E-01
4.72E-16
4.72E-16
3.00E-05
5.25E-16
1.00E-01
5.25E-17
5.25E-17
3.50E-08
1.75E-11
9.00E-01
2.05E-17
2.05E-17
2.27E-17
1.00E-01
2.27E-18
2.27E-18
1.30E-06
2.28E-17
9.00E-01
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1.00E-01
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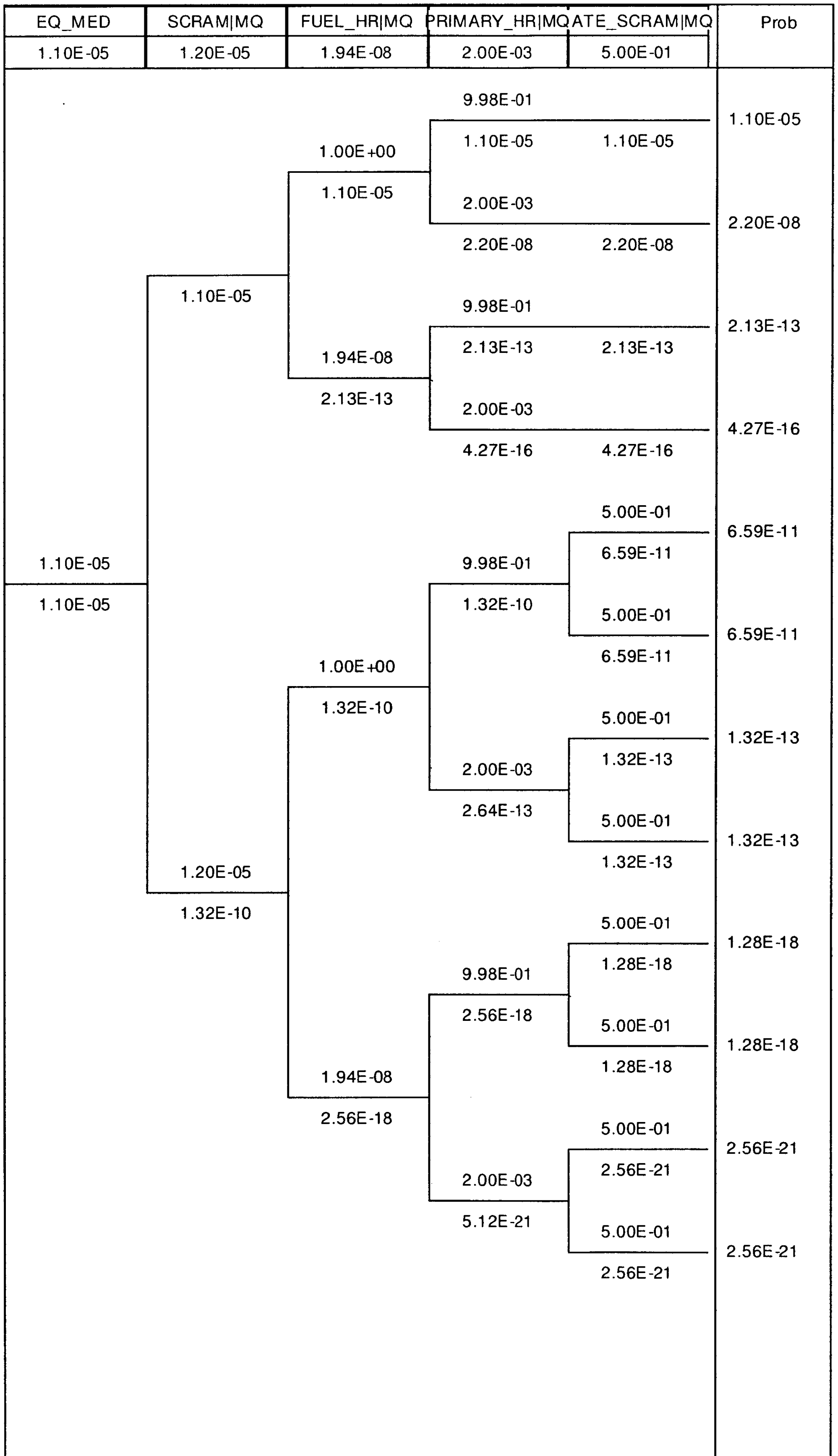
6.82E-23


EQ_MED
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FUEL_HR|MQ
PRIMARY_HR|MQ
ATE_SCRAM|MQ
Prob
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1.20E-05
1.94E-08
2.00E-03
5.00E-01
9.98E-01
1.10E-05
1.10E-05
1.10E-05
1.00E+00
1.10E-05
2.00E-03
2.20E-08
2.20E-08
2.20E-08
2.20E-08
1.10E-05
9.98E-01
2.13E-13
2.13E-13
2.13E-13
1.94E-08
2.13E-13
2.00E-03
4.27E-16
4.27E-16
4.27E-16
4.27E-16
5.00E-01
6.59E-11
1.10E-05
9.98E-01
6.59E-11
1.10E-05
1.32E-10
5.00E-01
6.59E-11
6.59E-11
1.00E+00
1.32E-10
5.00E-01
1.32E-13
1.32E-13
2.00E-03
2.64E-13
5.00E-01
1.32E-13
1.32E-13
1.20E-05
1.32E-10
5.00E-01
1.28E-18
1.28E-18
9.98E-01
2.56E-18
5.00E-01
1.28E-18
1.28E-18
1.94E-08
2.56E-18
5.00E-01
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2.56E-21
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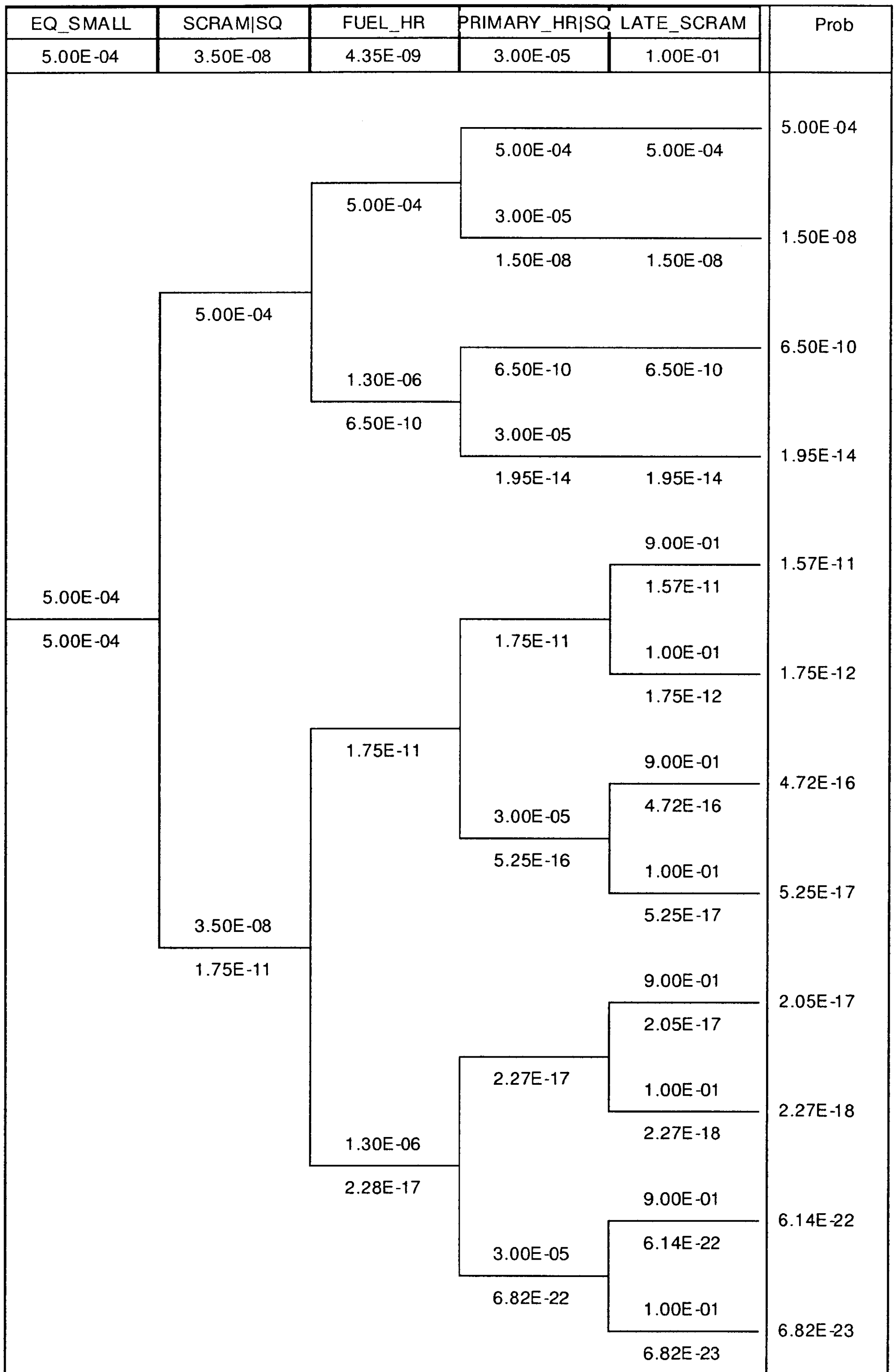
5.00E-01
2.56E-21
2.56E-21


EQ_SMALL
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SCRAM|SQ
3.50E-08
FUEL_HR
4.35E-09
PRIMARY_HR|SQ
3.00E-05
LATE_SCRAM
1.00E-01
Prob
5.00E-04
5.00E-04
5.00E-04
5.00E-04
3.00E-05
1.50E-08
1.50E-08
1.50E-08
1.50E-08
5.00E-04
6.50E-10
6.50E-10
6.50E-10
1.30E-06
6.50E-10
3.00E-05
1.95E-14
1.95E-14
1.95E-14
1.95E-14
9.00E-01
1.57E-11
1.57E-11
5.00E-04
5.00E-04
1.75E-11
1.00E-01
1.75E-12
1.75E-12
1.75E-11
9.00E-01
4.72E-16
4.72E-16
3.00E-05
5.25E-16
1.00E-01
5.25E-17
5.25E-17
3.50E-08
1.75E-11
9.00E-01
2.05E-17
2.05E-17
2.27E-17
1.00E-01
2.27E-18
2.27E-18
1.30E-06
2.28E-17
9.00E-01
6.14E-22
6.14E-22
3.00E-05
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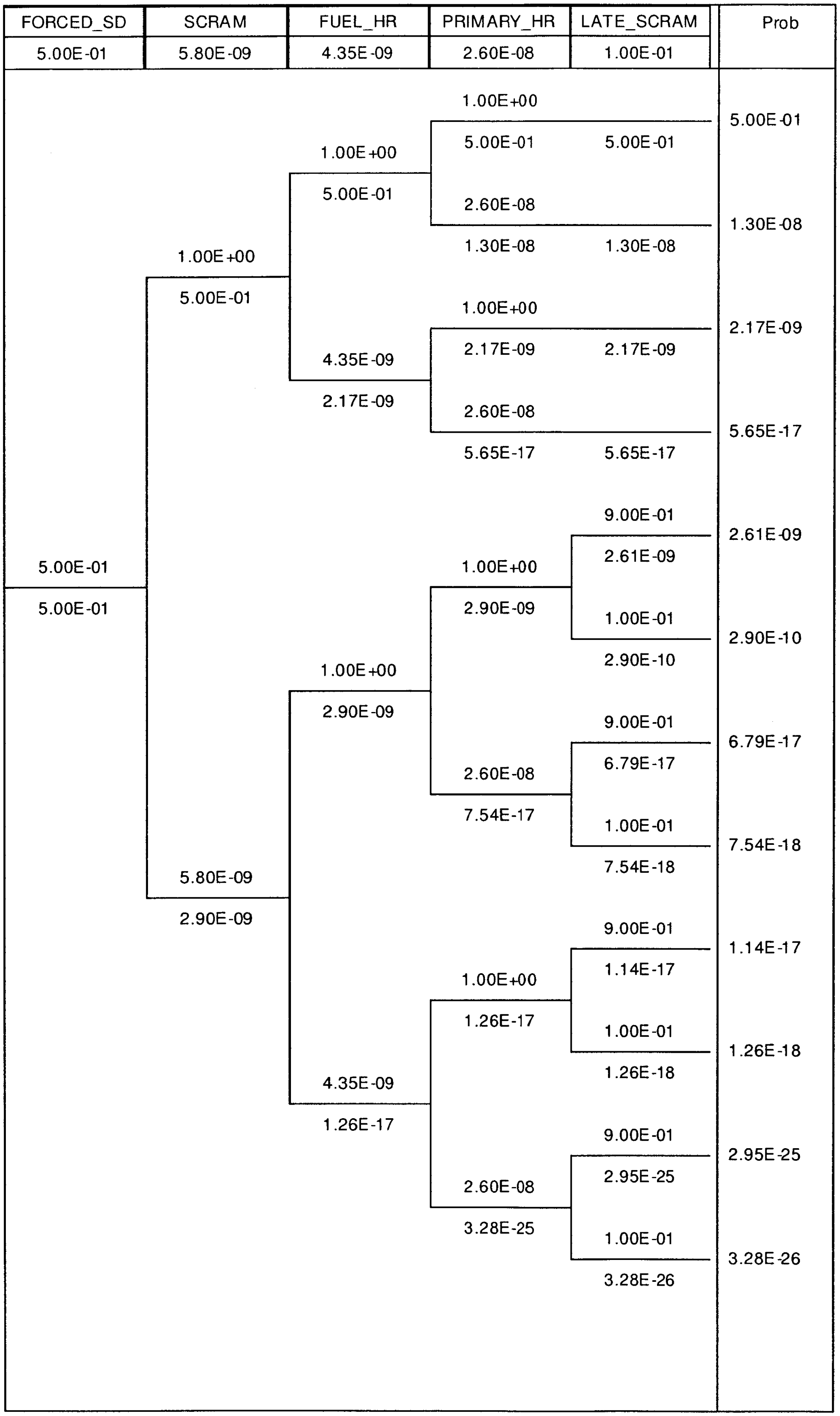
1.00E-01
6.82E-23
6.82E-23


FORCED_SD
5.00E-01
SCRAM
5.80E-09
FUEL_HR
4.35E-09
PRIMARY_HR
2.60E-08
LATE_SCRAM
1.00E-01
Prob
1.00E+00
5.00E-01
5.00E-01
5.00E-01
1.00E+00
5.00E-01
2.60E-08
1.30E-08
1.30E-08
1.30E-08
1.00E+00
5.00E-01
1.00E+00
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2.17E-09
4.35E-09
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1.14E-17
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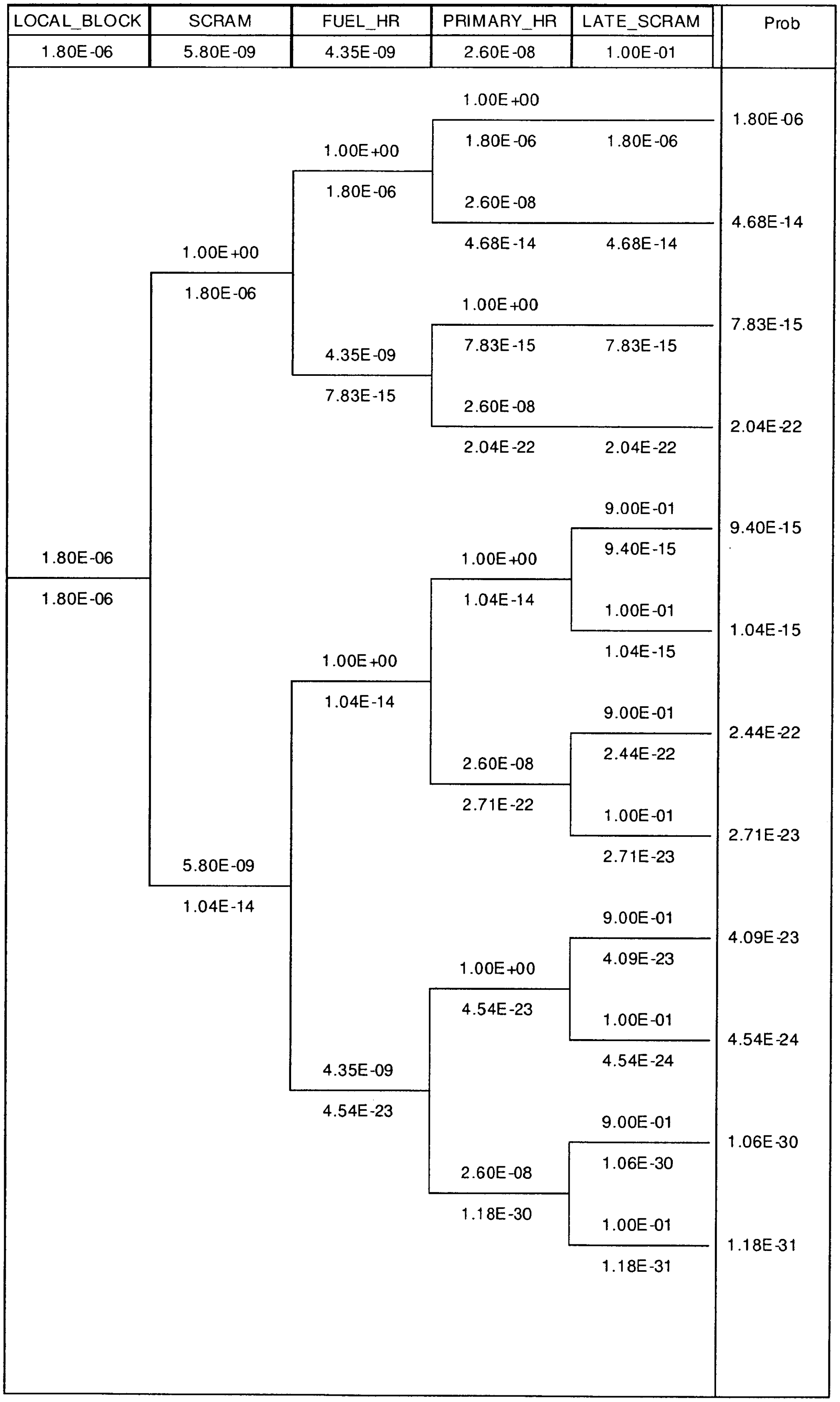
1.26E-18
1.26E-18
4.35E-09
1.26E-17
9.00E-01
2.95E-25
2.95E-25
2.60E-08
3.28E-25
1.00E-01
3.28E-26
3.28E-26


LOCAL_BLOCK
1.80E-06
SCRAM
5.80E-09
FUEL_HR
4.35E-09
PRIMARY_HR
2.60E-08
LATE_SCRAM
1.00E-01
Prob
1.00E+00
1.80E-06
1.80E-06
1.80E-06
1.00E+00
1.80E-06
2.60E-08
4.68E-14
4.68E-14
4.68E-14
1.00E+00
1.80E-06
1.00E+00
7.83E-15
7.83E-15
7.83E-15
4.35E-09
7.83E-15
2.60E-08
2.04E-22
2.04E-22
2.04E-22
9.00E-01
9.40E-15
9.40E-15
1.80E-06
1.80E-06
1.00E+00
1.04E-14
1.00E-01
1.04E-15
1.04E-15
1.00E+00
1.04E-14
9.00E-01
2.44E-22
2.44E-22
2.60E-08
2.71E-22
1.00E-01
2.71E-23
2.71E-23
5.80E-09
1.04E-14
9.00E-01
4.09E-23
4.09E-23
1.00E+00
4.54E-23
1.00E-01
4.54E-24
4.54E-24
4.35E-09
4.54E-23
9.00E-01
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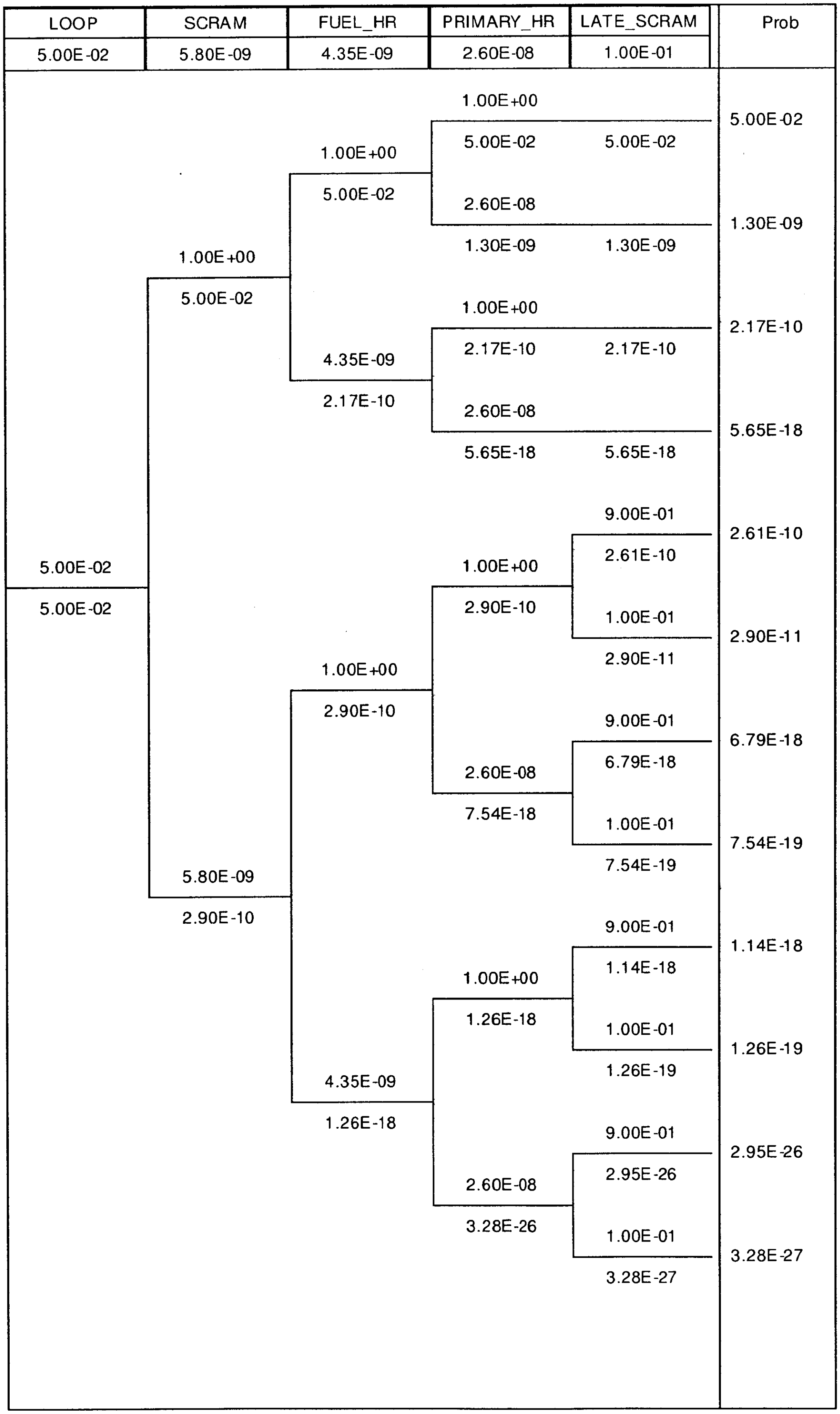
1.06E-30
2.60E-08
1.18E-30
1.00E-01
1.18E-31
1.18E-31


LOOP
SCRAM
FUEL_HR
PRIMARY_HR
LATE_SCRAM
Prob
5.00E-02
5.80E-09
4.35E-09
2.60E-08
1.00E-01
1.00E+00
5.00E-02
5.00E-02
1.00E+00
5.00E-02
2.60E-08
1.30E-09
1.30E-09
1.00E+00
5.00E-02
1.00E+00
2.17E-10
2.17E-10
4.35E-09
2.17E-10
2.60E-08
5.65E-18
5.65E-18
9.00E-01
2.61E-10
5.00E-02
1.00E+00
5.00E-02
2.90E-10
1.00E-01
2.90E-11
1.00E+00
2.90E-10
9.00E-01
6.79E-18
2.60E-08
7.54E-18
1.00E-01
7.54E-19
5.80E-09
2.90E-10
9.00E-01
1.14E-18
1.00E+00
1.26E-18
1.00E-01
1.26E-19
4.35E-09
1.26E-18
9.00E-01
2.95E-26
2.60E-08
3.28E-26
1.00E-01
3.28E-27
5.00E-02
1.30E-09
2.17E-10
5.65E-18
2.61E-10
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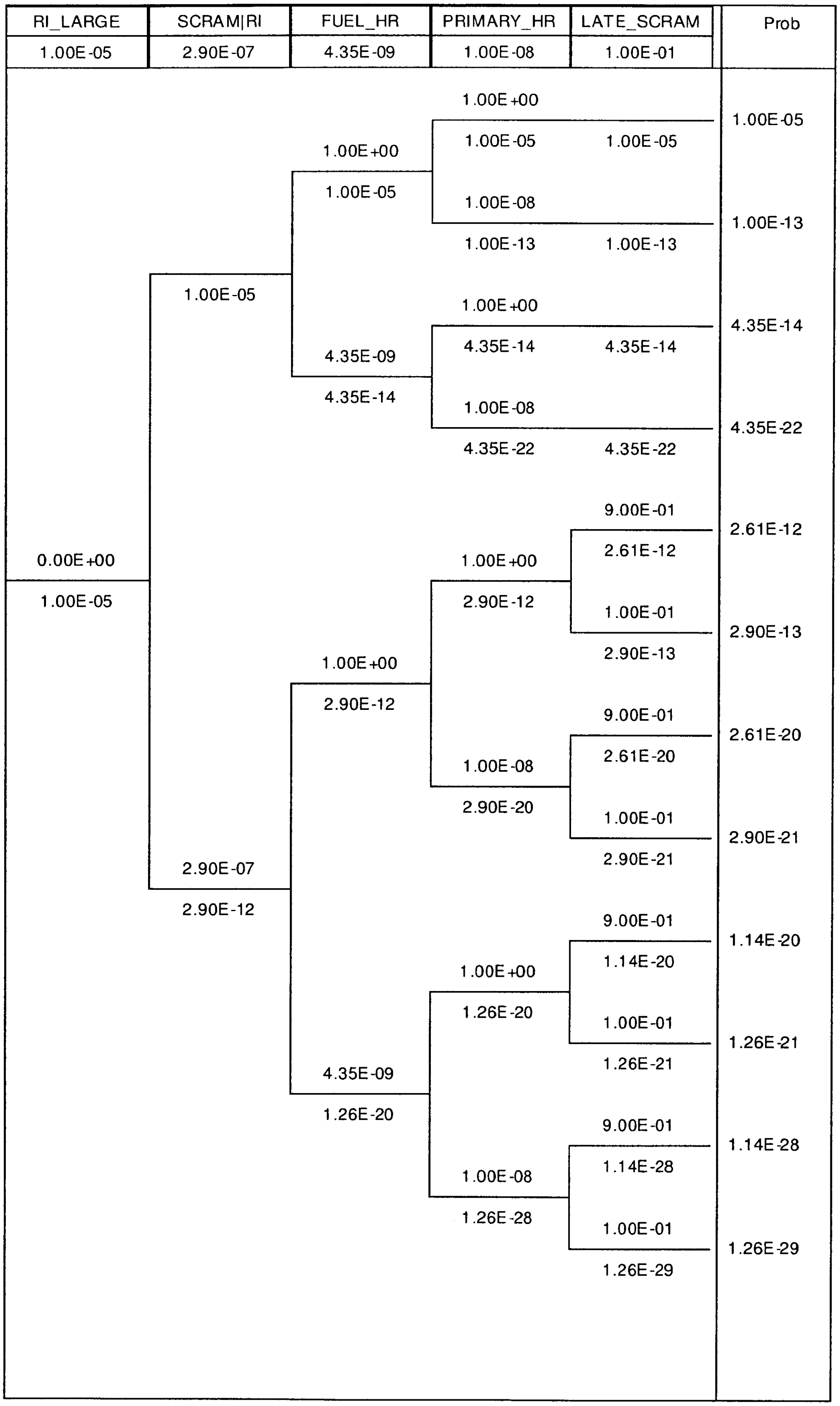
6.79E-18
7.54E-19
1.14E-18
1.26E-19
2.95E-26
3.28E-27


RI_LARGE
1.00E-05
SCRAM|RI
2.90E-07
FUEL_HR
4.35E-09
PRIMARY_HR
1.00E-08
LATE_SCRAM
1.00E-01
Prob
0.00E+00
1.00E-05
1.00E-05
2.90E-07
2.90E-12
1.00E+00
1.00E-05
4.35E-09
4.35E-14
1.00E+00
2.90E-12
4.35E-09
1.26E-20
1.00E+00
1.00E-05
1.00E-08
1.00E-13
1.00E+00
4.35E-14
1.00E-08
4.35E-22
1.00E+00
2.90E-12
1.00E-08
2.90E-20
1.00E+00
1.26E-20
1.00E-08
1.26E-28
1.00E-05
1.00E-13
4.35E-14
4.35E-22
9.00E-01
2.61E-12
1.00E-01
2.90E-13
9.00E-01
2.61E-20
1.00E-01
2.90E-21
9.00E-01
1.14E-20
1.00E-01
1.26E-21
9.00E-01
1.14E-28
1.00E-01
1.26E-29
1.00E-05
1.00E-13
4.35E-14
4.35E-22
2.61E-12
2.90E-13
2.61E-20
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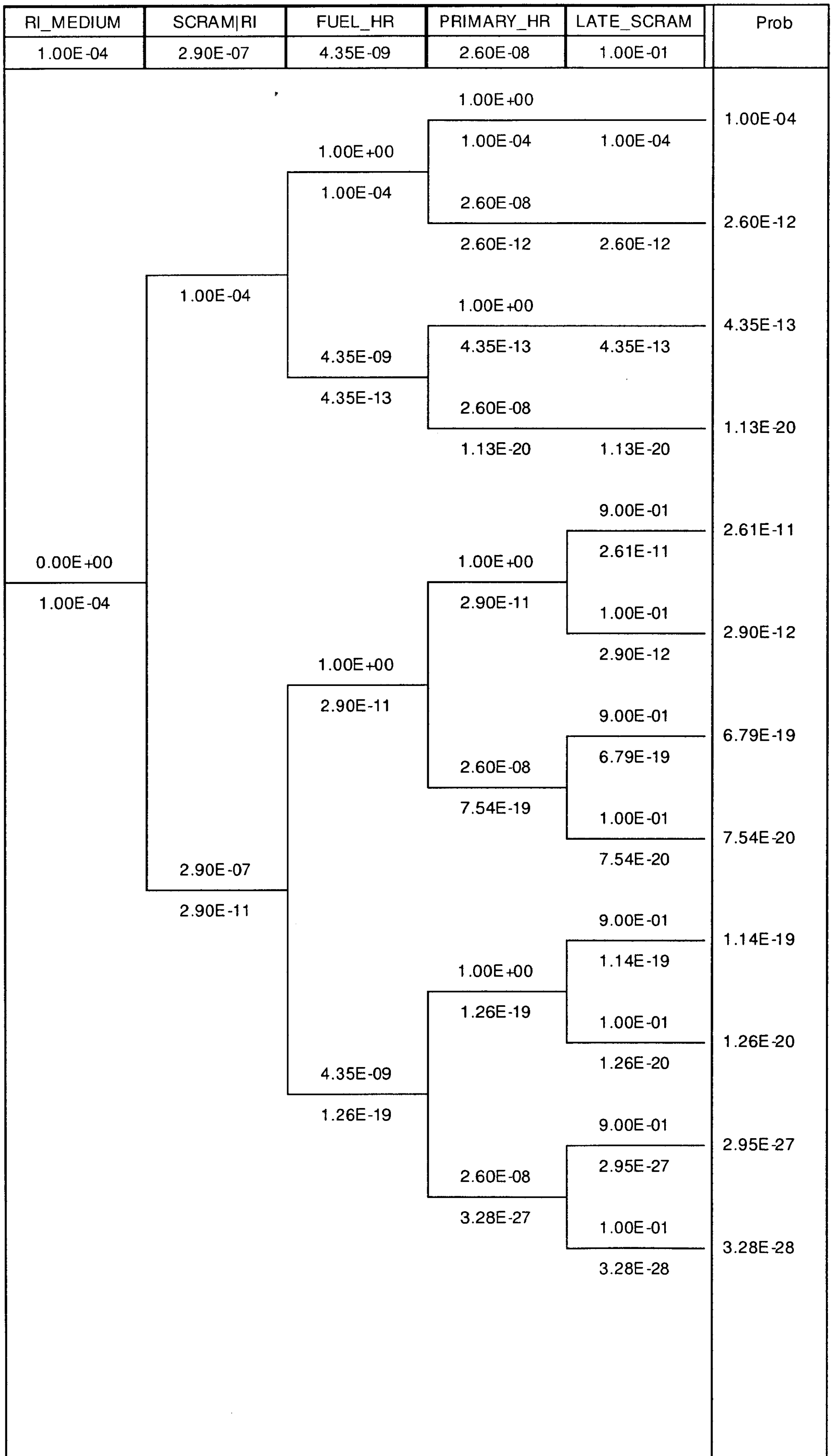
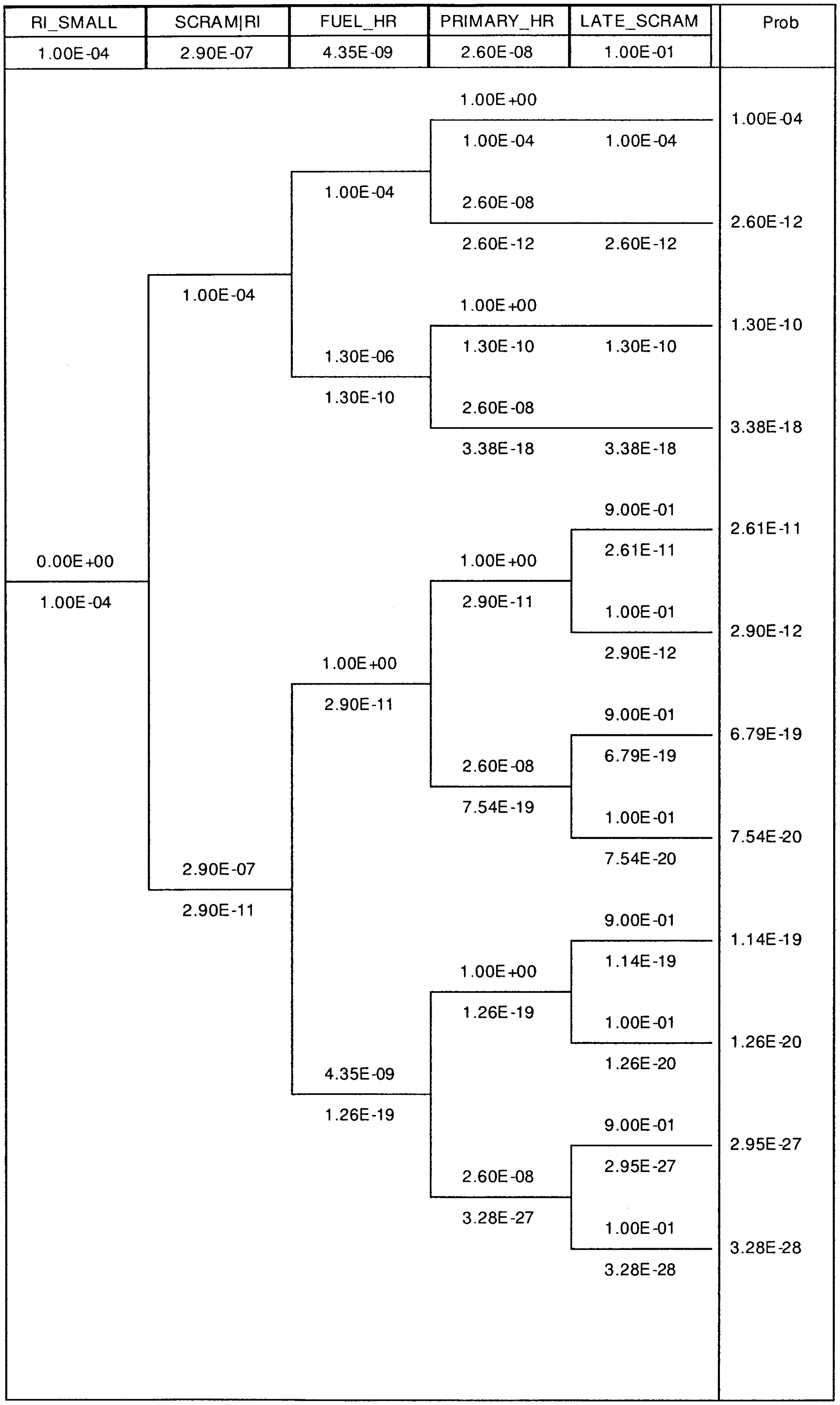
1.14E-20
1.26E-21
1.14E-28
1.26E-29


RI_MEDIUM
1.00E-04
SCRAM|RI
2.90E-07
FUEL_HR
4.35E-09
PRIMARY_HR
2.60E-08
LATE_SCRAM
1.00E-01
Prob
1.00E+00
1.00E-04
1.00E-04
1.00E+00
1.00E-04
2.60E-08
2.60E-12
2.60E-12
1.00E-04
1.00E+00
4.35E-13
4.35E-13
4.35E-09
4.35E-13
2.60E-08
1.13E-20
1.13E-20
9.00E-01
2.61E-11
0.00E+00
1.00E+00
1.00E-04
2.90E-11
1.00E-01
2.90E-12
1.00E+00
2.90E-11
9.00E-01
6.79E-19
2.60E-08
7.54E-19
1.00E-01
7.54E-20
2.90E-07
2.90E-11
9.00E-01
1.14E-19
1.00E+00
1.26E-19
1.00E-01
1.26E-20
4.35E-09
1.26E-19
9.00E-01
2.95E-27
2.60E-08
3.28E-27
1.00E-01
3.28E-28
1.00E-04
2.60E-12
4.35E-13
1.13E-20
2.61E-11
2.90E-12
6.79E-19
7.54E-20
1.14E-19
1.26E-20
2.95E-27
3.28E-28


RI_SMALL
SCRAM|RI
FUEL_HR
PRIMARY_HR
LATE_SCRAM
Prob
1.00E-04
2.90E-07
4.35E-09
2.60E-08
1.00E-01
1.00E+00
1.00E-04
1.00E-04
1.00E-04
1.00E-04
2.60E-08
2.60E-12
2.60E-12
2.60E-12
1.00E-04
1.00E+00
1.30E-10
1.30E-06
1.30E-10
1.30E-10
1.30E-10
2.60E-08
3.38E-18
3.38E-18
3.38E-18
9.00E-01
2.61E-11
0.00E+00
1.00E+00
2.61E-11
1.00E-04
2.90E-11
1.00E-01
2.90E-12
2.90E-12
1.00E+00
2.90E-11
9.00E-01
6.79E-19
6.79E-19
2.60E-08
7.54E-19
1.00E-01
7.54E-20
2.90E-07
7.54E-20
2.90E-11
9.00E-01
1.14E-19
1.00E+00
1.14E-19
1.26E-19
1.00E-01
1.26E-20
4.35E-09
1.26E-20
1.26E-19
9.00E-01
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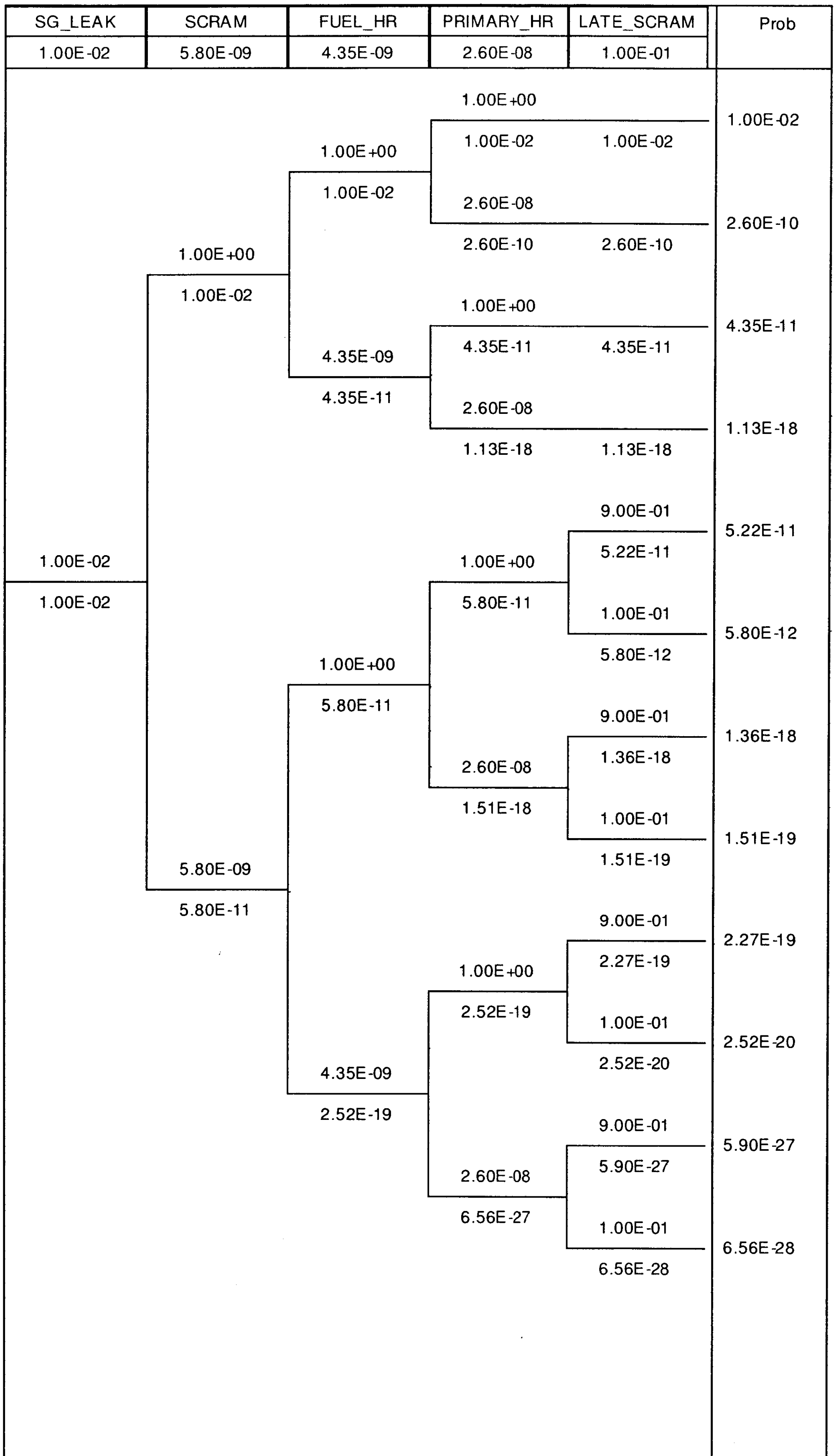
2.95E-27
2.60E-08
3.28E-27
1.00E-01
3.28E-28
3.28E-28


SG_LEAK
1.00E-02
SCRAM
5.80E-09
FUEL_HR
4.35E-09
PRIMARY_HR
2.60E-08
LATE_SCRAM
1.00E-01
Prob
1.00E+00
1.00E-02
1.00E-02
1.00E-02
1.00E+00
1.00E-02
2.60E-08
2.60E-10
2.60E-10
2.60E-10
1.00E+00
1.00E-02
1.00E+00
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4.35E-11
4.35E-11
4.35E-09
4.35E-11
2.60E-08
1.13E-18
1.13E-18
1.13E-18
9.00E-01
5.22E-11
5.22E-11
1.00E-02
1.00E-02
1.00E+00
5.80E-11
1.00E-01
5.80E-12
5.80E-12
1.00E+00
5.80E-11
9.00E-01
1.36E-18
1.36E-18
2.60E-08
1.51E-18
1.00E-01
1.51E-19
1.51E-19
5.80E-09
5.80E-11
9.00E-01
2.27E-19
2.27E-19
1.00E+00
2.52E-19
1.00E-01
2.52E-20
2.52E-20
4.35E-09
2.52E-19
9.00E-01
5.90E-27
5.90E-27
2.60E-08
6.56E-27
1.00E-01
6.56E-28
6.56E-28