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TITLE: THE EFFECT OF UNCERTAINTIES IN NUCLEAR REACTOR PLANT-SPECIFIC FAILURE DATA ON CORE DAMAGE FREQUENCY

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# The Effect of Uncertainties in Nuclear Reactor Plant-Specific Failure Data on Core Damage Frequency

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# EXECUTIVE SUMMARY

It is sometimes the case in PRA applications that reported plant-specific failure data are, in fact, only estimates which are uncertain. Even for detailed plant-specific data, the reported exposure time or number of demands is often only an estimate of the actual exposure time or number of demands. Likewise, the reported number of failure events or incidents is sometimes also uncertain because incident or malfunction reports may be ambiguous. In this report we determine the corresponding uncertainty in core damage frequency which can be attributed to such uncertainties in plant-specific data using a simple (but typical) nuclear power reactor example.

For the case in which all the plant-specific data used in the PRA are uncertain to the extent that each of the reported values may be in error by as much as a factor of 3, the width of the 90% uncertainty interval for the corresponding annual core damage frequency can increase by as much as a factor of 5. In addition, the mean core damage frequency can also increase by as much as a factor of 3.

Because it is now feasible to do so, we recommend that existing Level 1 PRA computer codes (such as the Level 1 codes in SAPHIRE) be internally enhanced and upgraded to accommodate such data uncertainties. This would provide a convenient way for PRA analysts to (1) properly account for such uncertainties in PRA applications, thus increasing the accuracy and precision of the PRA results; and (2) determine the driving data uncertainties which have the most relevant and influential effect on the PRA results.

## 1. BACKGROUND AND PURPOSE

An important aspect of Probabilistic Risk Assessment (PRA) concerns the accurate statistical representation of state-of-knowiedge uncertainties about the parameters of assumed stochastic models considered in the PRA. Apostolakis[1] refers to such uncertainties as *state-of-knowledge uncertainties* about the parameters of stochastic components in an appropriate "model-of-the-world." In particular, we focus our attention on such state-of-knowledge uncertainties in the estimated component *probability of failure upon demand p* for an assumed binomial distribution and the *event occurrence rate* $\lambda$ per unit time of an assumed Poisson distribution. The corresponding plant-specific binomial data consists of observing y failures in n total demands (usually at the basic-event or component level of the PRA). Similarly, the corresponding plant-specific Poisson data consists of observing x events (such as potential accident initiating events or component failures) in total exposure (or operating) time t.

It is sometimes the case that such reported plant-specific data are, in fact, only estimates which are uncertain. Although the statistical issue of uncertainties in (y, n) and (x, t) and their treatment is long-standing, it was first formally raised and discussed in PRA by Siu and Apostolakis[2] and Parry[3]. Mosleh[4] refers to such uncertainty as *hidden uncertainty*, because it is usually not explicitly quantified in the data analysis for PRA. Even for detailed plant-specific data, the reported exposure time t or number of demands is often only an estimate of the actual exposure time or number of demands. Mosleh[4] describes a case in which, for several possible reasons, a reported exposure time was subsequently found to be a factor of 3 smaller than the actual (true) exposure time.

The reported values of y or x can also be uncertain. For example, it is sometimes the case that the available information on a particular component incident (malfunction) report is so unclear and/or incomplete that a definitive classification of the event as having actually occurred is difficult or impossible. This leads to uncertainty regarding the true value of y or x.

Martz and Picard[5] and Martz, Kvam, and Atwood[6] present computationally convenient methods to account for uncertainties in either (x, t) or (y, n) when using Bayesian methods to estimate $\lambda$ or p, respectively. These methods bypass the need for numerical integration, thus making them easy to implement in practice. They also quantify the broadening effect that accounting for these uncertainties has on the usual 90% Bayesian credibility (or uncertainty) intervals for both $\lambda$ and p.

The purpose of this report is to determine the corresponding uncertainty in core damage frequency which can be attributed to uncertainties in both (x, t) and/or (y, n). To fully accomplish this purpose would require the use of numerous nuclear reactor plant-specific PRAs, which is clearly beyond the scope and resources available for this task. Thus, a single (but typical) simple example is used to make a preliminary assessment of the effect of accounting for (or, conversely, ignoring) such uncertainties. The example is described in Section 2 and Section 3 contains the results of the preliminary assessment. Some general recommendations are given in Section 4.

## 2. EXAMPLE

Consider the 2 nuclear power reactor standby emergency systems in Figure 1 which, among other things, are designed to operate in the event of a loss of offsite power (LOSP). The 2 systems depicted in Figure 1 are a 2-train standby emergency core cooling system (ECCS) and a 2-train standby containment spray system (CSS). Each of these 2 systems consists of 2 parallel trains having a pump, check valve (CV) and motor operated valve (MOV) in series. In addition to the pumps, valves, and storage tank indicated in Figure 1, there are also 2 emergency diesel generators (labeled DG-A and DG-B) which are used to provide emergency AC power for operating the motor operated valves and pumps in these 2 systems in the event of a LOSP. The particular diesel generator used to provide the AC power to operate the respective components is also indicated in Figure 1. We will also assume throughout that either one of the 2 parallel trains in each of these 2 systems is sufficient to accomplish the corresponding cooling and pressurization control functions; that is, for each system the success criterion we consider is 1-out-of-2.

Figure 2 shows the simple abbreviated event tree that we consider here in response to a LOSP event. The uppermost branch (or sequence) leads to no core damage, while the lower 2 accident sequences both lead to severe core damage. Sequence number 2 consists of a LOSP event (denoted by I) followed by failure of the ECCS (denoted by E) and success of the CSS (denoted by $\bar{C}$); thus, sequence 2 is denoted by $IE\bar{C}$ and leads to an anticipated small radionuclide release from the containment. Similarly, the bottom-most sequence (sequence number 3) in Figure 2 consists of the occurrence of the initiating event followed by the subsequent failure of both the ECCS and CSS. Thus, sequence 3 is designated IEC and leads to severe core damage and corresponding anticipated large radionuclide release from the containment.

Now consider fault trees for the failure of these 2 systems to operate on demand. Figures 3 and 4 show the fault trees for the ECCS and CSS top event failure to operate on demand (1-out-of-2 success criterion). Assuming that the similar component groups {DG-A, DG-B}, {E-PUMP-A, E-PUMP-B}, {C-PUMP-A, C-PUMP-B}, {E-MOV-A, E-MOV-B}, and {C-MOV-A, C-MOV-B} are each potentially susceptible to common cause failures, we consider the corresponding common cause basic events in Figures 3 and 4 given by CC-DG, CC-E-PUMP, CC-C-PUMP, CC-E-MOV, and CC-C-MOV. These events are used to account for all the unknown possible causes that potentially induce dependencies between the probability of failure of the 2 similar components in each of the respective groups.

Based on the fault tree models in Figures 3 and 4, Figures 5 and 6 give the minimal cut set representations for accident sequences 2 and 3 in Figure 2, respectively. Note that, while the representation given for sequence 3 contains only the 17 most probable minimal cut sets, the representation given for sequence 2 contains all the minimal cut sets. Because accident sequence 2 involves the success of the CSS, in order to remove logical inconsistencies the representation for sequence 2 requires the consideration and use of complementary CSS component events (which are depicted using the slash symbol "/" rather than a bar "—"). The expressions in Figures 5 and 6 will be used in Section 3 to quantify the corresponding accident sequence frequencies by propagating the basic event (component) plant-specific failure probabilities and associated uncertainties discussed below.

We consider the case of common, plant-specific data for the CV, MOV, diesel driven pump, and diesel generator. Table 1 contains the Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) plant-specific data[7] that we consider here for each of these components and for the required demand-dependent failure modes. Note also, however, that although these data are from different plants, they are representative of the data likely to be obtained from a typical plant and are adequate for our purposes here. The corresponding NUCLARR record number[7] is also indicated in Table 1.

As stated in the Introduction, we consider Bayesian estimation of the failure on demand probability p. For the case of a Jeffreys' noninformative prior distribution on p [which is a Beta(p; 0.5, 0.5) distribution; that is, a beta distribution with parameters $a = b = 0.5$] and y failures in n demands, the corresponding usual Bayesian point estimate (the posterior mean) of p is

$$\tilde{p} = \frac{y + 0.5}{n + 1}. \tag{1}$$

Note that the net effect of the prior is essentially to contribute 0.5 "prior" failures in 1 "prior" demand to the Bayes estimate. For each of the components and data listed in Table 1, the corresponding Bayesian point estimates $\tilde{p}$ are also given.

Now consider the LOSP initiating event. For a specific plant, Poisson data consisting of 3 LOSP events in 6 operating years were reported. Likewise, assuming a Jeffreys' noninformative prior distribution on the annual frequency of occurrence $\lambda$ [which is an improper Gamma($\lambda$; 0.5, 0) distribution; that is, a gamma distribution with shape parameter $a = 0.5$ and scale parameter $b = 0$] and x failures in exposure time t, the corresponding usual Bayesian point estimate (the posterior mean) of $\lambda$ is

$$\tilde{\lambda} = \frac{x + 0.5}{t}. \tag{2}$$

In the case where $x = 3$ and $t = 6$ years, we have $\tilde{\lambda} = 0.58$ events per year.

Let us now hypothetically assume that the values of y and n given in Table 1 are only "estimates" which are uncertain. Martz, Kvam, and Atwood[6] describe a methodology for use in accounting for uncertainties in y and n. For the case of uncertain n, they suggest using a subjective lognormal distribution to capture and express the uncertainty in n whose median (or mean) is given by the stated value (which is now considered to be only an estimate) of n. They then suggest approximating the true average posterior distribution of p by a beta distribution whose first 2 moments approximately match the corresponding moments of the true average posterior distribution of p. They also determined that this approximation is excellent in all of the cases that they have considered; consequently, we consider this beta approximation here. Also, a Jeffreys' noninformative Beta(p; 0.5, 0.5) distribution is used throughout.

Because of Mosleh[4]'s finding that exposure times can be in error by as much as a factor of 3, we treat this as the extreme (boundary) case and consider an approximate error factor of 3 in all cases in which n is uncertain. Thus, we consider a lognormal distribution having the specified mean listed in the "Demands n" column in Table 1 and corresponding error factor of 3 (at the 95% confidence level).

For the case of uncertain y or x, Martz, Kvam, and Atwood[6] propose using a maximum entropy distribution, having a specified finite support, whose mean is likewise given by the specified estimate of y listed in Table 1. We likewise consider an approximate factor of 3 error in y which we use to define the support for y. For each of the components in Table 1, as well as the LOSP initiating event, Table 2 gives the mean of y (or x) and associated support that we use to construct the corresponding maximum entropy distribution on y. Note that, except for LOSP, the upper limit of the support is a factor of 3 greater than the mean. The support for the LOSP uncertainty in x in Table 2 actually reflects true, plant-specific uncertainty in the number of LOSP events that should be counted as having occurred at the given facility.

Table 3 gives the beta distribution parameters a and b for the beta approximation to the average posterior distribution of p (see Reference 6) for each of the components for 4 different cases: no uncertainty in either y or n; only uncertainty in n; only uncertainty in y; and uncertainty in both y and n. The case in which there is no uncertainty in either y or n serves as a baseline case for comparison with the remaining 3 cases. Following Martz and Picard[5], Table 4 gives the gamma distribution shape and scale parameters a and b, respectively, for the gamma approximation to the average posterior distribution of the LOSP frequency per year $\lambda$ for these same 4 cases, which now involve x and t instead of y and n.

Now consider the increase in uncertainty in our Bayesian estimates of p and $\lambda$ as a consequence of accounting for the hypothesized uncertainty in (y, n) and (x, t). Table 5 gives the ratio of the width of the 90% Bayesian credibility interval for each of the 3 cases involving uncertainty in y and/or n (or x and/or t) to the width of the corresponding interval for the baseline case. For example, we see from Table 5 that, in the case of a CV, the 90% uncertainty interval for the probability of failure to operate per demand in which the y = 2 failures in n = 191 reported demands are both uncertain (to the extent described above) is over twice as wide (a factor of 2.2) than the corresponding interval in which there is no uncertainty in either of these values y or n. We also note that the indicated uncertainty in y or x appears to have less effect than the indicated uncertainty in n or t. Also, as expected, the case in which both y and n (or x and t) are uncertain has a greater effect than the case in which only y or n (or x or t) is uncertain.

## 3. CORE DAMAGE FREQUENCY RESULTS

We now consider the uncertainty in core damage frequency corresponding to the uncertainties in the component failure probabilities and LOSP frequency of occurrence presented in Section 2. We propagate the component uncertainties using Monte Carlo simulation in conjunction with the minimal cut set representations for core damage sequences 2 and 3 given in Figures 5 and 6. The Monte Carlo simulation was carried out to a depth of 10,000 replications.

The same 4 cases considered in Section 2 are likewise considered here: (1) no uncertainty in any of the (y, n) or (x, t) values; (2) uncertainty in only and all of the values of n and t; (3) uncertainty in only and all of the values of y and x; and (4) uncertainty in all of the (y, n) and (x, t) values. As in Section 2, the results from the first case serve as a baseline case for use in comparing the results from each of the remaining cases. In addition to these 4 cases, a fifth case is also considered. Because the number of plant-specific LOSP events x that occurred in the 6 year exposure time period is truly uncertain, it was decided to include the case in which only the LOSP x value is uncertain as a fifth case.

Table 6 gives the mean probabilities (the Bayesian point estimates) of each of the basic events in the fault trees in Figures 3 and 4 as well as the mean LOSP annual frequency of occurrence. We further assume here that the uncertainty in the probability p that the TANK fails on demand can be adequately expressed using a truncated lognormal distribution (truncated at 1.0) with parameters $\mu = -16$ and $\sigma = 1.4$. Note that these parameters produce a median value of p of $1.0 \times 10^{-7}$, a mean value of p of $2.7 \times 10^{-7}$, and an error factor of 10. Although all of the basic fault tree events are considered to be independent, because the plant-specific data pertains to each class of CVs, MOVs, diesel driven pumps, and diesel generators, we only considered 6 random variables in the simulation; namely, C-CV-A, C-MOV-1, C-PUMP-A, DG-A, TANK, and LOSP. All of the remaining basic events were considered to be perfectly correlated with these 6 basic events; thus, the same random probabilities generated for these 6 basic events were used in the Monte Carlo simulation for the remaining basic event probabilities in each corresponding class of components. For estimating the probability of each of the 2-component common cause basic events in Figures 3 and 4 we used a simple beta factor of 0.1 (see Table 6).

Table 7 gives the Monte Carlo results of the uncertainty analysis for the LOSP core damage sequence number 2 for each of the 5 cases. The row labeled "Mean Ratio" gives the ratio of the mean annual core damage frequency for each of the 4 cases to the baseline case. As in Section 2, the cases in which all values of n and t are uncertain yields means which are a factor of 2 or so larger than the baseline case. Similar results are observed for the median core damage frequency. The row labeled "Width of 90% Interval Ratio" gives the ratio of the width of the symmetric 90% Bayesian uncertainty interval on the annual core damage frequency for each of the 4 cases to the corresponding width for the baseline case. It thus measures the increase in the uncertainty of core damage frequency which can be directly attributed to uncertainties in the plant-specific data as indicated by each case. We observe that the cases in which all values of n and t are uncertain increase the overall uncertainty in core damage frequency by as much as a factor of 3. We also note that the case in which only the LOSP value of x is uncertain has virtually no effect on the mean, median, or width of the 90% interval.

Similarly, Table 8 gives the Monte Carlo uncertainty analysis results for sequence number 3 for each of the same 5 cases. The results are further exaggerated beyond those in Table 7. We see that, for the cases in which all the values of n and t are uncertain, the mean core damage frequency is more than a factor of 3 larger, while the width of the 90% uncertainty interval is between a factor of 4 to 5 wider, than the baseline case.

## 4. RECOMMENDATIONS

The results in Tables 7 and 8 are thought to be bounding results because all of the uncertainties considered for y, n, x, and t roughly represented an error factor of 3. We believe that a simultaneous error factor of 3 in all the plant-specific data values thus represents a clear bounding case. As to whether or not a fivefold increase in the uncertainty of core damage frequency is sufficiently important to merit further study of this issue, depends on the particular situation regarding the use and importance of the Level 1 PRA results. Such an increase either may or may not be sufficiently important to warrant further consideration of such data uncertainties. Because not only the uncertainty in core damage frequency, but the mean and median as well, also increase in proportion to the degree of uncertainty in the plant-specific data, it is our belief that more careful attention should be given in future PRAs to the consideration, accommodation, and propagation of such data uncertainties when there is justifiable reasons for their existence.

Further, because it is now feasible to do so, existing Level 1 PRA computer codes (such as the Level 1 codes in SAPHIRE) should be internally enhanced and upgraded to accommodate such data uncertainties. This would provide a convenient way for PRA analysts to (1) properly account for such uncertainties in PRA practice, thus increasing the accuracy and precision of the PRA results; and (2) determine the driving data uncertainties which have the most relevant and influential effect on the PRA results.

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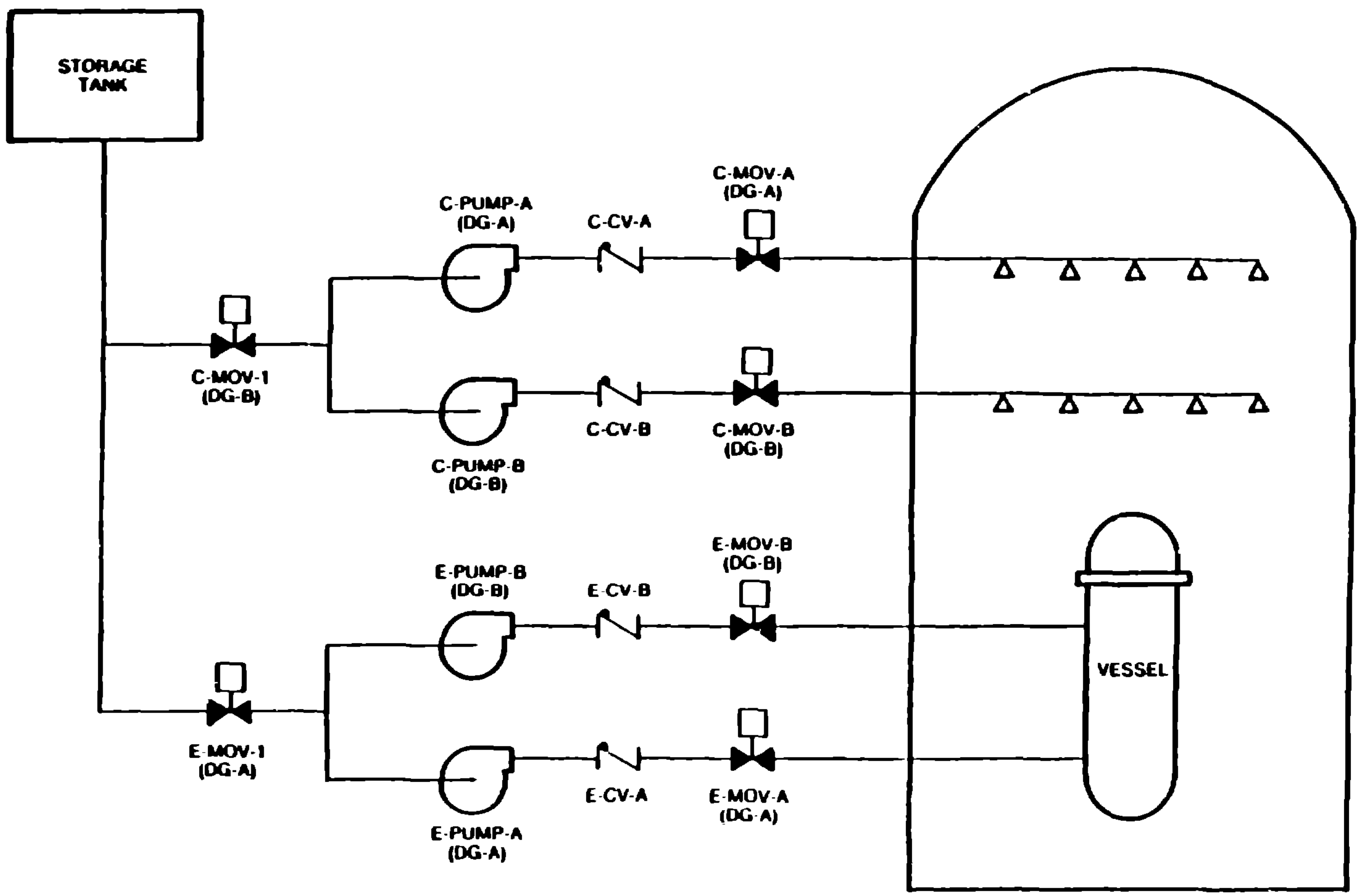


Figure 1. The Emergency Reactor Core Cooling and Containment Spray System Configurations for Use in the Event of Loss of Offsite Power in Our Nuclear Power Plant Example.

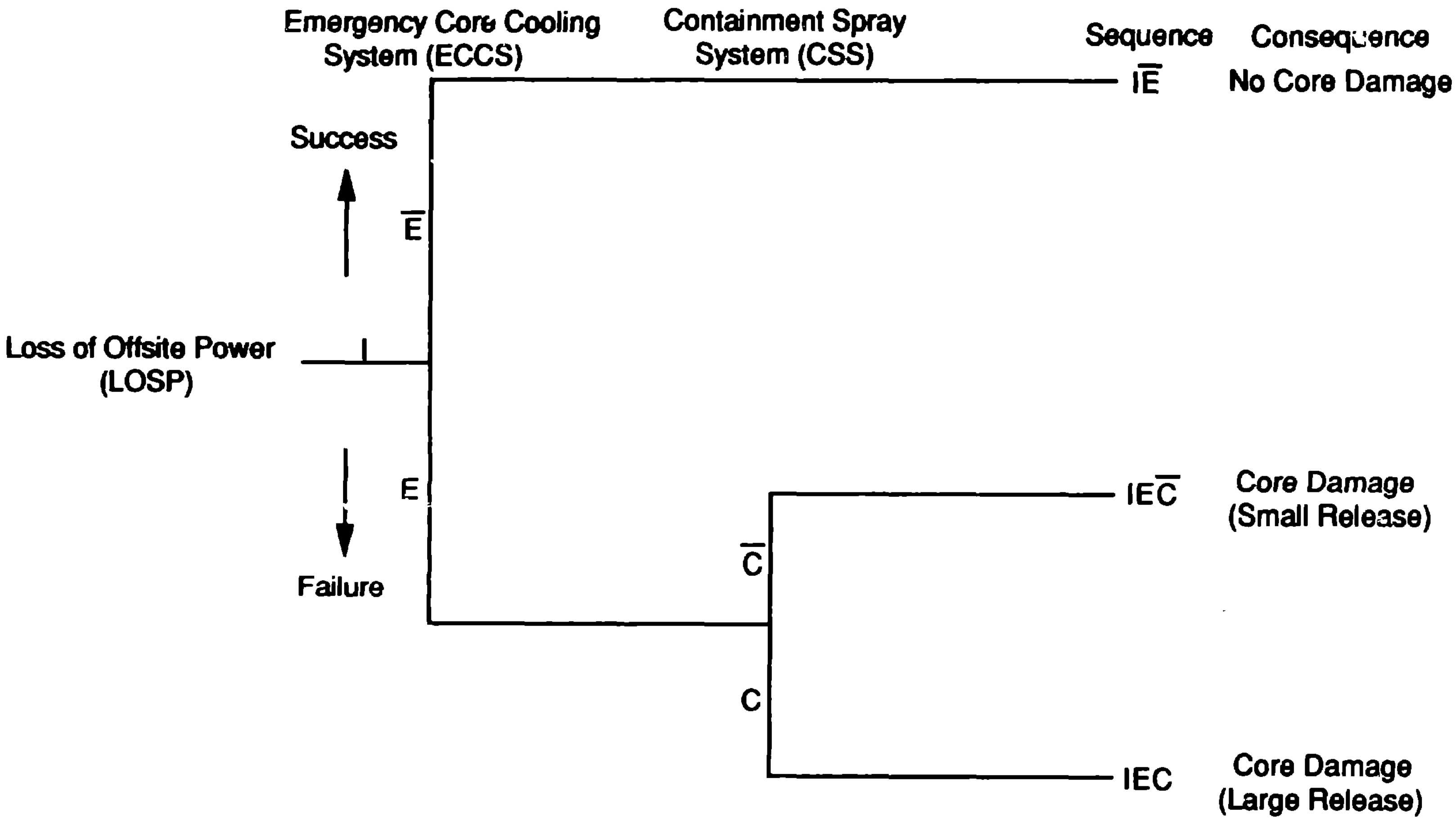


Figure 2. The Loss of Offsite Power Event Tree in Our Nuclear Power Plant Example.

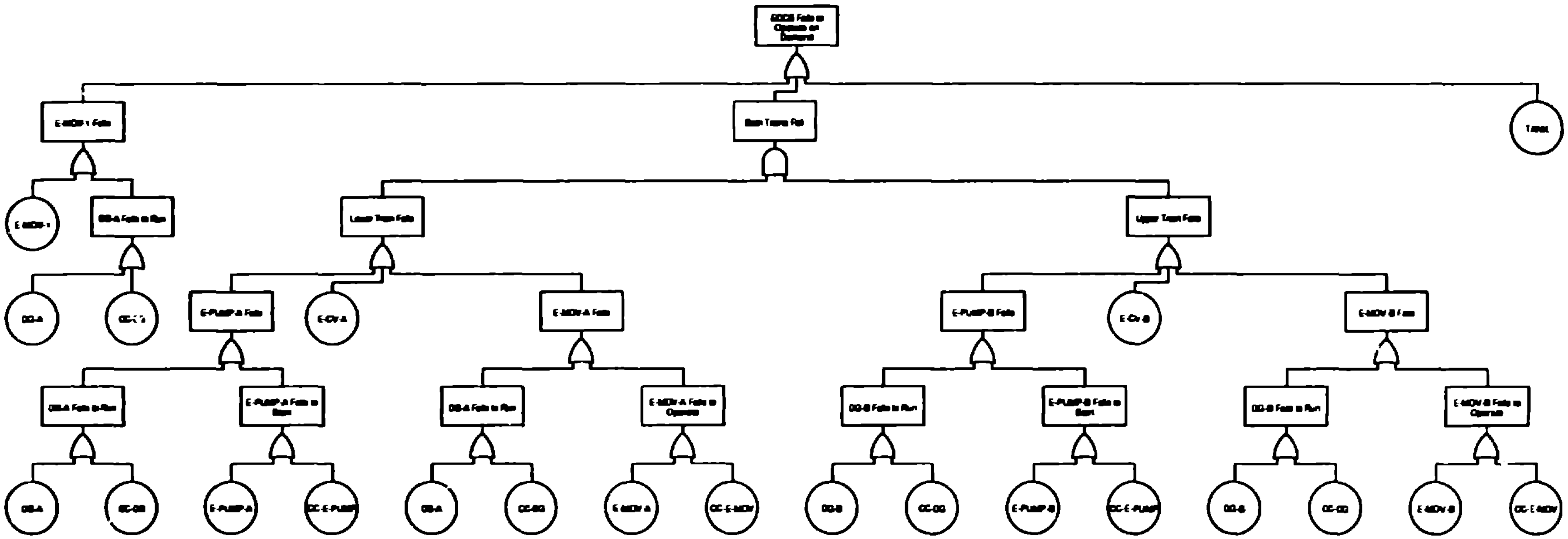

Figure 3. Fault Tree for the Failure of the ECCS to Operate on Demand (1-out-of-2 Success Criterion).

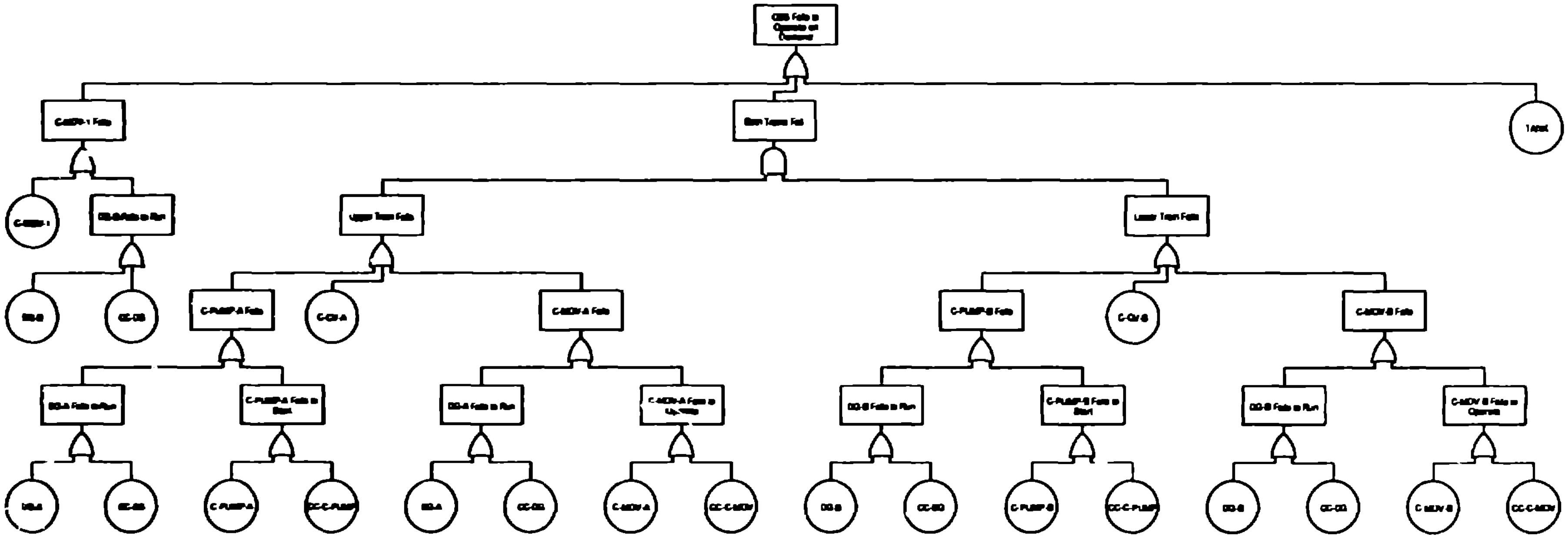

Figure 4. Fault Tree for the Failure of the CSS to Operate on Demand (1-out-of-2 Success Criterion).

SEQUENCE-2 =
LOSP * CC-E-MOV * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
LOSP * CC-E-PUMP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-MOV-1 * LOSP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
DG-A * LOSP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
LOSP * CC-E-MOV * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK */CC-DG * /CC-C-PUMP * /CC-C-MOV +
LOSP * CC-E-PUMP * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-MOV-1 * LOSP * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-PUMP-A * E-PUMP-B * LOSP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-MOV-B * E-PUMP-A * LOSP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-CV-B * E-PUMP-A * LOSP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-MOV-A * E-PUMP-B * LOSP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-MOV-A * E-MOV-B * LOSP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-CV-B * E-MOV-A * LOSP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-CV-A * E-PUMP-B * LOSP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-CV-A * E-MOV-B * LOSP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-CV-A * E-CV-B * LOSP * /C-CV-B * /C-MOV-1 * /C-MOV-B * /C-PUMP-B * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-PUMP-A * E-PUMP-B * LOSP * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-MOV-B * E-PUMP-A * LOSP * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-CV-B * E-PUMP-A * LOSP * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-MOV-A * E-PUMP-B * LOSP * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-MOV-A * E-MOV-B * LOSP * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-CV-B * E-MOV-A * LOSP * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-CV-A * E-PUMP-B * LOSP * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-CV-A * E-MOV-B * LOSP * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV +
E-CV-A * E-CV-B * LOSP * /C-CV-A * /C-MOV-1 * /C-MOV-A * /C-PUMP-A * /DG-A * /DG-B * /TANK * /CC-DG * /CC-C-PUMP * /CC-C-MOV

Figure 5. Minimal Cut Set Representation for LOSP Core Damage (Small Release) Accident Sequence (Sequence No.2).

SEQUENCE-3 =

CC-DG * LOSP +
DG-A * DG-B * LOSP +
DG-A * C-MOV-B * LOSP +
DG-B * E-MOV-A * LOSP +
DG-A * C-PUMP-B * LOSP +
DG-B * E-PUMP-A * LOSP +
E-MOV-1 * DG-B * LOSP +
DG-A * C-MOV-1 * LOSP +
CC-E-MOV * DG-B * LOSP +
DG-A * CC-C-MOV * LOSP +
DG-A * CC-C-PUMP * LOSP +
CC-E-PUMP * DG-B * LOSP +
DG-A * C-CV-B * LOSP +
E-CV-A * DG-B * LOSP +
E-MOV-1 * C-MOV-1 * LOSP +
E-MOV-1 * CC-C-MOV * LOSP +
CC-E-MOV * C-MOV-1 * LOSP

Figure 6. Minimal Cut Set Representation for LOSP Core Damage (Large Release) Accident Sequence (Sequence No. 3).

Table 1. NUCLARR[7] Plant-Specific Component Failure Data

| Component | NUCLARR Record No. | Failure Mode (/D) | NUCLARR System Code | Failures y | Demands n | $\tilde{p}$ |
|---|---|---|---|---|---|---|
| Check Valve | 461 | Fails to Operate | BA* | 2 | 191 | 1.3E-02 |
| Motor Operated Valve | 687 | Fails to Operate | BA* | 9 | 879 | 1.1E-02 |
| Diesel-Driven Pump | 130 | Fails to Start | BA* | 1 | 60 | 2.5E-02 |
| Diesel Generator | 1336 | Fails to Run | ••• | 3 | 179 | 1.9E-02 |

* Auxiliary/emergency feedwater system (PWR)

Table 2. Maximum Entropy Distribution Mean and Support Considered When y (or x) Is Uncertain

| Component/ Initiating Event | Mean of y (or x) | Support for y (or x) |
|---|---|---|
| Check Valve | 2 | {0,1,2,3,4,5,6} |
| Motor Operated Valve | 9 | {3,4,5,6,7,8,9,10,11,12,13,14,15,16,17,18,19,20,21,22,23,24,25,26,27} |
| Diesel Driven Pump | 1 | {0,1,2,3} |
| Diesel Generator | 3 | {0,1,2,3,4,5,6,7,8,9} |
| LOSP | 3 | {1,2,3,4} |

Table 3. Beta Posterior Distribution Parameter Values for All Combinations of Uncertainty in the Plant-Specific Component Failure Data (y, n)

| Component | Failures y | Demands n | Neither y nor n Uncertain a | Neither y nor n Uncertain b | Only n Uncertain a | Only n Uncertain b | Only y Uncertain a | Only y Uncertain b | Both y and n Uncertain a | Both y and n Uncertain b |
|---|---|---|---|---|---|---|---|---|---|---|
| Check Valve | 2 | 191 | 2.500 | 189.500 | 1.470 | 71.067 | 1.046 | 79.304 | 1.093 | 52.847 |
| Motor Operated Valve | 9 | 879 | 9.500 | 870.500 | 2.842 | 165.842 | 2.167 | 198.521 | 1.999 | 116.649 |
| Diesel Driven Pump | 1 | 60 | 1.500 | 59.500 | 1.012 | 25.638 | 0.852 | 33.790 | 0.825 | 20.902 |
| Diesel Generator | 3 | 179 | 3.500 | 176.500 | 1.793 | 57.483 | 1.149 | 57.963 | 1.240 | 39.757 |

Table 4. Gamma Posterior Distribution Parameter Values for All Combinations of Uncertainty in the Plant-Specific Initiating Event Data (x, t)

| | Events | Exposure Time (yrs.) | Neither x nor t Uncertain | | Only t Uncertain | | Only x Uncertain | | Both x and t Uncertain | |
|---|---|---|---|---|---|---|---|---|---|---|
| Initiating Event | x | t | a | b | a | b | a | b | a | b |
| Loss of Offsite Power (LOSP) | 3 | 6 | 3.500 | 6.000 | 1.935 | 2.432 | 2.674 | 4.583 | 1.772 | 2.227 |

Table 5. Ratios of the Widths of the 90% Posterior Credibility Intervals on $p$ and $\lambda$

| Component/Initiating Event | Only n (or t) Uncertain | Only y (or x) Uncertain | Both y and n (or x and t) Uncertain |
|---|---|---|---|
| Check Valve | 2.0 | 1.5 | 2.2 |
| Motor Operated Valve | 2.8 | 2.0 | 3.2 |
| Diesel-Driven Pump | 1.8 | 1.3 | 2.0 |
| Diesel Generator | 2.1 | 1.6 | 2.5 |
| LOSP | 1.8 | 1.1 | 1.9 |

Table 6. Mean Basic Event Probabilities and Mean Initiating Event Frequency

| Basic Event Name | Probability (/D) |
|---|---|
| C-CV-A | 1.3E-02 |
| C-CV-B | 1.3E-02 |
| C-MOV-1 | 1.1E-02 |
| C-MOV-A | 1.1E-02 |
| C-MOV-B | 1.1E-02 |
| C-PUMP-A | 2.5E-02 |
| C-PUMP-B | 2.5E-02 |
| DG-A | 1.9E-02 |
| DG-B | 1.9E-02 |
| TANK | 2.7E-07 |
| E-CV-A | 1.3E-02 |
| E-CV-B | 1.3E-02 |
| E-MOV-1 | 1.1E-02 |
| E-MOV-A | 1.1E-02 |
| E-MOV-B | 1.1E-02 |
| E-PUMP-A | 2.5E-02 |
| E-PUMP-B | 2.5E-02 |
| LOSP | 0.58* |
| CC-DG | 1.9E-03 |
| CC-E-PUMP | 2.5E-03 |
| CC-C-PUMP | 2.5E-03 |
| CC-E-MOV | 1.1E-03 |
| CC-C-MOV | 1.1E-03 |

* Frequency (/yr.)

Table 7. Uncertainty Analysis Results for LOSP Core Damage (Small Release) Accident Sequence Number 2

| Statistic | Case | | | | |
|---|---|---|---|---|---|
| | No (y, n) nor (x, t) Values Uncertain (Baseline) | All n and t Values Uncertain | All y and x Values Uncertain | Only LOSP x Value Uncertain | All (y, n) and (x, t) Values Uncertain |
| Mean | 2.84E-02 | 6.11E-02 | 2.85E-02 | 2.85E-02 | 6.10E-02 |
| ***Mean Ratio*** | *---* | ***2.2*** | ***1.0*** | ***1.0*** | ***2.1*** |
| Standard Error of the Mean | 1.75E-04 | 5.29E-04 | 2.33E-04 | 1.95E-04 | 5.79E-04 |
| Median (approximate) | 2.47E-02 | 4.64E-02 | 2.23E-02 | 2.40E-02 | 4.33E-02 |
| ***Median Ratio*** | *---* | ***1.9*** | ***0.9*** | ***1.0*** | ***1.8*** |
| Mode (approximate) | 1.90E-02 | 1.77E-02 | 1.13E-02 | 1.69E-02 | 1.52E-02 |
| Standard Deviation | 1.75E-02 | 5.29E-02 | 2.33E-02 | 1.95E-02 | 5.79E-02 |
| Variance | 3.05E-04 | 2.80E-03 | 5.42E-04 | 3.79E-04 | 3.36E-03 |
| Skewness | 1.54 | 2.27 | 1.99 | 1.58 | 2.31 |
| Kurtosis | 7.22 | 12.64 | 9.32 | 7.10 | 11.76 |
| Coefficient of Variation | 0.62 | 0.87 | 0.82 | 0.68 | 0.95 |
| 0.025 Quantile | 5.97E-03 | 5.56E-03 | 3.35E-03 | 4.61E-03 | 4.32E-03 |
| 0.05 Quantile | 7.87E-03 | 8.48E-03 | 4.74E-03 | 6.29E-03 | 6.58E-03 |
| 0.95 Quantile | 2.47E-02 | 4.64E-02 | 2.23E-02 | 2.40E-02 | 4.33E-02 |
| 0.975 Quantile | 6.16E-02 | 1.62E-01 | 7.46E-02 | 6.58E-02 | 1.75E-01 |
| Width of 90% Uncertainty Interval | 7.23E-02 | 2.00E-01 | 9.13E-02 | 7.86E-02 | 2.12E-01 |
| ***Width of 90% Interval Ratio*** | *---* | ***2.8*** | ***1.3*** | ***1.1*** | ***2.9*** |

Table 8. Uncertainty Analysis Results for LOSP Core Damage (Large Release) Accident Sequence Number 3

| Statistic | Case | | | | |
|---|---|---|---|---|---|
| | No (y, n) nor (x, t) Values Uncertain (Baseline) | All n and t Values Uncertain | All y and x Values Uncertain | Only LOSP x Value Uncertain | All (y, n) and (x, t) Values Uncertain |
| Mean | 1.92E-03 | 6.38E-03 | 2.07E-03 | 1.92E-03 | 6.41E-03 |
| ***Mean Ratio*** | *---* | ***3.3*** | ***1.1*** | ***1.0*** | ***3.3*** |
| Standard Error of the Mean | 1.76E-05 | 8.49E-05 | 3.20E-05 | 1.93E-05 | 1.01E-04 |
| Median (approximate) | 1.40E-03 | 3.50E-03 | 1.01E-03 | 1.35E-03 | 3.12E-03 |
| ***Median Ratio*** | *---* | ***2.5*** | ***0.7*** | ***1.0*** | ***2.2*** |
| Mode (approximate) | 7.53E-04 | 6.00E-04 | 3.17E-04 | 7.03E-04 | 1.20E-03 |
| Standard Deviation | 1.76E-03 | 8.49E-03 | 3.20E-03 | 1.93E-03 | 1.01E-02 |
| Variance | 3.09E-06 | 7.21E-05 | 1.02E-05 | 3.72E-06 | 1.02E-04 |
| Skewness | 2.63 | 3.64 | 5.00 | 3.11 | 5.23 |
| Kurtosis | 16.29 | 24.56 | 49.86 | 20.65 | 58.12 |
| Coefficient of Variation | 0.92 | 1.33 | 1.55 | 1.00 | 1.57 |
| 0.025 Quantile | 2.14E-04 | 1.74E-04 | 3.87E-05 | 1.64E-04 | 1.06E-04 |
| 0.05 Quantile | 2.98E-04 | 3.36E-04 | 7.71E-05 | 2.44E-04 | 2.11E-04 |
| 0.95 Quantile | 1.40E-03 | 3.50E-03 | 1.01E-03 | 1.35E-03 | 3.12E-03 |
| 0.975 Quantile | 5.32E-03 | 2.24E-02 | 7.53E-03 | 5.52E-03 | 2.35E-02 |
| Width of 90% Uncertainty Interval | 6.65E-03 | 2.93E-02 | 1.05E-02 | 7.15E-03 | 3.31E-02 |
| ***Width of 90% Interval Ratio*** | *---* | ***4.4*** | ***1.6*** | ***1.1*** | ***5.0*** |