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September 6, 1982

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Zion Station Units 1 and 2
Zion Probabilistic Safety Study
NRC Docket Nos. 50-295 and 50-304

References (a): February 24, 1982, letter from T. M. Novak to G. T. Klopp.

(b): March 10, 1982, letter from S. A. Varga to L. O. DelGeorge.

(c): May 18, 1982, letter from S. A. Varga to L. O. DelGeorge.

(d): June 29, 1982, letter from W. L. Stiede to H. R. Denton.

Dear Mr. Denton:

This is to provide Commonwealth Edison's response to the questions by the NRC staff and ACRS consultants on the Zion Probabilistic Safety Study. Responses are provided for the questions contained in Enclosures 1 and 3 of reference (c). Also provided are comments on the reviews of the study performed by Brookhaven (reference (a)) and Sandia (reference (b)).

Upon consideration of the questions received, Commonwealth Edison has elected to revise the event trees for turbine trip with loss of offsite power and ATWS contained in section 1 of the report. Page additions to the appropriate sections of the report have been included in this transmittal. The revisions to those sections do not change our previous conclusion that the risk from these event scenarios is acceptably low.

Fifty (50) copies of the question responses and the review comments have been provided for your use. Fifty (50) copies of the revised pages to the report have also been provided.

Please address questions regarding this matter to this office.

Very truly yours,

F. G. Lentine

F. G. Lentine
Nuclear Licensing Administrator

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ENCLOSURE 1 QUESTIONS

Summary of ACRS Review Questions and Comments
on the Zion Probabilistic Safety Study (PSS)

I. General Comments

It would be useful for the licensee to indicate the strong point of his study and the weak points (i.e., those points that are least subject to a rigorous defense.)

Response

The Zion Probabilistic Safety Study includes what we feel to be an appropriate discussion of such matters in Volume 1, section II. 10, "Reflections, Advancements, and Limitations".

II. METHODOLOCGY AND EXECUTION OF THE ZION PSS

II.1 Comments on Propagation of Uncertainties

(1) The implication of section 0.13 that a systematic methodology for generating uncertainties was used throughout the study is somewhat misleading. It suggests a possible disconnect between the methodology authors of section 0 and the engineers who actually arrived at the probability distributions assigned to the various branch points of the accident sequence. This seems especially true in the case of the containment matrix. (ANL, Sec. 2.1).

Response

We disagree with this comment. It is our position that systematic development of uncertainties is pursued in the Zion PSS. Due to the natures of the various analyses and subject matter being analyzed, the tools for establishing uncertainty varied. This does not imply a "disconnect." In fact, the authors of section were deeply involved with the establishment of uncertainty in all phases of the work, including the containment matrix.

II. METHODOLOGY AND EXECUTION OF THE ZION PSS

II.1 Comments on Propagation of Uncertainties

(2) Using the point estimate of risk to rank the release categories for which uncertainty calculations would actually be performed suggests two questions: First, were the aforementioned dominant categories the same for all risk measures? Second, would another ranking indicator, e.g., uncertainty in risk, lead to the same ranking. With this latter measure, the release category that contributed most to the uncertainty in risk would be ranked number one and so forth. Although, the answers to these questions may well support and vindicate the calculations actually performed in the study, we did not find any indication that these questions were addressed. (ANL, 2.1)

Response

First, with respect to the performance of uncertainty calculations, the release categories were not "ranked" as such. What was done, simply, was that all those release categories which were significant, either by virtue of frequency of occurrence or by virtue of severity (i.e., by contribution to any of the five damage indices) were given a full uncertainty treatment. Those categories whose contribution to the final curves was insignificant were not. That is all there is to it.

Second, the question seems to be asking whether a release category which is a small contributor to health effects frequency on the basis of mean value (i.e., mean frequency) could be, nevertheless, a large contributor on the basis of uncertainty. This actually cannot happen numerically, especially with our "logarithmic type" probability curves, since any sizable probability area on the high end of the scale will necessarily pull the mean value up and make it significant also. Thus, the release categories which dominate on the basis of mean frequencies also dominate when uncertainty is considered.

Third, on the general matter of adding probability curves, the situation is very comparable to adding a string of ordinary numbers. If there are some small numbers in the string which, in aggregate, do not affect the sum (to the level of round-off of the result) then in engineering parlance, we say that these small numbers can be "neglected" in the addition operation. However, they are not "neglected" in the sense of "negligence." In fact, we could argue that they are actually included. They just do not affect the result. The situation in adding probability curves is entirely analogous. Those curves which do not affect the final result curve can be omitted in the numerics.

Fourth, in the Zion calculations, a single release category, 2R, was the major contributor to all health effects. This category had both the highest mean frequency and also the largest uncertainty. The major source of 2R releases was seismically initiated melt sequences.

II. METHODOLOGY AND EXECUTION OF THE ZION PSS

II.1 Comments on Propagation of Uncertainties

(3) Review of the phenomenology associated with the containment matrix event trees has led to some question as to whether the probabilities assigned to the branch points were in some cases optimistic with respect to both value and uncertainty range. If probabilities and attendant uncertainties assigned to these branch points have been optimistic, short cuts used in assessing the uncertainty bands may be invalid. In many cases with respect to the containment matrix, it appears that the treatment was such that no uncertainty value was assigned. Specifically, uncertainties in branch points having 1-E or E probabilities were ignored -- obviously this is only justified if the confidence that is implied by assigning these probabilities is justified. (ANL, 2.1)

Response

It is our position that the split fractions assigned to the branch points of the containment event tree are not "optimistic". They represent best estimate values and uncertainty has been applied, as noted, for all but the $1-\epsilon$ or ϵ split fraction assignments. In these cases, the split frac- tion is used as a "place keeper" to insure that a consistant treatment of the event tree occurs. Such usage occurs where we are very certain that a branch will or will not be followed. The certainty in these cases derives from both the phonomenological evaluations in section 3 and the sensitivity studies and best estimate transient analyses in section 4.

High confidence was placed in the ϵ probability assignments. This was because while the physical parameters (pressure, temperature, etc.) of the accident sequence phenomenology may have fairly large uncertainty bands, the reference levels (containment failure pressure, hydrogen flame temperature criteria, minimum levels of water for debris bed coolability) were not approached in these cases. As a further check, sensitivity runs have been made on the value of ϵ. In the base case, ϵ was 10^{-4} in most cases. The internal risk of the plants is not changed significantly until ϵ nears 10^{-2}. These runs were done with the change in ϵ applied to all the ϵ's and then multiplied together without any monte carlo or other techniques used. Thus, uncertainties in physical parameters were followed and incorporated, but their impact was small.

II.2 Comments on Data Analysis

(1) To widen the published uncertainty bands associated with equipment failure rates, the lognormal distribution is fitted to "data" by matching the 20th and 80th percentiles to the ends of the published data range. Reference 0-17 of the Zion PSS is quoted as stating that 20-50% of "true values" fall outside judgementally obtained 98% confidence bounds. If we simply assume the pessimistic side of this estimate, that 50% of opinions on failure rates are wrong, then the 25th and 75th percentiles should be used. In any case, more justification for matching distributions should be provided. (ANL, C 2.2)

Response

The distributions of failure rates that are used reflect the study team's state of knowledge. Each case was evaluated individually and the reported distribution was the concensus of the analysts.

In the process of quantifying our state of knowledge, we felt that we had to take into account published evidence that experts tend to overestimate their knowledge (e.g., Reference 0.17 of the Zion report) as well as our own experience with generic distributions (Nuclear Engineering and Design, Vol. 56, pp. 321-329, 1980). We, therefore, decided initially to broaden the distributions. It must be pointed out--as unfortunately, it was not in the Zion report-- that this broadening was only an initial step and that the resulting distributions still had to withstand the scrutiny of the analyst. It turned out that the use of 20/80 percentiles resulted, in most (but not all) cases, in distributions that the team found acceptable.

In this context, our Reference 0.17 is seen to be only one of the various factors that affected our state of knowledge. The reviewers seem to assign to it much more weight than we did and they would like to see us be "pessimistic" and use the 25/75 percentiles in lieu of the 20/80 pair. Of course, there can be no "proof" that one must use one pair over the other. We simply did not feel that we had to distort our state of knowledge in order to be "pessimistic" and take literally all the statements contained in Reference 0.17. Besides, the real question is whether it would impact on the result in a significant way. Given that most of these generic distributions are specialized using Zion specific data, the effect of the use of 25/75 versus 20/80 is indeed insignificant.

II.2 Comments on Data Analysis

(2) The WASH 1400 bounds are not consistently used as the 20th and 80th percentiles. Easterling has written a few words on the treatment of the V sequence, a dominant sequence in the Zion and many other studies, and has shown that the sequence mean changes about four orders of magnitude depending on whether the WASH-1400 parameter bounds are used as 5th and 95th percentiles, or as 20th and 80th percentiles, a choice that seems to be highly subjective if not somewhat arbitrary. If this choice is arbitrary and if the Easterling calculations do reflect what was done in the Zion study (it is not always easy to tell), then one must conclude that the methodology allows one to get any answers one wishes within the four orders of magnitude. (JH, p.6)

Response

There is an old saying to the effect that consistency is the hallmark of mediocre minds. The point of this is that consistency itself is not necessarily a virtue; it depends on what we are consistent about.

In the Zion study, we have tried very hard to be consistent about using our brains and our good engineering judgment; about thinking through each situation rather than mindlessly using mechanical formulas or routine computer programs.

In particular, in the use of Bayes' theorem to quantify our state of knowledge regarding component failure rates, we have consistently used the prior distribution to encode our state of knowledge based on all the information we have except the specific operating experience of the Zion plant. For most components (but not all), we used as prior the distributions given in WASH-1400, stretched so that the 5th and 95th percentiles became 20th and 80th percentiles. We did this everywhere the resulting distribution reflected our state of knowledge. Where it did not, we of course did not use the 20-80 process. To do so would have been inconsistent and would have sacrificed the fundamental meaning of our Bayesian approach in order to adhere to a mechanical recipe.

What Easterling has done in his few words, is to produce a dramatic example of a well known fact; namely that if one has a probability distribution, say for a quantify λ, on a log scale, and if one broadens the distribution, then the mean of the distribution will change substantially. This is because on a log scale, the mean of the curve is entirely dominated by the high side tail. The rest of the course has essentially no impact on the mean. In particular, Easterling's example shows that if one is interested in the mean of lambda squared, the sensitivity is even stronger so that a change from 5-95 to 20-80 can give a change four or five orders of magnitude in the mean, even though the change in the overall appearance of the curves is rather small ("seemingly innocent" as he calls it).

We are, therefore, grateful to Easterling for providing us with this example to help us communicate a point we have been preaching for the last few years to anyone who would listen; namely, that the mean is a very nonrobust parameter of a probability curve on log scale and, therefore, is very unsuitable for use in risk comparisons or risk goals. Indeed, it is an unsuitable measure for expressing risk, period. We therefore preach that one should look at the whole probability curve. Any single parameter of the curve, especially the mean, is misleading. We need to learn to "think curves," and to regard "risk" as being quantified and expressed by the whole curve. From this point of view, an innocent change remains an innocent change.

For additional details concerning the V sequence and its quantification, see response to Question III.3(4) and NRC Estimation Methodology Question 6.

II.3 Comments on Human Error

(1) There appears to be a somewhat arbitrary decision to assign the 10/90 percentiles in matching the lognormal distribution to human error rates. The 20/80 percentiles were chosen to represent equipment failure rates. Our ignorance of human error rates exceeds our ignorance of equipment failure rates. Therefore, assuming the 20/80 choice to be correct for equipment failure, the choice of the 10/90 percentile band for human error rates appears to be optimistic and counter to our present state of understanding. The matched distribution should be broader than the analogous equipment distribution, i.e., it should be matched using, say, a 30/70 choice. Obviously, whatever choice is made should be defended with stronger arguments than are now provided. (ANL, C 2.3)

Response

There is a misunderstanding here in the sense that the reviewers talk about the choice of 20/80 and 10/90 pairs regardless of what the actual expert opinions are. While it is true that we know less about human error than we do about equipment failures, it is also true that the authors of the Human Reliability Handbook were very careful to reflect this uncertainty in the upper and lower bounds. Consequently, we did not feel that we had to stretch their distributions as much as those of failure rates. The process that we followed in deciding what distributions to use was the same as that described in our response to Question II.2(1).

II.3 Comments on Human Error

(2) The treatment of dependence appeared to be optimistic in predicting collaborative operator failure rates. We recommend human error experts be consulted. (ANL, 2.3)

Response

We disagree with the comment. For more extensive discussions, refer to our response to enclosure 3 questions "Human Reliability Analysis".

II.3 Comments on Human Error

(3) High stress situations are stated as being handled on a case-by-case basis. Several obvious questions should be answered: (1) What is the general basis for handling individual cases? (2) What is the impact of high stress situations on the results of this study? (3) Where and for what events is high stress behavior most critical? (4) How do high stress operator failures rates used in this study compare to low stress failure rates? (ANL, C 2.3)

Response

1. The general basis is to employ the basic values in the Human Reliability Handbook and, as appropriate, to adjust these values to account for available information, available time, the extent and nature of operator training and drills, and other pertinent factors that may apply.

2. The impact of such situations is very minor.

3. The judgement of what constitutes "most critical" will vary from individual to individual. None of the situations examined had any major impact on risk. Human error on the switchover to recircultion cooling for large LOCA initiators can be judged as one of the more important factors in core melt frequency. Due to timing, this is judged to be a high stress situation.

4. A general comparison is not felt to be germaine due to the variety of factors involved in various specific sequence evaluations.

II.3 Comments on Human Error

(4) Finally, the report states on p. 1.3.3 "While errors on commission to misunderstanding of correct or mostly correct indications (as at TMI-2) are not explicitly modeled, it is felt that the above approach on human error accounts for such events." Upon what is this feeling based? (WCL, p.5)

Response

In each event tree, long term cooling requires at least one successful human action. We view failure at that point in the event sequence quite broadly. It includes not only failure to follow identified procedural steps but also errors of commission due to misinterpretation of plant conditions which interfere with otherwise successful core cooling functions (e.g., Section 7.3). Our viewpoint is implemented by first assigning median human error rates based on most likely plant conditions and instrument displays. Then we consider the potential sources of variability along any particular accident sequence in assigning uncertainty bounds around those median human error rates. Because of possible variabilities with respect to exact plant conditions, instrument performance, confusion (i.e., the range of events competing for the operators' attention), and average operator capability, we usually assign broad human error uncertainties.

II.4 Comments on System Modeling and Simplification Procedures

(1) As with other studies, the Zion study team turned to event trees and fault trees for cataloguing accident sequences. However, their use of event trees is somewhat different. They have chosen to carry several support system faults (e.g., AC bus failures) in the event tree. Such an approach limits the number of support system failure states that can be explicitly modeled, and which ones are modeled is decided by the analyst based usually on a probabilistic argument. Such simplifications result in models that have limited utility for future studies. (JH, p.5)

Response

Commonwealth Edison has found that this modeling approach has led to a study that is very useful, from a utility's standpoint, as a long term tool for a variety of purposes. It was never our intent to develop specific models that would be directly applied to future studies of other plants. In fact, our review of such models, to date, leads us to the conclusion that they are very difficult to use without extensive PRA and computer resources which may not be available to utility users.

II.4 Comments on System Modeling and Simplification Procedures

(2) The fault trees also were treated in an abbreviated manner by drawing front line systems in block diagram form and deriving simplified trees to identify important cut sets. Such treatments require many subjective judgements by the analyst and are thus difficult to review and difficult to draw insights from (JH, p.5). There was concern that these simplifications may invalidate the study. (WCL, p.6)

Response

The block diagrams do not require any subjective judgments; whereas, very detailed fault trees do. We are surprised that the reviewers felt that the block diagrams were difficult to review. We believe that is the merit of developing block diagrams; i.e., ease of communication. The comment that these "simplifications" may invalidate the study is very strong and should be backed up. As it stands, all we can say is that it is not true.

II.5 Systems Interactions and Common Cause

Related to the above concerns is the problem of not-so obvious system interactions such as water in air lines, that were missed because the analysis was not carried out in enough detail. (WCL, p.3) Interactions between systems that are not connected but which can influence each other upon failure were not systematically treated, nor is there consideration of the potential for the adverse influence of cascading effects (PD, p.5). What assurance is there that the above concerns will not have a significant impact on the results of the study when they are addressed?

Instrument Air Systems were not modeled. Is instrument air used in safety system actuation? (WCL, p.5)

Response

Problems such as water in air lines generally lead to system instabilities that can initiate plant transients. They are imbedded in our initiating event data and no effort to segregate them for separate analysis was made. The physical plant was examined to identify significant potential for flooding and other cascade damage effects. None were identified that would be expected to occur with higher frequency than the seismic effects quantified in the study.

Instrument air is not used in safety system actuation at Zion.

II.6 The "Other" Category

What assurance is there that the "other" category indeed includes all events not analyzed and properly identifies their probability and consequences? (WCL, p.5)

Response

Obviously the category "other" provides no "assurance" that "all" events not analyzed are included. We have used the category "other" as a thinking device to cause us to pause, after doing our analysis, and reflect on what might have been omitted. As a result of this reflection, the analyst can use the category to make an allowance for what may have been omitted in his judgment, taking account in this judgment of what he knows about the system, of the degree of thoroughness of his analysis, the experience of the industry with this system, and of the evidence that scenarios or failure modes not otherwise in his analysis have/have not previously occurred. In this regard, such usage is superior to ignoring the issue and it does reflect an honest, subjective effort to close the "completeness gap".

III. PLANT ANALYSIS

III.1 General

(1) Do the event sequences (P. 1.5 - 183) include the out-of-service conditions permitted by Technical Specifications? (WCL, P.6)

Response

Yes. Unavailability due to maintenance and other technical specification limited activities are included in system fault tree and plant event tree quantifications.

III.1 General

(2) In Table II.4-12 the variances listed for the Zion plant specific events indicate the distributions are quite narrow whereas the PWR Population Generic variance indicate broad distributions for Initiating Event Categories 7, 8, 11a, and 13a. What effect do these broad distributions have on the final conclusions of the Zion PRA? (WCL, p.4)

Response

The reason for the narrow variances of the plant specific distributions is that the plant specific evidence is strong for these events (i.e., these events occur frequently--see Table II.4-11). In cases such as this, the evidence dominates; the posterior and the breadth of the prior are of no significance for the final results.

III.2 Specific Systems

(1) HPIS

The failure rate for HPIS seems very low; the Zion PSS used a median value of 1.2×10^{-2} (2 of 3 pumps) in Table II 4-15. The conditions and requirements assumed don't seem to explain such a large difference. Similarly, for small LOCAs, the corresponding HPIS differences are 5.8×10^{-9} (one of four pumps) for Zion vs $8. \times 10^{-3}$ (one of these pumps) for Surry. (PD, p.6)

Response

A review of the HPIS analyses in section 1.5 will reveal additional information on the derivation of the specific values. Commonwealth Edison has not made and does not intend to make detailed comparisons between the Zion PSS and other studies.

III.2 Specific Systems

(2) Engineered Safety Actuation System p. 1.5.3.2

The report states: "This analysis is carried out under the following assumptions: The system is in its normal operating mode prior to the initiating event." Since the ESF systems are in standby mode, how can it be justified that no operational errors have been made prior to actuation of the systems? (WCL, p.8)

Response

The statement quoted is intended to indicate that the system is assumed to be in a standby (ie nonactuated) mode prior to demand. The failure causes for the system and their quantification are presented in detail in section 1.5.2.2.3 (Volume 3)

III.2 Specific Systems

(3) Reactor Protection System Breakers

The report places great emphasis on using Bayes' Theorem to fold in plant specific data and considers this procedure as being conservative. But, is it conservative to take plant specific data which shows poorer performance than generic data, fold it in with generic data, and then use the result? As an example, the Zion data for the RPS Breakers shows

$$\frac{5\ \text{failures}}{612\ \text{tests}} = 8.2 \times 10^{-3}/\text{demand}$$

for a point estimate. On page 1.3-32 the unavailability of K-2 is that of scram breakers, wiring, and the CRDMs themselves:

Mean: 1.8×10^{-4} (failure per demand),
Variance: 5.2×10^{-8}.

Should selected plant specific Zion failure data have been used in the study without folding it in to generic data to obtain a more accurate measure of the risk at Zion? How many other Zion specific failure data values have been folded in to obtain lower failure rates than that representative of Zion? (WCL, p. 5&6)

Response

To respond to the first part of the question concerning the reactor trip breakers and event K-2, page 1.5-306 indicates that the value for a reactor trip breaker failing to open on demand is 9.79×10^{-3} (mean value). This value is higher than the value quoted above which is obtained by simply dividing the number of failures by the number of demands. This instance shows the true value of including plant specific evidence with a generic prior distribution (in this case, a mean value of 2.9×10^{-4}). To take this result and say that the unavailability of K-2 is much less (1.8×10^{-4}) and that K-2 includes scram breakers wiring and the CRDM themselves indicates a lack of understanding of the

RPS for Westinghouse PWRs. While it is true that K-2 includes the components described above, the breakers and logic are not single failure contributors. The Zion RPS logic (which includes the breaker and wiring) is arranged in a one-out-of-two configuration. That is, if either logic channel actuates its associated reactor trip breaker, the logic function has been completed. The details of the quantification of the RPS are presented in Section 1.5.2.2.2 of the ZPSS.

The question is answered in more general terms as follows. As the question says, the Zion report applies Bayes' theorem to combine generic and plant specific data. This is not considered a conservative procedure but an accurate procedure for reflecting our state of knowledge in light of both kinds of evidence. The problem that concerns the questions, that the plant specific performance is poorer than the generic, takes care of itself. That is what Bayes' theorem is all about. If the plant specific evidence is strong, it will overwhelm the generic in the Bayes' theorem calculations. If it is not strong but suggestive, Bayes' theorem will weight it exactly appropriately.

III.2 Specific Systems

(4) Trip Relay Failure

The report states: "Altnough the relays for a particular scram are arranged in parallel, diversity of scram signals requires coincident failure of two or more relays in series." The previous statement is not stated correctly. The relay contacts are arranged in parallel. Both contacts must open to open the scram string. If redundancy is claimed in the contact functions, then two sets of parallel contacts in series must fail to induce system failure. How does one conclude that functional redundancy exists? Does functional redundancy exist for all accident sequences? (WCL, p. 6)

Response

The trip relays deenergized by a particular scram signal are arranged in parallel. Both relays, rather than both relay contacts, must open to open the scram string. Failure of a single relay to open on demand could result in scram failure if this is the only relay set demanded to open. Diversity in scram signals for the different initiating events of interest results in at least two sets of parallel relays being demanded to open. Failure will result if at least one relay in each parallel set fails to open.

One concludes that functional diversity exists by constructing a matrix of initiating events versus scram signals actuated (expected). This was done very early in the ZPSS by the PRA analysts and the plant staff. Based on this exercise, it was determined that, for the initiating events analyzed in the ZPSS, at least two diverse scram signals will be developed in all cases where the reactor trip function is required.

III.2 Specific Systems

(5) Case 1: Failure of Power at Bus 147 p. 1.5-193

The report states: "If no safeguards actuation signal is present on either unit, which breaker first receives a closing signal is determined by the relative speeds of the bus undervoltage sensing relays." What this means is that if measurements were to be at Zion today, diesel generator 0 would be preferentially aligned with either Unit 1 or 2 each time there was loss of offsite power depending on the adjustment of the undervoltage relays. This preferential sequence would always occur unless the settings of the undervoltage relays were changed. Therefore there is not a 50% probability that diesel 0 will align with Unit 1. If the undervoltage relays are set such that diesel 0 automatically aligns with Unit 1. If the undervoltage relays are set such that diesel 0 automatically aligns with Unit 1, this alignment will always occur until the relay settings are changed and this p=1 for Unit 1 and p=0 for Unit 2 diesel 0 alignment. (WCL, p. 6)

Response

The reviewers are correct that if measurements were to be taken at Zion today, diesel generator 0 would be preferentially aligned to either bus 147 or bus 247, depending on which set of undervoltage relays operated first. However, the use of a 0.5 probability for the alignment of this diesel generator to Unit 1 correctly accounts for its operation in the context of the study. The 4,160V bus undervoltage relays are tested and calibrated annually during each refueling outage. The relays on bus 147 and bus 247 are adjusted during the respective unit refueling outages, approximately 6 months apart. Therefore, although one set of relays will certainly be aligned for the preferential operation of diesel generator 0 today, we cannot be sure that this will be the same 6 months from now. The Zion Probabilistic Safety Study evaluates the response of the Zion units to initiating events that may occur randomly in time over the life of the plant. In this context, for a given loss of offsite power event, our state of knowledge about the alignment of diesel generator 0 is evenly split. We know that it will certainly be preferentially aligned to one of the buses. However, we do not know which bus that will be and, without more evidence, must assign an equal probability to each.

III.2 Specific Systems

(6) Auxiliary Feedwater

Auxiliary feedwater system in the Zion PSS has an estimated failure probability that is almost an order of magnitude better than that estimated in WASH-1400 for the Surry plant. However, NUREG-0611 which included a comparison of all AFWSs of Westinghouse - designed operating plants, identified Zion as having an unavilability higher than Surry. This principally stemmed form the Zion plant having a single manual valve at the condensate storage tank which is shared by all three trains. The Zion study estimates that failure of this value can be detected, diagnosed and manually switched over with a probability of .993 (failure probability of $7x10^{-3}$). This appears to be a large amount of credit for this complex series of human actions.

In addition, the study apparently does not take into account the limited sustainability of the steam supply needed for the operation of the steam driven auxiliary feed pump. (PD p.5)

Response

It is our position that the analysis in the Zion PSS is appropriate and reasonable. Comparing this analysis with those done by others is valid only if one assumes that the basic boundary conditions and postulates are the same for all analyses. It should be recognized that the service water system at Zion is the Class I source of auxiliary feedwater for that plant. The switchover to this source of water is a well understood action by Zion operators.

The limited sustainability of the steam supply can be accomodated to a large extent by balancing flows and heat removal rates to keep the turbine driven pump operable until appropriate recovery actions can take place or until continued operation is no longer an issue.

III.3 Specific Sequences

(1) A problem with an ATWS sequence has been reported by Buslik. The human error probability of 0.004 was used that the operator would fail to open as necessary block valve in the 20 to 10 minutes time required following an ATWS. This, as he points out, appears extremely optimistic. Buslik also suggests that human error probability of 0.64 to 0.95 may be more appropriate in which case the ATWS core melt sequence becomes 5.8×10^{-5} and therefore an important sequence. This should be reviewed more closely. (JH, p. 7)

Response

Core Melt Due to ATWS

The reviewers correctly point out that the peak pressure following an ATWS occurs in about 2 minutes rather than 10 minutes as modeled in the study. However, we disagree strongly with their use of the Handbook of Human Reliability Analysis to requantify the necessary operator action. Furthermore, as stated in the Zion study, the ATWS analysis included overwhelming conservatisms. We have now revised that analysis and the changes are included as Attachment 1. Some of the more important changes are addressed in the following comments.

First, we address the reviewers' use of the Handbook. They use pages 17-20 and 17-24 which provide human error rates for the time immediately following a large LOCA and generic performance rules to be used in the absence of more specific information. In the large LOCA situation, human reliability is modeled as low (typical of very high stress) "not only because of the stress involved, but also because of a probable incredulity response. Among the operating personnel the probability of occurrence of a large LOCA is believed to be so low that, for some moments, a potential response would likely be to disbelieve panel indications. Under such conditions it is estimated that no action at all might be taken for at least 1 minute and that if any action is taken it would likely be inappropriate." (p. 17-9 of the Handbook)

This description does not apply to the ATWS case. First, all operators we have observed and interviewed respond to plant trip signals by immediately checking for turbine-generator trip (and decreasing load) and reactor trip (rod bottom lights). This is an almost automatic or "second nature" response with no hesitation (incredulity response) about completing those actions (trip the turbine-generator if it has not tripped, trip the reactor if it has not tripped, and carry out the required actions to shut down the reactor if reactor trip fails). Also, even though ATWS is hypothesized to have potentially severe effects, operators do not seem to be as "nervous" about it as about a large LOCA. The stress level would not immediately be especially high. Second, as clearly laid out in the recirculation system analysis, three reactor operators (ROs) are in the Zion control room at all times. One is assigned to each unit's panel and the third, the center desk man, immediately responds to the unit in trouble. So even in the first 2 minutes, two operators are available to support the ATWS. The shift engineer (SE) and shift technical advisor (STA), both SROs at Zion, may also be involved within the first 2 minutes. At least one of the two must be in the control room; say he is the STA. Then the SE is most likely there, but may be in an adjacent area or anywhere else in the plant, perhaps as far away as the switchyard or the forebay of the cribhouse. From discussions with plant operators, we believe the following discrete probability distribution is a reasonable model of the mean response time for the SE to arrive in the control room:

Time for SE to Reach Control Room (minutes)	Probability
0	0.80
0-1 (0.5)	0.10
1-5 (3)	0.08
5-20 (12.5)	0.02

Let us break the human response into two components: recognizing the failure to trip, and performing the required actions to protect the plant. In the recognition phase, it is only necessary to observe the presence of a trip condition and no actual reactor trip; i.e., no rod bottom lights. High readings on nuclear instruments reinforce this observation. For this phase, we see little or no dependence among the operators and model the situation as low dependence.

From the earlier discussion and the remarks on page 17-9 of the Handbook for "second nature" responses, it seems appropriate to consider the stress level optimum. The basic human error probability for this situation is 0.003. Then for low dependence, the center desk man's human error probability (HEP) is

$$\frac{1 + 19(.003)}{20} = 0.05$$

Since the STA (and SE if he is in the control room) will not respond as quickly and thus has less time to recognize the ATWS condition, we multiply his HEP by 2; i.e., 0.1. If the SE arrives in the control room within 1 minute, we again double his HEP to 0.2. Therefore, the total HEP for failing to discover the ATWS condition is

$$.8[0.003 \times 0.05 \times 0.1 \times 0.1] + .1[0.003 \times 0.05 \times 0.1 \times 0.2]$$

$$+ .1[0.003 \times 0.05 \times 0.1] = 3.0 \times 10^{-6}$$

After acknowledging very broad uncertainty in these results by assuming a lognormal distribution and assigning a range factor of 20, the mean HEP for recognizing the ATWS condition is 1.6×10^{-5}.

The first actions required of the operators, to manually trip the reactor and the turbine, are of a routine or automatic nature. To quote the handbook, "If personnel at a plant indeed have such frequent practice that the tasks in question could be regarded as 'second nature', the HEPs assigned to the moderately high level of stress will not apply, as the stress level will be closer to optimum." (page 17-9) We expect the manual trip to be attempted immediately, before the real significance of the ATWS condition is appreciated. Nevertheless, because the timing is short, we double the basic HEP for the RO; i.e., 0.006. Then for low dependence, the center desk man's HEP is

$$\frac{1 + 19(.006)}{20} = 0.056$$

As above, we double this to 0.111 for the STA and the SE if he is in the control room and double it again if he arrives within the first minute. Thus, the total HEP for failing to initiate a manual reactor trip is

$$0.8[0.006 \times 0.056 \times 0.111 \times 0.111] + 0.1[0.006 \times 0.056 \times 0.111$$

$$\times 0.222] + 0.1[0.006 \times 0.056 \times 0.111] = 7.83 \times 10^{-6}$$

Assigning a range factor of 20, the mean HEP is 4.11×10^{-5}.

If the reactor still has not tripped, it is apparent to the operators that a very unexpected condition exists. Despite their extensive training for this situation, we believe the operators will feel high stress as they begin to carry out the ATWS emergency procedure. The first step after attempting the manual trips of the reactor and turbine is to drive in the control rods. If this action begins within 1 minute, it should successfully terminate the ensuing pressure rise. Under these conditions, we assign an HEP of 0.25 to the RO. The center desk man may be closely working with the RO, so we consider this as a case of high dependence with an HEP of

$$\frac{1 + 0.25}{2} = 0.63$$

Because the STA and SE will be delayed in responding, probably until the RO's concern is voiced, we consider them moderately dependent

$$\frac{1 + 6(0.25)}{7} = 0.357$$

but double this value because of the time constraint to 0.71. Remember though that the required action is simple. In fact, all the STA really needs to do is say, "why aren't you driving rods?" and the event could be terminated. Finally, if the SE (or STA) is outside the control room, we give him no credit in helping the situation. Thus, the total HEP for failing to drive rods within 1 minute is

$$0.8[0.25 \times 0.63 \times 0.71 \times 0.71] + 0.2[0.25 \times 0.63 \times 0.71]$$

$$= 8.59 \times 10^{-2}$$

If we assign a range factor of 10; i.e., the upper bound is 0.859, then the mean HEP is 0.23 for failing to drive control rods given that automatic and manual trip have failed.

If the reactor has not been tripped and inward rod motion has not begun within 1 minute, and if the pressure is successfully controlled by the relief and safety valves, we next look for reactor shutdown by manually deenergizing power to the rods. Here, we assume high stress (0.25) for the RO in deciding to carry out the action, complete dependence for the center desk man (1.00), high dependence for the STA

$$\frac{1 + 0.25}{2} = 0.63$$

and moderate dependence for the SE

$$\frac{1 + 6(0.25)}{7} = 0.36$$

We neglect the SE if he has not returned within 5 minutes. Thus, the total HEP for deciding to disable power to the rods is:

$$0.98[0.25 \times 1 \times 0.63 \times 0.36] + 0.02[0.25 \times 1 \times 0.63]$$

$$= 5.87 \times 10^{-2}$$

Assigning a range factor of 10; i.e., the upper bound is 0.59 and the mean is 0.156.

Finally, the procedure specifies that the RO send the equipment operator ("A" man) to trip the breakers locally. Although he is not stressed, we double the basic HEP to 0.006 and the total HEP is 0.156 + 0.006 = 0.162.

Other important changes to the ATWS analysis include:

- o The fact that the Zion PORVs have been modified to prevent leakage. The PORV block valves are now kept open, so manual action is no longer required for pressure relief.

- o The fraction of time the PORV must open to control the ATWS pressure rise due to unfavorable moderator coefficient was erroneously given as 0.1 when it should have been 0.01.

- o A new branch has been added to account for the fact that most overpressure conditions will not disable the safety injection system. Most now branch to the small LOCA event tree.

Results of the revised analysis show the ATWS contributions to core melt and risk to be much smaller than calculated previously.

III.3 Specific Sequences

(2) The second area which has been pointed out by Kolb (8) is the credit given for spray injection given a core melt due to recirculation failure following a LOCA. This credit is given on the basis that 100,000 gallons of water will remain in the Refueling Water Storage Tank (RWST) when switch over to recirculation from injection occurs. This injection water provides another source of water to insure spray operation and reduce the probability of containment failure. The procedures we have indicate that an injection spray pump will be left on until the RWST is emptied and we have found no LOCA procedural steps for refilling the RWST. Thus, the RWST may be depleted of water when needed during core melt for containment protection. This could impact significantly the plant damage bin probabilities and perhaps the risk. Again, this has the character of providing credit for operator action beyond that which is typical of PRAs and therefore may deserve further review. (JH, p.7)

Response

Please refer to our response to enclosure 3, "Systems Analysis", question 1.

III.3 Specific Sequences

(3) A Third accident sequence, Station Blackout due to a LOP transient, is a dominant contributor to risk. The calculation is or has been pursued by Buslik, Fa[illegible]ling, and Kolb. The questions arising have to do with several factors, including the treatment of the increasing trend in the unavailability of the turbine-driven pump, the appropriateness of the LOP transient frequency prediction, and the onsite emergency power restoration assumptions. Depending on the way some of these are treated, the mean for this sequence could be approaching two orders of magnitude higher than the study predicts. This also deserves further investigation (JH, p. 7); as does the assumed quick recovery of offsite power as the grid margin is reduced. (PD, p. 5)

Response

First of all, we wish to acknowledge that the reviewers called our attention to errors in the loss of offsite power event tree model. We have now revised that analysis and the changes are included as Attachment 2.

The new model is more complete in terms of tracking sequences involving recovery from all electric power states. Along with correction of non-conservative numerical and logic errors, we corrected the overly conservative assumption that a seal LOCA leads to melt. With the recovery of electric power, bleed and feed cooling with high pressure injection can lead to success. It should be noted that the corrections lead to no changes in release category frequencies or consequences. However the following changes in plant damage state mean frequencies do occur:

Plant State	Old Frequency	Revised Frequency
SEFC	7.41-6	7.41-6
SEF	1.28-9	1.30-9
SEC	1.76-8	1.80-8
SE	6.53-10	4.50-9
SLFC	1.91-5	1.91-5
SLF	4.76-9	4.79-9
SLC	1.93-6	1.93-6
SL	1.25-8	1.26-8
TEFC	8.43-7	9.13-7
TEF	1.61-9	2.14-9
TEC	9.32-7	9.54-7
TE	2.27-7	2.29-7
AEFC	1.75-6	1.75-6
AEF	1.87-10	1.87-10
AEC	8.23-9	8.23-9
AE	1.05-11	1.05-11
ALFC	9.76-6	9.76-6
ALF	7.27-10	7.27-10
ALC	3.98-10	3.98-10
AL	2.52-13	2.52-13
V	1.05-7	1.05-7

Please refer to our response to NRC Estimation Methodology Question 3 for a discussion of the turbine-driven auxiliary feedwater pump unavailability. Our responses to NRC Systems Analysis Question 11 and Estimation Methodology Question 5 address the treatment of the loss of offsite power initiating event frequency and the distribution for time to recover offsite power.

III.3 Specific Sequence

4. There has also been concern over the low probability assessed for the V sequence at Zion (1.17×10^{-7} per year) which is considerably less than at Surry (4×10^{-6}) or Sequoyah (5×10^{-6}). A difference in system design may not explain the differences (PD, p. 4). The large effect of treating the WASH-1400 bounds as 5/95 percentiles instead of 20/80 percentiles was mentioned earlier.

Response

1. The low value of the "V" sequence when compared with that calculated in WASH-1400 is due to the following changes implemented by Zion in response to NRC directives and major design differences between the Zion units and the Surry plant analyzed in WASH-1400 and the Sequoyah plant analyzed in NUREG/CR-1659.

   a. Zion leak-checks both LPIS injection path check valves on all four injection paths after every plant cold shutdown. This leak-checking removes from consideration the probability of check valves failing to close after demand as both valves in each injection path are verified to be closed during the leak-check. This leaves only the rupture failure mode for the "V" sequence comparison.

   b. Each Zion low pressure injection path (refer to Figure 1.3.4.1-2, p. 1.3-91 of the ZPSS) contains an additional check valve inside containment which must fail (disc rupture) in order for high pressure fluid to enter the low pressure portion of the LPIS.

   c. The high pressure-low pressure boundary is located upstream of a normally open motor-operated valve outside the containment.

d. Each injection path contains a flow limiting orifice to limit RHR runout flow (flow into a broken RCS loop) to 1,500 to 2,000 gpm. This orifice should also limit the flow out of the containment in the event of a "V" type event.

e. Each of two LPI cold leg injection paths contain a high capacity (900 gpm) relief valve which discharges to the pressurizer relief tank inside containment.

These plant design and testing features were analyzed in Section 1.3.4.1.6.1 of the ZPSS and the results presented as the "V" sequence for the cold leg injection path. For a valid comparison, this result should be compared to the WASH-1400 "V" sequence analysis and the NUREG-CR-1659 analysis.

2. The dominant "V" sequence as presented in the ZPSS is a sequence not previously addressed or analyzed in WASH-1400, NUREG/CR-1659, or any PRA which existed prior to the publication of the ZPSS. This sequence is the failure (by combinations of rupture or disc remaining open) of the normal hot leg suction path to the RHR pumps. This path contains two normally closed motor-operated valves. The details of the analysis of this path are presented in ZPSS Section 1.3.4.1.6.3. The discussion of the 20/80 versus 5/95 percentiles is presented in the response to NRC question 6 under Estimation Methodology in Enclosure 1.

IV. CONTAINMENT ANALYSIS

IV.1 Assignment of Split Fractions in Containment Analysis

A barrier to review of the document is the lack of a clear correlation between accident phenomena and the split fractions assigned to the branch points in the containment event tree. The formal documentation and the method of incorporation of the analyses performed in Sections 3 and 4 to substantiate the assigned split fractions in Section 2.0 is lacking in detail. (ANL, C 3.0) Some of the problem lies in trying to lump phenomenological uncertainty with truely stochastic processes. (DP, p.1)

Response

An extensive effort was made, (including the provision of an example) to describe the correlation discussed in this comment. We expect some difficulty on the reviewers part since this entire approach is relatively new and innovative.

The second part of the comment (DP, P.1) regarding uncertainty indicates that the commentor has not read or does not understand material provided which describes the framework for treating uncertainty, probability and frequency. We suggest a review of section 0 might be helpful.

It was too time consuming and the report would have indeed been overly large for us to have tried to document and correlate all of the analysis and reasoning in the assignment of the containment event tree split fractions. Instead we wrote Section 2.5 of the Zion PSS to describe the methodology we used for an important and representative path in one of the containment event trees. We did go to the trouble of reporting and documenting in Section 2.6 all of the probabilities assigned for the thousands of paths of the trees along with all the other information one would need to reproduce the containment matrix for internal events.

In addition to the work provided in the report two papers have been published that further describe both the containment event tree and the methodology used to assign split fractions.[1,2] We see no need to expound further on the correlation or split fraction assigned until specific points are questioned by those taking the time to pinpoint relevant areas after reviewing and trying to understand the already ample supply of information provided in Sections 2, 3, 4, and 5.

References:

1. W. I. Toman and D. C. Richardson, "A Methodology to Evaluate and Quantify the Mitigative Aspects of Containment Structure", Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, September 20-24, 1981, Vol. I, pp. 275-283, 1982.

2. L. A. Wooten, N. J. Liparulo, D. F. Paddleford, "Quantification of Branch Point Probabilities for Class 9 Containment Event Trees", Proceeding of the International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, September 20-24, 1981, Vol. I, pp 284-292, 1982.

IV.2 Core Melt Coherency

(1) To what extent could the chemical heat liberated by the steam oxidation of Zirconium reduce the radial temperature gradient in the core during meltdown and cause the process to be more coherent than hypothesized in the Zion PSS (DP, p.4)

Response

Section 3.1.2 discusses the incoherency of the core heatup process and provides examples of the incoherency as Figures 3.1.2-1 through 3.1.2-3 for large LOCA, small LOCA, and transient core melts as calculated with the TMI-2 HEATUP code which includes a model for steam oxidation of clad and its associated heat input. In addition, analysis of a break in the pressurizer vapor space using Westinghouse thermal hydraulic and fuel rod computer codes indicates that the exothermic metal water reaction tends to increase the radial thermal gradient in the core thereby leading to strong incoherencies in the core melt progression. The analysis indicates that, at the time that fuel temperatures approach the melting temperature (4000°F) in the central regions of the core, the hottest axial region of the average fuel rod has not reached a temperature where metal water reaction is significant (1800°F). The analysis also indicates that at the point where approximately 10 percent of the core volume is above the melting temperature, the lower power level fuel assemblies which represent greater than 30 percent of the core volume have not begun to undergo significant metal water reaction of the fuel rod cladding.

Based on the results of this analysis, the exothermic metal water reaction of the fuel rod cladding tends to enhance the incoherent core melt progression.

IV.2 Core Melt Coherency

(2) To what extent does the self limiting nature of the oxidation process under high-hydrogen partial pressures, molten silver alloy from the control rods dissolving Zirconium cladding, ballooning of the fuel rods, and the eudothermic formation of eutectics of Zr, ZrO_2 and UO_2 cause the core melt process to be more incoherent than predicted in the Zion PSS (ANL, C 3.1)

Response

We believe many factors such as those listed contribute to a noncoherent core melt. This is discussed in 3.1.2. We see no compelling need to further address the contribution of each factor to the lack of coherency.

IV.3 In-Vessel Steam Explosions/Spike

(1) Probability

The ANL Group was in agreement with the basic phenomenology limiting in-vessel steam explosion as presented by the ZPSS authors; however, recent experimental data available after the ZPSS was prepared shows that, under the impact mode of contact, reproducible steam explosions' can occur at elevated system pressure. It is suggested that the authors of the ZPSS study evaluate this new data and its impact. In addition, additional justification for the assignment of $\approx 10^{-4}$ as the split fraction for the likelihood of breaching the pressure vessel should be requested. (ANL, C 3.2)

Response

Considering the voided condition within the primary system required to initiate core degradation, a highly confined impact mode of contact has no relevance to the reactor system configurations of interest. For those sequences leading to substantial degradation, the only molten fuel-water contact mode of significance is pouring of the molten debris into the lower plenum. Consequently, this was the contact mode specifically addressed in the ZPSS analyses.

IV.3 In-Vessel Steam Explosion/Spike

2. Steam Generator Tube Integrity

While the ZPSS method of calculation may be sufficiently accurate it is important to obtain an assessment from the analyst of the effect differences between the actual U-bend tubes and test conditions. The following comments seem appropriate: (ANL, C 3.2)

A. How valid are the results when the pressure loading source is dynamic

B. The dynamic pressure loading effects could be more severe on the U-bend tubes as compared to the tested straight tubes

C. How valid is the flow stress correlation given in Ref. (2) of the Appendix when it is applied to other materials not tested, such as Inconel 600

Response IV.3 (2)A

In Reference 1, Manning states that "the essential requirement (for cylinder design) was to determine the lowest value of the bursting pressure, but the difficulty in this is that experimenters tend to get erroneously high values because they raise the pressure too fast". It is inferred that static pressure capability is the more limiting consideration.

Referring to Figure 3.1.6-7 of Reference 2, the pressure rise is only 230 psi in about 25 seconds. This is a comparatively slow pressure rise and cannot be considered dynamic. It is also noted that the maximum pressure in the figure is 2560 psi.

Response IV.3 (2)B

No, the pressure loading effects cannot be more severe on the U-bend tubes than on the straight tubes. In the process of producing the U-bends,, cold working of the material occurs and this increases the yield strength of the tube. The burst pressure is proportional to the flow stress and since the flow stress is a function of the yield stress, the burst strength of the U-bend is generally higher than that of the straight tubes.

Response IV.3 (2)C

The flow stress formula expressed as a function of the sum of the yield strength and ultimate strength has been correlated for many ductile materials. It is noted that Reference 3 included four experiments using Type 316 stainless steel.

Extensive burst pressure testing of Inconel 600 steam generator tubing indicates that the flow stress is, in fact, proportional to the sum of the yield strength and ultimate strength of the material; the absolute value of flow stress, of course, will be different for different materials. Extrapolation of flow stress corresponding to different temperature for a given material is valid when based on the ratio of the sum of the yield strength and the ultimate strength at the respective temperature.

References:

1. Manning, W. R. D., "Burst Pressure as the Basis for Cylinder Design", J Pressure Vessel Tech., ASME Transactions, Volume 100, November 1978.

2. Zion Probabilistic Safety Study, Volume 6, page 3.1 - 108.

3. "Investigation of the Initiation and Extent of Ductile Pipe Rupture", BMI - 1908 Report, June 1971.

IV.4 In-Vessel Cooling

The event tree defines the nodal question to be...Do the conditions exist for in-vessel cooling of the core debris As noted in Section 2.5.1.5 several items are required for this to be true, they are: (1) human intervention is required to provide a source of water, (2) a heat sink, such as the secondary system, must exist as well as a water return path, (3) the debris can be quenched and particle sizes are sufficiently large tc allow coolability.

A. Heat sink and return path - The reflux mechanism cited, is certainly an effective heat removal method; however, if non-condensibles (such as air or hydrogen) accumulate in the steam generator region, it will pose additional thermal resistance and reduce heat transfer. We would suggest an evaluation by ZPSS of the effect of non-condensible gases upon the reflux mechanism.

Response

The Zion PSS does consider these effects. In short, the study concludes that non-condensibles would occupy the upper portion of the tube bundle leaving the lower portion, which would be cooled by feedwater, free to participate in the condensation and reflux process. Also, the study acknowledges the obvious, eg if the tube bundle is filled with non-condensibles, no reflux process will occur. Extensive evaluations of the process beyond that in the study are not warranted for the purposes of the study and will not be performed. The in-vessel cooling likelihood does not affect the risk. (The retention of this node serves to alert utility personnel to the possiblity of corrective action.)

IV.4 In-Vessel Cooling

B. Debris Coolability - The pool boiling critical heat flux has been chosen by the ZPSS authors for evaluation of the in-vessel coolability limit. The use of the critical heat flux limit is valid only for particles greater than 4-6 mm in diameter at atmospheric pressure. Based upon the possibility of small local explosions producing much finer particles and the small particles dominating the coolability, the ZPSS authors should be asked to reassess the choice of model.

Response

It is our position that the model is entirely correct. The chronological location of the in-vessel cooling node is such that we are considering in vessel cooling only prior to the release of molten material into the lower plenum of the vessel. The potential for any significant amount of material being involved in steam explosions prior to that point (and certainly the potential for fragmention dominating coolability at that point) is negligible.

The pool boiling critical heat flux was chosen to represent the quenching of debris with an unknown configuration. It was fully realized and discussed in the report that particle mixes smaller than a few millimeters can have dryout heat fluxes lower than the limit calculated by the critical heat flux. However, with the reflooding mechanisms considered for a core substantially overheated and degraded, no substantial particulation mechanism was discovered which would provide for such fine particulation of the entire core material. As a result, the global mechanism describing the hydrodynamic stability limit for quenching of the debris was used and this mechanism was compared to pertinent experimental results including the quenching rates observed in the "B" loop pump start in the TMI-2 accident. As illustrated in the report, this formulation provided an accurate first order assessment of that quenching rate.

No significant credit was taken for in vessel coolability. However debris coolability in the water filled cavity is expected and was taken credit for based on the tests described in Appendix 3.4.5 "Debris Bed Experiments," and the January 1982 issue of Nuclear Science and Engineering. Coolable debris beds are expected for particles smaller than the 4-6 minimum range in the vessel cavity based on the quantity and power generation expected in the debris deposited there. The above tests and models such as Lipinski's can support this for a reasonable range of particle size.

IV.5 Vessel Failure

(1) Considering the importance of the reactor vessel failure mode to the subsequent analyses for short term containment pressurization and long-term coolability, it would seem that a more exhaustive and convincing analysis of bottom head failure would be in order. The principal concern with the present analysis is with the realism of the assumption that failure of the instrument tube weld will lead directly to ejection of the guide tube from the penetration. Mechanisms which could prevent the tube from being ejected include: a) interference due to differential thermal expansion, b) interface pressure between the tube and vessel wall due to system pressure, c) pressure welding of the tube to the vessel was, and d) resistance to tube motion from external supports.

Considering the importance of understanding the expected mode of reactor vessel failure it would be appropriate to review additional information which needs to be provided for the containment. (ANL, C 3.2)

Response

Two seperate and independent vessel failure analyses are contained in the Zion PSS. The first of these is a "first principles" analysis while the second is a more sophisticated analysis descirbed in Appendix 3.4.6. The conclusions of these analyses are the same. For the dispersive events, with significant residual primary system pressure, the forces involved in expulsion, once the weldment is melted, are very great. None of the mechanisms noted in the question could be expected to have any effect on the process.

The reviewer questions the assumption that failure of the partial penetration weld will lead directly to ejection of the instrumentation tube, and suggests the following mechanisms to challenge the assumption:

Mechanism	Description
(a)	Interference due to differential thermal expansion
(b)	Interface pressure between the tube and vessel wall due to system pressure
(c)	Pressure welding of the tube to the vessel wall
(d)	Resistance to tube motion from external supports

In response to suggested mechanisms (b) and (c), it should be noted that the vessel has more compliance to pressure than the tube. In other words, the tube is stiffer than the vessel. This being the case, the pressure will tend to pull the vessel penetration radially outward away from the tube. For this reason, mechanisms (b) and (c) will not take place, and therefore cannot impede tube ejection.

In response to suggested mechanism (a), it should be noted that there is an initial diametral clearance between the tube and the vessel penetration. This clearance averages 0.0025" according to the manufacturing tolerance on the vessel drawings. In addition to the manufacturing clearance, there will be some additional clearance due to pressure effects already discussed in the preceding paragraph. Acting to take up this available clearance is the differential thermal expansion between the tube and the penetration which is $(7.85 \times 10^{-6} - 7.12 \times 10^{-6})$ $(1.5)\ \Delta T$ or $1.01 \times 10^{-6}\ \Delta T$ inches, where ΔT represents the difference between the ejection temperature and the temperature after fabrication of the vessel. The weld material is considered to lose all of its shear strength when T = 2000°F, and the differential expansion would then be 0.002". There is sufficient initial clearance to prevent mechanism (a). In response to suggested mechanism (d), the instrumentation tubes are only supported by the thimble guide tubes, which are slender and which also have a 90° bend. These tubes can give very little resistance to tube ejection. The tubes are supported by angle iron type of racks but they offer no resistance in the vertical direction from the interface between the instrument tunnel and the area under the reactor.

The Inconel guide tubes for the in-core instrument penetrations has a lower thermal conductivity than the carbon steel reactor pressure vessel. As debris accumulated on the vessel wall, energy would be conducted into the carbon steel wall as well as down the Inconel penetration stub. Since the carbon steel has a thermal conductivity about 3 times that of Inconel, the energy conducted down the stub would be effectively lost into the BPV wall at the welded junction on the inside wall of the vessel. This axial transmission down the Inconel rod heats the weld faster than the one-dimensional conduction attributed to the vessel wall and causes this local region of the vessel to expand as a result of the temperature increase. Since the energy is more easily conducted into the carbon steel, the thermal transport path will be short circuited at the weld and the net energy deposition will be into the reactor pressure vessel wall which will see local temperature increases faster than the Inconel penetration and will tend to grow more than the Inconel tube.

IV.5. Vessel Failure

(2) It was also thought likely that more than one tube would fail leading to a net discharge orifice that is larger than the one assumed in the Zion PSS. This would reduce the main driving force for the dispersal of core debris in the pressurized (small LOCA) scenarios and would cause them to look more like the unpressurized (large LOCA) cases. (DP, pp. 4-5)

Response

Sensitivity analyses were carried out for more than a single penetration failure and the analyses showed that the resultant dispersion from the reactor cavity was very insensitive to the assessment of how many local failures were assumed. It should be noted however that the time available for simultaneous failures is extremely short since the discharge of molten debris requires a time interval from a few seconds to about 10 sec. Consequently, the failure of additional ports must also occur within this time frame to have any effect on the overall process. Also, the radius ablated by the discharge of material encompasses at least one other penetration, and as a result the calculation used in the ZPSS effectively includes other simultaneous failures since the most likely site for an additional failure would be an adjacent penetration to the initial failure location. The summation of all this information was used in employing the analysis of a single failure location for the ZPSS as being more than sufficient to analyze the accident progression.

IV.6 Core Debris Dispersion

(1) Given vessel failure for a small break LOCA by failure of an instrument tube penetration, the authors predict a large fraction of the core debris ejected from the vessel will be dispersed out of the reactor cavity onto the containment floor. Although the ACRS consultants agreed in general with the driving mechanisms, several significant limitations were identified: a) the effects of crust formation on liquid surfaces, b) the interaction of molten core with ex-vessel concrete resulting in substantial gas release (ANL, C 3.4); (DP, pp. 7-8); instrumentation tubes running the length of the instrumentation tunnel might create a blockage greater then suggested by their cross-sectional areas. (DP, p. 6)

Response

The ACRS consultants raise several questions concerning the overall progression of the melt after leaving the vessel. Such considerations are exactly the reason why a containment event tree is used in analyzing the progression of events for the ex-vessel states. In addition to the containment event tree, several first-principle analyses were carried out to deduce the likely progression of events for those sequences with an elevated system pressure at the time of vessel failure. To carry out these basic analyses, considerations of crust formation and liberation were made, but given the rapid time frame of material ejection from the reactor pressure vessel and from the cavity, very thin debris crusts were calculated, i.e. sufficiently thin as to have no significant mechanical strength. As a result, crust formation during the dispersive phase of the accident progression were neglected.

Gas released from the concrete attack merely increases the net gas flow through the reactor cavity and instrument tunnel region up into the containment. As a result, such gas release would also tend to disperse molten debris and would only augment the natural process. However, the superficial velocities attendant to such concrete attack are orders of magnitude below those being released from the reactor pressure vessel during the dispersion process, as a result, this was also judged to be a negligible contributor to the overall event progression.

Guide tubes for the in-core instrument probes run through the instrument tunnel and up onto the containment floor. These were considered by the ACRS consultants to have some potential for impeding dispersion of the molten debris because they occupy part of the flow area within the instrument tunnel. Evaluating the cross-sectional area occupied by these tubes and their size effects supports, results in an area occupation of < 1 percent of the entire cross-sectional area. Consequently, these would have insignificant influence as frictional walls, but conceptually dynamic processes could occur where these could be torn or distorted and forced into a porous plug of the instrument tunnel. This is another reason why a containment event tree is created to represent those processes whereby dispersion forces would be insufficient to remve the material from the reactor cavity. As a result, the material would remain in the cavity, and undergo quenching or concrete attack depending upon the details of the accident sequence, i.e. is water delivered to the reator cavity on a continual basis.

IV.6 Core Debris Dispersion

(2) The part of the core that is not involved in the coherent portion of the meltdown (50 percent as estimated in the Zion PSS) will eventually melt and leave the vessel without the dynamic forces associated with the initial vessel failure. The Zion report seems to neglect this remainder of the core which will not be dispersed (DP, p. 4)

Response IV.6 (2)

In fact, the core debris remaining in the reactor vessel after the initial failure was considered in assessing the phenomenology important in the determination of the containment response, and its effect was also considered in quantification of the containment event tree and the resultant contributions to the source terms.

Areas we considered most important related to the debris left in the reactor vessl after melt through were: coolability of the debris in the reactor cavity, steam and hydrogen generation and the effect of the debris on the source term. Coolability is deemed to be likely for the worst case of a large break event with failure of injection and the after failure core debris coolability is encompassed by the arguments of Section 3.2.13, 3.2.14, and 3.2.15. Section 3.2.12 specifically addresses the debris disposition for the two types of cases, total pour and dispersion plus later pour. As for steam generation and hydrogen generation these issues are also discussed in section 3.2.12 - 3.2.15. Furthermore, section 4.2 discusses the assumptions made in the analyses regarding additional hydrogen generation and steaming rates. Where extra masses of hydrogen were assumed to be produced the extra percentage was determined from the entire core mass not just the dispersed mass. All of the analyses reported in section 4.3 take account of the entire mass of core debris not just the dispersed fraction. Concerning possible changes to the source terms, we note that the entire core with all of its fission product inventory was used in determining the fission product release (Section 5.4).

IV.7 Presence of Cavity Water (Node J)

Node J asks the question whether water will be in the ex-vessel cavity. The authors of the Zion PSS have assigned a probability of 1-E where E = 10^{-4} that water will be in the ex-vessel cavity. They state that this is based upon a detailed evaluation of the plant design. The reviewers were unable to find reference to the evaluation in the Zion report. Additional detail is warranted. (ANL, C 3.5)

Response

Refer to section 3.2.13 of the study for an assessment pertinent to this question.

IV.8 Basemat Penetration (Node K)

(1) The analysis of jet attack of the concrete assumes the form of a quasi steady state calculation of a molten jet attacking and ablating the surface. Several conservative assumptions in the calculations were used: The non-conservative assumption is that spallation the surface of the concrete does not occur. The calculated heat flux into the concrete of 20,000 kw/m^2 (page 3.2.8) is an extremely high value. The limited duration of the jet mode of attack does make penetration unlikely; however, the use of $\epsilon = 10^{-4}$ does require additional justification particularly with respect to the affects of spallation. (ANL, C 3.6).

Response

Concrete spallation was not included in the analysis since Sandia experiments, carried out with prototypic materials, showed that such mechanical breakup mechanisms were not significant in the overall thermal penetration of the concrete material.

IV.8 Basemat Penetration (Node K)

(2) The cavity sump is located at the far end of the instrumentation tunnel. Melt dispersed during discharge from the pressure vessel will collect in this sump. The concrete below the sump is especially thin (2-4 feet). Penetration of the concrete basemat at this location should be considered for all accidents in which molten core debris escapes the pressure vessel. The effect of the sump on the novel hydrodynamics of melt during high pressure discharge is uncertain. It is known that only small discontinuities in surface can have drastic influences on the flow of liquids over the surfaces. Other features of melt behavior, such as melt/concrete interactions, will provide even stronger effects on these hydrodynamics. (DP, pp. 5-6)

Response

The reactor cavity sump was considered in the evaluation. However, the key feature of assessment for concrete attack was the ability of water to be continually supplied to the reactor cavity instrument tunnel region. If this water was supplied, the debris was assessed to be coolable since the particle sizes generated by the concrete attack would be far larger than those precluding debris coolability. If the debris was not coolable, the containment failure resulted from overpressurization as a result of steam generation and long term concrete attack. Consequently, the crucial assessment is not the mechanics of material dispersion or the collection within the reactor cavity sump, but rather the evaluation of a continual supply of water into the reactor cavity and instrument tunnel regions, which is accident specific.

IV.9 Coolability of Ex Vessel Debris Bed (Node Q)

(1) The question addressed in the Node Q of the containment event tree is: ...Does the debris positioned on the reactor cavity and containment floor form a configuration which is initially coolable thereby preventing significant concrete attack A significant uncertainity with respect to the assessment of ex-vessel debris coolability is: What effect does the concrete and gas release from the concrete have upon the quenching process of the core melt in the ex-vessel cavity This effect can not be ignored as the authors indicate that up to 30 minutes may be required to quench the core. (ANL, C 3.7)

Response

Typical superficial steam velocities attendant to the quench process are 1 m/sec. During the early phases of concrete attack, the superficial gas velocities may be as high as 10 cm/sec and quickly decay from this value. As a result, the influence of gas liberated from the concrete influences calculations by only 10%, which is not significant for PRA assessment.

IV.9 Coolability of Ex Vessel Debris Bed (Node Q)

(2) There is a lack of sensitivity to the uncertainties associated with fragments debris beds. Real fragmentation processes will yield particles that are not monodisperse and probably not spherical. Non-spherical particles with a range of sizes will routinely pack more densely than assumed in ZPRA. Packing density increases, and consequently porosity and coolability decrease with increases in the mean particle size and increase in the breadth or size distribution. And stratification should be considered. (DP, p. 7)

Response

Assessments of the fragmentation sizes resulting from either film boiling fragmentation or those produced by gases liberated from the concrete during thermal attack result in particle sizes which were orders of magnitude greater than that required for establishing a coolable debris bed. Consequently, the effects due to non-uniform particle sizes, non-spherical particles, and stratification would be second order effects compared to the evaluation of water being supplied on a continuous basis.

(3) The probability of achieving a coolable bed in the ex-vessel cavity was assigned a probability of $1-\epsilon$ where $\epsilon=10^{-4}$ for all events where water is available. The phenomena associated with the quench of the core material which need additional evaluation are: 1) crust formation between the debris and the water, 2) gas release from the concrete hindering water reentry into the cavity, 3) reduction of the gas released from the concrete by metal constituents in the melt and additional energy generation, and 4) late entry of the remainder of the core materials into the ex-vessel cavity which seems to be omitted from the study (DP, pp. 4-6; ANL, C 3.7). The phenomena may effect the ultimate conclusion and need to be addressed in more detail to justify the ZPSS conclusion of $\epsilon=10^{-4}$ as a split fraction.

Response

In evaluating the coolability of core debris in the reactor cavity and instrument tunnel, effects of crust formation were considered, but the release of gas as a result of thermal attack of the concrete provided sufficient forces to break up any overlying crust. This was considered as an important mechanism for determining the debris size and distribution. Gas released from the concrete was considered, but the superficial velocity resulting from the thermal attack was negligible compared to the steam velocities produced by the quenching. As a result, the major influence of the gas was to produce a large size particulate bed. Reduction of the gases released by the concrete were included in the containment analysis through the incorporation of the INTER code for assessing the concrete attack and this included all of the materials which would be eventually collected within the reactor cavity including the later entry of the remaining materials not initially released from the reactor pressure vessel.

IV.9 Coolability of Ex Vessel Debris Bed (Node Q)

(4) The Zion PSS analysis seems based upon the notion that cooling of the debris is limited solely by the ability to supply coolant. During a core melt accident cooling of ex-vessel debris is limited not by the supply of coolant but by the ability to get heat out of the material. If the barrier to heat removal posed by the low thermal diffusivity of the largely oxidic material were properly recognized in the ZPRA, the ex-vessel hydrogen production would by greatly increased. The material stays hotter, longer, regardless of how large an excess of coolant is available. In fact, the large supply of coolant assures there is an excess of reactant for the hydrogen production process. (DP, p. 6)

Response

The barrier to heat removal posed by the low thermal diffusivity of oxidic material was recognized in the Zion Probabilistic Risk Assessment as was the potential for particulation due to gaseous products released from the concrete during thermal attack. This particulation exposes a large amount of area and allows the debris to be quenched on a comparatively rapid basis. Once quenching has occurred, even though the material may still be at high temperatures, the amount of additional oxidation which can be generated is very small. This was properly acknowledged in the ZPSS and was incorporated into the overall containment evaluation. In addition, substantial variations on the amount of hydrogen released were included in the uncertainty and sensitivity assessements in section IV. As shown, these had little influence on the overall assessment of containment integrity.

IV.10 Hydrogen Production

(1) The core exit temperature of 1093°C is quoted as the peak temperature prior to vessel failure which seems somewhat low and should be investigated. The importance of this calculation relates directly to the ultimate pressure in the containment because it is a contributor to the quantity of hydrogen generated as well as the pressure vessel structural integrity. (ANL, C 3.8)

Response

The gas temperatures quoted in Section 3.1.4 correspond to the mixed average temperature in a gas volume representing the outlet plenum and upper head region. This temperature was low because even though the temperature of the gas exiting the core is relatively high this condition persists only for a short time and had a very low flow rate. The large mass of gas in the outlet plenum for the high pressure transient case results in a substantially lower mixed temperature than for the lower pressure large and small break cases. The MARCH output for the structural temperatures in the outlet plenum was checked and found to be below the 2000°F temperature at which the steel reaction becomes significant.

IV.10 Hydrogen Production

(2) The contribution of the non-condensible gas generation to containment pressurization from core/concrete interactions during the approximately 30 minutes required to quench the debris in the ex-vessel cavity seems to have been ignored and should be included. (ANL, C 3.8)

Response

This is considered negligible since the pressure contribution from decomposition of a 1/2 foot depth of limestone concrete over a 500 ft^2 cavity via non-condensible gas addition is less than 1 psi.

Furthermore, it is not true that this was not considered since the pressure rise from noncondensibles was accounted for when concrete attack was allowed to occur.

IV.10 Hydrogen Production

(3) The production of hydrogen and other combustible and non-combustible gases, due to the portion of the core that leaves the vessel after the initial failure seems to be neglected. Would this be similar to the behavior of all the core in the large LOCA scenarios (DP, p. 4)

Response

As explained on page 3.1-29 the integrated containment analysis for the transient and small LOCA core melts have included a substantial allowance for reaction of clad after core plate collapse. This allowance was 50 percent of the previously unreacted clad and covers interaction with water in the vessel head or cavity and containment floor as well as that from limited concrete attack in the process of forming a coolable debris bed. This factor was applied to the whole core regardless of whether it was the core fraction involved in the initial coherent dispersive event or the fraction which had delayed entry to the cavity.

In addition other cases were analyzed with 100 percent clad reaction and/or additional hydrogen and carbon monoxide generation from melt-concrete reaction in the case of a non-coolable debris bed as part of the sensitivity study and to address all paths in the containment event tree.

(4) The effects on the concrete walls of the instrumentation tunnel due to the thermal radiation and steaming of the core debris that is dispersed there seems to be neglected. (DP, p. 7)

Response

Any concrete attack over the short duration of the dispersive blowdown period would not be significant as can be seen by applying erosion rates such as 100 cm/hr to a period of a few seconds. Thermal radiation from debris to walls over the debris in the instrument cavity tunnel were not modelled. In almost all cases water over the debris would quench the debris eliminating the thermal radiation source. For cases where debris was assumed to be non-coolable in spite of water cover, the water layer would absorb the upward radiative flux and only downward concrete attack would be significant and this was modelled. Finally in cases where the cavity would eventually dry out because of prolonged boiloff with no replenishment from fan coolers or spray - the containment was automatically assumed to fail at 12 hours due to steaming. Hence, some additional concrete attack from thermal radiation to walls above the debris would not be significant.

IV.10 Hydrogen Production

(5) The analysis of hydrogen generation neglects the solubility of ZrO_2 in liquid Zr and in some parts of the analysis, neglects the formation of eutectic between oxygen saturated zirconium and fuel. (DP, p. 3)

Response

The hydrogen generation approach has been to make use of generally available analytical models and the experimental data that does exist for clad reaction with steam and discharge of molten material into water. We consider the allowances made for hydrogen in the transient analysis to be consistent with releases observed from various experiments and tests as well as the TMI event. The data that does exist of course accounts for solubility and eutectic factors to the extent they were present.

The effects of eutectic formation were partially incorporated by assuming the melt had a lower temperature than that characteristic of molten uranium dioxide. This reduction in temperature was not found to be significant, and while it was recognized that eutectic formation between the cladding and fuel could also reduce the hydrogen produced during the accident sequence, the kinetics of such processes are sufficiently unknown such that the evaluation team felt that no credit could be taken for this mechanism in reducing the hydrogen production. As discussed in section IV of the ZPSS considerable uncertainties were applied to the hydrogen production rate and in the specific assessments carried out in section III, conservative evaluations were provided for hydrogen production in the lower plenum and in the ex-vessel configuration. Part of this conservative approach was the neglect of mixture properties for the melt (pure zirconium was assumed in assessing the hydrogen production outside of the core) and neglect of eutectic formation.

IV.11 Hydrogen Burn Analysis

The reviewers recommend consideration of the following questions by the ZPSS authors.

A. How are steam removal rates and/or steam gradients taken into consideration in the analysis (ANL, C 3.8)

Response

In ZPSS, the containment response analyses were performed using the COCCLASS 9 containment code. Steam removal by gross steam condensation as a result of heat transfer to containment walls, heat transfer to structures, and fan coolers or sprays and subsequent fall-out of the condensate to the sump is modeled in the code. The code conservatively assumes that the condensed matter falls into the containment sump water at the same rate as it forms. Diffusion of condensing steam through steam condensation gradients in a steam-air mixture near the walls is not modelled in the code.

Where large steam removal mechanisms are available from active sources such as fans or sprays the atmosphere is quickly well mixed such that a one node containment model is adequate. In cases where mixing is via natural mechanisms such as jetting, buoyancy, and natural convection the mixing is slower, but so is the steam condensation, and in such cases the atmosphere is seen to contain excessive steam such that hydrogen burn is not predicted.

IV.11 Hydrogen Burn Analysis

B. What justification is there that locally-high concentrations of hydrogen or CO could not build up in the time scale of their release or formation in the containment building during a given scenario? (ANL, C 3.8)

Response

Significant CO would result only from extensive core/concrete interactions. For those sequences in the containment event tree where such interaction is expected (i.e. dry cavity), accumulations of CO and steam in the cavity may occur. However, the thermal effects would dissipate the concentration to the upper areas of the containment where rapid mixing due to process dynamics is assured. Within the cavity, the mixture is inerted by steam.

Hydrogen, in significant quantities can be released from the primary system thru a pipe break, thru vessel failure or thru safety/relief valve operation. For a large LOCA, the fan coolers or sprays will insure mixing between and within the steam generator compartment and the upper plenum of the containment. For small LOCA's some hydrogen may be released into the steam generator compartment but most would be released into the cavity upon vessel failure. The dispersive nature of the failure would assure rapid mixing. For transient events, some hydrogen would be released thru the safety/relief valves to the pressurizer relief tank. As the rupture disk fails on this tank, the hydrogen would be released into a large annular area with good flow communication to the upper containment area. Within the annular area, steam inerting would be expected. Most of the hydrogen would be released upon vessel failure as with the small break.

Therefore, as a result of fan coolers, sprays, the open nature of the containment, and the dynamics of the processes involved, significant local pocketing of hydrogen or CO in a non-inerted environment is not judged to be physically realistic.

C. The flame temperature criteria becomes invalid when a homogenous H_2 concentration cannot be assumed and its use precludes the prediction of hydrogen detonation. (DP, p. 2)

Response

The flame temperature criteria determines whether, given a specific composition of the hydrogen mixture, it is possible to have global combustion, resulting in a significant pressure rise. In other words, it determines the limit of global combustion, at which the flame propagates in all directions. It does not preclude the prediction of hydrogen detonations, which are determined by the detonability limit, geometric requirements, and other conditions.

The flame temperature criterion provides a conservative estimate for the deflagration limit of hydrogen, air and steam mixtures. As explained in our response to the previous question (IV 11-B), atmosphere in the Zion containment is expected to be well mixed and there would not be significant concentration gradients. Hence, usage of the flame temperature criterion in the Zion PSS is justified. Hydrogen detonation was not precluded on the basis of the flame temperature criterion. In order for local detonation to occur in a gas pocket, the hydrogen concentration in the pocket must be at least 18 volume percent, and the geometry of the gas pocket should be such that it is confined by walls and contains many obstacles that can generate strong turbulence. Neither the concentration requirement nor the geometrical requirements would be satisfied in the Zion containment, and hence there can be no hydrogen detonations.

D. What effects would structures in the containment building have on the propagation of a postulated combustion wave, or conversely could any of the postulated combustion pressures damage structures or auxiliary safeguard systems (ANL, C 3.8)

Response

Interaction of the containment atmosphere flow with structures would generate turbulence which would cause a moderate increase in the flame propagation velocity. This turbulence is not expected to be any stronger than that generated by the spray droplets or by the jet flow from the break and fan cooler exhaust ports. Zion containment is essentially a large open volume and its geometry is not conducive to producing strong turbulence. Turbulence produced by the flow - structure interaction would not be strong enough to accelerate the combustion wave to velocities large enough to produce significant pressure waves. A study reported in Reference 1 indicates that for a spherical flame in stoichiometric methane - air mixture to produce an overpressure of 0.3 atmosphere, it must be accelerated to about 50 times its normal burning velocity. A similar increase in flame velocity will also be required for hydrogen-air mixtures. For concentrations below the stoichiometric limit, the flame velocity required to produce a significant pressure wave will increase rapidly with decreasing fuel concentration. The presence of steam in hydrogen mixture will further increase the required flame acceleration. Because of the open nature of the Zion containment, the flow-structure induced turbulence is not expected to increase the flame velocity by the two orders of magnitude necessary to produce blast waves. Hence, during hydrogen burn the containment pressure would increase gradually without producing significant pressure waves.

Ref (1) Strehlow, R. A., Luckritz, R. T., Adamczyk, A. A., and Shimpi, S. A., "The Blast Generated by Spherical Flames," Combustion and Flame, Vol. 35, 1979, pp,.297-310.

Structures in the Zion containment and engineered safeguard systems such as fan coolers and sprays are designed to withstand the postulated pressure transient for the design basis accident which results in a pressure rise from 0 to 47 psig in about 10 seconds. This is a more severe pressure transient than that could be caused by a hydrogen burn, during which the containment pressure could rise by up to about 60 psi in 20 to 60 seconds. Because of the inherent conservatism in the design analysis, the structures within the containment, and fan cooler and spray systems should be capable of withstanding a significantly more severe pressure transient than the design basis transient. As an example of this type of conservatism, note that the fan coolers are provided with relief devices in the housing and with backdraft dampers on the fan outlet which are designed to cope with design basis accident. Hence, the structures within the containment and auxiliary safeguard systems are expected to survive the pressure transient during a realistic hydrogen burn scenario.

IV.11 Hydrogen Burn Analysis

E. What was the basis and use of the hydrogen burn probabilities in Table 2.5.1.1 (See also page II.5-19). The range of temperature is 100°F in Table 2.5.1.1. Small changes in the hydrogen source term, available oxygen or steam may alter the calculated adiabatic flame temperature far more than 100°F. (ANL, C 3.8)

Response

The basis for the probabilities assigned in Table 2.5.1.1 were the results of the flame temperature and ignitor tests Westinghouse had conducted at Fenwal labs as cited in part in WCAP 5909. We recognized the fact that changes in the hydrogen source, steam source, oxygen concentration, containment temperature, and containment pressure would effect the flame temperature as well as the criterion calculated. This is why various runs were made where these parameters were varied to calculate differing flame temperatures. These runs were then used for the appropriate branches in the containment event tree by relying on knowledge as to the applicability of the runs. Dependent upon the assumptions made for the runs the engineer could then through table 2.5.1-1 determine a split fraction for a particular node of the tree.

IV.12 Containment Mass and Energy Loadings

(1) MARCH Code limitations and non-conservatisms need further consideration. (PD, p. 4)

Response

The Zion PSS contains substantial independent evaluation of core melt related phenomena in Section 3 and its Appendices and Section 4 and its Appendices. These relate to a broad variety of effects such as coherency, steam generation rates, hydrogen burn modelling, debris coolability, and containment pressure and thermal analysis. It is clear that MARCH modelling was not blindly followed or uncritically accepted. Additional comments regarding MARCH by others are identified in References 1 and 2.

(1) NUREG/CR-2285 Interim Technical Assessment of MARCH Code, November 1981.

(2) BNL Presentation to ACRS, May 21-22, 1981.

IV.12 Containment Mass and Energy Loadings

(2) The relation between the bounding cases and the most probable cases for the six sequence classes in the containment analysis in Section 4.3 needs to be clearly indicated. Some of the bounding cases would fail the containment and should be given a non-negligible weight in the probabilistic treatment of containment failure. (GS p.4)

Response

A wide variety of parametric and model variations were included in the sensitivity studies. In some cases, series of events were arbitrarily coupled together in a manner that would not be physically realizeable even under conservative assumptions. This was done in order to test the sensitivity of the analyses and in order to establish a high degree of confidence in the ability of the containment relative to the events being studied. Those cases which involved wholly non-mechanistic and non-realizeable postulates were properly given negligible weighting in the probabilistic treatment. We do not plan to change this.

IV.13 Containment Failure and Integrity

(1) In general, the approach used to calculate containment structural capability appears sound and well documented. Some of the questions which appear significant in the context of the high estimated failure pressure of the containment building are:

A. How and to what extent has failure to isolate the containment been considered? (PD, p.2)

Response

The question of containment isolation failure has been examined and found to have negligible influence on the risk. The vast majority of the lines which isolate are connected to systems which are closed on the containment side, the ex-containment side, or both. An isolation failure in these cases would not result in a release unless other failures are postulated. The combined likelihood and expected low levels of release insure that such events are negligible in terms of risk.

For lines not connected to closed systems, the isolation valves are closed and verified prior to power opertion except for the small purge line valves which may be actuated periodically to permit operator access to the containment. These latter valves receive an automatic isolation signal. Given the small fraction of time these valves are open and the failure rate of these valves to close on demand, we find that a failure to isolate these valves offers a negligible risk contribution.

IV.13 Containment Failure and Integrity

(1)B. Aside from major openings has there been a systematic review of all other penetrations (piping, electrical, etc.) to assure that, under the predicted high tempertures and pressures, no premature failure occurs in the sense that the minimum structural strength or leak-tightness of the vessel is degraded? (ANL, C 3.10: PD, p.2)

Response

A review was conducted, on a generic basis of all types of penetrations to insure their adaquacy.

IV.13 Containment Failure and Integrity

(1)C. How and to what extent have interfacing system integrity failures been considered in the recirculation mode under class 9 accident conditions of temperature, pressure, and radiation. (PD, p.3)

Response

The types of failures noted have been considered. Three environmental factors, temperture, pressure and radiation were considered. For the ECCS systems in the recirculation mode, the temperature and pressure conditions are not an issue since, with the exception of the recirculation sump valves, routine operation can involve pressures (and in the case of the RHR system, temperatures) higher than those of interest in the study. Temperature might be thought to have a long term impact on HPI system pump seals but, at worst, given seal cooling failure, such effects would be to increase seal leakage rather than to stop the pump from performing its function. Valve packing is generally a graphite packing qualified for higher temperatures than those expected. The sump valves need only open early in the sequence progression prior to the development of severe conditions.

The source term used to qualify the equipment for Zion was essentially TID-14844. In-containment radiation levels may exceed this level somewhat in time but ECCS failures in the time period of interest are not expected. Any such failures postulated would most likely involve valve packing leaks thru a closed leakoff system to radwaste. System failure would be extremely unlikely.

IV.13 Containment Failure and Integrity

(1)D. How has failure of the containment purge system been considered in the analysis of containment integrity? (PD, p.3)

Response

See response to IV.13 (1) A

IV.13 Containment Failure and Integrity

(1)E. How has the possible failure of fan coolers from aerosol plugging of filters been considered? (PD, p.3)

Response

See response to enclosure 3, Systems Analysis, question 2.

IV.13 Containment Failure and Integrity

(1)F. Primary containment ultimate capacity was determined to be 149 psia. It was calculated by Sargent & Lundy (Appendix 4.4.1). The analysis was supposed to cover: 1) containment structure, 2) penetrations, 3) rate of loads, 4) uncertainty bounds, 5) failure mechanisms. Only the first two items are dealt with, in some detail. (ZZ, p.2)

Response

The rate of loading was furnished to Sargent & Lundy based on reasonable bounding analyses from the transient cases in section 4. Therefore, Sargent & Lundy (S&L) did not have to perform any such analyses.

An assessment of uncertainty, based on Sargent & Lundy's work is contained in the Zion study (Figure 2.5.1.2) and in the cited S&L report (section 7).

The failure mechanism is discussed in section 8 of the S&L report. The criteria for establishing a failed condition is also in the S&L Report. We do not believe it is reasonable to attempt to predict an exact failure configuration and size and have therefore taken a conservative approach to defining a significant failure condition.

IV.13 Containment Failure and Integrity

(1)G. Containment structural capability analysis does not address the systems and structures attached to the containment wall. While 149 psia appear to be a reasonable number for the cylinder, containment bypass potential as a consequence of some pentration break away prior to this pressure has not been included in the study. (ZZ, p.4)

Response

The containment analysis considers the pentration to liner joint as part of the overall integrity evaluation.

IV.13 Containment Failure and Integrity

J. In order to more fully ensure integrity of the containment structure it would be helpful to have an evaluation of the effects of credible construction errors to see what effect they might have on downgrading the calculated structural strength of containment. (ANL, C 5.3)

Response

The initial testing program for Zion station included a pneumatic proof test of the containment. The test pressure employed was approximately 69 psia. During this test, containment response was accurately measured with theodilites and compared to calculated responses in some detail. A leak rate test followed the proof test (Ref. Zion FSAR section 5.2.) The ongoing tendon surveillance program has revealed no problems in this area.

These programs provide a high degree of assurance that no gross construction errors exist. It is expected that minor flaws such as very small holidays in the concrete would have no effect on the calculated strength. The uncertainty characterization for the containment capacity used in the Zion study is therefore judged to be entirely appropriate.

IV.13 Containment Failure and Integrity

(2) The 1 σ uncertainty in the containment structural failure pressure is of the order of $\pm$ 2 psi based only on structural property uncertaintly. Considering all the above factors, the use of 2 psi uncertainty band requires additional justification. (ANL, C 3.10)

Response

The uncertainty expressed in figure 2.5.1-2 of the Zion report represents our best estimate of the lower bound "s" curve of a family of "s" curves. For simplicity sake and in the interest of remaining within the state of the art, we have conceptually collapsed all of the curves onto this lower bound curve. The curve represents material uncertainties as well as other considerations.

IV.13 Containment Failure and Integrity

(3) Analysis consists of hand calculations and of an axisymmetric finite element computer calculation. It is strictly deterministic and the conclusion that confidence level of 95% is associated with the calculated containment internal pressure capability is not supported (presumably based on knowledge of materials property statistics.) (ZZ, p. 3)

Response

See response to previous question IV.13 (2).

IV.14 Core Retention Device

(1) A core retention device can reduce hydrogen generation, steam generation, aerosol formation both before and after containment failure, as well as retard basement erosion. Analysis in ZPRA focuses only on the basement erosion issue. (DP, p.9)

Response

Refer to sections 9.1.3.1.4, 9.1.3.2.7, and 9.1.3.3 of the Zion PSS. The comment above is incorrect.

IV.14 Core Retention Device

(2) Even with this focus on basement erosion the appraisal of the results is unusual. A minimal retention device will, according to the analysis, prevent basement penetration for two days. This is in contrast to less than a day predicted by ZPRA or less than a few hours if the cavity sump is considered as suggested above. Further, the probability of restoration of power during this two days is estimated to be quite high so that enforced cooling of the core debris becomes possible. The ability of a core retention device to contain the core debris until enforced cooling becomes available is not pointed out as a benefit of the device in the Zion PSS. (PD, p.9)

Response

Refer to section 9.2.4.1 of the Zion PSS. The comment is incorrect.

V. SOURCE TERMS

V.1 Comparison of Point Estimate Risk with Level 2 Risk

By assigning probability weights to the site matrix through use of "source term multiplier, U" the effective accident source and resulting consequence is reduced by over a order of magnitude. In light of the impact of source term reduction in the site matrix, the ZPSS authors should be requested to provide additional detail to support the probability-weighting values used. (ANL, C 1.0)

Response

Refer to sections 5.6 and 6.3 of the Zion PSS. For additional detail see containment and consequence analysis Question 2 of Enclosure 3.

V.2 Oxidative Source Term

(1) In the ZPRA the role of steam explosions is down played, and they analyze release from only half the core. The entire thrust of the ex-vessel interaction analysis in ZPRA involves very dynamic dispersal of core debris. The amount of material involved in the dispersal is the same as in the steam explosion events of the Reactor Safety Study. Further, the time hot debris is exposed to the oxidizing environment according to the ZPRA analysis is far longer than the exposure according to the Reactor Safety Study steam explosion analysis. An accentuation rather than a reduction of the Reactor Safety Study source term is clearly called for. (DP, p. 8)

(2) The error is compounded because coupling between sections on processes and the release of radioactivity is extremely weak. If there is one area where coupling should be strong it is between release of fission products from the fuel and the behavior of the fuel melt or debris during the accident. (DP, p. 8)

Response V.2 (1) & (2)

We disagree for the following reasons. The dispersive event in the ZPSS is not a steam explosion nor is it that similar in nature. It represents an ejection of core debris from the reactor cavity and its distribution onto the water covered containment floor by wave processes associated with gas blowdown from the reactor vessel bottom penetration as discussed in Section 3.2. In contrast with the steam explosion considered in the RSS which involved 50 percent of the core finely particulated and oxidized while airborne in the containment atmosphere, the dispersive event is one in which approximately 50 percent (or 80 percent in extremely unlikely case) of the core would be involved in flow from cavity to containment floor as a result of a gas jet. In this process it was considered unlikely that debris might be finely particulated and suspended in the containment atmosphere as in the case of a steam explosion. Therefore RSS category 2 without oxidation enhancement was used. To account fcr steam explosion source enhancement in a non dispersive (gravity drop) situation it was conservatively assumed to be equivalent to 50 percent of the total core mass being involved in a steam explosion that renders 25 percent oxidized as fine particulate matter in the RSS manner from the standpoint of a fission product release source term in the event of an early containment failure. This is clearly described on pages 5.3-1 and 5.3-2.

VI. CONSEQUENCES

VI.1 Completeness

(1) The Zion study was limited to examining offsite health impacts and did not specifically assess either the contamination of land areas or the financial consequences resulting from potential accidents. For some accidents, these other consequences are dominant.

Response

The Zion PSS did not assess land contamination or financial consequences as noted. Absent such an assessment and absent any agreed upon scaling relationship between these and other consequences, it does not appear that one can judge them to be "dominant" for any accidents.

VI.I <u>Completeness</u>

(2) Genetic effects are not included in the assessment of health effects.

<u>Response</u>

Refer to section 0.11.1 of the Zion PSS. The statement is incorrect.

VI.2 Emergency Actions

Assumption regarding emergency actions can have a [illegible] on calculated results.

(1) Early effects are very sensitive to assumed evacuation parameters, particularly the delay time before public movement. These sensitivities should be displayed in the reported results, and distributions for the parameters rather than best-estimates should be used. (DA, p. 4 and 5)

Response

The assumed point estimate for delay before evacuation was based on the belief that the extensive emergency planning in place provides the capability to rapidly warn evacuees and initiate response within 1 hour. Variations in this point estimate are recognized in the probabilistic treatment of the U values in the S matrices for estimating uncertainty.

VI.2 Emergency Actions

(3) The assumption is made that accident initiation times may be obtained by the random selection of times from a uniform distribution through the entire year. The general procedure practiced by utilities is to run reactors at or near 100% power because of their economic advantages over backup power. However, outages are required for maintenance and refueling. Because of economic advantages, it is assumed that the utility will attempt to plan these outages at times of low power demand (i.e., when the need for more expensive backup power is lower). Since the analysis implicity assumes that the outages are uniformly distributed through the year, two questions result:

1. Are outages distributed uniformly throughout the year?
2. If not, would the use of historical and/or anticipated outage distribution data have significant effect upon consequences?

An appraisal of this potential effect is recommended for the following reason. Outages may decrease the initiating event frequencies for a particular time of year. Clearly the weather, evacuation, and population data that apply during times of year (or day) when the initiating event frequencies are highest should be weighted more strongly than others. Sampling for a on-uniform distribution will recognize this potential effect. (ANL. 2.4)

Response

Basically, Edison makes every effort to have its nuclear plants on line during the months of June, July and August to accomodate summer peak loads. Refueling and maintenance outages may be viewed as uniformly distributed through the remaining months.

No significant effect on consequences is anticipated for this slight non-uniformity.

VI.2 Emergency Actions

(4) For large accidents, the required medical personnel and equipment may not be available for the assumed supportive medical treatment. This should be reflected in the results. (DA, p. 7).

Response

Supportive treatment was assumed to be available for persons receiving life-threatening doses. This assumption is the same as that used in the Reactor Safety Study and is described in WASH-1400, Appendix VI, Section 9.2.1. The following basis for the assumption is similar to that used for the RSS and is a summary of information in the reference cited.

Supportive treatment for persons receiving doses in the range of 350 to 600 rem to the bone marrow would require specialized facilities primarily consisting of adequate medical personnel and laboratory support. Specialized treatment would not be required immediately but could be begun up to about 3 weeks after the accident. Because of the time available, it is reasonable to assume that national medical resources would be available to aid exposed people. A partial inventory of facilities which could provide such support, including only those with approved programs for residencies in internal medicine, indicated that 2,500 to 5,000 people could receive such treatment in U.S. hospitals. Medical treatment for persons receiving less than 350 rem would not require special facilities.

The limited availability of special facilities is a constraint only in very low probability scenarios. In analyses used to produce the S1 matrix in the ZPSS, only a few scenarios resulted in estimates of more than 5,000 persons in the dose range of 350 to 600 rem. Explicit consideration of this resource constraint is not expected to change risk curves significantly except in the low frequency tail of the curve where early fatality consequences could be higher by a factor of 2.7. However, it should be recognized that the relatively high estimate of early fatalities in the worst scenario stems from the conservative assumption that no protective action is taken beyond the EPZ. Even an ad hoc action such as basement sheltering for those with basements would be expected to reduce early fatality consequences markedly. The reduction would be more than enough to balance the increase due to the apparent constraint in medical resource availability and could reasonably be expected to eliminate the constraint entirely.

VII. External Events

This is a "first of its kind" analysis of seismic risk integrated into a probabilistic risk assessment. As such, many of the methods used are without precedent and should be considered as extensions of the art rather than applications of it. Significant questions relative to the ZPSS seismic analysis are as follows:

(1) This study used a rapid seismic attenuation relation for predicting peak accelerations at the Zion site and a low cutoff (zero probability of ground accelerations above 0.65g). Such an assumption is open to question; there are competing theories which would give low probabilities of substantially higher peak accelerations. (ANL, C 4.0)

Response

The attenuation rate used in the study is supported by both empirical and theoretical results (Nuttli, 1979b*), and is found acceptable in reviews by Drs. Street and Trifunac. Also see response to NRC Staff and Consultants Questions, Seismic Fragility, Question 15.

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* This reference is in ZPSS Section 7.9.1.

VII. External Events

VII.(2) The use of a lognormal distribution for describing uncertainty and variability in fragilities is probably justified although the assertion that such distributions are accurate to the 0.01 level is optimistic. There is evidence that lognormal distributions show considerable variation with failure data at the .05 to .10 level. This becomes significant because, as is pointed out in the report, the seismic risk is associated with an interaction of the tails of the seismicity distribution and the fragility distributions. Further study of this issue is warranted. (ANL, C 4.0)

Response

For many variables considered in the development of the seismic fragility curves, sufficient data does not exist to exactly determine the distribution. However, for many variables such as material strength (References 1 and 2) and earthquake characteristics (Reference 3), the lognormal distribution has been shown to provide a reasonable approximation of the data.

An important relationship in regard to the determination of the seismic fragility curves is the central limit theorem which states that products and quotients of several variables tend to be lognormal even if the individual variable distributions are not lognormal. The approach used in determining the seismic capacities consists of identifying the factors of safety inherent in the design for each of a number of design parameters. The overall factor of safety consists of the product of the individual factors of safety for these parameters which includes contributions from material strength, earthquake characteristics, damping, etc. Thus, although the distribution associated with some of these variables may not be known, the central limit theorem provides assurance that the final distribution will be at least approximately lognormal. In view of the uncertainty associated with many of the variables contributing to the overall seismic capacity, considerable additional effort to exactly define the shape of the fragility curve is not considered to be warranted.

References

(1) Freudenthal, A. M., J. M. Garrelts, and M. Shinozuka, "The Analysis of Structural Safety," Journal of the Structural Division, ASCE, ST1, pp 267-325, February, 1966.

(2) Kennedy, R. P., "A Statistical Analysis of the Shear Strength of Reinforced Concrete Beams," Technical Report No. 78, Department of Civil Engineering, Stanford University, Stanford, California, April 1967.

(3) "A Study of Vertical and Horizontal Earthquake Spectra," WASH 1255, Nathan M. Newmark Consulting Engineering Services, prepared for USAEC, April 1973.

VII. External Events

VII.(3) It is the impression of the reviewers that the uncertainties, β_u, for the fragilities, particularly the equipment fragilities, are underestimated. The reasons are: (1) several studies attempting to predict loads on piping during seismic simulation tests have indicated discrepancies of 200 to 300 percent; (2) extrapolations of linear analyses to failure for equipment which may exhibit hardening nonlinearities (e.g., gaps) can grossly overestimate fragilities; and (3) there has apparently been no consideration given to uncertainty in quality control, design error, and installation error in the fragility calculations. (ANL, C 4.0)

Response

1. We are aware that there have been large discrepancies between predicted and observed piping responses, particularly if the systems are excited at low levels. At low levels of excitation small differences in damping can have a very large effect on response if the system fundamental frequency is in the amplified range of the applicable response spectrum. Also, mathematical modeling of damping at low levels is another source of uncertainty since friction at supports can be a significant contributor relative to material damping. When treated as equivalent modal damping, the mathematical model can contain significant uncertainty. At higher levels of acceleration coincident with the threshold of equipment failure, the response sensitivity to damping is greatly reduced and damping is dominated by material hysteresis rather than by a combination of friction and material damping. Other uncertainties in dynamic modeling tend to have less impact on resulting reponse.

If we examine the predicted range of uncertainty in response for piping and components that were designed by analysis and are flexible such that their dynamic response is amplified, we find that the predicted combined β for structural response and equipment response ranges from about 0.3 to 0.5, i.e., for a given ground acceleration input, the overall variability in equipment response is expressed as:

$$\beta_{C_{ER}}^2 + \beta_{C_{SR}}^2$$

where $\beta_{C_{ER}}$ is the combined random and uncertainty β's for equipment response and $\beta_{C_{SR}}$ is the combined random and uncertainty β's for structural response. The variation is in a large part due to the building and equipment frequency variation. The stiffer structures and equipment tend to have lower variability in response.

Using the properties of the lognormal distribution, the following table portrays the magnitude of uncertainty in equipment response as used in the probabilistic risk assessment study of Zion.

β_C	-1β	$+1\beta$	$\frac{+1\beta}{-1\beta}$
0.3	.74	1.35	1.82
0.4	.67	1.49	2.22
0.5	.61	1.65	2.70

β_C	-1.65β	$+1.65\beta$	$\frac{+1.65\beta}{-1.65\beta}$
0.3	.61	1.64	2.69
0.4	.52	1.93	3.71
0.5	.44	2.28	5.18

The range from plus to minus 1 β represents the 85 percent probability of nonexceedence and the plus to minus 1.65β range is the 95 percent probability of nonexceedence. The ratio of the maximum response/minimum response is quite significant and certainly addresses the large response uncertainty quoted in the question. We do not have good statistical data on test response versus predicted response for complex systems excited to high levels of motion; thus, the uncertainties in response portrayed above are estimated by examining what we consider realistic bounds on each of the variables that contributes to overall dynamic response and again examining the resulting overall response variability for reasonableness. Further research in this area would be valuable in quantifying uncertainty in response and the results may possibly show some increased uncertainty over that which has been estimated; however, we do not feel that the effect would be grossly different than utilized in the study.

2. In regards to extrapolation of linear analysis to failure levels for equipment, we have tried to estimate reasonable bounds on the limit load or stress for static loading and reasonable bounds on ductility. The estimated uncertainties in limit load are supported by collapse moment test data for simple components and material property variation available in the literature. The failure models and resultant uncertainty in the failure level are based upon simple failure mechanisms. In some cases for highly redundant systems, the uncertainty may be underestimated but the capacity is likewise underestimated. This would tend to bias the fragility to the low side. For components that were identified as significant contributors to risk, all assumptions were reexamined to assure that the best estimate of capacity was used and that possible bounds on capacity were included in the β's.

System ductilities were estimated by examining reasonable bounds on ductility. For components which could possibly fail in a brittle mode, the lower bound ductility was established at 1.0 with the estimated median value being in the range of 1.5. Even though inelastic energy absorption capability has been included in the capacity factor, we do not feel that we have overestimated median capacities and have attempted to anchor lower bounds to very conservative values of limit load and ductility, such that if there is any bias it is on the conservative side.

3. Response is the same as given to NRC Staff and Consultants questions, Seismic Fragility, Question 11.

VIII. OTHER COMMENTS

(1) Are design errors and fabrication errors included adequately? If not, how much change or uncertainty would be introduced by allowing for them?

Response

We believe that an adaquate treatment has been included. Both the age and operating experience of Zion as well as the extensive plant testing argue against any significant likelihood of such errors remaining undetected. The use of category "other" as well as the inclusion of broad uncertainties accounts for any remaining likelihood.

VIII. OTHER COMMENTS

(2) Are human errors of commission adequately treated in the analysis?

Response

See response to II.3(4) above and Section 0.19.4 and 0.19.5 in the ZPSS.

VIII. OTHER COMMENTS

(3) Is there a significant likelihood that a severe earthquake may not only cause a core melt accident but also directly cause a loss of containment integrity?

Response

There is no significant likelihood of direct containment failure from seismic events.

However, there is a small likelihood under very large earthquakes, for bumping of the reactor and auxiliary buildings with resulting penetration of the containment liner. This scenario is included in the seismic risk calculations.

VIII. OTHER COMMENTS

(4) Does the evaluation of operator error for a severe earthquake have to be reconsidered by view of the potential for major losses of information in the control room (combined with failure or spurious behavior of much equipment with is not seismically qualified?

Response

Given the dominant earthquake failure modes, there is little potential for operator interaction of any kind. No credit is taken for such action.

VIII. OTHER COMMENTS

(5) Has the failure of large pressure components within the containment been adequately treated?

Response

The question is somewhat lacking in specificity. We believe we have adaquately treated component failures.

ENCLOSURE 3 QUESTIONS

Summary of NRC Staff and Consultants Questions on the Zion Probabilistic Safety Study (PSS)

Systems Analysis

1. The ZPSS gives complete credit for the remaining water and refilling the RWST following dryout to allow for continued containment spray operation. Describe the plant procedure which instructs the operator to do this. Include a discussion of time CSIS will be available with respect to loss of low pressure recirculation, core melt, vessel failure and containment cooling. (See ACRS question III. 3(2))

Response

The plant analysis section of the Zion PSS contains event trees which include a nodal question regarding the functioning of the containment spray system in the recirculation mode. Such functioning involves the operation of the residual heat removal (RHR) pumps in a recirculation mode, drawing water from the containment recirculation sump and pumping through the residual heat removal heat exchangers to the spray ring headers. Close examination of the plant state assignments for the branch points in the event trees will reveal that, even for cases where such operation succeeded, no credit in terms of plant state assignment was taken for recirculation sprays. Instead, only where injection phase spray succeeded, was the "C" designator in the plant state applied.

This approach is essentially a simplification relative to plant event state development and represents a degree of conservatism in the study. For example, an SEF plant state means that the containment spray system failed to function during the injection phase. It is applied to both the success and failure branches of the recirculation spray question. Therefore, even if the recirculation spray system succeeds, no credit is taken for spray operation.

The application of the "C" designator to all sequences where injection phase spray operation succeeded is not unrealistic. Even without fan cooler operation, the Zion containment can tolerate significant pauses in spray operation without approaching overpressure failure limits. For example, given an SE plant state, the time to overpressure failure is at least 12 to 14 hours. The refueling water storage tank (RWST) has enough of a "heel" of water at switchover to recirculation to permit injection spray operation for about 30 minutes. This permits the sprays to reduce any initial pressure rise at the time of vessel failure. The Zion study examined (but did not set forth in the report) the ability of plant operators to refill, to some reasonable level, the RWST in the 12 to 14 hour time period prior to containment overpressure. Several sources of water exist which could be used for this purpose.

First of all, the two primary water storage tanks, which are connected on a common header, can each hold 200,000 gallons of reactor grade water. These tanks feed the primary water makeup pumps which in turn can refill the RWST through the boric acid blender or associated blender bypass lines.

Secondly, the RWST can be refilled by pumping water from the spent fuel pool to the RWST. The spent fuel pool inventory can easily be maintained from the demineralized water system or, if need be, by manual operation of fire hoses.

Thirdly, the RWST can be manually filled by connection to fire headers or by bringing a fire truck on site and filling the tank from the crib house forebay.

It is important to note that, in terms of containment heat removal, boration of spray water is not significant. Also, it is important to note that, at the time in question, boration is not an issue in terms of return to criticality. Lastly, pH adjustment for fission product retention can, if necessary, be achieved by refilling the sodium hydroxide tank on the spray system and/or by adding boric acid from the boric acid storage tanks thru either the charging pumps or the spray pumps.

Operation of the spray system to prevent overpressure containment failure does not need to be a continuous function as noted earlier. Approximately 12 to 14 hours are required to fail the containment absent any safeguards. If only one of the 3 methods of filling the RWST is employed, enough water can be added to the RWST in about 3 hours to allow the spray system to effectively hold the pressure well below the failure level. Such a process can continue on an intermittant basis until either the fan coolers or the RHR recirculation system can be placed into effective operation. We have assigned and do still assign a frequency of ε to the failure of such actions.

It is recognized that no post core melt procedure exists at Zion which suggests such an approach. In fact, no post core melt procedures of any kind exist for any specific scenarios anywhere in the industry or in any regulatory requirement to our knowledge. Since the actions of interest occur literally hours into such an event, with a full complement of emergency support services and personnel available, and since the actions of interest are glaringly obvious, it seems reasonable to assign a very low likelihood to any failure to recognize the need for such actions. Similarly, given the variety and diversity of means available to accomplish these actions, it seems reasonable to similarly view combined systems failures as very unlikely for the plant states of interest.

Therefore we find it totally unreasonable to simply <u>assume</u> that no refill of the RWST occurs for the plant states of interest. Such an assumption may be worthwhile in the context of a sensitivity study to assess the worth of long term spray availability. However, to extend that assumption beyond the realm of a sensitivity study and to represent it as "realistic", based on a lack of post core melt procedures, is grossly irresponsible and totally incorrect. Sandia, of course, in its' review, went even further and recommended that "no credit" be given to such operations since no procedures existed which call for such actions. Not only does this violate logic as noted above but it is totally contrary to the basic concept of realistic PRA. It constitutes a regression to arbitrary licensing practices. We clearly do not accept such reasoning in the context of risk assessment and cannot consider that such reasoning constitutes an valid basis for assessing risk or evaluating Zion Station.

System Analysis

2. The ZPSS assumes that the operation of the containment fan cooler system will not be adversely affected by the environment present in the containment following a core meltdown. Since many fan cooler design parameters are exceeded by the post core melt environment, provide an analysis to support this assumption

Response

We have performed a detailed analysis of fan coolers operability under degraded core conditions. During a degraded core event, the fan coolers may be challenged by aerosols generated during core degradation and core/water/concrete interaction, as well as elevated temperature and pressure in the containment. Even under these adverse conditions, our analysis showed that the fan coolers would not fail. The detailed analysis is given below.

The Zion containment fan cooler system is provided to filter, cool and dehumidify the reactor containment environment during normal and design basis accident conditions. During normal operation air is drawn through the inlet damper passing through the roughing filter plenum, cooled by the cooling coils and discharged by the fan to the distribution duct work.

If a severe accident occurs, the fan speeds are reduced (85,000 cfm to 53,000 cfm) and the air flow is re-routed from the return air duct through an accident flow inlet damper and into the filtration package plenum, through moisture separators, HEPA filters, cooling coils and discharged by the fan.

The Zion containment fan coolers are designed to survive the design basis LOCA transient during which the containment pressure is expected to rise to 47 psig and temperature to 271°F in about 10 seconds. The fan coolers are conservatively rated for up to three hours in a containment environment of 47 psig and 271°F and gas mixture density of 0.172 lb/ft/^{3}. As long as the fan coolers are operational during degraded core accidents, the long term containment pressure and temperature would be significantly below 47 psig and 271°F, respectively. At certain times during the accident, such as during hydrogen burns and vessel failure, containment conditions may exceed the design basis LOCA conditions, however such conditions should persist only for short durations. During an energetic hydrogen burn, the containment pressure can be expected to rise by about 60 psi in about 20 to 60 seconds and then decay to the pre-burn value in a short time. This is a slower pressure rise than that for the design basis accident, which results in a pressure rise of about 47 psi in about 10 seconds. Because of the inherent conservatism in the design analysis, and the relief capacity of the fan cooler housings, it is expected that the fan coolers should be capable of withstanding a significantly more severe pressure transient than this design basis transient. Hence, the fan coolers are expected to survive the pressure transient during a realistic hydrogen burn scenario.

The duration of thermal transient from a significant hydrogen burn would be short (1 to 2 minutes). Components of the fan coolers are relatively large, rugged in construction and hence should have substantial thermal inertia. During equipment survivability tests conducted at Fenwal, Inc. and Acurex, Corp., the peak surface temperatures measured for equipment with relatively low thermal inertia (such as, pressure transmitter limit switch) exposed to hydrogen burns were generally under 300°F. (References 1 and 2). As the components of the fan coolers have substantially greater thermal inertia than the small equipment tested in the above tests, their peak temperatures during hydrogen burns can be expected to be below 300°F. All components of the fan cooler are designed to withstand, without impairing operability, a post accident containment temperature of 271°F for 3 hours.[1,2] Because of conservatism in the design analysis, the fan coolers can be expected to operate at temperatures substantially above 271°F. The most critical component, the fan motor, is totally enclosed in a jacket which is water cooled. The peak component temperature anticipated during hydrogen burns (about 300°F) is only slightly higher than the design basis (271°F) and its duration (a few minutes) is substantially shorter than that assumed for the design basis transient (3 hours). Hence, a short duration thermal transient from a hydrogen burn should not have adverse effects on heat removal capability of fan operation.

The aerosols generated in vessel, wnich consist of fission products and other core materials, would be effectively removed in the pathways, i.e. core and internals structures, and piping, to the containment by natural deposition, either laminar or turbulent, and gravitational settling. Since the steam content in the reactor vessel is high, fission product removal by steam condensation is also significant. The retention of particulates in the primary system depends on the thermohydraulic conditions involving a core melt. For accident sequences with low steam flow rates in the primary system, such as transient or small break events, a small fraction of the total aerosol mass from the core would be released from the primary system. Given the long and winding paths, and low steaming rates for most of the accident sequences following core melting, it is expected that a very small fraction of the aeosol inventory in-vessel will escape into the containment.

If the containment sprays are operating, considerable airborne aerosols would be washed out of the atmosphere. Within the RCFC's the moisture separators would remove significant quantities of particulates prior to reaching the HEPA filters.

The quantity of aerosols generated during core degradation is primarily from core melt. As stated in ZPSS, in the case where water is available to form a coolable debris bed following vessel failure, very little aerosol would be released from the core debris since the core melt is quickly quenched. Even if one assumes that the debris bed is non-coolable, the basic plant geometry ensures that the core debris will be water covered when fan coolers are operating. This water cover will remove most of the aerosols prior to their release into the containment. Therefore, whether a coolable or non-coolable debris bed is formed, aerosols generated from core-concrete interaction are not an issue.

The above arguments that support our view of an insignificant effect of aerosols on the fan coolers are summarized as follows:

1. Aerosol generation would be limited to those generated in-vessel.

2. Aerosols would be mostly retained in the transport pathways, such as primary systems piping and structures, water in the reactor cavity and the containment sump, etc.

3. The fan cooler location, associated ducting, and moisture separators will preclude aerosols reaching the HEPA filters in a significant amount.

Thus the continued operation of the fan coolers is assured.

However, it can also be shown that the HEPA filters can absorb significant quantities of aerosols without affecting fan cooler performance. In the following analysis, we assume that the aerosol concentration is $3 gm/m^3$ (or 480 lbs. of aerosol in the containment). The aerosol concentration of $3 gm/m^3$ represents the peak concentration(3) calculated for a AD (i.e., a class I event) sequence. For a TMLB" (i.e., a class V event) sequence, if power is recovered at about an hour after vessel failure or later, the calculated aerosol concentration at that time is lower than $3 gm/m^3$(3). In this case, actuation of containment sprays would rapidly decrease the aerosol concentration. For a S_2D (i.e., a class III event) sequence, even though the peak aerosol concentration is higher than $3 gm/m^3$(3), the peak concentration exists for only a short time right after vessel failure and it would be rapidly decreased by containment sprays. Before the sprays are actuated, the aerosol concentration higher than 3gm/m^3 exists for only a short period of time. Using the pressure drop data for HEPA filters shown in Figure 1, and conservatively assuming a mean particle size of 2.6 microns (vs. the expected size of larger than 5 microns), the fan cooler characteristics can be calculated. Table 1 summarized the input data. The calculated RCFC flow was determined as a function of time following the accident and is shown in Table 2.

As the results indicate all the aerosols (i.e., approximately 480 lbs.) could be absorbed by the filters without degrading performance. In fact, 2" w.g. pressure drop is well below that required to plug the fan coolers, and is less than the maximum 4" w.g. recommended for normal operation. This conservative analysis demonstrates that filter plugging of the fan coolers by aerosals should not be a concern.

It should also be noted that this simple analysis took no credit for the moisture separator aerosol removal capability which would probably be significant.

Lastly, it should be noted that the retention of the HEPA filters in the Zion fan coolers served no useful purpose from a safety standpoint. No credit is taken in the FSAR safety analyses for the HEPA filters and the HEPA's are not used for cleanup in the normal operating modes. The HEPA's could be removed with no effect on either the FSAR safety analyses or the Zion PSS. In fact, the removal of the HEPA's would only have the beneficial effect of reducing the burden of testing and surveillance imposed on station personnel.

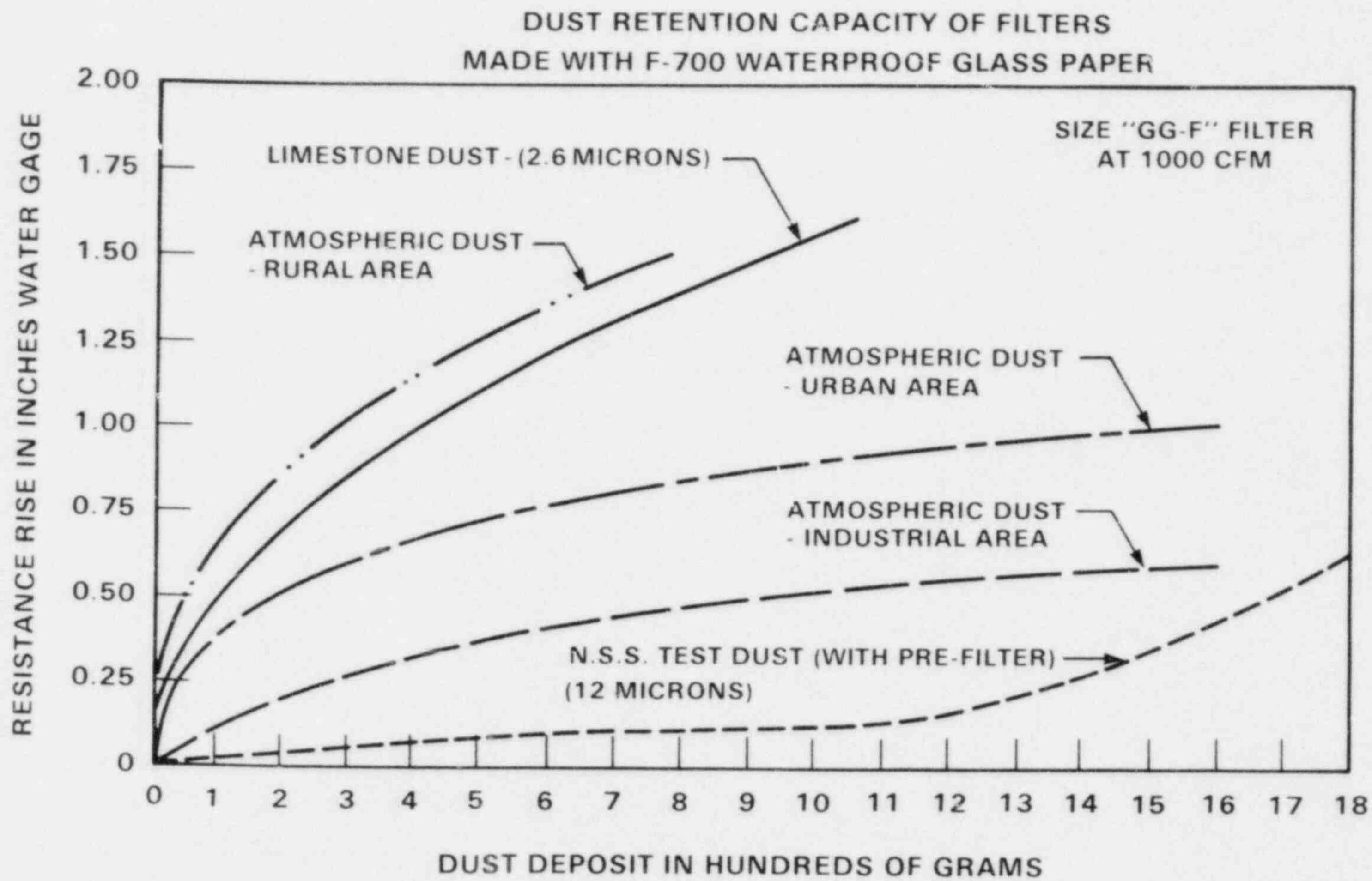

Figure 1. HEPA Filters with F-700 Medium, Separator-Type Dust Retention Capacity of Filters Made with F-700 Waterproof Glass Paper

TABLE 1

FAN COOLER CHARACTERISTICS

1.	Number of Reactor Containment Fan Coolers (RCFC)	5
2.	Number of filter units (24" x 24" x 12" each) per RCFC	45
3.	Post LOCA design air flow	53,000 cfm
4.	Containment conditions:	
	a. Temp.	271°F
	b. Press.	47 psig
	c. Air density	0.172 lb/ft^3
5.	Aerosol average particle size (limestone dust)	$2.6 \mu M$
6.	HEPA rise in pressure drop from in loading	See Figure 1
7.	Housing relief area	4.25 sq. ft.

TABLE 2

ZION PLANT-RCFC FLOW IN THE POST BEYOND DESIGN BASIS ACCIDENT

Time Following	3 RCFC Units Operating		5 RCFC Units Operating	
Accident	HEPA-dP	Flow	HEPA-dP	Flow
Min	Inches-H_2O	CFM	Inches-H_2	CFM
0 plus	0.14	171,000	0.14	285,000
4	0.98	159,000	0.64	273,000
10	1.40	154,000	1.10	263,000
20	1.75	150,000	1.33	258,000
30	1.92	148,000	1.45	256,000
60	2.20	146,000	1.60	253,000
100	2.20	146,000	1.66	252,000
120	2.20	146,000	1.66	252,000

References

1) Liparulo, N. J., Olhoeft, J. E. and Paddleford, D. F., Glow Plug Ignitor Tests in H_2 Mixtures, Westinghouse Electric Corporation, WCAP-5909, March 6, 1981.

2) Torok, R. C., "Equipment Survivability Scoping Tests - Hydrogen Control Studies Program," Paper present at Hydrogen Review Meeting at Dallas, Texas, February 3 and 4, 1982.

3) "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," NUREG-0772, U.S. Nuclear Regulatory Commission, June 1981.

Systems Analysis

3. The ZPSS assumes that safety system pumps do not require cooling as long as they are drawing from a cool water source. Please provide your analysis to support this assumption.

Response

The statement in the HPIS Analysis, (see ZPSS Section 1.5.2.3.1.2.4) concerning pump cooling is incorrect. The charging pumps require 300 gpm of component cooling water per pump for oil cooling; loss of the cooling water will result in pump failure in a short period of time. The safety injection pumps require 30 gpm of component cooling water for oil cooling; loss of the cooling water will result in pump failure in a short period of time. Because the HP injection pumps were assumed to operate for the loss of component cooling event, the correction of these errors requires further analysis to determine the exact impact on core melt frequency and plant risk. However, preliminary estimates indicate that there will be no significant impact on either core melt frequency or plant risk since all five CCW pumps must fail to lose all CCW flow.

The statement in the LPIS Analysis (see ZPSS Section 1.5.2.3.2.2.2) concerning pump cooling is correct. The RHR pumps require seal injection flow to maintain the integrity of the mechanical seal. The seal injection flow is provided by the pumped fluid. An attached seal water heat exchanger is provided for each RHR pump for use when the fluid being provided to the mechanical seal is hot. During the injection phase, the RWST water supplied to the mechanical seal is approximately 100°F which is approximately the same temperature as the component cooling water supplied to the seal heat exchanger. During the recirculation phase of accident recovery, loss of component cooling is assumed to fail the associated RHR cooling loop.

4. Past generic ATWS analyses by Westinghouse were based on the following important assumptions regarding the ability to limit the calculated peak pressure to below 3,200 psig:

   i) Availability of pressurizer safety and relief valves.

   ii) Capability (actuation circuitry diverse from the protection system) to automatically trip the turbine early (30 ∿ 60 seconds).

   iii) Capability (actuation circuitry diverse from the protection system) to automatically actuate all AFWS pumps (30 ∿ 60 seconds).

   If diverse actuation circuitry to perform mitigating functions defined under items ii) and iii) above are not implemented, provide your bases for the assumed peak pressures in the event of our ATWS.

   Why is core melt assumed to occur at 3,200 psig?

   What would be the effects on the steam generator tube integrity?

Response

Zion has the capability to trip the turbine early. The limiting case, complete loss of feedwater (trip of all main feedwater pumps), causes an automatic turbine trip. Partial loss of feedwater with no early turbine trip does not lead to peak pressures over 3,100 psig. All other analyzed transients generate automatic turbine trip signals independent of the reactor trip breakers.

Operational experience at Zion has shown that steam generator shrink following turbine trip causes low-low steam generator level alarms and start all AFWS pumps automatically.

Core melt was originally assumed to occur at 3,200 psig to simplify the ATWS event tree. This assumption has been dropped from our revised analysis. See response to Question III.3(1) in ACRS responses.

A 3200 psig pressure will not fail the tubes (Ref. WCAP-8330).

5. The ZPSS assumes the operator has a high probability of opening a PORV, during the early phase of an ATWS, in order to avoid exceeding 3,200 psi. Since Zion ATWS procedure does not instruct the operator to open the PORV, and only a few minutes are available to do so during an ATWS, the high probability of success used in the ZPSS needs further explanation. Estimate the time available more closely and give supporting evidence. Moreover, although with the resetting of the motor torque switches (see p. 1.3-40 of ZPSS) on the motors of the block valves, the motor-operated block valves are operable under primary system pressure, and it is not clear that they remain operable when there is an appreciable increase in system operating pressure above normal. DETERMINE THE MAXIMUM OPERATING PRESSURE AT WHICH THE BLOCK VALVES ARE OPERABLE, IF THIS PRESSURE IS LESS THAN THE PRIMARY REACTOR COOLANT SYSTEM BOUNDARY FAILURE PRESSURE, AND TAKE THIS PRESSURE INTO ACCOUNT IN CONSIDERING THE TIME AVAILABLE FOR OPERATOR ACTION. IF FAILURE OF ATWS PRESSURE SPIKE PROTECTION DOES NOT NECESSARILY LEAD TO CORE MELT, PRESENT SUPPORTING EVIDENCE (SEE QUESTION 4), AND REVISE THE EVENT SEQUENCES TO REFLECT THIS. QUANTIFY THE REVISED SEQUENCES (see ACRS question III.3(1)).

Response

See response to Question III.3(1) in ACRS responses.

Systems Analysis

6. Why are the "other" contributions to safety system unavailabilities always negligible?

Response

It is not true that the "other" contributions to safety systems unavailabilities are always neglibile. For example, the "other" category appears in the list of major contributors to the unavailability of the low pressure injection system (see ZPSS Table 1.5.2.3.2-11, Vol. 3, pp. 1.5-423).

Systems Analysis

7. The ZPSS gives credit for "feed and bleed" core cooling following a total loss of main and emergency feedwater. Is the "feed and bleed" success criteria consistent with results of the "feed and bleed" tests which were recently conducted at the LOFT facility. Provide your technical basis for assuming adequate core cooling via this mode.

Response

The LOFT experiment L9-1/L3-3 which was designed to investigate the response and recovery modes for a complete loss of main and auxiliary feedwater indicated that the LOFT facility had considerably more decay heat removal capability via the secondary than was envisioned or analytically predicted. However, due to the atypicalities between the LOFT facility and a Westinghouse designed PWR, the LOFT response does not necessarily verify the core cooling capability for "bleed and feed" decay heat removal. The technical basis for crediting "feed and bleed" as adequate for core cooling for total loss of feedwater is analytical studies performed for the Westinghouse Owners Group after TMI and documented in WCAP's - 9600, 9744, and 9914.

Systems Analysis

8. The ZPSS assume the reactor coolant pumps will leak at a rate of 300 gpm each approximately 30 minutes following a loss of seal cooling. Describe the analysis conducted which supports this flow rate.

Response

Highly conservative assumptions were made with respect to pump seal leakage following loss of RCP seal cooling. The base assumptions utilized in the analysis are summarized below. It should be noted, however, that operating experience and best estimate calculations indicate significantly extended times to seal failure from that assumed and that a major reduction in expected RCP leakage (< 50 gpm) is reasonable. However, the resources required to justify the best estimate values of time and leakage and the marginal benefits (reduced risk) associated with this change were unjustified. Thus, the upper bound values were used in the study.

Conditions

- RCS temperature and pressure at full operating conditions (2250 psig and 550°F).
- Site blackout.
- RCP's trip and coastdown.
- Seal injection and cooling water not available for 30 minutes.

Key Concerns

1. O-Ring Behavior

   O-Ring material is an elastomer that begins degradation of its mechanical properties at temperatures greater than 350°F. Some partial extrustion can be expected to occur. After 30 minutes at 550°F, some excessive cross-linking may occur resulting in permanent hardending. Some low pressure leakage may be expected.

Within 30 minutes, no catastrophic failure is expected but replacement is necessary.

2. No. 2 Seal Ring Graphitar

The carbon-graphite to stainless steel shrink fit would be lost in the 560°F to 610°F range. The 30-minute limit allows that this would be very unlikely. However, the design advantage of the compressive prestress will be greatly reduced as the temperature increases. Should for some reason, the No. 2 seal be placed in a high pressure mode, a tensile stress failure of the carbon-graphite may occur.

3. No. 1 Seal Aluminum-Oxide

Although there is a slight decrease in material properties at 500°F, the stresses remain far below safe limits.

4. Temperature Vs. Time

The moderating (mixing) effects of the injection water cavity is effective for approximately 10 to 15 minutes. Within 30 minutes, the seals will be subjected to full system temperature.

5. Double Delta Channel Seals

For the teflon based channel seals the expected temperatures will be significantly in excess of service temperature. Severe distortion of cross-section is likely. However, since the RCP is in the static mode, this distortion will have very little effect on the sealing capabilities. Subsequent replacement will be required.

6. Leakage With Total Seal Failure

If the seals were assumed to be totally removed, the leakrate could be estimated by considering only the restriction at the labyrinth seals, inlet holes, hydraulic plates, and the heat exchanger. Certain assumptions must be made about two-phase choke flow and the like. Conservative calculations indicate leakage for a typical 93A RCP to be from 200 to 300 gpm. Thus a 300 gpm leakage per pump was assumed in the study.

Systems Analysis

9. Why does the ZPSS not model safety system equipment room cooling but the Zion P&IDs shows several room cooling systems?

Response

The individual equipment room coolers consist of fans and heat exchangers. These heat exchangers are supplied with cooling water by the service water system. The fans are powered from ESS Motor Control Centers which are supplied from the same vital 4,160V switchgear buses as the associated pumps.

Loss of service water leads to loss of component cooling and/or diesel generator cooling. Either event is modeled by the existing plant event trees where the loss of equipment due to the loss of service water/component cooling water occurs in a shorter period of time than the failure of equipment due to loss of its associated room cooling heat exchangers.

Loss of the 4,160V switchgear buses is likewise modeled with the existing plant event trees.

Loss of individual room cooling fans could result in associated equipment failure over a period of time. However, the frequency of fan failure is much lower than the frequency failure of the associated equipment due to other causes.

For these reasons, equipment room cooling was not modeled in the ZPSS.

Systems Analysis

10. Justify the data used in the calculation of the interfacing systems LOCA (event V). Review of the data sources referenced would suggest this data does not exactly apply to the failure modes postulated. (See ACRS question III.3(4)).

Response

The data used for the rupture failure mode of check valves and gate valves is taken from the RSS for rupture of check valves and manual valves. The data appears to apply more to a valve body rupture than disc rupture; however, a review of data sources available at the time of the ZPSS did not reveal any data that was more directly applicable.

The data used for the failure mode disc transfer/remain open is not directly applicable for the demand event described; however, at the time of the ZPSS, no data source was available which could be used or referenced for this particular failure mode. Based upon work performed in support of other projects, the results using the data which was available at the time of the ZPSS are conservative.

Systems Analysis

11. Offsite power recovery is a major factor to which "station blackout" (loss of all AC power) and other loss of offsite power accident sequences is most sensitive. Depending on whether the recovery potential used in the Zion PRA or recovery based on other nuclear power experience is used, a significant difference can be obtained for the frequency of loss of offsite power related sequences. Justification should be provided for using the Zion PRA recovery model rather than the less optimistic values obtained based on actual loss of offsite power experience. (See ACRS question III.3(3)).

Response

The frequency and consequences of "station blackout" scenarios (loss of all onsite AC power) are affected most significantly by two factors in the Zion Probabilistic Safety Study models: (1) the frequency at which offsite power is lost (with subsequent failures of all diesel generators for a unit); and (2) the time required to restore power to the site from the offsite grid. Although power recovery is also possible by restarting or repairing one or more of the diesel generators, diesel generator recovery was not included in the Zion power recovery model. The reviewers note that the Zion offsite power failure frequency distribution and the distribution for time to recover offsite power are more optimistic than values based on "actual loss of offsite power experience." We believe that the Zion offsite power failure and recovery distributions accurately represent the combined information available from generic nuclear plant experience, specialized to the unique characteristics of the Zion site, the Commonwealth Edison transmission grid, the Zion onsite electric power system model, and the use of this information in the Zion Probabilistic Safety Study event trees.

LOSS OF OFFSITE POWER DATA

Many of the reviewers' comments regarding the frequency of the loss of offsite power initiating event and the estimated time to recover offsite power are based on data obtained from a report by Raymond F. Scholl, Jr., of the NRC (Reference 1). Mr. Scholl sent us a copy of his report, and our comments on the use of his data are included as item 3 of this response. However, we should also provide a few general comments which might serve to clarify the context and application of the loss of offsite power data as it was used in the Zion Probabilistic Safety Study.

The development of the data to be used in estimating the frequency of loss of power events must be consistent with its application in the plant response models. The loss of offsite power event is modeled in the Zion Probabilistic Safety Study as an event which causes a unit trip from power operatio and results in a loss of all offsite power supplies to the affected unit. Examination of the loss of offsite power events at several multiple unit sites shows that approximately 35% of these events occurred when at least one of the units was not operating at power. This percentage applies to power failures at multiple unit sites after all units at the site were in commercial service. If one includes the power failures which occurred after one unit was operating, but before all units were completed, the percentage of failures affecting fewer than all the units increases to approximately 50%. The point of the discussion is that the loss of power event is modeled as initiating a transient from power operation. Since a significant fraction of the power failures occur when units are shutdown, the data base should be adjusted to compensate for the availability of the units at the affected site. This also applies to the data for single unit sites, since at least 25% of the offsite power failures at these sites have occurred when the unit was shutdown. Therefore, although the reactors at multiple unit sites should not be treated independently for this initiating event, they should also not be treated as completely dependent. The use of a complete dependency assumption results in an overestimation of the frequency of the loss of offsite power event as it is modeled in the Zion Probabilistic Safety Study.

The data presented in the EPRI NP-801 report (Reference 2) provides a partial solution to this problem, since it reflects only those events which caused a unit trip from power operaton. Events which did not result in a unit trip or events which occurred when the unit was shutdown are not included in the data. Unfortunately, the data is indexed by reactor unit, and it cannot be used directly to provide common site events such as the frequency of offsite power failures which occur when all units at a site are operating at power (e.g., for a three-unit site, a common power failure event would be listed as three separate events if all three units were operating and all three tripped). Therefore, one cannot simply add the number of individual events documented in NP-801 for a multiple unit site and apply this data to the composite site event frequency.

Another important point to recognize when developing data for the loss of offsite power event is that the Zion in-plant power system model explicity accounts for the failures and maintenance of the system auxiliary transformers, which supply offsite power to the units during power operation and after a unit trip. A number of power failure events documented in the generic literature have been caused by single transformer failures. These events should not be included in the loss of offsite power initiating event data base to avoid double accounting for these failures in the Zion model. (They should be included in the offsite power failure data for a model which treats the transformers as part of the offsite grid.) Similarly, failures to transfer in-plant loads to the offsite source are also analyzed in the Zion power system model, and these events should be excluded from the offsite power failure data. (These events are not actual losses of offsite power, but for licensing compliance proposes they are often miscategorized as such in LER summaries.)

During the time since the Zion Pobabilistic Safety Study initiating event data was finalized, Pickard, Lowe and Garrick, Inc., has performed a more detailed review and analysis of the generic plant data for losses of offsite power at BWR and PWR sites. The basic event descriptions used in this analysis were obtained from three sources (References 3, 4, and 5) all of which rely heavily on the plant LERs. However, the information in

"Nuclear Power Experience" (Reference 5) often contains much more detail about the circumstances and causes of specific failure events than is available from the LERs, and it allows the data analyst to better understand the event and its applicability to the desired data base. The new PLG analysis includes events through December 31, 1981. The results of this study are summarized in Table 1 for the plants which were included in the Zion Probabilistic Safety Study initiating event data base.

Our analysis and comments on the use of the data developed by Scholl are included in item 3 of this response. We cannot determine the reasons for the differences between the revised PLG data and that reported in EPRI NP-801 because we do not have the basic event summaries used to develop the EPRI data. The PLG data base includes only those events which resulted in a loss of all offsite power to a unit. Events reported as partially degraded offsite power conditions (e.g., failure of one of two redundant offsite lines) are not included. However, events which did not result in a total loss of offsite power at the plant switchyard, but which caused failure of all sources available for automatic power transfer circuits and required operation of the diesel generators are included in the data base. (Manual switching operations to reenergize in-plant loads from available lines are included in the analysis of offsite power recovery actions, and these partial power failures are applicable to the initiating event data base.) Events attributed to single transformer failures and in-plant automatic transfer failures are not included in the data, because these events are analyzed within the plant power system model. The PLG data has been developed on a plant site basis, rather than a reactor unit basis, and notes are provided to describe the unit operating conditions when the power failure occurred.

A two-stage Bayesian update was performed using the data from Table 1. The Zion data was excluded from the plant population for the first step of the analysis. The plant population data was applied using the total number of offsite power failure events for each site and the total number of site years listed in Table 1 (i.e., not accounting for the effects of unit availability). The generic data was updated using the Zion site specific evidence of no failures in 9 site years. The resulting distribution provides the specialized calendar year frequency of loss of offsite power events at the Zion site, regardless of unit operating conditions. This distribution was then multiplied by the average Zion unit availability (0.71) to obtain the frequency of losses of offsite power to a Zion unit during power operation. The parameters of this updated and scaled distribution are

5th Percentile:	1.04×10^{-2}	failure/unit operating year
Median:	3.63×10^{-2}	failure/unit operating year
95th Percentile:	1.27×10^{-1}	failure/unit operating year
Mean:	4.85×10^{-2}	failure/unit operating year

The mean frequency for the loss of offsite power at a generic plant site from the given population, excluding the Zion evidence, was determined to be 0.194 events per site calendar year. The updated mean frequency for the loss of offsite power at the Zion site is 0.068 events per site calendar year.

There are several reasons for the differences between this distribution and the point estimates cited by the reviewers. One obvious source is the significant difference between the PLG data for losses of offsite power and the data developed by Scholl. The revised data base is also somewhat different from the original data used in the Zion Study which was obtained from EPRI NP-801. Therefore, our updated frequency for power failures per site calendar year is different from the value calculated using the EPRI data and correctly accounting for the effects of multiple reactor sites. Finally, since this data is applied to a specific Zion Probabilistic Safety Study transient event tree which models a unit trip from power operation, the site data has been scaled to account for the average Zion unit availability to develop the frequency of offsite power failures per reactor operating year.

Offsite Power Recovery Time Data

The reviewers' comments regarding the time for offsite power recovery are also based on the data collected by R. F. Scholl (Reference 1). It is noted that for 39 instances of total loss of offsite power for which the time to partial restoration of power was given, Scholl's data provides the following information.

- o Frequency of failure to recover power within 30 minutes: 0.41.
- o Frequency of failure to recover power within 60 minutes: 0.26.

The revised PLG summary of loss of offsite power events identifies a total of 58 events at all U.S. plant sites that have resulted in total power failure. As noted in the preceding section, these events do not include failures of single transformers or failures of in-plant power transfer circuits, since these failures were analyzed within the Zion plant electric power system model and no recovery was applied to power failure scenarios resulting from them. Power restoration time information was available for 42 of the 58 offsite power failure events (which includes 22 of the 34 Zion data base events summarized in Table 1). The offsite power recovery time data from these events is summarized below:

	Frequency of Failure to Recover Power	
	Within 30 Minutes	Within 60 Minutes
Overall (42 events)	.50	.38
Revised Zion data base (22 events)	.50	.41

This data seems to strongly reinforce the reviewers' observation that the Zion recovery time analysis is very optimistic compared with the generic data for offsite power restoration. However, there is also a significant amount of evidence available from the generic data to indicate that the

Zion power recovery time should be much lower than the plant population. Most of the sites experiencing extended power outages are located in areas subject to regional grid instability problems (e.g., St. Lucie and Turkey Point), unique supply line routing (e.g., Pilgrim), or have experienced localized outages directly attributable to some characteristic of the site (e.g., Millstone and Palisades). Furthermore, the Zion recovery model has been developed in response to a specific set of plant conditions which could have an important effect on the relative priorities assigned to offsite power restoration and, if they were to occur, could significantly reduce the recovery times reported for some sites.

The Zion Probabilistic Safety Study power recovery analysis is applied to plant conditions in which at least two of the three diesel generators for the affected unit are failed and no core cooling equipment is available (e.g., the turbine drive auxiliary feedwater pump has also failed). This situation is much different from all the events summarized in the generic data base, because in virtually all cases, onsite power was available from one or more diesel generators, and there was no immediate concern for loss of core cooling or loss of reactor coolant inventory. Of the 42 events for which recovery time data is available, 10 involved partial failures of onsite power sources or onsite switching failures which left one or more buses deenergized. Offsite power was restored to at least one bus within 30 minutes in all but one of these cases. The implications of this comparison are perhaps somewhat misleading, since none of these events involved significant offsite grid disturbances or widespread storm damage. However, in some cases involving regional power failures, there are indications in the reports that partial offsite power service was available and could have been connected to the plant if the onsite equipment had malfunctioned. The reasons for not reconnecting offsite power more quickly in these cases are not given in the reports.

As was noted in our discussion of the offsite power failure frequency data, the development and application of generic data must be consistent with its use and interpretation in the plant models. Since there has never been an instance of offsite power failure, severe degradation of onsite power supplies, and loss of core cooling capability, none of the generic data is directly applicable to the power recovery scenarios for the Zion Probabilistic Safety Study. The generic plant operating experience provided an important information input to the development of the Zion power recovery model. However, it was also recognized that many of the generic plant sites have local transmission grid and switchyard configurations which are much different from those at Zion and which could significantly affect the time required to restore power after a major disturbance.

The Zion site is somewhat unique among the population of plants because its switchyard is an important intertie point for the Commonwealth Edison transmission grid. A total of six transmission circuits are interconnected through a ring bus, and the transmission line rights of way diverge geographically soon after the lines leave the switchyard. Two of the lines connect to a neighboring utility in Wisconsin. It is also important to note that the Zion switchyard is directly adjacent to the plant buildings and that the switchyard equipment is normally operated by on-shift plant personnel.

The Zion offsite power recovery time distribution has two primary components, both of which are based on an analysis of the Zion plant site and evaluation of Commonwealth Edison transmission system experience. As with many scenarios modeled in the study, it was felt that this type of analysis would provide a more realistic estimate of expected performance than could be obtained from the evaluation of rather sketchy generic data from sites which are significantly different from Zion, under conditions very different from those being modeled in the study. The operator response time model accounts for normal shift manning, conflicting concerns about the failed diesel generators and the restoration of normal power, standard plant practice for switchyard operations, and expected operator performance under conditions not covered by specific procedures but involving operations familiar to plant personnel. It is conservative to the extent that it models the diesel generator trouble investigation and switchyard response actions as purely sequential events for a single operator when, in fact, it is possible that operators could respond to each location independently.

The offsite line restoration distribution presented in Secion 1.3.2.2 (page 1.3-15) is based on the evaluation of approximately 15 years of transmission line forced outage data for the entire Commonwealth Edison system (Reference 6). The line recovery distribution used in the study was developed from a model for the Zion site which treated the six transmission circuits as being equivalent to three pairs of totally coupled lines because of their routing and termination after leaving the site. A slight coupling was included between the two pairs of lines connecting to Commonwealth Edison facilities. The lines connecting to Wisconsin were treated as being essentially independent electrically from the other four circuits because of the intertie load shedding facilities which quickly disconnect these circuits from the Commonwealth Edison grid if instabilities develop in either of the utilities' transmission networks. Although the area is prone to severe thunderstorms, freezing rain, and heavy snowfalls, there is no evidence from the Commonwealth Edison records to indicate a significant frequency of physical damage to all of these circuits. (The 10^{-4} frequency for failure to recover power within 8 hours accounts for the fact that there have been no extended multiple line or transmission facility outages in the Commonwealth Edison system in more than 1,100 forced outage events due to all causes. Even though a series of severe tornadoes disabled five redundant transmission lines supplying the Dresden site on November 19, 1965, offsite power was restored to the site from at least one line within 4 hours after the initial power failure.)

We believe that the line recovery distribution used in the study provides a realistic model of the expected time to restore power from at least one of the six circuits at Zion and that the application of simply derived generic data is not relevant to the scenarios being modeled.

REVIEW OF SCHOLL OFFSITE POWER DATA

General Comments on the Scholl Report

We obtained a copy of Raymond F. Scholl's report on offsite power failures (Reference 1) and have completed a detailed review of the report and of the reviewers' use of Scholl's data. Scholl's work is one of the most comprehensive summaries of events affecting offsite power supplies that we have reviewed. His data was obtained from LERs and from the responses to information requests sent to all licensed operating plants. (A total of nine units at six sites did not respond to his requests, and only LER data was used for those units.) His summaries differentiate between partial and total power failure events and identify the specific cause of each failure where information was available. Since his raw data is simply a copy of a computer listing of events indexed according to plant docket number, type of failure (total or partial), and event date, it is difficult to correlate his data base to actual event descriptions. However, we were able to identify nearly all the events in the total power failure category, and we appreciate the thorough documentation provided in the report. Too few data bases provide this level of traceability, and Mr. Scholl's work is an important contribution to the effort needed to develop a comprehensive catalog of these events.

Unfortunately, although Mr. Scholl has accomplished an important data collection and categorization task, we disagree strongly with his methods for data reduction, and we see little practical use for the numerical results presented in the report summary and in Appendix B. Perhaps one of the most glaring deficiencies in Mr. Scholl's analysis of event frequencies is his treatment of the plant "age," or the number of years to which his failure data is applied. According to his definition, the plant "age" is based on the date of its first reported loss of offsite power through June 3, 1980. Therefore, since the first reported loss of power at Zion occurred on March 12, 1979 (this event is discussed below), the "age" of each Zion unit is assumed to be 1.23 years. Plants such as

Crystal River 3 and Duane Arnold reported no failures of offsite power and, therefore, their "ages" were set equal to zero (i.e., they were excluded from the data base). This is obviously an extremely biased treatment of the plant population success data, and the use of these "ages" to determine site specific and generic power failure frequencies is a gross misapplication of statistical analysis. It is not surprising that the study results obtain very pessimistic estimates of these failure frequencies, since the numerical methodology has precluded the correct accounting for many years of plant operations without power failures.

We have reviewed in detail only the basic event data for total losses of offsite power because these are the events of interest in the Zion data base. However, we note that Mr. Scholl included one partial loss of offsite power at both Zion units from an event on March 12, 1979. This event occurred with Zion Unit 2 in cold shutdown for refueling and Zion Unit 1 at power operation. Diesel generator 1B was out of service for maintenance. The reserve feed breaker from the Unit 2 offsite power source to the Unit 1 essential 4 kV buses was inadvertently removed from service for relay testing. This action administratively violated the technical specifications requirement for continuous operability of two sources of offsite power to Unit 1 during diesel generator maintenance. The action had no effect on bus voltages and would not have affected the automatic supply of offsite power following a Unit 1 trip, because the breaker in question can only be closed manually. Since the event was strictly an administrative oversight reportable under the plant's licensing criteria, it is difficult to justify its inclusion as an actual loss of power when developing data to be applied in estimating the frequency of power failure events. As noted, we did not review the partial power failure data in any detail, except for this single event. Therefore, we have no way of determining how many other events in this category could be of a similar nature, and we would certainly not apply the data without careful review.

Loss of Offsite Power Event Frequency Data

We attempted to verify each of the 109 total loss of offsite power events in Mr. Scholl's report. Unfortunately, 4 of the events were inadvertently truncated during reproduction of his computer listings for the report and were not included in our copy. Of the remaining 105 events, we were able to identify 94 by correlating the unit and event date with our own data base information (References 3, 4 and 5). In some cases, we also contacted the utility to obtain information about events for which we could find no written documentation. Of the 11 events which we could not trace, we suspect that at least 3 may be double accounting for events which occurred at the same unit within one or two days of the event in question, although this cannot be verified from the available information.

In general, there is an excellent correlation between Scholl's basic event data and the information used to develop the revised PLG data base discussed in item 1. Scholl's data includes at least 11 events which were due to auxiliary power transformer failure or failures of in-plant switching circuits. As noted in item 1, these types of events are analyzed as part of the Zion plant electric power system model and should be excluded from the loss of offsite power initiating event data base for the Zion Probabilistic Safety Study. This is simply a matter involving review of the data to ensure that it is compatible with the plant model. The transformer failures did result in power outages and, as such, they should be included in Scholl's data base. However, to avoid double accounting, they should be removed from the data applied to the Zion models. Since Scholl did not develop his data base specifically for use in the Zion Probabilistic Safety Study, it is not surprising that his data is not precisely compatible with the Zion models. However, because it was not developed for this purpose, it should not be broadly used as an authoritative source without first carefully examining its applicability to the study.

We are not going to discuss each of Scholl's data entries in this response, but one practice deserves some attention. In some cases, there is an obvious multiple accounting of single power failure events. We cannot determine the reasons for this practice, but we believe it may be related to Scholl's use of the data for cause and duration of specific events, and it may have resulted from the nature of his questionnaire responses. However, this multiple accounting has a significant effect on the reported frequencies of power failure at selected sites, and it provides totally misleading results for the application made in the Zion Probabilistic Safety Study. The most extreme example of this accounting practice involves the loss of all offsite power to Millstone Units 1 and 2 on August 10, 1976. The passage of Hurricane Belle caused severe salt spray coating of the entire Millstone switchyard, and several insulator flashovers resulted in the loss of offsite power to both units. Scholl's data base accounts for eight separate losses of offsite power from this single event (seven for Unit 1 and one for Unit 2). His event times indicate several partial restorations of power in rapid succession, but this event should be included as a single functional loss of power to the entire site. The other instances of this multiple accounting are events at Point Beach on October 13, 1973 (listed as two events for Unit 1 and one event for Unit 2), Indian Point on July 13, 1977 (listed twice for Unit 3), Beaver Valley on July 28, 1978 (listed twice), and San Onofre on April 22, 1980 (listed twice).

Scholl has reported his data on a reactor unit basis. As noted by some reviewers, it is more meaningful to index this data to the plant site, regardless of the number of operating reactors. A single loss of offsite power at a three-unit site is counted in the Scholl report as three separate power failures, one for each of the reactors. This accounting practice results in a very large total number of power failure events and explains most of the differences between Scholl's reported 109 total losses of offsite power and the 58 events included in the new PLG data base for all plant sites. Scholl's accounting could result in consistent frequencies for the loss of offsite power per plant site if the "ages" of the reactors at each site were correctly assessed. Unfortunately, his methodology for determining the plant "age" results in different success data for each unit at a site and makes it extremely difficult to assess a meaningful composite power failure frequency for the site.

TABLE 1

PLG REVISED LOSS OF OFFSITE POWER DATA BASE

	Site	Years*	Total Events**	Notes[β]
1.	Yankee Rowe	21	1	15
2.	Indian Point	19	5	3
3.	San Onofre	15	1	4
4.	Connecticut Yankee	14	4	5
5.	R. E. Ginna	12	1	6
6.	H. B. Robinson	11	0	
7.	Point Beach	11	0	
8.	Palisades	11	4	7
9.	Maine Yankee	9	1	16
10.	Surry	9	0	
11.	Oconee	9	0	
12.	Fort Calhoun	8	1	8
13.	Kewaunee	8	0	
14.	Arkansas One	7	2	9
15.	Three Mile Island	8	0	
16.	Calvert Cliffs	7	2	10
17.	Trojan	6	0	
18.	Millstone	11	1	11
19.	D. C. Cook	7	2	12
20.	Prairie Island	8	1	13
21.	Turkey Point	9	8	14
22.	Zion	9	0	
	Total	229	34	

*See note 1.
**See note 2.
[β]See following page.

Notes for Table 1

1. Time from "Date of Initial Criticality" listed in the NRC "Grey Book" summaries (NUREG-0020) through December 31, 1981, rounded to the nearest whole year.

2. A loss of offsite power event causes failure of all offsite power to a unit or failure of all automatically available offsite sources. Single transformer failures and failures of in-plant breaker transfer circuits are excluded.

3. Indian Point events are:

11/9/65:	Unit 1 shutdown (Northeast blackout).
12/70:	Unit 1 at cold shutdown.
7/20/72:	Unit 1 tripped from power operation.
7/12/77:	Unit 2 at cold shutdown; Unit 3 at 91% power.
6/3/80:	Unit 2 at 100% power; Unit 3 remained at power operation through the event, although the diesel generators were required to operate.

4. San Onofre 1 was at cold shutdown.

5. Connecticut Yankee events are:

7/27/68:	Unit operating condition not specified.
7/15/69:	Unit at 50% power.
7/19/74:	Unit operating condition not specified.
6/26/76:	Unit at cold shutdown.

6. Ginna tripped from power operation.

Notes for Table 1 (continued)

7. Palisades events are:

    9/2/71: Unit at cold shutdown.
    7/24/77: Unit at 100% power.
    11/25/77: Unit at 85% power.
    12/11/77: Unit at 100% power.

8. Fort Calhoun tripped from 98% power.

9. Arkansas One events are:

    4/8/80: Unit 1 operating condition not specified; Unit 2 shutdown.
    6/24/80: Unit 1 at 100% power; Unit 2 at 91% power.

10. One event which occurred on 12/20/73, during construction is not included in the data base for Calvert Cliffs. The other events are:

    4/11/78: Unit 1 at approximately 80% power; Unit 2 at approximately 75% power.
    4/13/78: Unit 1 at approximately 80% power; Unit 2 at approximately 50% power.

11. Millstone Unit 1 at 45% power; Unit 2 at 100% power. The event occurred during Hurricane Belle. Severe salt spray coated the switchyard and caused several failures due to insulator flashover.

12. D. C. Cook events are:

    2/1/75: Unit at 6% power (testing).
    9/1/77: Unit at 100% power.

13. Prairie Island Unit 1 at cold shutdown; Unit 2 at 100% power.

Notes for Table 1 (continued)

14. Turkey Point events are:

4/3/73:	Unit 3 at 73% power; Unit 4 under construction (testing).
4/4/73:	Unit 3 at 73% power; Unit 4 under construction (testing).
03/74:	Unit 3 at 73% power; Unit 4 at 60% power.
4/25/74:	Unit 3 at 97% power; Unit 4 at 98% power.
6/28/74:	Both units tripped from power operation.
5/16/77:	Unit 3 at 100% power; Unit 4 at cold shutdown.
4/4/79:	Unit 3 at shutdown; Unit 4 at approximately 90% power.
8/3/79:	Unit 3 at 100% power; Unit 4 at 100% power.

15. Yankee Rowe was shutdown. The event date is 11/9/65 (Northeast blackout).

16. Maine Yankee tripped from 47% power.

REFERENCES

1. Scholl, R. F., Jr., "Loss of Offsite Power, Survey Status Report, Revision 3," Report of the Systematic Evaluation Program Branch, Division of Licensing, U.S. Nuclear Regulatory Commission.

2. Electric Power Research Institute, "ATWS: A Reappraisal, Part III Frequency of Anticipated Transients," EPRI NP-801, July 1978.

3. Private sort of LER summaries from 1965 through April 30, 1981.

4. Battle, R. E., F. H. Clark, and T. W. Reddoch, "Review of Nuclear Power Plant Offsite Power Source Reliability and Related Recommended Changes to the NRC Rules and Regulations," NUREG/CR-1464, Appendix A, May 1980.

5. Petroleum Information Corporation, "Nuclear Power Experience," 1982.

6. Landgren, G. L., letter to G. T. Klopp, "Information Concerning the Commonwealth Edison Company's 345 kV Transmission Line Performance," February 18, 1981.

Systems Analysis

12. One area apparently not addressed concerning the "station blackout" dominant sequence was the probability and effects of DC power loss due to battery power depletion during an extended loss of AC power. Provide your analysis of this scenario to include:

   a. What is the battery life (in hours) using procedures currently available or proposed at Zion?

   b. What instrumentation and controls, particularly affecting AFWS operation, would be available, if DC bus 011-1 could be energized from Unit 2?

   c. Can the steam train AFWS continue to be operated following loss of AC and DC power? How would it be controlled if steam generator level indication is not available due to loss of DC power? Are there procedures to do this?

   d. Would the loss of DC power (before AC can be restored) affect the recovery of power to the plant; such as (1) affecting the ability to restore offsite power? (2) starting diesels without field flashing? 93) affect on plant sevurity systems os as to possibly hinder access to areas of the plant?

Response

A detailed quantitative analysis of the effects of DC power loss due to battery depletion during an extended loss of AC power was not included in the Zion Probabilistic Safety Study, and this analysis cannot be completed within the time contraints imposed by this response. However, a qualitative assessment of long term DC power availability can be made by examining the Zion DC power system configuration and elements of the offsite and onsite AC power failure and recovery models.

As shown in Zion Probabilistic Safety Study Figure 1.5.2.2.1-4 (page 1.5-272), each DC bus at Zion can be cross-connected to the corresponding DC bus for the opposite unit (i.e., buses 111 and 211, buses 112 and 212, buses 011-1 and 211, and buses 011-2 and 111). Each battery charger has sufficient capacity to supply the loads from both DC power divisions and, in fact, this capability is demonstrated when the buses crosstied during the monthly battery charging operations. (Study Section 1.3.4.13.5, page 1.3-312, provides a discussion of these operations and their treatment in the plant event models.) Each battery charger is supplied from a 480V AC essential power bus.

The loss of offsite power initiating event was modeled as failing the offsite power supply to both Zion units, requiring all five diesel generators to start and reenergize the onsite loads. The response of swing diesel generator "0" was analyzed in particular detail for this event to account for the fact that it can supply power to only one of the units at a time. The "station blackout" frequency distribution used in the plant event tree model evaluates the loss of all AC power to one unit, given a loss of offsite power. Under these conditions, it is very likely that one or more of the diesel generators for the other unit are operating normally to supply their associated loads, including the battery charger.

During extended power outages, many actions could be taken to recover onsite AC power an to increase the availability of DC power. These actions include restarting or recovery of failed diesel generators, selective shedding of nonessential DC loads such as turbine emergency oil pumps, feedwater pump emergency oil pumps, etc., and use of the DC bus crossties to supply two DC power divisions from an energized battery charger. None of these alternatives were evaluated in the Zion Probabilistic Safety Study models for the "station blackout" scenarios, and they could have a significant effect on the frequency of DC power loss due to battery depletion.

However, even without a detailed quantitative analysis of these AC and DC power recovery options, it is possible to address many of the reviewers' concerns about long term DC power failure and to place the Zion Probabilistic Safety Study analyses in their proper perspective.

a. The Zion station batteries are each rated at 996 ampere-hours. An 8-hour discharge test is performed each refueling outage to verify this capacity. No detailed information is available to support a specific expected battery life under "station blackout" conditions. The Zion emergency operating procedures for loss of offsite power specify that the major nonessential loads supplied by the batteries (e.g., the turbine emergency oil pump, air side seal oil backup pump, and the main feedwater pump emergency oil pumps) should be disconnected as soon as practical to conserve DC power for essential instrumentation and controls. Based on analyses performed for similar plants, it is expected that the Zion batteries would supply power for at least 2 to 4 hours if all AC power were lost. This time could be extended if additional nonessential loads were shed. However, verification or detailed quantitative evaluation of this time under realistic loading conditions is beyond the scope of this response.

b. Steam generator level and auxiliary feedwater flow are indicated in the main control room. The instrumentation for each steam generator is powered from a separate 120V AC instrument power bus. These instrument buses are connected to the AC and DC power distribution systems as shown in Zion Probabilistic Safety Study Figure 1.5.2.2.1-2 (p. 1.5-270). If all AC power were lost for a given unit, the DC batteries would supply instrument power as indicated (e.g., battery 111 would supply instrument bus 112, battery 112 would supply instrument buses 113 and 114, and battery 011 would supply instrument bus 111). Use of the DC bus crossties as discussed in the first section of this response could maintain the DC buses and the AC instrument buses energized from the battery chargers for the opposite unit. For example, if DC bus 011-1 were energized from Unit 2 (i.e., by closing the crosstie to DC bus 211), steam generator level and auxiliary feedwater flow indication would be available for steam generator 1A. Other crosstie arrangements could provide similar instrumentation for all four steam generators. Since the auxiliary feedwater flow control valves are AC motor-operated, these actions would not allow direct control of auxiliary feedwater flow from the main control room. However, if necessary, the valves could be operated locally at the direction of control room personnel. It should also be noted that redundant steam generator level and auxiliary feedwater flow indication (powered from the same sources as the control room indication) is available at the remote shutdown panels, which are located in close proximity to the auxiliary feedwater pumps on elevation 579' of the auxiliary building.

c. Refer to Zion Probabilistic Safety Study Section 1.5.2.3.9 (p. 1.5-681ff) for a description of the auxiliary feedwater system. The motor-operated main steam supply valves for the turbine-driven auxiliary feedwater pump are normally open, and the motor-operated auxiliary feedwater flow control valves are normally open at a preset throttle position. To start the turbine and supply flow from the turbine-driven pump, air-operated steam inlet valve FCV MS-57 must be opened. This valve fails to the open position on loss of air pressure or loss of DC power. Therefore, the turbine-driven pump would start automatically during scenarios involving loss of AC power or loss of DC power. The turbine is controlled by a mechanical governor which does not require AC or DC power. Thus, once started, the turbine would continue to run at a constant speed and would supply flow to all four steam generators as limited by the settings of the motor-operated flow control valves.

A detailed analysis of auxiliary feedwater flow control in scenarios for which no steam generator level or flow indication is available is beyond the scope of this response. As noted above, several alternatives are available for restoration of DC power to instrumentation for one or more steam generators for the affected unit. The auxiliary feedwater flow control valves may be operated manually to control flow from the turbine-driven pump. As long as level or flow indication was available for at least one steam generator, the operators could positively control level and flow for that steam generator. Setting all four flow control valves at the same position would provide reasonable assurance of similar conditions in all four steam generators. The operators could also simply leave the valves at their originally throttled positions for a relatively long period of time (hours) without fear of overfilling the steam generators. Accurate flow control during a very long term operation (several hours) could be achieved by performing heat balances based on reactor decay heat levels, steam generator atmospheric steam relief, and local measurement of condensate storage tank level to determine the total auxiliary feedwater flow rate.

These options were not assessed in the Zion Probabilistic Safety Study analysis. Zion plant emergency procedures address the operation and control of the turbine-driven auxiliary feedwater pump for situations involving total loss of AC power, but they do not provide any guidance for system control if all AC and DC power is lost. Because of the DC bus crosstie capability and the likelihood that one or more diesel generators will be supplying AC power to the unaffected unit, we believe that the operators will have at least partial instrumentation available for steam generator level and auxiliary feedwater flow, even if all the affected unit's batteries were to be depleted. This instrumentation would allow positive control of auxiliary feedwater flow by manual operation of the flow control valves.

d. Loss of all DC power before AC power was restored would affect the recovery of power to the onsite loads. The longest recovery time period applied in the Zion Probabilistic Safety Study models was 3 hours (i.e., if power had not been restored from the offsite grid within 3 hours, it was assumed to remain unavailable for the duration of the analysis period). Based on the selective shedding of nonessential loads, it is expected that DC power would remain available for switching operations for more than 3 hours following the loss of all AC power. The DC bus crosstie capability discussed previously in this response could also be used for switching control power.

(1) If all DC control power failed before offsite AC power were restored, the reconnection of the onsite buses to the offsite supply would require local manual (mechanical) circuit breaker operations and could significantly extend the time to restore power to the plant loads. However, because the maximum electric power recovery analysis period applied in the study was only 3 hours and because the batteries are expected to be capable of supplying switching control power for much longer than this period, the quantitative evaluation of battery power depletion after 3 hours would not affect the offsite power recovery results used in the study.

(2) Restarting or repair of failed diesel generators was not included in the Zion electric power recovery analysis. Therefore, the effects of DC power failures on diesel generator recovery have no effect on the analysis results.

(3) All Zion plant operators have keys to provide emergency access to security areas if the electrical security locks malfunction.

Systems Analysis

13. For anticipated transients without scram (ATWS), there is a sequence analyzed in the German Risk Study (see EPRI-NP-1804-SR, p. 5-32) where a pressurizer safety valve fails open, after a loss of main feedwater transient followed by failure of reactor trip. This sequence was assumed to lead to core melt, in the German Risk Study. The German Risk Study used a probability of 2.5×10^{-2} per demand for failure of pressurizer safety valve to reclose. With a mean frequency of loss of main feedwater of 5.2/year, a mean frequency of 3.7/year for turbine trip, a probability of 1.8×10^{-4} for failure of reactor trip, and a probability of .075 that one of the three pressurizer safety valves ticks open, one obtains

$$(5.2 + 3.7)\ (1.8 \times 10^{-4})\ (.075)/\text{year} = 1.2 \times 10^{-4}/\text{year}$$

for the probability of this sequence. IF THIS SEQUENCE WAS OMITTED BECAUSE IT DOES NOT LEAD TO CORE MELT, PLEASE GIVE SUPPORTING EVIDENCE.

Response

In our revised ATWS analysis [see the response to ACRS Question III.3(1)], failure of the pressurizer safety valve to close is explicitly modeled. There is no reason to expect this LOCA sequence to lead to a core melt unless no makeup is supplied. Many alternatives for automatic or manual actuation of makeup are available. First, normal charging is in operation and will supply high pressure water. Letdown should isolate and, when the VCT level falls, suction would automatically shift to the RWST. Thus not only providing makeup but also boration. Second, the operator will be alerted to the LOCA condition by increasing level in the PRT and increasing containment pressure. He has sufficient time to recognize the condition and initiate safety injection. As long as RCS pressure remains high, only the charging pumps have sufficient head to supply water. Third, although not explicitly analyzed in WCAP-8330, by combining the results of the accidental depressurization and loss of feedwater ATWS events, in this WCAP, RCS pressure and temperature trends can be inferred. It appears likely that the RCS will depressurize in much less than 10 minutes. If so, an automatic safety injection signal will be generated and the SI pumps could also supply sufficient makeup. Thus it is absured to assume guaranteed core melt for this sequence.

Systems Analysis

14. For a Small LOCA the ZPSS study (see p. 1.3-113 of ZPSS) asserts that success for High Head Injection requires delivery of water from one-out-of-two high head safety injection pumps or delivery of water from one-out-of-two charging pumps to the reactor coolant system. However, the Offshore Power Systems report (An Evaluation of the Residual Risk from the Indian Point and Zion Nuclear Power Plants, Report No. 36A75, February 1980) assumed that two out of the four pump set consisting of the two high head safety injection pumps and the two charging pumps were required. Please present evidence for the assertion that only one of four pumps is required for a small LOCA.

Response

Analytical studies performed for the Westinghouse owner's Group after TMI and documented in WCAP's 9600 and 9753 as well as the Westinghouse response to NUREG-0578 (Section 2.1.9) indicate that the flow from one hi-head safety injection pump or one charging pump is sufficient to prevent core damage. It should be noted that core top uncovery is predicted to occur for the larger (~ 2 inch) small breaks but that the maximum fuel rod clad temperature will not exceed apprixmately 1300°F.

Systems Analysis

15. In the Steam Generator Tube Rupture Event Tree (p. 1.3-130 of the ZPSS) an operator-action event OP-5 is considered. It is remarked on p. 1.3-42 of the ZPSS that "the operator could do nothing for a long period of time. In this mode water/steam would eventually pass through the steam generator relief valves". However, main steam lines are, generally speaking, not designed to take water loadings and may fail. A failure of a main steam line may result in sudden flashing of the water in the affected steam generator. The sudden cooldown of the reactor vessel may threaten reactor vessel integrity. JUSTIFY THE OMISSION OF THIS ACCIDENT SEQUENCE.

Response

The main steam lines at Zion can withstand a flooded condition. They were flooded for flushing purposes during construction cleanup. It is, furthermore, highly unlikely that they would be flooded given a water-steam mix flowing through the relief valves.

Systems Analysis

16. In the Steam Generator Tube Rupture event tree, failure of the High Pressure Injection System (HPIS) is considered only as a part of the OP-5 event. Failure of HPIS after a steam generator tube rupture is not assumed to lead to core melt, if no other failures occur. JUSTIFY THIS ASSUMPTION. Also, provide justification for assuming depressurization capability of AFW if failure of OP-5 occurs.

Response

Once the operator depressurizes the reactor coolant system to a pressure below the steam generator relief and safety valve setpoints, loss of RCS inventory has stopped. Although a two-phase condition may exist, continued cooling via the steam generators is possible (Reference Chapter 3, of the Zion Study, WCAP-9744, and WCAP-9601). Please also note that the reflux mode of heat transfer is discussed extensively in WCAP.

17. After a LOCA, in the recirculation mode of cooling the core, the residual heat removal pumps take suction from the containment sump. There are motor-operated valves (SI8811A and SI8811B) in the lines from the containment sump to the RHR pumps. The ZPSS used a value of 1.55×10^{-3} per demand for the mean failure frequency for these valves (failure mode: fail to open). The failure frequency is supposed to include failures in the local control circuitry for these valves (see p. 1.5-496 of ZPSS). However, the Offshore Power System study, following the Reactor Safety Study, used a value of .03 for the failure frequency of the local control circuitry of these valves. Moreover, common mode failures of the local control circuitry for these two valves may be of importance. EXPLAIN THE DIFFERENCES IN FAILURE FREQUENCY FOR THESE VALVES, AS CALCULATED BY OFFSHORE POWER SYSTEMS USING REACTOR SAFETY STUDY DATA AND AS GIVEN IN ZPSS. MOREOVER, ADDRESS THE PROBLEM OF COMMON MODE FAILURE OF THE LOCAL CONTROL CIRCUITRY FOR THESE VALVES.

Response

The OPS frequency of failure on demand, 0.03, is obtained using Reactor Safety Study (RSS) data. A review of the RSS was made to determine the source of this frequency of failure.

In Appendix II, Section 5.9 (Low Pressure Recirculation System or LPRS) of the RSS, the frequency of failure for an MOV that must change position is developed. The sum of the failures for this MOV is 3.3×10^{-2}, but this sum includes a check valve failure to open and the MOV breaker's failing to close. As MOV breakers are usually closed, removing the breaker failure and the check valve failure leaves a sum of 3.2×10^{-2} which appears to agree with the referenced value.

The LPRS value from the RSS (3.2×10^{-2}) is developed in the fault tree presented in Figure II 5-65 and quantified in Table II 5-32. A similar frequency of failure (1.9×10^{-2}) for an MOV is used in Section 5.6.4

[high pressure injection system (HPIS)], Appendix II, of the RSS. This is developed in the fault tree presented in Figure II 5-45 (Sheet 3) (MOV 1) and quantified in Table II 5-23. Both values are dominated by hourly failure rates for undetected failures based upon annual cycling of the valves (note, Zion valves are tested quarterly). Furthermore, these values do not agree with MOV failure rates on demand presented in other system analyses of Appendix II of the RSS (for example the RSIS and the HPRS) (1×10^{-3}) and in fact do not agree with the guidance presented in Appendices III and IV, Section 4.1.2 of the RSS. That section states that the failure of a valve to operate includes changing state from closed to open or open to closed. Section 4.1.4 states that "Available experience data do not permit separation of motor failure from pump failure. Therefore, separate motor failure rates for pump and valve drive motors should not be included." Similar statements are included in Sections 4.1.5, 4.1.6, and 4.1.11. In Table III 4.1, the failure of MOVs to operate on demand, Q_d, includes the driver, but does not include input control signals (such as the SICS signal). Finally, in Section 3.3 of Appendix III to the RSS, the failures presented for motor-operated valves include those failures typically associated with the motor control circuit (limit switches, torque switches, motors, contacts, etc.).

A review of other data sources indicates that the failures included in the HPIS MOV development and the LPRS MOV development are included in the frequency of MOV failure on demand. Therefore, requantifying these failures and summing with data that already includes such failures overestimates the frequency of failure of a single MOV on demand by as much as an order of magnitude.

The Zion MOV failure data used for updating the generic error distribution included as failures all control circuit failures (limit switches, torque switches, control switches, motors, contacts, etc.). For these reasons, the failure frequency of MOVs as presented in the ZPSS is more appropriate than the value presented in the OPS study and the RSS.

DEPENDENT FAILURES

To quantify the effect of dependent failures of motor-operated valves (SI8811A and SI8811B) in the lines from the containment sump to RHR pumps, OPS uses a value of 0.15 for the β factor. This number is based on the estimate in a BNL document in which the value of β was obtained by modifying the estimate of Fleming (General Atomic Report GA-A13284). The latter gave a β value of 0.23 for all valves. It also observed that the variation of β for a number of diverse equipment types had a fairly limited range (from about 0.1 to about 0.2). It was argued that since the failures considered in the model did not include control circuit failures, a number smaller than 0.23 was more apppropriate. Therefore, it was decided to use an average value of 0.15 for pumps and valves.

The mean value estimate (β = 0.014) used in the ZPSS was also arrived at subjectively. It basically represented the analyst's opinion that a smaller value was more representative of the type of dependencies modeled by β factor in the recirculation system analysis. Such dependencies did not, for instance, include those resulting from acts of test and maintenance mainly because such failures are treated explicitly in Zion systems analyses. It was therefore believed that the β factor for the analysis represents a subclass of the dependent failures usually used to estimate β factors.

However, as can be seen from the definition of β factor, it is not clear that by excluding certain classes of failure that the value of β would decrease. Depending on the number of dependent and independent failures removed from the data base, the value of β may decrease or increase.

Neither the OPS nor the ZPSS estimates of the β factor for MOVs can be fully justified on the basis of data. A recent PLG survey of about 200 MOV failures (PWR safety systems) and classification of these failure events in a manner consistent with PLG systems analysis approach indicate that for MOVs, β is about 0.06. This value represents the mean of the posterior distribution of β where some "potential dependent failures" are also included in the evidence by means of a weighted likelihood technique.

Preliminary review of a number of other failures not currently included in the data base indicate that the above estimate is slightly conservative and that the use of expanded data would result in a smaller value.

It must be pointed out that the above value is a generic estimate in the sense that it covers MOVs in various applications and systems configuration. To obtain a β factor more appropriate for the valves in question, the data base was further scrutinized and failures not judged as being applicable were excluded. The failures excluded consisted of common cause, potential common cause, and independent failures due to common environmental conditions at the valves (the valves in question are "canned" and thus are protected from common environmental failures), and those failures that are system specific (i.e., MOV failures caused by concentrated boric acid, typically isolation valves for the boron injection tank; failures due to high differential pressure, typically the safety injection system or high head injection system; and failures due to internal pressure buildup between the valve discs of large gate valves, precluded by design at Zion). The resulting mean value of β was 2.6×10^{-2}. This result indicates that the mean value used in the ZPSS is much closer to a realistic estimate of β than all other generic estimates discussed here.

Systems Analysis

18. Provide justification for not considering events which result in overcooling and pressurization of the reactor coolant system and would threaten reactor vessel integrity. Also provide the information requested on pressurized thermal shock as listed on page 19 of this enclosure.

Response

With respect to the overall pressurized thermal shock (PTS) issue, reference (1) addressed the major concerns in this area from a probabilistic viewpoint. This report was submitted to the NRC at the end of May, 1982. Further evaluations and discussions have been held between the WOG and NRC and an NRC position and required actions will be established tentatively in August, 1982. Once this is done, more specific answers to the accompanying PTS questions could be provided. However, the numerical results submitted in reference (1) along with discussions held between Westinghouse and NRC indicate that the values assumed for vessel failure beyond ECCS capability in the ZPSS are in close agreement. Thus the contribution to overall risk from PTS in this study is not expected to change based on recent NRC discussions.

Reference (1):

"Summary of Evaluations Related to Reactor Vessel Integrity", performed for the Westinghouse Owners Group, May 1982.

Human Reliability Analysis

"A major difficulty in reviewing the human reliability analysis (HRA) in the Zion probabilitistic risk assessment (PRA) resulted from the lack of adequate documentation to permit full evaluation of assumptions and estimates of human error probabilities (HEPs) employed. Nevertheless, considering only those system failure events we selected as critical for core melt or other risk, it was possible to understand sufficiently well the Zion estimates of unrecovered HEPs to permit an appropriate sensitivity analyses. Zion estimates of response times and/or HEPs were increased to larger values (e.g., doubled or quadrupled) in accord with more conservative analysis of the impact of human performance. We determined that the larger values had no material influence on the sequence calculations for the following events: loss of all AC power with failure of AFW; and failure of LPI in a large LOCA. In the events involving ATWS (loss of main feedwater ATWS and turbine trip ATWS), we did not accept the Zion HRA. No credit for human intervention was allowed in our analysis whereas the Zion HRA did allow ample credit for human intervention. We did accept the Zion estimates of human error probabilities for failure to initiate switchover to recirculation for either a large or a small LOCA and for failure to establish and maintain cooling by feed and bleed. We also accepted the Zion estimate for failure to open MOVs CC9412A and CC9412B (the isolation valves for the component cooling flow to the RHR heat exchangers) in a large LOCA event even though they optimistically assumed that four people would have to fail. In this particular case, although we disagreed with their aspect of their HRA, we judged that other conservatism in their analysis made up for the optimistic assumption of four people."

Response

The entire ATWS analysis has been revised. See response to ACRS Question III.3(1).

"Therefore, for all of the event sequences in which human error played a significant role, we have been able to provide an independent assessment of the impact of human performance despite many questions about the assumptions and estimated HEPs used in the Zion PRA. However, there are two potentially significant classes of human error that we are now able to evaluate. These are:

1. The apparent nonconsideration of some possibilities for common cause failure from human errors during calibration and during restoration procedures of safety-related components after test, maintenance, or calibration.

2. The possibly insufficient consideration of errors (apart from common cause errors) in restoring safety components after test, maintenance, or calibration."

Response

Human Errors During Calibration and During Restoration Procedures After Calibration. Page 1.5-298 of the ZPSS discusses the potential for common cause miscalibration of a particular set of RPS instrumentation during the refueling calibration tests. This potential is recognized and assigned a frequency of occurrence of 2.66×10^{-4} per set. This effect is not dismissed as the reviewer implies. Page 1.5-298 of the RPS report states that, due to the diversity in the types of instrumentation that provide trip signals, the effect of common cause miscalibration errors is minmized. The statement in the RPS analysis that most calibration activities, even if performed in error, do not result in an instrument that fails to provide a trip expresses the actual history of RPS instrumentation failures.

Because of the instrument diversity for RPS trip signals, at least two separate types (sets) of instrumentation must fail to provide a trip signal to cause RPS failure. On page 1.5-299 of the RPS, the frequency of common cause miscalibration of two sets of instrumentation is obtained by squaring the frequency of common cause failure of a single set. The result is presented as the frequency of RPS failure due to common cause miscalibration errors. During the course of the RPS analysis, the RPS instrumentaion calibration procedures were reviewed in detail to see if procedural or other mechanisms existed that could conceivably link the procedures for different types of instrumentation. No link was found.

The effect of human errors during the monthly logic channel tests was also considered. A review of PWR experience indicated that this type of failure occurs frequently. The RPS report does not mention these failures because they cause inadvertent reactor scrams. Reactor scram is success for the RPS analysis.

The statements presented previously also apply to the human error quantification performed in the ESAS analysis.

Human errors during restoration procedures of other safety related components after test or maintenance were explicitly quantified in the systems analysis performed for the ZPSS.

Human Reliability Analysis

"To enable us to evaluate the above two areas, we need to know what kinds of rules the Zion PRA personnel used to dismiss common-cause failure from human errors and to decide that all human errors related to restoration tasks would be recovered. We further need to know what kind of tagging and administrative control (including logs and other paper work) are used at the Zion plant to ensure that recovery from such errors is highly likely."

Response

We feel that common cause failures and other human error failures related to restoration after maintenance are not dismissed as stated but explicitly modeled and quantified.

However, the general rules used to evaluate the potential human errors of concern are:

a. Detailed review of all test and surveillance procedures associated with safety related equipment. These reviews enabled the systems analysts to perform the following:

   (1) Determine there was no effect due to testing or surveillance because the system lineup is not changed.

   (2) For cases where system lineup is changed, determine the indications available after test to determine incorrect lineups; where incorrect lineup could be corrected without other operator intervention due to automatic signals etc.; and finally, to assign frequencies of occurrence to the possible human errors and to the recovery actions prior to system actuation.

   (3) In addition, for those cases where human error could result in a misaligned train, the frequency of common cause errors which result in misaligned systems (coupling of errors) was calculated and its contribution was included in the frequency of system failure.

b. Review of post-maintenance testing and return to service procedures indicated the following:

   (1) Post-maintenance checkout procedures require the operation of safety related equipment as the final check prior to returning the equipment to operational status.

   (2) For equipment in normally operating systems, this check is a full flow test to the system.

   (3) For standby systems, the test and surveillance procedures are used, and the effects of human errors associated with these procedures have already been identified.

"There are some other types of information which could assist us in evaluating your assessments of human performance:

3. What degree of practice on the simulator and how often are in-plant walk-throughs given to operators in responding to multiple problems, e.g., ATWS with loss of feedwater?"

Response

All licensed (RO and SRO) Zion operating personnel are currently scheduled to receive 5 days of simulator practice annually. The simulator training includes multiple failure events to the extent required by approved Westinghouse and Commonwealth Edison training guidelines. An emergency planning drill is held quarterly, during which plant operators, administrative personnel, and offsite response personnel simulate an accident scenario and test the emergency response network. There is presently no other regular program of detailed in-plant walk-throughs to simulate emergency conditions or multiple failure scenarios. All licensed personnel are required to review all revisions to plant abnormal and emergency operating procedures as part of their periodic retraining program. Operators are normally scheduled for 1 week of classroom training on a continuing basis every 5 weeks throughout the year (i.e., 4 weeks on shift and 1 week in training).

"4. On what basis was it decided that the STA would be involved in detailed switch manipulations related to post-accident conditions?"

Response

Zion Probabilistic Safety Study Sections 1.5.2.3.4.4.2.1.1 (page 1.5-453 ff) and 1.3.3.9 (page 1.3-39 ff) describe the operator action scenarios involving the Shift Technical Advisor (STA). The analyses in these sections do not consider the STA as being involved in "detailed switch manipulations." In fact, in the recirculation switch-over analysis (page 1.5-454) it is stated that: "The shift technical advisor would not be involved in the detail. He is supposed to form an independent interpretation of the instrumentation readout." The discussion on page 1.5-455 provides the bases for assigning a low dependence between the STA and the rest of the control room operating team.

"5. At the Zion plant, is the STA to be as available as the SE? Since the STA is an SRO, does he also function at times as an RO or as an SE?"

Response

A Shift Technical Advisor (STA) is available on all shifts at Zion. The STA normally remains in the main control room or in the immediate vicinity throughout his shift. The STA desk is located in the "center desk" area of the control room, with access to both unit control boards. The STA does not function as an RO or as an SE during his normal working schedule. However, licensed STAs may be used as necessary to substitute for other shift supervisory personnel during their off-shift hours. Because of the management structure at Zion, the STA does not function as an RO. As a licensed SRO, he does receive initial training and experience in reactor control board operations, and he receives periodic simulator retraining.

Human Reliability Analysis

"6. Was the dependence assumption stated on page 0.15-5, Volume 2 of the Zion PRA used in any calculations of common cause human error in valve restoration tasks? If yes, it is quite possible that some very optimistic judgments were made."

Response

No, it was an example for illustrative purposes only. We believe that our treatment of dependence does reflect the uncertainties in our state of knowledge and it is neither optimistic nor pessimistic.

Dependent human errors were assessed for several scenarios in the systems analyses presented in Section 1.5.2. The general dependence relations stated in Section 0.15.2 were used as a consistent basis for these assessments. The specific valve restoration distribution presented on page 0.15-5 was developed as an example of the general methodology and was not used in any of these analyses. Each analyst developed a dependent error model that was compatible with the specific testing, maintenance, or operator action scenarios for his system. Specific examples of dependent error calculations can be found in Section 1.5.2.2.1.4.6.2 (page 1.5-207) for the electric power system, Section 1.5.2.2.2.4.4 (page 1.5-298) for the reactor protection system, Section 1.5.2.2.3.4.5 (page 1.5-325) for the safeguards actuation system, and Section 1.5.2.3.4.4.2.1.1 (page 1.5-453) for the recirculation switchover function.

Human Reliability Analysis

"7. What type of psychological scaling procedure was used to derive the histograms of operator response times and response time versus probabilities of error? What were the qualifications of the developers of these procedures in the field of psychological scaling?"

Response

The histograms for operator response times used in the Zion Probabilistic Safety Study were not derived by using a set of general psychological scaling procedures. The approach adopted in the derivation of these histograms was consistent with the general methodology applied throughout the study. Each scenario was studied in detail to determine the required operator actions, available alarms and instrumentation, conflicting concerns, and a "time window" for successful response. These factors were examined in the context of personnel availability, procedural guidance, general training, and physical locations of equipment relevant to the scenario. Historical experience with similar situations at other nuclear plants and at simulators was reviewed. Finally, because the study was being performed for a specific plant with a well trained staff and several years of operating experience, Zion station and Commonwealth Edison operating, supervisory, and engineering personnel were consulted for their expert opinions relating to the scenarios. This information was then used to develop the histograms used in the study, which we believe are the best format for expressing the available state of knowledge regarding these scenarios.

Estimation Methodology

1. What is the basis for claiming the assumed prior distributions of the Zion component failure rates and probabilities are "frequency distributions" - the "known results of experiments on populations"? If this assumption is false, how reliable are your estimates?

Response

There is misunderstanding here; we do not make the claim alleged. What we actually did was to interpret the distributions given in WASH-1400 as the WASH-1400 team's estimates of frequency distributions, i.e., population variability curves. We believe that is the way the WASH-1400 authors intended them. If we were to accept them as the "true" frequency distributions, we would have used them as prior probability distributions. In most cases, we did not accept them this way but rather expressed our own prior probability distributions using the WASH-1400 team's curves as "expert opinion" information input to us. In most cases, our prior distributions were broader than the WASH-1400 curves, reflecting more uncertainty in our minds than in theirs. This is a typical way to treat expert opinion.

Estimation Methodology

2. What is the rationale by which some WASH-1400 5/95 percentiles were treated on 20/80, for Zion, while others were accepted as 5/95? How sensitive are key results to these choices? (See ACRS question II.2(1)).

Response

See our response to ACRS Question II.2(1) and NRC Question 6 under Estimation Methodology.

The results are not, in general, very sensitve to these choices.

Estimation Methodology

3. Why was the increasing trend in the turbine-driven AFWS pump unavailability ignored? Discuss the effect of recent AFW system failures on the ZPSS AFW analysis.

Response

The ZPSS employed a plant specific maintenance data base that covered a time period through 1980. Several unusual maintenance events in both units are recorded in late 1979 and early 1980 in which repeated maintenance outages occurred for a turbine-driven AFWS pump. It is not clear if this represents an increasing trend in unavailability or an anomaly due to the increased emphasis placed on AFWS pump testing following the TMI-2 incident. In either case, these outages are responsible for the higher unavailability of AFWS pumps at Zion compared with most other pumps. Collection of data was suspended so that the study could be completed.

The effect of the recent Zion AFW system failures on the ZPSS AFW analysis is described below.

a. Sneak Circuit in the Motor-Driven AFWS Pumps created by a design modification in 1981 (see I&E Information Notice 82-01). During the performance of the AFWS analysis of ZPSS, detailed investigations into the control circuits for all AFWS components were undertaken. Based on the depth of this review, we feel that if this potential failure path had existed at the time of ZPSS, it would have been discovered and corrected.

b. Common Mode Miscalibration of AFWS Low Suction Pressure Trip Switches [See letter, K. Graesser to NRC; with RO81-30 (December 24, 1981), RO81-31 (January 5, 1982), RO81-32 (January 8, 1982) Docket 50304]. This potential error existed at the time of the ZPSS and should have been included in the system analysis as a common cause contributor to system failure due to: (1) miscalibration, and (2) nondiscovery. Further analysis may be necessary to determine the exact impact of this error; however, because of the allowable time for recovery from this type of failure (30 minutes), this event should have very little effect on plant risk.

Estimation Methodology

4. Provide justification for your statements about completeness, particularly given the arbitrary and inconsistent ways in which "other" failure probabilities were estimated. (Why was an average β-factor of 0.14 assumed for the failure of similar components in the low pressure injection and recirculation systems, but not for the reactor protection, engineered safeguards actuation, containment spray, containment fan cooling, component cooling water, service water, and auxiliary feedwater systems?)

Response

We make no claim about absolute completeness. Of course it is illogical to do so. What we do with the "other" category is to provide a formal device and discipline for causing us to reflect on our analysis and make an allowance for failure modes not otherwise included in our analysis. Since the idea of such an allowance was somewhat new at the time, and also subtle, the various systems analysts have applied it differently and perhaps inconsistently. To some extent, this is appropriate because the systems are different and the style of analysis in different systems is different. This category is not a panacea or a magical recipe to insure the impossible, of course. It is a device to help us think and to help us look again, thus bringing us closer to the ideal and unattainable goal of true completeness.

Estimation Methodology

5. What is the basis of the assumed distributions (page 1.3-15) from which the electric power recovery probabilities were derived? How do you resolve your estimates of .28 and .03 for failure to recover in 30 and 60 minutes, respectively, with industry-wide experience of .41 and .26? Since it takes a more unusual event than average to fail Zion's offsite power, shouldn't recovery be less likely than average?

Response

Please refer to our response to the NRC Systems Analysis Question 11 for a discussion of the general bases for the offsite power recovery time distributions.

The Zion Probabilistic Safety Study model for the frequency of the loss of offsite power initiating event did not assume that it takes "a more unusual event than average" to fail Zion's offsite power supply. The model was developed from a Bayesian analysis of historical generic power failure data, accounting for site to site variability in this data. The generic failure data was then updated with the evidence that Zion has never experienced a loss of offsite power. An essential element of the Bayesian model is the assumption that Zion is a representative member of the population of all nuclear plant sites. The evidence that Zion has not experienced a loss of power is important information that the Zion site may be less susceptible to this event than a "generic" site, and this data is reflected in the frequency distribution used for the study.

A detailed review of offsite power failure events indicates that most events do not involve widespread physical damage to the local transmission system. Typical failures involve some degree of damage to transmission lines or switching equipment, the effects of which are compounded by real and spurious signals causing the remaining supplies to trip (e.g., from overcurrent, frequency oscillations, human errors, spurious transfer trip signals, etc.). Therefore, historical evidence shows that in most power failure events, circuits are available to restore power within a few minutes if the appropriate switching can be accomplished to reenergize the lines. We have not performed any sensitivity studies to investigate the effects that the number of circuits and geographic diversity may have on the time to recover power. However, based on this general experience, it is expected that transmission systems incorporating a large number of geographically diverse circuits will achieve at least partial power restoration within much shorter time periods than systems containing fewer circuits with substantial right of way restrictions. Another important factor in determining restoration time is the availability of personnel for local switching operations and the availability of utility central system load control. Zion station personnel operate the switchyard equipment, and Commonwealth Edison has one of the most modern central power system control centers in the country. Since successful offsite power recovery was modeled as requiring only one of the six offsite lines to be reenergized, we believe that there is substantial evidence to indicate that the time for power recovery at Zion should be less than that at a "generic" plant site.

Estimation Methodology

6. Please explain the interfacing LOCA estimates. What is the basis for using WASH-1400 bounds for valve rupture? How sensitive are the final risk estimates to this assumption? Why is the interfacing systems LOCA estimated to occur so much less frequently than a large or medium LOCA 1.05(-7) vs. 9.4(-4)? (See ACRS question II.2(2)).

Response

The interfacing LOCA frequency of occurrence is explained in the ZPSS, page 1.3-72.

The bases for using WASH-1400 bounds for valve rupture are:

a. The initial quantification of the "V" sequence used the 20/80 distributions as presented in the methodology section. As presented in the Easterling letter referenced in ACRS Question II.2(2), the results indicated a frequency of occurrence 3 to 4 orders of magnitude higher than the published results. Based on this initial quantification, a data search was begun to determine the frequency of failure due to disc rupture. During this search, several interesting facts were found.

   (1) There have been no failures due to valve disc rupture in the nuclear power industry.

   (2) Phone conversations with valve manufacturers revealed no gate valve disc failures to the best of their knowledge. Typically, valve manufacturers are required to hydrostatically test valve discs and bodies to 150 percent of design pressure. They knew of no failures during hydrostatic testing for any of their product lines.

Because of this data, it was felt that, in the absence of detailed experience (number of valves in similar locations in the industry and the exposure hours at pressure for each valve), the distribution from WASH-1400 adequately described our state of knowledge concerning this failure mode. Referring again to Easterling's letter, if the results of two coincident valve ruptures is as large as 10^{-4}, then our current experience (no disc ruptures at a nuclear power plant) would be nearly impossible. To date, no valve disc rupture has occurred in either manufacturing facilities (where testing at 150 percent of design pressure occurs) or nuclear power plants and, to the best of our knowledge, no disc rupture has occurred in conventional power plants or process facilities.

As seen in the Easterling letter, the results are extremely sensitive to this assumption. However, after results are obtained, they must be compared with history and the differences resolved. In the case of the "V" sequence quantification, the results obtained by using the 20/80 distribution did not agree with the large amount of evidence which is available concerning this failure mode.

The reason the frequency of occurrence of the "V" sequence is so much less than either the large or medium LOCA frequency of occurrence should be obvious. The "V" sequence can only occur at certain specific locations and with very definite modes of failure. The large and medium LOCAs could occur at any point in the RCS where the piping size meets the definition of large or medium LOCA. Failure in the RCS piping can occur due to pipe rupture, weld failure, or other modes associated with leaks from high pressure systems.

Estimation Methodology

7. Why was pressure vessel rupture excluded from explicit quantification?

Response

Pressure vessel rupture was explicitly included as a special case of large LOCA in Section 1.3.4.1.5 of the ZPSS.

Seismic Fragility

The following questions were taken from the draft report (dated February 22, 1982): "Review of the Zion Probabilistic Safety Study, Seismic Fragility," prepared for Sandia National Laboratories by Jack R. Benjamin and Associates, Inc., which is Appendix D to March 5, 1982 Sandia letter report. The page numbers below each question refer to the draft report which gives background material for each question. The issues raised represent the most significant concerns for the seismic fragility study which should be addressed and resolved.

1. What seismic hazard curve values were used in the integration with the seismic fragility curves to obtain the frequency of core melt probability distribution? (pp. 5, 6, 21, 23, and 94)

Response

The results of Section 7.9.1 are presented in Table 3 of that section, as the conclusions therein state. However, it was necessary to change the reported parameter from peak sustained-based acceleration a_{ps} to sustained acceleration, a_s, on which the fragility analysis is based (Section 7.9.3). According to Section 7.9.1, page 10, this is accomplished by the use of Equation (3) so that the plotted values of acceleration are the tabulated values divided by 1.23. The second consideration was that there be a correspondence between the predicted maximum site intensity and the maximum damage on which the Modified Mercalli Intensity (MMI) scale and acceleration-related damage are based in the fragility analysis. This leads to a justification for truncation of the seismic hazard curves as discussed in detail in the response to question 15. Then, so that the seismicity and fragility analysis results would be compatible in the assembly process, the exceedance frequencies for the a_{max} infinity cases, Section 7.9.1, Table 3, were truncated at accelerations recommended in Section 7.9.3 and corresponding to each of the three $m_{b,max}$ cases (using $I_o = 2m_b - 3.5$, per Section 7.9.1, page 7) of each seismogenic zone. Accelerations seen in Section 7.9.1, Table 3, were converted to sustained acceleration, the above truncations were applied to the table, and the resultant table was actually used in the calculations.

Seismic Fragility

2. Is the definition of the damage effective ground acceleration appropriate for equipment which depends on functional operation as opposed to ductile strength capacity (e.g., service water pumps)? (pp. 17, 56, 63, and 78)

Response

Equipment response to a seismic event is dependent on the structural response at the equipment location. The damage effective ground acceleration is a measure of expected structural response as opposed to the elastic structural response that would be calculated by anchoring the earthquake spectra to the highest instrumental peak acceleration; thus, for equipment mounted above the basemat, damage effective ground acceleration is appropriate regardless of the failure mode of the equipment. For equipment mounted on the basemat which fails in a structural mode, several cycles of strong motion are required for the equipment to reach peak response, and again, damage effective ground acceleration is a more rational indicator of failure level than instrumental peak.

For equipment mounted on the basemat and which is acceleration sensitive, use of the instrumental peak may be more appropriate. However, most acceleration sensitive devices such as relays are mounted in flexible cabinets or racks which are structural elements requiring several cycles of strong motion to develop peak response. Again, damage effective ground motion is considered appropriate.

Examination of the equipment list for the Zion PRA study reveals that there are no known acceleration sensitive devices that are mounted on the basemat which have rigid links to the basemat. Most of the acceleration sensitive equipment is mounted in the auxiliary building well above the basemat and none of the acceleration sensitive equipment items are mounted in rigid structural elements connected to the basemat.

Seismic Fragility

3. What uncertainty was assigned to the inelastic energy absorption parameter for structures and equipment to account for the variability caused by using single-degree-of-freedom models for multi-degree-of-freedom prototypes? (pp. 50 and 66)

Response

There was an uncertainty value (β_u = 0.2) assigned to the use of the equation for ductility factor. Most of the median equipment ductilities were considered to be 2.0 or less resulting in a ductility factor of 1.41 or less. For a system ductility of 2.0 and considering the properties of the lognormal distribution, the β_u of 0.2 applied to the ductility factor equation would reduce the median ductility factor to 1.0 at 1.71 β's below the median; thus, the model indicates that about 4-1/2 percent of the time the median ductility factor would be less than 1.0. This may be biased on the conservative side, especially for ductilities of less than 2.0.

For very high values of ductility, the uncertainty should be increased above 0.2. However, in the development of fragilities for equipment and piping, median ductilities of no greater than 3.0 were considered and the estimated uncertainty, β_u = 0.2, for use of a SDOF ductility criterion for MDOF systems is considered reasonable.

SMA has conducted some limited studies of MDOF versus SDOF response to mutliple dynamic loads and a general conclusion that can be reached from the studies is that the SDOF models tended to overpredict response to the loads being considered. The studies were conducted for purposes other than the question under consideration and are not concl ve; but based on the trend observed, the use of SDOF response behavior to estimate the beneficial effect of ductility on MDOF systems would, if anything, be biased on the conservative side.

Seismic Fragility

4. Since a more detailed analysis for the effects of inelastic energy absorption was conducted in the SSMRP for the Auxiliary Building, shouldn't the results from this analysis be used to determine the fragility parameters for the concrete shear wall? (pp. 50 and 59)

Response

The analysis conducted for the SSMRP was for a "representative" shear wall structure. This representative structure was based on the Zion auxiliary building, but did not include the turbine building which is integrally connected to the auxiliary building along a common wall. The analysis of this reduced structure consisted only of a two-dimensional analysis for response in the E-W direction. The structure is nearly symmetric about the E-W axis and very little torsional response results from E-W excitation. However, the structure is highly nonsymmetric about the N-S axis and torsion is significant. In comparison to capacity to withstand N-S excitation, the capacity in th E-W direction is considerably higher (by a factor of more than two). The controlling common shear wall capacity results from N-S excitation so that the SSMRP evaluation is not directly applicable. While the uncertainty in the PSS common shear wall fragility could be expected to be reduced by conducting a similar nonlinear analysis for N-S excitation, a three-dimensional model which also includes the turbine building would be required.

Seismic Fragility

5. What effect does the absence of perfect dependence have on the fragility curves for piping systems and cable trays (i.e., cable systems)? (pp. 28, 73, and 79)

Response

The fragility value for piping is intended to apply to a run of pipe from anchor to anchor. In any piping run the number of critically stressed areas is limited to only a few locations. The fragility model conservatively assumes that the most critically stressed element is a butt weld joint, typically at an anchor. This type of element has the least margin against plastic collapse when stressed to code allowables when compared to the margins for other types of elements which typically have higher stress intensification factors, and frequently govern the piping design. Consideration is given to the fact that for a piping system to collapse, more than one plastic hinge must form. It is possible in some piping runs for more than one portion of the run to be a weak link and in these few instances the fragility description may tend to underestimate the probability of failure. This is believed to be approximately accounted for by basing the piping fragility description on the weakest element (butt joint at an anchor) and assuming that this type of element is the highest stressed in the pipe run.

Having established a fragility description for a pipe run from anchor to anchor, the systems analyst then makes an estimate of the number of such runs that might be present in a system.

Cable tray fragilities were based upon test data for various tray and support assemblies and are considered valid for each assembly. The systems analyst then considers the effect of several assemblies in series in developing the fragility curve for a safety system.

Seismic Fragility

6. Since the ductwork and dampers, batteries and racks, relief tank, and transformers have relatively low capacity values based on generic data, shouldn't specific analyses for these components be performed to develop the fragility curves? (pp. 11, 76, 101, and 102)

Response

Median acceleration capacities of the above items are:

Ductwork and Dampers	0.97g
Batteries and Racks	1.01g
Relief Tank	1.19g
Transformer	1.39g

If plant specific analyses were conducted for these items, the uncertainty on median capacity would be greatly reduced, but we feel that any revised median value would probably fall within the $\pm\ 1_{\beta}$ value of the initially estimated uncertainty. As discussed in Section 7.2.4 of the ZPSS, it is not necessary to model some of these components (such as the relief tank and ductwork) because their failure is not a consideration in the failure scenarios. The transformer failure is negated by the lower capacity of its insulators. Further, as seen by the Booleans in Section 7, core melt is dominated by two major structural failures, the containment building and the control building. Relative to these two major structural failure modes of 0.73g for soil failure beneath the containment building and 0.73g for shear failure of the auxiliary building shear wall, the above equipment capacities are relatively high. Therefore, plant specific analysis of the batteries and racks would not change the results and is unwarranted.

Seismic Fragility

7. What effect does the coarseness of the data points for the hazard and fragility curves have on the accuracy of the tails of the probability density function for frequency of core melt? (pp. 5, 6, and 94)

Response

The importance of greater discretization of acceleration and fragility curves could only be shown if parallel calculations were performed, which we have not accomplished. However, we believe there would be no significant difference in the mean values or in the distributions.

Seismic Fragility

8. What is the basis for the displacement versus acceleration curves shown in Figure 4-7 of Section 7.9.2? (pp. 56 and 75)

Response

As discussed in the report, base slab uplift was determined based on a nonlinear dynamic analysis of the containment building which was conducted by Sargent & Lundy (Reference 1). Initiation of base slab separation from the soil is predicted at slightly less than the DBE (0.17g), and a maximum uplift of 1.9 inches at approximately 0.7g. This is consistent with uplift predicted from other nonlinear dynamic analysis of reactor containment structures (Reference 2). From Reference 1, a relationship between moment and acceleration was available. Using both the tangent stiffness and secant stiffness approximation, the base rotation and contact area as a function of acceleration were developed from which the uplift as a function of acceleration was computed. The accelerations included a factor of 1.09 to account for the response spectrum developed by the time history compared to the median centered response spectrum used in this analysis.

The variability shown in Figure 4-7 is the composite variability expected from both randomness and uncertainty. In the range of base slab uplift of interest shown in Figure 4-7, the curves of displacement as a function of acceleration become steep in the higher acceleration range, but they are not vertical. It should not be implied that displacements increase without bound. For massive structures such as the reactor building to be overturned as rigid bodies by earthquake excitation is not considered credible. Although overturning moments can be generated which, if applied as a static moment, would predict overturning, the time duration of earthquake cycles is much too short to allow rigid body rotations of this magnitude to occur. Thus, the direction of excitation will reverse long before the structure can rotate an appreciable fraction of the total rotation necessary to cause instability. As an example, the rotation at the base slab for 1.9 inches of uplift is approximately 1.5×10^{-3} radians.

References

1. Amin, M. and Y. N. Chen, "Mean Ultimate Capacity of the Zion Station Containment Building," Sargent & Lundy, July 1980.

2. Kennedy, R. P., S. A. Short, D. A. Wesley, and T. H. Lee, "Effect on NonLinear Soil-Structure Interaction Due to Base Slab Uplift on the Seismic Response of a High Temperature Gas-Cooled Reactor (HTGR)," Nuclear Engineering and Design 38, (1976).

Seismic Fragility

9. What effect would consideration of a "best estimate" site specific ground response spectrum relative to the broad-banded spectrum used in the analysis have on the value of the factor, F (Section 7.9.3)? (pp. 43, 92, and 93)

Response

A "best estimate" site specific ground response spectrum is not available for Zion and the comparison of results with the broadbanded spectrum has not been made. Presumably, a site specific spectrum would have somewhat lower randomness since the broadbanded spectrum includes earthquakes from a wide range of site conditions and magnitudes. However, some additional uncertainty would be introduced by the use of a site specific spectrum which would tend to offset the reduced randomness. In the PSS, all the variability in the spectral shape parameter was attributed to randomness since the uncertainty associated with the spectrum was judged to be small. (The broad-banded spectrum was based on including a large majority of the applicable earthquake records available at the time it was developed.) It is judged the composite variability, β_C, would not be significantly changed. If the β_C associated with the spectral shape for a typical failure mode (for instance, the auxiliary building concrete shear walls, $\beta_C = 0.18$) were distributed equally between β_R and β_U, then $\beta_R = \beta_U \approx 0.13$. For this case, the total randomness, β_R, is reduced to 0.27, the total uncertainty, β_U, is increased to 0.31, and the total composite variability, β_C, remains unchanged at 0.41.

Seismic Fragility

10. How close do the electrical components, which were eliminated from Table 7.2-3, compare to the tested components that were used to develop the generic fragility data? (pp. 12, 26, and 78)

Response

The components tested in the Safeguards Program were typically from two or three manufacturers. It is not known if the Zion components were from any of the manufacturers that supplied safeguards equipment. In the safeguards program, a few components had lower than acceptable fragility and were not used. Since we did not know the similarity between those components tested in the safeguards program and those installed in Zion, we included the weak components from the safeguards program in the data base to develop generic fragilities.

Seismic Fragility

11. How do design and construction errors and aging affect the fragility curves and the subsequent systems analysis for the effects of seismic events? (pp. 10, 39, and 62)

Response

Design and construction errors were not treated explicitly in the Zion PSS. The possibility of design and construction errors was considered implicitly in many cases in establishing the uncertainty associated with a given failure mode, however.

In the case of primary coolant piping, the possibility of a large through wall flaw was considered as a lower bound on capacity and limit moment capacities of other piping were biased below the test data median to account for possible flaws. Unfortunately, essentially no data is available to quantify these effects for nuclear power plants. Although a number of discrepancies have been previously identified in nuclear power plants, the items identified to date have been modified as necessary or shown to have no safety implications. The code of Federal Regulations provides a strong incentive to continue the upgrade of discrepancies. Nevertheless, there is a possibility that design and construction errors which can affect the seismic capacity may exist.

It should be recognized that design and construction errors do not necessarily always result in a decrease in capacity. It is also possible to install higher strength bolts than specified, larger reinforcing bars or more closely spaced bars than required, or slip a decimal point in the conservative as well as in the unconservative direction of the analysis. However, the inspection and QA requirements for nuclear power plants are expected to produce fewer design and contruction errors than in typical civil and mechanical construction projects.

Some additional confidence exists in that structures and equipment are subjected to normal operating loads and static 1g vertical loads continually. In many cases, these loads are large, as for instance in the case of pressure, water hammer, and thermal loads in fluid systems, when compared to seismic loads. Low level dynamic loads resulting from cranes, forklifts, and other component handling equipment produce loads in structures which occur on a regular basis and which might serve to initiate some failures with very low capacities resulting from design and construction errors. Pressure tests of containment vessels, while producing different types of response than seismic, would likely provide an indication if significant construction errors exist in these structures. Finally, wind forces on structures produce lateral forces which may be at least comparable in magnitude to those developed by earthquakes, and at least for some wind velocities, occur on a much more frequent basis. Thus, although data on which to quantify accurate estimates of the effects of design and construction errors are not available, these are expected to be minimal, and are included to some degree in the uncertainties estimated for the Zion fragilities.

Aging effects were included quantitatively only in the strength of concrete. For pressure boundaries of equipment items such as pressure vessel nozzles or piping, the presence of possible flaws introduced by aging (thermal fatigue, intergranular stress corrosion cracking, etc.) was implicitly included for the primary coolant system by consideration of flawed weld joints as a lower bound on capacity. For other piping, limit moments derived from test data were biased on the low side of the median value to approximately account for flaws that may occur from aging.

Aging effects data on seismic resistance of electrical components are not available. The uncertainty bound on electrical equipment fragility tended to be greater than for mechanical components. The wider uncertainty was intended to address not only the generic treatment of electrical components but to some degree, address mild to moderate aging effects on electrical equipment fragility.

Most electrical equipment fabrication materials and procedures have been developed from field experience and environmental testing data to assure that insulation material will survive the equipment operating life and environment. Performance testing at least every 30 days assures that aging deterioration that would render a component inoperable is detected. While functional testing does not assure that aging has not deteriorated seismic resistance, it does identify aging problems in general and offers some assurance that gross aging degradation will not be present. Since the uncertainties bound is believed to account for mild to moderate aging degradation, the effect of aging on electrical equipment seismic fragility is believed to be implicitly included.

Seismic Fragility

12. Was the possibility of a LOCA followed by an aftershock, or the occurrence of a moderate earthquake when some safety related equipment is unavailable, considered in the analysis leading to the probability distribution of frequency of core melt? (pp. 13, 45, and 58)

Response

The frequency of a nonseismic induced LOCA followed by an earthquake would be too low to contribute or to be considered in the analysis. The frequency of a seismically induced LOCA or melt resulting from a transient, causing containment pressurization with time and without release, followed by an aftershock large enough to cause additional failures was not considered further also because of the low frequency of this meticulously timed scenario.

Seismic Fragility

13. Could the split of variability into randomness and uncertainty components be different than assumed in the analysis. If so, what would be the effect of a different split on the tails of the frequency of core melt density function? (pp. 43, 51, 52, 54, 55, 58, 59, 60, 65, 66, 67, and 72)

Response

Some difference in the split between randomness and uncertainty can be expected from different qualified individuals making the evaluations. In the Zion PSS, randomness was in general based on available data such as material strengths, earthquake characteristics, and comparisons of SRSS modal analysis results with absolute sum results. Uncertainty was based on estimates of our lack of knowledge. If most of the variability was judged to be either randomness or uncertainty, in many cases the total variability was lumped in either randomness or uncertainty. However, in all cases, the composite variability was judged to be a realistic value. Sensitivity analyses on results of somewhat different splits between randomness and uncertainty have not been conducted. However, the results are not expected to be significantly changed since the composite value would not be changed.

Seismic Fragility

14. In developing the values for the mode shape parameter for equipment, was the location of equipment relative to the location of the masses of the building model considered? If yes, how were they considered? (pp. 52, 62, 82, 84, and 89)

Response

Equipment response was considered to be uncoupled from structural response except in the case of the primary coolant system. The primary coolant system analysis conducted by Westinghouse included the containment structure in the model. Other components are light relative to the structure supporting them and uncoupling is justified.

Most equipment and piping under consideration are floor or wall mounted. The floors and walls are relatively stiff and in-structure response spectra for local modes are not expected to be significantly different from the spectra used in design.

Seismic Fragility

15. Why in Section 7.2.2 were maximum acceleration values assigned, while in the section on the hazard analysis (Section 7.9.1) maximum acceleration values were treated as being uncertain? (pp. 17 and 18)

Response

The upper bound of effective peak ground acceleration (EPA) was developed based on the existence of assumed upper bound on Modified Mercalli Intensity (I_{mm}) levels. Based on historic and geologic data, upper bounds for I_{mm} for the Zion site are considered realistic. However, fragility curves defining the frequencies of seismic induced failures for structures and equipment are much more easily incorporated in the overall risk assessment when the frequency of failure estimates are defined as a function of the EPA or similar mathematically quantifiable parameter. The EPA is the ground acceleration level at which a broad frequency content structural response spectrum should be anchored for the purposes of predicting structural damage. Site specific ground response spectra for the Zion site were not available for the PRA. For predicting structure and component damage at the Zion site, Structural Mechanics Associates, Inc. (SMA) has used the median broad frequency content structural response spectrum for alluvium sites defined in Reference 1. SMA has assumed 3 to 5 near-peak response excursions approaching the levels defined by this structural response spectrum anchored to the EPA.

This approach is most applicable when dealing with longer duration ground motions which contain a broad range of frequency contents such as the Taft recording from the 1952 Kern County, California, earthquake (local magnitude M_L = 7.2, range from causative fault = 40 km) or the highway test laboratory recording from the 1949 Olympia, Washington, earthquake (M_L = 7.0, range = 29 km). For such earthquakes, the EPA to which a broad frequency content response spectrum is anchored and the instrumental peak acceleration (IPA) should be essentially the same. Such records result in 3 to 5 structural response excursions approaching the levels defined by the structural response spectrum.

However, the frequency of strong ground motion at the Zion site will be mostly due to lower magnitude earthquakes ($M_L \leq 5.7$) and shorter ranges (less than 20 km). Ground motions from such earthquakes have characteristics like those recorded at the Gilroy Array from the 1979 Coyote Lake, California, earthquake ($M_L = 5.7$, range = 7 km) or the Melendy Ranch Barn record from the 1972 Bear Valley, California, earthquake ($M_L = 4.7$, range = 6 km). These records have narrow frequency content, and within the majority of the frequency range of interest (2 to 10 Hz) their structural response spectrum is seriously overpredicted by the Reference 1 broad frequency content median spectrum when this spectrum is anchored at the IPA (Instrumental Peak Ground Acceleration). Secondly, only a single cycle of strong structural response occurs from these records because of their limited duration and energy content. Thus, for these records the IPA cannot be used as a basis for predicting the level of structural response which is approached 3 to 5 times during an earthquake. Since structural damage predominantly depends upon multiple cycles of strong response, the IPA cannot serve as a good indicator of structural damage for these earthquakes. In these cases, damage is better described by an EPA which is much less than the IPA. This EPA represents the ground acceleration for an equivalent long duration record with broad frequency content which causes 3 to 5 response cycles and results in the same expected damage level as the actual record.

Effective Peak Versus Instrumental Peak and Sustained Peak Accelerations

SMA is currently engaged with Woodward-Clyde consultants in a research program sponsored by the Nuclear Regulatory Commission (NRC) to define effective ground motion parameters useful in predicting structural damage. This section briefly summarizes some of the tentative findings to date.

For the purpose of predicting elastic response of structures in the amplified acceleration frequency range (2 to 10 Hz), median broad frequency content response spectra such as those from Reference 1 are more accurately anchored to an EPA defined by:

$$A_E = 1.25 * A_{3F} \tag{1}$$

rather than the IPA. The quantity A_{3F} represents the thirdhighest acceleration peak of a filtered acceleration time-history record. The filter should be chosen to pass all frequency content below about 8 Hz and filter out all frequency content above 9 Hz. The quantity A_{3F} corresponds closely with what Nuttli (Reference 2) has defined as sustained peak acceleration.

Therefore, Equation (1), together with Nuttli's definition of sustained acceleration, can be used to define an EPA (A_E) to be used to estimate elastic response of a structure within the 2 to 10 Hz frequency range. However, elastic response is not a good measure of damage. Based upon current work, two ground motion timehistories with the same spectral acceleration values at the structure's natural frequency can lead to vastly different nonlinear response or damage for the same structure model. For instance, one can compare (Reference 3) the maximum nonlinear response (damage) from the Melendy Ranch Barn record (magnitude 4.7) with that computed from the Taft record (magnitude 7.2). It is found that the Melendy Ranch Barn record must be scaled to produce spectral accelerations between 1 and 2g's at the structural natural frequency to produce the same level of damage as a 0.5g spectral acceleration from the Taft record. Thus, for Melendy Ranch, the spectral acceleration must be 2 to 4 times as great as for Taft to produce the same level of structural damage. Similar conclusions are reached for the Coyote Lake records (magnitude 5.7) versus the Olympia record (magnitude 7.0) or Taft. Thus, for obtaining an EPA corresponding to a given level of structural damage, Equation (1) should be modified, as follows:

$$A_D = \frac{1.25}{F} * A_{3F} \tag{2}$$

The factor F must be established as a function of ground motion characteristics for a constant level of structural damage. For magnitudes greater than about 7.0 and ranges greater than about 40 km, F can be taken as unity and Equation (1) can be used to predict the EPA corresponding to structural damage. However, with magnitudes less than about 5.0 and ranges less than about 20 km, F should have a value greater than 2 for predicting structural damage. As a consequence, the EPA (A_D) should range from less than 0.6 to 1.25 times the sustained ground acceleration (A_{3F}) depending upon the earthquake magnitude and hypocentral range with the lower factor being appropriate to low magnitudes (less than about 5.0) and short hypocentral ranges (less than 20 km). The appropriate ratio of A_D/A_{3F} is strongly influenced by the duration of strong ground motion.

The SMA fragility curves for Zion are best anchored to the damage EPA defined by Equation (2). Because of the generally low magnitude of earthquakes which might result in strong motion at the Zion site, it is judged that the 90 percent confidence bounds on F are:

$$1.0 \leq F \leq 3.0 \tag{3}$$

Because of the tentative nature of the research conducted to date and the controversy of the subject of EPA versus IPA, it is recommended that F be conservatively selected for use in Equation (2) and that the EPA be defined by:

$$A_D \approx A_{3F} \tag{4}$$

with Nuttli's sustained peak acceleration being used to define A_{3F}.

Upper Bound Cutoff on Effective Peak Acceleration

The EPA is being used as a measure of damage to structures with a fundamental natural frequency in the 2 to 10 Hz frequency range. The I_{mm} is also a measure of damage. Although I_{mm} is a subjective scale, it probably correlates best to damage of conventional structures which generally have natural frequencies in the 0.3 to 3 Hz range. Because both EPA and I_{mm} are measures of damage capability of ground motion, these two quantities should be closely correlated with each other. Thus, one should be able to establish upper bounds on the EPA irrespective of frequency of exceedance if upper bounds exist on intensity.

Table 1 describes the earthquake effects (damage) corresponding to each of the I_{mm} scale levels. These damage descriptions can be used to define upper bounds on the EPA corresponding to a given I_{mm} level. Masonry A construction corresponds to earthquake resistant masonry structures designed to the Uniform Building Code (UBC) in California (Zone 4). Masonry B contruction is reinforced and represents well-engineered masonry structures in UBC Zones 0 or 1. Masonry C contruction represents well-constructed unreinforced masonry structures. The SMA methodology used to develop the fragility curves for structures and components at Zion will predict very substantial damage and/or at least partial collapse of 50 percent of these masonry structures for 3 to 5 cycles of the following EPA levels:

Masonry Type	50% Damage EPA Levels (g's)
C	0.25 - 0.3
B	0.4 - 0.5
A	0.6 - 0.8

Thus, very serious damage to a large number of Masonry A, B, and C structures would be predicted by the SMA methodology to correspond to EPA levels of less than 0.8, 0.5, and 0.3g's, respectively. The SMA methodology for predicting damage levels has been benchmarked against observed damage in past earthquakes in which substantial damage was observed for sustained ground motions corresponding to these levels. Based upon the damage descriptions in Table 1, serious damage to at least some Masonry A, B, C construction correspond to I_{mm} levels X, IX, and VIII, respectively. Comparing the EPA levels defined above for each of these levels of damage, one would estimate that I_{mm} of X would correspond to an EPA of 0.6 to 0.8g's or less, I_{mm} of IX corresponds to an EPA of 0.4 to 0.5g's or less, and I_{mm} of VIII corresponds to an EPA of 0.25 to

0.3g's or less based upon the described damage to masonry construction. Even considering uncertainty in the correlation between the two descriptors of damage (I_{mm} and EPA), an upper bound must exist on EPA for a given I_{mm} level. Otherwise, the two indicators of damage would be contradictory. Therefore, an upper bound on EPA can be estimated by assigning the EPA ground motion levels defined above to an intensity value one level lower than that for which a given type of masonry construction damage is considered appropriate. Thus:

Intensity, I_{mm}	Upper Bound EPA (g's)
IX	0.8
VIII	0.5
VII	0.3
VI	0.2

The EPA values given in this table are judged to represent conservative upper bounds for the corresponding intensity levels. These EPA levels would result in the prediction of substantially more damage than that from which the intensity level is defined.

If upper bound intensity levels are defined for the Zion site, then the EPA levels should also be limited to being below the upper bound levels defined above when the SMA fragility curves are used to predict structure and component damage. Unless these limits to EPA are applied, one would predict substantially more damage than could possibly correspond to a given upper bound intensity level. Even with these limits, it is judged that the level of predicted damage would correspond to at least one intensity level higher than the upper bound intensity level.

References

1. "A Study of Vertical and Horizontal Earthquake Spectra," WASH-1255, Nathan M. Newmark Consulting Engineering Services, prepared for USAEC, April 1973.

2. Nuttli, O. W., "The Relation of Sustained Maximum Ground Acceleration and Velocity to Earthquake Intensity and Magnitude," Paper S-73-1 published in State-of-the-Art for Assessing Earthquake Hazards in the United States, U.S. Army Engineer Waterways Experiment Station, Vicksburg, Mississippi, 1979.

3. Kennedy, R. P., S. A. Short, N. M. Newmark, "The Response of A Nuclear Power Plant to Near-Field Moderate Magnitude Earthquakes", Paper K 8/1, Sixth International Conference on Structural Mechanics in Reactor Technology, Paris, France, August 17-21, 1981.

Seismic Fragility

16. What is the basis for the median capacity of 1.16g for the reactor pressure vessel internals? (p. 67)

Response

The median acceleration capacity is derived from the product of the equipment capacity factor F_c, the equipment response factor, F_{ER}, and the structural response factor, F_{SR}. The derivation of F_c is described in Section 7.9.2 of the report, pages 5-14 and 5-15, and is 1.66. F_c is based on the ratio of the collapse moment of the CR Guide Tube of 1.86 My (determined from experimental data) and the response computed for a Housner spectrum anchored to 0.5g and equal to 1.12 My. Note that the ratio of the 0.5g Housner spectrum to the Zion ZPA of 0.17g is accounted for in the reponse factor.

The response factor accounts for:

a. Ratio of Housner spectrum anchored to 0.5g to the Zion ZPA of 0.17g. F = 2.94 (note that 0.18g was used in the original calculation resulting in a conservative bias of 5.8 percent).

b. Response for design damping of 2 percent versus median damping of 5 percent:

$$F_D = 1.22$$

c. Mode shape frequency and mode combination factors of 1.0 each.

d. A combination of earthquake components factor of 0.93 to account for slight unconservatism in the design basis earthquake component combination versus a median centered combination.

The resulting response factor is 3.33 (3.15 originally calculated).

The structural response factor of 1.3 is listed in Table 5-3 of the report.

Multiplying the three factors times the DBE peak ground acceleration of 0.17g results in a median ground acceleration capacity of 1.22 (1.16 originally calculated).

Seismic Fragility

17. What is the basis for the fragility parameter values for the control rod drive mechanisms? (p. 68)

Response

The control rod drive mechanism capacity is stated in Table 5-6 to be 3.33g and is the product of the capacity factor of 5.50, the equipment response factor of 2.74, and the structural response factor of 1.30. The strength factor of 5.50 is developed on page 5-20 of the report and is based upon the ratio of the CRD housing strength to the response to a generic design spectrum. The difference between the generic design spectrum and the Zion DBE 5 percent damped in-structure response spectrum is accounted for in the equipment response factor.

The equipment response factor of 2.74 is the product of the following factors:

a. Qualification Method - At the fundamental frequency of 5.8 Hz, the ratio of the generic design spectral acceleration to the 5 percent damped spectral acceleration for the Zion DBE at the RPV support is 1.76.

b. Spectral Shape Factor - The Zion DBE in-structure spectra were described to be a factor of 1.67 conservative relative to the spectra that would have resulted from the specified ground spectrum.

c. Damping Factor - In this case, the damping factor is 1.0 since a 5 percent median damped spectrum was used to develop the qualification method factor.

d. Mode Shape, Frequency, and Mode Combination - These factors are all unity.

e. Earthquake Component Combination Factor - The appropriate factor is 0.93 for vertical components that respond predominantly in the horizontal direction. This results from comparing the design basis earthquake component combination criteria to estimated median centered response.

The structural response factor of 1.3 is listed in Table 5-3 of the report.

Multiplying the three factors times the DBE peak ground acceleration of 0.17g results in a median ground acceleration capacity of 3.33g.

Seismic Hazard (See Page 5.4 and Appendix A Pages A.2 to A.5 of Sandia letter report of March 5, 1982)

1. How can the proposed Wisconsin Arch or Wisconsin Arch-Michigan Basin seismogenic zones be justified on the basis of the known seismicity or the deep-seated geological structure?

2. Why shouldn't the cumulative magnitude-recurrence curve determined by Nuttli and Herrmann (1978) be used?

3. Isn't the epicenter for the May 26, 1909, earthquake near Aurora, Illinois, as given by Docekal (1970)?

4. Shouldn't the best estimate of $m_{b,max}$ be 6.0 and not 5.8?

Response

Our response to the above questions will be made by addressing the general questions on seismogenic zones, activity rates, and the 1909 earthquake.

Seismogenic Zones

For calculation of seismic hazard, one of the most important factors associated with seismogenic zones is the closest distance between the zone and the site. In addition to seismicity, geology, and tectonic evidence, the following details were considered in drawing seismogenic zones for the ZPSS:

1. Historical seismicity in northern Illinois was represented by a zone (the Wisconsin Arch zone) which restricted seismicity to distances greater than 45 km from the Zion site. This zone and its closest approach represent and are consistent with zones suggested by Nuttli and Herrmann (1978) and by Street (review of ZPSS).

2. Historical seismicity in northern Illinois was represented by a zone (the Wisconsin Arch-Michigan Basin zone) which encompassed the site, allowing earthquakes in that zone to occur at the site.

3. Historical seismicity in northern Illinois was considered to be representative of the Central Stable Region, and was modeled as such.

Arguments about the exact location of boundaries of these zones are of little relevance for the Zion site, because all other factors being equal, these boundaries do not significantly affect seismic hazard calculations.

The one exception is the eastern boundary of the Wisconsin Arch zone, as stated above, this boundary as used in the ZPSS is consistent with zones suggested by other investigators. Thus, the zones used in the ZPSS represent the range of seismogenic zones which might be proposed to represent seismicity in the northern midwest.

The Wisconsin Arch-Michigan Basin zone used in the ZPSS has been criticized because it cuts across major basement structure and is therefore unlikely. This is correct and the low relative likelihood of this zone is reflected in its assigned subjective probability (30 percent). The reason for including it at all is to present results for a zone which represents seismicity in the northern midwest and which encompasses the site, as discussed above. The Wisconsin Arch-Michigan Basin zone adequately serves that purpose.

Activity Rates

An important parameter in seismic hazard evaluations is the activity rate associated with each important seismogenic zone. Arguments about the location and size of specific historical earthquakes can be deferred by comparing activity rates used in the ZPSS study with those published elsewhere. Such a comparison is shown in the following table.

Zone	Activity Rate for $m_b \geq 4$	Activity Rate for $m_b \geq 5$
ZPSS Wisconsin Arch	0.0695	0.0083[1]
Nuttli and Herrmann (1978)[2] Northern Illinois	0.055[3]	0.0068[3]
Street - Northern Illinois (Figure 7.9.1-4)	0.15	0.011

(1) Calculated using modal b-value of 0.92.
(2) Study referenced in ZPSS Section 7.9.1.
(3) Corrected to plot cumulative data at lower ends of magnitude intervals.

The rates used for the ZPSS Wisconsin Arch zone are slightly higher than (but consistent with) those reported by Nuttli and Herrmann (1978) for the equivalent zone, and lower than those suggested by Street (Figure 7.9.1-4). The rates for the ZPSS zone were derived by recognizing uncertainties in historical earthquake locations, and by including in the Wisconsin Arch zone all large events near its boundary. This includes the large earthquakes in northern Illinois reported by Street (Figure 7.9.1-3). The larger rates derived by Street are a result of his larger zone (which includes many small earthquakes in northern Illinois and southern Wisconsin). In the seismic hazard analysis for Zion, the effect of these larger rates (30 percent larger for $m_b \geq 5$) would be counteracted by the larger area of seismicity (more earthquakes to the north, west, and south). Therefore, seismic hazard results for the suggested zone would be close to those reported for the ZPSS Wisconsin Arch zone. This would particularly be the case for higher accelerations (low annual frequencies) which are governed by the larger magnitudes.

1909 Earthquake

Arguments about the location of the May 26, 1909, earthquake are irrelevant for the present discussion. This earthquake has been included in the historial seismicity and used to derive seismic parameters (activity rate and maximum magnitude) for all seismogenic zones used to represent seismicity in northern Illinois. Exactly where the 1909 earthquake occurred is not important; the seismogenic zones used in the ZPSS reflect the uncertainty on where a similar event might occur in the future.

Street (review of ZPSS) indicates magnitude estimates for the 1909 event between 5.1 and 5.6. The best estimate of $m_{b,max}$ used in the ZPSS is 5.8, with discrete values of 5.6, 5.8 and 6.0 representing the uncertainty in $m_{b,max}$ and adequately bounding the range of estimates given for the 1909 earthquake.

Containment and Consequence Analysis

1. In Section 9 of the ZPSS, you describe features to mitigate the consequenes of core-melt accidents. On pages 9.0-3 and 9.0-4, you discuss a limitation of the filtered vented containment system, namely that risk reduction from a filtered vent is limited because of seismically induced failures. What would be the risk reduction from a filtered-vent if it was as robust (relative to seismic events) as the containment building itself, that is, it maintained its performance capability and met its design requirements for those seismic events that dominate risk yet allow the containment building to remain intact? What would be the additional cost for such a seismic qualification

Response

By idealistically portraying the filtered vent system as having a robustness identical to the containment building itself, and by ignoring concerns about operability, release category 2R becomes negligible in overall risk contribution. The net result is that for level 2 risk the early fatalities index at the 100 fatality level is reduced by approxi- mately a factor of 10 at the 90% confidence level and by less than this (perhaps a factor of 3 or 5) at the 50% confidence level from the ZPSS base case. Although this change would reduce the already very low risk, the intent of this section was two fold first to identify potential miti- gation features and second to identify equivalent systems. Thus the second part of this section should be addressed, namely what is the impact of seismically upgrading the diesel containment spray system to the same level as the containment building. This change would result in essentially the same risk reductions as discussed above.

Also, earthquakes which disable the plant control room and control systems would most likely render effective operator action on a filtered vent very unlikely. Thus, there would be little possibility for human operation of the vent or intervention in the event of a vent problem.

For both cases (diesel sprays and filtered vent) the conclusions of the ZPSS remain the same. Both would provide a reduction in risk, but this reduction is not significant especially in the context of the already very low risk. Furthermore, of the options reviewed, the diesel driven containment spray is clearly the best alternative.

The early fatality risk for seismic events for Zion results largely from sequences in which there is loss of long-term containment heat removal capability which in turn leads to long-term containment overpressure failure. The Zion study radioactivity release category applied to these sequences was category 2R. Category 2R led to a significant likelihood of early fatalities in the Zion study.

However the conclusion cited above is an artifact of the conservative Zion calculations. In the subsequent study for the Indian Point plant, radioactivity release from the containment for delayed containment overpressure failures was evaluated in more detail. The containment failure mode for such sequences is expected to be such that radioactivity release would occur over a number of hours. In addition radioactivity available for release would be markedly reduced by natural processes in the time between source release to the containment and containment failure. This would reduce the likelihood of early fatalities from a 2R release category to insignificant levels. Therefore, the benefit from a filtered vent would be reduced to an even more insignificant level.

Containment and Consequence Analysis

2. The risk of early fatalities drops by two orders of magnitude when going from your point estimate of risk to your level two estimate. As we understand it, this reduction is believed to be justified based on the recognition that the source term (release category) an site analysis in the point estimate is conservative. The method used to quantitatively account for the conservatisms is the use of U-factors discussed in Sections 5 and 8. Some aspects of this approach appear to be justified (e.g., the accounting for the delayed release in the 2R category), but other aspects are unclear. If you are going to quantify the conservatisms of your point estimate and display them (as level two curves), then more justification as to how you got there is justified. Please provide justification for the U-factors displayed on pages 8.6-3 and 4 and include your understanding of the phenomenology and accident progressions involved. (See ACRS question V.1).

Response

The U factors displayed on pages 8.6-3 and 8.6-4 are composed of 2 components:

1. Modification to the source term values to account for conservatism (or optimism) in the consequence calculation methodology. For this component of the U-factor, the source terms were used as surrogates to represent uncertainty in consequence calculations. (See Section 6.3.2.3 of ZPSS).

2. Discrete probability distributions or histograms developed to represent the conservatism in the source terms calculated by applying the source term calculational methodology of the Reactor Safety Study. The assignment of numerical probability values to the individual bins in the histograms of Figure 5.6.2-1 was based on engineering judgement. The assigned values represent an estimate of the

likelihood that the source term values calculated with the Reactor Safety Study (RSS) methodology or some fraction of the RSS value might be expected to be released to the environment.

The method for combining the two components of the U factors (developed in the form of histograms) is set forth in Section 8.6. Presented below is additional explanation of the basis for source term histogram estimates of Figure 5.6.2-1.

Source Term Histogram

The basic RSS methodology for calculating source term release to the environment was considered to be the following:

a. It is assumed that fission products released from the core are transported through the reactor system to containment without attenuation.

b. Fission product depletion from the containment atmosphere is calculated by applying the CORRAL Code.

c. It is assumed that there is no source term attentuation along the leakpath from the containment to the environment.

The basic RSS methodology for calculating the release of radionuclides from containment to the environment in the event of a core melt accident is generally acknowledged in the technical community to produce estimates which are conservatively high. However, existing data and models are not yet sufficiently developed to provide a basis for calculating better estimates of radioactivity release. Therefore, histograms were utilized to express engineering judgement of the degree of conservatism in the release valves in a form that can readily be incorporated into the overall risk analyses. The point estimate source terms produced by the RSS methodology then in reality represent an upper percentive estimate and that upper percentive estimate has in effect been replaced by more realistic radioactive release valves. The release category for which application of the histogram had the longest impact on risks was release category 2R. The point estimate valves for this release category (calculated with the RSS methodology) assumed early containment failure; the histograms were used in part to account for the very significant depletion by natural processes of the radioactivity available for release (upon containment failure), by natural processes during the 8 or more hours between fission product release to the containment and containment overpressure failure.

For each of the histograms, a particular accident sequence was considered in developing the histogram for that release category. The accident sequence selected was the most probable sequence for that release category. The appropriate sequence is indicated in the discussion in Section 5.6.2.1 through 5.6.2.3. Probability distribution histograms were only developed for those release categories which were significant contributors to risk. The bins selected for the histograms were a) the source value calculated with the RSS methodology, b) one-half of source value calculated with the RSS methodology, c) one-tenth the source value calculated with the RSS methodology.

A discussion of the basis for the probability values for each of the histograms follows:

1. Release Category 2 (see Section 5.6.2.1)

The V sequence is the dominant sequence considered for this case. For this sequence some retention of fission products during transport through the reactor system and RHR system to the break is expected. Additionally, the auxiliary building is likely to experience local rather than gross failure as a result of the V sequence blowdown so that some holdup and deposition of fission products in the auxiliary building is likely. Overall reduction of the calculated source term by at leat a factor of 1/2 seemed the most likely situation. Values near the calculated source term might occur if gross failure of the auxiliary building were to occur so a value of 0.25 was assigned. Reduction by as much as a factor of 10 is less likely and would only occur if the auxiliary building were to be relatively leak tight at the time of fission product release. Hence a value of 0.15 was assigned.

2. Release Category 2R (Section 5.6.2.3)

The dominant accident sequence considered in estimating a source probability distribution for release Category 2R was a core melt sequence initiated by a transient with loss of all AC power and with loss of secondary heat sink. Loss of power continues for the long-term and eventually a delayed containment failure occurs from pressure buildup in containment. Containment failure occurs 10 to 12 hours after accident initiation. Containment sprays are not functional for this sequence. Such a sequence generally corresponds to the TMLB' sequence of the RSS.

In the absence of CORRAL calculations the category 2R source term assumed in this study was the same as that utilized for category 2. That is, essentially complete release to the containment for transportable fission products were assumed. It was further assumed that fission products released to containment remain in the containment atmosphere and available for release for a number of hours. In fact, the substantial period of time between fission product release (at vessel failure) and containment failure will permit much of the iodine to deposit on surfaces and also provide time for significant aerosol settling and deposition. As a result, a source term of the magnitude of a factor of 10 less than that assumed in the study is regarded as more likely. Even smaller source terms are possible while source terms as large as the point value are regarded as very unlikely. In addition, for such a sequence it is likely that the melt release component will flow through the hot leg to the pressurizer out the safety valve and to the quench tank for the period between core melt release and reactor vessel melt through. Relatively complete retention, of radioiodine and particulate material so transported, by the water in the pressurizer and/or quench tank is possible.

The assigned probability values of 0.45 for source terms a factor of 10 below those calculated with RSS methodology and 0.25 for an even greater reduction reflect the judgement that point value source terms for this release category are conservatively high by a significant factor.

3. Release Category 8B (Section 5.6.2.3)

The dominant accident sequence considered in developing a source term probability distribution for release Category 8B was a core melt sequence initiated by a small pipe break with failure of either ECC injection or ECC recirculation; containment sprays are assumed to be functional. A significant fraction of the fission products released from the core debris are likely to be eventually transported to the containment, but most will be washed out by the sprays. Some retention in the reactor system not accounted for in the calculation is also likely. CORRAL reoresentation of the containment processes is regarded as reasonably good although greater reduction of radioiodine than accounted for in CORRAL is likely. A value of 0.5 was selected for the factor of 1 bin and 0.4 for the factor of 1/2 bin to account for the situation where reactor system retention is effective. For some small breaks or transient sequences where water lutes may exist along the trans- port path, reduction by factors as great as 10 can occur; a factor of 0.1 was assigned to this bin.

3. The post-vessel-failure accident progressions are dominated by a blowout of core materials from the reactor cavity into the containment proper, and a settling out on the containment floor according to the ZPSS. Assuming that 25-50 percent of the core inventory is so dispersed, how do you claim that there is virtually no effect on the containment sprays in the recirculation mode If significant amounts of this core particulate gets into the sumps, the sprays could fail or be seriously degraded. In order for us to better understand your position, please provide:

   (a) the range of particle sizes you expect with a full justification of why;

   (b) the settling characteristics and final disposition of these core materials within the containment with fan coolers on and with fan coolers off;

   (c) the magnitude and composition of particulates that you believe would fail the sprays.

Response

Please refer to our response to enclosure 3, systems analysis, question 1. The Zion study takes no credit for post vessel failure spray recirculation. However, we do expect that such an operation would be successful.

Materials which maybe present in the containment as fragmented matter include:

1. concrete
2. paint
3. steel trash
4. core debris

The containment sump is protected by a sump screen system consisting of a number of layers of trashscreens. The outer layers are gross trash-screens, and the innermost layer of screening has hole sizes of 1/4 inch diameter. Therefore the largest particle in the recirculation piping is 1/4 inch in diameter.

Westinghouse tests have indicated the following regarding suspended particulate of varying specific gravity substances.

Substance	Specific Gravity	Upper Limit of Suspended Particle Size**
Concrete, Glass	2.4 - 4.0	0.02 inch
Steel	8	0.015 inch
Paint	0.9 - 3.0	0.25 inch*

* size dictated by sump screen
** containment floor water velocity = 1/2 fps

Thus, it is seen that large diameter high density particulate would not reach sump, any high density material reaching the sump is limited to very small particulate because of gravitational settling in the low flow velocity field over the containment floor, and that low density particulate is limited to 1/4 inch by screens. Thus any suspended solids are less than 1/4 inch diameter.

The spray/recirculation pump vendors have been previously surveyed about the reliability of their pumps operating with suspended solids. All vendors agree that their pump characteristics are favorable for pumping slurries and that problems should not occur unless the pumps are stopped for significant periods after pumping a slurry.

It should also be noted that the spray nozzle has an internal diameter of 3/8 inch, and the above information is independent of fan cooler operation.

A range of debris sizes from a few tens of microns up to large globules several inches thick may result from a wave type dispersion of molten core debris from the cavity in the event of a pressurized vessel melt-thru event. It is expected that by far the predominant mass would be associated with relatively large particulate (several mm diameter to several centimeters). The recirculation sump is about 45° from the instrument tunnel opening and because of this and existing wall geometry is not in a direct path of material which could be driven up the sloped wall of the tunnel.

The bulk of this material would immediately collect on the floor as large debris chunks, and the remainder should settle in the low water flow velocity that is directed toward the sump except for suspended fines wich are not anticipated to degrade the spray recirculation.

Containment and Consequence Analysis

4. You claim very low probabilities for basemat penetration (0.02 percent) in your contaiment matrix; yet, your analysis indicates that:

   (a) the reactor cavity will eventually be dry for such damage stages as TE and SE;

   (b) a large fraction of the core material will not be blown out of the reactor cavity (in particular, the 50 percent remaining in the vessel after initial blowdown);

   (c) basemat penetration times for non-coolable debris beds are in the range of one (1) day.

   Please support this claim which appears to be inconsistent with the points raised above. Also reference where in Section 4 you perform a dry-cavity overpressurization analysis. Please provide a containment temperature history plot consistent with your dry-cavity overpressurization analysis.

Response 4(a)

We note that on page 2.6-37 of ZPSS that the probability for basement failure assigned for the TE and SE states was conservatively estimated as 0.9999 not 0.0002. No other split fraction for this class was assigned (i.e. via a conditional). This value could be conservatively assigned since late overpressure sequences with 0.9999 probability ended in a late overpressure which is a much worse release category.

The question in the containment event tree just prior to basemat penetration is the late overpressure question. We assigned a value of 0.9999 for late overpressure for the TE and SE states. This leads to a worse release than the basemat release and therefore it was not very important whether we assigned 1.0 or 0.0 to the basemat question.

Response (b) and (c)

For the TE and SE type of plant states the most likely accident progression is dispersal of the core debris which is generated initially. The dispersal will drive the core debris out of the reactor cavity and onto the floor area in the containment. This will be followed by a later degradation and disposition of the bulk of the remaining core into the lower reactor cavity region.

The majority of the core debris which is dispersed from the lower reactor cavity will quench initially, and then begin to boil water from the containment floor. During this time period, the water on the floor is being depleted due to the boiling induced evaporation and being replenished due to condensation on structural surfaces in the containment. Since the evaporation rate exceeds the condensation rate, eventually the containment floor dries out. At this point in the transient the dispersed core debris will begin to attack concrete and transfer heat directly into the containment atmosphere. Our calculations indicate that due to the large surface area available, the majority of the core debris will transfer heat directly into the atmosphere.

The core debris remaining and that generated long term in the lower reactor cavity will evaporate the water in that area, and then probably begin to attack concrete.

The containment pressure and temperature transient will result from the integrated effect of the effects from the dispersed core debris and that which remains in the lower reactor cavity.

In the risk study performed, the containment was conservatively assumed to fail at about 12 hours into the accident. For the best estimate accident sequence the lower reactor cavity did not dry out and thus concrete attack did not begin until after more than approximately 9 hours into the accident. Given that only a limited amount of core concrete reaction would occur, it was judged that only a rapid pressure rise (spike) or gradual overpressure would fail the containment.

Response (b) and (c) (Cont'd)

Our conclusion is therefore that for the TE and SE events concrete attack will occur but will not cause basemat penetration prior to the end of an approximate 12 hours time interval which was assumed for the containment failure times TE and SE (i.e., class V events). This is conservative we would agree in light of code results that shown much slower penetration rates as possibly alluded to in part c of the above question.

In summary in the Zion PSS we assumed a conservative containment failure time of 12 hours, and a conservative containment failure mode of late overpressure vs containment basemat failure.

Containment and Consequence Analysis

5. You present three containment matrices: one representing your best estimate; one, your more conservative estimate: and one, your more optimistic estimate. When we assess the impact of the differences between the three, we see virtually no change in risk. This implies that you have sufficient knowledge of containment phenomenology and failure behavior that no important uncertainty exists. We do not share this position. Please clarify whether we have interpreted your position correctly.

Response

The lack of sensitivity of overall risk to containment matrices with branch probabilities representing best estimate, optimistic, and conservative estimates to a large extent is explained by the fact that overall risk was dominated by events in which no containment safeguards are available or in which the containment was bypassed. These have containment failure probability of unity and hence are not very sensitive to containment matrix uncertainties. For other core melt transients it would require a major stackup of unexpected phenomena in order to reach the very substantial containment failure limit. We have identified what we consider to be a reasonable range of phenomena and associated uncertainties to investigate this aspect. Many of these uncertainty ranges were very substantial (i.e., order of magnitude) and in fact had major impact on certain release categories. The fact that the overall results were not greatly impacted does not necessarily imply that this insensitivity would also be observed for a different mixture of core melt sequence types or for different containment capabilities.

The largest shifts in probability occurred at Node H, In-Vessel Debris Bed Coolability. Containment Classes I, II, V, and VI already had only an ϵ chance of succeeding here, and containment classes III and IV, in the worst case reduced their success probability by a factor of 1000. Most non-ϵ modes had detrimental changes in probability in the worst case of factors of two or five. Those nodes with an ϵ containing split fraction were situations where the evidence was overwhelming even if large changes in physical parameters occured. For instance, if the best estimate containment pressure at Node R was 25 psia and several other extreme phenomenological events are piled on top of that, the resultant pressure may be 100 psia which is still well below the 149 psia failure pressure. Thus, the ϵ probability of containment failure is unchanged.

An important conservation in calculating the worst case "C" matrix is that the several nodes which had more pessimisstic probability assignments were combined as worst case upon case. In other words, the extremes of the probability ranges were multiplied together. No distribution, convolutions, DPD arithmetic, or monte carlo treatments were used in these calculations.

Containment and Consequence Analysis

6. It's not clear to us that the emergency planning assumptions used in the consequence analysis are consistent with emergency planning procedures presently in place, or planned for the future. Please provide a comparison of your assumption with in-place procedures.

Response

It was assumed that implementing procedures would incorporate EAL guidance in NUREG-0654 (Rev. 1) or similar guidance in its predecessor NUREG-0610. That is, it was assumed that notification of authorities by plant per- sonnel would be prompt and, in time, notification of the populace by authorities would also be prompt. The simulation of evacuation was designed to represent routes and travel times estimated by Stone and Webster* for evacuation of a zone approximately 11 miles in radius in accordance with emergency evacuation plans in effect at the time. The Stone and Webster study estimated travel times of 3.0 to 22.4 hours for evacuation of the EPZ and identified a single route change which could reduce the upper estimate to 16 hours. The simulation of the evacuation plan in the ZPSS incorporated very low travel speeds--3 mph maximum and 1 mph in elements in the NW sector beyond 4 miles. Lower speeds were assumed for elements containing large numbers of people in special facilities requiring substantial mobilization time. The travel time for EPZ clearance calculated from the simulation is 11.5 hours, somewhat shorter than the maximum of 16 hours but much longer than the minimum of 3 hours.

*"Preliminary Evacuation Time Study of the 10-Mile Radius Emergency Planning Zone at the Zion Station," Stone and Webster Engineering Corporation, January 1980.

Containment and Consequence Analysis

7. We would like to see risks presented as a function of distance for individuals, also costs of mitigation and interdiction. Please provide graphical displays for selected consequence categories.

Response

Commonwealth Edison has employed a selected display of health effects to portray the risks associated with Zion Station. The form of that presentation is one which we find both lucid and useful. We do not see any justification for the additional analysis work requested in terms of assessing Zion Station. We therefore decline to perform such analyses.

Containment and Consequence Analysis

8. Are the supportive medical treatment assumptions used to correlate dose versus early fatality consistent with the available resources to provide such treatment to the exposed persons? Please explain the bases for your answer.

Response

Please refer to our response to ACRS question VI.2.4.

Containment and Consequence Analysis

9. Are the early fatality consequences determined for the release category with the largest release fractions for particulates consistent with the early fatality consequences estimates determined for categories with much smaller release fractions? Please explain the bases for your answers.

Response

The question is unclear. Early fatality consequences are clearly highest for the early releases with large particulate release fractions. The smaller release fraction release categories generally have longer release periods and larger warning times which reduce consequences. The dose versus health effect relationships for early fatalites incorporate thresholds which are not exceeded for small release fraction accidents.

Containment and Consequences Analysis

10. We are not sure how accidents involving the spent fuel pool were considered. Please provide an explanation.

Response

Please refer to Volume 1, Section I.4, page I-14 of the Zion study. Non-core accidents were evaluated elsewhere (NUREG-CR/0603) and found to be very small contributors to risk.

Containment and Consequence Analysis

11. The evacuation assumptions should consider external accident initiators, or should be shown to not produce substantially different consequence estimates. Please provide justification for your assumptions considering both internal and external accident initiators.

Response

Please refer to our response to enclosure 1, (question VI.2 2).

Containment and Consequence Analysis

12.1 The LASL analyses of the Zion Containment shell deformation under pressures approaching shell failure levels (in the 125 psig range) predict upward vertical displacements of over 3" at the dome apex. This displacement appears to result from combined saucering of the basemat and axial strains in the cylindrical containment wall. Additional horizontal displacements (up to about 1.5" at midheight) occur due to shell distortions because of the radial growth of the shell.

Response

It must be noted that the Zion Probabilistic Safety Study contains a study of the containment ultimate capacity and the behavior of the containment at elevated pressures. We can be responsive to questions covering that effort. We have not reviewed the LASL work in any detail. Absent such a review, we suggest that questions on the LASL work be addressed to LASL since they have a ready frame of reference for responses. Our responses to questions 12.1 and subsequent questions referring to LASL work represent a best effort to provide information, but should not be construed as an endorsement of the LASL work referenced. Moreover, unless otherwise noted, our responses reflect our considered judgement regarding specific questions based on the LASL work. No detailed analyses have been performed to duplicate the LASL study and assess effects therefrom.

Containment and Consequence Analysis

12.1(1) The LASL analyses of the Zion Containment shell deformation under pressures approaching shell failure levels (in the 125 psig range) predict upward vertical displacements of over 3" at the dome apex. This displacement appears to result from combined saucering of the basemat and axial strains in the cylindrical containment wall. Additional horizontal displacements (up to about 1.5" at midheight) occur due to shell distortions because of the radial growth of the shell.

What potential structural failures could exist (under these conditions) in the fuel transfer tube at its point of passage from the containment shell to the spent fuel building?

Response

The fuel transfer tube is located 13'-7" above the basemat and, therefore, the displacements at the transfer tube will be considerably less than the maximum movement noted in the above question. The transfer tube is equipped with bellows, expansion joints (see FSAR Figure 5.1-14); and any differential movement between the containment and fuel building will be accommodated by this assembly.

Containment and Consequence Analysis

12.1(2) The LASL analyses of the Zion Containment shell deformation under pressures approaching shell failure levels (in the 125 psig range) predict upward vertical displacements of over 3" at the dome apex. This displacement appears to result from combined saucering of the basemat and axial strains in the cylindrical containment wall. Additional horizontal displacements (up to about 1.5" at midheight) occur due to shell distortions because of the radial growth of the shell.

What effect would structural failures have on potential loss of containment integrity?

Response

As described in the Containment Ultimate Capacity Report, it has been determined that up to the calculated ultimate pressure capacity, no structural failures would occur and, therefore, there would not be any loss of containment integrity.

Containment and Consequence Analysis

12.1(3) The LASL analyses of the Zion Containment shell deformation under pressures approaching shell failure levels (in the 125 psig range) predict upward vertical displacements of over 3" at the dome apex. This displacement appears to result from combined saucering of the basemat and axial strains in the cylindrical containment wall. Additional horizontal displacements (up to about 1.5" at midheight) occur due to shell distortions because of the radial growth of the shell.

At what pressure level (below 125 psig) would any failure first appear?

Response

As described in the Containment Ultimate Capacity Report it has been determined that no structural failures would occur at or below the calculated ultimate pressure capacity. Beyond that pressure, structural failures are anticipated.

Containment and Consequence Analysis

12.2(1) The LASL analysis of Zion indicates radial displacements at the midheight of the cylindrical shell of about 1.5" (at about 125 psig). At the elevation level of various groups of piping penetrations in these plants, the combined vertical displacements of perhaps 2" plus radial displacements would result in net movement of the penetrations of several inches. If the ends of any piping runs (both inside and outside the containment) are substantially restrained (i.e., attached to internal components that do not move with the containment or to adjacent tunnels or buildings) at points relatively near the penetration, there is a potential for placing very severe loads on penetration components.

To what extent are any piping runs restrained so that they cannot move to accommodate shell deformations; what are the anchorage/ support mechanisms for each pipe run and the distances between penetrations and the closest major pipe support inside and outside the shell?

Response

The major piping (mainstream and feedwater) are not restrained at the containment wall. The anchor point is outside the containment and the penetrations have bellows to allow for thermal expansion and containment movement. The remaining piping penetrations are the anchor points for piping into/out of the containment. The location of the pipe support closest to the containment wall is a function of the respective pipe line size, internal temperature and wall thickness. All of the connected piping is designed to accommodate appropriate thermally induced displacement. We foresee no problems in accommodating the motion projected in the Zion PSS (Section 4.4). See FSAR questions and answers.

Containment and Consequence Analysis

12.2(2) The LASL analysis of Zion indicates radial displacements at the midheight of the cylindrical shell of about 1.5" (at about 125 psig). At the elevation level of various groups of piping penetrations in these plants, the combined vertical displacements of perhaps 2" plus radial displacements would result in net movement of the penetrations of several inches. If the ends of any piping runs (both inside and outside the containment) are substantially restrained (i.e., attached to internal components that do not move with the containment or to adjacent tunnels or buildings) at points relatively near the penetration, there is a potential for placing very severe loads on penetration components.

Are any major penetrations (such as equipment hatch or personnel hatch) or purge lines, etc., sufficiently restrained outside the containment shell so as to potentially be heavily loaded due to the gross shell deformation?

Response

Review of the equipment-personnel hatch and emergency air lock details show no restraint outside the containment shell sufficient to heavily load the penetrations. The purge line outside the containment beyond the isolation valve is a thin wall pipe which is not capable of restraining the containment deformation.

Containment and Consequence Analysis

12.2(3) The LASL analysis of Zion indicates radial displacements at the midheight of the cylindrical shell of about 1.5" (at about 125 psig). At the elevation level of various groups of piping penetrations in these plants, the combined vertical displacements of perhaps 2" plus radial displacements would result in net movement of the penetrations of several inches. If the ends of any piping runs (both inside and outside the containment) are substantially restrained (i.e., attached to internal components that do not move with the containment or to adjacent tunnels or buildings) at points relatively near the penetration, there is a potential for placing very severe loads on penetration components.

What are the loads and strains induced upon penetrations by shell deformation; what are the probabilities of failures in welds and distortions of hatch seals or valve seats, that could cause loss of isolation capability; can such failures occur concurrently inside and outside the shell (due to shell distortion) so as to cause loss of double sealing capability or of any pressurized zones between seals?

Response

Load induced by shell deformation is considered an anchor displacement load which is a secondary type load.

Based on our judgement, the probabilities of such failures are very insignificant.

We don't anticipate a concurrent double failure or postulated breaks.

12.2(4) The LASL analysis of Zion indicates radial displacements at the midheight of the cylindrical shell of about 1.5" (at about 125 psig). At the elevation level of various groups of piping penetrations in these plants, the combined vertical displacements of perhaps 2" plus radial displacements would result in net movement of the penetrations of several inches. If the ends of any piping runs (both inside and outside the containment) are substantially restrained (i.e., attached to internal components that do not move with the containment or to adjacent tunnels or buildings) at points relatively near the penetration, there is a potential for placing very severe loads on penetration components.

Could the vertical axis of the containment shell tilt under the postulated high pressures (as a result of non-symmetrical saucering of the basemat or restraints imposed by structural elements below the general level of the basemat)? Could such tilting increase the loads upon penetrations, due to increased displacement relative to external connections?

Response

The change in stiffness of the mat complex due to the keyway below the mat may cause local asymmetrical dishing of the slab. However, it is expected that any non-symmetrical displacement or rotations due to the restraining effects imposed by these structural elements, which are located relatively close to the center of the circular mat, will attenuate toward the containment shell resulting in insignificant tilting of the containment vertical axis.

12.2(5) The LASL analysis of Zion indicates radial displacements at the midheight of the cylindrical shell of about 1.5" (at about 125 psig). At the elevation level of various groups of piping penetrations in these plants, the combined vertical displacements of perhaps 2" plus radial displacements would result in net movement of the penetrations of several inches. If the ends of any piping runs (both inside and outside the containment) are substantially restrained (i.e., attached to internal components that do not move with the containment or to adjacent tunnels or buildings) at points relatively near the penetration, there is a potential for placing very severe loads on penetration components.

At what pressure level are any of the above failures likely to first appear? How would any leak paths grow as a function of pressure?

Response

No failures are expected at pressure levels up to about 125 psi gauge. At pressures beyond that, failures are expected to develop and become larger with further increase in pressure. No relationship between leak path and pressure has been defined.

Containment and Consequence Analysis

12.3(2) Is there any possibility that thermal expansion of piping would cause excessive stresses in piping penetration welds (with or without any allowance for relief due to any concurrent growth of the containment shell)?

What would be the potential strains in welds; are any strain levels expected to constitute structural failure or loss of containment integrity?

Response

We expect that the strain levels will be limited to such values that they will not cause structural failure or loss of containment integrity.

As explained in the response to question 12.3 (4), the loads at the penetration due to temperature increases and shear deformations are not additive for piping inside the containment.

Containment and Consequence Analysis

12.3(3) Is there any possibility that thermal expansion of piping would cause excessive stresses in piping penetration welds (with or without any allowance for relief due to any concurrent growth of the containment shell)?

Are there any accident conditions in which such temperature stresses and strains could occur without the concurrent presumed relieving effect of shell growth under pressure?

Response

Such conditions are not expected.

Containment and Consequence Analysis

12.3(4) Is there any possibility that thermal expansion of piping would cause excessive stresses in piping penetration welds (with or without any allowance for relief due to any concurrent growth of the containment shell)?

Are the strains due to temperature increases in piping and those due to shell distortions geometrically related (i.e., non-aligned or out-of-phase) in any way so as to be additive (rather than tending to relief)?

Response

They are not additive (i.e., they are tending to relief). For piping outside the containment only the strains due to shell distortions are of importance since no temperature increase is anticipated.

Containment and Consequence Analysis

12.4(1) In the Zion plant, the electrical penetrations are apparently all of the D. G. O'Brien type, utilizing glass hermetic seals around conductors plus epoxy potting compounds and other materials. Limited equipment qualification testing appears to have been done on these penetration assemblies or the individual materials. All equipment testing was apparently conducted for pressure and temperature conditions pertinent to design bases currently reflected in licensee safety analysis reports; no tests appear to be available for conditions now being analyzed for more severe degraded core sequences. For the Zion penetrations, qualification tests employed conditions of 265F to 273F for 49 hours at 46 psig. Although an IE review concluded that there was a "high likelihood" of satisfactory operability under the design conditions, reservations were expressed (even for design basis conditions) that the potting materials might be suspect under high relative humidity, that the tests did not adequately simulate saturated steam conditions, that epoxies and polymer materials have shown aging effects under combined accident conditions of temperature, irradiation, and water/spray chemistry, and that some uncertainty about service life may exist for conductor insulation and jacket materials.

What are the specific materials used for assuring containment integrity?

Response

The specific material are stainless and carbon steels, fused glass, ferrous or molybdenum alloys.

12.4(2) In the Zion plant, the electrical penetrations are apparently all of the D. G. O'Brien type, utilizing glass hermetic seals around conductors plus epoxy potting compounds and other materials. Limited equipment qualification testing appears to have been done on these penetration assemblies or the individual materials. All equipment testing was apparently conducted for pressure and temperature conditions pertinent to design bases currently reflected in licensee safety analysis reports; no tests appear to be available for conditions now being analyzed for more severe degraded core sequences. For the Zion penetrations, qualification tests employed conditions of 265F to 273F for 49 hours at 46 psig. Although an IE review concluded that there was a "high likelihood" of satisfactory operability under the design conditions, reservations were expressed (even for design basis conditions) that the potting materials might be suspect under high relative humidity, that the tests did not adequately simulate saturated steam conditions, that epoxies and polymer materials have shown aging effects under combined accident conditions of temperature, irradiation, and water/spray chemistry, and that some uncertainty about service life may exist for conductor insulation and jacket materials.

What mean service life and standard deviation data exist for Zion penetration assemblies under combined severe accident conditions of concurrent temperature, pressure, radiation, steam, and water/spray chemistry? Can any existing life service data be extrapolated to the potentially more severe conditions of degraded core events?

Response

Similar electrical penetrations have been tested to temperatures and pressures beyond the original Zion design temperatures and pressure. Temperatures in excess of $500^{0}F$ under steam conditions and pressures of 280 psi have repeatedly shown no loss of seal integrity. Leak rates of 1.0×10^{-8} std cc/sec throughout these conditions have been maintained. As for radiation exposure, the seal is completely unaffected by exposures of 6.4×10^{15} Rads.

12.4(3) In the Zion plant, the electrical penetrations are apparently all of the D. G. O'Brien type, utilizing glass hermetic seals around conductors plus epoxy potting compounds and other materials. Limited equipment qualification testing appears to have been done on these penetration assemblies or the individual materials. All equipment testing was apparently conducted for pressure and temperature conditions pertinent to design bases currently reflected in licensee safety analysis reports; no tests appear to be available for conditions now being analyzed for more severe degraded core sequences. For the Zion penetrations, qualification tests employed conditions of 265F to 273F for 49 hours at 46 psig. Although an IE review concluded that there was a "high likelihood" of satisfactory operability under the design conditions, reservations were expressed (even for design basis conditions) that the potting materials might be suspect under high relative humidity, that the tests did not adequately simulate saturated steam conditions, that epoxies and polymer materials have shown aging effects under combined accident conditions of temperature, irradiation, and water/spray chemistry, and that some uncertainty about service life may exist for conductor insulation and jacket materials.

Since many electrical penetrations exist in Zion, to what extent can mean life and standard deviation be relied on to guarantee that no penetration will fail under accident conditions? Since only a few assemblies were tested, what is the confidence level in the available service life data?

Response

Sufficient number of penetrations have been tested to have a high confidence level that all penetrations will perform satisfactorily under accident conditions, since the tests have been conducted, with favorable results, beyond the requirements of the Zion temperature and pressure conditions.

12.4(4) In the Zion plant, the electrical penetrations are apparently all of the D. G. O'Brien type, utilizing glass hermetic seals around conductors plus epoxy potting compounds and other materials. Limited equipment qualification testing appears to have been done on these penetration assemblies or the individual materials. All equipment testing was apparently conducted for pressure and temperature conditions pertinent to design bases currently reflected in licensee safety analysis reports; no tests appear to be available for conditions now being analyzed for more severe degraded core sequences. For the Zion penetrations, qualification tests employed conditions of 265F to 273F for 49 hours at 46 psig. Although an IE review concluded that there was a "high likelihood" of satisfactory operability under the design conditions, reservations were expressed (even for design basis conditions) that the potting materials might be suspect under high relative humidity, that the tests did not adequately simulate saturated steam conditions, that epoxies and polymer materials have shown aging effects under combined accident conditions of temperature, irradiation, and water/spray chemistry, and that some uncertainty about service life may exist for conductor insulation and jacket materials.

If the severe accident temperatures do reach the 400F range, what is the probability of failure of sealants in several penetrations? What is the likely number, nature and extent of such failures?

Response

No sealants are used for the pressure boundary of the electrical penetrations.

Containment and Consequence Analysis

12.4(5) In the Zion plant, the electrical penetrations are apparently all of the D. G. O'Brien type, utilizing glass hermetic seals around conductors plus epoxy potting compounds and other materials. Limited equipment qualification testing appears to have been done on these penetration assemblies or the individual materials. All equipment testing was apparently conducted for pressure and temperature conditions pertinent to design bases currently reflected in licensee safety analysis reports; no tests appear to be available for conditions now being analyzed for more severe degraded core sequences. For the Zion penetrations, qualification tests employed conditions of 265F to 273F for 49 hours at 46 psig. Although an IE review concluded that there was a "high likelihood" of satisfactory operability under the design conditions, reservations were expressed (even for design basis conditions) that the potting materials might be suspect under high relative humidity, that the tests did not adequately simulate saturated steam conditions, that epoxies and polymer materials have shown aging effects under combined accident conditions of temperature, irradiation, and water/spray chemistry, and that some uncertainty about service life may exist for conductor insulation and jacket materials.

Is there any possibility of synergistic effects on sealant failure as a result of the major displacements that occur in the concrete shell?

Response

No sealants are used for the pressure boundary of the electrical penetrations.

12.4(6) In the Zion plant, the electrical penetrations are apparently all of the D. G. O'Brien type, utilizing glass hermetic seals around conductors plus epoxy potting compounds and other materials. Limited equipment qualification testing appears to have been done on these penetration assemblies or the individual materials. All equipment testing was apparently conducted for pressure and temperature conditions pertinent to design bases currently reflected in licensee safety analysis reports; no tests appear to be available for conditions now being analyzed for more severe degraded core sequences. For the Zion penetrations, qualification tests employed conditions of 265F to 273F for 49 hours at 46 psig. Although an IE review concluded that there was a "high likelihood" of satisfactory operability under the design conditions, reservations were expressed (even for design basis conditions) that the potting materials might be suspect under high relative humidity, that the tests did not adequately simulate saturated steam conditions, that epoxies and polymer materials have shown aging effects under combined accident conditions of temperature, irradiation, and water/spray chemistry, and that some uncertainty about service life may exist for conductor insulation and jacket materials.

If internal isolation barriers (seals) fail, what is likely to be the gradient of accident conditions toward the outer barrier? Can any estimate be made of the additional time to failure of consecutive barriers? Can any pressurized zones between seals be maintained in the event of failure of either barrier?

Response

Assuming the question refers to the inside containment bulkhead, of the electrical penetrations, determination of a gradient would be difficult since the type of improbable failure is unknown.

12.4(7) In the Zion plant, the electrical penetrations are apparently all of the D. G. O'Brien type, utilizing glass hermetic seals around conductors plus epoxy potting compounds and other materials. Limited equipment qualification testing appears to have been done on these penetration assemblies or the individual materials. All equipment testing was apparently conducted for pressure and temperature conditions pertinent to design bases currently reflected in licensee safety analysis reports; no tests appear to be available for conditions now being analyzed for more severe degraded core sequences. For the Zion penetrations, qualification tests employed conditions of 265F to 273F for 49 hours at 46 psig. Although an IE review concluded that there was a "high likelihood" of satisfactory operability under the design conditions, reservations were expressed (even for design basis conditions) that the potting materials might be suspect under high relative humidity, that the tests did not adequately simulate saturated steam conditions, that epoxies and polymer materials have shown aging effects under combined accident conditions of temperature, irradiation, and water/spray chemistry, and that some uncertainty about service life may exist for conductor insulation and jacket materials.

Are there any significant ameliorating effects (on penetration assembly and seal life) due to the existence of large heat sinks nearby or are any portions of penetration seal assemblies too exposed to containment accident conditions to benefit from such effects?

Response

Such effects need not be considered since the material makeup of the bulkheads is sufficient to withstand the accident without this additional margin.

Containment and Consequence Analysis

12.5(1) For Zion, as for other reactor plants, it is presumed that large double isolation valves have elastemeric seats which have been qualified for selected design basis accident conditions, that large hatches have double inflatable seals which are similarly qualified, and that, for both valves and hatches, the zones between the double seals are pressurized to prevent leakage. Conversations with manufacturers indicate that commonly used elastomers may not be good for temperatures significantly higher than design basis conditions and that even high temperature alternatives (e.g., fluorosilicone) may not be suitable for steam conditions.

What are the resistances of Zion seat and seal materials to higher temperature (up to 450F) with concurrent adverse conditions of steam, water/spray chemistry, radiation and pressure? What is the average service life under such conditions?

Response

As part of Equipment Qualification Report submitted to the NRC (October 30, 1980) none of the process fluid containment isolation valves were identified to have elastemeric seats, but have stellited seating surfaces. The service life is expected as 40 years + 1 year accident. The containment isolation valves in the main purge lines (42 inch diameter) have been identified as having "Nordel" compound seating material. The manufacturer has stated that this material is qualified to a temperature of $300^{o}F$ and a pressue of 62 psig. In addition, the material will tolerate conditions of 100% relative humidity, sodium hydroxide spray, and radiation exposures of 1×10^{8} rads. We would expect that minor seal degradation on the inboard valve might be expected in the first 24 hours of a core melt event. However, given the pressurization system, any leakage would largely be into the containment for most sequences.

Containment and Consequence Analysis

12.5(2) For Zion, as for other reactor plants, it is presumed that large double isolation valves have elastemeric seats which have been qualified for selected design basis accident conditions, that large hatches have double inflatable seals which are similarly qualified, and that, for both valves and hatches, the zones between the double seals are pressurized to prevent leakage. Conversations with manufacturers indicate that commonly used elastomers may not be good for temperatures significantly higher than design basis conditions and that even high temperature alternatives (e.g., fluorosilicone) may not be suitable for steam conditions.

Are alternative seal materials available that could resist both higher temperature and steam concurrently? What are their probable service lives and related confidence levels?

Response

The isolation valves (for the process piping) have been identified as having stellited seating surfaces (or equal). Resistance to a steam environment is expected to be suitable, as well as to a high temperature environment. The service level is expected to be 40 years + 1 year accident; the confidence level is high.

12.5(3) For Zion, as for other reactor plants, it is presumed that large double isolation valves have elastemeric seats which have been qualified for selected design basis accident conditions, that large hatches have double inflatable seals which are similarly qualified, and that, for both valves and hatches, the zones between the double seals are pressurized to prevent leakage. Conversations with manufacturers indicate that commonly used elastomers may not be good for temperatures significantly higher than design basis conditions and that even high temperature alternatives (e.g., fluorosilicone) may not be suitable for steam conditions.

In the event of failure of an inner barrier, can the pressurized zone be maintained? If it cannot be maintained, what would be the effect on containment isolation integrity? What effect will failure of an inner barrier have upon loads imposed upon outer barrier; what are the time parameters for loads imposed on outer barriers?

Response

The volume in the process piping between the inner and outer containment isolation valves is pressurized with air from the safety-related penetration pressurization system located in the auxiliary building. Failure of an inner barrier (inner valve) would result in in-leakage of air into the containment from the P. P. system unless the containment pressure exceeded 62 psia. Similarly, failure of an outer barrier would result in out-leakage from the penetration volume to the auxiliary building. In either case, containment isolation integrity would not be lost. Failure of either the inner or outer barriers would not impose significant additional loads on the other barrier because of the resulting in or out leakage.

Containment and Consequence Analysis

IV.13(1) In general the approach used to calculate containment structural capability appears sound and well documented. Some of the questions which appear significant in the context of the high estimated failure pressure of the containment building are:

H. Temperature effects in the concrete are dismissed as not important (Vol. 7, Sect. 4, App. 4.4.1). Is this conclusion valid for all accident sequences, partially those in which sustained high containment temperatures are likely? Can more detail be provided on concrete temperature calculations, such as liner to concrete heat transfer assumptions, and film heat transfer coefficients used? (PD, p.2; ZZ, p.3)

Response

The temperature effects on the containment structure can be separated into three categories: structural loads, effects on material prcperties, and effects on liner stresses and strains.

Forces and moments due to thermal loads will be greatly reduced, if not relieved, under the conditions of concrete cracking and reinforced yielding due to pressurization at the ultimate capacity level.

Degradation of material properties was not indicated by the magnitude of the peak temperature of temperature-time histories.

Although, the effect of high temperature on the liner may effect the containment capacity, the containment analysis is conservatively done with and without the liner as a structural member. The liner strains are evaluated to guarantee that the liner remains intact and therefore the containment leaktightness is not jeopardized.

Containment and Consequence Analysis

IV.13(1) In general the approach used to calculate containment structural capability appears sound and well documented. Some of the questions which appear significant in the context of the high estimated failure pressure of the containment building are:

I. Transverse shear stresses should be looked at in more detail at the location of cylinder attachment to foundation mat for potential leak path formation. Thermal effects can aggrevate the stresses. (ZZ, 03.)

Response

Transverse shear at the containment-mat intersection has been investigated in detail (see ultimate capacity report, Section 3) and was not a controlling failure mode. Although temperature will cause transverse shears in this area, the effects will be small due to the cracked state of the containment. (See also response to Question IV 13.1.H). Therefore, the temperature effects are not expected to cause shear to become the controlling failure mode.

It should be noted that in the original design, the meridional prestressing tendons were designed for both seismic and pressurization effects. For a Class 9 accident, seismic loads are not postulated to occur simultaneously with the pressure loads. Also the meridional tendon stresses do not control the ultimate capacity of the containment. Therefore due to this available "margin", the concrete shear transfer mechanism should remain viable.

Containment and Consequence Analysis

IV.13(1) In general the approach used to calculate containment structural capability appears sound and well documented. Some of the questions which appear significant in the context of the high estimated failure pressure of the containment building are:

K. Sargent and Lundy does not really calculate failure of containment in the sense of liner rupture but rather attempts to establish a floor or minimum pressure capability of containment. It would be useful to seek their opinion as to just how the ultimate failure and release of radioactivity would occur. (ANL. C3.19)

Response

As stated in the Ultimate Capacity Report, the analysis was preformed to determine a realistic engineering lower bound ultimate containment capacity to withstand internal pressure.

It is estimated that ultimate failure of the containment structure will occur by rupture of the hoop tendons at mid-cylinder elevation, although release of radioactivity would probably occur via extensive cracking or other failures prior to gross structural collapse. In fact, a 16 square inch opening might well relieve enough pressure to prevent such gross failure. Such a relieving event would be expected to occur at the lower bound capacity value discussed above.

RESPONSE TO SANDIA LETTER REPORT
OF MARCH 5, 1982
TO THE USNRC ENTITLED:
"REVIEW AND EVALUATION OF THE ZION
PROBABILISTIC SAFETY STUDY"

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1.0 INTRODUCTION

The Sandia "Letter Report" (Reference 1) on the Zion Probabilistic Safety Study (ZPSS, Reference 2) is, in general, a careful and detailed review of our work. Indeed, their discussion and incisive commentary has, in several cases, helped us clarify our own thinking and has led us to new insights. We believe that Sandia has made a positive contribution to the process of understanding and using PRA in nuclear reactor safety. At the same time, there are some points on which we vigorously disagree. The Sandia Letter Report is a lengthy document (326 pages) and covers a wide range of topics from the ZPSS. It was a primary source document for the long list of formal NRC questions (Reference 3) which the Commonwealth Edison Company has answered in detail (Reference 4). Therefore, we believe that all important questions raised by Sandia have already been addressed. Nevertheless, we feel that a direct response to the Letter Report is appropriate, if only to illustrate the linkage between our answers to the NRC and the Sandia document. In the process, we will focus on several additional points we find significant.

Selecting a coherent format for responding to this wide ranging report has not been easy. Recognizing that the cover letter (Reference 5) to the Sandia report and the report itself implies that the important findings are included in Chapter 5, Summary and Conclusion, we have selected Chapter 5 as the skeleton format for our response. Each item raised there is addressed and in some cases, additional related items from the bulk of the report are also discussed.

2.0 RESPONSE TO SECTION 5.1 IMPORTANT FINDINGS

Sandia's comments are listed sequentially in this style type and single spaced. Our responses follow in this style type at space and a half spacing as shown here.

> Among the important findings of our [Sandia's] review are the following grouped by topic:
>
> Initiating Events
>
> - The initiating events covered in the ZPSS seem to be relatively complete compared to those addressed in previous PRAs, and their estimates of initiating event frequencies appear reasonable.

No response required.

> Event Trees
>
> - The treatment of the containment spray system (CSS) is questionable. The ZPSS assumes that the CSS can be used throughout an accident in the injection mode rather than having to draw from the sump. They assume that the operator will act to conserve the water in the refueling water storage tank by sparingly operating the pumps and that, if depleted, the tank can be refilled. The assumption regarding the operation of the pumps is at variance with existing procedures made available in this review; no procedure exists for refilling the tank. Given these circumstances, it is recommended that such credit should not be given. This significantly changes the frequency of some risk significant plant damage states. (See paragraph 5.2.1)

In the Zion event tree analysis, no credit was taken for the recirculation mode of containment spray operation. The "C" designator was assigned only to plant states or event sequences in which injection phase spray succeeded. This is not considered to be unrealistic, since the Zion containment can tolerate, even without fan cooler operation, significant pauses in spray operation without approaching overpressure failure limits. Approximately 12 to 14 hours are required to fail the containment in the absence of any safeguards. Refilling the RWST can be quickly accomplished and spray operation can continue on an intermittent basis until either the fan coolers or the RHR recirculation system can be placed into effective operation. Since these actions occur literally hours into such an event and are obvious, and given the emergency support services and personnel available as well as the variety and diversity of means available to accomplish these actions, it is unreasonable to assume that no refill of the RWST occurs for the plant states of interest.

As a result, a low likelihood assigned to failure of operator recovery actions such as the preceeding one is not unduly optimistic. See response to NRC System Analysis Question 1 for further details.

- Operator recovery actions (such as the one noted above) were often assumed to be performed with negligible failure probability. This assumption appears to be overly optimistic.

We disagree strongly with this appraisal. The reasons for assigning a low likelihood of human failure to the previous example are detailed in our response to NRC System Analysis Question 1.

- Core melts caused by overpressure failure of containment (e.g., S_2C type accidents as in WASH-1400) were not considered. However, this would have negligible effect on risk. (See paragraph 2.2)

This scenario was considered. However, it was not considered a core melt sequence because the Zion RHR pumps are designed to operate under such conditions.

Success Criteria

- Success criteria used in the analysis appear to be reasonable and consistent with those used in PRAs of similar plants.

No response required.

Human Reliability Analysis

- The human reliability analysis reflected a diligent and sincere effort to use accepted human reliability analysis methods. Some general problems, however, were recognized. Among them were:

    - undue optimism in the assessment of credit for human redundancy;

    - optimistic assessments of human performance under stress, especially for the case of multiple problems;

The main thrust of the comments on undue optimism employed in the assessment of credit for human redundancy is centered around the ATWS study. That part of the study has been revised since the publication of the ZPSS. The key arguments of the revision are given here.

To establish the basic human error probability (HEP), first, all operators we have observed and interviewed respond to plant trip signals by immediately checking for turbine-generator trip (and decreasing load) and reactor trip (rod bottom lights). This is an almost automatic or "second nature" response with no hesitation (incredulity response) about completing those actions (trip the turbine-generator if it has not tripped, trip the reactor if it has not tripped, and carry out the required actions to shut down the reactor if reactor trip fails). Also, even though ATWS is hypothesized to have potentially severe effects, operators do not seem to be as "nervous" about it as, say, about a large LOCA. The stress level would not immediately be especially high.

Second, as clearly laid out in the recirculation system analysis, three reactor operators (ROs) are in the Zion control room at all times. One is assigned to each unit's panel and the third, the center desk man, immediately responds to the unit in trouble. So even in the first 2 minutes, two operators are available to support the ATWS. The shift engineer (SE) and shift technical advisor (STA), both SROs at Zion, may also be involved within the first 2 minutes. At least one of the two must be in the control room; say he is the STA. Then the SE is most likely there, but may be in an adjacent area or anywhere else in the plant, perhaps as far away as the switchyard or the forebay of the crib-house. From discussions with plant operators, we believe the following discrete probability distribution is a reasonable model of the mean response time for the SE to arrive in the control room:

Time for SE to Reach Control Room (minutes)	Probability
0	0.80
0-1 (0.5)	0.10
1-5 (3)	0.08
5-20 (12.5)	0.02

Let us break the human response into two components: recognizing the failure to trip; and performing the required actions to protect the plant. In the recognition phase, it is only necessary to observe the presence of a trip condition and no actual reactor trip; i.e., no rod bottom lights. High readings on nuclear instruments reinforce this observation. For this phase, we see little or no dependence among the operators and model the situation as low dependence.

From the earlier discussion and the remarks on page 17-9 of the human reliability handbook (Reference 6) for "second nature" responses, it seems appropriate to consider the stress level optimum. The basic human error probability for this situation is 0.003. Then for low dependence, the center desk man's human error probability (HEP) is

$$\frac{1 + 19(.003)}{20} = 0.05$$

Since the STA (and SE if he is in the control room) will not respond as quickly and thus has less time to recognize the ATWS condition, we multiply his HEP by 2; i.e., 0.1. If the SE arrives in the control room within 1 minute, we again double his HEP to 0.2. Therefore, the total HEP for failing to discover the ATWS condition is

$$.8[0.003 \times 0.05 \times 0.1 \times 0.1] + .1[0.003 \times 0.05 \times 0.1 \times 0.2]$$

$$+ .1[0.003 \times 0.05 \times 0.1]$$

$$= 3.0 \times 10^{-6}$$

After acknowledging very broad uncertainty in these results by assuming a lognormal distribution and assigning a range factor of 20, the mean HEP for recognizing the ATWS condition is 1.6×10^{-5}.

The first actions required of the operators, to manually trip the reactor and the turbine, are of a routine or automatic nature. To quote the handbook, "If personnel at a plant indeed have such frequent practice that the tasks in question could be regarded as 'second nature,' the HEPs assigned to the moderately high level of stress will not apply, as the stress level will be closer to optimum." (page 17-9) We expect the manual trip to be attempted immediately, before the real significance of the ATWS condition is appreciated. Nevertheless, because the timing is short, we double the basic HEP for the RO; i.e., 0.006. Then for low dependence, the center desk man's HEP is

$$\frac{1 + 19(.006)}{20} = 0.056$$

As above, we double this to 0.111 for the STA and the SE if he is in the control room and double it again if he arrives within the first minute. Thus, the total HEP for failing to initiate a manual reactor trip is

$$0.8[0.006 \times 0.056 \times 0.111 \times 0.111]$$

$$+ 0.1[0.006 \times 0.056 \times 0.111 \times 0.222]$$

$$+ 0.1[0.006 \times 0.056 \times 0.111]$$

$$= 7.83 \times 10^{-6}$$

Assigning a range factor of 20, the mean HEP is 4.11×10^{-5}.

If the reactor still has not tripped, it is apparent to the operators that a very unexpected condition exists. Despite their extensive training for this situation, we believe the operators will feel high stress as they begin to carry out the ATWS emergency procedure. The first step after attempting the manual trips of the reactor and turbine is to drive in the control rods. If this action begins within 1 minute, it should successfully terminate the ensuing pressure rise. Under these conditions, we assign an HEP of 0.25 to the RO. The center desk man may be closely working with the RO, so we consider this as a case of high dependence with an HEP of

$$\frac{1 + 0.25}{2} = 0.63$$

Because the STA and SE will be delayed in responding, probably until the RO's concern is voiced, we consider them moderately dependent

$$\frac{1 + 6(0.25)}{7} = 0.357$$

but double this value because of the time constraint to 0.71. Remember, though, that the required action is simple. In fact, all the STA really needs to do is say, "why are you not driving rods?" and the event could be terminated. Finally, if the SE (or STA) is outside the control room, we give him no credit in helping the situation. Thus, the total HEP for failing to drive rods within 1 minute is

$$0.8[0.25 \times 0.63 \times 0.71 \times 0.71] + 0.2[0.25 \times 0.63 \times 0.71]$$

$$= 8.59 \times 10^{-2}$$

If we assign a range factor of 10; i.e., the upper bound is 0.859, then the mean HEP is 0.23 for failing to drive control rods given that automatic and manual trip have failed.

If the reactor has not been tripped and inward rod motion has not begun within 1 minute, and if the pressure is successfully controlled by the relief and safety valves, we next look for reactor shutdown by manually deenergizing power to the rods. Here, we assume high stress (0.25) for the RO in deciding to carry out the action, complete dependence for the center desk man (1), high dependence for the STA

$$\frac{1 + 0.25}{2} = 0.63$$

and medium dependence for the SE

$$\frac{1 + 6(0.25)}{7} = 0.36$$

We neglect the SE if he has not returned within 5 minutes. Thus, the total HEP for deciding to disable power to the rods is

$$0.98[0.25 \times 1 \times 0.63 \times 0.36] + 0.02[0.25 \times 1 \times 0.63]$$

$$= 5.87 \times 10^{-2}$$

Assigning a range factor of 10; i.e., the upper bound is 0.59 and the mean is 0.156.

Finally, the procedure specifies that the RO send the equipment operator ("A man") to trip the breakers locally. Although he is not stressed, we double the basic HEP to 0.006 and the total HEP is 0.156 + 0.006 = 0.162.

Other important changes to the ATWS analysis include hardware changes and more careful modeling of physical phenomena. These are fully discussed in Attachment 2 to our response to the BNL Peer Review of the ZPSS (Reference 7).

- personal estimates of operater performance rather than using simple measurements;

Actual measurement of operator performance involved in responding to emergency plant operation is, in most cases, unrealistic. As a result, any in-plant simulation of this kind will usually underestimate the operator reaction/response time. This is mainly due to the lack of emergency stress level, simultaneous interfering events, and competing demands for operator actions in any attempt for this measurement. We believe that expert assessment of operator performance is the only way to include the preceding considerations and to obtain realistic estimates. As a specific example, consider the histograms of operator response over time in Section 1.3.2.2, restoration of offsite power supply of the ZPSS. Much longer (and more realistic) times are shown than one would observe by timing an operator walking through these actions.

- inadequate documentation of the use of expert opinion;

The approach adopted in the use of expert opinion was consistent with the general methodology applied throughout the study. Each scenario was studied in detail to determine the required operator actions, available instrumentaion, conflicting concerns, and available response time in the context of personnel availability, procedural guidance, general training, physical locations of equipment, and historical experience relevant to the scenarios. Zion station and Commonwealth Edison operating, supervising, and engineering personnel were consulted for their expert opinions relating to the scenarios. This information was then used to develop the histograms used in the study, which are considered to be the best format for expressing the available state of knowledge regarding these scenarios. In addition, all the histograms thus developed were properly documented in the appropriate sections of ZPSS. In view of the preceding arguments, one will disagree with the statement "the method used was merely to ask people their opinions and to calculate some estimate of central tendency--probably the mean" on page 2.5-6 of the Sandia Letter Report.

- optimistic assessments of dependence among tasks done by the same person;

Dependent human errors were assessed for several scenarios in the systems analyses presented in Section 1.5.2. The general dependence relations stated in Section 0.15.2 were used as a consistent basis for these assessments. The specific valve restoration example on page 0.15-5 of ZPSS was not used in any of the systems analyses. Each analyst developed a dependent error model that was compatible with the specific testing, maintenance, or operator action scenarios for his system. We believe that our treatment of dependence does reflect the uncertainties in our state of knowledge and it is neither optimistic nor pessimistic. See response to NRC Human Reliability Analysis Question 6 (Reference 4) for further details.

- apparent nonconsideration of some possibilities for common cause failures from human errors;
- possible insufficient consideration of errors in restoring safety components after test, maintenance, or calibration.

The procedures used in ZPSS to evaluate potential human errors are:

1. Detailed review of all test and surveillance procedures associated with safety related equipment. This review enabled the system analysts to determine the following:

   a. The effect due to testing/surveillance for cases where system lineup is not changed.

   b. For cases where system lineup is changed, determine whether indications are available to detect incorrect lineups after test and whether incorrect lineups could be corrected without operator intervention due to automatic signals, etc. Frequencies of occurrence are then assigned to the possible human errors and to the recovery actions prior to system actuation.

c. In addition, for those cases where human error could result in a misaligned train, the frequency of common cause errors (coupling of errors) which results in misaligned systems was calculated and its contribution was included in the frequency of system failure.

2. Review of post-maintenance testing and return to service procedures indicated the following:

   a. Post-maintenance checkout procedures require the operation of safety related equipment as the final check prior to returning the equipment to operational status.

   b. For equipment in normally operating systems, this check is a full flow test to the system.

   c. For standby systems, the test and surveillance procedures are used, and the effects of human errors associated with these procedures have already been identified.

In light of the preceding procedures, common cause failures from human errors and errors in restoring safety components after test, maintenance, or calibration are systematically identified and evaluated in ZPSS. See response to NRC Human Reliability Analysis Questions 1 and 2 (Reference 4) for a specific example in the reactor protection system.

> Of these issues, the first three are the most significant to the results. In particular, realistic assumptions of operator response to ATWS events involving loss of secondary cooling significantly change the frequency of core melt. (see paragraph 5.2)

Very significant conservatisms were included in the original ATWS analysis in the areas of options for human response and physical phenomena. Sandia's conclusion is based on incomplete consideration of the model. One serious error with respect to ATWS timing was discovered. If that error alone is corrected, an increase in core melt frequency can be calculated. However, when we observe such a change, it is prudent to review the entire model to see if conservative (bounding) assumptions are now driving the results. In this case, they were and therefore, we have completely revised the model as described earlier. In addition to the changes in human response analysis, other important changes to the ATWS analysis include:

- The fact that the Zion PORVs have been modified to prevent leakage. The PORV block valves are now kept open, so manual action is no longer required.

- The fraction of time the PORV must open to control the ATWS pressure rise due to unfavorable moderator coefficient was erroneously given as 0.1 when it should have been 0.01.

- A new branch has been added to account for the fact that most overpressure conditions will not disable the safety injection system. Most now branch to the small LOCA event tree.

Results of the revised analysis show the ATWS contributions to core melt and risk to be much smaller than calculated previously.

Estimation Methodology

- Zion's estimates of maintenance unavailabilities appear to be consistent with Zion Data.

No response required.

- The treatment of uncertainty associated with estimates from existing data sources is inconsistent. Generally, 5 and 95 percent bounds from WASH-1400 were used as 20% and 80% limits at Zion. Notable exceptions to this were Zion's treatment of interfacing system LOCAs, pressure vessel rupture, and pipe ruptures. In all three cases, substantially higher estimates would have been obtained had their general rule been followed. The results are highly sensitive to this assumption.

The failure rate distributions used in ZPSS reflect the study team's state of knowledge. Each case was evaluated individually and the reported distribution was the consensus of the analysts. In the process of quantifying our state of knowledge, the generic distributions were broadened to take into account the evidence that experts tend to over-estimate their knowledge. It must be noted that this broadening was only an initial step and that the resulting distributions still had to withstand the scrutiny of the analyst. It turned out that the use of 20/80 percentiles resulted, in most (but not all) cases, in distributions that the team found acceptable.

In the particular case of interfacing system LOCAs, the results did not agree with the large amount of evidence which is available concerning this failure mode (disc rupture). It is noted that, to date, no valve disc rupture has occurred in either manufacturing facilities (where hydrostatic testing at 150 percent of design pressure occurs) or nuclear power plants and, to the best of our knowledge, no disc rupture has occurred in conventional power plants or process facilities. Because of this, it was felt that, in the absence of detailed experience (number of valves in similar locations in the industry and the exposure hours at pressure for each valve), the distribution from WASH-1400 adequately described our state of knowledge concerning this failure mode.

See response to NRC Estimation Methodology Question 6 and ACRS Question II.2(1) for further details.

- The Bayesian methodology used to estimate accident sequence rates is somewhat oversold, but it does have the positive effect of highlighting the importance of plant-specific data. Where Zion data exist and are used to modify ZPSS's prior to probability distributions, the effect of the prior distributions is generally unimportant with respect to Zion's estimated accident sequence rates. Where Zion data are not available or used, the estimates are quite sensitive to the assumed prior distribution.

We do not think that we have oversold anything. Section 0 of the ZPSS is devoted to methodology and clearly establishes the philosophical foundations of the whole study. One of the reasons warranting such an explicit exposition is that the "Bayesian methodology applied to risk assessment is also new," as Sandia states (page 2.6-1)

The Bayesian methodology is not used only to specialize the generic distributions. It is a much broader framework in which risk analysis is performed. For example, state-of-knowledge uncertainties are emphasized everywhere in the ZPSS, while "point" or "best" estimates are, in general, eliminated, as are other similarly ill defined concepts as "optimistic" and "pessimistic" analyses. The propagation of uncertainties is done through standard methods of combining distributions and in a rigorous way, avoiding the perplexing notion of confidence intervals.

The fact that the data tend to dominate the posterior (specialized) distributions is as it should be. When the Zion evidence is weak, the prior (subjectively assessed) distribution is important and the results are sensitive to it. Again, this is not surprising. What is surprising is that these observations are used to diminish the importance of Bayesian methods. Two obvious conclusions seem to escape the reviewers:

1. Classical methods are useful only as approximate methods when the evidence dominates the posterior distribution.

2. The subjective assessment of distributions is extremely important, but has nothing to do with Bayesian methods. It is an issue that must be addressed on its own right. By using 20/80 percentiles, we have demonstrated that we do care about the prior distributions and we have tried to take into account relevant results from experimental psychology and from our past experience (References 25, 26, and 23 of Section 0 of ZPSS).

In general, the Sandia review of the methodology is done from the point of view that classical methods are the right way of doing things, while "Bayesian methodology is controversial ..." (page 2.6-1), as if classical methods were not equally controversial. An example in which this attitude leads to erroneous results is provided by the discussion on pages 2.6-16 and 17. In classical statistics, the notion of an estimator and criteria for "best" estimators are introduced. This is, essentially, a consequence of the ad hoc nature of these estimators. In the Bayesian framework, the idea of estimators is not used. Bayes' theorem, based on logic, tells us how to handle the data and we do not need any ad hoc criteria.

What Sandia does on page 2.6-17 is to take the criterion of "unbiasedness" (a classical notion) and apply it to an expression that Sandia calls an "estimator," which it is not. Sandia, then, proceeds to erroneously conclude that $\alpha^2 + p^2$, as an estimator of p^2, could be seriously biased. We repeat that $\alpha^2 + p^2$ is simply the mean of the distribution of p^2 in terms of the distribution of p. It is simply a logical result that guarantees consistency. It is not an estimator of p^2.

- The inclusion of the β-factor for accounting for "other," dependent causes of failure is inconsistent. In some cases, such as diesel generator failures, such dependent failures could be significant to the electrical system failure probability.

We do agree that our use of the β-factor numerical estimates is not what it ought to be. We should have compared the values used in individual systems. Even though it is arguable whether the results would have been affected significantly, the credibility of our work would certainly have benefited.

There is a slightly sarcastic tone in Section 2.6.8 where our attitude toward completeness is briefly summarized. Specific counter examples, however, proving that we have erred are lacking. A particular target of the reviewers seems to be the "other" category of failure causes.

This category is not meant to solve everything. We simply acknowledge that we cannot think of everything and ask the question "how likely is it that something that we have not thought of will happen?" One may disagree with our specific numbers, but not with the fact that the category "other" does exist. Would it have been better for us to completely ignore this category? There is a hint in the Sandia review that perhaps we should have, as the following passage from Section 2.6.8 indicates:

> The basis for Zion's assumed personal probability distribution for the β-factor is vague. A typical statement is the following:
>
> > "Most of the observed coupled failures in the industry involved motor-operated or air-operated valves that had to change position on demand. The frequent partial tests and the full annual system test indicate that an unforseen (sic) common cause failure is of low frequency. We express this state of knowledge by taking a β-factor with range
> >
> > 1.0×10^{-3} to 5.0×10^{-2} which yields Mean:
> >
> > 1.4×10^{-2} (p. 1.5-375)."
>
> Of course, and any two beings possessing the same knowledge would do the same (see p.0.4-2). It would have been more straightforward for the authors to say, "We will model explicitly those dependencies we are aware of and deem important, such as by conditioning on electric power, and omit any others, because we feel they have negligible probability and we prefer not to arm-wave them away with a fudge factor.

We believe that to follow this recommendation would be wrong. While we do model those dependencies that we are aware of, we go beyond that by including a frequency for the "other" category. Sandia is, in effect, telling us to set the frequency of the "other" category equal to zero, hardly a defensible position.

The reviewers also seem to be uncomfortable with the reasoning that has led us to this particular distribution for β. Lacking any specific objections to what we have done, all we can say is that this kind of reasoning is used extensively in PRAs when data is not available. In fact, the NRC human reliability handbook, developed at Sandia, is a compilation of human error rates derived in essentially the same way.

External Events

- The seismic analysis was, in general, difficult to review due to lack of clear documentation. The following observations are made:

  - sequences considered included only those events caused by the seismic event; that is, combinations of seismic and nonseismic events leading to core melt were not considered;

In some projects, the inclusion of nonseismic events in the seismic analysis might contribute to a higher frequency of the top event, say core melt. The combination of acceleration frequency sufficient to cause a failure leading to an initiating event, such as loss of offsite power, and the unavailability of some components in a key system due to seismic failure and others due to nonseismic reasons (maintenance, random failure, etc.) would have to be greater than the much more frequent nonseismic causes of loss of offsite power and the nonseismic unavailability of key systems. Further in order for nonseismic unavailabilities to be considered, the logic would have to show that the earliest seismic failures have to be followed by additional nonseismic failures that cause loss of a key system before getting to the top event. This is not the case for Zion. Here, as seen in Table 7.2-3 of ZPSS, the earliest seismic failures after seismically initiated loss of offsite power are the service water pumps, followed closely by the failure of the auxiliary building, interconnecting water piping, and water storage facilities. The loss of any of these leads directly to core melt and essentially the unavailability of additional components for other reasons do not visibly increase core melt frequencies from seismic initiated events.

  - the overall methodology used seems, in general, to be appropriate;

No response required.

- the treatment of the choice of the boundaries of the seismogenic zones and the rate of seismic activity are questionable;

One of the most important factors for seismic hazard analysis is the closest distance between the site and important seismogenic zones. The alternative zone suggested in the Sandia Letter Report (Figure 2.7.1-1) is quite similar to the ZPSS Wisconsin Arch Zone in that the closest distance to the site is several tens of kilometers. Differences between the two zones in terms of their northern, western, and southern boundaries, are of little significance for the Zion site. In fact, all other parameters being equal, the alternative zone (Figure 2.7.1-1) would probably indicate somewhat less seismic hazard at Zion than the ZPSS Wisconsin Arch Zone, because the alternate zone speads seismicity farther to the north, west, and south (away from the site). The ZPSS Wisconsin Arch Zone was drawn using structural geology, as explained in ZPSS Section 7.9.1, rather than relying solely on historical seismicity; the effect on seismic hazard at Zion is small as described below.

The other delineated zone in the ZPSS study (the Wisconsin Arch-Michigan Basin Zone) has been criticized because it cuts across major basement structure and is therefore unlikely. This is correct, and the low relative likelihood of this zone is reflected in its assigned subjective probability (30 percent). The reason for including it at all is to present results for a zone which represents seismicity in the northern midwest, and which encompasses the site. The Wisconsin Arch-Michigan Basin Zone adequately serves that purpose.

A second important parameter in seismic hazard evaluation is the activity rates associated with important zones. Arguments about the location and size of specific historical earthquakes can be avoided by comparing activity rates used in the ZPSS study with those published elsewhere. Such a comparison is shown in the following table.

Zone	Activity Rate for	
	$m_b \geq 4$	$m_b \geq 5$
ZPSS Wisconsin Arch	0.0695	0.0083[1]
Nuttli and Herrmann (1978)[2] Northern Illinois	0.055[3]	0.0068[3]
Street - Northern Illinois (Figure 7.9.1-4)	0.15	0.011

1. Calculated using model b-value of 0.92.

2. Study referenced in ZPSS Section 7.9.1.

3. Corrected to plot cumulative data at lower ends of magnitude intervals.

The rates used for the ZPSS Wisconsin Arch Zone are slightly higher than (but consistent with) those reported by Nuttli and Herrmann (1978) for the equivalent zone, and lower than those suggested by Street (ZPSS Figure 7.9.1-4). The rates for the ZPSS zone were derived by recognizing uncertainties in historical earthquake locations, and by including in the Wisconsin Arch Zone all large events near its boundary. This includes the large earthquakes in northern Illinois reported by Street (Figure 7.9.1-3). The larger rates derived by Street are a result of his larger zone (which includes many small earthquakes in northern Illinois and southern Wisconsin). In the seismic hazard analysis for Zion, the effect of these larger rates (30 percent larger for $m_b \geq 5$) would be counteracted by the larger area of seismicity (more earthquakes to the north, west, and south, as explained above). Therefore, seismic hazard results for the suggested zone would be close to those reported for the ZPSS Wisconsin Arch Zone. This would particularly be the case for higher accelerations (low annual frequencies) which are governed by the larger magnitudes.

In summary, the alternative suggested seismogenic zone for northern Illinois would produce seismic hazard results similar to those reported in the ZPSS. Activity rates used in the ZPSS are consistent with those reported by other investigators.

- the summary of seismicity in the main body of the Zion report does not faithfully restate the conclusions and reproduce the results contained in the appendices (Zion report, Module 6, paragraph 7.9.1);

ZPSS Section 7.9.1 provides a table of sustained-based ground acceleration frequencies with no truncation of maximum acceleration. The fragility analysis is based on sustained ground accelerations and recognizes limits of damage from experienced accelerations. Therefore, the table in Section 7.9.1 had to be modified to correlate with the fragility analysis. Although seismic hazard curves were prepared, calculations were based on the discretized values of Table 3 in Section 7.9.1, modified. This is discussed in greater detail in the response to NRC Seismic Fragility Question 1 (Reference 4).

- the imposition of an upper bound of effective peak acceleration is unusual and, if relaxed, would probably lead to moderate increases in final mean seismic risk estimates;

Experienced limits of structure and equipment damage from earthquakes of various magnitudes indicate that there are upper bounds in the maximum ground accelerations that probably relate to limits in the earthquake energy and frequency contents. This suggests that upper bounds should be placed on predictions of accelerations in relation to maximum earthquake magnitudes from which the accelerations are predicted. This is discussed in greater detail in the response to NRC Seismic Fragility Question 15 (Reference 4).

- the definition of damage effective ground acceleration used in the analysis may not be appropriate for electrical and mechanical equipment, buried pipe, and equipment which depends on functional operation as opposed to ductile strength capacity.

Effective peak ground acceleration (EPA) was chosen as a descriptor for the fragility curves developed by SMA because it was considered to most appropriately characterize the majority of seismically induced failure modes. Some modes of failure such as those resulting from impact, etc., could instead be characterized by parameters such as relative displacements. However, such an approach using different parameters is difficult to implement in the overall risk assessment. It is difficult to rank the various failure modes as to order of occurrence if different parameters are used, and the seismic hazard curves must be developed in a manner consistent with the fragility curve. Obviously, the development of seismic hazard curves in terms of a descriptor such as relative displacements for specific structures or equipment items is not considered practical.

For structures and many items of equipment, EPA is considered the most valid descriptor, even for items where functionality is involved. For modes of failure involving very little inelastic energy dissipation, very little or no credit is taken for ductility reduction in response. Such modes of failure may involve elastic buckling, for instance, or a localized failure in a structure where very little system ductility exists.

For buried pipe, the mode of failure, while ductile, is based on the maximum strain in the soil (together with anchor point motions, if significant). The soil strain, however, is dependent on soil velocity rather than acceleration. Normally, maximum ground velocities occur in the frequency range below 2 Hz. It is expected that lower magnitude earthquakes will have frequency contents primarily in the 2 to 10 Hz range so that the velocity would be overpredicted using the median broad frequency spectra used for Zion keyed to an instrumental peak acceleration (IPA). The buried piping for Zion has a very high expected seismic capacity.

For mechanical equipment with failure modes related to functionality, EPA is also considered to be a reasonable descriptor for many items. For instance, shaft binding in valve operators and pumps and fans does not usually occur until elastic limits have been exceeded, and this normally requires several response cycles near resonance to develop malfunction rather than a single acceleration pulse. These mechanical components which possess some ductility that may be relied upon before failure occurs may therefore be considered to behave in the same manner as structures for which EPA is considered as a valid descriptor, even though the equipment mode of failure is functional. This applies to this type of equipment mounted either low in the structure where the equipment is subjected to essentially ground input motion, and also high in the structure if the structure remains essentially elastic. In addition, many of these components have significantly higher fragility levels than the structures so that for equipment mounted high in the structure, the structure fragility will control. Again, the structure capacity is considered to be appropriately characterized by EPA.

Some electrical and a few mechanical equipment items are expected to be acceleration sensitive. Such items as electrical relays, for instance, may trip or otherwise malfunction as the result of a single acceleration pulse. No yielding or other structural damage may occur, (although several cycles of cabinet response may be required) so that in such cases instrumentral peak acceleration (IPA) may be expected to be a more valid descriptor than EPA. Most of these modes of failure are recoverable either automatically or by a manual reset, and many have relatively high acceleration capacities compared to other items required for plant safety.

- Design and construction errors and aging were not considered.

Design and construction errors and aging were considerations in the fragility analysis and were included within the uncertainty expressed on the median capacities assigned. This is discussed in greater detail in the response to NRC Seismic Fragility Question 11 (Reference 4).

- The assumption of lognormal distribution for all variables needs further justification.

Lognormal distributions for the variables considered in the fragility analysis are as valid as other possible distributions, particularly if limited credibility is given to the "tails" of these distributions. For greater discussion, see the response to ACRS External Events Question VII.2 (Reference 4) and to the next question in this section.

- The estimate of the mean frequency of core melt due to seismic events appears reasonable (assuming proper resolution of the questions that have been raised). However, changes to the tails of the probability density function for core melt are expected to be substantial.

The lognormal distribution can be expected to provide reliable results through the majority of the fragility curve so long as one is not primarily interested in the tails of the distribution below 1 to 2 percent. As discussed in the report developed by SMA for seismically induced failures in support of the Zion Probabilistic Safety Study, use of the lognormal distribution for estimating very low failure fractions is considered to be conservative. The degree of conservatism introduced into the probability of release is dependent not only on the conservatism in the fragility description but also on the seismic hazard description at the low seismic levels. However, if the occurrence of the seismic hazard for low input levels is large enough, it is apparent that very low level earthquakes can govern the seismic induced release.

This is considered unrealistic for engineered structures and equipment found in nuclear power plants. For low level earthquakes, it is expected that below some threshold, there is virtually no chance of failure due to seismic excitation. Material strength data, for instance, normally does not fall to very low values compared to the median value but instead normally exhibits some lower bound. Other variables such as damping also indicate both lower and upper bounds which are not zero or infinite damping. Extensive studies have been conducted to develop response spectra from available earthquake records and while dispersion exists about the median values, spectra with essentially zero or infinite response do not occur. For these, as well as other variables contributing to the seismic fragility of a given structure or component, it is apparent that some lower and upper bound cutoffs on the tails of the dispersion exist. Since the overall fragility curves are based on a combination of these variables, it is expected a threshold exists below which no failure will occur and this is supported by experience. Although quantitative data is lacking, this threshold value is expected to be approximately minus two lognormal standard deviations using the "best estimate" or composite fragility variability. The composite lognormal standard deviation, β_C, is expected to provide the most realistic basis for the cutoff rather than randomness or uncertainty since the composite value combines the effects of both dispersions. A similar cutoff may be expected to exist on the upper bound of the frequency of failure. In this case, plus three lognormal standard deviations based on the composite fragility curve is expected to provide a more accurate estimate.

It is therefore recommended that for the Zion PSS that where a significant contribution to risk arises from very low level earthquakes, that the above truncation be applied in order to eliminate any excess conservatism introduced by the use of the lognormal distribution at very low frequencies of failure. An example is shown in Figure 1 for a typical fragility curve. In this example, the median effective peak ground acceleration for failure is approximately 1.1g with a random lognormal standard deviation (β_R) of 0.12 and lognormal standard deviation associated with uncertainty (β_R) of 0.20. The "best estimate" or composite standard deviation (β_C) is therefore approximately 0.23. Using the approach discussed above, the cutoff ground acceleration associated with minus two lognormal standard deviations is approximately 0.69g and intersects the composite fragility curve at a frequency of failure of approximately 0.02. The upper bound cutoff at plus three standard deviations occurs at approximately 2.2g.

Quantitative data is not available to support any variation of dispersion resulting from uncertainty as a function of frequency of failure. It is difficult to imagine, however, that the dispersion would be smaller at frequencies of failure at the very low or very high levels compared to the 0.5 range. Therefore, it is not recommended that the shape of any of the fragility curves be modified, but rather that the simple minus two and plus three lognormal standard deviation cutoffs be applied to any confidence level fragility curve.

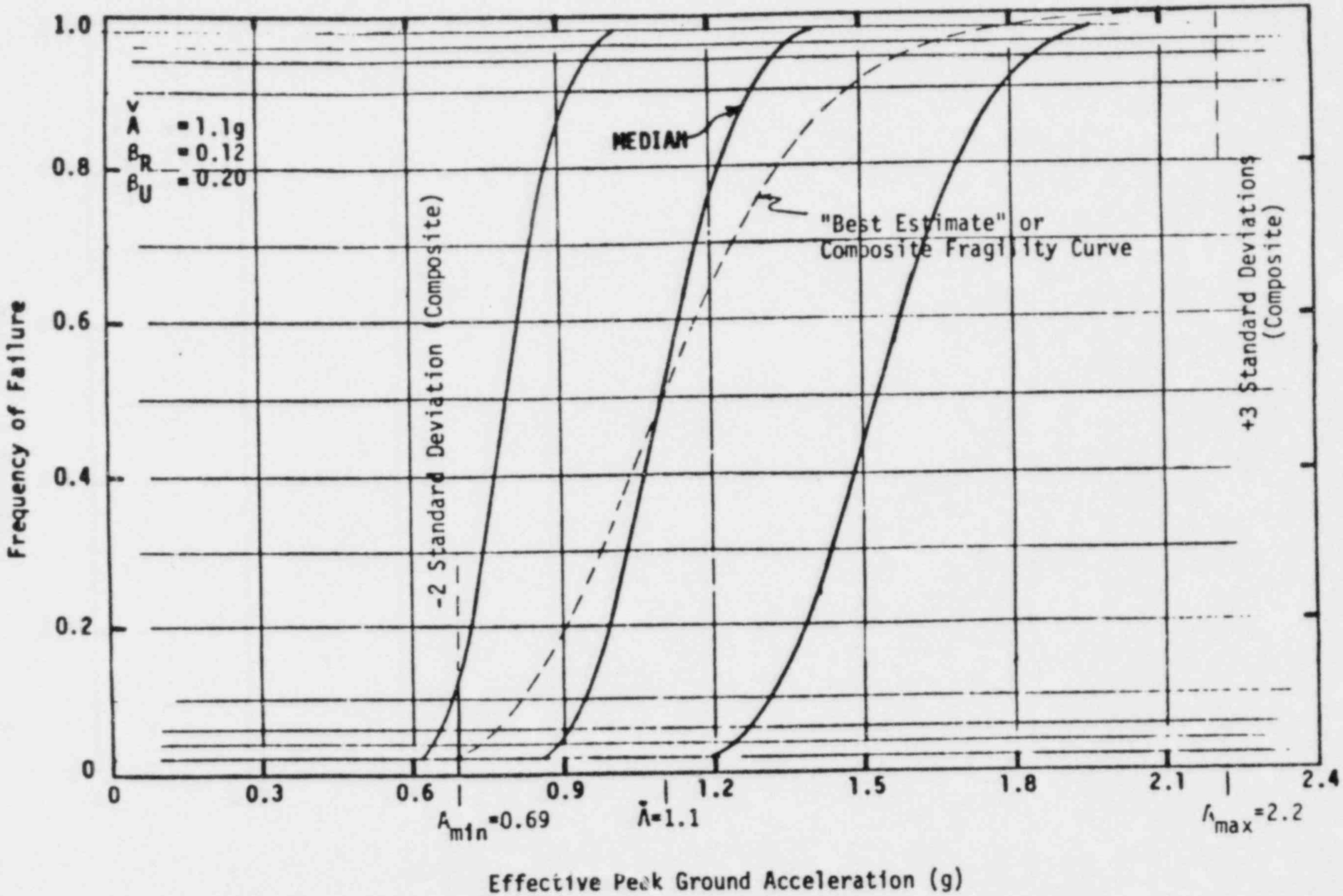

FIGURE 1. TYPICAL FRAGILITY CURVE

- Because of the need for subjective input for probabilistic studies of seismic hazard, and the resulting large uncertainties, the results of these studies are more appropriate for making relative comparisons of seismic hazard than determining absolute values.

As stated in the methodology section of the ZPSS, the approach taken by the ZPSS study team is to quantify the state of knowledge about the scenarios identified. Due to the nature of the analysis, expert opinion and engineering judgment are used in all situations along with whatever data is available. In the case of seismic analysis, the result is a probability curve against the frequency of core melt. This curve quantifies our uncertainty on the basis of all information available. This curve quantifies the risk therefore, and may indeed be said to "be" the risk. It is acutally the absolute "value" of risk for the plant at hand. Of course, it is useful also to compare these curves for different plants to obtain an idea of the relative risk at the different plants.

> The methods used in the treatment of fire were generally acceptable. However, fires in the control room, cable chases, or diesel generator areas were not considered. Consideration of control room fires could approximately double the estimate of core melt due to fires, although not substantially increasing the overall frequency of core melt.

We are pleased that the methodology is found acceptable. Furthermore, we are not surprised that concerns are expressed about the possibility of fires in other locations. A complete fire risk analysis ought to consider all the locations in the plant. Such an effort, however, would be very time consuming. We therefore decided to do a preliminary screening of the locations and then to concentrate on the ones that were judged to require a detailed investigation. We admit that we could have done a better job documenting our preliminary screening. The following are the reasons that led us not to investigate in detail the control room, cable chases, diesel generator areas, and the containment.

Control Room

- The control room is directly above the inner cable spreading room and the instrumentation and controls on the control panels follow a configuration similar to that of the inner cable spreading room cable routing.

- The most critical area of the control room is the panel controlling the safety equipment and the panel controlling the charging pumps and the PORVs. A fire in these cabinets would have the same impact on the plant as a fire in area Z in Figure 7.3-7 of Section 7.3 of the ZPSS. Thus, the turbine-driven auxiliary feedwater pump would remain operable in the case of a control room fire.

- The operators would detect any fire almost immediately and attempt to extinguish it in a short period of time.

- The ventilation system for the control room is designed such that it takes suction from the inside of the control panels, passes the air through filters, and then discharges it into the room. Thus, not all control room fires would lead to an evacuation of the area. It is judged that only a small fraction of control room fires would lead to an evacuation of the control room.

- The power breakers of all the pumps can be closed locally, thus bypassing the control room. This can be achieved even in the case of a loss of DC power.

- All the valves are in their safe position. A hot short causing them to move to a nonsafe position is deemed to be of small conditional frequency given a control power fire. These valves are all outside the containment and can be manually aligned to their safe position if needed.

- A fire in the electrical panel (the panel controlling the electric power system), in the worst case, would lead to a loss of AC power to the affected unit.

Because of these reasons, we believe that the analysis of fires in the control room would lead to a change of the low tail of the frequency distribution of core damage due to fires in the cable spreading room. This, however, would not lead to doubling the estimate of core melt due to fires.

Sandia suggests that the estimate of 3.9×10^{-6} per reactor year for core damage developed by Gallucci and Hockenbury (<u>Nuclear Engineering and Design</u>, Vol. 64, p.135, March 1981) be added to our mean value of 4.6×10^{-6} ry^{-1}. We do not believe that this is appropriate for the following reasons:

1. This estimate was derived for a "generic" BWR.

2. Gallucci and Hockenbury's methods are different from ours. Their reasoning for assigning frequencies to various events is not detailed in the cited reference.

3. The authors themselves state in the abstract of their paper: "Conservative estimates of core-damage probabilities due to fire were obtained; application of the methodology to a particular BWR including specific knowledge of cable locations, fire retardants, detectors, etc., would result in considerably lower probabilities."

4. Sandia adds the Gallucci estimate to our mean value (the result being approximately 10^{-5} ry^{-1}) and states: "Hence, as a revised estimate one could state that the 50 percent probability of core melt from the fire may be on the order of 10^{-5} per reactor year, (compared to the total frequency of core melt given as 6.7×10^{-5})." It is wrong to say that there is a 50-percent probability, etc., because our estimate is a mean value, not a median. In fact, our median is smaller (9.4×10^{-7} ry^{-1}).

Cable Chases

There is only one area in the Zion station that resembles a cable chase. It is an inaccessible underground tunnel containing cables enclosed in conduits. Since the area is inaccessible and the cables are enclosed in conduits, the fire hazard of the area is deemed to be much smaller than the areas analyzed in Section 7.3 of the ZPSS.

Diesel Generators

We agree with Sandia's statement (Section 2.7.2.2) that:

> In the case of Zion, the two plants are provided with five diesel generators--two each, plus one swing generator. Combined with fire suppression systems and adequate separation, it is unlikely that fire in a diesel generator area would prevent safe shutdown. (At Zion, the diesel generators are separated by 3-hour rated barriers, and each room provided with CO_2 suppression systems.)

Containment

The fire areas within the containment were found to contain a smaller number of vital components than the critical areas analyzed in Section 7.3 of the ZPSS. It was found that a small LOCA cannot occur from a containment fire and safe shutdown can be achieved by the components outside the containment.

The reactor coolant pumps are located inside the missile barrier and therefore are well separated from the containment fan coolers and RHR related equipment which are located outside the missile barrier. There are no cables or other components above the reactor coolant pumps. Therefore, the PORVs are not judged to be readily exposed to an RCP fire. Thus, it is concluded that RCP related fires do not pose a threat to safe shutdown and they can be considered as one of the causes leading to reactor trip.

The following comment does not appear in the summary, but in Section 2.7.2.2.

> The model of fire propagation from one cable tray to another is inadequate and the calculated values for propagation times are significantly above what has been experienced from cable tray fire tests performed at Sandia.
>
> The model is indeed in need of significant improvement, but there seems tc be a large margin of conservatism built into the assumptions.

It is not clear to us what portion of the fire propagation model was deemed to be inadequate by the reviewers, or indeed what they would consider to be an adequate model. The model used was constructed to conservatively incorporate the physical characteristics of fire in an explicit calculation of the likelihood of fire growth in a general cable tray array. Our uncertainties in the model's predictions are largely due to our uncertainties in the characteristics of the initiating fire, our uncertainty in the physical properties of the cables at the Zion plant, and to the relatively primitive state of the art in cable tray fire modeling; these uncertainties were handled explicitly throughout. The conservatism of the model and the quantitative treatment of uncertainties were both recognized by the reviewers. While we agree with them that improvements can be made in the model, we also believe that the model used was entirely adequate for our purposes.

To address in detail the reviewers' statement concerning the discrepancy in predicted and experimental propagation times would require a lengthy discussion of our probabilistic model for fire growth as well as the physical model. Instead, we summarize the reasons underlying the wide distribution of calculated propagation time (whose large tail and mean may be the source of the reviewer's concern) in the following three points:

1. The event {no propagation given a fire in a critical location} was conservatively neglected. Very long fire propagation times were assigned to scenarios where the physical fire propagation model actually predicted that fire propagation would not occur.

2. Most of the longer propagation times are associated with fires initiated with minimal external fuel. As noted in point 1, the assumption that these fires will certainly spread, although possibly after a long time, is conservative. The shorter times are associated with fires initiated with more highly combustible transient fuels.

3. The long propagation times are also a function of the cable tray vertical separation distance. If the trays are brought closer together, the propagation times will certainly decrease.

- The treatment of floods does not appear to be appropriate for the purposes of a probabilistic risk assessment.

The approach adopted in the Zion flooding analysis was, first, to review and evaluate all the possible sources of flooding. The detection and isolation provisions were then considered. Finally, system logic as well as structure and component fragilities were examined. However, due to the low frequency of occurrence and the reliability of flood mitigating features, no flooding scenarios with significant impact on core melt frequency or plant risk are identified.

Accident Sequence Analysis

- Differences in the treatment of offsite power recovery and turbine driven pump maintenance unavailability would substantially increase the estimated frequency of the sequence "turbine trip due to loss of offsite power: loss of all AC power, failure of auxiliary feedwater." This sequence currently dominates the risk due to internal events. (see paragraph 5.2.1)

The reviewers' comments regarding the time for offsite power recovery are based on the data collected by R. F. Scholl (Reference 8). This data led the reviewers to the conclusion that the Zion recovery time analysis is very optimistic compared with the generic data for offsite power restoration. However, there is a significant amount of evidence available from the generic data to indicate that the Zion power recovery time should be much lower than the plant population. It must also be noted that the Zion recovery model has been developed in response to a specific set of plant conditions which could have an important effect on the relative priorities assigned to offsite power restoration and, if they were to occur, could significantly reduce the recovery times reported for some sites. We believe that the application of generic data must be consistent with its use and interpretation in the plant models.

The generic plant operating experience provided an important information input to the development of the Zion power recovery model. However, it was also recognized that many of the generic plant sites have local transmission grid and switchyard configurations which are much different from those at Zion and which could significantly affect the time required to restore power after a major disturbance. Most of the sites experiencing extended power outages are located in areas subject to regional grid instability problems, unique supply line routing, or have experienced localized outages directly attributable to some characteristic of the site.

The Zion site is somewhat unique among the population of plants because its switchyard is an important intertie point for the Commonwealth Edison transmission grid. It is also important to note that the Zion switchyard is directly adjacent to the plant buildings and that the switchyard equipment is normally operated by onshift plant personnel.

The Zion Probabilistic Safety Study power recovery analysis is applied to plant conditions in which at least two of the three diesel generators for the affected unit are failed and no core cooling equipment is available (e.g., the turbine-driven auxiliary feedwater pump has also failed). This situation is much different from all the events summarized in the generic data base, because in virtually all cases, onsite power was available from one or more diesel generators, and there was no immediate concern for loss of core cooling or loss of reactor coolant inventory. Since there has never been an instance of offsite power failure, severe degradation of onsite power supplies, and loss of core cooling capability, none of the generic data is directly applicable to the power recovery scenarios for the Zion Probabilistic Safety Study.

The Zion offsite power recovery time distribution has two primary components, both of which are based on an analysis of the Zion plant site and evaluation of Commonwealth Edison transmission system experience. The operator response time model accounts for normal shift manning, conflicting concerns about the failed diesel generators and the restoration of normal power, standard plant practice for switchyard operations, and expected operator performance under conditions not covered by specific procedures but involving operations familiar to plant personnel. As with many scenarios modeled in the study, it was felt that this type of analysis would provide a more realistic estimate of expected performance than could be obtained from the evaluation of rather sketchy generic data from sites which are significantly different from Zion, under conditions very different from those being modeled in the study. Please refer to response to NRC System Analysis Question 11 and Estimation Methodology Question 5 for further details.

Regarding Sandia's comment on turbine-driven auxiliary feedwater pump maintenance unavailability, the ZPSS employed a plant specific maintenance data base that covered a time period through 1980. Several unusual maintenance events in both units are recorded in late 1979 and early 1980 in which repeated maintenance outages occurred for a turbine-driven AFWS. It is not clear if this represents an increasing trend in unavailability or an anomaly due to the increased emphasis placed on AFWS pump testing following the TMI-2 incident. In either case, these outages are responsible for the higher unavailability of AFWS pumps at Zion compared with most other pumps. Collection of data was suspended so that the study could be completed.

The effect of the recent Zion AFW system failures on the ZPSS AFW analysis is described below.

1. Sneak Circuit in the Motor-Driven AFWS Pumps Created by a Design Modification in 1981 (See I&E Information Notice 8201). During the performance of the AFWS analysis of ZPSS, detailed investigations

into the control circuits for all AFWS components were undertaken. Based on the depth of this review, we feel that if this potential failure path had existed at the time of the ZPSS, it would have been discovered and corrected.

2. Common Mode Miscalibration of AFWS Low Suction Pressure Trip Switches (See letter, K. Graesser to NRC; with R081-30 (December 24, 1981), R081 (January 5, 1982), R081-32 (January 8, 1982) Docket 50304). This potential error existed at the time of the ZPSS and should have been included in the system analysis as a common cause contributor to system failure due to: (1) miscalibration, and (2) nondiscovery. Further analysis may be necessary to determine the exact impact of this error; however, because of the allowable time for recovery from this type of failure (30 minutes), this event should have very little effect on plant risk.

- The treatment of interfacing system LOCA is highly sensitive to the treatment of WASH-1400 estimates. The discussion of the model for this sequence and the selection of bounds contrary to Zion's normal assumptions are not adequate.

This question was addressed earlier in response to Sandia's second comment under Estimation Methodology; see page 21.

- Under the assumptions used in the Zion analysis, a less optimistic treatment of operator response to an ATWS event increases the core melt probability substantially. Given this, ATWS events would dominate core melt.

Sandia concurs with the Brookhaven reviewers concerning the ZPSS ATWS analysis (Reference 9). BNL had correctly pointed out that the peak pressure following an ATWS occurs in about 2 minutes rather than 10 minutes as modeled in the study. However, we disagree strongly with their use of the handbook of human reliability analysis to requantify the necessary operator action. They used pages 17-20 and 17-24 which provide human error rates for the time immediately following a large LOCA and generic performance rules to be used in the absence of more specific information. Due to the differences in stress level and incredulity/ second nature response involved in large LOCA and ATWS events, we feel it is inadequate to use the preceding human errror rate in this case. In addition, some important changes have been made since the original analysis was performed. This includes the fact that the Zion PORVs have been modified to prevent leakage. The PORV block valves are now kept open, so manual action is no longer required for pressure relief. Furthermore, as stated in the Zion study, the ATWS analysis included overwhelming conservatisms. We have now revised that analysis and the results show the ATWS contributions to core melt and risk to be much smaller than calculated previously. Please refer to response to ACRS Plant Analysis Question III.3.(1) (Reference 4) for further details and our response to an earlier Sandia comment under human reliability analysis.

- Zion's treatment of AFWS motor-driven pump failure is noticeably more optimistic than Zion data alone would indicate.

The analysis that leads to the conclusion stated in this comment is contained in Section 2.6.3. In effect, the reviewers are using one of the many possible ways for checking the effect of the prior distribution on the posterior distribution. The basis of their investigation is to determine a fictitious number of failures, f_{PRIOR}, over a fictitious number of hours, T_{PRIOR}, which are determined by

$$f_{PRIOR} = f_{POST} - f_{ZION}$$

$$T_{PRIOR} = T_{POST} - T_{ZION}$$

where f_{POST} and T_{POST} is a similar pair determined from the posterior distribution by the method of matching moments, and f_{ZION} and T_{ZION} is the actual Zion evidence.

The results of this exercise are listed in Table 2.6-1 of the review. As the reviewers correctly note

> The contributions of the priors to the final results vary considerably.

There are several cases in which f_{PRIOR} is negative; i.e., the effect of the prior is to reduce the actual number of failures observed at Zion. This is what happens in the case of AFWS motor-driven pump failures, where $f_{ZION} = 4$ and $n_{ZION} = 462$, while $f_{PRIOR} = 1.3$ and $n_{PRIOR} = 80$ (the number of demands plays the role of the number of hours in this case).

We view this method as a clever exercise that can help the analysts check the implications of their prior distributions along with other checks that are routinely used; e.g., the reasonableness of the mean values and the 95th percentiles, etc. We believe that we have demonstrated that we are sensitive to the issues that may arise in choosing a prior distribution. For example, we have broadened some generic distributions, as described in Section 0 (Methodology) of the Zion study.

The role that the prior distribution and the data should play in shaping our state of knowledge (i.e., our posterior distribution) is determined by Bayes' theorem. It would theoretically be inappropriate to change the prior, because the data happens to tend to place the posterior distribution on the tail of the prior. The fact that the prior distribution has a tail (on the high side) means that we allow a small probability to the failure rate being there. It also means that the posterior distribution can be in that range when the plant evidence is strong enough to overpower the prior distribution. All this, of course, is taken care of by Bayes' theorem and we do not need to worry, for instance, about how strong the plant evidence must be.

In the particular case of AFWS motor-driven pumps, we have a situation in which the plant evidence of 4 failures over 462 tests tends to place the posterior on the high tail of the prior distribution. This evidence, however, is not strong enough to completely dominate the posterior distribution. We should also emphasize that the prior distribution does not dominate either. Furthermore, in several other cases, the data tend to shift the posterior distribution toward the low range of the prior, which in effect, tends to artificially increase the number of failures, a fact that does not seem to bother the reviewers presumably because it is "conservative."

Of course, all these arguments come down to the perennial issue of whether the prior distributions are the consensus distributions of all parties concerned. The reviewers recognize this, but do not elaborate too much on it. They state:

> Whether or not the contributions of the prior distributions are fair and just, depends on the actual information contained in the source documents.

All that we can say, at this point, is that the prior distributions in the Zion study indeed represented our state of knowledge at the time that they were developed. We also acknowledge that others may disagree and we are prepared to discuss specific concerns.

- The ZPSS scope of analysis, as far as completeness of the systems analysis is concerned, is consistent with PRAs done in the past.

No response required.

- Core melt/systems interactions are not addressed in portions of the report reviewed. Of particular concern is the assumption that containment fans will continue to operate in a post core melt environment, although their design limits will have been exceeded. This is a very important assumption when considering risk for Zion, particularly in light of comments made earlier with respect to containment spray operation. Given the importance of this issue and the lack of adequate information to resolve it, we have presented in Section 5.2.1 plant damage state frequency estimates assuming the fans could work and assuming they cannot work.

The Zion plant was examined to identify significant potential for core melt/systems interactions. A detailed analysis was performed to determine the fan coolers operability under degraded core conditions. During a degraded core event, the fan coolers might be challenged by aerosols generated during core degradation and core/water/concrete interaction, as well as elevated temperature and pressure in the containment. Even under these adverse conditions, the analysis showed that the fan coolers would not fail. This analysis is given in the response to NRC System Analysis Question 2 (Reference 4).

- The treatment of station blackout does not appear to adequately treat common mode diesel failures and common mode DC power failures. Depletion of station batteries in long-term blackout situations does not appear to have been addressed.

The "station blackout" was defined and evaluated in the ZPSS as the loss of all AC power to one unit, given a loss of offsite power supply to both Zion units. Under these conditions, it is very likely that one or more of the diesel generators for the other unit are operating normally to supply their associated loads, including the battery chargers. Each DC bus at Zion can be cross-connected to the corresponding DC bus for the opposite unit. If all AC power were lost for a given unit, use of the DC bus crossties could maintain the DC buses and the AC instrument buses energized from the battery chargers for the opposite unit. In addition, the Zion emergency operating procedures for loss of offsite power specify that the major nonessential loads supplied by the batteries should be disconnected as soon as practical to conserve. As such, with the consideration of possible actions to recover onsite AC power (e.g., restarting or recovery of failed diesel generators) and to increase the availability of DC power, we believe that depletion of station batteries in long term blackout situations is very unlikely and thus has no significant effect on either the core melt frequency or the plant risk. Please refer to the response to NRC System Analysis Question 12 (Reference 4) for further details.

- Reactor coolant pump seal LOCAs are not explicitly treated in the initiating events, but this does not appear to be significant.

The reactor coolant pump seal LOCAs are implicitly covered in subcategory b, "miscellaneous small LOCAs" of category 3, "Small LOCAs."

At the time the data base for the Zion PRA was compiled, there had been no reported reactor coolant pump seal leakages which had resulted in a reactor trip and which were large enough to be categorized as small LOCAs. Such an incident occurred at Arkansas 1 in May 1980.

Because there was plant specific evidence for Zion for this initiating event, a change in the population data of this magnitude snould make very little difference in the results.

- Assumptions regarding feed and bleed may affect the core melt frequency, but are not expected to influence risk.

In the post-TMI licensing environment, Westinghouse performed a variety of generic analyses to demonstrate the inherent safety of Westinghouse designed PWRs. Some of these analyses included loss of coolant accidents induced by the complete loss of all feedwater. The complete loss of all feedwater will result in a small LOCA through the pressurizer PORVs as the secondary decay heat removal degrades. These analyses were reported in WCAP-9600, "Report on Small Break Loss of Coolant Accidents in Westinghouse NSSS Systems" and WCAP-9744, "Loss of all Feedwater Induced Loss of Coolant Accident Analyses." The purpose of those analyses was to investigate the minimum operator action time in the event that no feedwater was available for the removal of decay heat. WCAP-9915, "PORV Sensitivity Study for LOFW-LOCA Analyses," reports the results of a number of sensitivity studies to this type of analysis and provides an indication of the relationship between PORV capacity, steam generator dryout time, and operator action time.

The loss of feedwater induced LOCA is a small LOCA through the pressurizer PORVs. The analysis method utilized the small break LOCA evaluation model, WFLASH, which is described in WCAP-8200 entitled "WFLASH: A Fortran IV Computer Program for Simulation for Transients in a Multi-Loop PWR." The conservative Appendix K inputs were used, such as 120 percent of ANS decay heat. WFLASH has been verified and approved by the U.S. Nuclear Regulatory Commission for use in conservative evaluation of small LOCAs in a Westinghouse PWR. Numerous sensitivity studies have been performed to verify the inputs of the WFLASH model for small LOCAs.

The loss of feedwater induced LOCA analysis was initialized so that the steam generator water level used in WFLASH resulted in a steam generator dryout time which matched calculation by an accepted independent process. The WFLASH loss of feedwater LOCA analysis uses conservatively low liquid and vapor heat transfer coefficients and secondary metal heat capacity is not modeled to reduce secondary heat removal capability. The pressurizer PORVs were modeled as pressure dependent break flow paths to the containment. The mass flow rate through the PORVs is based on the rated PORV flow which is used to determine an appropriate flow area for use with the evaluation model break flow rate. This methodology will result in a conservative short operator action time.

Westinghouse has developed unambiguous procedures for the recognition and consequential mitigation of a complete loss of feedwater. This procedure was submitted for NRC review in December of 1981. It has been included in the emergency response guideline and is being integrated into the critical safety function monitoring. Westinghouse has also held training seminars on this issue.

The loss of fluid test facility near Idaho Falls, Idaho, performed experiment L9-1/L3-3 on April 15, 1981. This experiment was designed to examine the decay heat removal capability in the complete loss of feedwater scenario. The L9-1/L3-3 loss of fluid test terminated the main feedwater and waited until there was a high hot leg temperature before tripping the reactor. This provided a very conservative inventory on the secondary side, and was more limiting than the Zion criteria in that respect. Additional decay heat removal capability extended the time scale of events just as a longer dryout time would. Most of the transient behavior predicted in WCAP-9600 and 9744 was observed. The pressurizer PORV showed decay heat removal margin and successful operability under this scenario condition. Westinghouse believes a detailed modeling of the loss of feedwater LOCA scenario would also show additional operator action time margin for Westinghouse PWRs. The WCAP-9600 and 9744 calculations were designed to be conservative.

Westinghouse does not believe that this test should be used to verify the WFLASH model due to loss of fluid test atypicalities, but believes the WFLASH model has sufficient verification for conservative analyses of this nature.

Based on the Westinghouse analyses, sensitivity studies, and the additional decay heat removal capability demonstrated in the loss of fluid test facility, Westinghouse believes that primary bleed and feed is a viable mode of decay heat removal in the event of a loss of all feedwater in the Zion nuclear units. In fact, the reviewers concur with our assumption as stated on page 4-4 of the Sandia Letter Report:

> It should be noted that we feel that feed and bleed core cooling should be given credit as was done in the ZPSS. Recent tests at the LOFT facility suggest that feed and bleed is a biable core cooling option.
>
> - Alternate assumptions regarding the need for pump cooling during injection following an accident could significantly influence risk. (see paragraph 4.8)

The statement in the HPIS analysis (see ZPSS Section 1.5.2.3.1.2.4) concerning pump cooling is incorrect. Component cooling water is required for the oil cooling of charging pumps and safety injection pumps. Loss of the cooling water will result in pump failure in a short period of time. However, no significant impact on either core melt frequency or plant risk is expected. Please refer to response to NRC System Analysis Questions 2 and 9 (Reference 4) for further details.

REFERENCES

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4. "Response to U. S. NRC and ACRS Review Questions on the Zion Probabilistic Safety Study," Commonwealth Edison Company, September 1982.

5. Letter from W. R. Cramond, Sandia National Laboratories, to S. Newberry, USNRC, dated March 5, 1982.

6. Swain, A. D., and H. E. Guttmann, "Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, October 1980.

7. "Response to the BNL Peer Review of the Zion Probabilistic Safety Study," Commonwealth Edison Company, September 1982.

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RESPONSE TO THE "BNL PEER REVIEW OF THE
ZION PROBABILISTIC SAFETY STUDY"

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1. INTRODUCTION

The "BNL Peer Review of the Zion Probabilistic Safety Study" (Reference 1) represents the first extensive review of our work (Reference 2) funded from sources outside the project. We are grateful to the BNL staff for the insights we have gained from their comments and commend them for the depth of review provided in such a short time. Complementing the Zion Study Review Board's general level direction on the suitability of our methods and the reasonableness of our results, the BNL review raises detail level questions concerning specific models and calculations.

Careful consideration of the questions raised by BNL leads us to the following conclusions. We find that modification to some of our methods and models would be justified in the interest of enhanced clarity and correctness. We disagree with some of the BNL comments and find that some are rooted in uncritical acceptance of generic data. Finally, we find that none of the concerns lead to significant changes in the calculated consequences of events at Zion. Unfortunately, we also find that the BNL review is lacking any real assessment of the overall quality of the Zion study. We note with some surprise that the reviewers seemingly fail to appreciate, perhaps even to notice the two principal aims of the study: (1) to treat the issue of uncertainty rigorously, and (2) to be site specific.

Past criticisms of PRAs have emphasized the need to quantify, propagate, and display the uncertainties in a rigorous and visible manner. The Zion team has been very sensitive to this requirement and has made it clear that uncertainties are the dominant theme of the study. Section 0 (Methodology) of the study addresses in detail both the philosophical basis and the mathematics of uncertainties in PRAs.

Throughout the study, the frequencies of various events are never single "point" estimates. A probability distribution is always presented, which reflects the state of knowledge of the study team. These probability-of-frequency distributions contain both generic and plant specific information. They are a complete statement of what we know. For communication purposes, we have also reported several characteristic values (percentiles and mean values) that summarize these distributions. This is the appropriate context for evaluating these "point" estimates. In fact, the study has pointed out the dangers of relying too much on mean values (Section 0).

The BNL review does not appear to appreciate these issues. Several short calculations that are presented are done with point estimates and using generic data as opposed to the probabilistic and plant specific calculations in the Zion PSS. It is, of course, appropriate for the BNL review to do such calculations as a rough ball park check on the more exact calculations. We welcome and value them from that point of view and regard the fact that they come as close as they do, generally speaking, as confirmation of our results.

Finally, we wonder at BNL's silence on several advances in the state of the art that the Zion study has made. Several of these are the rigorous handling and displaying of uncertainties that we mentioned earlier; the matrix formulation that is valuable both as a communications tool and a framework for the easy evaluation of changes; the identification and avoidance of certain pitfalls; the rigorous combination of generic and site specific data through the two-stage Bayesian procedure; the suggestion that certain generic distributions should be broadened; the new analysis of degraded core phenomena and containment response; the development of models for situations where none existed before; e.g., for fires; and, the improvement on site modeling. The report of the Lewis committee on WASH-1400 is highly respected because it identifies both the positive and the negative features of the RSS. We believe that the BNL review has been focused on negative criticism, which, as we shall show in the following sections, is very often unjustified.

2. COMMENTS ON SECTION 2, INTERNAL EVENTS

2.1 TREATMENT OF DC POWER

In Section 2.1, page 7, of their comments, the BNL reviewers note that they could not ascertain whether or not the loss of DC power was included in the AC power analysis for the loss of offsite power initiating event. Failure of DC power was included in the AC power analysis under all system operating conditions, although its contribution to the failure of AC power was determined to be negligible. The treatment of DC power is discussed in the development of the AC power failure state equations in Section 1.5.2.2.1.4.5 (page 1.5-192 ff). In this section, it is noted that the DC system was assumed to be in a normal configuration immediately prior to the initiating event, and failures of DC power are evaluated over the system operating period after event initiation. No recovery from DC power failures was analyzed. Failure of power at any of the three DC buses would result in a trip of the associated reactor and would also place the AC power system in a degraded condition due to the loss of switching control power. The bounding treatment of these initiating events and their effects are addressed in Section 1.3.4.13.5 (page 1.3-312).

2.2 LOSS OF OFFSITE POWER DATA

Many of the BNL reviewers' comments regarding the frequency of the loss of offsite power initiating event and the estimated time to recover offsite power are based on data obtained from a report by Raymond F. Scholl, Jr., of the NRC (Reference 3). Mr. Scholl sent us a copy of his report, and our comments on the BNL use of his data are included as item 2.8 of this section. However, we should also provide a few general comments which might serve to clarify the context and application of the loss of offsite power data as it was used in the Zion Probabilistic Safety Study.

The BNL reviewers are correct in their observation that multiple reactors at a single plant site should not be treated independently for the purpose of developing loss of offsite power frequency data (Reference 4). It is true that offsite power disturbances tend to affect the power supplies to all units at a given site, rather than selectively affecting a single reactor. The model for the Zion in-plant electric power system fully acknowledges this fact and explicitly accounts for the common loss of power to both units when evaluating the response of swing diesel generator "0."

The development of the data to be used in estimating the frequency of loss of power events must be consistent with its application in the plant response models. The loss of offsite power event is modeled in the Zion Probabilistic Safety Study as an event which causes a unit trip from power operation and results in a loss of all offsite power supplies to the affected unit. Examination of the loss of offsite power events at several multiple unit sites shows that approximately 35% of these events occurred when at least one of the units was not operating at power. This percentage applies to power failures at multiple unit sites after all units at the site were in commercial service. If one includes the power failures which occurred after one unit was operating, but before all units were completed, the percentage of failures affecting fewer than all the units increases to approximately 50%. The point of the discussion is that the loss of power event is modeled as initiating a transient from power operation. Since a significant fraction of the power failures occur when units are shutdown, the data base should be adjusted to compensate for the availability of the units at the affected site. This also applies to the data for single unit sites, since at least 25% of the offsite power failures at these sites have occurred when the unit was shutdown. Therefore, although the reactors at multiple unit sites should not be treated independently for this initiating event, they should also not be treated as completely dependent. The use of a complete dependency assumption results in an overestimation of the frequency of the loss of offsite power event as it is modeled in the Zion Probabilistic Safety Study.

The data presented in the EPRI NP-801 report (Reference 5) provides a partial solution to this problem, since it reflects only those events which caused a unit trip from power operaton. Events which did not result in a unit trip or events which occurred when the unit was shutdown are not included in the data. Unfortunately, the data is indexed by reactor unit, and it cannot be used directly to provide common site events such as the frequency of offsite power failures which occur when all units at a site are operating at power (e.g., for a three-unit site, a common power failure event would be listed as three separate events if all three units were operating and all three tripped). Therefore, one cannot simply add the number of individual events documented in NP-801 for a multiple unit site and apply this data to the composite site event frequency.

Another important point to recognize when developing data for the loss of offsite power event is that the Zion in-plant power system model explicity accounts for the failures and maintenance of the system auxiliary transformers, which supply offsite power to the units during power operation and after a unit trip. A number of power failure events documented in the generic literature have been caused by single transformer failures. These events should not be included in the loss of offsite power initiating event data base to avoid double accounting for these failures in the Zion model. (They should be included in the offsite power failure data for a model which treats the transformers as part of the offsite grid.) Similarly, failures to transfer in-plant loads to the offsite source are also analyzed in the Zion power system model, and these events should be excluded from the offsite power failure data. (These events are not actual losses of offsite power, but for licensing compliance proposes they are often miscategorized as such in LER summaries.)

During the time since the Zion Pobabilistic Safety Study initiating event data was finalized, Pickard, Lowe and Garrick, Inc., has performed a more detailed review and analysis of the generic plant data for losses of offsite power at BWR and PWR sites. The basic event descriptions used in this analysis were obtained from three sources (References 6, 7, and 8) all of which rely heavily on the plant LERs. However, the information in "Nuclear Power Experience" (Reference 8) often contains much more detail about the circumstances and causes of specific failure events than is available from the LERs, and it allows the data analyst to better understand the event and its applicability to the desired data base. The new PLG analysis includes events through December 31, 1981. The results of this study are summarized in Table 1 for the plants which were included in the Zion Probabilistic Safety Study initiating event data base. For reference purposes, Table 2 is a reproduction of the information contained in Table 3 of Reference 2, which compares the results of R. F. Scholl's work with the data used in the Zion study.

Our analysis and comments on the BNL use of the data developed by Scholl are included in item 2.8 of this response. We cannot determine the reasons for the differences between the revised PLG data and that reported in EPRI NP-801 because we do not have the basic event summaries used to develop the EPRI data. The PLG data base includes only those events which resulted in a loss of all offsite power to a unit. Events reported as partially degraded offsite power conditions (e.g., failure of one of two redundant offsite lines) are not included. However, events which did not result in a total loss of offsite power at the plant switchyard, but which caused failure of all sources available for automatic power transfer circuits and required operation of the diesel generators are included in the data base. (Manual switching operations to reenergize in-plant loads from available lines are included in the analysis of offsite power recovery actions, and these partial power failures are applicable to the initiating event data base.) Events attributed to single transformer failures and in-plant automatic transfer failures are not included in the data, because these events are analyzed within the plant power system model. The PLG data has been developed on a plant site basis, rather than a reactor unit basis, and notes are provided to describe the unit operating conditions when the power failure occurred.

A two-stage Bayesian update was performed using the data from Table 1. As suggested by the BNL review comments, the Zion data was excluded from the plant population for the first step of the analysis. The plant population data was applied using the total number of offsite power failure events for each site and the total number of site years listed in Table 1 (i.e., not accounting for the effects of unit availability). The generic data was updated using the Zion site specific evidence of no failures in 9 site years. The resulting distribution provides the specialized <u>calendar year</u> frequency of loss of offsite power events at the Zion site, <u>regardless</u> of unit operating conditions. This distribution was then multiplied by the average Zion unit availability (0.71) to obtain the frequency of losses of offsite power to a Zion unit <u>during power operation</u>. The parameters of this updated and scaled distribution are

5th Percentile:	1.04×10^{-2}	failure/unit operating year
Median:	3.63×10^{-2}	failure/unit operating year
95th Percentile:	1.27×10^{-1}	failure/unit operating year
Mean:	4.85×10^{-2}	failure/unit operating year

The mean frequency for the loss of offsite power at a generic plant site from the given population, excluding the Zion evidence, was determined to be 0.194 events per site calendar year. The updated mean frequency for the loss of offsite power at the Zion site is 0.068 events per site calendar year.

There are several reasons for the differences between this distribution and the point estimates cited by the BNL reviewers. One obvious source is the significant difference between the PLG data for losses of offsite power and the data developed by Scholl. The revised data base is also somewhat different from the original data used in the Zion Study which was obtained from EPRI NP-801. Therefore, our updated frequency for power failures per site calendar year is different from the value calculated by BNL using the EPRI data and correctly accounting for the effects of multiple reactor sites. Finally, since this data is applied to a specific Zion Probabilistic Safety Study transient event tree which models a unit trip from power operation, the site data has been scaled to account for the average Zion unit availability to develop the frequency of offsite power failures per reactor operating year.

2.3 OFFSITE POWER RECOVERY TIME DATA

The BNL reviewers' comments regarding the time for offsite power recovery are also based on the data collected by R. F. Scholl (Reference 1). It is noted that for 39 instances of total loss of offsite power for which the time to partial restoration of power was given, Scholl's data provides the following information.

- Frequency of failure to recover power within 30 minutes: 0.41.
- Frequency of failure to recover power within 60 minutes: 0.26.

The revised PLG summary of loss of offsite power events identifies a total of 58 events at all U.S. plant sites that have resulted in total power failure. As noted in the preceding section, these events do not include failures of single transformers or failures of in-plant power transfer circuits, since these failures were analyzed within the Zion plant electric power system model and no recovery was applied to power failure scenarios resulting from them. Power restoration time information was available for 42 of the 58 offsite power failure events (which includes 22 of the 34 Zion data base events summarized in Table 1). The offsite power recovery time data from these events is summarized below:

	Frequency of Failure to Recover Power	
	Within 30 Minutes	Within 60 Minutes
Overall (42 events)	.50	.38
Revised Zion data base (22 events)	.50	.41

This data seems to strongly reinforce the BNL reviewers' observation that the Zion recovery time analysis is very optimistic compared with the generic data for offsite power restoration. However, there is also a significant amount of evidence available from the generic data to indicate that the Zion power recovery time should be much lower than the plant population. Most of the sites experiencing extended power outages are located in areas subject to regional grid instability problems (e.g., St. Lucie and Turkey Point), unique supply line routing (e.g., Pilgrim), or have experienced localized outages directly attributable to some characteristic of the site (e.g., Millstone and Palisades). Furthermore, the Zion recovery model has been developed in response to a specific set of plant conditions which could have an important effect on the relative priorities assigned to offsite power restoration and, if they were to occur, could significantly reduce the recovery times reported for some sites (e.g., Indian Point).

The Zion Probabilistic Safety Study power recovery analysis is applied to plant conditions in which at least two of the three diesel generators for the affected unit are failed and no core cooling equipment is available (e.g., the turbine drive auxiliary feedwater pump has also failed). This situation is much different from all the events summarized in the generic data base, because in virtually all cases, onsite power was available from one or more diesel generators, and there was no immediate concern for loss of core cooling or loss of reactor coolant inventory. Of the 42 events for which recovery time data is available, 10 involved partial failures of onsite power sources or onsite switching failures which left one or more buses deenergized. Offsite power was restored to at least one bus within 30 minutes in all but one of these cases. The implications of this comparison are perhaps somewhat misleading, since none of these events involved significant offsite grid disturbances or widespread storm damage. However, in some cases involving regional power failures, there are indications in the reports that partial offsite power service was available and could have been connected to the plant if the onsite equipment had malfunctioned. The reasons for not reconnecting offsite power more quickly in these cases are not given in the reports.

As was noted in our discussion of the offsite power failure frequency data, the development and application of generic data must be consistent with its use and interpretation in the plant models. Since there has never been an instance of offsite power failure, severe degradation of onsite power supplies, and loss of core cooling capability, none of the generic data is directly applicable to the power recovery scenarios for the Zion Probabilistic Safety Study. The generic plant operating experience provided an important information input to the development of the Zion power recovery model. However, it was also recognized that many of the generic plant sites have local transmission grid and switchyard configurations which are much different from those at Zion and which could significantly affect the time required to restore power after a major disturbance.

The Zion site is somewhat unique among the population of plants because its switchyard is an important intertie point for the Commonwealth Edison transmission grid. A total of six transmission circuits are interconnected through a ring bus, and the transmission line rights of way diverge geographically soon after the lines leave the switchyard. Two of the lines connect to a neighboring utility in Wisconsin. It is also important to note that the Zion switchyard is directly adjacent to the plant buildings and that the switchyard equipment is normally operated by on-shift plant personnel.

The Zion offsite power recovery time distribution has two primary components, both of which are based on an analysis of the Zion plant site and evaluation of Commonwealth Edison transmission system experience. As with many scenarios modeled in the study, it was felt that this type of analysis would provide a more realistic estimate of expected performance than could be obtained from the evaluation of rather sketchy generic data from sites which are significantly different from Zion, under conditions very different from those being modeled in the study. The operator response time model accounts for normal shift manning, conflicting concerns about the failed diesel generators and the restoration of normal power, standard plant practice for switchyard operations, and expected operator performance under conditions not covered by specific procedures but involving operations familiar to plant personnel. It is conservative to the extent that it models the diesel generator trouble investigation and switchyard response actions as purely sequential events for a single operator when, in fact, it is possible that operators could respond to each location independently.

The offsite line restoration distribution presented in Secion 1.3.2.2 (page 1.3-15) is based on the evaluation of approximately 15 years of transmission line forced outage data for the entire Commonwealth Edison system (Reference 9). The line recovery distribution used in the study was developed from a model for the Zion site which treated the six transmission circuits as being equivalent to three pairs of totally coupled lines because of their routing and termination after leaving the site. A slight coupling was included between the two pairs of lines connecting to Commonwealth Edison facilities. The lines connecting to Wisconsin were treated as being essentially independent electrically from the other four circuits because of the intertie load shedding facilities which quickly disconnect these circuits from the Commonwealth Edison grid if instabilities develop in either of the utilities' transmission networks. Although the area is prone to severe thunderstorms, freezing rain, and heavy snowfalls, there is no evidence from the Commonwealth Edison records to indicate a significant frequency of physical damage to all of these circuits. (The 10^{-4} frequency for failure to recover power within 8 hours accounts for the fact that there have been no extended multiple line or transmission facility outages in the Commonwealth Edison system in more than 1,100 forced outage events due to all causes. Even though a series of severe tornadoes disabled five redundant transmission lines supplying the Dresden site on November 19, 1965, offsite power was restored to the site from at least one line within 4 hours after the initial power failure.)

We believe that the line recovery distribution used in the study provides a realistic model of the expected time to restore power from at least one of the six circuits at Zion and that the application of simply derived generic data is not relevant to the scenarios being modeled.

2.4 OFFSITE POWER RECOVERY HISTOGRAM

We have rechecked the calculations for the offsite power recovery histogram and agree with the BNL reviewers that there is a numerical error in the assignment of the discretized interval porbabilities. This error apparently arose from a combination of interpolation errors and truncation of a portion of the long "tail" of the line recovery distribution. The histogram combination calculations have been corrected, and the results agree with the BNL calculations for the 60-minute time interval. However, we disagree with the BNL results for the 30-minute interval. The corrected histogram for offsite power recovery yields the following values for the failure to recover power:

- Frequency of failure to recover power within 30 minutes: 0.292.
- Frequency of failure to recover power within 60 minutes: 0.046.

The BNL results are summarized in Section 2.1, page 11, of their comments. They have calculated the frequency of failure to recover power within 30 minutes to be 0.365 and the frequency of failure to recover power within 60 minutes to be 0.046. We suspect an interpolation error in their calculation for the 30-minute interval, although we are not sure of the source of this error. Our results were obtained through a convolution of the power recovery histograms shown in the Zion Probabilistic Safety Study, Section 1.3.2.2, page 1.3-15. For computation purposes, the probability values were assigned to the midpoints of the respective time intervals for these histograms. Linear interpolation was used between adjacent points on the resulting cumulative probability distribution to obtain the probability values associated with intermediate recovery times.

The calculations on the following two pages illustrate the effects that our histogram corrections have on the mean values of the distributions for P_{F30} and P_{F60}, which are described in the Zion Probabilistic Safety Study, Section 1.3.2.3 (pages 1.3-17 and 18). The value for P_{F30} increases from 6.04×10^{-4} to 6.34×10^{-4}, or an increase of approximately 5%. The change in P_{F60} is from 7.49×10^{-5} to 1.04×10^{-4}, or an increase of a factor of approximately 1.4. This increase is somewhat smaller than the difference between the single 60-minute point values of 0.046 and 0.030, because of the redistribution of the probabilities over the entire recovery time histogram. This redistribution affects the time integral of power failure and recovery as illustrated in the detailed calculations.

We sincerely appreciate the BNL comments pointing out this error. Resolution of the differences in the numerical values for the 30-minute recovery would not have a significant impact on the event sequence of interest (Sequence 44 of Event Tree 11b, as shown in the Zion

Probabilistic Safety Study, Figure 1.3.4.11-2) since the conditional frequency for this sequence is affected most strongly by the cumulative value for recovery through 60 minutes.

RECALCULATION OF P_{F30}

(Refer to Zion Probabilistic Safety Study Pages 1.3-17 and 1.3-18)

t (minutes)	$p_{\overline{89}}(t)(hr^{-1})$	$P_{os}(t+30)$	$X(t,30)(hr^{-1})$
0^+	5.25×10^{-4}	.708	3.72×10^{-4}
30	5.75×10^{-4}	.954	5.49×10^{-4}
60	6.25×10^{-4}	.985	6.16×10^{-4}
90	6.75×10^{-4}	.992	6.70×10^{-4}
120	7.25×10^{-4}	.994	7.21×10^{-4}
150	7.75×10^{-4}	.995	7.71×10^{-4}
180	8.25×10^{-4}	.998	8.23×10^{-4}
210	8.75×10^{-4}	.9989	8.74×10^{-4}
240	9.25×10^{-4}	.9994	9.24×10^{-4}
270	9.75×10^{-4}	.9995	9.75×10^{-4}
300	1.00×10^{-3}	.9995	1.00×10^{-3}
330	1.00×10^{-3}	.9995	1.00×10^{-3}
360	1.00×10^{-3}	.9999	1.00×10^{-3}

Using mean values: $P_{\overline{89}}(6\text{ hours}) = 7.08 \times 10^{-3}$

$P_{\overline{89}}(0) = 1.83 \times 10^{-3}$

$P_{os}(30) = .708$

$$\int_{0^+}^{6} X(t,30)dt = 5.15 \times 10^{-3}$$

$$P_{F30} = 6.34 \times 10^{-4}$$

RECALCULATION OF P_{F60}

(Refer to Zion Probabilistic Safety Study page 1.3-18)

t (minutes)	$\overline{p_{89}}(t)(hr^{-1})$	$P_{os}(t+60)$	$X(t,60)(hr^{-1})$
0^+	5.25×10^{-4}	.954	5.01×10^{-4}
30	5.75×10^{-4}	.985	5.66×10^{-4}
60	6.25×10^{-4}	.992	6.20×10^{-4}
90	6.75×10^{-4}	.994	6.71×10^{-4}
120	7.25×10^{-4}	.995	7.21×10^{-4}
150	7.75×10^{-4}	.998	7.73×10^{-4}
180	8.25×10^{-4}	.9989	8.24×10^{-4}
210	8.75×10^{-4}	.9994	8.74×10^{-4}
240	9.25×10^{-4}	.9995	9.25×10^{-4}
270	9.75×10^{-4}	.9995	9.75×10^{-4}
300	1.00×10^{-3}	.9995	1.00×10^{-3}
330	1.00×10^{-3}	.9999	1.00×10^{-3}
360	1.00×10^{-3}	.9999	1.00×10^{-3}

Using mean values: $\overline{P_{89}}(6 \text{ hours}) = 7.08 \times 10^{-3}$

$\overline{P_{89}}(0) = 1.83 \times 10^{-3}$

$P_{os}(60) = .954$

$$\int_{0^+}^{6} X(t,60)dt = 5.23 \times 10^{-3}$$

$P_{F60} = 1.04 \times 10^{-4}$

2.5 EFFECT OF CHANGES ON EVENT TREE 11b, SEQUENCE 44

As a result of the BNL comments, we have recalculated the frequency for the loss of offsite power initiating event using the revised PLG generic data base discussed in item 2.2. We believe that this data represents the type of power failure event being modeled better than the data originally obtained from EPRI NP-801. As suggested by BNL, the offsite power failure data has been developed on a plant site basis rather than a reactor unit basis. The average Zion unit availability has been included in the data updating process to develop a distribution for the frequency of loss of offsite power during single Zion unit power operation. The Zion site data has not been included in the generic plant population. The mean frequency for loss of offsite power to an operating reactor from this analysis is

$\phi_{LOP} = 4.85 \times 10^{-2}$ event/reactor year

We do not agree with the BNL comments regarding the use of generic data for the time for offsite power recovery. We believe that the characteristics of the Zion site and the Commonwealth Edison experience are sufficient to justify a departure from the use of purely generic data for this evaluation. Furthermore, there is some indication from the generic data that the times for partial power restoration could be somewhat shorter at a number of the plant sites under conditions similar to that being modeled for this event sequence. Therefore, we have retained the models for operator response and offsite line recovery described in the Zion Probabilistic Safety Study, Section 1.3.2.2.

We have corrected the numerical errors in the power recovery histogram as described in item 2.4. The revised mean conditional frequency for the failure of power at buses 148 and 149 for longer than 60 minutes is:

$$P_{F60} = 1.04 \times 10^{-4}$$

We have retained the conditional frequency for the failure of the turbine-driven auxiliary feedwater pump:

$$L\text{-}2 = 4.9 \times 10^{-2}$$

Using these mean values, we obtain the revised unconditional mean frequency for Sequence 44 of Event Tree 11b.

$$\phi_{LOP} \times P_{F60} \times (L\text{-}2) = 2.5 \times 10^{-7}$$

The unconditional frequency for this sequence from Table 8.4-7 (page 8.4-16) of the Zion Probabilistic Safety Study is 2.0×10^{-7}. Because the failure of electric power causes failure of both containment sprays and fan coolers, this core melt scenario leads to the important core damage state TE. While it is an important contribution to risk due to internal events, it has no effect on the plant risk curves when all events (internal and external) are considered.

2.6 TREATMENT OF DIESEL GENERATOR RECOVERY

The Zion Probabilistic Safety Study electric power system models did not include any credit for the restoration of power due to the recovery of failed diesel generators or switching of the output from swing diesel generator "0." The BNL reviewers note this in their comments Section 2.2.1, page 12. However, they also note that the diesel generator recovery time model presented in Zion Probabilistic Safety Study Section 1.3.2.4 (pages 1.3-20 and 1.3-21) is optimistic compared with the data in Table 10 of NUREG/CR-1362 (Reference 10). On page 35 of NUREG/CR-1362, the authors provide the following information about their classification of diesel generator repair times.

> "Again, we note that this classification is mostly determined subjectively and is based on our experience with diesel generators. However, there were LERs that stated exactly how

long it took to repair the diesel generator. But the number of these LERs is small, approximately 19% of the 425 one-liners, compared to the number of LERs that did not state how long the repair took."

The Zion diesel generator recovery time distribution was developed from a combination of our experience with diesel generator repairs, an analysis of the Zion plant staffing, and a summary of the site specific diesel generator failure and repair time data from Zion. Two important points should be noted when comparing this distribution with the information from NUREG/CR-1362. The Zion recovery distribution models the time to repair a failed diesel generator following initial personnel response to the diesel generator room, and it assumes that the recovery efforts will be essentially continuous until the diesel generator is returned to service. (This would certainly be the case for any scenarios contributing to core melt.) Therefore, long repair times which might be observed in the generic data due to failures occurring on maintenance off-shift hours without immediate personnel call-out are not directly applicable to this model. (They would be applicable to routine failures under nonemergency circumstances.) If a diesel generator failure simply places the plant in a limiting condition for operation under the technical specifications, no special effort is made to mobilize repair personnel unless continued plant operation is threatened. Therefore, one would expect the generic plant population data for recovery times to be somewhat longer than those observed in emergency situations.

It should also be noted that Zion is somewhat unique among the plant population in that a large engineering staff is assigned to the plant site. One of the groups in this plant technical staff is assigned specific responsibility for the diesel generators, and they provide an important source of onsite engineering expertise. Zion plant experience has shown the response times of these engineers to be equal to or less than that of maintenance personnel during situations requiring rapid repairs. This factor has been included in the diesel generator recovery time distribution as a slight downward shift in the percentage of intermediate time repairs (4 to 8 hours) and an increase in the percentage of repairs completed in less than 4 hours, because of the troubleshooting expertise of the technical staff personnel.

Finally, it should be noted that the Zion diesel generator recovery distribution is only slightly different from the information in NUREG/CR-1362, if one excludes the category of "unknown/not applicable" data from Table 10 of the NUREG. The following table summarizes this comparison.

Recovery Time Period	NUREG/CR-1362, Table 10			Zion Distribution (page 1.3-20)
	Recovery From Failure to Start	Recovery From Failure to Continue to Run	Average Recovery From All Failures	
0-1 Hour	.31	.16	.25	.30
1-4 Hours	.27	.28	.28	.40
4-8 Hours	.25	.25	.24	.15
8-24 Hours	.09	.14	.11	.05
> 24 Hours	.08	.17	.12	.10

We believe that the slight differences between the Zion Probabilistic Safety Study distribution and the NUREG/CR-1362 information are totally justified by the nature of the NUREG data, the availability of Zion engineering personnel, the evidence from the site specific diesel generator data, and the scenarios for which this recovery model was developed.

2.7 TREATMENT OF DIESEL GENERATOR COMMON CAUSE FAILURES

2.7.1 GENERAL COMMENTS AND REVIEW OF THE MILLSTONE DEGRADED VOLTAGE EVENT OF July 5, 1976

In Section 2.2.2 of their comments, the BNL reviewers note that the development of the conditional frequency for the reactor coolant pump seal failure event (LS) assumes that the Zion Unit 1 and Unit 2 diesel generators fail independently. They are correct in their interpretation of the treatment of this event. Common cause failures were evaluated for each unit combination of diesel generators (i.e., using the terminology of Zion Probabilistic Safety Study, Section 1.3.3.8, for P_A, P_B, P_C, P_D, and P_E). However, when the diesel generator failure distributions were combined, they were combined as if the Unit 1 and Unit 2 distributions were independent. The dominant contributing scenario was determined to be one in which diesel generator "0" energized bus 247, diesel generators 1A and 1B failed (P_C, including common cause failures), and diesel generators 2A and 2B failed (P_E, including common cause failures). As noted by BNL, the distributions for P_C and P_E were combined as if they were independent, implicitly assuming that any common cause failures affecting diesel generators 1A and 1B would not simultaneously affect diesel generators 2A and 2B. The way in which the distributions were combined also implies no treatment of common cause failures simultaneously affecting all five diesel generators, regardless of the status of diesel generator "0."

We appreciate the BNL concerns about these possible inter-unit common cause effects and agree that they should have received an explicit treatment in the original analysis for event LS. However, we disagree with the implications of the example used to illustrate the potential for this type of failure. The Millstone event of July 5, 1976, involved a unit trip which caused the surrounding grid voltage to droop from 352 kV to 333 kV. (The entire state of Connecticut was being supplied by the two Millstone units and one small fossil plant when the trip occurred.) The in-plant 4 kV buses had been manually transferred to the reserve offsite power supply in preparation for a normal unit shutdown. This is an important departure from a unit trip from power operation which requires the bus undervoltage relays to automatically initiate the source transfer. When the trip occurred, all 4 kV bus voltages remained above the undervoltage relay setpoints of 2,912V. Therefore, no signal was developed to shed the bus loads and start the plant diesel generators. However, the compound voltage drops were large enough that several 480V AC motor control center contactors had insufficient voltage to operate and the equipment failed to start. Because the control power for these contactors was derived from 480/120V AC stepdown transformers at the contactors, the increased control power demand also caused the control power fuses to blow.

The BNL comments state that the diesel generators were not capable of automatic start. It is true that they did not and would not automatically start under the given 4 kV bus voltage conditions. However, they would have started normally if bus voltage had decreased below the undervoltage setpoints, as would have been the case during a total loss of offsite power event. The undervoltage relay setpoints would have to be set at effectively zero bus voltage in order to prevent the diesel generators from sensing a low voltage condition during a true loss of power event. Although relay miscalibration is often cited as a contributing cause for this event, it is important to note that the relays were set at the minimum voltage specified in the plant technical specifications. The setpoints should have been higher, but the relay technicians had performed their calibrations correctly. (A test circuit design error had also caused approximately 5% voltage differential between the desired and actual setpoints, but this error is negligible when considering a full loss of bus voltage.) Furthermore, there is no mention of degraded performance of operating 4 kV or 480V AC equipment, indicating that the low voltage conditions were not sufficient to adversely affect the normal bus loads. We are not familiar with the source of control power for 4 kV circuit breakers at Millstone, but this control power is DC at Zion and would not be significantly affected by degraded AC bus voltage. We believe that it is reasonable to assume that if a similar situation were to exist at Zion and plant conditions required the automatic actuation of safeguards equipment, the current inrush from the 4 kV loads automatically closing onto their buses would cause sufficient additional losses to drop bus voltage below the undervoltage relay setpoints and would automatically start the diesel generators. If the load breakers closed and bus voltage remained above the undervoltage setpoints, then it is likely that the control room operators would quickly become aware of the condition and manually start and load the diesel generators.

This is a somewhat lengthy discussion about the Millstone event, but it should be evident that this specific condition would not have resulted in the type of diesel generator failure event envisioned by the BNL reviewers. It is true that a number of 480V loads failed, and the event certainly represents an unanticipated common cause failure contribution for this equipment. However, simple extrapolation of this event to a potential common cause failure scenario for all five Zion diesel generators following a loss of offsite power is inappropriate.

We also disagree strongly with the apparent BNL rationale for estimating the conditional frequency of this common cause diesel generator failure as 10^{-4} per loss of offsite power event. This type of reasoning is equivalent to inferring guilt by association, with no evidence to support their claim. One of the principal reasons for performing comprehensive analyses like the Zion Probabilistic Safety Study is to provide a logical framework for the identification and assessment of specific contributors to risk. We welcome comments identifying scenarios we may not have included, and we encourage the reviewers to challenge our assessment of the scenarios we did include. This type of interactive evaluation will certainly result in a more meaningful product. However, we feel that responses based only on vague assumptions are simply counterproductive.

2.7.2 ESTIMATE OF THE COMMON CAUSE CONTRIBUTION TO THE UNAVAILABILITY OF ALL FIVE ZION DIESEL GENERATORS

2.7.2.1 Undetected Mechanical Maintenance Errors

Having made these points, we do agree that there could be scenarios which couple the Unit 1 and Unit 2 diesel generators, and these scenarios should be evaluated in our analysis. As noted by the BNL reviewers, the type of common cause failure of concern is one which is not revealed by the normal plant testing program, but surfaces only during an actual demand on the diesel generators from a loss of offsite power. One possible source of these failures could be an undetected diesel generator maintenance error as described in Zion Probabilistic Safety Study, Section 1.5.2.2.1.4.6.2 (page 1.5-207). Extending the evaluation of these errors to four diesel generators (the minimum number of failures necessary to achieve event LS) provides the following results.

Unavailability of four diesel generators due to undetected maintenance errors:

Mean: $Q_{ME4} = 2.39 \times 10^{-7}$

We have used a 95% effectiveness for the detection of general maintenance errors during the normal diesel generator testing program. Errors of the general type considered in this analysis (e.g., rags in the oil system) have occurred on single diesel generators at Zion, and these errors were detected during either the first or second routine test after maintenance. The coupling factor used to determine the dependence for the third and fourth errors is the same value (0.081) used for the second error. This treatment provides results which are slightly conservative compared with those which would be obtained through a cascading application of the low dependence expression recommended in the NRC Human Reliability Handbook NUREG/CR-1278 (Reference 11).

2.7.2.2 Testing and Maintenance Performed During Refueling Outages

The only common maintenance performed on the Zion diesel generators is the routine diesel engine overhaul done just before or during a unit refueling outage. Diesel generator relay testing and calibration of the 4 kV bus relays is also done during each unit refueling outage. The automatic start of each diesel generator from an undervoltage condition at its respective 4 kV bus is functionally tested during each quarterly safeguards actuation test on each unit (PT-10). The undervoltage starting circuits and the loss of power and safeguards actuation load sequencers are functionally tested for each diesel generator during the integrated safeguards actuation test performed during each unit refueling outage (TSS-15.6.35). Since the refueling outages are scheduled approximately 6 months apart for each of the Zion units, there are no common maintenance procedures or tests which affect all five diesel generators within a short period of time. We are thus faced with a series of operations which affect all the diesel generators on one unit, followed approximately 6 months later by a similar series of operations which affect all the diesel generators on the other unit. (Swing diesel generator "0" is normally overhauled with the Unit 1 diesel generators, and its bus undervoltage relays and sequencers are tested for each of its load buses during the respective unit outage.)

We believe that the contribution from undetected maintenance errors described above provides a conservative quantification of the effects from errors during the diesel engine mechanical overhaul activities. This is especially true when one considers the long times between successive refueling outages and the numerous diesel generator tests performed between these outages.

2.7.2.2.1 4 kV Bus Relay Testing and Calibration

It is extremely difficult to identify specific common cause failure scenarios resulting from undetected human errors affecting the 4 kV bus relays or the diesel generator load sequencers. However, it is possible to provide a rough conservative estimate of the magnitude of these contributions based on evaluation of the equipment, personnel actions, and testing procedures. The 4 kV bus undervoltage and fault protection relays are tested each refueling outage to verify that their setpoints are within the specified tolerances. All the relays for each 4 kV bus are tested by the same personnel on the same day. Each of the three buses for a unit is tested on a different day during the outage, as plant conditions allow. The same test personnel normally perform all the relay checks during a given outage, although different personnel may perform the tests during successive outages. All relay testing is performed according to approved written procedures. The error of interest in this analysis is an adjustment made to a set of relays common to all 4 kV buses that would prevent the diesel generators from automatically reenergizing the buses following a loss of offsite power. As noted in our discussion of the Millstone event, this scenario requires a gross miscalibration error or effective blocking or bypassing of the bus undervoltage sensing circuits or diesel generator output breaker closing circuits. It seems most likely that this type of error would result from a procedural omission or common personnel misinterpretation. It should be noted that the personnel performing these tests are not assigned to the Zion plant staff, but are members of the Commonwealth Edison Operational Analysis Department, who are specially trained to perform these tests and perform them on a regular basis throughout all the company's transmission and generation facilities (i.e., the test personnel are very familiar with these operations and are not simply assigned this task once each unit refueling outage). Considering the personnel and the nature of the testing, a frequency of 5×10^{-3} error per bus test is a very conservative estimate for general errors which could result in the bus relay setpoints being left outside their tolerance band. For simplicity, we will assume complete dependence among the errors at all three buses for a given unit. The assumption of a lower coupling factor could reduce the frequency of errors affecting more than one bus by a factor of 5 to 10, but we will retain the complete dependence assumption to account for possible common procedural errors and the fact that the same personnel perform all adjustments on a given unit within a relatively short period of time.

Before a unit is returned to criticality after a refueling outage, a comprehensive safeguards actuation test is performed which includes a functional test of the diesel generator automatic start and output breaker closing operations (PT-10). During this test, normal power is interrupted to each 4 kV bus, and the associated diesel generator is verified to automatically start from the bus undervoltage condition and reenergize the bus. The test is performed by licensed Zion plant operators. The results of this test must be reviewed and approved before the unit may be returned to operation. Since this test functionally verifies the operation of the same circuits required to sense bus undervoltage and start the diesel generators during a loss of offsite power event, it is very likely that the test would detect any personnel errors which were serious enough to prevent these automatic operations. However, we assign a conservative frequency of 2×10^{-3} error per test to account for the omission of the procedural step which checks this function. As with the relay test personnel errors, complete dependence is assumed for these errors for all three unit buses, even though the test procedure has separate sections for each bus and normally requires two or more shifts to complete with at least two different groups of personnel. Using these very conservative rough estimates for human errors, we estimate that an upper bound for the frequency of a unit being returned to power operation with all three of its buses in a condition which prevents automatic diesel generator loading on loss of offsite power is on the order of 1×10^{-5} per refueling outage. It is emphasized that this is a very simplistic point estimate calculation to provide a point of comparison for this response only. We are confident that a more sophisticated analysis of the scenario, including a realistic assessment of the uncertainties involved, would result in a mean frequency significantly lower than this estimate.

A safeguards actuation test identical to that performed prior to returning to power after a refueling outage is also performed once each calendar quarter during unit operation. Because of the staggered nature of the unit refueling outages, it is virtually certain that at least one of these tests will be performed during the intervening period from one unit startup to the next unit's refueling outage. If the first test failed to detect the relay problems, then it is certainly possible that the second test will not be a totally independent check of the condition. However, because of the intervening period and the likelihood that a completely different set of personnel will be performing the test, we also believe that the assignment of complete dependence between these successive tests is unwarranted. We conservatively assign a conditional error frequency of 0.1 error per test for the failure of the second test to disclose the bus relay deficiencies, given the fact that they were not discovered by the refueling outage test. We again conservatively assume complete dependence among all three unit buses for this test. We now have a point estimate on the order of 10^{-6} for the frequency of entering a second unit refueling outage with all the relays from one unit in a condition which would prevent the diesel generators from automatically starting. Even if we were to assume complete dependence for the relay test personnel errors for the second unit (recognizing that they may be different personnel but will probably be using the same test procedures) and were to assume complete dependence for the failure of the functional safeguards actuation test to detect these errors, we are faced with an extremely conservative point estimate for the common relay failure condition which is approximately equal to the mean value of the distribution obtained from our detailed analysis without these common cause failures.

Again, it must be emphasized that the preceding analysis is a very simplistic treatment of this scenario. We believe that a more detailed analysis, including better estimates for the human error frequencies, better models for the interbus and intertest error dependencies, and a rigorous quantification of our uncertainties would result in a mean unavailability of all power due to this scenario which was no greater than this point estimate, and it would very likely be lower. It should also be recognized that this scenario, unlike the coincident failure of four or five diesel generators, presents a very strong potential for operator recovery actions. Following the loss of offsite power and failure of all five diesel generators to start automatically, the operator must simply start and load one of the diesel generators onto one of the buses supplying a component cooling pump within approximately 30 minutes following the initial power failure. All required operations can be performed from the control room, and the operators would have few distractions to compete with their efforts to restore power under these conditions.

2.7.2.2.2 Diesel Generator Load Sequencer Testing

Scenarios involving the diesel generator loss of power load sequencers can be addressed in the same manner as the 4 kV bus relays. If these sequencers were to completely fail to operate or if they were misadjusted to close several load breakers at the same time, the associated diesel generators would either start automatically and run unloaded or could fail due to excessive loading transients. Therefore, although the sequencers do not affect the diesel generator automatic starting circuits or the diesel generator output breaker closing circuits, certain sequencer failures have the same effects as would failures of the diesel generators themselves. Each diesel generator has a load sequencer for loss of power conditions and a sequencer for safeguards actuation conditions, the difference between the two being the source of the actuation signal and the order in which equipment is reenergized. Since the loading on each bus is slightly different, each sequencer has its own time settings to provide a combination of the desired plant mechanical equipment operation and a relatively uniform loading transient for each diesel generator.

Each of these sequencers is functionally tested each unit refueling outage to verify its overall operation and the closing time of each load breaker. The diesel generators are started, the sequencers are allowed to reenergize the bus loads, and the sequencers are adjusted only if they fail to operate within the specified times. If a sequencer fails the initial test, it must be retested after adjustment to verify that all loads cycle onto the bus as required. The test is normally performed at the beginning of a unit refueling outage and takes between 1 and 2 days to complete. It is performed under the direction of plant technical staff engineering personnel with the assistance of licensed plant operators. The engineers are responsible for verifying the operation and timing of the load sequencers. Since the sequencers are adjusted only if they do not meet the operating success criteria, we believe that the most likely sources of errors during this testing are misinterpretation of the success criteria or a procedural deficiency which would lead to erroneous results. As with the 4 kV bus relays, these errors would have to be severe enough to functionally disable the diesel generators, but they could not be so severe as to be obvious to the personnel interpreting the test results. Rather wide variations in the precise timing and sequence of the bus reloading operation will not significantly impair diesel generator performance. We also believe that extreme sequencer settings which either failed to reenergize all the equipment or which simultaneously reclosed all the load breakers would be detected by the test engineers, because these are precisely the conditions which the sequencers are supposed to prevent. We cannot evaluate how severely misadjusted the sequencers must be to cause diesel generator failure, nor can we precisely evaluate the likelihood that the test engineers will recognize these settings as being "unusual." However, the existing sequencer settings have been verified to provide acceptable diesel generator loading response during several tests on each bus, and we were informed by plant personnel that the settings have not been altered since unit preoperational testing. Therefore, any procedural change requiring an adjustment of the sequencers would be considered an unusual event.

The first performance of the refueling test normally provides an integrated test of the sequencers and the diesel generators by actually starting all the bus loads. However, we were told by plant personnel that subsequent tests which might be required to reverify the timing circuits or check specific adjustments could be performed with the bus load breakers racked to the test position (i.e., the test would verify the timing sequence and the operation of the circuit breakers, but the diesel generators would not be loaded).

The nature of this test has led us to postulate two general scenarios which we believe to provide the most significant potential for common cause errors: (1) the sequencer timing setpoints could be specified incorrectly in the test procedure, and the test personnel could reset the sequencers to these erroneous values; and (2) the sequencer setpoints could be specified correctly, but the test personnel could misinterpret the procedure and inadvertently reset the sequencers to incorrect values.

Changes to the setpoints listed in the procedure would require an active decision by a member of the plant engineering staff and would have to be reviewed and approved by at least three members of station management. Typical frequencies for routine errors of omission in the preparation of procedural revisions are on the order of 5×10^{-3} error per revision. However, in this case the revision must specify precisely the wrong settings to cause failure of all three sequencers. We believe that the frequency of these errors of commission is at least one to two orders of magnitude lower than the typical omission errors cited in the generic literature. Therefore, we estimate a frequency of 5×10^{-4} for errors which specify the incorrect sequencer timing setpoints. This value represents an estimate of the frequency of errors per procedure revision. Although there is normally no need to change these timing specifications, we conservatively use this value to represent the probability that the times are incorrectly specified in any given revision of the procedure to account for the relatively high frequency of minor changes normally recorded for test procedures.

The sequencers would be adjusted only after the initial (successful) performance of the test demonstrated a deviation between the actual loading times and those specified in the procedure. Changes made before the test would cause the diesel generators to fail and would be corrected. Since at least two engineers are normally responsible for the performance of this test, we believe that they would question the validity of a procedure calling for the adjustment of setpoints which are not normally changed following the completion of a successful diesel generator loading operation. An error frequency of 10^{-2} error per test seems reasonable for the failure of both test engineers and the test review personnel to question the changes and approve the sequencer readjustments. Since the diesel generators are not necessarily loaded during a retest after sequencer adjustment, it is possible for the incorrect settings to exist for approximately 1 year until the following refueling outage test for the affected unit.

It is difficult to assess a coupling factor between the sequencer tests conducted during alternate unit outages. An assumption of complete dependence between the procedural errors is supported by the generally low frequency of procedural changes affecting the setpoints (i.e., if an error existed for one test, it is very likely that the same error would be in place 6 months later when the test was performed on the other unit). However, since it is likely that a different set of test engineers would be assigned to the second test, the use of a complete dependence factor for failure to recognize the erroneous settings is difficult to justify. For simplicity in this rough estimate calculation, we will assume complete dependence for these errors to account for the possibility that the same personnel will conduct the test and will "remember" how to readjust the sequencers. This results in an extremely conservative point estimate on the order of 5×10^{-6} for the complete unavailability of all five diesel generators due to successive sequencer misadjustments in response to procedural errors. As noted in the calculation for the 4 kV bus relays, we strongly believe that a more sophisticated analysis of the sources of the procedural errors, the procedure review and approval process, the response of the test personnel, and the dependencies between successive tests could reduce this estimate significantly. The second test errors would result in a maximum period of approximately 6 months during which all the diesel generator sequencers could be misadjusted. Failure of the diesel generators to automatically load during the next outage test would identify the errors, as would any actual loading demands during the intervening period from real or spurious power failure conditions.

Misinterpretation of the correct setpoints is unlikely because adjustments are made only after the initial diesel generator loading test. The test personnel would have to decide that something was wrong with the correct loading sequence, initiate a procedure revision, and readjust the sequencers to the incorrect settings. As noted before, inadvertent extreme adjustments that would either prevent the sequencers from functioning or would cause all loads to be energized simultaneously are considered very remote. The effects of this scenario are the same as the first. The assumption of complete dependence between the personnel performing the tests during successive unit refueling outages is extremely difficult to justify in this scenario. We estimate the overall likelihood of this scenario to be at least a factor of 10 below the situation involving a preexisting documented procedural error. For the test engineers to reject acceptable results and adjust the sequencers to precisely the wrong settings requires either deliberate or nearly irrational behavior, and we believe that the number of people normally involved with the performance and review of these tests is adequate to ensure a negligible contribution from these actions.

Depending on the nature of the adjustment error, recovery from sequencer failures following a loss of offsite power could be very simple. If the sequencers failed to reclose the circuit breakers, the diesel generators would remain running unloaded and tied to their respective buses, but no equipment would be automatically restarted. The control room operators would simply have to reclose the breakers from the control board switches to restart the equipment, since the manual start function is not inhibited. Because of the design conservatisms for the diesel generators, it is not certain that a diesel generator would fail even if all its loads were reenergized simultaneously. However, the protective circuits which would normally trip the diesel generator from an overload condition are bypassed during an emergency start. Therefore, if the most extreme sequencer settings could cause the diesel generators to fail from overload, the diesel engines or the generator windings could be damaged, and it is unlikely that these failures could be recovered quickly.

2.7.2.3 Loss of Diesel Engine Cooling

Two additional scenarios associated with the diesel engines have been suggested as potential common cause failure contributors. Many plants in warmer climates have experienced extensive problems with biofouling of raw water heat exchangers by fresh water clams, mussels, and sponges. Since the Zion diesel generators are cooled by the plant service water system, this failure mechanism would certainly be an important common cause candidate if these organisms were prevalent in Lake Michigan. However, the general climate and average temperature of Lake Michigan are not conducive to their growth, and there has been no evidence of significant biofouling at Zion or any of the other power plants operating in the upper Midwest. Furthermore, because the diesel generators are operated at load for at least an hour during each monthly test, it is likely that degraded cooling capacity would be discovered during these routine tests because of unacceptably high engine operating temperatures. Therefore, we have ruled out heat exchanger biofouling as a candidate for common cause failures of all the Zion diesel generators. Failures of the diesel generators resulting from loss of the plant service water supply have been treated explicitly in the loss of service water turbine trip event tree 11c and by accounting for the dependencies between the operating diesel generators and the available service water pumps in degraded electric power states in all other event trees.

2.7.2.4 Fuel Oil Contamination

It has also been suggested that all five diesel generators could fail during operation because of contaminated or improper fuel oil. Although the fuel is stored in separate tanks for each diesel generator, it is common practice to "top off" several of the tanks from a single truckload of fuel. However, the fraction of the fuel tank inventory replaced during these operations is normally small, in the range of 10% to 20% of the tank capacity, and it is rare to have all five tanks filled from the same fuel shipment. Each fuel shipment is sampled from the delivery truck and each fuel oil storage tank is sampled quarterly to verify the condition of the fuel oil. Furthermore, plant data collected for the Zion Probabilistic Safety Study indicates that each diesel generator is run approximately once every 2 weeks, for routine operability tests and for nonroutine tests required by the plant technical specifications in response to maintenance on redundant safeguards equipment. It is believed that the frequent diesel generator testing, the fuel oil sampling, the relatively small fuel volume percentage replaced during a single fuel delivery, and the generally low frequency of contaminated fuel shipments combine to make this scenario a negligible common cause failure contributor for all five diesel generators during a loss of offsite power event.

2.7.2.5 Seismic Events

Finally, we should mention an important common cause contributor to failure of offsite power and failure of all five diesel generators that has been explicitly quantified in the Zion Probabilistic Safety Study. The mean frequency of a seismic event of sufficient acceleration to cause failure of the plant switchyard and unrecoverable damage to all five diesel generators is estimated to be approximately 5.6×10^{-6} event per reactor year.

2.7.2.6 Summary

We again emphasize that the rough point estimate calculations performed in this section are meant only to identify the scenarios which we believe to be the most significant potential contributors to common cause failures of all five diesel generators. The primary purpose of this exercise was to provide a point of comparison with the BNL estimate and to emphasize our methodology for the identification and assessment of specific scenarios rather than assigning simplistic estimates without any supporting analyses.

If one simply adds the common cause failure contributions quantified in this section without the seismic contributor, one obtains a point estimate on the order of 7×10^{-6} for the unavailability of all five diesel generators from common cause failures. However, as noted in each calculation, these are extremely coarse estimates, and we believe that a more sophisticated analysis, including the quantification of our uncertainties, would result in a mean unavailability somewhat lower than this value. This is a point estimate for the conditional unavailability of all five diesel generators following a loss of offsite power. The mean frequency for the loss of offsite power to the Zion site is approximately 0.068 event per site calendar year, derived in item 2. Therefore, a rough estimate of the frequency for loss of all power to both Zion units from the loss of offsite power and common cause failures of all five diesel generators is approximately 4.8×10^{-7} event per site calendar year, excluding seismic events. This frequency is approximately 12 times less than that due to seismic events which fail the offsite power supply and fail all five diesel generators. Therefore, even if one were to use the conservative estimates for common cause failures derived in this section, these failures are strongly dominated by the seismic contribution to total AC power failure, which is explicitly quantified in the Zion Probabilistic Safety Study. (This summary analysis does not account for the availability of the Zion units, since the availability effects would apply equally to the loss of offsite power events and the seismic events.)

Of course, it must also be recognized that many of the failure scenarios involving the bus relays and load sequencers provide a very high likelihood of rapid diesel generator power recovery by simply operating circuit breakers from the control room panels. These recovery actions have not been assessed, but since at least 30 minutes is available for their performance, it is very likely that the net common cause contribution to event LS, including recovery, would be much less than the contribution evaluated in Zion Probabilistic Safety Study, Section 1.3.3.8, from the independent treatment of the Unit 1 and Unit 2 diesel generators. (Recovery from the failures evaluated in that section would follow the general diesel generator recovery time model discussed in item 2.6, which is much more pessimistic about the recovery of a failed diesel generator within 30 minutes.)

We cannot identify any justification for the BNL estimate of 10^{-4} for the unavailability of all five diesel generators due to common cause failures, excluding external events. We also believe that the distribution reported in Zion Probabilistic Safety Study, Section 1.3.3.8, for the power failure contribution to event LS provides a realistic assessment of this scenario.

8. Review of Scholl Offsite Power Data

8.1 General Comments on the Scholl Report

We obtained a copy of Raymond F. Scholl's report on offsite power failures (Reference 3) and have completed a detailed review of the report and of the BNL use of Scholl's data. Scholl's work is one of the most comprehensive summaries of events affecting offsite power supplies that we have reviewed. His data was obtained from LERs and from the responses to information requests sent to all licensed operating plants. (A total of nine units at six sites did not respond to his requests, and only LER data was used for those units.) His summaries differentiate between partial and total power failure events and identify the specific cause of each failure where information was available. Since his raw data is simply a copy of a computer listing of events indexed according to plant docket number, type of failure (total or partial), and event date, it is difficult to correlate his data base to actual event descriptions. However, we were able to identify nearly all the events in the total power failure category, and we appreciate the thorough documentation provided in the report. Too few data bases provide this level of traceability, and Mr. Scholl's work is an important contribution to the effort needed to develop a comprehensive catalog of these events.

Unfortunately, although Mr. Scholl has accomplished an important data collection and categorization task, we disagree strongly with his methods for data reduction, and we see little practical use for the numerical results presented in the report summary and in Appendix B. Perhaps one of the most glaring deficiencies in Mr. Scholl's analysis of event frequencies is his treatment of the plant "age," or the number of years to which his failure data is applied. According to his definition, the plant "age" is based on the date of its first reported loss of offsite power through June 3, 1980 (Reference 1, page 1). Therefore, since the first reported loss of power at Zion occurred on March 12, 1979 (this event is discussed below), the "age" of each Zion unit is assumed to be 1.23 years (Reference 1, Table 5). Plants such as Crystal River 3 and Duane Arnold reported no failures of offsite power and, therefore, their "ages" were set equal to zero (i.e., they were excluded from the data base). This is obviously an extremely biased treatment of the plant population success data, and the use of these "ages" to determine site specific and generic power failure frequencies is a gross misapplication of statistical analysis. It is not surprising that the study results obtain very pessimistic estimates of these failure frequencies, since the numerical methodology has precluded the correct accounting for many years of plant operations without power failures.

We have reviewed in detail only the basic event data for total losses of offsite power because these are the events of interest in the Zion data base and they are the events from which BNL extracted their data. However, we note that Mr. Scholl included one partial loss of offsite power at both Zion units from an event on March 12, 1979. This event occurred with Zion Unit 2 in cold shutdown for refueling and Zion Unit 1 at power operation. Diesel generator 1B was out of service for maintenance. The reserve feed breaker from the Unit 2 offsite power source to the Unit 1 essential 4 kV buses was inadvertently removed from service for relay testing. This action administratively violated the technical specifications requirement for continuous operability of two sources of offsite power to Unit 1 during diesel generator maintenance. The action had no effect on bus voltages and would not have affected the automatic supply of offsite power following a Unit 1 trip, because the breaker in question can only be closed manually. Since the event was strictly an administrative oversight reportable under the plant's licensing criteria, it is difficult to justify its inclusion as an actual loss of power when developing data to be applied in estimating the frequency of power failure events. As noted, we did not review the partial power failure data in any detail, except for this single event. Therefore, we have no way of determining how many other events in this category could be of a similar nature, and we would certainly not apply the data without careful review.

2.8.2 LOSS OF OFFSITE POWER EVENT FREQUENCY DATA

We attempted to verify each of the 109 total loss of offsite power events in Mr. Scholl's report. Unfortunately, 4 of the events were inadvertently truncated during reproduction of his computer listings for the report and were not included in our copy. Of the remaining 105 events, we were able to identify 94 by correlating the unit and event date with our own data base information (References 6, 7 and 8). In some cases, we also contacted the utility to obtain information about events for which we could find no written documentation. Of the 11 events which we could not trace, we suspect that at least 3 may be double accounting for events which occurred at the same unit within one or two days of the event in question, although this cannot be verified from the available information.

In general, there is an excellent correlation between Scholl's basic event data and the information used to develop the revised PLG data base discussed in item 2.2. Scholl's data includes at least 11 events which were due to auxiliary power transformer failure or failures of in-plant switching circuits. As noted in item 2.2, these types of events are analyzed as part of the Zion plant electric power system model and should be excluded from the loss of offsite power initiating event data base for the Zion Probabilistic Safety Study. This is simply a matter involving review of the data to ensure that it is compatible with the plant model. The transformer failures did result in power outages and, as such, they should be included in Scholl's data base. However, to avoid double accounting, they should be removed from the data applied to the Zion models. Since Scholl did not develop his data base specifically for use in the Zion Probabilistic Safety Study, it is not surprising that his data is not precisely compatible with the Zion models. However, because it was not developed for this purpose, it should not be broadly used as an authoritative source without first carefully examining its applicability to the study.

We are not going to discuss each of Scholl's data entries in this response, but one practice deserves some attention. In some cases, there is an obvious multiple accounting of single power failure events. We cannot determine the reasons for this practice, but we believe it may be related to Scholl's use of the data for cause and duration of specific events, and it may have resulted from the nature of his questionnaire responses. However, this multiple accounting has a significant effect on the reported frequencies of power failure at selected sites, and it provides totally misleading results for the application made in the Zion Probabilistic Safety Study. The most extreme example of this accounting practice involves the loss of all offsite power to Millstone Units 1 and 2 on August 10, 1976. The passage of Hurricane Belle caused severe salt spray coating of the entire Millstone switchyard, and several insulator flashovers resulted in the loss of offsite power to both units. Scholl's data base accounts for eight separate losses of offsite power from this single event (seven for Unit 1 and one for Unit 2). His event times indicate several partial restorations of power in rapid succession, but this event should be included as a single functional loss of power to the entire site. The other instances of this multiple accounting are events at Point Beach on October 13, 1973 (listed as two events for Unit 1 and one event for Unit 2), Indian Point on July 13, 1977 (listed twice for Unit 3), Beaver Valley on July 28, 1978 (listed twice), and San Onofre on April 22, 1980 (listed twice).

Scholl has reported his data on a reactor unit basis. As noted by the BNL reviewers, it is more meaningful to index this data to the plant site, regardless of the number of operating reactors. A single loss of offsite power at a three-unit site is counted in the Scholl report as three separate power failures, one for each of the reactors. This accounting practice results in a very large total number of power failure events and explains most of the differences between Scholl's reported 109 total losses of offsite power and the 58 events included in the new PLG data base for all plant sites. Scholl's accounting could result in consistent frequencies for the loss of offsite power per plant site if the "ages" of the reactors at each site were correctly assessed. Unfortunately, his methodology for determining the plant "age" results in different success data for each unit at a site and make it extremely difficult to assess a meaningful composite power failure frequency for the site.

The BNL reviewers were apparently aware of many of the shortcomings of Scholl's report, because they modified his data significantly for their comments on the Zion Probabilistic Safety Study. Unfortunately, they did not document this fact, nor did they provide any reasons for this modification. Table 3 provides a comparison between the data from Scholl's report and the data which BNL has reported as being derived from Scholl's work (Reference 4, page 6 and Table 3). The Table 3 column labeled "Scholl Modified" is our attempt to combine Scholl's data for multiple reactors at a single site and for multiple events resulting from a single functional cause. We believe that the data in this column best represents the applicability of Scholl's work to the frequency of total failures of offsite power per plant site (including transformer and switching failures not directly applicable to the Zion plant models). We cannot explain the differences between this column and the BNL data. The BNL comments do not mention these differences, nor do they acknowledge the fact that BNL has modified Scholl's work, leaving the reader to believe that their conclusions are drawn directly from Scholl's report.

We have also discussed Scholl's use of the plant "age" data. We cannot identify the source or the rationale for the BNL data for "years in operation," except that the data is obviously different from Scholl's. A brief reference to this data is made on page 6 of Reference 4, where it is noted that different units of the same site have been treated as one unit with total operating time being the time of the oldest individual unit. Since the times are not generally consistent with Scholl's, we cannot determine their source. Of course, this success time data must be compatible with the failure event data, and it is of equal importance in determining the failure rates applied in the BNL review. Since many of the BNL comments expressed concern about the subjectivity and lack of traceability in the Zion Probabilistic Safety Study, we have difficulty understanding their reasons and methods for these undocumented modifications of Scholl's data.

2.8.3 DATA FOR TIME TO RESTORE OFFSITE POWER

The BNL reviewers also refer to the Scholl report to justify their use of generic data for the time to restore offsite power. The general inapplicability of this approach for the Zion Probabilistic Safety Study is discussed in item 2.3 of this response. However, as with the frequency data, BNL appears to have modified Scholl's data to arrive at their estimates for the time to restore power. Table 4 presents our summary of Scholl's 44 power failure events for the sites in the Zion data base, with the corresponding times to partial and total offsite power recovery as noted in his data entries. This data provides the following information:

- Frequency of failure to restore offsite power within 30 minutes: 0.58.
- Frequency of failure to restore offsite power within 60 minutes: 0.36.

These statistics are more pessimistic than those derived by the BNL reviewers and are comparable to those summarized in item 2.3 for the new PLG data base. However, it is important to note from Table 4 that the time for partial power restoration was available for only 18 of the 44 events. Successful offsite power restoration was defined for the Zion model as recovery of any one of the six offsite lines. The average time to partial power restoration is approximately half the time to total power recovery for the events in which both times are given. Therefore, it is possible that the partial power recovery times for the remaining events could be significantly lower than the total recovery times, and the generic failure recovery statistics could be improved. We simply note this data and reemphasize the fact that generic recovery times under nonemergency conditions are not directly applicable to the scenarios modeled in the Zion Probabilistic Safety Study.

2.8.4 SUMMARY

In summary, we believe that R. F. Scholl's work is an important contribution to the collection and categorization of detailed data on the failures of offsite power at U. S. nuclear power plants. We disagree strongly with his methodology for computing annual power failure rates, because of his biased application of the population success data. We also believe that his use of "normalized" annual event rates as a comparative measure of unit experience versus statistical "targets" is inappropriate. However, Scholl's statistical treatment of the data apparently has not strongly influenced the BNL reviewers.

It is evident that BNL modified Scholl's data for their use in the Zion Probabilistic Safety Study review, although there is no mention of this fact in their comments. The BNL data is questionable to the extent that their failure events may be influenced by Scholl's multiple accounting and their operating years may be influenced by Scholl's "age" data. We cannot verify either of these possibilities. It is regrettable that the BNL analysts neglected to document their departures from Scholl's work, because their comments lead us to believe they adopted Mr. Scholl's data directly from his report.

2.9 FAILURE FREQUENCY FOR MOTOR-OPERATED VALVES

The BNL reviewer criticizes the frequency of failure assigned to motor-operated valves and the mean β factor used to include common mode failures for the containment sump to RHR pump suction valves. The BNL reviewer proposes a frequency of failure of 0.03 and a β factor of 0.15, while the Zion PRA assigned a frequency of failure of

1.55×10^{-3} and a β factor of 0.014. This section addresses these conflicts.

- Frequency of Failure

  The BNL frequency of failure on demand, 0.032, is obtained using Reactor Safety Study (RSS) data. A review of the RSS was made to determine the source of this frequency of failure.

  In Appendix II, Section 5.9 (Low Pressure Recirculation System or LPRS) of the RSS, the frequency of failure for an MOV that must change position is developed. The sum of the failures for this MOV is 3.3×10^{-2}, but this sum includes a check valve and the MOV breaker's failing to close. As MOV breakers are usually closed, removing the breaker failure rate and the check valve failure rate leaves a sum of 3.2×10^{-2} which appears to agree with the BNL value.

  The LPRS value (3.2×10^{-2}) is developed in the fault tree presented in Figure II 5-65 and quantified in Table II 5-32. A similar frequency of failure (1.9×10^{-2}) for an MOV is used in Section 5.6.4 [high pressure injection system (HPIS)], Appendix II, of the RSS. This is developed in the fault tree presented in Figure II 5-45 (Sheet 3) (MOV 1) and quantified in Table II 5-23. Both values are dominated by hourly failure rates for undetected failures based upon annual cycling of the valves; however, Zion valves are tested quarterly. Furthermore, these values do not agree with MOV failure rates on demand presented in other system analyses of Appendix II of the RSS (for example the RSIS and the HPRS) (1×10^{-3}) and in fact do not agree with the guidance presented in Appendices III and IV, Section 4.1.2 of the RSS. That guidance is that the failure of a valve to operate includes changing state from closed to open or open to closed. Section 4.1.4 states that "Available experience data do not permit separation of motor failure from pump failure. Therefore, separate motor failure rates for pump and valve drive motors should not be included." Similar statements are included in Sections 4.1.5, 4.1.6, and 4.1.11. In Table III 4.1, the failure of MOVs to operate on demand, Q_d, includes the driver, but does not include input control signals (such as the SICS signal). Finally, in Section 3.3 of Appendix III to the RSS, the failures presented for motor-operated valves include those failures typically associated with the motor control circuit (limit switches, torque switches, motors, contacts, etc.).

  A review of other data sources (Reference 14) indicates that the failures included in the HPIS MOV development and the LPRS MOV development are included in the frequency of MOV failure on demand. Therefore, requantifying these failures and summing with data that already includes such failures overestimates the frequency of failure of a single MOV on demand by as much as an order of magnitude.

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The Zion MOV failure data used for updating the generic error distribution included as failures all control circuit failures (limit switches, torque switches, control switches, motors, contacts, etc.). For these reasons, the failure frequency of MOVs as presented in the Zion PRA is more appropriate than the value presented in the BNL review.

- Dependent Failures

To quantify the effect of dependent failures of motor-operated valves (SI8811A and SI8811B) in the lines from the containment sump to RHR pumps, BNL uses a value of 0.15 for the β factor. This number is based on the estimate of another BNL document (Reference 15) in which the value of β was obtained by modifying the estimate of Fleming (Reference 16). The latter gave a β value of 0.23 for all valves. It also observed that the variation of β for a number of diverse equipment types had a fairly limited range (from about 0.1 to about 0.2). It was argued in Reference 15 that since the failures considered in the model did not include control circuit failures, a number smaller than 0.23 was more apppropriate. Therefore, it was decided to use an average value of 0.15 for pumps and valves.

The mean value estimate (β = 0.014) used in the Zion PRA (Reference 2) was also arrived at subjectively. It basically represented the analyst's opinion that a smaller value was more representative of the type of dependencies modeled by β factor in the recirculation system analysis. Such dependencies did not, for instance, include those resulting from acts of test and maintenance mainly because such failures are treated explicitly in Zion systems analyses. It was therefore believed that the β factor for the analysis represents a subclass of the dependent failures usually used to estimate β factors.

However, as can be seen from the definition of β factor, it is not clear that by excluding certain classes of failure that the value of β would decrease. Depending on the number of dependent and independent failures removed from the data base, the value of β may decrease or increase.

Neither BNL (Reference 1) nor Zion PRA (Reference 2) estimates of the β factor for MOVs can be fully justified on the basis of data. A recent PLG survey of about 200 MOV failures (PWR safety systems) and classification of these failure events in a manner consistent with PLG systems analysis approach indicate that for MOVs, β is about 0.06. This value represents the mean of the posterior distribution of β where some "potential dependent failures" are also included in the evidence by means of a weighted likelihood technique.

Preliminary review of a number of other failures not currently included in the data base indicate that the above estimate is slightly conservative and that the use of expanded data would result in a smaller value.

It must be pointed out that the above value is a generic estimate in the sense that it covers MOVs in various applications and systems configuration. To obtain a β factor more appropriate for the valves in question, the data base was further scrutinized and failures not judged as being applicable were excluded. The failures excluded consisted of common cause, potential common cause, and independent failures due to common environmental conditions at the valves (the valves in question are "canned" and thus are protected from common environmental failures), and those failures that are system specific (i.e., MOV failures caused by concentrated boric acid, typically isolation valves for the boron injection tank; failures due to high differential pressure, typically the safety injection system or high head injection system; and failures due to internal pressure buildup between the valve discs of large gate valves, precluded by design at Zion). The resulting mean value of β was 2.6×10^{-2}. This result indicates that the mean value used in the Zion PRA (Reference 2) is much closer to a realistic estimate of β than all other generic estimates discussed here.

2.10 TURBINE TRIP, LOSS OF OFFSITE POWER EVENT TREE

First of all, we wish to acknowledge that the BNL review called our attention to errors in the loss of offsite power event tree model. We have now revised that analysis and the changes are included as Attachment 1 to this response.

The new model is more complete in terms of tracking sequences involving recovery from all electric power states. Along with correction of nonconservative numerical and logic errors, we corrected the overly conservative assumption that a seal LOCA leads to melt. With the recovery of electric power, bleed and feed cooling with high pressure injection can lead to success. It should be noted that the corrections lead to no changes in release category frequencies or consequences. However the following changes in plant damage state mean frequencies do occur:

Plant State	Old Frequency	Revised Frequency
SEFC	7.41-6	7.41-6
SEF	1.28-9	1.30-9
SEC	1.76-8	1.80-8
SE	6.53-10	4.50-9
SLFC	1.91-5	1.91-5
SLF	4.76-9	4.79-9
SLC	1.93-6	1.93-6
SL	1.25-8	1.26-8
TEFC	8.43-7	9.13-7
TEF	1.61-9	2.14-9
TEC	9.32-7	9.54-7
TE	2.27-7	2.29-7

Plant State	Old Frequency	Revised Frequency
AEFC	1.75-6	1.75-6
AEF	1.87-10	1.87-10
AEC	8.23-9	8.23-9
AE	1.05-11	1.05-11
ALFC	9.76-6	9.76-6
ALF	7.27-10	7.27-10
ALC	3.98-10	3.98-10
AL	2.52-13	2.52-13
V	1.05-7	1.05-7

On page 2-13 of Reference 1, BNL states that a reactor coolant pump seal LOCA will occur after 15 minutes with no component cooling water or seal injection flow. It appears much more likely that such failure requires at least 30 minutes to develop (Internal Westinghouse Electric Corporation Memo from C. L. Gottshall to D. J. Lokay, NS-MSE-845, "RCD Seal Behavior," dated February 18, 1977). We did not take advantage of the opportunity to avoid the LOCA by recovering power.

2.11 BLEED AND FEED MODEL

BNL states (page 2-1) that no analysis is given to support the bleed and feed model used in the study. The Westinghouse reports "Report on Small Break Accidents for Westinghouse NSSS System," WCAP-9600, Vol. III, June 1979, and "Loss of Feedwater Induced Loss of Coolant Analysis Report," WCAP-9744, May 1980, cited in the study provide the basis for our model. Because of the design of the Zion charging pumps, the model is conservative for Zion. A recent sensitivity study by Westinghouse, WCAP-9915, December 1981, gives additional support.

2.12 AEROSOL PLUGGING OF THE FAN COOLERS

BNL estimates the probability of fan cooler plugging by aerosols, given a core melt and failure of the fan coolers, as 0.1 (page 2-15). Based on the Zion fan cooler design, this event appears to be impossible for the following reasons:

- There are only two ways to generate substantial aerosols. First, the vaporization release during concrete attack leads to small (1μ) size particles that can levitate easily. However, if either the fan coolers or containment sprays are operating, water will be present in the reactor cavity and concrete attack will not proceed. If the fan coolers are not working, we are not interested in plugging. Secondly, the dispersive event can produce large (100 to 1000μ) particles which would tend to settle out quickly if levitated above the basemat level.

- Entry to the fan coolers is very high in the containment.

- Even if the filters should become plugged, the Zion fan coolers can pass enough flow (there is a filter bypass flow path) to provide adequate cooling.

- The aerosols will not be sufficient to completely plug the fan coolers.

2.13 ACCIDENTS DURING HOT STANDBY AND COLD SHUTDOWN

Although LOCAs during hot standby are included in the Zion PSS, BNL is essentially correct in stating that core melt accident sequences during hot standby and cold shutdown (pages 2-6 and 2-7) are not included. While they can surely be modeled within the existing methodology, it is felt that they do not significantly contribute to risk and therefore it would not be particularly valuable to analyze them in detail. Most important in this assessment are the following:

- When not operating at power, the plant is not susceptible to the most frequent disturbances, trips due to control system instabilities and spurious signals requiring successful operation of standby systems.

- Reduced decay heat levels mean much longer recovery and response times and much smaller makeup requirements. For example, in situations with the head removed and RHR pump failure, losses are by boiloff and any low capacity, low head source of water will be sufficient. The operators are not limited to the RWST and safety injection pumps.

2.14 REACTOR COOLANT PUMP TRIP FOR SMALL LOCA

Tripping of the reactor coolant pumps for specific small LOCAs is based on analysis included in WCAP-9600 where calculations were based on limiting PCT to 1800F. Failure to trip the pumps does not lead to the kind of extensive core damage of interest to the risk study and is therefore not included in the Zion PSS. It is worth noting that more recent Westinghouse calculations using the NOTRUMP computer code developed specifically for analyzing small break situations shows even less severe results. Also, if high head injection works, reactor coolant system pressure never falls to the point (1,200 psia) at which RCP trip is required.

2.15 MULTIPLE INSTRUMENT TUBE SMALL LOCA

The frequency of this event is included in the small LOCA frequency of the Zion PSS. Even though it is below the core, the same success criteria hold as for any other small LOCA. Analysis and discussions of this event were provided as part of RESAR414 by Westinghouse and were reviewed by the NRC and ACRS. Although probabilities were not calculated, qualitative mechanistic arguments were presented.

2.16 CORE MELT DUE TO ATWS

The BNL reviewers correctly point out that the peak pressure following ATWS occurs in about 2 minutes rather than 10 minutes as modeled in the study. However, we disagree strongly with their use of the Handbook of Human Reliability Analysis to requantify the necessary operator action. Furthermore, as stated in the Zion study, the ATWS analysis included overwhelming conservatisms. We have now revised that analysis and the changes are included as Attachment 2 to this response. Some of the more important changes are addressed in the following comments.

First, we address BNL's use of the Handbook. They use pages 17-20 and 17-24 which provide human error rates for the time immediately following a large LOCA and generic performance rules to be used in the absence of more specific information. In the large LOCA situation, human reliability is modeled as low (typical of very high stress) "not only because of the stress involved, but also because of a probable incredulity response. Among the operating personnel the probability of occurrence of a large LOCA is believed to be so low that, for some moments, a potential response would likely be to disbelieve panel indications. Under such conditions it is estimated that no action at all might be taken for at least 1 minute and that if any action is taken it would likely be inappropriate." (Page 17-19)

This description does not apply to the ATWS case. First, all operators we have observed and interviewed respond to plant trip signals by immediately checking for turbine-generator trip (and decreasing load) and reactor trip (rod bottom lights). This is an almost automatic or "second nature" response with no hesitation (incredulity response) about completing those actions (trip the turbine-generator if it has not tripped, trip the reactor if it has not tripped, and carry out the required actions to shut down the reactor if reactor trip fails). Also even though ATWS is hypothesized to have potentially severe effects, operators do not seem to be as "nervous" about it as about a large LOCA. The stress level would not immediately be especially high. Second, as clearly laid out in the recirculation system analysis, three reactor operators (ROs) are in the Zion control room at all times. One is assigned to each unit's panel and the third, the center desk man, immediately responds to the unit in trouble. So even in the first 2 minutes, two operators are available to support the ATWS. The shift engineer (SE) and shift technical advisor (STA), both SROs at Zion, may also be involved within the first 2 minutes. At least one of the two must be in the control room; say he is the STA. Then the SE is most likely there, but may be in an adjacent area or anywhere else in the plant, perhaps as far away as the switchyard or the forebay of the crib-house. From discussions with plant operators, we believe the following discrete probability distribution is a reasonable model of the mean response time for the SE to arrive in the control room:

Time for SE to Reach Control Room (minutes)	Probability
0	0.80
0-1 (0.5)	0.10
1-5 (3)	0.08
5-20 (12.5)	0.02

Let us break the human response into two components: recognizing the failure to trip and performing the required actions to protect the plant. In the recognition phase, it is only necessary to observe the presence of a trip condition and no actual reactor trip; i.e., no rod bottom lights. High readings on nuclear instruments reinforce this observation. For this phase, we see little or no dependence among the operators and model the situation as low dependence.

From the earlier discussion and the remarks on page 17-9 of the handbook for "second nature" responses, it seems appropriate to consider the stress level optimum. The basic human error probability for this situation is 0.003. Then for low dependence, the center desk man's human error probability (HEP) is

$$\frac{1 + 19(.003)}{20} = 0.05$$

Since the STA (and SE if he is in the control room) will not respond as quickly and thus has less time to recognize the ATWS condition, we multiply his HEP by 2; i.e., 0.1. If the SE arrives in the control room within 1 minute, we again double his HEP to 0.2. Therefore, the total HEP for failing to discover the ATWS condition is

$$.8[0.003 \times 0.05 \times 0.1 \times 0.1] + .1[0.003 \times 0.05 \times 0.1 \times 0.2]$$

$$+ .1[0.003 \times 0.05 \times 0.1] = 3.0 \times 10^{-6}$$

After acknowledging very broad uncertainty in these results by assuming a lognormal distribution and assigning a range factor of 20, the mean HEP for recognizing the ATWS condition is 1.6×10^{-5}.

The first actions required of the operators, to manually trip the reactor and the turbine, are of a routine or automatic nature. To quote the handbook, "If personnel at a plant indeed have such frequent practice that the tasks in question could be regarded as 'second nature,' the HEPs assigned to the moderately high level of stress will not apply, as the stress level will be closer to optimum." (page 17-9) We expect the manual trip to be attempted immediately, before the real significance of the ATWS condition is appreciated. Nevertheless, because the timing is short, we double the basic HEP for the RO; i.e., 0.006. Then for low dependence, the center desk man's HEP is

$$\frac{1 + 19(.006)}{20} = 0.056$$

As above, we double this to 0.111 for the STA and the SE if he is in the control room and double it again if he arrives within the first minute. Thus, the total HEP for failing to initiate a manual reactor trip is

$$0.8[0.006 \times 0.056 \times 0.111 \times 0.111] + 0.1[0.006 \times 0.056 \times 0.111$$

$$\times 0.222] + 0.1[0.006 \times 0.056 \times 0.111] = 7.83 \times 10^{-6}$$

Assigning a range factor of 20, the mean HEP is 4.11×10^{-5}.

If the reactor still has not tripped, it is apparent to the operators that a very unexpected condition exists. Despite their extensive training for this situation, we believe the operators will feel high stress as they begin to carry out the ATWS emergency procedure. The first step after attempting the manual trips of the reactor and turbine is to drive in the control rods. If this action begins within 1 minute, it should successfully terminate the ensuing pressure rise. Under these conditions, we assign an HEP of 0.25 to the RO. The center desk man may be closely working with the RO, so we consider this as a case of high dependence with an HEP of

$$\frac{1 + 0.25}{2} = 0.63$$

Because the STA and SE will be delayed in responding, probably until the RO's concern is voiced, we consider them moderately dependent

$$\frac{1 + 6(0.25)}{7} = 0.357$$

but double this value because of the time constraint to 0.71. Remember though that the required action is simple. In fact, all the STA really needs to do is say, "why aren't you driving rods?" and the event could be terminated. Finally, if the SE (or STA) is outside the control room, we give him no credit in helping the situation. Thus, the total HEP for failing to drive rods within 1 minute is

$$0.8[0.25 \times 0.63 \times 0.71 \times 0.71] + 0.2[0.25 \times 0.63 \times 0.71] = 8.59 \times 10^{-2}$$

If we assign a range factor of 10; i.e., the upper bound is 0.859, then the mean HEP is 0.23 for failing to drive control rods given that automatic and manual trip have failed.

If the reactor has not been tripped and inward rod motion has not begun within 1 minute, and if the pressure is successfully controlled by the relief and safety valves, we next look for reactor shutdown by manually deenergizing power to the rods. Here, we assume high stress (0.25) for the RO in deciding to carry out the action, complete dependence for the center desk man (1.00), high dependence for the

$$\text{STA} \quad \frac{1 + 0.25}{2} = 0.63$$

and moderate dependence for the

$$\text{SE} \quad \frac{1 + 6(0.25)}{7} = 0.36$$

We neglect the SE if he has not returned within 5 minutes. Thus, the total HEP for deciding to disable power to the rods is:

$$0.98[0.25 \times 1 \times 0.63 \times 0.36] + 0.02[0.25 \times 1 \times 0.63] = 5.87 \times 10^{-2}$$

Assigning a range factor of 10; i.e., the upper bound is 0.59 and the mean is 0.156.

Finally, the procedure specifies that the RO send the equipment operator ("A" man) to trip the breakers locally. Although he is not stressed, we double the basic HEP to 0.006 and the total HEP is 0.156 + 0.006 = 0.162.

Other important changes to the ATWS analysis include:

- The fact that the Zion PORVs have been modified to prevent leakage. The PORV block valves are now kept open, so manual action is no longer required for pressure relief.

- The fraction of time the PORV must open to control the ATWS pressure rise due to unfavorable moderator coefficient was erroneously given as 0.1 when it should have been 0.01.

- A new branch has been added to account for the fact that most over-pressure conditions will not disable the safety injection system. Most now branch to the small LOCA event tree.

Results of the revised analysis show the ATWS contributions to core melt and risk to be much smaller than calculated previously.

2.17 COMPONENT COOLING WATER

For the event "Loss of Component Cooling During Plant Operation," the results of the CCW analysis presented in the PRA are not appropriate. Because this event results in a plant transient (reactor trip) and eventual loss of RCP cooling, it has been analyzed separately.

System success requirements for this analysis are: (1) two running CCW pumps, and (2) one operable heat exchanger.

For the purposes of this analysis, the CCW system is quantified for failure during a single year of plant operation. The results of the analyses are presented below.

A. Single Failures

Rupture or gross leakage in the CCW system piping will result in rapid system degradation which will lead to system failure. From the system analysis, 30 major sections of pipe were identified where rupture or leakage results in system failure. This led to a frequency of system failure of

$$\phi_{\text{pipe leakage}} = 30\ (8.6 \times 10^{-8}) = 2.7 \times 10^{-8}/\text{hour}$$

The results per year at Zion are obtained as follows:

$$\phi_{\text{pipe leakage}} = (2.7 \times 10^{-8})(8{,}760)(0.9) = 2.1 \times 10^{-4}/\text{year}$$

since at least one unit is operating about 0.9 of the time [0.7 + 0.7 - (0.7)(0.7)].

Operator action to recover the CCW system after large leaks or ruptures is extremely likely due to the immediate indications available (low flow, low discharge pressure, and sump level alarms in the area served by CCW). However, failure of the pump suction header could result in failure of the CCW pumps if no operator action is taken to secure the CCW within a short period of time. There are five major sections of pipe in this area. Using the frequency of pipe failure from the CCW systems analysis, the frequency of pipe rupture in this area is:

$$\phi_{\text{pipe rupture}} = (5)(8.6 \times 10^{-10}) = 4.3 \times 10^{-9}/\text{hour}$$

$$= (4.3 \times 10^{-9})(8{,}760)(0.9) = 3.4 \times 10^{-5}/\text{year}$$

Using .5 as the frequency of operator error (failing to secure the running and standby CCW pumps), the contribution to system failure from suction piping rupture is

$$\phi_{\text{CCW/pipe}} = 1.7 \times 10^{-5}/\text{year}$$

B. Heat Exchanger Failure

Failure under the conditions of this analysis is defined as failure of all heat exchangers during plant operation. A detailed review was conducted to determine the frequency of heat exchanger gross failures (failure that results in complete loss of function over a short period of time). No failures were found in Zion operating experience. Potential common mode failures, such as mussel or clam fouling have not been detected at the Zion units. The predominant mode of failure identified for the CCW heat exchanger is gradual degradation of heat transfer capacity over long periods of time. This type of failure will be detected and corrected prior to complete system failure.

Failure of the heat exchanger isolation valves can result in rapid failure of a single heat exchanger train. This effect is quantified below. Each heat exchanger has four valves where failure can result in failure of a single heat exchanger train. This results in a failure of a heat exchanger train of:

$$\phi_{HX} = 4(5.28 \times 10^{-8})/\text{hour}$$

$$= 2.1 \times 10^{-7}/\text{hour}$$

Over a year, this results in a frequency of failure of a single operating heat exchanger train of:

$$\phi_{HX} = (2.1 \times 10^{-7})(8,760)(0.9)$$

$$= 1.7 \times 10^{-3}/\text{year}$$

Failure of the standby heat exchangers can be caused by failure of the inlet isolation valve to open and operator error in aligning the heat exchanger. Errors in alignment are detectable immediately and are excluded in this analysis. The split fraction for failure of a manual valve to open on demand is assumed to be the same as a check valve failing to open on demand; 4.32×10^{-5}. The split fraction for failure of the standby heat exchanger is:

$$f_{HXSTBY} = [3 \times 5.28 \times 10^{-8}/\text{hour}]\ 8,760 + 4.32 \times 10^{-5}$$

$$= 1.4 \times 10^{-3}$$

For failure of three heat exchangers, the results are:

$$\phi_{\text{Heat Exchanger}} = (1.7 \times 10^{-3})^2 (1.4 \times 10^{-3})$$

$$= 4.0 \times 10^{-9}/\text{year}$$

C. Pump Trains

From plant operating history, the probability of having zero pumps in standby during plant operation is 7.7×10^{-3}. From this state, system failure will occur if two of the three operating pumps fail. Failure of a single operating pump in this condition will result in plant power reduction and eventual shutdown, but will not result in loss of the CCW system. The frequency of failure of an operating CCW pump is 1.87×10^{-6}/hour.

Both units can only remain at power with three pumps operating and no standby pumps for 1 week. After 1 week, one unit must shut down; the other unit may remain in operation indefinitely. The frequency of failure of the CCW system from this state is expressed in the following equation:

$$P(F)_{pumps} = P(OS)\ P(run)\ \left(\frac{8,760 \text{ hours}}{\text{year}}\right)\ (.01)$$

$$\phi_{pumps} = (7.7 \times 10^{-3})(1.87 \times 10^{-6}/\text{hour})(8,760 \text{ hours/year})(.01)$$

$$= 1.3 \times 10^{-6}/\text{year}$$

(.01 is the beta factor used for the frequency of failure of more than 1 operating pump due to common cause.)

D. System Failure

The frequency of system failure is the sum of the frequency of system failure due to piping failure, heat exchanger failure, and pump failure. The result is:

$$\phi_{system} = 1.7 \times 10^{-5} + 4.0 \times 10^{-9} + 1.3 \times 10^{-6}$$

$$= 1.8 \times 10^{-5}/\text{year}$$

This result is dominated by piping failures which are not recoverable by operator action. Other causes of system failure such as fire, earthquake, etc., have been considered in the PRA analysis.

2.18 COLD OVERPRESSURIZATION

At the time of the Zion PSS, it was felt that overcooling and cold shutdown overpressurization events would not significantly increase the mean frequency of catastrophic vessel failure over the Zion value of 3×10^{-7} per year taken from WASH-1400. Recently, the Westinghouse Owners Group has been conducting extensive work in that area. Evaluations of that new work should be available within the next few months. Moreover, it is important to realize that Zion has installed cold overpressurization protection. After cooldown and depressurization in preparation for going onto closed loop residual heat removal, Zion procedures require the operator to open the PORV block valves and to turn a single switch for automatic low pressure protection. That switch changes the relief setpoint for both PORVs to a low value appropriate for cold shutdown conditions.

2.19 REACTOR COOLANT PUMP MISSILES

Reactor coolant pump missiles following large LOCA are not included in the Zion model. Pump flywheels must be in compliance with NRC Regulatory Guide 1.14. Reference 12 finds that the conditional probability of missile generation given a LOCA is much less than 10^{-8}.

2.20 ZION UNIQUE FEATURES TO PREVENT CONTAINMENT LEAKAGE

The Zion units have two systems which serve to limit the fission product release from the containment and ensure no leak paths exist before or after an initiating event. These systems were unique at the time the plants were licensed. The two systems (described in the Zion FSAR in Sections 6.6.5 and 6.6.6) are the isolation valve seal water system and the containment penetration and weld channel pressurization system. They are described briefly below.

Isolation Valve Seal Water System. The isolation valve seal water system assures the effectiveness of those containment isolation valves that are located in lines connected to the reactor coolant system, or that could be exposed to the containment atmosphere during any condition which requires containment isolation, by providing a water seal (and in a few cases a gas seal) at the valves. The system provides a simple and reliable means for injecting seal water between the seats and stem packing of the globe and double disc types of isolation valves, and into the piping between closed diaphragm type isolation valves. This system operates to limit the fission product release from the containment. It is designed as an engineered safety feature and provides assurance that the containment leak rate is lower than that assumed in the accident analysis should an accident occur.

The system consists of a pressurized seal water tank, compressed nitrogen bottles, and the necessary piping and valves to individual components. System operation is initiated either manually or by any automatic safety injection signal. The components served by the isolation valve seal water system are listed in the FSAR.

Containment Penetration and Weld Channel Pressurization System. The containment penetration and weld channel pressurization system provides means for continuously pressurizing the positive pressure zones incorporated into the containment penetrations and the channels over the welds in the steel inner liner in the event of a loss of coolant accident (LOCA). It is designed as an engineered safety feature and provides assurance that the containment leak rate in the event of an accident is lower than that assumed in the accident analysis.

A regulated source of clean and dry compressed air (100 psig) from either the instrument air system (normal supply--three compressors) or the penetration pressurization system (backup supply--three compressors) is supplied to all containment penetrations and inner liner weld channels. A standby source of gas pressure from compressed nitrogen cylinders (at about 50 psig) will automatically deliver nitrogen at this reduced pressure in the event the normal and backup air sources are lost. Each one of these sources supplies four independent air receivers which in turn supply compressed air or gas to the penetrations and weld channels in four pressurization system zones as shown in the FSAR. The system alarms on low pressure or high air flow.

TABLE 1

PLG REVISED LOSS OF OFFSITE POWER DATA BASE

Site	Years*	Total Events**	Note
1. Yankee Rowe	21	1	15
2. Indian Point	19	5	3
3. San Onofre	15	1	4
4. Connecticut Yankee	14	4	5
5. R. E. Ginna	12	1	6
6. H. B. Robinson	11	0	
7. Point Beach	11	0	
8. Palisades	11	4	7
9. Maine Yankee	9	1	16
10. Surry	9	0	
11. Oconee	9	0	
12. Fort Calhoun	8	1	8
13. Kewaunee	8	0	
14. Arkansas One	7	2	9
15. Three Mile Island	8	0	
16. Calvert Cliffs	7	2	10
17. Trojan	6	0	
18. Millstone	11	1	11
19. D. C. Cook	7	2	12
20. Prairie Island	8	1	13
21. Turkey Point	9	8	14
22. Zion	9	0	
Total	229	34	

*See note 1.
**See note 2.

Notes for Table 1

1. Time from "Date of Initial Criticality" listed in the NRC "Grey Book" summaries (NUREG-0020) through December 31, 1981, rounded to the nearest whole year.

2. A loss of offsite power event causes failure of all offsite power to a unit or failure of all automatically available offsite sources. Single transformer failures and failures of in-plant breaker transfer circuits are excluded.

3. Indian Point events are:

   11/9/65: Unit 1 shutdown (Northeast blackout).
   12/70: Unit 1 at cold shutdown.
   7/20/72: Unit 1 tripped from power operation.
   7/12/77: Unit 2 at cold shutdown; Unit 3 at 91% power.
   6/3/80: Unit 2 at 100% power; Unit 3 remained at power operation through the event, although the diesel generators were required to operate.

4. San Onofre 1 was at cold shutdown.

5. Connecticut Yankee events are:

   7/27/68: Unit operating condition not specified.
   7/15/69: Unit at 50% power.
   7/19/74: Unit operating condition not specified.
   6/26/76: Unit at cold shutdown.

6. Ginna tripped from power operation.

7. Palisades events are:

   9/2/71: Unit at cold shutdown.
   7/24/77: Unit at 100% power.
   11/25/77: Unit at 85% power.
   12/11/77: Unit at 100% power.

8. Fort Calhoun tripped from 98% power.

9. Arkansas One events are:

   4/8/80: Unit 1 operating condition not specified; Unit 2 shutdown.
   6/24/80: Unit 1 at 100% power; Unit 2 at 91% power.

10. One event which occurred on 12/20/73, during construction is not included in the data base for Calvert Cliffs. The other events are:

   4/11/78: Unit at approximately 80% power; Unit 2 at appro..imately 75% power.
   4/13/78: Unit 1 at approximately 80% power; Unit 2 at approximately 50% power.

11. Millstone Unit 1 at 45% power; Unit 2 at 100% power. The event occurred during Hurricane Belle. Severe salt spray coated the switchyard and caused several failures due to insulator flashover.

12. D. C. Cook events are:

    2/1/75: Unit at 6% power (testing).
    9/1/77: Unit at 100% power.

13. Prairie Island Unit 1 at cold shutdown; Unit 2 at 100% power.

14. Turkey Point events are:

4/3/73:	Unit 3 at 73% power; Unit 4 under construction (testing).
4/4/73:	Unit 3 at 73% power; Unit 4 under construction (testing).
03/74:	Unit 3 at 73% power; Unit 4 at 60% power.
4/25/74:	Unit 3 at 97% power; Unit 4 at 98% power.
6/28/74:	Both units tripped from power operation.
5/16/77:	Unit 3 at 100% power; Unit 4 at cold shutdown.
4/4/79:	Unit 3 at shutdown; Unit 4 at approximately 90% power.
8/3/79:	Unit 3 at 100% power; Unit 4 at 100% power.

15. Yankee Rowe was shutdown. The event date is 11/9/65 (Northeast blackout).

16. Maine Yankee tripped from 47% power.

TABLE 2*

POPULATION TIME AND EVENT DATA

LOSS OF OFFSITE POWER

		DOCKET #	"ZION" # FAILURES	"ZION" YEARS IN OPERATION	"SCHOLL" # FAILURES	"SCHOLL" YEARS IN OPERATION
1.	Yankee Rowe	29	9	15	1	15
2.	Indian Point 1	3	3	12	7	12
	Indian Point 2	247	1	5		
	Indian Point 3	286	3	3		
3.	San Onofre	206	0	8	4	12
4.	Con Yankee		4	8	4	
5.	R.E. Ginna	244	1	6	3	11
6.	H.B. Robinson		1	5	1	9
7.	Point Beach 1	266	3	5		
	Point Beach 2	301	0	4	4	9
8.	Palisades	255	0	4	6	9
9.	Maine Yankee	309	0	3	1	8
10.	Surry 1	280	2	4	0	8
	Surry 2	281	1	4		
11.	Oconnee 1	269	0	4		
	Oconnee 2	270	0	2	1	6
	Oconnee 3	287	0	3		
12.	Fort Calhoon	285	4	3	2	7
13.	Kewaunee	305	0	2	1	6
14.	Arkansas One	368	2	2	1	6
15.	TMI-1	289	0	2	0	6
16.	Calvert Cliffs	317	0	1	3	6
17.	Trojan	344	0	1	0	5
18.	Milistone 2	336	1	1	0	5
19.	D.C. Cook 1	315	0	1	1	5
20.	Prairie Island 1	282	0	3	0	6
	Prairie Island 2	306	0	2		
21.	Turkey Point 3	250	0	4	12	8
	Turkey Point 4	251	0	3		
22.	Zion 1	295	0		0	8
	Zion 2	204	0	11		
		TOTAL:	34	131	53	167

*Reproduced from Reference 2, Table 3

TABLE 3

COMPARISONS OF BNL AND SCHOLL DATA*

Site	Scholl Total Failures[1]	Scholl Modified[2]	Scholl "Age" (Years)[3]	BNL Total Failures[4]	BNL Years in Operation[4]
1. Yankee Rowe	1	1	16.98	1	15
2. Indian Point	7	4	7.87	7	12
3. San Onofre	4	3	12.37	4	12
4. Connecticut Yankee	5	5	12.1	4	-[6]
5. R. E. Ginna	3	3	11.5	3	11
6. Robinson	0	0	9.39	1	9
7. Point Beach	7	3	9.32	4	9
8. Palisades	6	6	8.91	6	9
9. Maine Yankee	1	1	1.76	1	8
10. Surry	0	0	5.98	0	8
11. Oconee	3	1	6.54	1	6
12. Fort Calhoun	2	2	6.76	2	7
13. Kewaunee	1	1	1.56	1	6
14. Arkansas One	2	1	5.28[5]	1	6
15. Three Mile Island	0	0	3.01	0	6
16. Calvert Cliffs	5	3	6.45	3	6
17. Trojan	0	0	4.13	0	5
18. Millstone	8	1	8.94	0	5
19. D. C. Cook	1	1	5.29	1	5
20. Prairie Island	0	0	3.16	0	6
21. Turkey Point	16	8	6.25	12	8
22. Zion	0	0	1.23	0	8
Total	72	44		52	

*See referenced footnote on following page.

Notes for Table 3

1. From Reference 1, Appendix A Table for LOP B. Total events for all units at the site.

2. Combination of single failures affecting multiple units on the same date and single failures listed more than once for the same unit. This summary was prepared by PLG from the data in Column 1 and should be the total number of unique losses of offsite power experienced at the site, including failures of transformers and in-plant switching circuits.

3. From Reference 1, Table 5 and Appendix B Table for Failure Rate. The age here is the largest unit age listed for all the units at a site.

4. From Reference 2, Table 3.

5. Only the age of Arkansas One Unit 2 is listed in Scholl's report.

6. BNL data does not list operating years for Connecticut Yankee.

TABLE 4

SCHOLL OFFSITE POWER RECOVERY TIMES*

Site	Time to Offsite Power Recovery (Hours)		
	Date	Partial	Total
1. Yankee Rowe	11/9/65	0.55	4.18
2. Indian Point	7/20/72	--	0.92
	5/6/77	--	--
	7/13/77	--	6.47
	6/3/80	2.42	14.75
3. San Onofre	6/6/73	--	4.98
	6/7/73	--	--
	4/22/80	0.07	0.07
4. Connecticut Yankee	4/27/68	0.48	0.93
	7/15/69	--	1.93
	7/19/72	0.02	0.02
	1/19/74	0.33	1.75
	6/26/76	0.27	0.28
5. R. E. Ginna	12/5/68	--	--
	3/4/71	--	0.50
	10/21/73	0.67	1.28
6. Point Beach	2/5/71	--	6.37
	4/27/74	--	--
	10/13/73	0.92	5.42
7. Palisades	9/2/71	0.93	0.93
	10/17/74	--	0.50
	9/24/77	--	4.75
	9/27/77	--	--
	11/25/77	--	3.50
	12/11/77	--	1.50
8. Maine Yankee	8/31/78	0.02	0.02
9. Oconee	1/4/74	--	1.00
10. Fort Calhoun	3/13/75	--	--
	8/22/77	0	1.83

*From Reference 1, Appendix A Table for LOP B.

TABLE 4 (continued)

Site	Time to Offsite Power Recovery (Hours)		
	Date	Partial	Total
11. Kewaunee	1/17/80	16.33	158.70
12. Arkansas One	4/7/80	--	0.32
13. Calvert Cliffs	12/20/73	--	--
	4/11/78	0.27	11.20
	4/13/78	--	5.83
14. Millstone	8/10/76	13.84*	31.24**
15. D. C. Cook	2/19/80	--	0.67
16. Turkey Point	4/3/73	--	--
	4/4/73	0.17	--
	3/1/74	--	--
	4/25/74	--	0.33
	6/28/74	0.18	0.93
	5/16/77	--	--
	4/4/79	7.23	8.77
	8/3/79	--	--

*Sum of partial recovery times for seven events for Unit 1 (see item 8.2).

**Sum of total recovery times for seven events for Unit 1 (see item 8.2).

3. COMMENTS ON SECTION 3.1, FIRES

3.1 PAGE 1

The discussion of probabilities and frequencies is basically correct. Let us take advantage however, of the reviewer's coin example to clarify the terminology a bit further. The reviewer uses "frequency," as we do, to mean a basic parameter in the random process model we are postulating to approximate the real coins behavior. This parameter manifests itself or can be interpreted on the one hand as the outcome of an infinite experiment, and on the other hand, as our state of confidence about the outcome of any finite experiment. We use the word "probability" to describe this state of confidence; the classicist uses it as the name for the parameter of the model. Moreover, while we may be uncertain about the value of the coins parameter, we nevertheless do not talk about this parameter as being a "random" variable. No, we say it is a fixed variable about which we are uncertain. We reserve the term "random variable" for quantities which truly change or fluctuate. Thus, if 47 heads are obtained in 100 trials, the frequency does not change but our state of knowledge about it does. While this comment has little to do with the fire analysis, we feel that it is important to establish a common language and understanding of what probabilistic analysis is all about, especially because the reviewer talks about "confusion" later. In this context, we never say that the frequency of heads is 0.5 but, rather, we assign a probability distribution to that frequency to express our state of knowledge about it and then we use Bayes' theorem to update our knowledge in the light of the evidence of 47 heads in 100 trials. This procedure is described in detail in Section 0 of the Zion Probabilistic Safety Study.

3.2 PAGE 1

> "There is confusion in the fire analysis, ...in the use of these concepts--the frequency of a random variable and the probability of a parameter."

The reasons provided later in the review do not justify this statement. We will return to it after we respond to the reviewer's technical objections.

3.3 PAGE 2

The fire growth part produces the <u>mean</u> time τ_V (and τ_H) for vertical propagation, not the <u>actual time</u> t_V. Similarly, the suppression model produces the <u>mean</u> suppression time τ_S from the available information in the literature and our own judgment.

3.4 PAGE 2-3

Equations (1-5) are correct and they reflect what we have done in the analysis.

3.5 PAGE 4

The discussion of Q_p shows a misunderstanding of what we have done. The pilot fuel, call it Q, is a truly random variable with a frequency distribution. The mean of this distribution we do not know. This mean value is representative of the conditions under which the fire starts. This is what we call Q_p and, since we do not know its value, we present a state-of-knowledge histogram. We then argue that the statistical variability about Q_p; i.e., whether the pilot fuel is 400 or 450 Btu with some frequencies is not as crucial as the state-of-knowledge uncertainty. This is tantamount to using a delta fraction for the frequency distribution of Q.

The problem is that the reviewer does not interpret Q_p as a mean value, but rather as a random variable (what we call Q above).

3.6 PAGE 5

The review also shows a misunderstanding of our suppression model. Our model is as follows:

The actual suppression time at Zion is a random variable with frequency distribution.

$$g_s(t_s;\tau_s) = \frac{1}{\tau_s} \exp\left(-t_s/\tau_s\right)$$

We do not know the mean value τ_s and, as usual, we express what we know about it in terms of a probability histogram. Our state of knowledge about τ_s is shaped by the Zion cable spreading room fire suppression capability and by the generic information that the General Atomic report contains. The GA report does not give us statistical data on t_s in the true sense of the words, contrary to what the reviewer claims. It gives us estimated times to "control" the fires. We have, therefore, used these estimates as an additional input to our analysis and not as "the" data. If one wanted to insist and use them as statistical data, one would use the "two-stage" Bayesian approach that was used in the analysis of the initiating event data of the Zion Probabilistic Safety Study.

The reason why we keep the frequency distribution of t_s, something which we did not do in the case of the growth time, is that the state-of-knowledge uncertainty cannot be claimed to overwhelm the statistical uncertainty. For example, an estimated suppression time of, say, 23 minutes, could have come from an exponential with mean 15 minutes or 30 minutes. The statistical variability is comparable to the state-of-knowledge variability.

What the reviewer does in deriving his $\overline{g}_s(t_s)$ is to completely ignore the site specific information, to completely trust the GA estimates, to use our subjective numbers; i.e., the grouping of the estimates, and to ignore the statistical variability of t_s. This is at least as crude a model as ours and it certainly does not justify his claim that we are "confused."

3.7 PAGE 6

The reviewer feels that longer suppression times should also be considered because the Browns Ferry fire was suppressed about 7 hours after it started. The study team judged that that fire was a lesson about the use of water in suppressing fires and that the same hesitation would not be repeated. Furthermore, the fires that we consider are not deep seated in cable penetrations; therefore, the effectiveness of other means for suppression would not be inhibited to the same extent. In any case, this is a legitimate difference of opinion. The important question is how significantly the results are affected.

The important part of the histogram for the conditional (given a fire) $F(t_s > t_v)$, i.e., the values greater than 0.01, is due to strong fires that grow very rapidly (the rising part of the curve of Figure 7.3-3). These fires can only be suppressed by quick action. The question is, then, by how much would one reduce the probabilities of the shorter mean suppression times (5 and 15 minutes) because one fire, due to special reasons at another plant, was allowed to burn for 7 hours. We do not believe that there would be a substantial reduction and, therefore, the conditional histogram would not change much.

As an example, we actually modified our histogram of mean suppression times as follows:

Probability:	0.39	0.22	0.22	0.11	0.06
τ_s (minutes):	5.0	15.0	30.0	60.0	420.0

This means that we very conservatively treat the Browns Ferry time of 7 hours (420 minutes) as a mean time to suppression for a class of fires similar to that at Browns Ferry. The new median value of the conditional frequency of two trays being in fire is about 6×10^{-2} (our value equals about 3×10^{-2}) and the new 95th percentile remains at about 0.8; i.e., our value. The mean value moves from about 0.17 to about 0.22. The fact that the reviewer also derives a value of 0.22 is entirely coincidental, since he does not include the Browns Ferry fire in his calculations. Again we emphasize that the above example is purely for illustrative purposes, and we reiterate that the original histogram given in the Zion study for τ_s still represents our state of knowledge.

3.8 CONCLUDING COMMENT

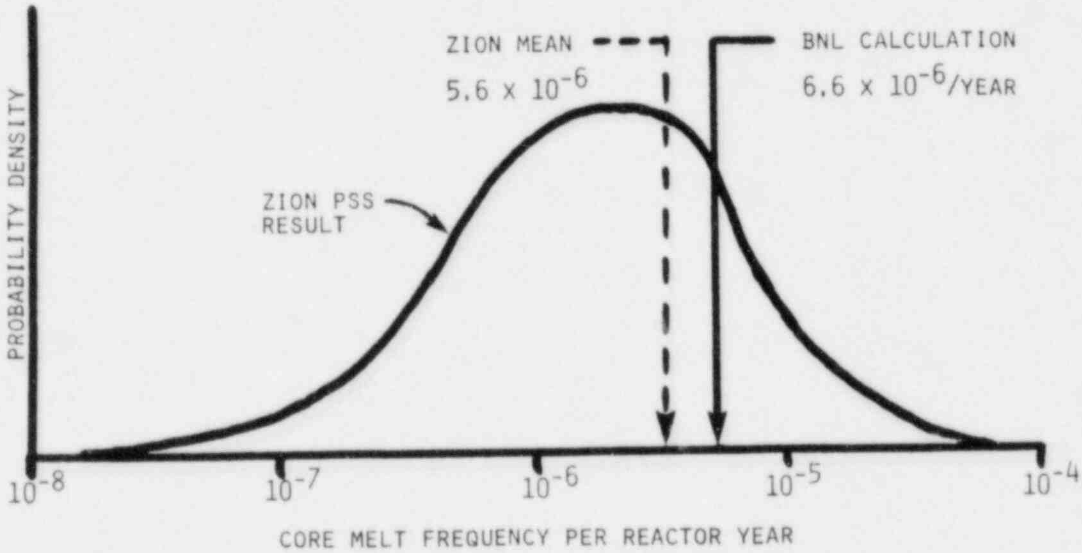
In reading the reviewer's comments we get the impression that he has failed to approach what we do in the proper context. We stated in the report that the state of the art of fire risk analysis had not been, when we started, sufficiently advanced. In our modeling efforts we have been very careful to state all our assumptions and to quantify our uncertainties. Because of this care, the uncertainty band is very large. This is exactly as it should be, because this is what our state of knowledge is. We firmly believe that the models that we have developed are reasonable. There is certainly room for refinements, and these will reduce the uncertainties.

Most of the assumptions that we have made are conservative. Apparent considerations must be evaluated in the appropriate context. For example, the reviewer complains about our reducing the importance of the 7 hour Browns Ferry suppresion time and he completely ignores our conservative assumption that the growth of the fire is not inhibited by the suppresion efforts. He does not even bother to look more carefully and assess the importance of this nonconservatism, as we did in comment number 7.

We are suprised that the reviewer has nothing to say when he "corrects an error in the statistical treatment" and all that he finds is that the mean core melt frequency is increased from 1.8×10^{-6}/year to 2.4×10^{-6}/year. He obviously ignores the fact that our histogram for this frequency ranges from 3×10^{-11} to 2.3×10^{-5} per year. Finally, we are convinced that the propagation of uncertainties in the analysis has been done correctly and rigorously. Nothing in the reviewer's comments justifies his claim about confusion.

4. COMMENTS ON SECTION 3.2, EARTHQUAKES

First of all, no actual review was performed. Instead, BNL assessed the sensitivity of the core melt frequency to the use of SSMRP seismic hazard function in place of the Zion seismicity curves. The result of this assessment is compared with the Zion probability curve below. The means differ by 18%, which reflects the differences in the seismicity curves and in the details of the numerical computation process. In light of these differences, 18% should be regarded as remarkable agreement. Moreover, the direction of the difference, BNL's mean higher than Zion's, is consistent with the fact that the SSMRP seismicity curve does not have the cutoffs present in the Zion curves.



5. COMMENTS BY S. KAPLAN ON THE BNL MEMORANDUM FROM I. A. PAPAZOGLOU TO R. A. BARI, January 19, 1982

"On the Two-Stage Bayesian Procedure for Determining Plant-Specific Frequencies for Initiating Events" (Reference 4)

I shall present my comments in the order in which they come up in reading Papazoglou's memorandum from front to back.

The first paragraph is a summary of the PLG two-stage Bayesian procedure and is correct.

The second paragraph cites two "methodological problems" in the PLG approach as follows:

(i) In the application of the method, if the second stage is to be used, the data for the specific plant should be excluded from the set of plant data input in the first stage to avoid double counting.

(ii) If the two-stage technique is applied a second time to incorporate additional information, it yields inconsistent results; that is treating the information in two successive packages gives a different result than treating it in one lump.

With respect to point (i), Papazoglou has previously pointed this out to me in person. The point is subtle, it has a minor effect numerically on Zion, as seen in Table 6 of Reference 4 and is still not accepted by everyone. Nevertheless, I personally think Papazoglou's correction is right, have incorporated it in a revised version of Reference 2 and in the IEEE publication, and have acknowledged him there as the source of the correction. The revised version of Reference 2 was put out in September 1981 and sent to Papazoglou at that time.

Item (ii) is actually just a restatement of item (i). The apparent inconsistency stems not from the two-stage technique itself but from applying it with the plant specific data included in stage 1. With this data excluded, the property of "noninformative sampling stopping" is present as in the conventional Bayesian approach because the second stage, of course, is a conventional Bayes approach.

In paragraph three, the "methodologically correct technique" mentioned refers again to item (i), excluding the plant specific data from the first stages. The rest of the paragraph is correct.

Paragraph four, beginning at bottom of page two, makes a good point about the overestimation of time. It should be noted, however, that while the effective operating years is not the sum of the two plants' individual years, neither is it just the larger of the two. Rather it is in between because with two plants on a site there is a reactor operating, and subject to LOOP, for a greater fraction of the year.

Paragraph five says "the identified errors in the logic...do not introduce any significant numerical error for the Zion application." Again, there is only one error, item (i) previously mentioned, and we agree with the conclusion on numerical impact. With respect to the treatment of LOOP events the numerical error could be noticeable percentagewise, with respect to the frequency of LOOP events themselves. However since in Zion such events contribute in only a minor way to public health risk, we think the numerical effect of this error on the final risk curves is insignificant. Similarly, the effect on the final risk curves would be inconsequential even if the vastly different data set of Scholl, Table 3 (Reference 4), were to be used.

With respect to the choice of lognormal curves, the issue of course is whether within the class of such curves chosen in any instance there are good approximations to the true population distribution. If the true distribution were bimodal for example one should use a class of bimodal shapes. The choice of shapes is in fact part of the prior distribution to stage 1 and thus reflects the set of information E_1. If one uses vastly wrong shapes, there will be "sensitivity." For any set of realistic shapes there should not be much sensitivity. In the applications we have dealt with, we feel that lognormal shapes are a good approximation to the physical situation, but of course we would be interested in hearing evidence to the contrary.

Section 2, beginning on page 4, provides a description of the two-stage technique. With the exception of the correction (i) noted earlier, this so-called "BNL technique" is identical to that put forth in Reference 2. Yet it is claimed here as being "developed at BNL." One could get the impression here that on the basis of the minor correction (i), BNL is attempting to claim the entire two-stage approach as its own.

6. REFERENCES

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3. Scholl, R. F., Jr., "Loss of Offsite Power, Survey Status Report, Revision 3," Report of the Systematic Evaluation Program Branch, Division of Licensing, U.S. Nuclear Regulatory Commission.

4. Memo from I. A. Papazoglou to R. A. Bari, January 19, 1982.

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6. Private sort of LER summaries from 1965 through April 30, 1981.

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14. EG&G Idaho, Inc., "Data Summaries of Licensee Event Reports of Valves at U.S. Commercial Nuclear Power Plants," NUREG/CR-1364, June 1980.

15. BNL Critique of the Offshore Power.

16. Fleming, K. N., and P. H. Raabe, "A Comparison of Three Methods for the Quantitative Analysis of Common Cause Failures," Paper X.3 in the Procedure of the ANS/ENS/OECD Topical Meeting on Probabilistic Analysis of Nuclear Reactor Safety, May 8 through 10, 1978, Los Angeles, California.

ATTACHMENT 2

REVISION TO THE ZION PROBABILISTIC SAFETY STUDY - ATWS ANALYSIS

REVISION 1 TO THE ZION PROBABILISTIC SAFETY STUDY
ATWS ANALYSIS

The Zion Probabilistic Safety Study model for ATWS events has been revised to eliminate excessive conservatisms and to correct an error with respect to event timing in the original analysis. The only observable effect on the overall study results is a very slight reduction in core melt frequency. While it is appropriate to provide the revised analysis, the major cost involved in propagating these results throughout the remainder of the study cannot be justified. Therefore, we recommend that you keep the original pages for reference in tracking results through the study and add these new pages for information and decision-making purposes. Changes to Sections 1.3.3.9 and 1.3.3.11 are included here.

New operator actions analyses OP-6 and OP-7 are added to Section 1.3.3.9. These replace the old OP-6:

6. OP-6 -- Rods in By One Minute and OP-7 -- Manually Deenergize and RCCAs Fall: Manual Actions to Shut Down the Reactor in ET-14 ATWS. Following ATWS, some initial rapid actions by the operators are nearly certain to occur. If the situation continues to degrade, stress builds and proper interpretations and successful actions become less likely. Peak reactor coolant system pressures occur at about 2 minutes into the transients, so early actions must occur quickly.

   First, all operators we have observed and interviewed respond to plant trip signals by immediately checking for turbine-generator trip (and decreasing load) and reactor trip (rod bottom lights). This is an almost automatic or "second nature" response with no hesitation (incredulity response) about completing those actions (trip the turbine-generator if it has not tripped, trip the reactor if it has not tripped, and carry out the required actions to shut down the reactor if reactor trip fails). Also even though ATWS is hypothesized to have potentially severe effects, operators do not seem to be as "nervous" about it as about a large LOCA. The stress level would not immediately be especially high. Second, as clearly laid out in the recirculation system analysis (Section 1.5.2), three reactor operators (ROs) are in the Zion control room at all times. One is assigned to each unit's panel and the third, the center desk man, immediately responds to the unit in trouble. So even in the first 2 minutes, two operators are available to support the ATWS. The shift engineer (SE) and shift technical advisor (STA), both SROs at Zion, may also be involved within the first 2 minutes. At least one of the two must be in the control room; say he is the STA. Then the SE is most likely there, but may be in an adjacent area or anywhere else in the plant, perhaps as far away as the switchyard or the forebay of the cribhouse. From discussions with plant operators, we believe the following discrete probability distribution is a reasonable model of the mean response time for the SE to arrive in the control room:

Time for SE to Reach Control Room (minutes)	Probability
0	0.80
0-1 (0.5)	0.10
1-5 (3)	0.08
5-20 (12.5)	0.02

Let us break the human response into two components: recognizing the failure to trip and performing the required actions to protect the plant. In the recognition phase, it is only necessary to observe the presence of a trip condition and no actual [illegible] or trip; i.e., no rod bottom lights. High readings on nuclear instruments reinforce this observation. For this phase, we see little or no dependence among the operators and model the situation as low dependence.

From the earlier discussion and the remarks on page 17-9 of the handbook for "second nature" responses, it seems appropriate to consider the stress level optimum. The basic human error probability for this situation is 0.003. Then for low dependence, the center desk man's human error probability (HEP) is

$$\frac{1 + 19(.003)}{20} = 0.05$$

Since the STA (and SE if he is in the control room) will not respond as quickly and thus has less time to recognize the ATWS condition, we multiply his HEP by 2; i.e., 0.1. If the SE arrives in the control room within 1 minute, we again double his HEP to 0.2. Therefore, the total HEP for failing to discover the ATWS condition is

$$.8[0.003 \times 0.05 \times 0.1 \times 0.1] + .1[0.003 \times 0.05 \times 0.1 \times 0.2]$$

$$+ .1[0.003 \times 0.05 \times 0.1] = 3.0 \times 10^{-6}$$

After acknowledging very broad uncertainty in these results by assuming a lognormal distribution and assigning a range factor of 20, the mean HEP for recognizing the ATWS condition is 1.6×10^{-5}.

The first actions required of the operators, to manually trip the reactor and the turbine, are of a routine or automatic nature. To quote the handbook, "If personnel at a plant indeed have such frequent practice that the tasks in question could be regarded as 'second nature,' the HEPs assigned to the moderately high level of stress will not apply, as the stress level will be closer to optimum." (page 17-9) We expect the manual trip to be attempted immediately, before the real significance of the ATWS condition is appreciated. Nevertheless, because the timing is short, we double the basic HEP for the RO; i.e., 0.006. Then for low dependence, the center desk man's HEP is

$$\frac{1 + 19(.006)}{20} = 0.056$$

As above, we double this to 0.111 for the STA and the SE if he is in the control room and double it again if he arrives within the first minute. Thus, the total HEP for failing to initiate a manual reactor trip is

$$0.8[0.006 \times 0.056 \times 0.111 \times 0.111] + 0.1[0.006 \times 0.056 \times 0.111$$

$$\times 0.222] + 0.1[0.006 \times 0.056 \times 0.111] = 7.83 \times 10^{-6}$$

Assigning a range factor of 20, the mean HEP is 4.11×10^{-5}.

If the reactor still has not tripped, it is apparent to the operators that a very unexpected condition exists. Despite their extensive training for this situation, we believe the operators will feel high stress as they begin to carry out the ATWS emergency procedure. The first step after attempting the manual trips of the reactor and turbine is to drive in the control rods. If this action begins within 1 minute, it should successfully terminate the ensuing pressure rise. Under these conditions, we assign an HEP of 0.25 to the RO. The center desk man may be closely working with the RO, so we consider this as a case of high dependence with an HEP of

$$\frac{1 + 0.25}{2} = 0.63$$

Because the STA and SE will be delayed in responding, probably until the RO's concern is voiced, we consider them moderately dependent

$$\frac{1 + 6(0.25)}{7} = 0.357$$

but double this value because of the time constraint to 0.71. Remember though that the required action is simple. In fact, all the STA really needs to do is say, "why aren't you driving rods?" and the event could be terminated. Finally, if the SE (or STA) is outside the control room, we give him no credit in helping the situation. Thus, the total HEP for failing to drive rods within 1 minute is

$$0.8[0.25 \times 0.63 \times 0.71 \times 0.71] + 0.2[0.25 \times 0.63 \times 0.71]$$

$$= 8.59 \times 10^{-2}$$

If we assign a range factor of 10; i.e., the upper bound is 0.859, then the mean HEP is 0.23 for failing to drive control rods given that automatic and manual trip have failed.

To calculate OP-6, the human factors analysis above is coupled with the hardware analysis of Section 1.5.2 using the fault tree of Figure 1.3.3-9. In the reactor protection system analysis of Section 1.5.2, results are given for automatic reactor trip. If the operator pushes the reactor trip button, all logic circuits are bypassed and the trip coils for the reactor trip breakers are energized (automatic reactor trip only deenergizes the undervoltage coils--no breaker failures-to-trip have been observed when the trip coils are energized). When the motor generator sets are deenergized, all circuitry is bypassed. These results are summarized in the following table.

UNAVAILABILITY FOR REACTOR TRIP

	Automatic Reactor Trip	Manual Reactor Trip	Manually Deenergize and RCCAs Fall
Rods	2.98×10^{-6}	2.98×10^{-6}	2.98×10^{-6}
Doubles	1.70×10^{-4}	2.70×10^{-6}	0
Test and Maintenance	6.20×10^{-6}	$< 1.0 \times 10^{-7}$	0
Common Cause	4.61×10^{-7}	4.61×10^{-7}	0
System	1.78×10^{-4}	6.14×10^{-6}	2.98×10^{-6}

Data for failure of the rods to insert in response to an operator driving rods (failure of CRDCS) can be obtained from a recent NRC report.* We obtain the following distribution:

5th Percentile: 3.14×10^{-5}

50th Percentile: 2.70×10^{-4}

95th Percentile: 2.32×10^{-3}

Mean: 6.35×10^{-4}

*Hubble, W. H., and C. F. Miller, "Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants January 1, 1972 to April 30, 1978," NUREG/CR-1331, EGG-EA-5079, February 1980.

If the reactor has not been tripped and inward rod motion has not begun within 1 minute, and if the pressure is successfully controlled by the relief and safety valves (PR-1, discussed in 1.3.3.11), we next look for reactor shutdown by manually deenergizing power to the rods. Here, we assume high stress (0.25) for the RO in deciding to carry out the action, complete dependence for the center desk man (1.00), high dependence for the

$$\text{STA} \quad \frac{1 + 0.25}{2} = 0.63$$

and moderate dependence for the

$$\text{SE} \quad \frac{1 + 6(0.25)}{7} = 0.36$$

We neglect the SE if he has :ot returned within 5 minutes. Thus, the total HEP for deciding to disable power to the rods is:

$$0.98[0.25 \times 1 \times 0.63 \times 0.36] + 0.02[0.25 \times 1 \times 0.63]$$

$$= 5.87 \times 10^{-2}$$

Assigning a range factor of 10; i.e., the upper bound is 0.59 and the mean is 0.156.

Finally, the procedure specifies that the RO send the equipment operator ("A" man) to trip the breakers locally. Although he is not stressed, we double the basic H⁻ to 0.006 and the total HEP is 0.156 + 0.006 = 0.162.

To calculate OP-7, the human factors analysis above is coupled with the reactor trip hardware analysis results shown earlier, using the fault tree of Figure 1.3.3-10.

1.3.3.10 Safeguards Actuation (Unchanged)

1.3.3.11 ATWS Pressure Relief

ATWS Pressure Relief (PR-1) is used only in the ATWS event tree. The conservative ATWS pressure spike criterion of 3,200 psia (References 1.3-8, 1.3-10) can be exceeded if certain moderator temperature coefficient (MTC) and reactor coolant system relief capability criteria are not met. The fault tree, Figure 1.3.3-11, depicts a discretized 3-state model for MTC. The MTC relief criteria and their conditional frequencies given ATWS have been assessed from the analyses of References 1.3-8 and 1.3-10, the technical specific limitation on power escalation rate and the power history at Zion:

MTC	Frequency				Pressure Relief Criteria
pcm/^{0}F	5%	Median	95%	Mean	
>-5	2.9-6	1.0-4	3.4-3	1.0-3	Pressure will exceed 3,200 psia
-5 >MTC>-7	2.9-5	1.0-3	3.4-2	1.0-2	Three safety valves must open and one PORV must be opened to limit the system pressure to less than 3,200 psia
<-7	9.6-1	9.99-1	9.999-1	9.89-1	Only three safety valves must open to limit system pressure

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

Data for PORVs was given in (1) above. Safety valve failure to open data from Section 1.5.1 is:

Mean: 1.63 x 10^{-6}

Variance: 1.81 x 10^{-12}

Quantifying the fault tree for PR-1 we obtain:

AC buses available	Mean
Power at 149	1.0 x 10^{-3}
No power at 149	1.1 x 10^{-2}

Secure Pressure Relief (PR-1) is quantified using the fault tree of Figure 1.3.3-12. Data on the failure of relief and safety valves to close after passing two phase flow is scanty. Recent EPRI testing seems to support the use of generic data for PORVs:

5th Percentile: 6.7 x 10^{-3}

50th Percentile: 2.0 x 10^{-2}

95th Percentile: 6.0×10^{-2}

Mean: 2.5×10^{-2}

but implies that safety valves may perform much more poorly after passing water than the 2.9×10^{-3} mean generic value for passing steam. We have decided to use a conservative value of 0.1 failures per demand for the safety valves. Then the result for PR-2 is 0.3.

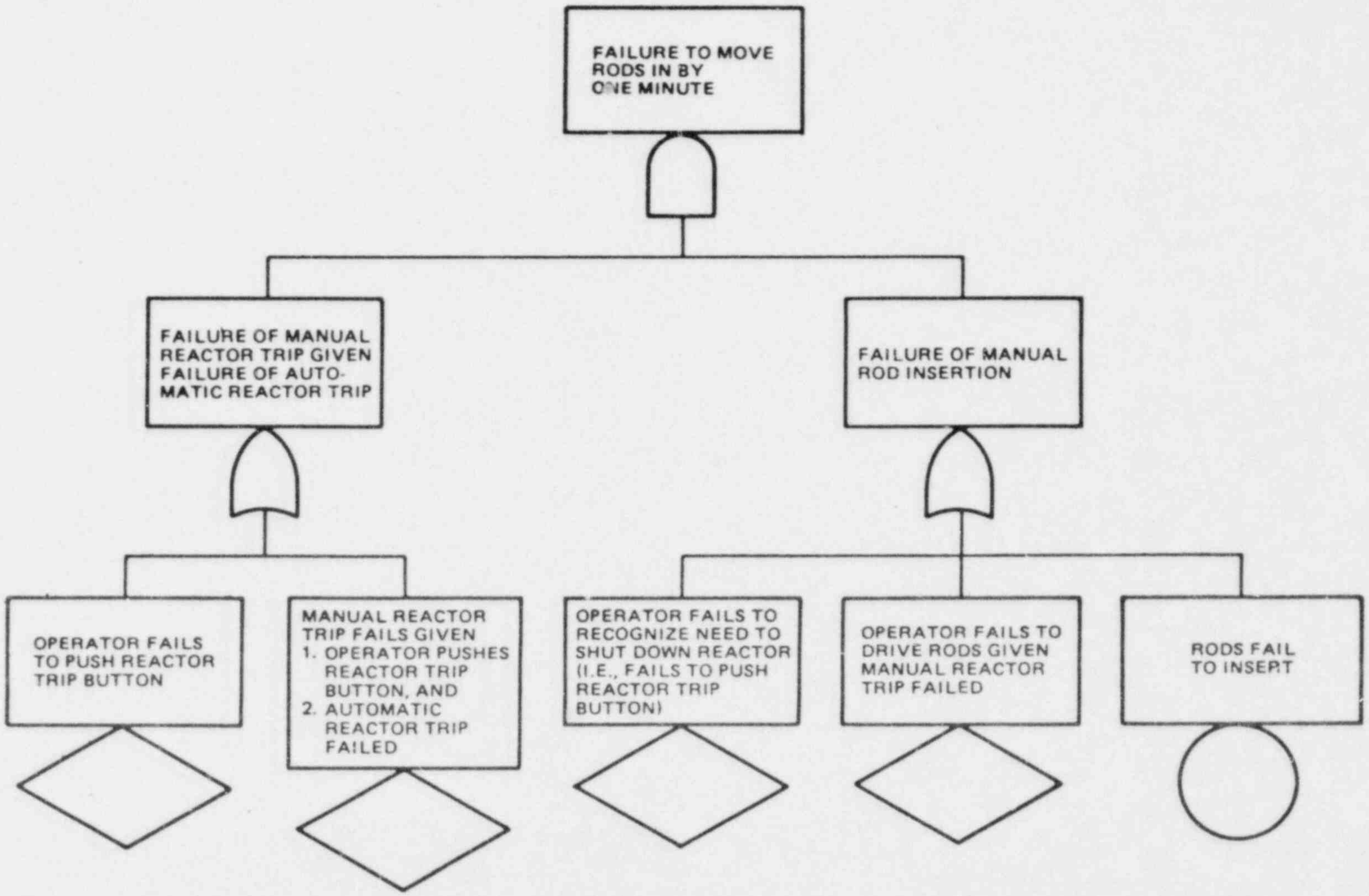

Figure 1.3.3-9. Fault Tree for OP-6, Rods In By One Minute

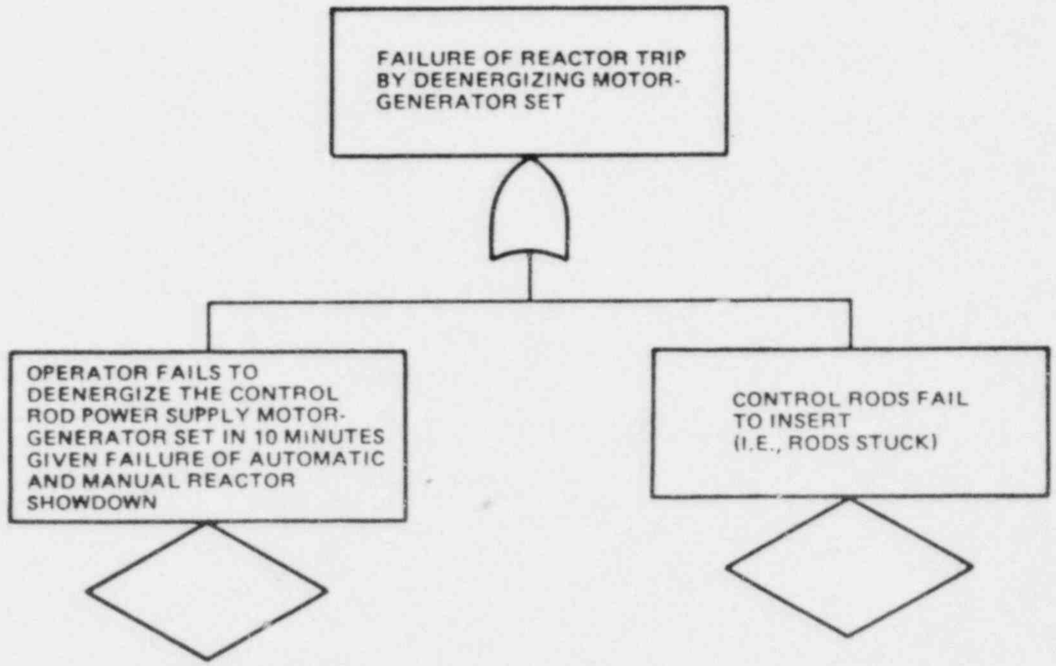

Figure 1.3.3-10. Fault Tree for OP-7, Manually Deenergize and RCCAs Fall

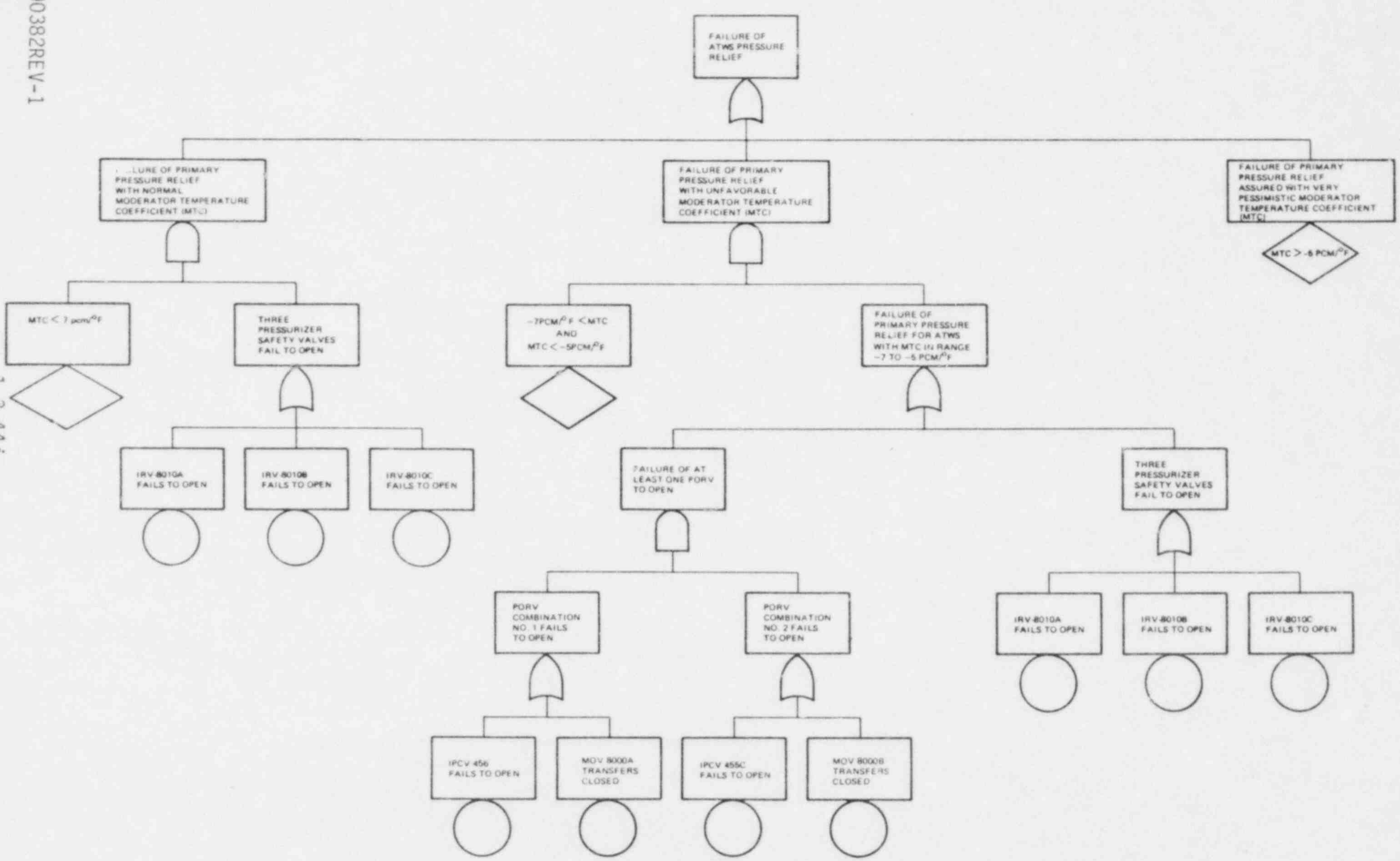

Figure 1.3.3-11. Fault Tree for PR-1, Failure of Primary Pressure Relief for ATWS

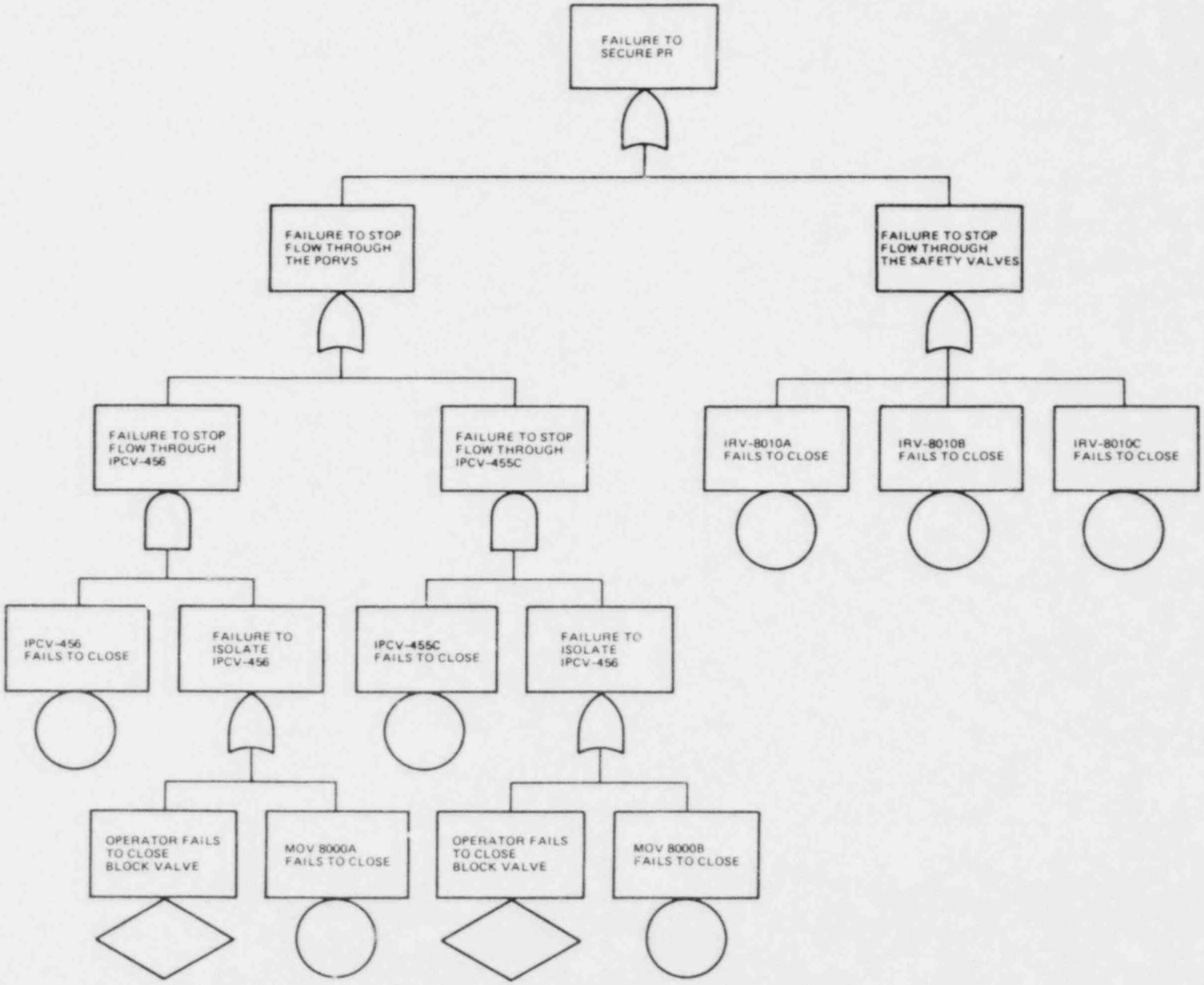

Figure 1.3.3-12. Fault Tree for PR-2, Secure PR

REVISION 1 TO THE ZION PROBABILISTIC SAFETY STUDY
ATWS ANALYSIS

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1.3.4.14 ATWS - Event Tree 14

The ATWS event tree (Figure 1.3.4.14-1, Event Tree 14) applies to all transients for which failure to insert rod cluster control assemblies into the reactor occurs following a reactor trip demand or violation of safety limits. The event tree uses the following symbols to identify system and operator functions:

- ET-14 - ATWS Failure to SCRAM
- PL - Initial Power Level > 80%
- TT-2 - Turbine Trip/MSIV Closure
- L-1 - Auxiliary Feedwater and Secondary Cooling
- L-2 - Auxiliary Feedwater and Secondary Cooling - ATWS
- OP-6 - Rods In by 1 Minute
- PR-1 - ATWS Pressure Relief
- SO - Safety Injection System Operable
- OP-7 - Manually Deenergize and RCCAs Fall
- PR-2 - Secure PR
- OP-2 - Primary Cooling Bleed and Feed
- R-3 - Recirculation Cooling
- CF-2 - Fan Coolers
- CS - Containment Spray
- NA - NAOH Addition

ATWS events encompass a wide spectrum of initiating conditions and transient progression trends (heatup or cooldown). These cannot be adequately modeled in just one event tree unless certain preconditions are conservatively imposed. These preconditions are:

- All excessive cooldown event initiators result in generation of a main feedline isolation. Thus, all of these events ultimately result in a heatup sequence. No credit is taken for safety injection boration or increase in RCS liquid inventory, due to this signal.

- All depressurization events result in generation of a safety injection signal. Generation of this signal results in loss of main feedwater. No credit taken for boration or increase in RCS liquid inventory due to this signal.

- All power excursion transients are beyond the capability of the feedwater system to maintain steam generator level.

By imposing the above assumptions, which are conservative from a system overpressurization and core melt basis, all of the ATWS events essentially reduce down to a loss of feedwater-type ATWS events. Although this basis is conservative, it adequately models the major decisions required to identify core melt and consequential failures. The transient then progresses as described below.

The loss of or inadequate supply of main feedwater events produce a large imbalance in the heat source/sink relationship. The secondary system can no longer remove all of the heat that is generated in the reactor core. This heat buildup in the primary system is indicated by rising reactor coolant system temperature and pressure, and by increasing pressurizer water level, which is due to the insurge of expanding reactor coolant. Water level in the steam generators drops as the remaining water in the secondary system, unreplenished by main feedwater flow, is boiled off. When the steam generator water level falls to the point where the steam generator tubes are exposed and primaryto-secondary system heat transfer is reduced, the reactor coolant temperature and pressure increase is maintained as the pressurizer fills and releases water through the safety and relief valves. The peak pressure attained in the primary system depends upon the ability of the pressurizer safety and relief valves to release the reactor coolant volumetric insurge to the pressurizer. The volumetric relief capacities of these valves are reduced when the pressurizer fills and water is passed instead of steam. During an ATWS, the heat source/sink mismatch causes the reactor coolant temperature and coolant expansion rate to increase and the core reactivity (due to moderator and fuel temperature defects) and power to drop. Reducton of the pressurizer safety and relief valve volumetric relief capacity (due to filling the pressurizer and relieving water) early in the transient when core power is still relatively high, will result in a higher peak reactor coolant system pressure than the peak pressure that would result from reduction of pressurizer relief and safety valve capacity later in the transient, when core power is lower. All other ATWS transients (nonloss of feedwater initiators) will result in a less than or similar power mismatch between heat source/sink. Thus, this case will be considered as the bounding scenario.

There are several mechanisms by which a plant may be shut down following an ATWS event. These include initiation of a safety injection process, an emergency boration process, a normal boration process, a manual reactor trip, or manual insertion of the control rods.

A manual reactor trip signal is processed both directly to the trip breakers and through the protection logic. If this action should fail to deenergize the control rod drive mechanisms, the operator can trip the control rod power supply motor-generator set supply breakers to trip the reactor. If the control rods are tripped, the shutdown banks drop into the core in approximately 2 seconds, inserting more than 4% negative reactivity.

If safety injection is used, borated water is supplied from the boron injection tank through the charging pumps or low head safety injection pumps. Boron concentration in the boron injection tank is 20,000 ppm. At nominal reactor coolant system pressure the safety injection flow is approximately 60 lb/sec. for Zion using the centrifugal charging pumps.

If emergency boration is used, borated water is supplied from the boric acid tank through the boric acid pumps into the normal charging system. Boron concentration in the boric acid tank is approximately 4% boric acid by weight. The charging flow is generally in the range of the normal charging flows.

If a standard boration is used, borated water is supplied through the chemical volume and control system, through the boric acid blender. The source of the borated water is the boric acid tank.

For a manual trip, safety injection signal, or emergency boration signal, the times after operator action required to reach a point where only decay heat is being removed from the core are typically approximately 40 seconds, 3 minutes, and 45 minutes, respectively. At this time, the operator would be able to proceed with normal plant procedures for cooling the reactor coolant system to conditions that permit the use of the residual heat removal system. Sufficient feedwater is available for plant cooldown using either the main feedwater system, the auxiliary feedwater system, or both.

The units have enough capacity to continue auxiliary feed flow at the maximum rate for about 3 hours (much longer at lower feed rates) without a change in the auxiliary feedwater system lineup. Thereafter, an adequate supply is provided by the service water system.

1.3.4.14.1 Initiators

ATWS is not defined as a single event. Rather ATWS is the combination of two events. The first being an initiator which results in plant operation outside of the operating control band; and, second, the failure to insert the rod cluster control assemblies into the reactor following reactor trip demand or violation of safety limits. In this study, the term ATWS takes on a broader definition than that established in WASH-1270 (Reference 1.3-16). All Transients Without Scram are included, not just condition II events (Faults of Moderate Frequency as defined in ANSI-N18.2-1973, Reference 1.3-17), as is traditionally analyzed (see WCAP 8330, References 1.3-8 and 1.3-10). To obtain the frequency of the ATWS initiating events, all of the event trees (Figures 1.3.4.3-1 through 1.3.4.12-1) were used to identify where the above combination of conditions occur. Where this combination occurred (initiating event - no reactor trip), a transfer was made out of the original event tree and into the ATWS event tree. Thus, the entry point (first node) frequency for ATWS is the summation of a large number of transient combinations from the following trees:

- Small LOCA
- Steam Generator Tube Rupture
- Steam Line Break Inside Containment
- Steam Line Break Outside Containment
- Loss of Feedwater Flow
- Full or Partial Closure of One Main Steam Isolation Valve
- Loss of Primary Flow
- Core Power Increase
- Turbine Trip
- Spurious Safety Injection Actuation

Although the systems/functions used to stabilize the plant for an ATWS are similar to all event trees, the timing, definition of success/failure, and types of consequential failures warrant separate treatment.

The conditional split fractions going to ATWS from each of the other event trees, in each electric power state, are given as M^{β} in Part 1 of Tables 1.3.4.14-4 through 1.3.4.14-14.

1.3.4.14.2 Systems and Operator Functions

The top events of the ATWS event tree, events PL through NA above, are described in detail in Sections 1.3.3 and 1.5.2 as to their overall performance and success probabilities. In this section, each of the above systems are described as they specifically relate to the ATWS events.

1. PL - Initial Power Level> 80% - An upward branch (PL>80%) means that the ATWS event can potentially lead to reactor coolant system pressures large enough to induce a LOCA and even disable safety injection. A downward branch (PL<80%) means that large pressure spikes cannot occur for all events with the exception of those scenarios which also include failure of TT-2. For these scenarios, OP-2 or OP-6 will be required; however, the challenge to the system is significantly less than for the scenarios in the upward branch of PL. This conclusion is based on very conservative assumptions with respect to component stress limits. In addition, since a large number of transient initiators are grouped together, transient progression (pressure rises) are substantially overpredicted.

2. TT-2 - Turbine Trip/MSIV Closure - Success of this function involves rapid interruption of steam flow through the turbine via closure of the turbine throttle valve or main steamline isolation valve closure. The purpose of this function is to minimize steam generator water losses during the early phases of the transient, thus providing for a longer time before significant reductions occur in steam generator tube heat transfer. This will reduce pressurizer insurges which would reduce the potential for significant overpressures during the early phases of the transient. Failure to stop turbine steam load is assumed to result in severe overpressurization of the reactor coolant system. Although the transient analysis of WCAP-8330 allows up to 30 seconds before termination of steam flow to the turbine, the current analysis takes no credit for operator action to trip the turbine or close the MSIVs.

3. L-1 - AFWS Actuation and Secondary Cooling. The auxiliary feedwater system success requires the start of one motor driven pump or the turbine driven pump in response to an automatic actuation signal or in response to operator action. The automatic signals to start the auxiliary feedwater pumps for this event tree come from low-low steam generator level.

   Secondary cooling is achieved by removal of heat from the steam generator by the automatic or manual opening of a relief valve(s) associated with a steam generator(s) receiving auxiliary feedwater. Additional heat removal means are safety valves, steam dumps to the main condenser, or blowdown. Only the steam reliefs and safeties are modeled for secondary cooling as there is adequate secondary provided by this means alone.

Heat removal is only required from one steam generator to provide required cooling to the primary system. Given these conditions, the reactor core will be cooled by single-phase or two-phase natural circulation to the steaming steam generator (References 1.3-6, 1.3-14).

4. L-2 - AFWS Actuation and Secondary Cooling, ATWS Pressure Spike Protection - Success of this function involves providing feedwater flow to all steam generators from one turbine or two motor driven auxiliary feedwater pumps. Success may also be achieved by providing the equivalent flow from the main feedwat· · system, however, no flow from this source is modeled. Success of this function also requires that there is an adequate heat transport mechanism from the reactor core to the steam generators. Thus, overpressurizations which result in loss of significant primary inventory or loss of natural circulation will fail this function even with the auxiliary feedwater system functional.

5. OP-6 - Rods in by 1 Minute - Should automatic reactor trip fail, many important failure modes are bypassed when the manual trip button is pressed. Also, unless the rods are physically hung up in the core, they can be driven using the rod control system. Success of OP-6 requires that the operator try a manual trip and if it fails, that he try to drive rods.

6. PR-1 - ATWS Pressure Relief - When PL>80%, reactor coolant system relief is required to prevent exceeding 3,200 psia. Depending on the value of the moderator temperature coefficient (see Section 1.3.3.11), such capability

   - Can be supplied by three pressurizer safety valves
   - Can be supplied by three safety valves and one power-operated relief valve
   - May not be possible.

   The 3,200 psia criterion is a conservative lower bound on the pressure at which the reactor coolant system could fail with loss of safety injection function (References 1.3-8 and 1.3-10). Failure of a relief valve to reclose is not considered at this node (see node PR-2).

7. SO - Safety Injection System Operable - Table C-1 of WCAP-8330 examines the maximum pressures for reactor coolant system components. From that information, it is conservatively estimated that the chance of failing the safety injection system given RCS pressure exceeding 3,200 psia is less than 1/100.

8. OP-7 - Manually Deenergize and RCCAs Fall - Success of this function involves the operator manually tripping the reactor within 10 minutes of transient initiation by opening the control rod power supply motor-generator set supply breakers.

9. PR-2 - Secure PR - When auxiliary feedwater is successful and the reactor is shut down, the transient can be terminated by securing the blowdown. This requires reseating of all three safety valves and reseating of the PORVs or manual closure of their block valves.

10. OP-2 - Primary Cooling Bleed and Feed - Success of this function must satisfy three conditions. First, the operator must initiate safety injection (manual or automatic) within 25 minutes following the initiating event. Simultaneously, the operator must depressurize the reactor coolant system to below the shutoff head of the safety injection system via the pressurizer power-operated relief valves. For the Zion Units, PORV operation at the setpoint (2,350 psia) is sufficient to allow adequate energy transport and liquid makeup (SI shutoff head of 2,600 psia). A detailed discussion of this procedure and the supporting analysis is provided in WCAP-9744 (Reference 1.3-18). Formal procedures to achieve success of this function have been developed in response to NUREG-0578 Section 2.1.9.c requirements. These procedures (Instructions to Mitigate Inadequate Core Cooling) are currently under review and will be implemented pending final approval. Aside from the core cooling aspects of this procedure, during an ATWS operation of the safety injection system, provides a diverse method for tripping the reactor due to boration. Thus, the 2000 ppm borated water from the RWST used for high heat injection will assure adequate shutdown as well as the 20,000 ppm borated water used in emergency boration.

11. R-3 - Recirculation Cooling. When L-1 (or L-2) is failed and short term cooling is provided via bleed and feed, long term cooling must be accomplished using at least one residual heat removal (RHR) pump. Success of R-3 is dependent upon service water, component cooling water, and fan cooler operation; it can be accomplished in two ways. If the plant is completely depressurized to less than 150 psia, low pressure recirculation is possible. If the plant is depressurized to less than 600 psia, either high pressure recirculation or preferred normal RHR cooling is possible. At greater than 600 psia, only high pressure injection can be used. Success depends upon the operator recognizing what action to take and successful operation of RHR and other equipment.

12. CF-2 - Fan Coolers. Containment fan coolers provide long term cooling for the containment atmosphere. They prevent overpressure and can effectively cool recirculation water from the containment sump (Reference 1.3-11). CF is only displayed on the event tree for core melt sequences when the fan coolers are useful in protecting the containment from overpressure failure. For CF-2, the probability that CF fails (less than three out of five operate properly and service water cooling is available) is calculated conditional on the failure of R-3.

    Recall that reasons for R-3 failure include service water failure (this also fails the fan coolers) and failure of both CCW to the RHR heat exchangers and failure of the fan coolers. This conditional probability calculation is described in Section 1.3.3. This complicated calculation could be avoided by restructuring the tree as SW-CF-R rather than R-CF but the latter structure describes the transient sequence progression more clearly.

13. CS - Containment Spray. Containment spray protects the containment from overpressure failure and scrubs the containment atmosphere of radionuclides. These functions are important if core melt occurs because they reduce the chance and severity of releases. Event CS includes successful generation of an automatic or manual actuation signal. Automatic actuation occurs on high-high containment pressure of 23 psig, coincident with an SI signal. One out of three pumps is required for success. Intermittent manual operation can be effective as long as water is available in the RWST. In this event tree, it is very unlikely that sprays will provide any useful function. They are required if bleed and feed cooling is required and R-3 fails. In such a case, it is most likely that the operator will use the remaining RWST water to continue bleed and feed to save the core rather than save it for sprays after melt. Two cases of its use are possible. The RWST is refilled after melt but before containment overpressure and loss of all electric power (therefore, no R-3) and successful intermittent operation of the diesel-driven spray pump.

14. NA - Sodium Hydroxide Addition - Success of this system involved proper lineup of the sodium hydroxide tank with the containment spray system to provide a high pH containment spray for additional scrubbing of radioactive iodine from the containment atmosphere.

1.3.4.14.3 Tree Structure

1. Displayed Dependencies.

a. For PL is greater than 80%:

(1) When TT-2 fails L-2, OP-6, and OP-7 also fail. This is because TT-2 failure is conservatively assumed to result in very large, early system overpressurizations. The result is that auxiliary feedwater is inadequate to prevent steam generator dryout, the pressurizer insurge cannot be controlled via the PORVs or safety valves, manual reactor trip is not available in this time frame and the overpressurization may result in consequential failure of the safety injection system check valves and piping.

(2) When L-2 fails the same consequential failures are assumed to occur as in a., however, the anticipated overpressurization would be less than the previous case. Thus, the probability of such failures would be less. This case leads to core melt.

(3) When SO succeeds, the sequence branches to the small LOCA tree.

(4) When CS fails, NA cannot succeed.

b. For PL is less than 80%:

(1) The ATWS pressure spike is not severe. Therefore, L-1 can provide AFW (L-2 does not branch).

(2) When turbine trip/MSIV closure (TT-2) succeeds, steam flow from the steam generators is blocked and ATWS pressure relief PR-1 is not required.

(3) When TT-2 fails, the steam generators rapidly boil dry and, although the auxiliary feedwater system could supply water to the steam generators for cooling, current operating procedures will very likely cause the operators to secure flow in the auxiliary feedwater system. Thus, when TT and MS fail, L-1 (auxiliary feedwater system) is modeled as failed. Also, PR may now be required to avoid seriously high pressure even from low power level.

(4) When L-1 auxiliary feedwater and secondary cooling succeed and long term shutdown is ensured by OP-7 success, no further branches are necessary and the transient is ended.

(5) When L-1 fails or the steam generators remain unisolated but core cooling is provided via primary bleed and feed, that is, OP-6 succeeds or L-1 succeeds but OP-6 is needed for long term shutdown, eventually the system is placed in long term recirculation cooling. If R-3 succeeds, and the core is protected, no further branches are necessary and the transient is ended.

(6) All remaining cases lead to core melt and containment protection is examined. When CS fails, NA cannot succeed.

2. Systems Interaction. Most dependencies caused by systems interactions have been discussed in the previous section. Others exist, as follows:

a. All active systems depend on electric power. This is handled in the quantification of the event tree by making separate quantification runs of the tree for each state of electric power. The results from each run are then combined based on the likelihood of the plant being in a specific electric power configuration following the tube rupture.

b. Diesel generators require service water (SW) for cooling.

c. Containment Fan Coolers (CF-2) require SW for successful operation.

d. Successful recirculation cooling (R-3) requires either component cooling supply to the RHR heat exchangers or the successful operation of the fan coolers in conjunction with the operation of the RHR pumps. However, fan cooler contribution to core cooling is only applicable when the recirculation sump is utilized for water supply. R-3 requires service water and component cooling water for RHR pump cooling.

3. Degraded Core States. When PL>80% (or when PL<80% and TT-2 fails) all degraded core states are assumed to follow from a rupture of the reactor coolant system (small LOCA). Therefore, they are assigned to the accident sequence types SE and SL (for early and late failures). When PL<80% pressure spikes causing reactor coolant system rupture do not occur. Therefore, all degraded core states branch to the ATWS tree (ET-14) or are assigned to the accident sequence types TE and SL (for early and late failures) in accordance with Table 1.3.4.0-2 since all are transient sequences. Depending on success or failure of fan coolers (CF) and containment spray (CS) the assignments are as follows (see Figure 1.3.4.14-1).

Plant Event Sequence Category	System Failure Responsible For Core Melt	CF*	CS*
SEFC	PL>80% and	1	1
SEF	[TT-2 or L-2 or	1	0
SEC	PR-1 or OP-2]	0	1
SE	PL<80% and SO	0	0
TEFC	PL<80% and OP-2	1	1
TEF		1	0
TEC		0	1
TE		0	0
SLFC	R-3	1	1
SLF		1	0
SLC		0	1
SL		0	0

* 1 = Success
0 = Failure

The ATWS event differs from other core melt sequences in that the reactor is not shut down quickly. For all other events, only decay heat is factored into the core melt sequences and containment mass/energy releases.

The assessment of the impact of this additional energy on the assignment of containment tree categories requires a study of two issues. First, since the ATWS events release a significant amount of steam to the containment via the relief and safety valves during the early phases of the transient, will these releases significantly change the initial conditions for entry to the containment tree? Second, will these higher energies significantly affect containment transient progression (e.g., much sooner core melt, larger hydrogen generation, higher CR heat fluxes in debris bed, etc.)?

The first concern is resolved in WCAP 8330 Appendix D. This document compares (accidental depressurization) mass and energy releases and containment pressures, for the most, limiting ATWS event and a LOCA. The comparison shows that the ATWS releases and pressures are only a small fraction of that expected for a LOCA and, in fact, the total containment pressure rise is <8 psig compared to approximately 40 psig for a LOCA. Thus, the initial conditions upon which the containment event tree would be entered are less than that assumed for a normal small LOCA or transient event.

The second concern can also be addressed by the analysis in WCAP 8330. Although the transients presented show very large energy inputs above decay heat valves, the ability to sustain power levels above decay heat is primarily a function of secondary cooling. For all cases where loss of L-2 occurs, the reactor core would be driven significantly subcritical via the moderator density coefficient and core voiding once steam generator dryout had occurred. An example of this is provided in the station blackout analysis of WCAP 8330 where the core remains significantly subcritical until after auxiliary feedwater is available. Since core melt does not occur for any event except those involving a loss of L-2, the core will shutdown quickly on voids and moderator effects. This would assure that from a degraded core sequence standpoint, the status of the reactor does not affect transient progression.

1.3.4.14.4 Quantification

The ATWS event tree is a continuation of all plant event trees except:

- ET-1 and ET-2, Large and Medium LOCAs - the reactor is initially shut down by voiding in the core region.
- ET-11b, Turbine Trip - Loss of Offsite Power - ATWS is treated explicitly in the tree revised to include recovery of offsite power.
- ET-13, Reactor Trip.

Following failure of reactor trip, the other trees branch to the ATWS tree in a particular state of electric power with split fractions shown as M^{β} in Tables 1.3.4.14-4 through 1.3.4.14-14. The branch point split fraction data for the ATWS tree (conditional on ϕ and electric power state) is developed in Sections 1.3.3 and 1.5.2, and is summarized in Table 1.3.4.14-1.

When mean value data are propagated through the ATWS event tree, the results displayed in Table 1.3.4.14-2 are obtained. We call this 8 x 21 matrix M^{α}. Dominant sequences from the ATWS tree with all AC power available (most likely state) are listed in Table 1.3.4.14-3. Those that are shown to be dominant with respect to risk will be discussed in Section 8 where risk is quantified. The probability of frequency of those sequences will be requantified to properly account for uncertainty.

M_1^{α} shows the split fractions to each plant event sequence category conditional on an ATWS in an event tree for other than a turbine trip initiating event and a specific state of electric power. Likewise, M_2^{α} shows the split fractions to each plant event sequence category conditional on an ATWS in a turbine trip event tree and a specific state of electric power. The results of the ATWS tree quantification, M^{α}, are combined with ATWS results of the applicable plant event trees, M^{β}, in Tables 1.3.4.14-4 through 1.3.4.14-14.

Event PL has been quantified by reviews of the plant trip data from Zion 1 and 2. The approximate power levels for all losses of main feedwater (ϕ_7) and turbine trips (ϕ_{11b}) have been identified. These transients provide the bulk of the input to the ATWS event tree (>90% on a mean frequency basis). Based on that data, PL>80% for 48 out of 99 transients, we have assigned a mean value of 0.5 to PL.

TABLE 1.3.4.14-1

ET-14 - FAILURE TO SCRAM

AC Electric Power at 147, 148, 149

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-14	ATWS Failure to Scram	-	-	-	1.0	
PL	Power Level >80%	-	-	-	5.0-1	1.3.4.14
TT-2	Turbine Trip/MSIV Closure	2.1-5	1.1-4	5.5-4	1.8-4	1.3.3.2
L-1	Auxiliary Feedwater and Secondary Cooling	1.0-6	3.9-6	7.1-6	4.2-6	1.3.3.6
L-2	Auxiliary Feedwater and Secondary Cooling - ATWS	2.9-3	5.7-3	1.1-2	6.2-3	1.3.3.6
OP-6	Rods in by 1 Minute	-	-	-	8.0-3	1.3.3.9(6)
PR-1	ATWS Pressure Relief	-	-	-	1.0-3	1.3.3.11
SO	Safety Injection System Operable	-	-	-	1.0-2	1.3.4.14
OP-7	Manually Deenergize and RCCAs Fall	-	-	-	6.5-1	1.3.3.9(6)
PR-2	Secure PR	-	-	-	2.7-1	1.3.3.11
OP-2	Primary Cooling Bleed and Feed	7.5-5	1.4-3	2.7-2	3.0-3	1.3.3.9(2)
R-3	Recirculation Cooling	9.9-6	1.2-4	1.5-3	3.8-4	1.3.3.7
CF-2	Fan Coolers	5.8-6	4.6-5	3.6-4	1.0-4	1.3.3.7
CS	Containment Spray	6.6-6	7.4-5	6.5-4	2.2-4	1.5.2
NA	Sodium Hydroxide Addition	1.0-4	5.1-4	1.2-3	7.5-4	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

TABLE 1.3.4.14-1 (continued)

ET-14 - FAILURE TO SCRAM

AC Electric Power at 148, 149

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-14	ATWS Failure to Scram	-	-	-	1.0	
PL	Power Level >80%	-	-	-	5.0-1	1.3.4.14
TT-2	Turbine Trip/MSIV Closure	2.1-5	1.1-4	5.5-4	1.8-4	1.3.3.2
L-1	Auxiliary Feedwater and Secondary Cooling	1.0-6	3.9-6	7.1-6	4.2-6	1.3.3.6
L-2	Auxiliary Feedwater and Secondary Cooling - ATWS	2.9-3	5.7-3	1.1-2	6.2-3	1.3.3.6
OP-6	Rods in by 1 Minute	-	-	-	8.0-3	1.3.3.9(6)
PR-1	ATWS Pressure Relief	-	-	-	1.0-3	1.3.3.11
SO	Safety Injection System Operable	-	-	-	1.0-2	1.3.4.14
OP-7	Manually Deenergize and RCCAs Fall	-	-	-	6.5-1	1.3.3.9(6)
PR-2	Secure PR	-	-	-	2.7-1	1.3.3.11
OP-2	Primary Cooling Bleed and Feed	7.5-5	1.4-3	2.7-2	3.0-3	1.3.3.9(2)
R-3	Recirculation Cooling	9.9-6	1.2-4	1.5-3	3.8-4	1.3.3.7
CF-2	Fan Coolers	5.8-6	4.6-5	3.6-4	1.0-4	1.3.3.7
CS	Containment Spray	1.9-4	3.4-3	4.0-2	8.7-5	1.5.2
NA	Sodium Hydroxide Addition	5.8-4	2.0-3	6.9-3	3.0-3	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

TABLE 1.3.4.14-1 (continued)

ET-14 - FAILURE TO SCRAM

AC Electric Power at 147, 149

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-14	ATWS Failure to Scram	-	-	-	1.0	
PL	Power Level >80%	-	-	-	5.0-1	1.3.4.14
TT-2	Turbine Trip/MSIV Closure	2.1-5	1.1-4	5.5-4	1.8-4	1.3.3.2
L-1	Auxiliary Feedwater and Secondary Cooling	1.5-4	3.9-6	7.1-6	4.2-6	1.3.3.6
L-2	Auxiliary Feedwater and Secondary Cooling - ATWS	3.4-2	7.2-2	1.5-1	8.0-2	1.3.3.6
OP-6	Rods in by 1 Minute	-	-	-	8.0-3	1.3.3.11
PR-1	ATWS Pressure Relief	-	-	-	1.0-3	1.3.3.11
SO	Safety Injection System Operable	-	-	-	1.0-2	1.3.4.14
OP-7	Manually Deenergize and RCCAs Fall	-	-	-	6.5-1	1.3.3.9(6)
PR-2	Secure PR	-	-	-	2.7-1	1.3.3.11
OP-2	Primary Cooling Bleed and Feed	7.5-5	1.4-3	2.7-2	3.0-3	1.3.3.9(2)
R-3	Recirculation Cooling	5.0-3	8.2-3	1.3-2	8.6-3	1.3.3.7
CF-2	Fan Coolers	2.2-3	9.5-3	4.0-2	1.4-2	1.3.3.7
CS	Containment Spray	1.9-4	3.4-3	4.0-2	8.7-3	1.5.2
NA	Sodium Hydroxide Addition	5.8-4	2.0-3	6.9-3	3.0-3	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

TABLE 1.3.4.14-1 (continued)

ET-14 - FAILURE TO SCRAM

AC Electric Power at 147, 148

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-14	ATWS Failure to Scram	-	-	-	1.0	
PL	Power Level >80%	-	-	-	5.0-1	1.3.4.14
TT-2	Turbine Trip/MSIV Closure	2.1-5	1.1-4	5.5-4	1.8-4	1.3.3.2
L-1	Auxiliary Feedwater and Secondary Cooling	1.5-4	3.2-4	6.3-4	3.7-4	1.3.3.6
L-2	Auxiliary Feedwater and Secondary Cooling - ATWS	3.4-2	7.2-2	1.5-1	8.0-2	1.3.3.6
OP-6	Rods in by 1 Minute	-	-	-	8.0-3	1.3.3.9(6)
PR-1	ATWS Pressure Relief	-	-	-	1.1-2	1.3.3.11
SO	Safety Injection System Operable	-	-	-	1.0-2	1.3.4.14
OP-7	Manually Deenergize and RCCAs Fall	-	-	-	6.5-1	1.3.3.9(6)
PR-2	Secure PR	-	-	-	2.7-1	1.3.3.11
OP-2	Primary Cooling Bleed and Feed	-	-	-	1.0	1.3.3.9(2)
R-3	Recirculation Cooling	5.0-3	8.2-3	1.3-2	8.6-3	1.3.3.7
CF-2	Fan Coolers	2.2-3	9.5-3	4.0-2	1.4-2	1.3.3.7
CS	Containment Spray	1.9-4	3.4-3	4.0-2	8.7-3	1.5.2
NA	Sodium Hydroxide Addition	5.8-4	2.0-3	6.9-3	3.0-3	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

TABLE 1.3.4.14-1 (continued)

ET-14 - FAILURE TO SCRAM

AC Electric Power at 149

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-14	ATWS Failure to Scram	-	-	-	1.0	
PL	Power Level >80%	-	-	-	5.0-1	1.3.4.14
TT-2	Turbine Trip/MSIV Closure	2.1-5	1.1-4	5.5-4	1.8-4	1.3.3.2
L-1	Auxiliary Feedwater and Secondary Cooling	1.5-4	3.2-4	6.3-4	3.7-4	1.3.3.6
L-2	Auxiliary Feedwater and Secondary Cooling - ATWS	3.4-2	7.2-2	1.5-1	8.0-2	1.3.3.6
OP-6	Rods in by 1 Minute	-	-	-	8.0-3	1.3.3.9(6)
PR-1	ATWS Pressure Relief	-	-	-	1.0-3	1.3.3.11
SO	Safety Injection System Operable	-	-	-	1.0-2	1.3.4.14
OP-7	Manually Deenergize and RCCAs Fall	-	-	-	6.5-1	1.3.3.9(6)
PR-2	Secure PR	-	-	-	2.7-1	1.3.3.11
OP-2	Primary Cooling Bleed and Feed	1.0-4	1.8-3	3.0-2	3.8-3	1.3.3.9(2)
R-3	Recirculation Cooling	1.9-2	3.1-2	5.0-2	3.2-2	1.3.3.7
CF-2	Fan Coolers	3.3-1	7.2-1	1.6	8.0-1	1.3.3.7
CS	Containment Spray	2.0-2	5.6-2	1.7-1	7.0-2	1.5.2
NA	Sodium Hydroxide Addition	-	-	-	1.0	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

TABLE 1.3.4.14-1 (continued)

ET-14 - FAILURE TO SCRAM

AC Electric Power at 148

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-14	ATWS Failure to Scram	-	-	-	1.0	
PL	Power Level >80%	-	-	-	5.0-1	1.3.4.14
TT-2	Turbine Trip/MSIV Closure	2.1-5	1.1-4	5.5-4	1.8-4	1.3.3.2
L-1	Auxiliary Feedwater and Secondary Cooling	1.5-4	3.2-4	6.3-4	3.7-4	1.3.3.6
L-2	Auxiliary Feedwater and Secondary Cooling - ATWS	3.4-2	7.2-2	1.5-1	8.0-2	1.3.3.6
OP-6	Rods in by 1 Minute	-	-	-	8.0-3	1.3.3.9(6)
PR-1	ATWS Pressure Relief	-	-	-	1.1-2	1.3.3.11
SO	Safety Injection System Operable	-	-	-	1.0-2	1.3.4.14
OP-7	Manually Deenergize and RCCAs Fall	1.6-5	4.7-4	1.4-2	6.5-1	1.3.3.9(6)
PR-2	Secure PR	-	-	-	2.7-1	1.3.3.11
OP-2	Primary Cooling Bleed and Feed	-	-	-	1.0	1.3.3.9(2)
R-3	Recirculation Cooling	1.4-2	1.8-2	2.3-2	1.8-2	1.3.3.7
CF-2	Fan Coolers	4.0-1	5.3-1	7.1-1	5.4-1	1.3.3.7
CS	Containment Spray	6.4-3	8.4-3	1.2-2	8.9-3	1.5.2
NA	Sodium Hydroxide Addition	-	-	-	1.0	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

TABLE 1.3.4.14-1 (continued)

ET-14 - FAILURE TO SCRAM

AC Electric Power at 147

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-14	ATWS Failure to Scram	-	-	-	1.0	
PL	Power Level >80%	-	-	-	5.0-1	1.3.4.14
TT-2	Turbine Trip/MSIV Closure	2.1-5	1.1-4	5.5-4	1.8-4	1.3.3.2
L-1	Auxiliary Feedwater and Secondary Cooling	2.8-2	4.4-2	7.0-2	4.9-2	1.3.3.6
L-2	Auxiliary Feedwater and Secondary Cooling - ATWS	3.4-2	7.2-2	1.5-1	8.0-2	1.3.3.6
OP-6	Rods in by 1 Minute	-	-	-	8.0-3	1.3.3.9(6)
PR-1	ATWS Pressure Relief	-	-	-	1.1-2	1.3.3.11
SO	Safety Injection System Operable	-	-	-	1.0-2	1.3.4.14
OP-7	Manually Deenergize and RCCAs Fall	-	-	-	6.5-1	1.3.3.9(6)
PR-2	Secure PR	-	-	-	2.7-1	1.3.3.9(6)
OP-2	Primary Cooling Bleed and Feed	-	-	-	1.0	1.3.3.11
R-3	Recirculation Cooling	-	-	-	1.0	1.3.3.7
CF-2	Fan Coolers	-	-	-	1.0	1.3.3.7
CS	Containment Spray	6.4-3	8.4-3	1.2-2	8.9-3	1.5.2
NA	Sodium Hydroxide Addition	-	-	-	1.0	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

TABLE 1.3.4.14-1 (continued)

ET-14 - FAILURE TO SCRAM

No AC Electric Power

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-14	ATWS Failure to Scram	-	-	-	1.0	
PL	Power Level >80%	-	-	-	5.0-1	1.3.4.14
TT-2	Turbine Trip/MSIV Closure	2.1-5	1.1-4	5.5-4	1.8-4	1.3.3.2
L-1	Auxiliary Feedwater and Secondary Cooling	2.8-2	4.4-2	7.0-2	4.9-2	1.3.3.6
L-2	Auxiliary Feedwater and Secondary Cooling - ATWS	3.4-2	7.2-2	1.5-1	8.0-2	1.3.3.6
OP-6	Rods in by 1 Minute	-	-	-	8.0-3	1.3.3.9(6)
PR-1	ATWS Pressure Relief	-	-	-	1.1-2	1.3.3.11
SO	Safety Injection System Operable	-	-	-	1.0-2	1.3.4.14
OP-7	Manually Deenergize and RCCAs Fall	-	-	-	6.5-1	1.3.3.9(6)
PR-2	Secure PR	-	-	-	2.7-1	1.3.4.11
OP-2	Primary Cooling Bleed and Feed	-	-	-	1.0	1.3.3.9(2)
R-3	Recirculation Cooling	-	-	-	1.0	1.3.3.7
CF-2	Fan Coolers	-	-	-	1.0	1.3.3.7
CS	Containment Spray	-	-	-	1.0	1.5.2
NA	Sodium Hydroxide Addition	-	-	-	1.0	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

TABLE 1.3.4.14-2

ATWS EVENT TREE QUANTIFICATION

1. ATWS Split Fractions Given No Previous Turbine Trip for Specific Electric Power States, M_1^α

PLANT EVENT SEQUENCE CATEGORY

	Power Available at	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
	147, 148, 149	4.11-05	6.64-09	5.00-09	8.01-13	3.62-06	5.79-10	2.85-10	4 56-14	7.80-06	[illegible]-09	7.80-10	1.25-13	0	0	0	0	0	0	0	0	0
	148, 149	4.11-05	3.17-08	5.00-09	3.86-12	3.62-06	2.79-09	3.49-10	2.69-13	7.80-06	6.01-09	7.80-10	6.01-13	0	0	0	0	0	0	0	0	0
	147, 149	4.03-04	3.12-07	5.74-06	4.43-09	2.53-04	1.95-07	2.96-06	2.28-09	8.23-06	6.34-09	1.17-07	9.00-11	0	0	0	0	0	0	0	0	0
$M_1^\alpha(ATWS\|\overline{TT}) =$	147, 148	3.17-03	7.62-07	4.50-05	1.08-08	2.06-04	4.94-08	2.29-06	5.49-10	2.74-03	6.59-07	3.90-05	9.35-09	0	0	0	0	0	0	0	0	0
	149	7.63-05	6.01-06	3.35-04	2.64-05	3.32-05	2.61-06	2.00-03	2.50-05	1.96-06	1.54-07	7.84-06	6.18-07	0	0	0	0	0	0	0	0	0
	148	1.46-03	1.27-05	1.76-03	1.53-05	0	0	1.88-03	1.71-06	1.27-03	1.10-05	1.49-03	1.29-05	0	0	0	0	0	0	0	0	0
	147	0	0	5.12-03	2.82-05	0	0	3.77-12	3.13-04	0	0	2.67-02	2.32-04	0	0	0	0	0	0	0	0	0
	None	0	0	0	4.32-02	0	0	0	0	0	0	0	2.70-02	0	0	0	0	0	0	0	0	0

2. ATWS Split Fractions Given A Previous Turbine Trip for Specific Electric Power States, M_2^α

PLANT EVENT SEQUENCE CATEGORY

	Power Available at	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
	147, 148, 149	3.99-05	6.45-09	4.86-09	7.77-13	3.54-06	5.67-10	2.80-10	4.47-14	7.80-06	1.25-09	7.81-10	1.25-13	0	0	0	0	0	0	0	0	0
	148, 149	3.99-05	3.08-08	4.86-09	3.75-12	3.54-06	2.73-09	3.42-10	2.63-13	7.80-06	6.01-09	7.86-10	6.01-13	0	0	0	0	0	0	0	0	0
$M_2^\alpha(ATWS\|TT) =$	147, 149	4.02-04	3.10-07	5.72-06	4.42-09	2.52-04	1.94-09	2.95-06	2.27-09	8.23-06	6.34-09	1.17-07	9.00-11	0	0	0	0	0	0	0	0	0
	147, 148	3.08-03	7.40-07	4.37-05	1.05-08	2.06-04	4.94-08	2.29-06	5.48-10	2.74-03	6.59-07	3.90-05	9.35-09	0	0	0	0	0	0	0	0	0
	149	7.61-05	5.99-06	3.34-04	2.63-05	3.26-05	2.57-06	1.99-03	2.48-05	1.96-06	1.54-07	7.85-06	6.18-07	0	0	0	0	0	0	0	0	0
	148	1.42-03	1.23-05	1.71-03	1.49-05	0	0	1.88-03	1.71-06	1.27-03	1.10-05	1.49-03	1.29-05	0	0	0	0	0	0	0	0	0
	147	0	0	5.03-03	2.75-05	0	0	3.77-02	3.13-04	0	0	2.67-02	2.32-04	0	0	0	0	0	0	0	0	0
	None	0	0	0	4.31-02	0	0	0	0	0	0	0	2.70-02	0	0	0	0	0	0	0	0	0

NOTES:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior

E - Early melt
L - Late melt

F - Fan coolers are operating
C - Containment sprays are operating

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TABLE 1.3.4.14-3

ATWS EVENT TREE DOMINANT SEQUENCES
WITH ALL AC POWER AVAILABLE

Plant Event Sequence Category	Conditional Split Fraction	Dominant Sequences		
		Sequence No.	Failed Branch Points	Conditional Split Fraction
SEFC	4.11-5	37	L-1, SO	3.10-5
		23	OP-6, OP-7, OP-2	7.73-6
		10	OP-6, PR-2, OP-2	1.12-6
		44	TT-2, SO	8.99-7
SEF	6.64-9	39	L-2, SO, CS	4.96-9
		25	OP-6, OP-7, OP-2, CS	1.24-9
		12	OP-6, PR-2, OP-2, CS	1.80-10
		46	TT-2, SO, CS	1.44-10
SEC	5.00-9	40	L-2, SO, CF-2	3.10-9
		36	L-2, (SLOCA), SA-1	8.67-10
		26	OP-6, OP-7, OP-2, CF-2	7.74-10
		13	OP-6, PR-2, OP-2, CF-2	1.12-10
		47	TT-2, SO, CF-2	8.99-11
SE	8.00-13	42	L-2, SO, CF-2, CS	4.96-13
		36	L-2, (SLOCA), SA-1, CS	1.39-13
		28	OP-6, OP-7, OP-2, CF-2, CS	1.24-13
		15	OP-6, PR-2, OP-2, CF-2, CS	1.80-14
		49	TT-2, SO, CF-2, CS	1.44-14
SLFC	3.62-6	36	L-2, (SLOCA), R-2	1.43-6
		53	PL, OP-6, OP-7, R-3	9.84-7
		17	OP-6, OP-7, R-3	9.77-7
		4	OP-6, PR-2, R-3	1.42-7
SLF	5.79-10	36	L-2, (SLOCA), R-2, CS	2.28-10
		55	PL, OP-6, OP-7, R-3, CS	1.58-10
		19	OP-6, OP-7, R-3, CS	1.56-10
		6	OP-6, PR-2, R-3, CS	2.27-11

TABLE 1.3.4.14-3 (continued)

ATWS EVENT TREE DOMINANT SEQUENCES
WITH ALL AC POWER AVAILABLE

Plant Event Sequence Category	Conditional Split Fraction	Dominant Sequences		
		Sequence No.	Failed Branch Points	Conditional Split Fraction
SLC	2.85-10	56	PL, OP-6, OP-7, R-3, CF-2	9.84-11
		20	OP-6, OP-7, R-3, CF-2	9.77-11
		36	L-2, (SLOCA), SW	6.82-11
		7	OP-6, PR-2, R-3, CF-2	1.42-11
		82	PL, TT-2, R-3, CF-2	3.38-12
SL	4.56-14	58	PL, OP-6, OP-7, R-3, CF-2, CS	1.58-14
		22	OP-6, OP-7, R-3, CF-2, CS	1.56-14
		36	L-2, (SLOCA), SW, CS	1.09-14
		9	OP-6, PR-2, R-3, CF-2, CS	2.27-15
TEFC	7.80-6	59	PL, OP-6, OP-7, OP-2	7.79-6
TEF	1.25-9	61	PL, OP-6, OP-7, OP-2, CS	1.25-9
TEC	7.80-10	62	PL, OP-6, OP-7, OP-2, CF-2	7.79-10
TE	1.25-13	64	PL, OP-6, OP-7, OP-2, CF-2, CS	1.25-13

NOTE: The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior
E - Early melt
L - Late melt
F - Fan coolers are operating
C - Containment sprays are operating

Values are presented in abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

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TABLE 1.3.4.14-4

ATWS CONTRIBUTIONS TO THE SMALL LOCA EVENT TREE

1. Split Fractions for ATWS and AC Bus Availability

	ATWS and 147, 148, 149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
M^{β} =	[1.78-4	3.33-8	4.77-9	4.77-9	4.10-10	4.10-10	4.10-10	3.33-11]

2. Conditional ATWS Contributions

PLANT EVENT SEQUENCE CATEGORY

	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC
$M^{\beta} \cdot M^{\alpha}$ =	[7.42-9	1.21-12	4.13-12	1.51-12	6.54-10	1.06-13	1.68-12	1.41-13	1.42-9	2.33-13	1.20-11

	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
$M^{\beta} \cdot M^{\alpha}$ =	1.03-12	0	0	0	0	0	0	0	0	0]

Notes:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior

E - Early melt
L - Late melt

F - Fan coolers are operating
C - Containment sprays are operating

TABLE 1.3.4.14-5

ATWS CONTRIBUTIONS TO THE STEAM GENERATOR TUBE RUPTURE EVENT TREE

1. Split Fractions for ATWS and AC Bus Availability

	ATWS and 147, 148, 149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
M^{δ} =	[1.02-10	1.97-14	2.78-15	2.78-15	2.39-16	2.39-16	2.39-16	1.92-17]

2. Conditional ATWS Contributions

PLANT EVENT SEQUENCE CATEGORY

	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF
$M^{\delta} \cdot M^{a}$ =	[4.29-15	6.99-19	2.39-18	8.72-19	3.78-16	6.16-20	9.76-19	8.15-20	8.19-16	1.35-19

	TEC	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
$M^{\delta} \cdot M^{a}$ =	6.96-18	5.94-19	0	0	0	0	0	0	0	0	0]

Notes:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior

E - Early melt
L - Late melt

F - Fan coolers are operating
C - Containment sprays are operating

TABLE 1.3.4.14-6

ATWS CONTRIBUTIONS TO THE STEAM BREAK INSIDE CONTAINMENT EVENT TREE

1. Split Fractions for ATWS and AC Bus Availability

	ATWS and 147, 148, 149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
M^{β} =	[1.79-4	3.38-8	4.78-9	4.78-9	4.18-10	4.18-10	4.18-10	3.38-11]

2. Conditional ATWS Contributions

PLANT EVENT SEQUENCE CATEGORY

	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF
$M^{\beta} \cdot M^{\alpha}$ =	[7.42-9	1.21-12	4.13-12	1.51-12	6.54-10	1.07-13	1.68-12	1.41-13	1.42-9	2.33-13

	TEC	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
$M^{\beta} \cdot M^{\alpha}$ =	1.20-11	1.03-12	0	0	0	0	0	0	0	0	0]

Notes:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior

E - Early melt
L - Late melt

F - Fan coolers are operating
C - Containment sprays are operating

TABLE 1.3.4.14-7

ATWS CONTRIBUTIONS TO THE STEAM BREAK OUTSIDE CONTAINMENT EVENT TREE

1. Split Fractions for ATWS and AC Bus Availability

	ATWS and 147, 148, 149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
H^{β} =	[1.79-4	5.77-8	2.39-9	2.39-9	2.79-9	2.79-9	1.99-10	2.39-10]

2. Conditional ATWS Contributions

PLANT EVENT SEQUENCE CATEGORY

	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
$H^{\beta} \cdot H^{\alpha}$ =	[7.41-9	1.25-12	7.79-12	1.02-11	6.53-10	1.12-13	1.05-11	1.40-13	1.41-9	2.57-13	1.01-11	6.40-12	0	0	0	0	0	0	0	0	0]

Notes:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior   E - Early melt   F - Fan coolers are operating
T - Transient behavior   L - Late melt   C - Containment sprays are operating

TABLE 1.3.4.14-8

ATWS CONTRIBUTIONS TO THE LOSS OF MAIN FEEDWATER EVENT TREE

1. Split Fractions for ATWS and AC Bus Availability

	ATWS and 147, 148, 149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
M^{β} =	[1.81-4	9.28-10	2.44-9	2.44-9	2.75-9	2.75-9	2.09-10	2.36-10]

2. Conditional ATWS Contributions

PLANT EVENT SEQUENCE CATEGORY

	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
$M^{\beta} \cdot M^{\alpha}$ =	[7.41-9	1.25-12	7.79-12	1.02-11	6.53-10	1.12-13	1.05-11	1.41-13	1.41-9	2.57-13	1.01-11	6.04-12	0	0	0	0	0	0	0	0	0]

Notes:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior

E - Early melt
L - Late melt

F - Fan coolers are operating
C - Containment sprays are operating

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TABLE 1.3.4.14-9

ATWS CONTRIBUTIONS TO THE LOSS OF RCS FLOW EVENT TREE

1. Split Fractions for ATWS and AC Bus Availability

	ATWS and 147, 148, 149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
M^{β} =	[1.63-4	8.30-10	2.17-9	2.17-9	2.49-9	2.49-9	1.90-10	2.13-10]

2. Conditional ATWS Contributions

PLANT EVENT SEQUENCE CATEGORY

	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
$M^{\beta} \cdot M^{\alpha}$ =	[7.41-9	1.25-12	7.79-12	1.02-11	6.53-10	1.12-13	1.05-11	1.40-13	1.41-9	2.57-13	1.01-11	6.40-12	0	0	0	0	0	0	0	0	0]

Notes:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior

E - Early melt
L - Late melt

F - Fan coolers are operating
C - Containment sprays are operating

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TABLE 1.3.4.14-10

ATWS CONTRIBUTIONS TO THE CORE POWER EXCURSION EVENT TREE

1. Split Fractions for ATWS and AC Bus Availability

	ATWS and 147, 148, 149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
M^{β} =	[2.36-11	7.37-15	3.18-16	3.18-16	3.59-16	3.59-16	2.67-17	3.08-17]

2. Conditional ATWS Contributions

PLANT EVENT SEQUENCE CATEGORY

	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC
$M^{\beta} \cdot M^{\alpha}$ =	[9.64-16	1.63-19	1.01-18	1.33-18	8.49-17	1.46-20	1.37-18	1.82-20	1.84-16	3.34-20	1.31-18

	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
$M^{\beta} \cdot M^{\alpha}$ =	8.32-19	0	0	0	0	0	0	0	0	0]

Notes:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior

E - Early melt
L - Late melt

F - Fan coolers are operating
C - Containment sprays are operating

TABLE 1.3.4.14-11

ATWS CONTRIBUTIONS TO THE TURBINE TRIP EVENT TREE

1. Split Fractions for ATWS and AC Bus Availability

	ATWS and 147, 148, 149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
M^{β} =	[1.80-4	5.85-8	2.54-9	2.54-9	2.74-9	2.74-9	2.15-10	2.34-10]

2. Conditional ATWS Contributions

PLANT EVENT SEQUENCE CATEGORY

	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC
$M^{\beta} \cdot M^{\alpha}$ =	[7.20-9	1.21-12	7.60-12	1.02-11	6.39-10	1.09-13	1.87-11	1.39-13	1.41-9	2.57-13	1.01-11

	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
$M^{\beta} \cdot M^{\alpha}$ =	6.40-12	0	0	0	0	0	0	0	0	0]

Notes:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior

E - Early melt
L - Late melt

F - Fan coolers are operating
C - Containment sprays are operating

TABLE 1.3.4.14-12

ATWS CONTRIBUTIONS TO THE TURBINE TRIP LOSS OF OFFSITE POWER EVENT TREE

1. Split Fractions for ATWS and AC Bus Availability (from Sequence 15)

	ATWS and 147, 148, 149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
M^{B} =	[2.94-6	0	0	0	0	0	0	0]

2. Conditional ATWS Contributions (Including the Direct Contributions from Sequences 52-64)

PLANT EVENT SEQUENCE CATEGORY

	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
M(ATWS) =	[1.17-10	1.90-14	1.43-14	2.28-18	1.04-11	1.67-15	8.23-16	1.31-19	2.29-11	3.68-15	2.30-15	3.68-19	0	0	0	0	0	0	0	0	0]

Notes:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior

E - Early melt
L - Late melt

F - Fan coolers are operating
C - Containment sprays are operating

TABLE 1.3.4.14-13

ATWS CONTRIBUTIONS TO THE TURBINE TRIP LOSS OF SERVICE WATER EVENT TREE

1. Split Fractions for ATWS and AC Bus Availability

	ATWS and 147, 148, 149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
M^{B} =	[1.79-4	0	0	0	0	0	0	5.95-8]

2. Conditional ATWS Contributions

PLANT EVENT SEQUENCE CATEGORY

	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
$M^{B} \cdot M^{a}$ =	[0	0	7.40-9	2.96-9	0	0	1.57-6	2.52-10	0	0	1.40-9	1.85-9	0	0	0	0	0	0	0	0	0]

Notes:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior

E - Early melt
L - Late melt

F - Fan coolers are operating
C - Containment sprays are operating

TABLE 1.3.4.14-14

ATWS CONTRIBUTIONS TO THE SPURIOUS SAFETY INJECTION EVENT TREE

1. Split Fractions for ATWS and AC Bus Availability

	ATWS and 147, 148, 149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
M^{β} =	[1.81-4	5.66-10	4.81-9	4.81-9	4.25-10	4.25-10	4.25-10	3.40-11]

2. Conditional ATWS Contributions

PLANT EVENT SEQUENCE CATEGORY

	SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC
$M^{\beta} \cdot M^{\alpha}$ =	[7.42-9	1.21-12	3.65-12	1.50-12	6.54-10	1.07-13	1.17-12	1.40-13	1.42-9	2.30-13	1.16-11

	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
$M^{\beta} \cdot M^{\alpha}$ =	1.02-12	0	0	0	0	0	0	0	0	0]

Notes:

Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

A - Large LOCA behavior
S - Small LOCA behavior
T - Transient behavior

E - Early melt
L - Late melt

F - Fan coolers are operating
C - Containment sprays are operating

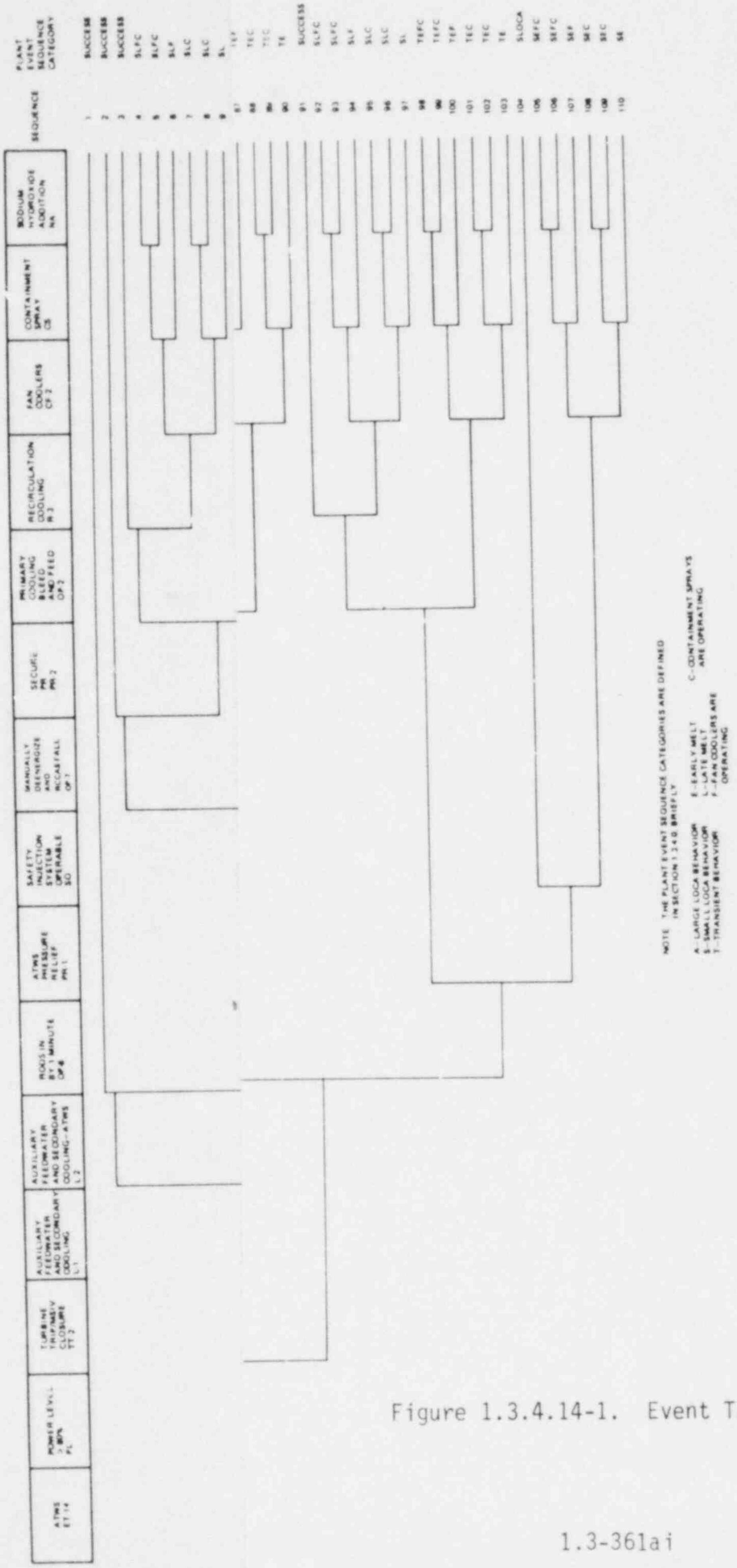

Figure 1.3.4.14-1. Event Tree 14

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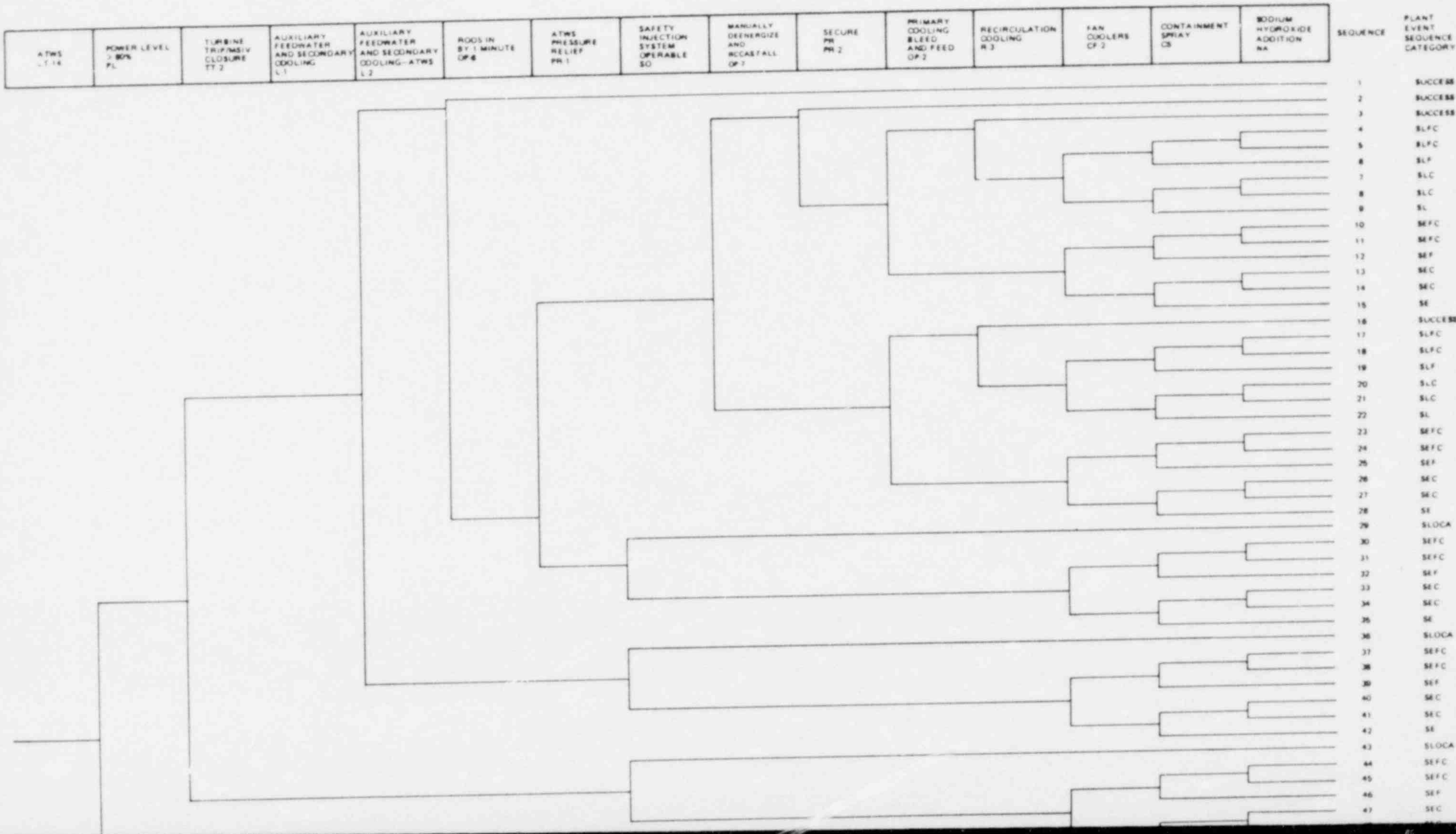

ATWS LT-14
POWER LEVEL > 80% PL
TURBINE TRIP/MSIV CLOSURE TT-2
AUXILIARY FEEDWATER AND SECONDARY COOLING L1
AUXILIARY FEEDWATER AND SECONDARY COOLING—ATWS L2
RODS IN BY 1 MINUTE OP-6
ATWS PRESSURE RELIEF PR-1
SAFETY INJECTION SYSTEM OPERABLE SO
MANUALLY DEENERGIZE AND RECASTALL OP-7
SECURE PR PR-2
PRIMARY COOLING BLEED AND FEED OP-2
RECIRCULATION COOLING R-3
FAN COOLERS CF-2
CONTAINMENT SPRAY CS
SODIUM HYDROXIDE ADDITION NA
SEQUENCE
PLANT EVENT SEQUENCE CATEGORY
1 SUCCESS
2 SUCCESS
3 SUCCESS
4 SLFC
5 SLFC
6 SLF
7 SLC
8 SLC
9 SL
10 SEFC
11 SEFC
12 SEF
13 SEC
14 SEC
15 SE
16 SUCCESS
17 SLFC
18 SLFC
19 SLF
20 SLC
21 SLC
22 SL
23 SEFC
24 SEFC
25 SEF
26 SEC
27 SEC
28 SE
29 SLOCA
30 SEFC
31 SEFC
32 SEF
33 SEC
34 SEC
35 SE
36 SLOCA
37 SEFC
38 SEFC
39 SEF
40 SEC
41 SEC
42 SE
43 SLOCA
44 SEFC
45 SEFC
46 SEF
47 SEC

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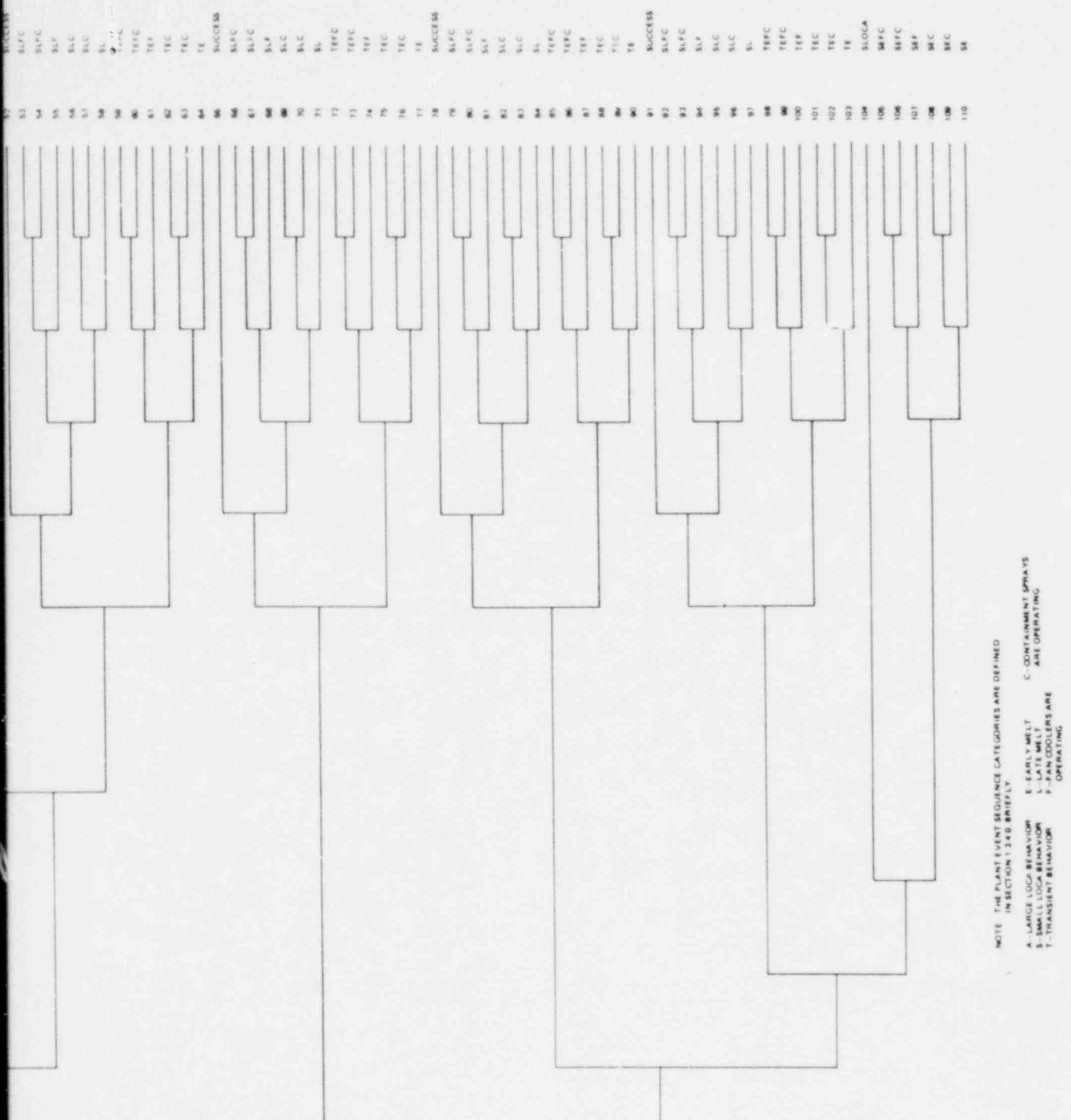


Figure 1.3.4.14-1. Event Tree 14

ATTACHMENT 1

REVISION TO THE ZION PROBABILISTIC SAFETY STUDY -
LOSS OF OFFSITE POWER ANALYSIS

REVISION 1 TO THE ZION PROBABILISTIC SAFETY STUDY
LOSS OF OFFSITE POWER ANALYSIS

The Zion Probabilistic Safety Study model for Turbine Trip - Loss of Offsite Power events has been revised to clarify the analysis and correct nonconservative approximations in the original analysis. There are no observable effects on the overall study results. While it is appropriate to provide the revised analysis, the major cost involved in propagating these results precisely throughout the remainder of the study cannot be justified. Therefore, we recommend that you keep the original pages for reference in tracking results through the study and add these new pages for information.

The Turbine Trip - Loss of Offsite Power Analysis of Section 1.3.4.11.4 is revised on the following pages.

B. Turbine Trip - Loss of Offsite Power

1. Base Case Quantification. The frequency of the Turbine Trip - Loss of Offsite Power Initiating Event, ϕ_{11b} is developed in Section 1.5.1 and shown along with the electric power state split fraction data (conditional on ϕ_{11b}) in Table 1.3.4.11b-1. The other branch point split fraction data for the event tree (conditional on ϕ_{11b} and electric power state) is developed in Sections 1.3.4.0 and 1.5.2 and is summarized in Table 1.3.4.11b-2.

When mean value data is propagated through the Turbine Trip Event Tree (Event Tree 11), the base case results displayed in Table 1.3.4.11b-3 are obtained. Important plant split fractions that are dominated by complete or partial failure of electric power will be corrected below based on the recovery of offsite power bounding technique described in Section 1.3.4.0. Many degraded states of electric power appear important for this event tree because it is initiated by a loss of offsite power. The recovery analysis is conducted in two steps. First, recovery of the less severe degraded electric power states is modeled; that is, all degraded electric power states except power unavailable at buses 148 and 149, and buses 147, 148, and 149. Next, a more realistic model is used for recovery from the two cases involving the combined loss of buses 148 and 149.

Recall from Section 1.3.2 that the chance of recovery offsite power within 1 hour is 0.958. We revise the data tables (1.3.4.11b-2) to account for recovery within 1 hour by using "power at 147, 148, and 149" values for all electric power states for the recoverable functions. The functions that are so adjusted are OP-2 (primary cooling bleed and feed), R-3 (recirculation cooling), CF-2 (fan coolers), CS (containment spray), and NA (sodium hydroxide addition), all of which can be successful if full power is restored within 1 hour. We then weight the original calculation by 0.042, the chance that offsite power is not restored, and weight the revised calculation by 0.958. This calculation is done for all electric power states except the two involving the combined failure of buses 148 and 149.

For the two most severe electric power states, the detailed recovery model from Section 1.3.2 is used. Here, the frequency of losing both buses 148 and 149 at any time over the first 6 hours of operation and then not recovering them for 30 minutes (and 60 minutes) is calculated.

The revised event tree, ET-11b, Turbine Trip Due to Loss of Offsite Power (Buses 148 and 149 Failed) is used for this quantification. In that tree, the following new definitions and data apply:

$$EP\text{-}0 = \begin{pmatrix} \text{The chance that buses 148} \\ \text{and 149 are both failed} \\ \text{at t=0} \end{pmatrix} = 1.85 \times 10^{-3}$$

$$LS = \begin{pmatrix} \text{RCP seal LOCA (i.e., the} \\ \text{chance of LS given EP-0).} \end{pmatrix} = \frac{1.76 \times 10^{-6}}{1.85 \times 10^{-3}} = 9.5 \times 10^{-4}$$

$$EP\text{-}30 = \begin{pmatrix} \text{The chance that buses 148} \\ \text{and 149 are lost for} \\ \text{greater than 30 minutes} \\ \text{given EP-0.} \end{pmatrix} = \frac{6.02 \times 10^{-4}}{1.85 \times 10^{-3}} = 3.2 \times 10^{-1}$$

$$EP\text{-}60 = \begin{pmatrix} \text{The chance that buses 148} \\ \text{and 149 are lost for} \\ \text{greater than 60 minutes} \\ \text{given EP-0 and EP-30.} \end{pmatrix} = \frac{7.49 \times 10^{-5}}{(1.85 \times 10^{-3})(3.2 \times 10^{-1})} = 1.2 \times 10^{-1}$$

If electric power is lost for more than 30 minutes (down branch at EP-30), the auxiliary feedwater system must function without AC power: L-1*. If EP-30 and L-1* fail, success is still possible using bleed and feed cooling. If electric power is lost for less than 60 minutes (EP-60 success), the bleed and feed, recirculation cooling, and containment cooling questions are asked, conditional on AC power being available. If not, core and containment cooling systems are assumed failed.

The results for the complete electric power recovery model are displayed in Table 1.3.4.11b-3. The plant split fractions do not include contributions via the failure of reactor trip, ATWS, event tree. These conditional split fractions without the ATWS tree contributions are labeled M_{11b}. The same table gives the split fractions, M11b, going to the ATWS event tree (ET-14) conditional on electric power state. Later, in Section 1.3.4.14, the conditional plant matrix due to ATWS, M(ATWS), is calculated and in Section 1.4 M(ATWS) and all the M_i' are combined into the complete plant matrix, M.

Dominant sequences are listed in Table 1.3.4.11b-4. Those that are shown to be dominant with respect to risk will be discussed in Section 8 where risk is quantified. The probability of frequency of those sequences will be requantified to properly account for uncertainty.

TABLE 1.3.4.11b-1

TURBINE TRIP - LOSS OF OFFSITE POWER
INITIATING EVENT AND ELECTRIC POWER DATA

Description	5%	Median	95%	Mean
ϕ_{11b}- Turbine Trip-Loss of Offsite Power (Annual Frequency)	8.27-3	3.84-2	1.69-1	5.76-2
AC Power Available at Buses (Given ϕ_{11b}):				
147, 148, 149	3.50-1	3.80-1	4.11-1	3.80-1
148, 149	4.36-1	4.51-1	4.67-1	4.51-1
147, 149	2.08-2	3.44-2	5.71-2	3.61-2
147, 148	2.08-2	3.44-2	5.71-2	3.61-2
149	3.66-2	4.49-2	5.50-2	4.52-2
148	3.66-2	4.49-2	5.50-2	4.52-2
147	1.52-3	2.96-3	3.79-3	3.22-3
None	2.81-3	3.80-3	5.13-3	3.86-3

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

TABLE 1.3.4.11b-2

ET-11b - TURBINE TRIP LOP

AC Electric Power at 147, 148, 149

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-11	Turbine Trip	4.0-3	1.6-2	6.1-2	2.2-2	1.5.2
K-3	Reactor Trip	-	-	-	1.8-6	1.5.2
LS	Reactor Coolant Pump Seal LOCA	-	-	-	0.0	1.3.3
L-1	AFWS Actuation and Secondary Cooling	1.0-6	3.9-6	7.1-6	4.2-6	1.5.2
OP-2	Primary Cooling and Bleed and Feed	3.6-3	5.9-3	9.4-3	6.1-3	1.5.2/1.33
R-3	Recirculation Cooling	9.9-6	1.2-4	1.5-3	3.8-4	1.5.2/1.33
CF-2	Fan Coolers	5.7-8	5.4-7	5.2-6	1.4-6	1.5.2/1.33
CS	Containment Spray	6.6-6	7.4-5	6.5-4	2.2-4	1.5.2/1.33
NA	Sodium Hydroxide Addition	1.0-4	5.1-4	1.2-3	7.5-4	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

TABLE 1.3.4.11b-2 (continued)

ET-11b - TURBINE TRIP LOP

AC Electric Power at 148, 149

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-11	Turbine Trip	4.8-3	1.8-2	7.2-2	2.6-2	1.5.2
K-3	Reactor Trip	-	-	-	1.8-6	1.5.2
LS	Reactor Coolant Pump Seal LOCA	-	-	-	0.0	1.3.3
L-1	AFWS Actuation and Secondary Cooling	1.0-6	3.9-6	7.1-6	4.2-6	1.5.2
OP-2	Primary Cooling and Bleed and Feed	3.6-3	5.9-3	9.4-3	6.1-3	1.5.2/1.3.3
R-3	Recirculation Cooling	9.9-6	1.2-4	1.5-3	3.8-4	1.5.2/1.3.3
CF-2	Fan Coolers	5.7-8	5.4-7	5.2-6	1.4-6	1.5.2/1.3.3
CS	Containment Spray	1.9-4	3.4-3	4.0-2	8.7-3	1.5.2/1.3.3
NA	Sodium Hydroxide Addition	5.8-4	2.0-3	6.9-3	3.0-3	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

TABLE 1.3.4.11b-2 (continued)

ET-11b - TURBINE TRIP LOP

AC Electric Power at 147, 149

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-11	Turbine Trip	3.5-4	1.4-3	6.0-3	2.1-3	1.5.2
K-3	Reactor Trip	-	-	-	1.8-6	1.5.2
LS	Reactor Coolant Pump Seal LOCA	-	-	-	0.0	1.3.3
L-1	AFWS Actuation and Secondary Cooling	1.5-4	3.2-4	6.3-4	3.7-4	1.5.2
OP-2	Primary Cooling and Bleed and Feed	3.6-3	5.9-3	9.4-3	6.1-3	1.5.2/1.3.3
R-3	Recirculation Cooling	5.0-3	8.2-3	1.3-2	8.6-3	1.5.2/1.3.3
CF-2	Fan Coolers	2.2-3	9.5-3	4.0-2	1.4-2	1.5.2/1.3.3
CS	Containment Spray	1.9-4	3.4-3	4.0-2	8.7-3	1.5.2/1.3.3
NA	Sodium Hydroxide Addition	5.8-4	2.0-3	6.9-3	3.0-3	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

TABLE 1.3.4.11b-2 (continued)

ET-11b - TURBINE TRIP LOP

AC Electric Power at 147, 148

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-11	Turbine Trip	3.5-4	1.4-3	6.0-3	2.1-3	1.5.2
K-3	Reactor Trip	-	-	-	1.8-6	1.5.2
LS	Reactor Coolant Pump Seal LOCA	-	-	-	0.0	1.3.3
L-1	AFWS Actuation and Secondary Cooling	1.5-4	3.2-4	6.3-4	3.7-4	1.5.2
OP-2	Primary Cooling and Bleed and Feed	-	-	-	1.0	1.5.2/1.3.3
R-3	Recirculation Cooling	5.0-3	8.2-3	1.3-2	8.6-3	1.5.2/1.3.3
CF-2	Fan Coolers	2.2-3	9.5-3	4.0-2	1.4-2	1.5.2/1.3.3
CS	Containment Spray	1.9-4	3.4-3	4.0-2	8.7-3	1.5.2/1.3.3
NA	Sodium Hydroxide Addition	5.8-4	2.0-3	6.9-3	3.0-3	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

TABLE 1.3.4.11b-2 (continued)

ET-11b - TURBINE TRIP LOP

AC Electric Power at 149

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-11	Turbine Trip	4.6-4	1.8-3	7.2-3	2.6-3	1.5.2
K-3	Reactor Trip	-	-	-	1.8-6	1.5.2
LS	Reactor Coolant Pump Seal LOCA	-	-	-	0.0	1.3.3
L-1	AFWS Actuation and Secondary Cooling	1.5-4	3.2-4	6.3-4	3.7-4	1.5.2
OP-2	Primary Cooling and Bleed and Feed	4.3-3	6.7-3	1.0-2	6.9-3	1.5.2/1.3.3
R-3	Recirculation Cooling	1.9-2	3.1-2	5.0-2	3.2-2	1.5.2/1.3.3
CF-2	Fan Coolers	3.3-1	7.2-1	1.6-0	8.0-1	1.5.2/1.3.3
CS	Containment Spray	2.0-2	5.6-2	1.7-1	7.0-2	1.5.2/1.3.3
NA	Sodium Hydroxide Addition	-	-	-	1.0	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

TABLE 1.3.4.11b-2 (continued)

ET-11b - TURBINE TRIP LOP

AC Electric Power at 148

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-11	Turbine Trip	4.6-4	1.8-3	7.2-3	2.6-3	1.5.2
K-3	Reactor Trip	-	-	-	1.8-6	1.5.2
LS	Reactor Coolant Pump Seal LOCA	-	-	-	0.0	1.3.3
L-1	AFWS Actuation and Secondary Cooling	1.5-4	3.2-4	6.3-4	3.7-4	1.5.2
OP-2	Primary Cooling and Bleed and Feed	-	-	-	1.0	1.5.2/1.3.3
R-3	Recirculation Cooling	1.4-2	1.8-2	2.3-2	1.8-2	1.5.2/1.3.3
CF-2	Fan Coolers	4.0-1	5.3-2	7.1-1	5.4-1	1.5.2/1.3.3
CS	Containment Spray	6.4-3	8.4-3	1.2-2	8.9-3	1.5.2/1.3.3
NA	Sodium Hydroxide Addition	-	-	-	1.0	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

TABLE 1.3.4.11b-2 (continued)

ET-11b - TURBINE TRIP LOP

AC Electric Power at 147

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-11	Turbine Trip	2.6-5	1.2-4	5.3-4	1.8-4	1.5.2
K-3	Reactor Trip	-	-	-	1.8-6	1.5.2
LS	Reactor Coolant Pump Seal LOCA	-	-	-	0.0	1.3.3
L-1	AFWS Actuation and Secondary Cooling	2.8-2	4.4-2	7.0-2	4.9-2	1.5.2
OP-2	Primary Cooling and Bleed and Feed	-	-	-	1.0	1.5.2/1.3.3
R-3	Recirculation Cooling	-	-	-	1.0	1.5.2/1.3.3
CF-2	Fan Coolers	-	-	-	1.0	1.5.2/1.3.3
CS	Containment Spray	6.4-3	8.4-3	1.2-2	8.9-3	1.5.2/1.3.3
NA	Sodium Hydroxide Addition	-	-	-	1.0	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11 x 10^{-5}.

TABLE 1.3.4.11b-2 (continued)

ET-11b - TURBINE TRIP LOP

No AC Electric Power

Code	Description	5th Percentile	Median	95th Percentile	Mean	Source
ET-11	Turbine Trip	6.2-5	2.1-4	6.9-4	2.2-4	1.5.2
K-3	Reactor Trip	-	-	-	1.8-6	1.5.2
LS	Reactor Coolant Pump Seal LOCA	-	-	-	4.6-4	1.3.3
L-1	AFWS Actuation and Secondary Cooling	2.8-2	4.4-2	7.0-2	4.9-2	1.5.2
OP-2	Primary Cooling and Bleed and Feed	-	-	-	1.0	1.5.2/1.3.3
R-3	Recirculation Cooling	-	-	-	1.0	1.5.2/1.3.3
CF-2	Fan Coolers	-	-	-	1.0	1.5.2/1.3.3
CS	Containment Spray	-	-	-	1.0	1.5.2/1.3.3
NA	Sodium Hydroxide Addition	-	-	-	1.0	1.5.2

Note: Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

TABLE 1.3.4.11b-3

TURBINE TRIP - LOSS OF OFFSITE POWER EVENT TREE QUANTIFICATION

1. Conditional Split Fractions Including ATWS [1,2]

PLANT EVENT SEQUENCE CATEGORY [3]

		SEFC	SEF	SEC	SE	SLFC	SLF	SLC	SL	TEFC	TEF	TEC	TE	AEFC	AEF	AEC	AE	ALFC	ALF	ALC	AL	V
M_{11b}	=	[0	0	0	1.75-6	2.12-7	9.25-9	3.96-7	2.97-8	2.08-4	2.83-7	1.61-4	1.84-4	0	0	0	0	0	0	0	0	0]
M'_{11b}	=	[1.23-8	9.40-11	1.50-14	7.10-8	4.22-8	4.49-10	1.67-8	1.25-9	1.39-6	9.18-9	3.88-7	3.50-6									

2. Split Fractions to ATWS Conditional on Electric Power State (from Sequence 15 of ET-11b)

		ATWS and 147,148,149	ATWS and 148, 149	ATWS and 147, 149	ATWS and 147, 148	ATWS and 149	ATWS and 148	ATWS and 147	ATWS and None
M^{β}_{11b}	= [	6.76-7	8.08-7	6.58-8	6.58-8	8.08-8	8.08-8	0	0]

Notes:

1. M_{11b} = Six-Hour Electric Power Bounding Model

   M'_{11b} = Results including Electric Power Recovery

2. Values are presented in an abbreviated scientific notation, e.g., 1.11-5 = 1.11×10^{-5}.

3. The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

   A - Large LOCA behavior
   S - Small LOCA behavior
   T - Transient behavior
   E - Early melt
   L - Late melt
   F - Fan coolers are operating
   C - Containment sprays are operating

TABLE 1.3.4.11b-4

TURBINE TRIP, LOSS OF OFFSITE POWER EVENT TREE DOMINANT SEQUENCES[1]

Plant Event Sequence Category[2]	Conditional Frequency[3]	Dominant Sequences				
		Tree[4]	Sequence	Bus 14__	Failed Branch Points	Conditional Frequency
SEFC	1.23-8	C	37	-	EP-0, LS, OP-2	7.57-9
		C	50	-	EP-0, LS, EP-30, OP-2	3.14-9
		C	63	-	EP-0, K3, EP-30	1.59-9
SEF	9.40-11	C	39	-	EP-0, LS, OP-2, CS	6.65-11
		C	52	-	EP-0, LS, EP-30, OP-2, CS	2.75-11
SEC	1.50-14	C	40	-	EP-0, LS, OP-2, CF	1.06-14
		C	53	-	EP-0, LS, EP-30, OP-2, CF	4.39-15
SE	7.10-8	C	56	-	EP-0, LS, EP-30, EP-60	7.08-8
SLFC	4.22-8	C	17	-	EP-0, EP-30, L-1*, R-3	9.60-9
		B	3	9	L-1, R-3	6.00-9
		B	3	8	L-1, R-3	6.00-9
		B	3	7,9	L-1, R-3	4.89-9
		B	3	7,8	L-1, R-3	4.89-9
		A	3	7,9	L-1, R-3	4.62-9
		A	4	9	L-1, R-3	4.15-9
		B	3	8,9	L-1, R-3	6.81-10
		B	3	7,8,9	L-1, R-3	5.70-10
		C	31	-	EP-0, LS, R-3	4.69-10
		C	44	-	EP-0, LS, EP-30, R-3	1.94-10

NOTES:

1. These sequences are dominant with respect to the revised ET-11b that accounts for electric power recovery.

2. The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

   A - Large LOCA behavior E - Early melt F - Fan coolers are operating
   S - Small LOCA behavior L - Late melt C - Containment sprays are operating
   T - Transient behavior

3. Values are presented in abbreviated scientific notation; e.g., $1.11\text{-}5 = 1.11 \times 10^{-5}$.

4. Tree means: A - ET-11 No recovery, weight 0.042.
   B - ET-11 Recovery of all AC within 60 minutes, weight 0.958.
   C - ET-11b.

TABLE 1.3.4.11b-4 (continued)

TURBINE TRIP, LOSS OF OFFSITE POWER EVENT TREE DOMINANT SEQUENCES[1]

Plant Event Sequence Category[2]	Conditional Frequency[3]	Dominant Sequences: Tree[4]	Sequence	Bus 14__	Failed Branch Points	Conditional Frequency
SLF	4.49-10	A	5	9	L-1, R-3, CS	3.13-10
		C	19	-	EP-0, EP-30, L-1*, R-3, CS	8.43-11
		A	5	7,9	L-1, R-3 CS	4.07-11
		C	33	-	EP-0, LS, R-3, CS	4.12-12
		C	46	-	EP-0, LS, EP-30, R-3, CS	1.71-12
		B	5	9	L-1, R-3, CS	1.32-12
		B	5	8	L-1, R-3, CS	1.32-12
		B	5	7,9	L-1, R-3, CS	1.08-12
		B	5	7,8	L-1, R-3, CS	1.08-12
		B	5	8,9	L-1, R-3, CS	1.50-13
		B	5	7,8,9	L-1, R-3, CS	1.26-13
SLC	1.67-8	A	7	9	L-1, R-3, CF-2	1.66-8
SL	1.25-9	A	8	9	L-1, R-3, CF-2, CS	1.25-9
TEFC	1.39-6	A	9	7,8	L-1	5.41-7
		A	10	8	L-1	3.20-7
		C	23	-	EP-0, EP-30, L-1*, OP-2	1.55-7
		B	9	9	L-1, OP-2	9.69-8
		B	9	8	L-1, OP-2	9.69-8
		B	9	7,9	L-1, OP-2	7.89-8
		B	9	7,8	L-1, OP-2	7.89-8
		B	9	8,9	L-1, OP-2	1.10-8
		B	9	7,8,9	L-1, OP-2	9.21-9

NOTES:
1. These sequences are dominant with respect to the revised ET-11b that accounts for electric power recovery.

2. The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

   A - Large LOCA behavior E - Early melt F - Fan coolers are operating
   S - Small LOCA behavior L - Late melt C - Containment sprays are operating
   T - Transient behavior

3. Values are presented in abbreviated scientific notation; e.g., $1.11\text{-}5 = 1.11 \times 10^{-5}$.

4. Tree means: A - ET-11 No recovery, weight 0.042.
   B - ET-11 Recovery of all AC within 60 minutes, weight 0.958.
   C - ET-11b.

TABLE 1.3.4.11b-4 (continued)

TURBINE TRIP, LOSS OF OFFSITE POWER EVENT TREE DOMINANT SEQUENCES[1]

Plant Event Sequence Category[2]	Conditional Frequency[3]	Dominant Sequences				
		Tree[4]	Sequence	Bus 14__	Failed Branch Points	Conditional Frequency
TEF	9.18-9	A	11	7,8	L-1, CS	4.76-9
		A	11	8	L-1, CS	2.88-9
		C	25	-	EP-0, EP-30, L-1*, OP-2, CS	1.36-9
		B	11	7	L-1, OP-2, CS	2.13-11
		B	11	8	L-1, OP-2, CS	2.13-11
		B	11	7,9	L-1, OP-2, CS	1.74-11
		B	11	7,8	L-1, OP-2, CS	1.74-11
		B	11	8,9	L-1, OP-2, CS	2.42-12
		B	11	7,8,9	L-1, OP-2, CS	2.03-12
TEC	3.88-7	A	13	8	L-1, CF-2	3.76-7
		A	12	7,8	L-1, CF-2	7.68-9
TE	3.50-6	C	44	-	EP-0, EP-30, L-1*, EP-60	3.50-6

NOTES:

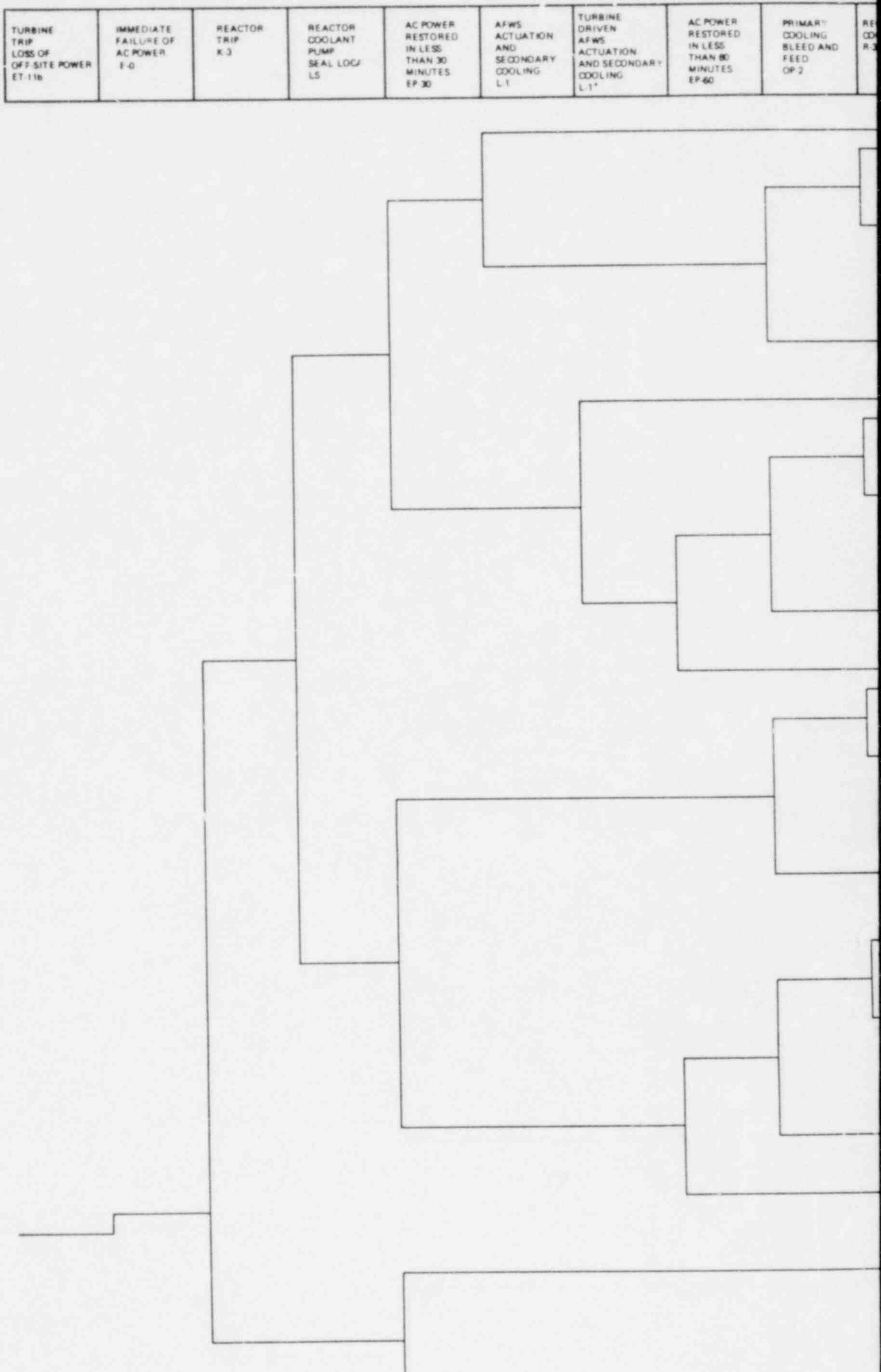
1. These sequences are dominant with respect to the revised ET-11b that accounts for electric power recovery.

2. The Plant Event Sequence Categories are defined in Section 1.3.4.0, briefly:

   A - Large LOCA behavior E - Early melt F - Fan coolers are operating
   S - Small LOCA behavior L - Late melt C - Containment sprays are operating
   T - Transient behavior

3. Values are presented in abbreviated scientific notation; e.g., $1.11\text{-}5 = 1.11 \times 10^{-5}$.

4. Tree means: A - ET-11 No recovery, weight 0.042.
   B - ET-11 Recovery of all AC within 60 minutes, weight 0.958.
   C - ET-11b.



NOTE: THE PLANT EVENT SEQUENCE CATEGORIES ARE DEFINED IN SECTION 1.3.4.0. BRIEFLY:

A – LARGE LOCA BEHAVIOR
S – SMALL LOCA BEHAVIOR
T – TRANSIENT BEHAVIOR

E – EARLY MELT
L – LATE MELT
F – FAN COOLERS ARE OPERATING

C – CONTAINMENT SPRAYS ARE OPERATING

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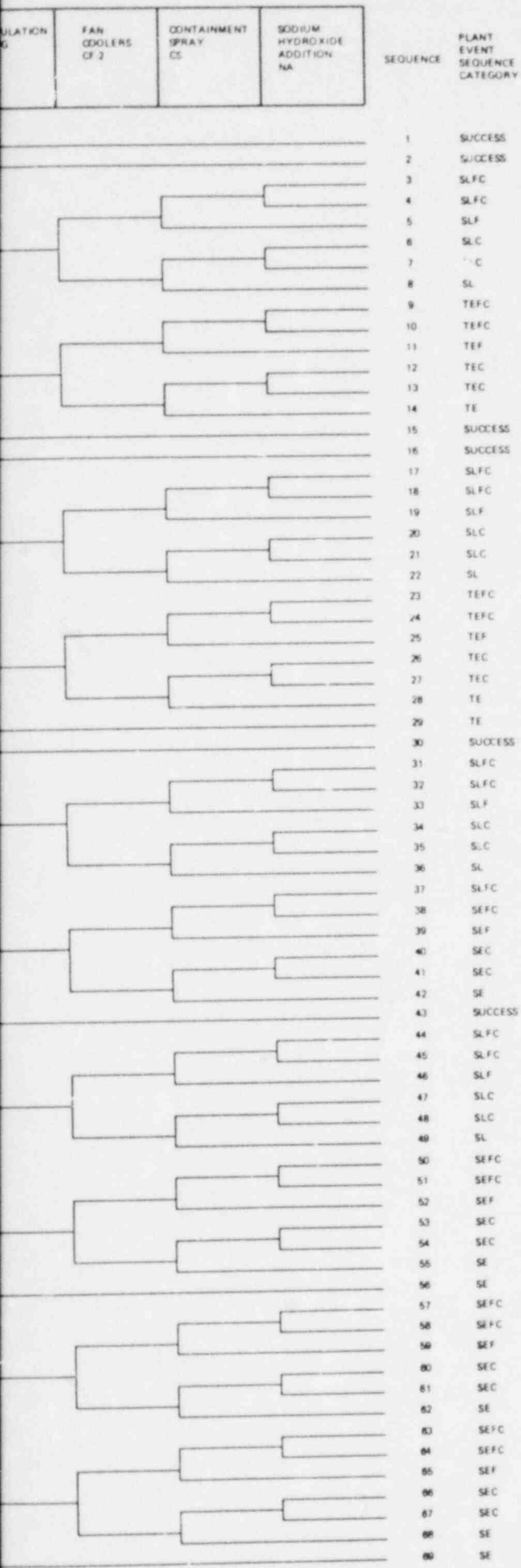

Figure 1.3.4.11-2. Event Tree 11b, Turbine Trip Due to Loss of Offsite Power (Buses 148 and 149 Failed)