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Department of Nuclear Energy

January 15, 1982

Dr. Ashok Thadani, Chief
Reliability and Risk Assessment Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Ashok:

SUBJECT: BNL Review of the Zion Probabilistic Safety Study

This letter provides our review comments on the Zion Probabilistic Safety Study (ZPSS) which was submitted to you by Commonwealth Edison. This review was conducted by Dr. Arthur J. Buslik, Dr. Ioannis A. Papazoglou, and myself. The review process benefitted from many helpful discussions with Dr. W.T. Pratt of BNL who is reviewing, under contract with NRR/DSI/RSB, those portions of the ZPSS which contain the analysis of physical phenomena.

Because our review was a short term, limited-manpower effort, we could not make a final evaluation on the soundness of the ZPSS. However, we have been able to develop preliminary impressions, identify points for further examination (by others) and provide specific comments in selected areas (as per mutual agreement between BNL and RRAB).

The attached memorandum (A.J. Buslik to R.A. Bari, 1/15/82) identifies several specific issues of concern. As per agreement with Scott Newberry, additional comments will be transmitted to you in one week, provided that they become available from I.A. Papazoglou.

The following general comments are provided below on the topics of scrutability, subjectivity, and comparison with WASH-1400.

Scrutability

Within particular topic areas, the report is reasonably understandable. In other areas there is difficulty in understanding the report. This is due in part to a lack of documentation and in part to the appearance of arbitrary assumptions. It is not clear that all of the information presented in the report is actually utilized. Furthermore, the overall integration of information is not transparent. The new notation for plant damage states is misleading (see attached memorandum) and is inconvenient and unnecessarily confusing when referenced against WASH-1400 accident sequences.

The matrix notation is helpful but its significance is overplayed and its description is an unnecessary distraction in the main body of the report.

Subjectivity

[redacted] If [redacted]thermore, additional contributors may appear.

The containment analysis appears to be the largest contributor to the low risk calculated in the ZPSS. It also appears to be the area with the greatest degree of subjectivity. Much of the physical analysis is treated in a manner which lacks justification. The final "containment matrix" is highly suspect and is the subject of examination in the BNL/RSB program. Specific issues and concerns have been brought to the attention of NRR through Dr. J.F. Meyer of RSB.

Comparison with WASH-1400

On technical grounds, it is difficult to understand the point of comparing the final results of WASH-1400 with those for ZPSS.

The methodologies, data, and assumptions used in the two studies are sufficiently distinct that relative information or conclusions drawn on plant or site risk cannot be obtained until all differences in methodologies, data, and assumptions are specifically identified and enumerated.

We hope that these comments and the attached memorandum are useful to you. If you have any questions on any of this material, please do not hesitate to contact me.

Warm regards,

Bob

Robert A. Bari, Head
Engineering and Risk
Assessment Division

RAB:sd
attachment
cc.: A.J. Buslik
R.E. Hall
J. Hickman, Sandia
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W.T. Pratt

BROOKHAVEN NATIONAL LABORATORY

MEMORANDUM

DATE: January 18, 1982

TO: R. A. Bari

FROM: A. J. Buslik AJB

SUBJECT: BNL Peer Review of the Zion Probabilistic Safety Study

1.0 INTRODUCTION

This report documents the results of our peer review of the Zion Probabilistic Safety Study, Reference 1. Section 2 considers internal events, and Section 3 external events. Section 4 is a discussion and summary.

2.0 INTERNAL EVENTS

2.1 Completeness of the Analysis, System Interactions, System Success/Failure Criteria

In this section, comments on the completeness of the analysis will be made. System success criteria which require further study will be noted. System interactions which do not appear to be treated properly will be noted. Two sequences for which more detailed analyses were made will be discussed separately, in sections 2.2 and 2.3. These are accident sequences initiated by loss of offsite power, and small loss of coolant accidents followed by failure of emergency coolant recirculation.

The plant state (or core damage state) used in the analysis of Reference 1 tells whether the sequence type is a large loss of coolant accident (identifier A), a small loss of coolant accident (identifier S), a transient (identifier T), or an interfacing LOCA (identifier V), whether the core melt is early (E) or late (L), whether the containment sprays are operating (C), or not (blank space), and whether the containment fan coolers are operating (F) or not (blank space). However, the containment spray identifier (C) refers only to the operation of the containment sprays in the emergency coolant injection phase and not in the recirculation phase. An accident in which a small Loss of Coolant Accident (LOCA) occurred, followed by failure of emergency coolant recirculation, failure of the containment recirculation spray, and failure of the containment fan coolers, would have the identifier SLC, if the containment spray were to operate in the injection phase. This is the same core damge state identifier that would be used if the containment recirculation spray were operating, yet the containment failure mode and radioactive release might be very different for the two sequences.

In both transient-initiated sequences and small loss of coolant accidents the analysis of Reference 1 assumes that "feed and bleed", where primary coolant is injected into the reactor coolant system by the high pressure injection system, and released through the pressurizer relief valves, is adequate for successful decay heat removal, when the auxiliary feedwater system is unavailable. No analysis is given to support this assumption.

Although loss of component cooling water is considered as an accident initiator, there are potential systems interactions which are not discussed. If component cooling water is lost to the reactor coolant pump seals then reactor coolant pump seal injection flow must be maintained by the charging pumps to prevent reactor coolant pump seal failure. But it would appear that the changing pumps require component cooling water for cooling. If the charging pumps fail, then one has a small LOCA through the failed reactor coolant pump seals. The high head safety injection pumps can be used to take water from the refueling water storage tank (RWST), and inject it into the primary coolant system. In general, these pumps require component cooling water. It is possible that when they are pumping the relatively cool water from the RWST that they do not require component cooling water. But ultimately one has to go to emergency coolant recirculation. The point is that all this must be analyzed, and it does not appear to be discussed in the report. It is not at all clear how the frequency of core melt due to the loss of component cooling water initiator was arrived at.

A sequence which was important in the German Risk Study (See Reference 2, EPRI-NP-1804-SR) was one which involved loss of offsite power, reactor trip, lifting of the pressurizer relief valves, failure to close of the pressurizer relief valves, and failure of the diesel generators. This results in a small LOCA through the pressurizer relief valves, with failure of the High Pressure Injection System (HPIS), and leads to core melt. An incident which involved loss of offsite power, reactor trip, lifting of pressurizer relief valves, and failure to close of the pressurizer relief valves, occurred at the Beznau reactor in Switzerland, on August 20, 1974 (Reference 3, ORNL/NSIC-176, p. 58). This is a Westinghouse-designed reactor. The pressurizer relief valves lifted because of failure of a turbine by-pass valve. A very rough estimate of the frequency of this sequence is obtained as follows:

Event	Frequency
Loss of Offsite Power	.08/yr
Pressurizer Relief or Safety Valve Lifts	.1
Pressurizer Valve Fails to Reclose	.02
Diesel Generators fail to Energize any of Buses 147, 148, or 149	$1x10^{-3}$

The product of the frequencies of these events is $1.6x10^{-7}$/yr, which is comparable to the value obtained for other loss-of-offsite power sequences that were considered. This estimate of the frequency of this sequence is very rough. In the German Risk Study (See p.5-16 of EPRI NP-1804-SR), the pressurizer relief valves were assumed to lift on power failure. The frequency of pressurizer relief or safety valve lifting, given loss of offsite power, depends in part on whether the plant is operated with the block valves in series with the pressurizer relief valves in a closed position. Presumably the lifting of a pressurizer safety valve, with its higher set point, would occur more rarely. From a telephone conversation with George Klopp of Commonwealth Edison, it was learned that the Zion plants are, in fact, operated with the pressurizer block valves closed.

If the pressurizer relief valves are operated in this mode, it raises a question concerning the ATWS sequences. According to page 1.3-333 of the Zion Probabilistic Safety Study, ATWS pressure relief may require operation of one power operated relief valve on the pressurizer, in addition to the three safety valves on the pressurizer. This will occur if the moderator temperature coefficient is not sufficiently negative. On page 1.3-44 of the Zion Probabilistic Safety Study this condition is assumed to occur about 10% of the time.

The operator must open the block valve in less than 10 minutes. A mean frequency of $4x10^{-3}$ is assigned to the event that the operator fails to open the block valve in this 10 minute or less time period. Considering the high stress conditions under which the operator is acting, it would appear that this frequency could be significantly higher. If the human error frequency were P, instead of .004, then the frequency of core melt due to ATWS would be $6.7x10^{-6}$/yr + $8x10^{-5}$/yr x (P - .004). This can be seen by considering the following sequence of events, which leads to core melt (state SEFC):

EVENT	FREQUENCY
1. Either loss of main feedwater <u>or</u> turbine trip	(5.2+3.7)/yr = 8.9/yr*
2. Power level greater than 8%	.5**
3. Failure to trip	$1.8x10^{-4}$***
4. Mod. Temp. Coeff. between -7 and -5 pcm	.1****
5. Operator Fails to open block valve	P

* See Table 1.5.1-50 of Reference 1.

** See p. 1.3-340 of Reference 1.

*** See Tables 1.3.4.7-2 and 1.3.4.11a-2 of Reference 1.

**** See p. 1.3-44 of Reference 1.

The frequency of this sequence is $8x10^{-5}$P/yr. Since Reference 1 obtains a core melt frequency due to ATWS of $6.7x10^{-6}$/yr (see Table II.2.1 in Volume 1 of Reference 1), and since P = .004 was used in Reference 1, one obtains $6.7x10^{-6}$/yr + $8.0x10^{-5}$ (P - .004) for the frequency of core melt due to ATWS.

The estimate of .004 for the human error of failing to open the block valve was based on analogy to the human error in switchover to low pressure recirculation. However, there is at least 30 minutes after a large LOCA before switchover to low pressure recirculation is required, while according to p. 1.3-44 of Reference 1 the operator in an ATWS has "less than 10 minutes" to open the pressurizer block valve. The question is, how much less? Generic studies indicate that the peak pressure in an ATWS is reached in about 2 minutes (Ashok Thadani, private communication). If the operator must respond within two minutes, then, according to the Handbook of Human Reliability Analysis (see p. 17-20 of Reference 9), the human error frequency is .95, and P = 95 may be a good estimate, although the skill and training of the operators must be taken into account. Two minutes or so after an accident, credit is given for only one operator, according to the Handbook (see p. 17-24 of Reference 9). If 10 minutes were available, then the probability that the first operator fails to open the block valve is about .8, according to the Handbook (p. 17-20 of Reference 9). Credit is given here for the shift supervisor also being present. Assuming complete independence of the errors of the two people, one obtains .64 for the probability of failing to open the block valve. For P = .95 the core melt frequency due to ATWS is 8.2×10^{-5}/yr, and for P = .64 the core melt frequency due to ATWS is 5.8×10^{-5}/yr. These are important contributors to core melt frequency. A careful, detailed human error analysis of this sequence must be made, taking into account the skills and training of the operators, and the precise times available for their action. The Handbook (Reference 9, p. 17-25) says that, in some cases, in coping with large LOCA, the human error frequencies may be much lower than those given on p. 17-20 of the Handbook (Reference 9).

When looking for completeness, one can look at the relatively minor incidents that occurred which might have led to a serious accident if other things went wrong. Even if the conditional frequency of a serious accident, given the event sequence that acutally occurred, is relatively low, one would like this possible serious accident sequence to be included in the logical framework of the risk study. The reason is that there may be a very large number of such accident sequences, so that the frequency of the aggregate of these sequences may be appreciable.

One incident that occurred which could have led to something serious, and which does not appear to be included in the logical framework of the Zion Probabilistic Safety Study is a human-error initiated incident which occurred at Zion 2 on July 12, 1977. This incident resulted in the water level in the reactor being drawn down to the point where the pressurizer heaters were uncovered. It occurred when unit 2 was in hot standby. A reactor protection logic system test was to be done. Because of a series of administrative and operator errors, a number of instruments were jumpered so that dummy signals were present, when in fact these instruments were not supposed to be jumpered. These instruments displayed values which were not related to the actual values of the parameters they were supposed to measure, and the control systems responded to the dummy signals, not to the true values of the parameters.

Certain core-melt accident sequences which could occur during cold shutdown do not appear to be adequately addressed in the 7ion Probabilistic Safety Study. These are accident sequences in which the steam generators are unavailable for decay heat removal. In preparation for refueling, with the reactor vessel head bolts loosened, the option of returning to hot shutdown conditions and removing decay heat through the steam generators is not available. The steam generators may also be unavailable because of eddy current testing of the steam generator tubes. There have been at least four incidents during shutdown when residual heat removal pumps have become airbound during cold shutdown and the steam generators have been unavailable because the steam generator manway covers were removed in preparation for eddy-current testing. These events were (1) an event at Beaver Valley on September 4, 1978, (2) an event at Ginna on May 3, 1972, (3) an event on April 18, 1980 at Davis Besse, and (4) an event on April 19, 1980 at David Besse. When eddy-current testing of steam generator tubes is done, the water level in the reactor is lower than usual, and airbinding of the residual heat removal pumps becomes more likely. Charging pumps could in general be used to maintain a water level above the core, but maintenance on charging pumps is frequently done during periods of cold shutdown.

Although accidents occurring during cold shutdown may not be adequately addressed in the study, there do not appear to be any fundamental weaknesses in the methodology which prevent them from being handled as well as the other accident sequences considered. The human-error initiated accident sequences are more difficult, however, as the incident at Zion-2 referred to earlier, in which a variety of instruments were jumpered, demonstrates.

BNL was supplied with a list of initiating events by the Reliability and Risk Assessment Branch. We give here some comments concerning the treatment of these events in the Zion Probabilistic Safety Study.

For event 1 on this list (see Table 2.1), Station Blackout, the reactor coolant pump seal failure event was considered, but there are deficiences in the analysis, as will be discussed in Section 2.2. The loss of d.c. after a finite time was apparently (insofar as the authors could ascertain) not explicitly considered.

Event 4, Reactor coolant pump trip for a small LOCA is apparently not included, but its effect on the event sequence is unclear.

Event 6, a multiple instrument tube LOCA below core level is a small loss of coolant accident, and should be included in the frequency for small loss of coolant accidents.

Event 7, overcooling events leading to pressurized thermal shock, and event 8, overpressurization during cold shutdown, are not included in the Zion Probabilistic Safety Study, as being possible causes of reactor vessel rupture. (The related problem of radiation embrittlement changing the nil ductility transition temperature of the reactor vessel is also not considered.) The frequency of reactor vessel rupture is taken from Wash-1400 -see p. 1.3-71 of the Zion Probabilistic Safety Study.

For the large LOCA, event 9, reactor coolant pump missiles are not considered; turbine missiles as an accident initiator are considered briefly on p. 7.8-1 of the Zion Probabilistic Safety Study.

Event 12, a stuck open pressurizer safety relief valve is included as an accident initiator as part of the small LOCA frequency. This means that any difference in, say, the frequency of operator error for a pressurizer safety valve lifting and for other possible small LOCA's would not be considered. However, considering the operator training after the Three Mile Island Accident it is not very likely that the operator would turn off the High Pressure Injection System because of high pressurizer water level if the pressurizer pressure is low.

We were not able to ascertain, in the time available to us, whether containment isolation, event 15 of Table 2.1, was included or not. The signal to provide containment isolation is provided by the Engineered Safeguards Actuation System, discussed on p. 1.5-312. of Reference 1. There is also a discussion of containment isolation on p. 1.1-9 of Reference 1. However, the event trees looked at do not appear to address whether or not containment isolation took place. Moreover, the plant state description (i.e., core damage state) is not sufficiently detailed to say whether containment isolation took place. For example, SEFC says nothing about containment isolation. On p. 2.4-3 of Reference 1 it is indicated that containment isolation failure is sufficiently unlikely so that its contribution to risk is negligible; however, no analysis seems to be given in the report.

Event 19, loss of ventilation in the auxiliary building, is not considered as an accident initiator, but it is not clear that this is significant. Certain accident sequences could result in loss of ventilation because of loss of AC power; this was apparently not addressed directly but may not be important.

Event 21, reactor coolant pump seal failure, should be included as part of the small LOCA initiating event frequency. However, from Table 1.5.1-47 of Reference 1, and the comment there that there were no small LOCA's except for pressurizer relief or safety valve opening, it is evident that no reactor coolant pump seal loss of coolant accidents were included in the data base. There have been at least two reactor coolant pump seal failures - at H.B.

Robinson, unit 1, on May 1, 1975, where the containment was flooded to a depth of 12.5 inches, with a total leak of 132,500 gallons, and at Arkansas unit 1, on May 10, 1980 where the leak rate reached a maximum of 90 gallons per minute.

Table 2.1 gives, in addition to the initiating events supplied to us by the Risk and Reliability Analysis Branch, an indication as to whether the event is included in the Zion Probabilistic Safety Study, and some brief comments, or else reference to the text of this section, or other sections of this report for more details.

2.2 Loss of Offsite Power Initiator

2.2.1 Electric Power Recovery Models

The offsite power recovery model used in the Zion Probabilistic Safety Study is optimistic compared to generic data on the recovery of offsite power. There is no plant-specific data on the recovery of offsite power after a total loss of offsite power, since this has never happened at the Zion plant. According to the model used in the Zion Probabilistic Safety Study, the frequency that offsite power is not recovered within 30 minutes is .28, and the frequency that offsite power is not recovered in 60 minutes is .03. To obtain, from generic data, an estimate of the frequency distribution for the time to restoration of offsite power, we made use of a report by Raymond F. Scholl, Jr., of the NRC [Loss of Offsite Power - Survey Status Report, Revision 3]. For 39 instances of total loss of offsite power in which the time to partial restoration of offsite power was given, the frequency with which offsite power was not recovered within 30 minutes was .41, (as contrasted to the Zion Probabilistic Safety Study value of .28) and the frequency with which offsite power was not recovered within 60 minutes was .26 (as contrasted to the Zion Probabilistic Safety Study value of .03). Admittedly, there are deficiencies in this direct application of generic data to the Zion plant, but it is felt that the results obtained are closer to the truth than that obtained in the Zion Probabilistic Safety Study. Considering only failure-to-start of the diesel generators, and not failure to run, the probability of a loss of power to busses 148 and 149 because of failure of the diesel generators, coupled with failure to restore offsite power in 30 minutes, is:

$$P_{F30} = (1.83\times10^{-3}) \times .41 = 7.5\times10^{-4},$$

as contrasted to the value

$$P_{F30} = 6.02\times10^{-4}$$

obtained in the Zion Probabilistic Safety Study. Here 1.83×10^{-3} is the probability of failure of the diesel generators to supply power to busses 148 and 149. The difference is more significant when one considers the probability offsite power is not restored within 60 minutes, coupled with failure of the diesel generators to supply power to bases 148 and 149.

One obtains

$$P_{F60} = (1.83\times10^{-3}) \times .26 = 4.75\times10^{-4}, \text{ as}$$

opposed to the value

$$P_{F60} = 7.49\times10^{-5}$$

obtained in the Zion Probabilistic Safety Study. This is approximately a factor of six higher.

Even if one accepted the model for restoration of offsite power given in the Zion Probabilistic Safety Study, the probability offsite power is not restored within 60 minutes should be .046, not .03. The Zion Probabilistic Safety Study model for recovery of offsite power, gives the time to recovery of offsite power as the sum of two variables, which we will call t' and t". The time t' is the time for the operator to reach the 345kv relay house, check the diesel generators, check the 345kv relays and open the unit disconnects. The time t" is the time to restore power after above local operations have been performed. The frequency functions for t' and t" are histograms, so that they can be written as linear combinations of the unit step function:

$$g_1(t') = \sum_i a_i \, u(t'-t'_i)$$
$$g_2(t") = \sum_j b_j \, u(t"-t"_j),$$

where $g_1(t')$ is the frequency function for t', $g_2(t")$ is the frequency function for t", u(t) is the unit step function, equal to zero when its argument t is negative, and equal to unity otherwise. By use of the fact that the Laplace transform of a convolution is the product of the Laplace transforms of the two functions being convoluted, it is possible to determine the frequency function of t = t'+t" as

$$g(t) = \sum_{i,j} a_i b_j \; (t-t_i'-t_j")u(t-t_i-t_j'),$$

and numerical evaluation leads to the result that the probability offsite power is not restored in 30 minutes is .365, not the .28 calculated in the Zion Probabilistic Safety Study, and the probability that offsite power is not restored in 60 minutes is .046, not the .03 obtained in the Zion Probabilistic Safety Study.

We now estimate the change obtained in the frequency of sequence 44 in Table 1.3.4.11b-4 of Reference 1 when generic data is used to estimate P_{F60}, instead of the model used in the Zion Probabilistic Safety Study. This sequence is initiated by loss of offsite power, followed by loss of emergency power to buses 148 and 149, and failure to restore power for a period of 60 minutes. It also includes failure of the turbine driven auxiliary feedwater pump. The conditional frequency of this sequence, given the occurrence of the loss of offsite power initiator becomes

$$P_{F60} \times .049 = 2.32\times10^{-5},$$

when $P_{F60} = 4.75\times10^{-4}$ is used; if the Zion Probabilistic Safety Study value of 7.49×10^{-5} is used for P_{F60} one obtains 3.67×10^{-6} for this sequence.

The failure frequency of the turbine-driven auxiliary feedwater pump is .049 per demand. We estimate the initiator frequency for this event as .081/yr instead of .058/yr. The difference obtained here arises mainly from the fact that one cannot merely take the sum of the operating times of both units when updating the generic data distribution by the plant-specific data. This is a consequence of the fact that a loss of offsite power event can affect both units simultaneously. The Zion unit operating the longest has been operating 8.5 years. We obtain therefore, for sequence 44, a frequency of

$$.08/\text{yr} \times 2.32\text{x}10^{-5} = 1.9\text{x}10^{-6}/\text{yr}.$$

This sequence leads to core damage state TE. This sequence is identified as sequence 14 in Table II.2-1 in Volume 1 of Reference 1. The frequency obtained there is $2\text{x}10^{-7}$/yr; our estimate is a factor of 10 higher.

The diesel generator recovery model is also somewhat optimistic compared to the data given in Table 10 of NUREG/CR-1362 [Data Summaries of Licensee Event Reports of Diesel Generators at U.S. Commercial Nuclear Power Plants]. However, no credit was given for diesel generator recovery in the Zion Probabilistic Safety Study - the model developed was never used. The neglect of diesel generator recovery increases the calculated core melt frequency somewhat, but the conservatism introduced is not great, since, according to NUREG/CR-1362, only 23% of all diesel generator repairs take less than 1 hour.

2.2.2 Loss of Offsite Power Followed by Reactor Coolant Pump Seal Failure

There are errors in the calculations of event sequences 45 and 51 in the loss of offsite power event tree (see Table 1.3.4.11b-4 of Reference 1) of the Zion Probabilistic Safety Study. These sequences involve the reactor coolant pump seal failure after loss of offsite power and failure to supply emergency power to buses 147, 148, 149, 248, and 249. This results in loss of component cooling water, as well as loss of seal injection flow, to the reactor coolant pumps, and results in the failure of the reactor coolant pump seals. Possibly because of typographical errors, it is not possible to ascertain that sequence 51 involves the event LS, the reactor coolant pump seal LOCA, from Table 1.3.4.11b-4 of Reference 1. It is necessary to refer to the event tree,

Figure 1.3.4.11-2 of Reference 1). In any event, the conditional frequency of sequence 45, given the offsite power initiator, should be just the frequency of the event LS. Reference to p. 103-37 of Reference 1 shows that the event LS implies failure of buses 147, 148, 149, 248, 249. Nothing more is required, after the loss of offsite power initiator, since it is assumed that only a 15 minute loss of component cooling water and seal injection flow is sufficient to cause the reactor coolant pump seal LOCA. Thus, the conditional frequency of sequence 45, using the Zion Probabilistic Safety Study calculation of the frequency of event LS, is 1.8×10^{-6}, not the 3.09×10^{-8} given in Table 1.3.4.11b-4 of Reference 1. Sequence 51 should be the event LS followed by loss of power for more than 60 minutes. However, the event LS already implies failure of the diesel generators to energize buses 147, 148, and 149. Thus, in order to obtain loss of power for 60 minutes, given the offsite power initiator, and given the event LS, it is only necessary not to restore offsite power in 60 minutes. This event has, according to the Zion Probabilistic Safety Study, the frequency .03. One obtains therefore, for sequence 51, the conditional frequency (given loss of offsite power) of

$$1.8\times10^{-6} \times .03 = 5.4\times10^{-8}.$$

However, the probability offsite power is not restored within 60 minutes is .26, not .03, if generic data is used (see section 2.2.1). Moreover, the probability of the event LS is calculated assuming diesel generators associated with the two different units fail independently. (Common mode failures for the diesel generators associated with the same unit are considered. Also, external events are handled separately.) The dominant contributor to the event LS consists of the event product

$$ABC,$$

where

A = event that buses 148 and 149 are not energized by the diesel generators DGIA and DBIB.

B = event that buses 248 and 249 are not energized by the diesel generators DG2A and DG2B.

C = event that diesel generator DGO energizes bus 247 and not bus 147.

The frequency of the event LS is essentially, using the model of reference 1,

$$P(LS) = P(ABC) = P(A)P(B)P(C) = .5\ P(A)P(B),$$

since the probability DGO swings to either unit on loss of offsite power is .5. With

$$P(A) = P(B) = 1.85 \times 10^{-3}$$

one obtains $P(LS) = 0.5\ (1.85 \times 10^{-3})^2 = 1.71 \times 10^{6}$. But P(ABC) may be much greater than P(A)P(B)P(C) if there are common mode failures connecting the Unit 1 and Unit 2 diesel generators.

The type of common mode failure one is concerned about is the type which is not revealed by testing, but is only revealed during a real loss of offsite power event. A "near-miss" to this kind of event happened at Millstone Unit 2 on July 6, 1976. A voltage reduction on the grid occurred which resulted in a reactor trip but was not sufficiently great to actuate certain undervoltage relays. The diesel generators were not capable of automatic start. It really would not have mattered how many of them there were - they would have all responded in the same way. If one attempts to obtain an upper bound to the frequency of this type of event from data, one gets an upper bound estimate of the order of 10^{-2}. There have been some 400 reactor-years of PWR experience in the U.S., and the average rate of loss of offsite power is about .27/yr, so that there have been about 100 total loss of power incidents. If one says that there have been no incidents of common mode failure of the diesel generators in these 100 incidents, one obtains, at a 50% confidence level, 7×10^{-3} as an upper bound to the probability of this event. This seems too high to use as an estimate of the common mode faiure of all the diesel generators. However, it seems difficult to justify a number much below 10^{-4} per demand. It is rare to find systems without diverse subsystems, but which depend only on redundancy for reliability, to have a much lower unavailability. Then we obtain for the frequency of the event sequence 45 (from Table 1.3.4.11b-4 of Reference 1).

$$P(\text{sequence } 45) = .08/\text{yr} \times P(LS) = .08/\text{yr} \times 10^{-4}$$

$$= 8 \times 10^{-6}/\text{yr},$$

where, as before, we have used .08/yr instead of .058/yr as the frequency of loss of offsite power. This sequence 45 leads to core damage state SEFC. Sequence 51 of the same table would now be calculated as

$$P(\text{sequence } 51) = .08/\text{yr} \times P(LS) \times P(\text{offsite power not restored in 60 minutes})$$

$$= .08/\text{yr} \times 10^{-4} \times .26 = 2 \times 10^{6}/\text{yr},$$

and leads to core damage state SE.

2.3 Small Loss of Coolant Accidents Followed by Failure of Emergency Coolant Recirculation

The Zion Probabilistic Safety Study, Reference 1, does not explicitly consider simultaneous failure of both emergency coolant recirculation and containment recirculation sprays because of components common to both systems.

The core damage state descriptor (e.g., SL) gives no information about whether recirculation sprays are operating or not. Apparently the authors of the Zion Probabilistic Safety Study have tacitly assumed that recirculation spray operation is irrelevant to the containment response and radioactive release. If this were the case, then there would be no need to consider common mode failures of the recirculation spray and emergency coolant recirculation.

However, after a core melt aerosols are generated in containment. If containment recirculation sprays are not operating, then these aerosols may interfere with the operation of the containment for coolers. In the BNL critique (Reference 7) of the Offshore Power Systems risk assessment for the Zion plant, the probability of failure of the fan coolers, given a core melt and failure of the containment recirculation spray, was taken as .1. This was based on an estimate made by M. A. Taylor of the NRC. If this probability is valid, and if failure of emergency coolant recirculation, failure of containment recirculation spray, and failure of containment fan coolers leads to delayed overpressure failure of the containment (without the radioactivity removal that would occur if the containment recirculation spray were operating), then the common mode failure of emergency coolant recirculation and containment recirculation sprays is significant.

In the recirculation mode of cooling the core, for a small LOCA, the residual heat removal (RHR) pumps take suction from the containment sump; the water leaving the RHR pumps goes through heat exchangers and then goes to the containment sprays (by one path) and to the hi-head pumps for core cooling (by another path). There are motor-operated valves (SI8811A and SI8811B) in the lines from the containment sump to the RHR pumps. These valves are normally closed; failure to operate of both of these valves leads to failure of both the emergency coolant recirculation function and the containment recirculation spray function.

The Zion Probabilistic Safety Study, Reference 1, used a value of 1.55×10^{-3} per demand for the mean failure frequency of these motor operated valves (see Table 1.5.2.3.4-6 of Reference 1) and used a mean β-factor of .014 (see p.1.5-462 of Reference 1) to include common mode failures. The failure frequency of 1.55×10^{-3} is supposed to include failures in the local control circuitry for these valves (see p. 1.5-496 of Reference 1). However, following the Reactor Safety Study, both the Offshore Power System Study (Reference 8) and the BNL critique (Reference 7) of this study used a value of .03 for the failure frequency of the local control circuitry of the valves. Moreover, the BNL critique of the Offshore Power Systems Study used a β-factor of .15, an order of magnitude higher than that used in Reference 1. Including the failure frequency for valves due to failures of local control circuitry, one obtains a failure frequency of .032 per demand for one of the valves, using Reactor Safety Study data. Using the β-factor of .15, one obtains a frequency of (.032) (.15) = .0048 for common mode failure of both valves (valves SI8811A and SI8811B). There is other common equipment in the containment spray recirculation system and emergency coolant recirculation system, but the major portion of the common mode failure of these two systems comes from the common mode failure of these motor-operated valves. Using the mean small LOCA frequency of .0354 from the Zion Probabilistic Safety Study, Reference 1, and assuming a containment fan cooler failure frequency of .1 given failure of emergency coolant recirculation and containment recirculation spray, one obtains

$$(.0354)\ (.0048)\ (.1) = 1.7\times10^{-5}/\text{yr},$$

as the frequency of a sequence which may result in delayed overpressure failure of the containment. It is therefore of importance to determine whether this sequence does indeed lead to delayed containment overpressure failure, and moreover to recheck the Reactor Safety Study value of .03 for the failure frequency of the local control circuitry for a motor-operated valve, such as the valve SI8811A in the line from the containment sump to one of the RHR pumps.

TABLE 2.1

EVENT TO BE CONSIDERED IN RISK STUDIES

EVENT	REMARKS
1. Station Blackout	Included
(a) RCP seal failure	Included - deficiencies in analysis, see Section 2.2
(b) Loss of D.C. after finite time	Not Included - see remarks, this section
2. Loss of D.C. power	Included as a reactor trip initiator
3. Loss of instrument and control power	Not Included
4. RCP trip for a small LOCA	Not Included - see comments, this section
5. SDV LOCA	Not Applicable to PWR
6. Multiple instrument tube LOCA below core level	Included as part of small LOCA frequency see comments, this section
7. Overcooling Events (as pressurized thermal shock)	Not Included
8. Overpressurization during cold shutdown	Not Included
9. Large LOCA	Included
(a) RCP missiles	Not Included
(b) Other missiles	Not Included
10.. Steam Operation tube failure	Included as an initiator
11. ATWS	Included - but see comments, this section
12. Stuck open S/R valve	Included as an initiator as part of small LOCA frequency - but see comments, this section, concerning stuck open S/R valve after loss of offsite power.
13. Break in RHR during cold	Included in section on internal flooding, in Chapter 7.
14. Loss of main feedwater	Included

TABLE 2.1 (cont.)

EVENT TO BE CONSIDERED IN RISK STUDIES

EVENT	REMARKS
15. Containment Isolation	?, see comments, this section
16. Turbine Trip	Included
17. Loss of component cooling water	Included - but see comments, this section
18. Loss of service water	Included
19. Loss of ventilation in auxiliary building	Not Included - see comments, this section
20. Pipe breaks in auxiliary	Included in section on internal flooding
21. RCP seal failure	Treated as a small LOCA, but RCP seal failures that have occurred are not included in data base; see comments, this section
22. Boron Dilution	
(a) Shutdown	Not Included
(b) At power	Included as a power excursion event, (event tree 10)
23. Excess feedwater events	Included as part of turbine trip initiation frequency
24. Loss of instrument and control	Eliminated as an initiator during preliminary screening - see Section 1.3.1 of Zion Probabilistic Safety Study, p. 1.3-7.

3.0 FIRES AND EARTHQUAKES

3.1 Fires

The purpose of this section is to critique certain aspects of the fire analysis, for cable tray fires, of Reference 1.

We first make some remarks about terminology. Reference 1 makes use of two concepts, the first of which it calls "frequency", and the second of which it calls "probability". The word "frequency", in this usage, denotes a concept which someone belonging to the frequentist school of probability theorists would call probability. The word "probability", as used in Reference 1, refers to a degree-of-belief. Just as a frequentist would say that the probability of an event is not exactly known from statistical data, but that only an estimate of this probabiity can be obtained, so, in Reference 1, the authors talk about a probability of a frequency. The word "frequency", as used here, is not to be confused with "observed frequency", in a finite sequence of trials. For example, if one tosses a coin 100 times, and obtains 47 heads, then the observed (relative) frequency is .47, while the frequency is .5, if the coin is unbiased (the word "frequency" denoting what the frequentist would call probability). A random variable may take on various values with different frequencies. If the random variable is continuous, one can talk about a frequency function (corresponding to the frequentist's probability density function). If one is uncertain as to the exact value of a parameter, one assigns different degrees of belief to different values of the parameter. The parameter then has a probability density function, if it is assumed that it takes on a continuous set of values. The probability density function for a parameter serves much the same function as the frequentist's confidence interval - one gets an estimate of the uncertainty in the parameter.

There is confusion in the fire analysis of Reference 1 in the use of these concepts - the frequency of a random variable and the probability of a parameter. In order to understand this confusion, and what effects on the fire analysis it has, let us first outline the basic model used in the analysis of core melt sequences due to fires in the cable spreading room. The analysis is subdivided into the following parts:

(1) Data on cable tray fires is used to estimate the frequency of fires initiated in a cable tray - a probability of frequency curve is obtained.

(2) The CMPBRN code is used to determine the time t_V for propagation of the fire vertically to an adjacent cable tray, and the time t_H for propagation of the fire horizontally to an adjacent cable tray. In this calculation, no fire suppresion activity is assumed to take place.

(3) A family of frequency distributions for the fire suppression time, t_S, are generated from data on fire suppression times, with a probability (i.e., degree-of-belief) assigned to each member of this family.

(4) If the fire suppression time, t_S, is greater than the propagation time, t_V (or t_H), the fire is assumed to have propagated vertically (or horizontally) to the adjacent cable tray.

(5) From the analysis corresponding to parts (1) through (4) above, one can calculate the frequency of fires in cable spreading rooms which involve two or more trays. One must then consider what fractions of these fires can, combined with other events, cause a core melt.

For example, one core melt sequence considered is a sequence in which the fire disables the charging pumps and the motor driven pumps of the auxiliary feedwater system, and the turbine-driven pump of the auxiliary feedwater system is unavailable for reasons other than the fire. The calculation of the frequency of this sequence involves estimating the conditional frequency of occurrence of a fire at a particular location in the cable spreading room, given that a fire in the cable tray room has occurred.

Our analysis, in this section, focuses on parts (2), (3), and (4) of the fire analysis, and then only on some of the statistical aspects. The frequency at which a cable tray fire will propagate to a tray above the tray in which the fire has started, given a fire has started in the cable tray, is given by

$$F(t_s>t_v;\ \varepsilon_s,\ \varepsilon_v^{(1)},\ \varepsilon_v^{(2)}) = \int_0^{\infty} dt_s g_s(t_s;\ \varepsilon_s)\int_0^{t_s} dt_v g_v(t_v;\ \varepsilon_v^{(1)},\ \varepsilon_v^{(2)}) \quad (1)$$

Here the notation F(...) denotes the frequency of an event, in much the same way that pr(...) denotes, for the frequentist, the probability of an event. However, here we are dealing with a family of frequencies, indexed by the parameters ε_v, $\varepsilon_v^{(1)}$, $\varepsilon_v^{(2)}$. Each of these parameters has a degree of belief associated with it. If we assume that ε_s varies continuously, then we may associate a probability density function $h_s(\varepsilon_s)$ with it, so that $h\,(\varepsilon_s{}')$ represents the probability that the parameter ε_s lies between $\varepsilon_s{}'$ and $\varepsilon_s{}' + d\varepsilon_s$. Similar statements hold for $\varepsilon_v^{(1)}$, $\varepsilon_v^{(2)}$. (The parameters ε_s, $\varepsilon_v(1)$, $\varepsilon_v^{(2)}$ are assumed independent, although this is really not necessary.) The function $g_s(t_s;\ \varepsilon_s{}')$ represents the frequency function for the time to suppression of the fire, when the parameter ε_s has the value $\varepsilon_s{}'$. Similarly, $g_v(t_v;\ \varepsilon_v^{(1)},\ \varepsilon_v^{(2)})$ is the frequency function for the time for vertical propagation of the fire from one cable tray to the tray above it. The parameters $\varepsilon_v^{(1)}$, $\varepsilon_v^{(2)}$ may be viewed as parameters which enter into the CMPBRN code; the number of such parameters has been limited to two only for ease of exposition. Denote by $h_v^{(1)}(\varepsilon_v^{(1)})$ and $h_v^{(2)}(\varepsilon_v^{(2)})$ the probability density functions for $\varepsilon_v^{(1)}$ and $\varepsilon_v^{(2)}$. We shall limit our discussion to an analysis of the mean frequency at which t_s exceeds t_v. The mean frequency is given by

$$\overline{F}(t_s>t_v) = \int F(t_s>t_v;\ \varepsilon_s,\ \varepsilon_v^{(1)},\ \varepsilon_v^{(2)})h_s(\varepsilon_s)h_v^{(1)}(\varepsilon_v^{(1)})h_v^{(2)}(\varepsilon_v^{(2)}) \times d\varepsilon_s d\varepsilon_v^{(1)} d\varepsilon_v^{(2)} \qquad (2)$$

Use of Eq. (1) leads to

$$\overline{F}(t_s>t_v) = \int_0^\infty dt_s \overline{g}_s(t_s) \int_0^{t_s} dt_v \overline{g}_v(t_v), \qquad (3)$$

where

$$\overline{g}_s(t_s) = \int_{-\infty}^{+\infty} g_s(t_s;\ \varepsilon_s)h_s(\varepsilon_s)d\varepsilon_s \qquad (4)$$

and

$$\overline{g}_v(t_v) = \int_{-\infty}^{+\infty} g_v(t_v;\ \varepsilon_v^{(1)},\ \varepsilon_v^{(2)})h_v^{(1)}(\varepsilon_v^{(1)})h_v^{(2)}(\varepsilon_v^{(2)})d\varepsilon_v^{(1)}d\varepsilon_v^{(2)} \qquad (5)$$

Thus, the mean frequency at which t_s exceeds t_v depends only on the mean of the frequency functions for t_s and t_v.

The first difficulty with the analysis of cable tray fires in Reference 1 is the statistical treatment of t_v. The random variable t_v is treated in

Reference 1 as having an infinitely sharp frequency function, centered about a value τ_v which is a function of the parameters $\varepsilon_v(j)$:

$$g_v(t_v;\ \varepsilon_v(1),\ \varepsilon_v(2),\dots\ \varepsilon_v(n)) = \delta(t_v-\tau_v) \tag{6}$$

where

$$\tau_v = \varphi(\varepsilon_v(1),\ \varepsilon_v(2),\ \dots\ \varepsilon_v(n) \tag{7}$$

(We assume more than two parameters $\varepsilon_v(j)$ now.) The frequency function for t_v is uncertain, but once the unknown parameters are fixed, one obtains an infinitely sharp function. The parameters $\varepsilon_v^{(1)}$, $\varepsilon_v^{(2)}$, ... $\varepsilon_v^{(n)}$ are parameters which are used in the CMPBRN code which calculates τ_v. One of these parameters (say $\varepsilon_v^{(1)}$) is called Q_p in Reference 1. It refers to the heat content (in BTU) of the pilot fuel. The pilot fuel could be any of a number of materials, such as lunch wrappers or oily wiping cloths, both of which have been found in cable trays [See Reference 4, Kazarians and Apostolakis, NUREG/CR-2258, p. 91]. The function $h_v^{(1)}(Q_p)$ is given from p. 7.3-8 of Reference 1 as

$$\begin{aligned} h_v^{(1)}(Q_p) &= .1\delta(Q_p-400) + .44\,\delta(Q_p-2000) \\ &+ .44\,\delta(Q_p-10000) + .02\,\delta(Q_p-40000) \end{aligned} \tag{8}$$

The problem is that $h_v^{(1)}(Q_p)$ is really a frequency function, not a probability density function measuring degree of belief. A certain fraction of cable tray fires will be caused by lunch wrappers, some by oily wiping cloths, and so forth, and $h_v^{(1)}(Q\ )$ measures the relative fraction of fires which would be caused by pilot fuels with various values of Q_p. However, the misinterpretation of $h_v^{(1)}(Q_p)$ does not affect the mean frequency function for t_v; it affects the uncertainty bands but it is unclear how significant this is. To see that if $h_v^{(1)}(Q_p)$ were a frequency function instead of a probability density function, the mean frequency function is unchanged, one notes that if $h_v^{(1)}(Q_p)$ were a frequency function, then $g_v(t_v;\ \varepsilon_v^{(1)},\ \varepsilon_v^{(2)},\ \dots)$ would be the frequency function for t_v conditional on $\varepsilon_v(1) = Q_p$ having a specified value. The frequency function for fixed values of $\varepsilon_v(2)$, ... $\varepsilon_v(n)$ would then be

$$g_v(t_v;\ \varepsilon_v^{(2)},\ \dots\ \varepsilon_v^{(n)}) = \int_{-\infty}^{+\infty} g_v(t_v;\ \varepsilon_v^{(1)},\ \varepsilon_v^{(2)},\ \dots\ \varepsilon_v^{(n)})h_v^{(1)}(\varepsilon_v^{(1)})d\varepsilon_v^{(1)}$$

and the same formula for $g_v(t_v)$ as obtained in Eq. (5) (but extended to more than two $\varepsilon_v^{(j)}$), would be obtained.

The value of $\overline{g}_v(t_v)$ can be obtained from the graph of the (cumulative) distribution of τ_v given in Figure 7.3-3 of Reference 1 . (The quantity τ_v is called τ_v^* in Reference 1.) This graph represents the uncertainty distribution for τ_v. If $h(\tau_v)$ denotes the uncertainty distribution for τ_v, then

$$\begin{aligned}\overline{g}_v(t_v) &= \int g_v(t_v;\ \tau_v)h(\tau_v)d\tau_v \\ &= \int \delta(t_v-\tau_v)h(\tau_v)d\tau_v \\ &= h(t_v),\end{aligned}$$

since according to Eq. (6), $g_v(t_v;\ \varepsilon_v^{(1)},\ \ldots\ \varepsilon_v^{(n)})$ is just a δ function. Then the quantity

$$\int_0^{t_s} \overline{g}_v(t_v)dt_v$$

which enters into Eq. (3) can be read directly from Figure 7.3-3.

The difficulties with the statistical treatment of t_s are somewhat more substantive. Data on fire suppression times were obtained (see p. 7.3-10 of Ref. 1). To be used correctly, this data on the relative frequency of suppressing fire in various lengths of times should be used as our best estimate of $\overline{g}_s(t_s)$. If this is done, one obtains for $\overline{g}_s(t_s)$ the function

$$\begin{aligned}g_s(t_s) = &.4\ \delta(t_s\text{-5 min}) + .3\ \delta(t_s\text{-15 min}) \\ &+ .2\ \delta(t_s\text{-30 min}) + .1\ \delta(t_s\text{-60 min})\end{aligned}$$

However, for reasons which are unclear, the authors of Reference 1 said that this data on fires represents the uncertainty distribution in a mean time to fire suppression τ_s, not a sample estimate of the distribution of t_s. The actual frequency function for t_s, in Reference 1, is

$$g_s(t_s;\ \tau_s) = \frac{1}{\tau_s}\ e^{-t_s/\tau_s}$$

and the mean frequency function for t_s, according to Reference 1, is related to $\overline{g}_s(t_s)$ by

$$\overline{g}_s'(t_s) = \int \frac{1}{\tau_s} e^{-t_s/\tau_s} \overline{g}_s(\tau_s)d\tau_s.$$

Since, as discussed earlier, the values of

$$\int_0^{t_s} g_v(t_v)dt_v$$

can be read directly from Figure 7.3-3 of Reference 1, one can, from Eqs. (3) and (10) obtain our revised value for $\overline{F}(t_s>t_v)$. This value is

$$\overline{F}(t_s>t_v) = .22.$$

Coupled with the mean value of cable tray fires of $7.2\text{x}10^{-3}$/year obtained on page 7.3-4 of Reference 1, one obtains a mean probability of $1.6\text{x}10^{-3}$/year for cable tray fires involving two adjacent trays. This compares to a value of $1.2\text{x}10^{-3}$/year obtained in Reference 1, as given at the bottom of Table 7.3-2. The effect of this change is to change the mean core melt frequency due to cable tray fires from $1.8\text{x}10^{-6}$/year, as given on p. 7.3-1 of Reference 1, to $2.4\text{x}10^{-6}$/year.

There is a possibly important nonconservatism present in the fire analysis of Reference 1. This arises from the neglect of the Browns Ferry fire in determining the frequency distributions for the time to fire suppression. Although Reference 1 states that the Browns Ferry fire was included in the data base, this is only as far as determining the frequency of cable tray fires. The Browns Ferry fire took about 7 hours to control; the longest time for suppression of a fire considered in Reference 1 (see p. 7.3-10 of this reference) is 85 minutes. The time to suppression for cable tray fires used in Reference 1 was derived by Siu in Reference 5 (NUREG/CR-2269) on p. 108ff. He states there that the reason for omitting the Browns Ferry fire was that the long time required to put the fire out was due to the hesitation of plant personnel to use water to put out the fire, and that it is unlikely that that would occur again.

3.2 Earthquakes

The purpose of this section is to assess the sensitivity of the calculated frequency of seismically-induced core melt to the seismic hazard function. In particular, the effect of replacing the seismic hazard used in Reference 1 with the best estimate seismic hazard curve from the Seismic Safety Margins Research Program (SSMRP) will be determined. The SSMRP seismic hazard curve is given in Figure 11 of Reference 6, NUREG/CR-2015, Vol. 1, and is for the Zion site. Table 6 of Reference 6 gives the same information in different form, but there are errors in it; the corrected values were obtained in a private communication from Larry George. The SSMRP program reduced the seismic hazard function by .703 to take into account the plant availability factor, and because they were considering only accidents occurring during normal operation. We have not done this.

The frequency (i.e., what the frequentist calls probability) of a given plant state, given an earthquake with peak ground acceleration a, will be taken from the Reference 1 analysis; this information is presented in Table 7.2-4 of Reference 1. This table gives the conditional frequency of various seismic plant matrices. A row of the plant seismic matrix corresponds to a particular value of the peak ground acceleration. The various columns correspond to the various plant states. The only plant state of interest is that designated by SE, an early core melt which involves a small loss-of-coolant accident (which for the dominant accident sequences consists of failure of the reactor coolant pump seals after failure of all the diesel generators, and loss of offsite power). We shall determine only the mean value of the seismically-induced core melt frequency, and not consider the uncertainty estimates. From Table 7.2-4 of Reference 1, by averaging the values for the five equally likely seismic matrices given, one obtains for $\overline{F}(SE|a)$, the mean frequency of plant state SE, given an earthquake with peak ground acceleration a, the values given in the table below:

a	$\overline{F}(SE\|a)$
.225 g	.0022
.275 g	.0154
.35 g	.1358
.45 g	.464
.55 g	.796
.65 g	.950
.75 g	.994
.85 g	.9996

The mean value of the seismically-induced core melt frequency is given by

$$\overline{F}(SE) = \int \overline{F}(SE|a)g(a)da,$$

where g(a)da is the frequency (per year) of peak ground acceleration between a and a + da at the Zion site. The integral in Eq. (1) can be approximated by

$$\overline{F}(SE) = \sum_{i} \overline{F}(SE|\overline{a}_i) \int_{a_i}^{a_{i+1}} g(a)da,$$

where, for numerical convenience, the cut point a_i are chosen so as to agree with those of Figure 6 of the SSMRP study, NUREG/CR-2015, Vol. 1. The a_i are values at the center of each interval (a_i, $a_i + 1$). The values of $\overline{F}(SE|\overline{a}_i)$ were obtained from the above table of F(SE|a) by graphical interpolation. The table below gives the values of $\overline{F}(SE|\overline{a}_i)$ and $\int g(a)da$ for each interval.

a_i	$\overline{F}(SE\|\overline{a}_i)$	$\int_{a_i}^{a_{i+1}} g(a)da$
.225	.0022	3.6×10^{-4}/yr
.375	.205	1.8×10^{-5}
.525	.665	2.2×10^{-6}
.675	.965	5.75×10^{-7}
.865	1.0	2.3×10^{-8}
above .98	1.0	7.3×10^{-8}

One obtains from Eq. (2) the result

$$F(SE) = 6.6x10^{-6}/yr.$$

This differs by 18% from the result of $5.6x10^{-6}$/yr obtained in Reference 1.

4.0 Discussion and Summary

Some of the more important points made in this report are:

1. The frequency of core melt due to ATWS is 6.7×10^{-6}/yr + 8.0×10^{-5} (P-.004), where P is the probability that the operator fails to open the pressurizer block valve, in those cases where it is required. Values of P of .64 or even .95 may be appropriate, leading to values of the frequency of core melt due to ATWS in excess of 5.8×10^{-5}/yr.

2. We estimate accident sequences in which a reactor coolant pump seal LOCA occurs after loss of offsite power to have a much higher probability than was obtained in Reference 1, partly because of errors made in Reference 1 in the evaluation of this sequence, and partly because of consideration of common mode failures which would fail the diesel generators of both units. Moreover, the use of generic data to estimate the frequency function for the time to recover offsite power leads to a higher calculated probability that offsite power is not restored in one hour. The frequency of the sequence involving loss of offsite power, a reactor coolant pump seal LOCA, and failure to restore power in 60 minutes is estimated to have a frequency of 2×10^{-6}/yr. Using the assumptions made in Reference 1 this would lead to core damage state SE.

3. The core damage state descriptor (e.g., SLC), does not distinguish between cases where the containment recirculation sprays are operating or where they are not.

4. For sequences involving a small LOCA followed by failure of emergency coolant recirculation, significant differences in failure data for certain motor-operated valves, and, in particular, differences in the failure frequencies for the local control circuitry of these valves, were found between the Reactor Safety Study and Reference 1. Consideration of common mode failure of emergency coolant recirculation and containment recirculation sprays because of common components, and consideration of possible failure of fan coolers in a post core-melt environment with failed recirculation sprays, leads to a sequence which may result in delayed overpressure failure of the containment, with a frequency of 1.7×10^{-5}/yr.

5. The analysis given in Reference 1 of accidents initiated by loss of component cooling water does not address systems interactions involving failure of the charging pumps to maintain reactor coolant pump seal flow because the charging pumps require component cooling water.

6. The study may suffer from a lack of completeness as regards human-error initiated accidents and because of neglect of accidents occurring during cold and hot shutdown.

7. Corrections of an error in the statistical treatment in the fire analysis leads to only a moderate increase in the probability of core melt due to cable tray fires. The calculated frequency of core melt due to cable tray fires increases from 1.8×10^{-6}/yr to 2.4×10^{-6}/yr.

8. The use of the SSMRP seismic hazard function instead of the seismic hazard function of Reference 1 leads to a very moderate increase of 18% in the seismically-induced core melt frequency.

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