CT-1911

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SUBJECT: Review of "EPRI/WOG Analysis of Decay Heat Removal Risk at Point Beach" (NSAC-113), Final Report, Oct. 21, 1987.

Dear Paul,

Pursuant to your request, I have reviewed the subject document and my comments and observations are transmitted herewith for information and use by you and the Sub-committee on Decay Heat Removal. It should be noted at the outset that the review necessarily suffers from some rather severe limitations. In particular, resources did not permit an examination of the plant, a review of training procedures and emergency guidelines, or verification of much of the data and information which is referred to but not elaborated upon in the EPRI/WOG study. This limitation was found to be particularly significant in the area of recovery actions which played an important role in the EPRI/WOG study. Verification of the quantitative impact of recovery actions generally requires detailed knowledge of the plant layout as well as training and operating procedures. Thus, only an identification of some potentially questionable aspects of recovery could be included as part of the review.

In addition to the subject document, the review made considerable use of two related documents (References 1 & 2) for backup information.

In PRA reviews, I generally find it useful at the outset to identify and quantitatively rank the accident sequences which are dominant contributors to CMP and risk. This provides valuable perspective in evaluating which assumptions, data, and other issues are important in deriving the results. This approach is particularly useful when two PRAs are being compared, which is the case in this review (I note in this regard that the title of the subject document is somewhat of a misnomer in that the EPRI/WOG is not, as stated on page 2-1, an independent evaluation of DHR risk but rather a re-analysis of the NRC effort, as provided in Ref. 1, and is subject to the same limitations and omissions in scope).

The dominant core melt sequences will be considered first. Table 1 gives

the dominant sequences contributing to core melt probability for the two studies, with the sequences listed in decreasing probability from the NRC study. The table provides the assessed probability of each sequence and the percent contribution of each to the overall CMP for the two studies. The last column in the table provides the ratio of the NRC assessed probability to the EPRI/WOG probability for each sequence. This table was derived from information presented in the EPRI/WOG report.

Table 1 reveals that there are significant differences between the two studies. Not only is the CMP for the EPRI/WOG substantially lower (a factor of 31, with a total CMP of 1.0E-5), but, except for seismic initiators, the ranking of the dominant sequences is considerably different. (For some unexplained reason, the Reference 3 review indicates, on pages 5 of the cover letter and 5-1 of the report, that the NRC and EPRI/WOG CMP results for Point Beach are only a factor of 10 different) For the EPRI/WOG study, three of the NRC dominant sequences (Sequences 1,5, and 6) were either found to be negligible or, in the case of sequence #6, would not occur at all. The most significant difference between the two studies in terms of the sequence that contributes most to the change in core melt frequency is the internal flood sequence (#1). The EPRI/WOG assessment that the probability of this sequence is <E-8 accounts for almost 1/3 of the factor of 31 difference in the CMP between the two studies.

The two studies agree, however, in one general aspect. Both show significant contributions to CMP from "external" events. For the NRC study, 55% of the CMP contribution comes from external events (sequences 1,2, and 5). For the EPRI/WOG study, 74% of the CMP is from external events, all from seismic initiated accidents. The next most significant sequence in the EPRI/WOG study (#7) would need to have a probability increase of almost a factor of 10 to be an equivalent contributor.

It should be noted that a small part of the difference in the two results can be attributed to a design change to be made at the plant which was considered by EPRI/WOG but not considered in the NRC study. This design change is the installation of a back-up DC power supply in the form of seismically qualified batteries. Based on results in the studies, it appears that consideration of these batteries in the NRC study would result in a reduction in the CMP from 3.1E-4 to 2.8E-4, producing a modest reduction in the difference in CMP between the two studies from 31 to 28. Two sequences would appear to be affected by this change; the seismic sequence (#1 in Table 1) and the loss of off-site power sequence (#7). In the former case, it appears the NRC CMP from seismic initiators would be reduced from 6.1E-5 to 3.0E-5, which would reduce the difference between the two studies for seismic CMP from a factor of 8 to a factor of 4.

TABLE 1- COMPARISON OF CMP DOMINANT SEQUENCES BETWEEN THE NRC AND EPRI/WOG STUDIES

Sequence (designator)	NRC CMP	NRC % Cont.	EPRI/WOG CMP	EPRI/WOG % Cont.	NRC / EPRI/WOG
1. Internal Flood	7.7E-5	25	<E-8	negl.	>7700
2. Seismic	6.1E-5	20	7.4E-6	74	8
3. Small Loca, ECC recirc. failure (S_2MH_1'H2')	4.7E-5	16	5.8E-7	5.8	81
4. Long term station black-out (LTSB)	3.6E-5	12	5.4E-7	5.4	70
5. Internal Fire	3.2E-5	10	6.3E-8	negl.	500
6. Transient LOCA w/failure of ECC recirc (T_3QH_1'H2')	2.5E-5	8	N/A*	0	∞
7. LOOP with loss of feed-water and F&B (T_1MLE)	6.7E-6	2	7.7E-7	8	9
8. All other sequences	2.5E-5	8	6.5E-7	6.5	38
TOTALS	3.1E-4	100	1.0E-5	100	31

*According to the EPRI/WOG study, this sequence would not occur because the transients considered in this sequence (reactor/turbine trip with feedwater and offsite power available) would not reach the relief valve setpoints.

Since the major thrust of these studies is an evaluation of plant improvements on the basis of value/impact, an important aspect of the results is the relative off-site impact of the accident sequences. In this regard, the most significant accidents are those which result in early and gross failure of the containment function. According to Sect. 9 of the EPRI/WOG study, the expected person-rem/yr. for the EPRI/WOG study would be about a factor of 30 less than NRC if the NRC source term methodology were used in the EPRI/WOG study. This difference is essentially the same as the CMP difference between the two studies. However, use of the EPRI/WOG base case source terms (an application of the IDCOR methodology) would result in a further reduction of a factor of about 8 between the studies based on information given in Sect. 9. Thus, the base case results from the EPRI/WOG study on the basis of expected person-rem/yr. would be about a factor of 240 less than the NRC study.

Since both of the studies conclude that none of the suggested improvements are justified on a value/impact basis, this review will concentrate on evaluating the reasons for the large discrepancy in CMP.

The remainder of this review consists of: a) an examination of the differences for the sequences in Table 1, and b) miscellaneous comments and observations, primarily on the EPRI/WOG study but including some consideration of the NRC study.

A. Examination of Differences in Dominant Accident Sequences

1. Internal Flood (sequence #1)- This sequence, according to Sect. 3 (Pg. 3-26) and Appendix E of the NRC study (Ref. 1) involves flooding of the service water pump room as a result of a rupture in the fire main which apparently traverses the room. This is assumed to cause failure of all four SWS pumps which subsequently causes failure of the PCS, CCW, and loss of all safety injection pumps, RHR pumps and the two motor driven AFS pumps. The probability of this sequence is assessed at 7.66E-5/yr, the most dominant core melt sequence (25% of the total CMP). This probability is based on the product of the following factors:

Flooding probability= 2.2E-2/yr.
SWS pump failure probability= 0.1/flood event (based on judgement)
AFW steam turbine driven pump failure probability = 3.4E-2/demand

In the EPRI/WOG analysis, the probability of this sequence is assessed to be <10E-8 (see Table 1), a negligible contributor. This large reduction is based on two factors: a) the EPRI/WOG analysis computes the probability of flooding at 3.73E-5/yr. (Pg. 5-8), and b) according to the EPRI/WOG

study (Pg. 8-10) the HPI system does not depend on CCW, thus the option to provide feed and bleed cooling would still be available even if AFW were lost.

The NRC assessment of flood probability is based on the frequency of auxiliary building floods from data (not service water pump room flood data as stated on Pg. 8-10 of the EPRI/WOG study) as given in Table E1 of Appendix E. This data is referenced to be from a 1984 report by Kazarian and Fleming. I am not familar with this report and can make no judgement at this time regarding its applicability to the scenario postulated. However, the flood probability of 2.2E-2/yr seems high; it would lead to an expectation of some two such flooding incidents/yr for the U.S. nuclear plant population.

The EPRI/WOG flooding frequency is based on a correlation by Thomas (Pg. 5-8), and assumes that a rupture of a specific "T" in the fire main piping is required to cause flooding. I am not familar with either the configuration of the fire main piping in the SWS room or the Thomas correlation, and therefore cannot judge the applicability to the scenario being evaluated. However, the explanation of the correlation is not complete in the EPRI/WOG study, and I cannot reproduce the result given by applying the formula. The probability does seem quite low, implying that no such flooding would be expected in the entire U.S. reactor population during their 30 year lifetime with considerable margin.

With respect to the dependence between CCW and HPI, I do not have sufficient plant information to make an independent evaluation. The EPRI/WOG study merely states (pg. 8-10) that the dependence does not exist, while the NRC study assumes that it does. However, if feed and bleed is a viable means of core cooling for this sequence, even a factor of 0.1 in F&B success probability would reduce the NRC sequence probability to a minor contributor (2.5%).

In summary, without evaluation of additional information, I am unable to judge which analysis is more appropriate for this scenario. The NRC assessment seems excessively conservative while the EPRI/WOG result appears overly optimistic.

2. Seismic- As indicated in Table 1, the NRC estimates a CMP of 6.1E-5/yr for seismic initiated events, while the EPRI/WOG estimate is a factor of 8 lower. Page 6-13 to 6-14 of the EPRI/WOG study lists six factors which are alleged to contribute to the difference. As indicated previously, half of this factor appears to be based on the installation of new seismic resistant batteries at Point Beach which were not considered in the NRC study. The important remaining differences appear to be 1) the result of a reduced seismic hazard curve used by EPRI/WOG (Pg. 6-14) and 2) less fragility assumed for the RWST. In the case of the reduced seismic hazard

curve, the EPRI/WOG study assumed that the NRC curve should be reduced by a factor of 2 for less than 3XSSE and a factor of 5 for greater than 3XSSE. These reductions are said to be based on discussions with EPRI (Pg. 6-13).

The factor of four difference in the two results (neglecting the effect of the new batteries) would not generally be considered very significant given the large uncertainties normally attributed to seismic CMP estimates. However, the factor becomes more significant in this comparison because of the overwhelming dominance (74%) of seismic CMP in the EPRI/WOG study. For example, if the EPRI/WOG seismic CMP were raised by a factor of eight to be consistent with the NRC result, the total CMP for the two studies would be different by less than a factor of five rather than 31.

I have insufficient information to judge the fragility of the RWST. With respect to the seismic hazard curve, I performed a comparison of the NRC result with a hazard curve prepared for the Zion site, the closest site to Point Beach for which I have independent seismic hazard information (the sites are about 125 miles apart). The Zion hazard curve is contained in the Zion PRA (Ref. 3).

The NRC study does not contain an actual hazard curve, but merely lists return frequencies for ranges of accelerations which are multiples of the SSE (Pg. 3-11 of Ref. 1). On page C-6, the SSE is listed as 0.12g, thus, the return frequencies can be related to a range of accelerations. If these relationships are plotted on the Zion seismic hazard curve (Fig. 7.2-1, Pg. 7.2-18 of Ref. 3) a comparison can be made. The comparison is somewhat complicated by the fact that the Zion curve is actually a family of 9 curves each of which is assigned a probability of being applicable to the Zion site. However, the NRC data falls above (assuming that the frequencies apply to the midpoint of the acceleration range) the upper bound of all of the curves for accelerations less than about .35g., and is above all but two of the curves (with a combined probability of 0.216) for accelerations above .35g. On this basis, the NRC data seems quite conservative, and the EPRI/WOG revision would be more consistent with the Zion curve, even though the proposed revision produces a change which is inconsistent with the trend of the Zion curve (ie, the NRC curve is further outside the Zion bounds at lower accelerations where the proposed EPRI/WOG revision is less significant).

In an attempt to add further perspective on this issue, I have compiled seismic core melt frequencies from a number of PRAs, all for plants east of the Mississippi. The result of the compilation is presented in Table 2. The NRC and EPRI/WOG results both fall within the range of results in the table. The EPRI/WOG results are quite close to the Zion PRA results, the closest site to Point Beach (note however, that the SSMRP result for Zion

TABLE 2- COMPARISON OF SEISMIC CORE MELT PROBABILITES FROM SELECTED PRAS

PLANT	TOTAL CMP	SEISMIC	UNCERTAINTIES	ACCELERATION
Zion	7E-5	5.6E-6	2E-8/3E-5(1)	Becomes important at .6g
Indian Pt.-2	5E-4	1.4E-4	7E-6/5E-4(1)	-
Indian Pt.-3	2E-4	3.1E-6	4E-12/2E-5(1)	Major failures start at .6g
Millstone-3	1E-4	9.4E-5	-	Becomes important at .3g
Seabrook	2E-4	2.8E-5	-	Max contribution at .7g
Limerick	5E-5	5.7E-6	1E-9/3E-5(2)	-
Oconee 3	3E-4	6.3E-5	-	-
SSMRP(3)	-	2E-4	-	-
Point Beach* (NRC)	3.1E-4	6.1E-5	-	Major contribution .36-6g (Pg. 3-11)
Point Beach* (EPRI/WOG)	1.0E-5	7.4E-6	-	-

(1) These values are described in the PRAs as the 90% confidence intervals
(2) These values are the 5% and 95% bounds
(3) This study is the Siesmic Safety Margin Research Program for the Zion plant sponsored by the NRC, see "Application of the SSMRP Methodology to the Seismic Risk at the Zion Nuclear Power Plant", Lawrence Livermore National Labs, M. P. Bohn, et al, May, 1983.
*these studies exclude consideration of large break LOCAs and ATWS

is considerably higher, even in excess of the NRC result for Point Beach). Not much can be made of this comparison because of the very wide range in the results. At best, it can be stated that neither the NRC nor EPRI/WOG result is outside the (rather considerable) range of seismic CMPs for other nuclear plants east of the Mississippi.

3. Small LOCA, ECC recirculation failure- As shown in Table 1, the EPRI/WOG probability estimate for this sequence is a factor of 81 lower than the NRC evaluation. The factors contributing to the difference, according to page 8-4 of the EPRI/WOG study are: 1) frequency of small break LOCA, 2) CCW success criteria, and 3) operator actions. These will be considered separately.

The frequency of small break LOCAs in the NRC study was estimated at 2E-2/yr., while the EPRI/WOG study uses 3E-3. This difference contributes a factor of 7 to the total difference (a factor of 81) in this sequence. The NRC frequency is dominated (Pg. 5-3 of EPRI/WOG) by pump seal LOCA events and is based on an internal NRC memorandum (to D. Eisenhut from T. Murley) which is based on an LER search. However, the EPRI/WOG study argues (Pg. 5-3) that the majority of these events were failures which resulted in very small leak rates which would not trigger containment spray operation and thus would not require ECC recirculation (note that the sequence definition implies small break LOCAs which require ECC recirculation). The EPRI/WOG frequency is also stated to be consistent with the Oconee PRA which is based on industry data, and is consistent with other PRAs. In my opinion, the EPRI/WOG value is more reasonable. The NRC value would imply two small LOCAs requiring ECC recirculation per yr in a 100 reactor popullation. To my knowledge, no such events have yet occurred (also stated by the EPRI/WOG report) in 400 reactor years of experience. The EPRI/WOG estimate is some three times higher than that used in WASH-1400 which seems to adequately account for the possibility of large leak pump seal LOCAs. It is also my understanding that Westinghouse has implemented procedures for recovery of losses of pump seal injection and cooling which are the likely causes of pump seal failure. (I note that the NRC study assumes, Pg. A-1, as does the EPRI/WOG assessment, that pump seal failures as a consequence of station blackout are not considered. I questioned this assumption at a previous ACRS sub-committee meeting and was told that the assumption is based on the presumption that the pump seal LOCA issue under these conditions will be resolved under another unresolved safety issue, and the resolution will eliminate this event from consideration. It seems plausible that if the issue is resolved by some design or operational change, this change will also affect the probability of spontaneous pump seal LOCAs).

With regard to operator actions, the NRC study assumes a human failure probability for failure to initiate recirculation of 3E-3 (table 4-4 of EPRI/WOG), while EPRI/WOG uses 1E-4, which accounts for a factor of 3.3

in the probability difference for the sequence being considered. However, page 2-11 of the NRC study indicates that a human failure probability of 1E-3 was used for this sequence, which would account for a factor of 10 difference. Based on the NRC study results, it appears that 1E-3 was actually used. Without examining the actual proceedures, their availability, and the timing for operator actions, I have no basis to judge which assessment is more realistic. Given the long times involved for the scenario under consideration, and the increased attention which has been placed on recirculation procedures since WASH-1400, I suspect that the EPRI/WOG assessment is more realistic. It also seems consistent with other PRA results as indicated in Table 4.4 of the EPRI/WOG study.

The NRC study assumes that high pressure recirculation is dependent on the RHR pumps which are in turn dependent on CCW (Pg. 2-15). The EPRI/WOG study apparently assumes that no such dependency exists, but I could find no discussion of this dependency. The only reference to high pressure injection dependency on CCW which I could find appears on Pg. 8-10, and this apparently is related only to the injection mode of high pressure ECC and not the recirculation mode (see item 1 above). Without detailed plant information on dependencies, I have no basis to make a judgement on this issue. The discussion on Pg. A-10 of the NRC study indicates a dependency, but acknowledges that it does not apply for some sequences. The small break LOCA event tree in Appendix B (Pg. B-10) does not include CCW as a heading for the sequence being considered here.

4. Long term station blackout (LTSB)- A factor of 70 difference exists between the NRC and EPRI/WOG assessment for this sequence. I was unable to determine the quantitative significance of all of the factors which appear to contribute to the reduction. The factors which appear to be relevant include:

Difference in LOOP initiating frequency (factor of 1.3 per Table 5-1)
Difference in Diesel Generator CC failure (factor of 3.3, Table 5-1)
Installation of new station batteries (unknown factor)
Manual operation of turbine driven AFW pump (unknown factor)

The difference in LOOP initiating frequency is not significant and will not be considered further.

The diesel generator common cause failure for the EPRI/WOG study is estimated at 5.0E-4 on page 5-4 and 1.5E-3 for NRC. The basis for the EPRI/WOG value is stated (Pg. 5-2) to be that the common cause values were based on "methods consistent with current industry practice', and were derived using the NRC result as a starting point, "compared to other PRAs and further updates were based on plant design and experience as well as opportunity for recovery". Further justification is given on 5-11. Part of the basis is related to the relatively low common cause probability

provided in the Millstone 3 PRA. However, this value appears to be related to the low single diesel generator failure probability used in the Millstone PRA based on plant specific tests of the diesels which would not be applicable to Point Beach. These low values were criticized in an NRC review of the Millstone 3 study(5). Based on a survey of diesel generator common cause failure data, the EPRI/WOG value seems low. From Reference 4, the probability of failure on demand for 2 of 2 diesel generators (the Point Beach configuration according to Appendix A, Pg. A-21 of the NRC study) varies from 2.3E-3 to 7.8E-3/demand from seven sources based on U.S. data (this assumes a 4 week test interval for the data source dependent on test interval). It is not clear from reference 4 if or how recovery was accounted for, nor is it stated what time interval was assumed as the mission time for the failures. However, under plant blackout conditions it is my opinion that recovery operations will be quite difficult due to several factors, including potential lighting problems, inability to operate repair equipment which depends on AC power, and problems with limited and non-renewable air supplies for repeated start attempts. It is my view that the NRC result is more reasonable; it is already below the lower bound of the reference 4 survey.

With respect to the additional batteries, they appear to be of benefit only to maintain steam generator liquid level measurements (Pg. 4-11 and A-9) for use when manually controlling the steam turbine driven AFW pump, which is discussed below.

As stated on Pg. 8-2, the largest factor appears to be the EPRI/WOG assumption that local manual control of the steam turbine driven AFW pump could be accomplished after loss of both AC and normal DC, with communication between the local control personel and the plant operators (Pg. 3-6 and 4-11). I am skeptical that such operation would likely be successful. First of all, according to page 3-6, under these plant blackout conditions, a diesel fire pump needs to be started to provide cooling for the turbine driven pump. This requirement is not acknowledged in the procedure described on page 4-11. It is not stated how long the turbine pump can operate without normal cooling (supplied by service water), but it is estimated (Pg. 3-8) that it would take 10-15 minutes for the operator to reach the AFW equipment. It is not clear if the pump would have already been automatically started and be operating without cooling or have tripped from overheat. Further, the time to reach the equipment should consider that the plant is in a blackout condition, ie- no elevators, limited lighting, disabled security systems if dependent on AC power, etc. Local operation of the steam admission valve to the turbine would appear very difficult unless provisions have been made to protect the operator from the excessive heat, and the noise from pump operation, diesel fire pump, and steam flow could make communication with the operating room, necessary to control steam generator level, very difficult. Time is limited to accomplish all of the required operations and stabilize flow (see

related comment # 8 following). Without more detailed information, it is not possible to determine if all of these potential problems have been adequately considered, but they do not appear to have been systematically evaluated in the EPRI/WOG report. Consequently, I am inclined to accept the NRC assessment as being more realistic until a more definitive analysis of the operation can be evaluated.

5. Internal fire- The EPRI/WOG fire assessment produces an estimated CMP from fires which is a factor of 500 below the NRC assessment. Without additional plant information, including layout information for critical components, fire protection systems and philosophy, control of combustibles, fire barriers, etc. I cannot make a definitive judgement on the fire assessment for either study. On the basis of information in the EPRI/WOG assessment (Pg. 6-4 et seq) it appears that 1) neither study provides a comprehensive fire risk assessment utilizing state of the art analysis, and 2) the NRC assumptions with respect to fire risk appear overly conservative.

6. Transient LOCA with failure of ECC recirculation- This sequence, according to EPRI/WOG, will not occur because, as noted previously, this transient would not reach the relief valve set points. I have no basis to judge if the EPRI/WOG evaluation is valid or not.

7. Loss of off-site power with loss of feedwater and feed and bleed- The EPRI/WOG assessment estimates that the probability of this sequence is a factor of 9 less than NRC. According to page 1-13 of EPRI/WOG, the key reason for the change is the addition of new batteries at the Point Beach site. This assessment appears valid.

B. Miscellaneous Comments and Observations

1. The NSAC Perspective section of the report refers (Pg. V) to a core melt probability target in the NRC safety goal. There is no reference to a core melt probability in the published version of the safety goal. A more meaningful comparison would be public health risks which are the exclusive quantitative provisions of the goal.

2. Pg. 1-1: It is stated here that Point Beach was one of 6 plants selected in the A-45 effort "to represent a broad range of reactor and DHR system designs, so as to form the basis for a consistent set of generic or group-generic new licensing requirements for DHR at U.S. reactors." This description of the selection process seems to ignore what appears to be the single most important selection criteria: "Point Beach was identified in the initial qualitative screening as having sufficient potential vulnerabilities (to decay heat removal) to warrant additional study" (Pg. 1-3 of Ref. 1).

3. Pg. 1-12: It is stated here that "The EPRI/WOG analysis of Point Beach resulted in significant reductions in core melt frequency (of the NRC results) due to special emergencies. In the case of seismic analysis, the major changes occurred as a result of considering a variety of recovery actions for earthquakes less than three times the SSE." This explanation seems inadequate and inconsistent with the NRC result which indicates (Pg. 3-11) that the major contribution to core melt from seismic events comes from accelerations between three and [illegible] times the SSE.

4. Pg. 1-15: It is indicated here that recovery actions including use of water from the spent fuel pool have been added to the NRC model. If spent fuel is stored in the spent fuel pool which has to be immersed to prevent overheating, this may not be a reasonable alternative to consider.

5. Pg. 3-4 and 3-5: The basis is not provided for assumptions and statements given here. In particular, the assumptions regarding the time that PORV block valves are closed and the statement that relief valve set points would not be reached for the transients considered.

6. Sect. 3: Numerous assumptions which are stated to be conservative are listed in this section (cf. Pg. 3-11, 3-12, 3-14, 3-15). However, no information is provided to enable an evaluation of the effect of these conservatisms. Such information would be useful in providing additional perspective on the results.

7. Pg. 5-1: It is indicated here that Point Beach has operated 16 years, thus providing valuable plant specific data. However, neither the EPRI/WOG or NRC studies appear to account for plant aging effects. An ongoing NRC program to examine this issue has found that plant aging effects can increase the CMP for older plants.

8. Pg. A-9: The times used here to evaluate human error probabilities for operating the turbine driven AFW pump upon loss of DC appear optimistic. It is stated that 10-15 minutes would be required to manually start the pump, including local operation of the DC steam admission valves. This is stated to leave 15-20 minutes available for recovery before steam generator dryout. However, on page 3-8 it is stated that it takes the personnel 10-15 minutes just to reach the pumps after being dispatched by the plant operating staff. Thus, it would appear that only 5-10 minutes is available for recovery. Furthermore, since this scenario appears only to be important in conjunction with loss of AC (otherwise the motor driven AFW pumps would be available), the diesel driven fire pump must also be started to provide cooling to the turbine driven pump.

9. Pg. 9-1: It is not clear why the EPRI/WOG expected dose for case 1 (interpretation of NRC calculations) is so much lower (a factor of 15) than

the NRC Case Study results. Further, it is also not clear why the IDCOR source term results are higher than the interpretation of the NRC calculations when the IDCOR source term essentially eliminates all early containment failures.

10. Pg. C-2: It is not clear why the entries under containment failure mode probability are identical for all accident types, nor is it clear why the failure mode probabilities do not sum to 1.0.

11. Pg. C-10: It is not clear why the La releases for the DH (etc.) failure mode have not been reduced by a factor of 2, nor why the I-Br has not been released by a factor of 2 for the MT cases.

12. Pg. C-11: The Xe-Kr release fractions are greater than unity (1.7) for Accident Type 4.

13. Appendix C, general: It is not clear what evacuation assumptions were made in the EPRI/WOG consequence analysis. However, it should be noted that the overwhelming dominance of earthquake initiated accidents to CMP may mean that evacuation is degraded or impossible for most risk significant sequences due to off-site damage (communications, road ways, etc.) from the event. While this factor may not be significant for value-impact analysis which consider on-site costs (since they will dominate as stated on Pg. 4-1 of Ref. 2) it may be important for analyses which do not consider such costs. (Inclusion of on-site costs has been a controversial issue, for example, see Pg. 5-2 of Ref. 2)

14. As an observation, it is worth emphasizing the discussion in Sect. 7 which brings out some important points relevant to adding additional safety systems which are consistent with my own concerns. In summary, it is noted that consideration of such systems must be done carefully, with emphasis given to the details of the sequences which they are being installed to prevent, and recognition of interfaces and dependencies with other plant systems. Such systems can impose a detrimental burden to the operators, can introduce additional accident sequences, and can exacerbate some sequences previously found to be insignificant contirbutors. (Ref 2 also provides some cogent observations in this regard.)

I hope this review is of use to you and the Sub-committee

Sincerely,

P. R. Davis

CC: Dave Ward, ACRS

REFERENCES

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2. Letter, Gerald Neils, Chairman NUMARC Working Group on DHR to D. Ericson, Jr., Sandia National Laboratories, June 22, 1987.

3. Zion Probabilistic Safety Study, Commonwealth Edison Co., 1981.

4. "Common Cause Failure Data: Experience From Diesel Generator Studies", S. Hirschberg and U. Pulkkinen, Nuclear Safety Magazine, Vol. 26, No. 3, May-June 1985.

5. "A Review of the Millstone 3 Probabilistic Safety Study", NUREG/CR-4142, A. A. Garcia, et al, Lawrence Livermore National Laboratory, Oct. 1985.