ENVIRONMENTAL PROTECTION AGENCY

WASHINGTON, D. C. 20460

November 1974

Comments by the Environmental Protection Agency

on

Reactor Safety Study

AN ASSESSMENT OF ACCIDENT RISKS IN U.S. COMMERCIAL

NUCLEAR POWER PLANTS

UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
WASHINGTON, D.C. 20460

27 NOV 1974

Mr. Saul Levine
Project Staff Director
Reactor Safety Study
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Mr. Levine:

The Environmental Protection Agency's comments from the initial phase of its review of WASH-1400 ("An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants") are transmitted with this letter.

Because the assessment reported in WASH-1400 is expected to be a principal step toward establishing the accident risk associated with nuclear power plants, we are reviewing it in two phases. The first phase is represented by the enclosed preliminary comments based on a two-month review effort. The second phase will include an in-depth review of selected aspects of the study with technical assistance being provided to EPA through a contract with Intermountain Technologies, Inc. This second phase should be concluded by May 1975, at which time our intent is to issue a final report detailing all of our comments. During this period of continuing review we hope to maintain a close liaison with the Atomic Energy Commission so that our final report will reflect an up-to-date awareness of any resolution attained regarding comments by EPA or others on the draft report.

We have reviewed the work plan for our continuing effort with members of your staff as well as others in the technical community. We are also including it as a part of our review comments so that others may be cognizant of our planned efforts.

Our initial review indicates that the Reactor Safety Study provides an innovative forward step in risk assessment of nuclear power reactors. The general methodology and approach utilized in determining risk levels developed in the Reactor Safety Study appear to provide a meaningful basis for obtaining useful assessments of accident risks at nuclear power plants. Certainly, significant improvements in obtaining and utilizing nuclear plant operating data could considerably narrow the uncertainty range of risk estimates. We do, however, believe that certain aspects of

the report require modification and information additions. In particular, the consequence modeling assumptions appear to underestimate the health effects resulting from the accident sequences associated with the larger releases of radioactivity. It is uncertain what the impact of this apparent underestimation may be on the resultant risk assessment.

Although the report does not make an absolute judgment on nuclear power plant accident risk acceptability, the comparative risk approach highlighted in the summary and the main volume of the study will certainly imply an acceptability judgment to the average reader. EPA recognizes that the comparative risk approach is a first step in addressing this question, but by itself is misleading. However, studies in progress by EPA and others indicate that judgments on "risk acceptability" are extremely complex, with comparative risk evaluations representing only one of numerous inputs which must be considered.

We are interested in the plans for application of this methodology to other reactor systems and other components of the nuclear fuel cycle. Certainly, we would recommend that studies of this type should be considered by the applicable AEC successor and that their intent in these areas be publicly stated.

We would be pleased to discuss our comments with you if they require any clarification.

Sincerely yours,

W. D. Rowe

W. D. Rowe, Ph.D.
Deputy Assistant Administrator
for Radiation Programs (AW-558)

Enclosure

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INTRODUCTION AND CONCLUSIONS

Review Perspective

The Environmental Protection Agency has completed a preliminary review of the draft report "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, prepared by the Atomic Energy Commission. Our review process will continue through April 1975 at which time we will issue a final set of comments. During this period of continuing review, we hope to maintain a close liaison with those responsible for the Study so our final comments will reflect an up-to-date awareness of any resolution attained regarding comments by EPA or others on the draft report.

EPA's review of the Study cannot be considered as exhaustive in that many of the calculational details and data base have not been checked. Our focus has rather been one of emphasizing a review of major assumptions, concepts, methodology and approach. Although EPA's resources are limited when compared with those utilized in the development of the Study, it was deemed necessary to do as comprehensive a review as possible due to the many significant implications the Study has with regard to areas of EPA responsibility. EPA's primary concerns deal with the health and safety of the public and the protection of the environment from the consequences of accident releases. In this respect, we are involved in the planning for mitigation of these potential consequences, including the development of appropriate Federal guidance, and in assuring that the public risks incurred are societally acceptable. Within this context we attempt to maintain cognizance of accident analysis activities so that we can be continually aware of both the probability of accidents and the consequences for such accidental releases. Due to the importance we attach to this subject and the broad range of subject matter considered in the Study, we believe it is imperative that it receive a thorough and critical review by the general technical community and the public. We realize that much of this review that we suggest is already underway or planned. However, we feel that comments developed on the Study should be referenced in the final version of the Study and copies of these reviews should be publicly available.

In continuing its review, EPA has contracted with Intermountain Technologies, Inc. to assist us in the evaluation of the range of applicability of the various analytical models and assumptions utilized in the assessment. The preliminary work plan for this effort is presented in an attachment to our comments. If, in the initial stages this detailed review of the selected failure mode paths or critical source term parameters indicates a general agreement with the Study's evaluation, that portion of the investigation will be terminated and other failure mode paths or source term parameters may be substituted in this work plan.

Review Format

Following this Introduction and Conclusions section, our review takes up individual groups of volumes of the WASH-1400 document by first presenting general comments and then specific comments. This sequence begins with the main volume of the Study and continues with Appendix VI (environmental consequences); Appendices V, VII, and VIII (accident sequence, meltdown processes, and radioactivity releases); Appendices I, II, III, and IV, (definitions of failure pathways), Appendix X (design adequacy) and the summary volume, in that order. The last section of our review presents Additional Comments in order of the Study volumes themselves. These latter comments were not felt to be of the same level of significance as those referred to in the previous sections of our review.

Main Comments and Conclusions

EPA has made a broad spectrum of specific comments on the Study, realizing that they have varying degrees of impact on final results. However, as the document is bound to be used as a reference for many follow-on studies and analyses, we feel it is desirable to make it as complete and accurate as possible in all its facets. EPA's main comments and conclusions, although of a preliminary nature, are as follows:

1. The Study is innovative in both its concept and methodology and provides an innovative forward step in risk assessment of nuclear power reactors. In this respect, the AEC is to be commended. The general methodologies and rationale developed in the Study to determine risk levels appear to provide a meaningful basis for obtaining useful assessments of accident risks of nuclear power plants.

2. Appendix VI (environmental consequences) received particular attention in our review due to its pertinence to EPA concerns. This appendix was found to be quite weak in a number of respects and not up to the general thoroughness that appears to permeate many other sections. Our preliminary review indicates, for example, that if the recommendations of the BEIR Report are followed, the consequences estimated in the Study may be low, in certain cases, by factors of 2 to 5. In addition, the evacuation model assumed for the reference case consequence calculation also appears somewhat overly optimistic. Based on the information presented in the Study, this could increase consequences by at most a factor of 2 to 4 (i.e., no evacuation). Therefore, the combination of these factors could result in an underestimate, by about an order of magnitude, of the consequences associated with the "high" release accident sequences. Since the high release accident sequences are significant, but not dominating, contributors to the overall risk assessment, the resultant assessed

risk magnitude would be increased but by a lesser factor. It is suggested that appropriate modifications should be made or the rationale for utilizing other assumptions should be provided.

Furthermore, the description of certain critical portions of the overall calculational process should be significantly expanded to permit a clear understanding of the relationships between the radioactive material releases, its dispersion, population distributions, and the resulting health effects.

3. Although the Study indicates that no absolute judgment on nuclear power plant acceptability is intended, the comparative risk approach highlighted in the summary may well imply an acceptability judgment to the average reader. It should be further pointed out in the report that the comparative risk approach is only a first step in addressing this question and by itself can be misleading. It can be noted that studies in progress by EPA, National Science Foundation, and others, indicate that judgments on "risk acceptability" are extremely complex, with comparative risk evaluations representing only one of numerous inputs which must be considered.

4. As can be expected with such a voluminous report, a number of apparent inconsistencies, format difficulties, and cases of insufficient supporting information were encountered. Particularly in Appendix II there were inconsistencies in identification of components and levels of detail in the various fault trees and system descriptions. There were also problems with the lack of a readily accessible glossary of abbreviations and with inadequate cross-referencing among appendices. It is suggested that the Study be subjected to the necessary editing to eliminate abbreviations wherever possible, glossaries be added for those abbreviations used (e.g., foldout in Appendix I) and the cross-referencing between appendices be improved. The formats employed in Appendices III and VII are worthy of consideration for use in all appendices.

5. There is some concern relative to a lack of certainty as to what the follow-on actions in this program area will be. This is intensified by the recent reorganization of the AEC and its functions and lack of definition as to where this effort will be picked up and continued. We would expect that some follow-on effort should be directed toward additional verification that the design, operational, or other variations among the 100 nuclear plants to which the Study is applied, do not significantly affect the overall risk calculated by the Study. Of major interest would be consideration of other plant designs such as Westinghouse 2 and 4 loop reactor coolant systems, and BWR Mark II and III containment designs. Other items such as the use of hydrogen recombiners and differing modes of containment spray injection should also be considered for

examination. It is realized that in many of these cases what may appear as a significant difference on the system or component level may not significantly change the overall risks but some documentation of this should be presented in order to show that to be the case. A further concern relative to continuing effort in this area but not related to this specific Study is the application of this methodology to other reactor systems and other components of the nuclear fuel cycle. Certainly we would recommend that these studies should be considered by the applicable AEC successor and that their intent in these areas be publicly stated.

ASSESSMENT OF ACCIDENT RISKS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS

MAIN VOLUME

General Comments

The main volume presents a well written introduction to and summary of various analyses presented in the supporting appendices; therefore, comments on the material within this volume are generally covered elsewhere in this review. The discussion in Section 5.3 pertaining to the process of assessing release category probabilities was especially informative. The assignment to a category release probability of a 10% contribution from adjacent categories certainly adds a significant element of conservatism to the resulting probability values. The Study also attributes additional conservatism to the Monte Carlo process used to assess failure rate median values. However, the degree of conservatism attached to the Monte Carlo process throughout the Study, relative to its ability to compensate for wide ranges in available input data, may be somewhat misleading, especially if the log normalized data are "processed" through a series of "and" or "or" gates. It would appear that, in such cases, the Monte Carlo process would be expected to yield a point estimate similar to that attained through a straight additive or multiplicative process of input value median value failure rates, with an associated error factor. Although the Monte Carlo process is statistically correct, a further explanation of this process indicating the differences between it and the point estimate approach should be presented with regard to the evaluation of associated error factors. However, it should be noted that statistical techniques such as this, although appropriate analytical methodology, can never conclusively show that all critical pathways to an accident occurrence have been considered.

Chapter 6 of the main document presents a comparison of the nuclear accident risks to other societal risks. Although the Study does not make an absolute judgment on nuclear power plant accident risk acceptability, the comparative risk approach certainly implies an acceptability judgment to the average reader. EPA recognizes that the comparative risk approach is a first step in addressing this question; however, studies in progress by the EPA and others indicate that judgments on "risk acceptability" are extremely complex, with comparative risk evaluations representing only one of numerous inputs which must be considered.

Specific Comments

The question of applicability of the Study results to all current commercial water reactors is very pertinent. The discussion on page 27 appears to be the only place in the entire report that the question is considered, and only a brief general assessment is attempted. It is generally recognized that there are certain design differences in Babcock & Wilcox and Combustion Engineering plants as well as the Westinghouse plants of four-loop design (more common than the three-loop system selected). Similarly, the BWR containment design, in particular, has undergone two major changes (the Mark II and Mark III containments) since the reference design Peach Bottom plant, which would be expected to at least change the details of the containment response analyses. It would appear that the Study could benefit significantly by recognizing these design differences and presenting the necessary arguments which support the thesis that these design and response differences at the system design level do not have a major effect on the overall risk assessment. Further discussion appears warranted and any continuing analyses by the AEC to further verify this conclusion should be included.

On page 45, the safety improvement analogy with the aircraft and automobile industries regarding increasing safety with development is questionable. In both of these cases, safety improvements were accomplished by utilizing accident experience data. Although there has been significant variance in safety between particular designs in these industries, hopefully, similar significant differences will not be the case in nuclear plant safety designs. Furthermore, these industries have received increasing government control with the rise in concern over inadequacies.

On page 104, the argument for overprediction of fission product release from molten fuel appears to be partially contradicted by the discussion in the second paragraph under <u>Meltdown Release Component</u> on page 8 of Appendix VII. Similarly, the large surface area to volume ratio of the molten fuel described on page 119 should enhance the release of isotopes rather than "limit" it, as indicated. The discussion on page 109 of reduced doses associated with wind direction change conditions may be offset by increased evacuation difficulties. It is not clear if this has been considered.

<u>CALCULATION OF REACTOR ACCIDENT CONSEQUENCES</u>

Appendix VI

<u>General Comments</u>

Appendix VI appears to need substantial modification and information additions, especially with regard to the health effect calculations. The approach and methodology, although possibly adequate for the purposes of the Reactor Safety Study, should not be presented as the approach and methodology which calculates the consequences of accidents "as realistically as is now possible," as indicated in the USAEC's Interim General Statement of Policy on WASH-1400, dated August 2?, 1974.

Although some of the factors affecting consequences are adequately discussed for the purposes of this report, there is no description of the overall calculational process which would permit a clear understanding of the relationships between the radioactive material releases, the dispersion, the population distributions, and the resulting health effects. Obviously, many refinements to the various calculation models are available. Those which were assessed and found negligible in effect for the purposes of this report should be discussed to give a better appreciation of the range of applicability of the calculation model used.

Specific Comments

The reasons given in Section 6.2 of Appendix VI for selection of critical radioisotopes do not support the omission of plutonium-241. Data presented by the USAEC in the draft WASH-1327, "Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in LWR," on pages I-13, I (A)-2, and II-20, indicate that for exposure by inhalation of plutonium within a few years after its production in the LWR uranium fuel cycle, Pu-241 contributes more to the dose than Pu-239. Similarly, the data referenced by the Study (ICRP-II) and the plutonium isotopic mix of WASH-1327 indicate that Pu-241 contributes more to the dose than Pu-239 in the majority of the organ doses considered in Table VI-16 of Appendix VI, the proportion of their contributions depending upon the solubility of the plutonium aerosol.

The discussion of meteorological models and assumptions should be expanded to discuss the expected calculational differences incurred with the Study's use of a simplified model as opposed to the more conventional but complex models in general use. For example, it is inferred from the discussion on page 16 of Appendix VI that much of the meteorological frequency information is taken from greater heights than the release heights predicted in this Study. Since wind velocity generally increases with altitude, using such information will tend to decrease the estimated downwind dose levels. The acute health effects, therefore, could be underestimated. Furthermore, the uniform distribution in the crosswind direction used in the atmospheric model, as described in Section 6.4 of Appendix VI, is also likely to produce an underestimate of acute health effects, since the sector averaged dose estimates should be lower than actual peak doses. Finally, without consideration for wind meander, the constant angular widths chosen appear to broaden the plume more than would be expected (Ref. Figure A.2, page 408, of "Meteorology and Atomic Energy - 1968"), again contributing to lower peak doses and thus fewer acute health effects. It is judged unlikely, however, that any underestimate of acute health effects resulting from the treatment of meteorological information and the dispersion model is greater than a factor of two.

The model selected to account for the effect of evacuation on the calculation of medical consequences is described on page 31 of this appendix. An EPA report, "Evacuation Risks - An Evaluation," EPA-520/6-74-002, is referenced in support of certain assumptions used in the evacuation model. Although a number of parametric calculations relating to the evacuation model assumptions are presented in Table VI-21 (including a no evacuation case study), we believe the base case evacuation model to be overly optimistic.

The EPA report on evacuation risks was primarily directed at an assessment of the risk of death and injury, and the costs associated with past evacuations of population groups. The data and information utilized in this study were obtained by contacting persons and organizations involved with previous evacuations precipitated by natural or man-made causes. Factors which were hypothesized to influence the time required for the historic data base evacuations included: (1) time lapse before onset of incident, (2) availability of evacuation plans, (3) time of day, (4) weather conditions, (5) population size, (6) area size, (7) characteristics of the area, (8) conditions of roads, (9) nature of incident, (10) warning time, and (11) population density. No correlation with evacuation times could be determined for parameters (1) through (9). Similarly, since warning time was generally not separable from the time associated with accomplishment of the evacuation, no quantitative evaluation of this parameter was made. A correlation of evacuation times with population density, however, was performed assuming independence from other parameters. A trend showing an increase in time required for evacuation was indicated as population density decreased. In applying this conclusion to evacuations which may result from potential nuclear accidents, an element of caution needs to be exercised. It should be remembered that the data on historic evacuations generally include situations applying to small areas or in the case of larger areas, when there is a lengthy forewarning time. More significant is the fact that evacuation travel distances were almost always short and safe destination points were generally obvious.

Since the evacuations called for in the larger consequence accidents appear to involve evacuation areas of a few hundred square miles, the application of evacuation time requirements from the EPA evacuation study to areas of this size is questionable.

Although the Appendix VI discussion states that an outermost limit for evacuation of 20 miles was assumed, it is not clear how this evacuated population segment is treated in terms of actual dose received. Similarly, in the assessment of property damage on page 63, the effectiveness of the 10 rem calculated yearly dose as a basis for temporary evacuation is unclear since the expected dose rate as a function of time is not indicated.

The assumption stated on page 63 regarding a first year projected dose of 10 rem as the criteria for determining the decision to evacuate may be unwarranted. A suggested value of 10 to 20 rem is cited, but the reference, although relevant, does not contain such a suggestion. In it, the recommendation is made that for small population groups, the use of evacuation as a protective action be considered if the anticipated exposure during 30 days might exceed a whole body dose of 2 rad or a thyroid dose of 10 rad. The reference suggests that under less favorable circumstances evacuation might not be considered as a protective action unless larger exposures were anticipated. The reference does not support the implication that 5 rem per year or less is acceptable because it is below the occupational dose limit, nor does it suggest 10 rem or any other projected dose as a criterion for decontamination.

The evaluation of health effects appears to require significant modification and information additions. Our preliminary indications are that if the BEIR Report (The Effects on Populations of Exposure to Low Levels of Ionizing Radiation, NAS/NRC, November, 1972) recommendations are followed, the assessed latent cancers are low by a factor of four (excess mortality by a factor of two), genetic effects are low by a factor of five, and acute deaths and acute illnesses are low by an undetermined factor. Although such changes in calculated health effects do not necessarily alter the results of the Study, either these changes should be made or the rationale for utilizing other assumptions should be justified. The following discussion highlights our evaluation in this regard.

Section 6.6 appears to totally neglect the 2% of the population in utero. A list of applicable references related to this subject is included in the additional comments section of our review. This rather important segment of the population should be considered with regard to the effects of radiation.

On page 34 and 48, a statement is made that the BEIR risk estimates are upper limit values. In fact, the BEIR Committee estimates, as they state, may be too high <u>or too low.</u> Although the value of zero could not be excluded by the data, <u>use of this number</u> was rejected on a number of cogent bases (pages 2 and 88 of the BEIR Report). Also, on pages 34 and 48, the somatic excess death risk used (50-165 deaths/10^6 man-rem) only reflects the range of absolute risk estimates from the BEIR Report. Consideration should be given to the estimates which, in combination with the absolute risk estimates lead to the BEIR Committee "most likely" numbers (150-200 deaths/10^6 man-rem). Using BEIR "most likely" numbers for a lifetime plateau instead of a 30 year plateau for expression of effects, the EPA estimate of 200 excess deaths/10^6 man-rem has been developed (page 167-174 BEIR).

The BEIR Committee genetic effects estimates referred to on page 35 may be construed as meaning that 10^6 man-rem would produce 10-100 dominant diseases and 1-100 congenital anomalies in the first generation after exposure. However, this is 1/5 of the total impact expected. An additional 40-400 dominant diseases and 4-400 congenital anomalies should be attributed to future generations (page 53 BEIR).

The reference for the numbers in rads used as criteria for estimating acute effects is not presented. The average dose which will cause fatalities to 50% of the people so exposed in 60 days is given by Lushbaugh, Comas, Edwards, and Andrews (Sect 17 in AEC CONF 680410, 1964) as about 235 rads. These authors estimates for the dose which will cause fatalities to 10% of the people so exposed, LD_{10}, is of the order of 75-80 rads with a range of about 40-120 rads. This estimate is of the same order of magnitude as the estimate of less than 5% mortality in the dose range of 40-140 rads given in NCRP Report #29. Therefore, the statement on page 37 that there is little chance of death from doses below 100 rads appears somewhat optimistic for estimating the possible effects on a large population. A more accurate estimation of effects would be made using an appropriately justified probit analysis with perhaps a cutoff at 40-50 rads.

On page 50, a statement regarding the deleterious genetic changes expected per 10^6 man-rem of exposure is presented, which also appears to be a misinterpretation of the BEIR Report. The BEIR Committee estimated that the average mutant persisted in the population for five generations not "...in the first and also in all generations..." Therefore, the total increments shown in Table VI-14 should be five times greater.

Similarly, the quotation from the BEIR Report (page 91) appearing on pages 52-53 is truncated to an extent that, in our opinion, a misinterpretation of the BEIR Report results. The paragraph quoted continues: "By extrapolation, it can be estimated that the number of deaths per 0.17 rem per year in the entire U.S. population may range roughly from 3,000 to 15,000 with the most likely value falling in the range of 5000 to 7000 (or 3500 per 0.1 rem per year)." Utilizing this last estimate, the excess mortality from all forms of cancer calculated in WASH 1400 would be almost doubled.

Possible clinical effects from acute radiation exposure other than death and radiation sickness are discussed only briefly on page 46. For example, temporary aspermia in the male has been observed following exposures as low as 12.5 R. The personal trauma of being unable to reproduce or of it being recommended that no attempt be made to conceive a child for some extended period after exposure is not negligible, at least for normal "peacetime" operations. Furthermore, the disruption of the homeostasis of the finely tuned endocrine system, while possibly amenable to hormone replacement therapy, does not necessarily represent insignificant individual trauma or financial burden. Therefore, a significant expansion of this presented discussion appears warranted.

ACCIDENT SEQUENCES, REACTOR MELTDOWN PROCESSES AND RADIOACTIVITY RELEASES Appendices V, VII and VIII

General Comments

These appendices, which follow various accident sequences through the meltdown process and associated releases of radioactivity, represent a significant effort to quantify the consequences of reactor meltdown accidents. It is recognized that to present a meaningful discussion of the many accident sequences evaluated, to relate these sequences to the timing and physical processes associated with a reactor meltdown, and to predict the resulting radioactivity releases via several containment failure mechanisms is a formidable task both technically and documentarily.

Of these three appendices, Appendix V (possibly because it pulls together much of the information presented in Appendices VII and VIII) appears to require some additional effort to resolve apparent inconsistencies and to supply additional information on accident sequences other than the large LOCA. Furthermore, the Study should highlight and expand the sensitivity analyses on ECCS functionability and the evaluation and significance of the various containment failure mode probabilities.

Our comments on both Appendices VII and VIII are dealt with in the specific comment section which follows.

Specific Comments

Appendix V

One problem in reviewing Appendix V involves apparent inconsistencies between the various tables which relate accident sequences to release categories. For example, on pages 21 and 24 (Tables V-3 and V-4), it is not clear how these lists were compiled. Both tables do not include some of the dominant large LOCA sequences from Table V-6 (e.g. AF-γ and ACD-α in category 1, and AF-δ in category 3) but do include sequences which are not considered dominant (e.g., ACDGI-α). Based on the discussion (page 140), it is also not obvious why sequence ACDGI-α is classified as release category 1 instead of category 3. Furthermore, in comparing the probabilities given in Table V-6 with the relative containment failure mode probabilities listed in table 2 page 124 of the attachment, certain sequences listed as "other large LOCA accident sequences" appear to be significant contributors to a release category probability (e.g., category 2, AHF - α, 3×10^{-10}; AH-δ, 4×10^{-11}). If these contributions are numerically correct, the sum appearing at the bottom of the table must only represent the sum of the listed dominant sequence probabilities. Since the large LOCA's do not dominate the probabilities of Table V-16, information similar to that presented in Table 2 of the attachment, page 124, applicable to small LOCAs and transients would appear pertinent for inclusion in this appendix.

We would like to emphasize at this point that inclusion in a release category probability estimate of a 10% contribution from adjacent release categories adds considerable conservatism to certain summed release category probabilities; however, an attempt should be made to correct and clarify the interpretation of these summary tables.

In the discussion of the smoothing of release category probabilities, p. 50, it is not clear how the smoothing technique necessarily swamps any common mode failure contribution. The presentation would also be clarified if even just an illustration were included which would show the bar chart in Figure V-1 reversed in relation to the severity categories.

Considering the interest attached to the ECCS functionability, the discussion on pages 52-55 is especially pertinent. This sensitivity analysis discussion might be considerably improved by not only relating the ECF contributions to overall release category probabilities but also to the "large LOCA" contribution. This latter relationship would show a larger percentage contribution. For example, given a large LOCA (A) Followed by ECF (E), one accident sequence would be AE- ε with the same consequences as AD- ε (category 7). The probability would be AE- ε = (1×10^{-4}) (10^{-2}) $(\sim 1) = 10^{-6}$ assuming the high end of the ECF failure rate. Although this and other sequences would have a moderate influence on large LOCA release categories, the limited impact on the overall release category probability would be highlighted. Since the ECF failure probabilities are of general interest, it would appear appropriate to identify the rationale for assuming the failure occurrence range utilized (10^{-2}- 10^{-5}). Considerable confusion is also caused by not including ECF sequences in Table V-16 while including such sequences in Table V-6.

With regard to the BWR transient tree quantification on page 68, it is not clear from this discussion, in conjunction with Table V-19, which transients were slow enough such that credit for reserve shutdown can be taken .

In attachment 1, Table 2, certain sequences are shown with "containment rupture - vessel steam explosion" failure mode probabilities of zero which are nevertheless estimated as 0.01 in Table V-6. Since similar tables are not included for S_1 and S_2 initiating events, the relationship between the various containment failure mode probabilities shown in Table V-7 and V-8 cannot be determined (e.g., the relationship between S_2 C-δ and S_2 C-α).

Appendix VII

The information contained in Appendix VII is well presented, sufficiently documented, and based on our preliminary review, presents a reasonable appraisal of the extent of radioactivity releases.

In discussing the meltdown release component for alkaline earths and noble metals (p.p. 11; 13), the probable values selected appear somewhat low if consistency with the selection basis of other released components (e.g., halogen, alkali metals) is to be maintained. In fact, the text, in discussing the alkaline earths, indicates a release range of 2-20% and suggests that the probable value should lie in the upper portion of this range, yet selects 10% as a most probable value.

On page C-1, last sentence, it is not clear what is referred to by "...the LOCA's postulated; i.e., successful ECC and recovery," since the Study is concerned with many LOCAs in which successful ECC and recovery are not assumed.

In outlining the accident sequence and core response on page C-2, the basis for the 100% rod failure at a maximum clad temperature of 2200 °F should be stated since this failure value appears to be quite conservative in view of vendor calculations (eg. **Surry**, Final Safety Analysis Report). Also, in the discussion of six critical points involved with the evaluation of the LOCA prompt release fission product source-term, the term "release coefficient" (escape fraction) should be made consistent with nomenclature used elsewhere in the report.

In Appendix K, p. K-19, it is not clear if the text is implying that a potential important pathway for release of fission products, between the containment shell and cofferdam, was not considered in the Study.

Appendix VIII

The discussion under "Limitations" on page 3 of Appendix VIII contains a disclaimer regarding the potential non-applicability of "these studies" (presumably core meltdown studies) to other PWRs and BWRs. As mentioned previously under the specific comments on the main volume of the Study, the discussion on this topic should be expanded.

In discussing the basic assumptions for the analysis of degraded accident behavior (p. 7), the basis for assuming that "core melting would take place without significant metal-water reaction and that there would be no possibility of steam explosions in the reactor vessel" under conditions of accumulator and pumped ECI failure needs further explanation to account for the possibility of residual water being left in the vessel from the blowdown process. With regard to the accident time scale, the 10- 11 second time quoted for essentially complete primary system depressurization and the time of accumulator discharge appear a factor of 2-3 too short compared to the results in the paper, "Comparison of Thermal-Hydraulic Response of LOFT and a Large PWR to LOCA Conditions," authored by P. Davis and J. Ductone, presented at the topical meeting on water reactor safety. CONF-730304, March 1973.

On page 12, the dismissal of the potential for a large energy release from a steam explosion between the molten core and water laden gravel seems to be contradicted by the Armco Incident described on page D-2.

In describing containment response (p-13), the assumption of LPRS cavitation at the time of containment failure seems pessimistic, if Regulatory Guide 1 is followed. The guide states, "Emergency core cooling and containment heat removal systems should be designed so that adequate net positive suction head (NPSH) is provided to system pumps assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present prior to postulated LOCA."

The meltdown sequence discussion, which includes CSRS and LPRS failures (p. 17), describes the molten core vessel penetration and interaction with the water in the reactor cavity and the CSRS water. Our understanding is that the CSRS water should not be expected in the reactor cavity except for possibly a small amount of leakage. If this interpretation is correct, all sources for reactor cavity water need further clarification. Similarly, on page 21 it appears that CSIS is assumed not to deliver water to the reactor cavity while the opposite is true for CSRS.

The assessment of containment failure mode probabilities includes the probability of containment failure resulting from a steam explosion estimated as $P \approx 10^{-2}$ (+1,-2). Since we are not aware of any discussion which indicates that this probability is sequence dependent, the probabilities associated with certain sequences in Table V-16 are not understood (e.g., $S_1D-\alpha$, 3×10^{-8} while $S_1D-\varepsilon$, 6×10^{-6} and no $S_2C-\alpha$ which should be at least 2×10^{-8} based on $S_2C-\alpha$ of 2×10^{-6}). The latter example may be eliminated because containment overpressure failure occurs before initiation of core meltdown. If this or other sequences are logical exceptions to the containment failure probability associated with vessel steam explosion, the exceptions should be discussed.

The assessments of containment failure probabilities from hydrogen combustion or overpressurization both are strongly dependent on the assumed normal distribution of containment failure pressures of about 100 psia with a 15 psi standard deviation and containment melt-through time (for which the meaning of the skewed distribution, 13(+10,-5) hours, is not clear). Given the information in figures 4 through 9, general correlation with containment failure mode probabilities listed in table 2, attachment 1, Appendix V could be observed. However, such was not the case for sequences AHF-δ and AD-δ . It would be helpful if the text provided an example of such a calculation, which would define the various probabilities listed in the text.

Additional clarifying remarks would seem appropriate at several points within Appendix A, which discusses the thermal evaluation model. On page A-16, the assumptions used for the core temperature distribution and vessel water inventory following blowdown should be stated and justified.

Regarding the fission product release fraction equation, the basis referred to should be specifically referenced. Similarly, the reasoning for assuming no change in steam properties due to hydrogen mixing should be presented. Under the heading "Convective Heat Transfer" the values chosen for T_W and h_B should be discussed since it would appear that h_B should vary with $Q_{D,R}$ ΔT, and pressure.

On page A-33, a question arises as to whether vessel failure can occur by fracture due to thermal stress occurring when the molten core contacts the lower vessel head.

It is not clear on Page A-36 if the continued addition of water on top of the core melt could cause a steam explosion similar to the East German incident described on page B-3.

The containment failure mode evaluation presented in Appendix E considers several factors which could affect the ultimate containment strength. Further discussion or clarification of the potential significance of these factors on the assumed 100 ± 15 psia failure pressure appears warranted. Since the assumed failure pressure could alter the containment failure mode probabilities for several accident sequences, an indication of the sensitivity of the release category probabilities to a change in the assumed containment failure pressure should be provided.

DEFINITION OF FAILURE DATA AND PATHWAYS

(Appendices I, II, III, AND IV)

General Comments

Our review of Appendices I and II has, for the most part, been limited to questions regarding the treatment of (1) specific failure pathways which are not acknowledged in these appendices or (2) the rationale for dismissal of other failure pathways or their relationship to assigned containment failure mode probabilities. Review of certain sections of Appendix II is presently anticipated in our continuing review of the Study.

In our limited review of Appendix IV, the role of the "common mode failure" in the overall risk assessment process is difficult to assess. Some methods, quantification techniques, causes and results are discussed in Appendix IV, but the material necessary to properly understand the total role and significance of common mode failures and to determine that a reasonable degree of completeness has been developed appears to be spread through Appendices I, II, III and V. Many summary statements in the earlier and later sections of the report assert the significance of common mode failures without quantification or reference when, in fact, the needed material is in other appendices. Further cross-referencing to such analyses and quantifications which support the assertions of this appendix could resolve these concerns.

Specific Comments

Appendix I

In the LOCA functional event tree development (p. 13 footnote), it appears that the containment building purge system has a probability of failure which is not acknowledged. Similarly, the possibility of containment overpressure failure prior to core melt which is treated in subsequent appendices, is not included in the discussion on page 21.

In the development of the PWR small LOCA event tree, S_2, p-132, it would seem possible that vessel melt-through could occur while the primary system pressure is above the accumulator injection pressure. After vessel melt-through, the primary system pressure would be rapidly reduced, allowing possible accumulator water injection onto the molten core, potentially causing containment rupture from a steam explosion. It could not be determined if the containment failure mode probabilities for S_2 LOCA's considered this possibility.

With regard to PWR reactor vessel rupture, p. 141, it is not clear how the polar crane presents an effective missile barrier for the entire upper portion of the containment.

On page 145, the reason for not considering rupture of steam generator tubes and subsequent overpressurization of the secondary system with potential for rupture outside the containment, should be stated. Similarly, on page 159 it is not obvious why the situation of automatic trip failure occurring with loss of electric power sequence was eliminated from consideration.

Appendix II, Vol. 2

The discussion of the electrical power system, offsite common mode failures (p. 33), does not specifically indicate whether earthquakes have been considered in assessing common mode failures, especially for power subsystems which are not specifically designed for earthquake response (diesel fuel system). Similarly, in the text on page 37, a discussion of how the failure analysis of the diesel generator system accounts for the failure modes discussed would be an important addition to this section.

The evaluation of the reactor protection system (RPS) dismisses the impact of a pressurizer vapor space rupture on the RPS (signal initiation) failure probability. Since it is possible for the low pressurizer level signal not to function due to frothing, the effect of such a failure should be addressed. On page 155, under CSIS failure modes, it appears that the Refueling Water Storage Tank (RWST), "suction line plugged," should also be listed under single failure resulting in unavailability of RWST water.

The Consequence Limiting Control System (CLCS) description on page 174 is confusing since it appears from this discussion that the operator may not be able to switch from CSIS to CSRS until the containment pressure falls to -0.5 psig (such a pressure may not occur in time to permit successful switch of these systems).

The results of analysis of system interfaces under Low Pressure Injection System (LPIS) indicates, on page 230, that momentary unavailability of water at the start of LPIS pump operation is not considered a failure, while unavailability of water to the CSRS is considered a failure for the same reason (Appendix I, p. 102). A clarification of this situation appears warranted. Also, in listing the single failure-failure modes for LPIS (p. 234), consideration should be given to pipe ruptures between J3 and either V4 and V5 (figure II-53) and to RWST pump suction drain plug.

In the examination of potential faults for the Low Pressure Recirculation System (LPRS) on page 498, it is not clear that pipe ruptures between P1, P2 and J3 will not cause system failure. The flow would split between the path to the cold leg and the rupture and, depending on relative flow resistances, the delivery of water to the cold leg may not achieve the necessary 300 gpm.

Appendix II - Volume 3

In the evaluation of the BWR electric power system, an assumption is made on page 27 that all emergency buses are available immediately prior to a LOCA based on the Technical Specification requirement that the reactor be shutdown if an emergency bus is not available. However, it appears there should be a finite probability that all emergency buses are not available which should be dependent on the failure probability of the failure detection system. Also in discussing offsite power common mode failures, the omission of earthquake as an initiating event should be addressed.

Discussion on page 134 relates failure of vacuum breaker valves in the open position to the defeat of the vapor suppression function. It would seem appropriate that the assumptions regarding the two or more valve failures required should be justified with calculations or referenced to pertinent information. Similarly, the assumption on page 243 that rupture of branch piping of 2-inch diameter or less will not significantly affect core spray injection system operation at the time of a LOCA or during injection requires justification.

Appendix III

This appendix on failure data is well written, well organized and appears to be appropriately integrated into the Study. In our continuing review, reflected in the attached work plan, EPA does intend to perform a selective review of the failure rate data base.

Appendix IV

Our review of Appendix IV, to date, has been limited, especially with respect to the analysis and quantifications applied in the Study; therefore, our comments at this time are very general in nature. The concept and influence of the "common mode failure" appear to need additional development. A distinction should be provided between the causative effects of certain common mode failures in initiating accident situations vs the influence effect of common mode failures resulting during an established accident sequence (e.g., the influence of a check valve slam and subsequent water hammer damage as a common mode initiator vs the consequence of such an event occurring during an accident in progress).

The completeness of the consideration given to potential sources of common mode failures also appears to require some expansion. Although, through searching in other appendices, it is evident that particular types and areas of common mode failures are considered, sources, such as the requirement for pump inlet subcooling for emergency coolant recirculating systems, pump bearing lubrication systems, instrument and component service water and air, heat tracing for plateout prevention, control room and cable tray fires, drain plugging of storage water systems, etc, ought to be discussed and treated in Appendix IV to support the claims and assertions developed.

Finally, the treatment of the method of screening for relevant common mode sources discussed on pages 18 through 39 appears credible but is unsupported and in need of tabular listing of (or reference to) the "numerous" types of sources considered in order to demonstrate that the method is indeed comprehensive in identifying all conceivable component, system, and operation vulnerabilities to common mode failures. The examples cited are useful but create questions of "what else," "how many," and what does a complete list look like.

DESIGN ADEQUACY

(Appendix X)

General Comments

This appendix does not appear to be tied in with the rest of the Study. There is no mention of how the results of this Study were utilized in the risk assessments. Since Appendix X indicates that a significant number of the systems examined were either not properly qualified, not properly analyzed, or didn't meet current standards, it would seem very important that these deficiencies be readily traceable to the quantitative risks, or that they be shown to be peculiar to the plant analyzed.

Specific Comments

The section on seismic loads (p. 47) appears incomplete in that current design response spectra were not evaluated for the structures and equipment. On pages 52 and 57, it is stated that the current spectra would increase seismic loads (by as much as a factor of 2). It is not clear what these increases mean relative to the general seismic vulnerability of the 100 plants and what risks are associated with the increases.

SUMMARY REPORT

General Comments

The summary document is a relatively well written volume, which satisfies its intent through a question and answer format. Our comments on certain quantifications of assessed impact are incorporated into our review of the Appendix VI volume. Of particular interest was the discussion comparing the Study predicted consequences with the earlier WASH-740 evaluation. It would appear that the significance of the four factors leading to differences in the two studies is substantial. Plume rise and evacuation and possibly population, as treated in WASH-1400, have relatively little impact on consequence when compared to the effect of the differences in assumed release of radioactivity to the environment. The Study indicates that given a PWR core meltdown event, a chance of only about one in one hundred exists that the resultant containment failure mode will be other than melt-through with its relatively insignificant radioactivity release. A somewhat similar case exists for the BWR meltdown event where a chance of only one in ten exists that the containment failure mode will be other than containment isolation failure in the drywell with, again, a relatively insignificant release of radioactivity. It would appear appropriate that the discussion of this variable in the summary document should be expanded.

Additional Comments

Main Volume

Clarifications

1. Page 122, Table 5.2 - It is not clear why AB, ACHF, S_1B and S_2B do not lead to containment failure by overpressure since loss of containment heat removal should lead to overpressure failure.

2. Page 126, Section 5.3.2.1 - The definition of a large LOCA being a rupture equivalent to a hole greater than 6 inches in diameter is not consistent with the definition used by vendors, AEC-Regulatory, etc., which is a 0.5 ft^2 (9" hole). It is not clear why a different definition was chosen here.

3. Page 154, item (1) - The SL-1 accident was a military power reactor nuclear accident which resulted in 3 fatalities. It appears that the statement ignores the SL-1 accident.

4. Page 216, last sentence in Section 6.4.7 and Figure 6-10 - The statement that the calculated probability of a dam failure resulting in 10,000 fatalities "...agrees with the extrapolation of the data..." does not appear justified. A straight line can be drawn through the three data points (as was done in Figure 6.9), and, if anything, an upward inflection of the curve is indicated by the data, rather than downward as drawn, to include the calculated point.

Editorial

1. Page 146, Section 5.4.4 - The source for the probability of aircraft impact accidents should be referenced.

2. Page 150, Section 5.4.6 - Near site explosions, which must be considered for reactor sites, are not mentioned.

3. Page 200, Section 6.4.1, 1st sentence - The reference does not agree with the reference at the end of Table 6.8.

4. Page 204, reference 1 - This reference appears incorrect, "...North Atlantic Hurricanes..." since the Galveston hurricane is apparently included (#1, described on page 200).

5. Page 205 - The average number of tornado fatalities is stated as 118 while the division indicated yields a value of 46.

Appendix I

Clarifications

1. Page 83, Figure I-13 - It is not clear why the success path for containment leakage is chosen as the drywell and the failure path, the wet well. Wet well leakage should produce the lesser consequences due to fission product scrubbing in the torus (see Appendix I, page 37).

2. Page 102, 3rd sentence, and page 134, item F. - The basis for the CSRS failure assumption is not clear since CSRS should eventually operate.

3. Page 199 - Further justifications of the unanticipated transient probability of 10^{-5} per year should be presented.

4. Page 205, Figure I-28 (also Footnote 1, page 207) - The RPS failure probability for unanticipated transients (Part C) has been increased from 4×10^{-7} to 4×10^{-6} to account for the fact that only the scram system may be effective for reactor shutdown. This reduction does not agree with the fault tree at the bottom of page 68, Appendix V, which assigns the failure of RPS to scram a value of 1.3×10^{-5}.

Editorial

1. Page 45, 3rd paragraph, 1st sentence - It appears that CR-VSE should be CR-CSE.

2. Page 77, Figure I-10 - The LPIS is missing from the ECI segment of this figure.

3. Page 198, 1st paragraph - The apparent distinction between a transient which causes a LOCA and a transient which causes a ruptured reactor coolant system is not clear.

4. Page 222, 4th sentence under RHRS - This sentence is not complete.

5. Page 233 - These footnotes appear to be used in Table I-13, but the heading does not match the heading for Table I-13.

6. Page 259, item 5 - This item appears out of place in that it is not a "...design feature provided to keep the likelihood of loss of pool water small..."

Appendix II, Vol. 1

Editorial

1. Page 11, last paragraph - This disclaimer paragraph seems to indicate that if data did not exist for a particular system failure contribution, it was not considered.

Appendix II, Vol 2

Clarifications

1. Page 253, item 3 at top of page - There does not appear to be any basis for the assumption that pipe ruptures of 2 inch diameter or less will not cause failure of accumulator injection.

2. Page 385, Introduction - It is not clear if the SICS analysis also applies to the small break case.

3. Page 490, top of page, and page 529, 1st paragraph - It is not clear how realignment of the LPR system to the hot legs will prevent an "undesirably high boron concentration or accumulation of residue and debris in the core that could result from continuous boiling." LPR system water injected in the hot legs will enter the upper plenum, run down the outer (cold) core and core structure region into the lower plenum, and be available for boiling in the hot central core region.

4. Page 490, top of page - It appears that closure of V_{10} is also required to effect the realignment.

5. Page 501, 2nd paragraph - It is not clear why air suction from the RWST occurs for this failure in view of the discussion under item (2), page 498.

Appendix II, Vol. 3

Clarifications

1. Page 92, Section 3 - It is not clear why Q_{MED} (1.3×10^{-5}) is considerably different from the RPS unavailability shown in Figure II-131 (2.47×10^{-6}).

Editorial

1. Page 359, item 2 - The use of the term "suppression chamber" is inconsistent with "wetwell" and "vapor suppression system" used elsewhere in the report.

Appendix III

Editorial

1. Page 187, first line - The bibliography section mentioned here appears to be missing.

Appendix IV

Clarifications

1. Page 43 - Results of the susceptibility analysis are presented but no specific reference is given to where the analysis is presented and the specific fault and event trees to which it was applied.

Editorial

1. Glossaries and definitions are sorely needed for this appendix, not only to track the latter sections in relation to Appendix II, but to understand the distinctions between the PWR and BWR treatments.

2. Page 8 - The treatment of ideas at this early stage in the appendix requires the reference to other unspecified appendices in order to understand the terms used and messages developed. An introductory tutorial treatment with a description of the other appendices which intimately interface with Appendix IV is needed.

3. Page 8-15 - This section could benefit by specific cross references, examples and limited numerical results to give significance and meaning to this important portion of the report.

4. Pages 40-41 - The list of "classes of potential common mode mechanisms" could benefit by a sub-category of items under each major topic to provide an index of completeness, e.g., where would failure causes fall for wearout due to exercising a given component, or for partial or delayed performance due to degradation from lack of service, or for transient behavior of a component (check valve water hammer).

5. Pages 40-63 - Although Sections 3.3 through 4.0 portray a reasonable description of the methods applied to the "quantifications" in the study, the interpretation could be considerably aided by examples with numerical results or tabulations, such as that of Table IV-4 on coupling probability.

6. Pages 65 and 87 - These two sections are intended to treat PWRs and BWRs separately and this should be stated in the introductory paragraphs.

7. Pages 65-98 - This Section, "Summary of Results," acknowledges the performance of the "fault analysis" in Appendix II and from those results identifies selected "sequences in the event tree...chosen because...(of) some potential susceptibility for common modes" and develops "impact" conclusions as "insignificant," "minor impact," etc. The support for and meaning of these conclusions should be identified.

The event sequences selected for the follow-on discussions appear without comparative discussion to other cases which have been dismissed. Although these discussions improve one's insight to the "controlling" common mode sequences, tabulations or some form of overall results presentation should be developed to enable the reader to gain a "feel" for the relative influence or "impact" of other sequences which could be important to plants of newer design than those chosen for analysis. The companion treatment given to the BWRs (page 87), although different in style, is equally obscure in portraying understanding and confidence that the treatment of common mode failures is comprehensive and complete.

APPENDIX VI

Clarifications

1. The description of the release and dispersion calculation in Appendix VI appears sketchy in that there is not a clear description of the radioactive material release magnitudes as a function of time over the release durations presented. Thus, any interaction of the airborne release with the population being evacuated cannot be evaluated. The description suggests that the fraction of core inventory released is modeled as a uniform release over the indicated duration of release. An alternative model could be a distribution of discrete releases as shown in Figure J-8 of Appendix VII. A clarification of this subject is in order.

2. The discussion of the consequence calculation and population distribution patterns of Appendix VI does not describe the model of the population distributions used for calculation of consequences within 70 miles; i.e., it is not discernible from the information presented whether the sector population, originally obtained as a function of distance from the reactor, was averaged over the first 70 miles or averaged over segments of sectors using differences in the cumulative populations from Table VI-6, or whether some other distribution model was used.

3. A more careful explanation of the population averaging method on page 24 would be helpful. In particular, the top 1% sectors are reflected in the peak case consequence results. The range of populations averaged into the top 1% would clarify the nature of the top population category.

4. The application of the plume broadening for meander over extended periods of time, described in Section 6.43 of Appendix VI, needs to be specified more clearly. Table VI-2 shows categories PWR 6 and PWR 7 having a duration of release of 10 hours and all other categories having shorter releases; it is not clear whether categories PWR 6 and PWR 7 are the only categories having "releases that last for many hours," i.e. categories to which the broadening was applied, or whether the broadening was applied to shorter releases as well.

5. Because this appendix does not present the necessary information regarding individual organ or whole body doses as a function of release category and downwind distance, several questions arise as to the significance of certain omissions from Table VI-16; namely, (1) lung dose contribution from noble gas inhalation, (2) consideration of Pu-241, Am, Cm, and U releases, (3) releases of longer lived isotopes, such as I-129 and H-3, and (4) any possible significant release of activation products.

6. With regard to the evacuation model, clarification is needed of the manner in which the warning time for evacuation T_j (time between awareness of impending core melt and leakage for accident type j) was determined. It is observed that, in Table VI-2, this time is constant for each reactor type and independent of the containment failure mode, and also that for release category PWR I, awareness of impending core melt is immediate at the outset of the accident.

7. Page 36, Section 6.6.3. - This section is based on available data and is apparently extended for standard man only. The uncertainties in the estimates, particularly as they apply to differences in age and state of health, should be at least underscored and, if possible, explored further.

8. Page 37 - The listing of peripheral blood element response should be compared to data given by Wald (Chapter 23, Haematological Parameters after Acute Radiation Injury, pp. 253-264 in Manual on Radiation Haematology, IAEA Technical Report Series No. 123, 1971).

9. Page 47, Section 6.6.4.4. - Reference and justify assumptions, particularly the "...slightly increased number of induced mutations." If a value judgment is to be made, a frame of reference must be established.

10. Although reference is made to the BEIR Report, the discussion regarding Table VI-13 is misleading. This table, taken from p. 171 of the BEIR Report, refers to the "...absolute risk for those aged 10 or more at the time of irradiation..." This is neither the complete estimate of the BEIR Committee nor the only population considered.

11. Page 49, Section 6.6.4.3 - This section does not mention the rather generalized "increased ill-health" considered in the BEIR Report.

12. The discussion of thyroid illness on pages 53 and 54 appears to need considerable clarification. In particular, the apparent treatment of production of nodules as an illness requiring a surgical process is not understood. For an estimate of nonfatal malignancies, reference to the BEIR Report would seem appropriate.

13. Page 54, first paragraph - The assumptions on incidence of nodule formation following thyroid exposure discussed on page 54 should be justified. For example, data in Reference 42 of the subject draft report suggest that, in a mixture of external and internal radiations, gamma and beta exposures are equivalent. The BEIR Committee points out studies evidently showing a species difference in response to beta irradiation of the thyroid and also points out the problems in some available human and I-131 data (a thyroid ablating dose is used).

While Reference 42 does mention thyroid nodularity incidences ranging from 0.47% to 1.6%, it should be pointed out that the 1.6% incidence was in a population of 30 to 59 years of age and 0.47% was in a general population. The 0.36% to 1.7% values in controls in various studies reflect small numbers in the populations and, perhaps, the regions of the country from which the populations were derived.

Lilien, et al (AM Lilienfeld, M. L. Levin and I. I. Kessler, Cancer in the United States, Harvard University Press, 1972), suggest a thyroid cancer incidence rate of $40/10^6$ persons based on state tumor registry data. Even if the ratio of fatal to occult cancers of 1 to 100 (ABCC Tech Report 25-68) is used and the incidence of $40/10^6$ thyroid cancers is considered fatal, the total incidence of thyroid cancer would be $4000/10^6$ persons. The relationship between these occult carcinomas and the total number of nodules has not been established yet, but some nodules are occult thyroid carcinomas. The nodules, as pointed out in Reference 42, represent malignant and benign tumors, but also nodular goiter, Hashimoto's thyroiditis, colloid diseases, local hyperplasia, local lymphnodes, etc.

14. Table VI-15 is somewhat misleading in that it apparently refers only to acute or subacute fatality and to "illness" in which thyroid should not be included since nodularity is not an "illness." The table does not include all effects, e.g. effects of pituitary injury or carcinogenesis, aspermia, etc.

15. References pertaining to in utero acute fatality and acute somatic injury are as follows: Evaluation for the Protection of the Public in Radiation Accidents; IAEA Safety Series # 21, IAEA Geneva (1967); Nokkentved, K. Effect of Diagnostic Radiation on the Human Fetus; Munksgaard, Copehagen (1968); Griem, M. L. The Effects of Radiation on the Fetus; Lying in: Journal of Reproductive Medicine 1:367-372 (1968); Hammer-Jacobsen, E. Therapeutic Abortion on Account of

X-ray Examination During Pregnancy; Danish Medical Bulletin. 6:113-122 (1959); Brent, R.L. and Gorson, R.O. Radiation Exposure in Pregnancy Current Problems in Radiology Vol. 215 (1972); Graham, S., Levin, M. L., Lilienfeld, A.M., Schuman, L.M, Gibson, R., Dowd, J.D., and Hempelmann, L. Preconception, Intrauterine, and Postnatal Irradiation as Related to Leukemia. pp. 347-371 in Epidemiological Approaches to the Study of Cancer and Other Chronic Diseases National Cancer Institute Monograph 19, NCI (1966).

16. There is also no indication that individual organ doses have been aggregated as "organ-rem" for summation in the estimate of "latent" cancers and genetic effects. Estimates of some isotopes and the distribution of organ doses and variations with age can be obtained from such publications as ICRP-17 (ICRP Publication #17, Protection of the Patient in Radionuclide Investigations, Pergamon Press. 1971).

17. Page 55, Section 6.7.3 - The use of ICRP-2 dose models, while defining what was done, does not seem adequate in light of advances in the field of physiology and dosimetry. As pointed out by Eve (I.S. Eve., "A Review of the Physiology of the Gastrointestinal Tract in Relation to Radiation Doses from Radioactive Materials," Health Physics 12:131-161, 1966) residence times and mass of contents for the GI tract used in ICRP-2 may be in error by factors of 2 or 3 in various segments and the values used for the stomach may be in error by a factor of 24 when residency time for inhaled material is being evaluated.

Dolphin and Eve (G.W. Dolphin and I.S. Eve, "Dosimetry of the Gastro-intestinal Tract", Health Physics, 12:163-172, 1966) suggest that differences of the order of a factor of 2 result, when a more sophisticated GI tract model is used rather than the ICRP-2 model.

Eve also made pertinent comments on the dose to the ovary from GI tract contents and the insensitivity of mucosal cells to radiation exposure at a depth of less than 140 microns.

The lack of information on particulate aerosol characteristics of the expected releases used in this section precludes applying the more accurate Task Group Lung Model or determining the extent of departure from the simple ICRP-2 model which would be expected. However, the current biological half-times for the various isotopes could be employed.

18. In the evaluation of damage from an accident, the health effects and dollar costs appear to be considered as mutually exclusive. This fails to consider the dollar costs of health effects. There is of course, the obvious cost of lost productivity but it is also noted, for instance, that thyroid nodules are passed off as being surgically treatable with no consideration as to the dollar cost of that treatment.

19. In Section 6.8.4, Non Core Accidents, Table VI-23 appears to over estimate the consequences by up to three orders of magnitude.

Editorial

1. In Section 6.4.4, for the phrase in parentheses, "vertical velocity toward the ground," substitute "ratio of the ground concentration to the integral over time of the adjacent air concentrations." This substitution will avoid furthering the false impression that the deposition velocity is indeed the vertical velocity toward the ground.

2. The PWR 7 category description on page II of Appendix VI needs a few more words of clarification, since the sprays do not act on the leakage occurring upward around the containment.

3. In the second paragraph on page 14 of Appendix VI, insert the word "acute" before the word "illness."

4. On page 14, the sentence "It was found, in particular, that the wind blew 0.1% of the time toward the 0.1% highest population density sector" needs clarification. The explanation on page 110 of the main volume is much clearer.

5. In Section 6.5.1, the reference to the isolated Idaho Falls site is of questionable interest, since Idaho Falls is not the site of any commercial nuclear power plant.

6. Table VI-6, on page 28 of Appendix VI, needs correction in that it shows, for categories 11 and 12, that the cumulative population decreases as the distance increases from 2 miles to 5 miles.

7. Page 32 - Experience with human radiation effects is not small and includes much more than Japanese data. The experience with acute effects is much less.

8. Page 35, Section 6.6.2 - The question of prophylaxis and adverse effects thereof is an open question. The fact that the treatment may be worse than the disease in some cases should also be considered.

9. Table VI-II, page 40 indicates up to 5% mortality at 165 rad (250 R) and a cutoff around 100 rad (150 R). Uncertainties in population response suggest that there must be a range around these values and that effects at lower exposure levels are possible.

10. Page 47, Section 6.6.4.2 - There is some confusion about the data studied by the BEIR Committee. Probably most of the data is on

relatively acute exposure to low LET radiation, the type most applicable to the emergency situation studied in the subject report.

11. In Section 6.8 of Appendix VI, the last sentence on page 67 implies that a Monte-Carlo type of determination was employed, as contrasted to the assertion in the second paragraph on page 3.

12. The title to figure VI-8 on page 76 should be changed since the thyroid nodules do not include all thyroid consequences to be expected.

<u>Appendix VII</u>

<u>Editorial</u>

1. Page C-2, item 2 - The core, taken as a whole, cannot "heatup" from sensible heat as stated here.

2. Page C-9 - The "Little Mamu" program should be referenced to supporting documentation.

3. Page I-2, equation (3) - Since this equation involves an integration over time, a distinction in the various time parameters is required since C_g is a function of "t".

<u>Appendix VIII</u>

<u>Clarifications</u>

1. Page A-3, 1st paragraph under <u>Fission-Product Release</u> - It appears that the pin rupture temperature was assumed to be 1500°F in the BOIL code calculations. This does not correspond to either of the two temperatures cited in Appendix VII.

2. Page A-12, last sentence under <u>Bottom Flooding</u> - The meaning of this sentence is not clear, particularly the reference to "these" flooding rates, and the reasoning that heatup of cores at elevated temperatures is not prevented.

<u>Editorial</u>

1. Page 6, top of page - Nomenclature problem: The ECR system described here appears to be the same as the LPRS system used in most of the rest of the Study documentation.

2. Page 7, last sentence - The starting time for CSIS is important. The fact that it must operate for a considerable length of time has nothing to do with start time considerations.

3. Page 8, 1st paragraph under Core Meltdown - It is not clear what is included in SIS failure (not previously defined).

4. Page 34, Accident Time Scale - A discussion similar to this for the PWR case would clarify the PWR containment discussion.

5. Page A-1, 1st paragraph under Core Heatup Calculations - In view of the application of the core heatup results to other PWRs and BWRs, the statement that some of the results apply only to the specific designs considered needs elaboration.

6. Page A-6, equation (A-9) - Q_{MELT} apparently should be Q_{QUENCH}.

7. Page E-9 - The pressures in this assessment should be labelled psig or psia, whichever is appropriate.

Appendix X

Clarifications

1. Page 6, first paragraph - Although the site geology is described, a description of what the plant is actually built on is not mentioned, as was done for the BWR on page 7.

2. Page 45, Note (4) - The Bijlaard formulae have not been defined in the text.

3. Page 94, first paragraph - It is indicated that the LHSIS (LPIS elsewhere) injects into the RCS hot legs. Figure II-53 of Appendix II, Vol. 2, shows injection into the cold legs and the text associated with the figure also indicates cold leg injection.

4. Page 94, third paragraph - The discharge pressure of 300 psig does not appear compatible with the 225-foot head stated on page 275 of Appendix II, Vol. 2.

5. Page 168, item 2 at bottom of page - This item states that the assumption of a 40° tilt of the MSIV actuator axis is a conservative assumption since "one expects vertical installation to be the usual practice." It is not clear why the actual orientation for the Surry plant was not determined in order to establish the validity of this conservatism. (Figure 28 shows an MSIV with about a 40° tilt to the actuator).

Editorial

1. The nomenclature used for the various reactor systems is not consistent with the rest of the Study. Examples are:

App A, Page 18 - Low Head Safety Injection System vs Low Pressure Injection Systems

High Head Safety Injection System vs High Pressure Injection Systems plus Accumulator Systems

Containment Recirculation Spray Systems vs Containment Spray Recirculation Systems

Core Spray Systems vs Core Spray Injection System

Residual Heat Removal Systems vs Post Accident Heat Removal.

Additional Comments

Summary Report

Editorial

1. Page 2, 1st sentence - The sources for the results in Figures 1, 2, & 3 should be identified, and the figures explained in more detail (ie, time period covered, population covered, etc).

2. Page 8, 1st paragraph, last sentence - Depending on schedules and definitions, this statement may be incorrect. Fort St. Vrain (330 MWe-HTGR) should start up this year, and Fulton 1 (1140 MWe-HTGR) is scheduled for startup in 1979.

3. Page 26, Section 2.21, 1st paragraph - A more effective qualification of the WASH-740 results would be to quote the cover letter transmitting the Study to the JCAE in March 1959. This letter, presumably written by the authors of the report, says, in part:

"Pessimistic values, leading to great hazards, were chosen for the numerical values of many uncertain factors which influence the final magnitude of the resulting damage. It can therefore be concluded that these theoretical estimates are greater than the damages which would actually result in the unlikely event of such an accident."

CONTRACT WITH INTERMOUNTAIN TECHNOLOGIES, INC.

CONTINUING WASH-1400 REVIEW TASKS

A. Failure Mode Paths Selected for Review

1.	BWR-Reactor Protection System	Review to determine credit taken for backup Boron injection under BWR transients selected following investigation of BWR Reactor Protection System.
2.	BWR-Transient #1	
3.	BWR-Transient #2	
4.	BWR-Transient #3	
5.	PWR-Electric Power Systems	Independent evaluation of Electric Power System Availability-
6.	PWR-High Pressure Injection System	Review to determine the extent that possible troublesome break locations have been accounted for.
7.	PWR-Small Break #1	
8.	PWR-Small Break #2	
9.	PWR-Loss of Power Transient	Review relationships considered between this accident sequence and specific containment failure modes.
10.	PWR-Low Pressure Injection System	Review to determine range of applicability of assumed failure paths and sensitivity of results on accident risk magnitude.
11.	PWR-Low Pressure Recirculation System	

B. Critical Radiological Source Term Parameters Selected for Review

1. Core conditions prior to meltdown calculation

    a. vessel residual water

    b. blowdown heat-transfer

    c. blowdown duration

2. Core meltdown calculation

3. Containment response

    a. failure pressure

    b. containment safeguards and containment pressure response.