Wood-Leaver and Associates, Inc.

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RESPONSES TO BNL QUESTIONS ON THE RELEASE AND CONSEQUENCE ANALYSIS REPORTED IN THE BIG ROCK POINT PRA

Prepared For

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1.0 INTRODUCTION

The purpose of this document is to respond to questions and comments raised in the Brookhaven National Laboratory (BNL) critique of the Big Rock Point (BRP) Probabilistic Risk Assessment (PRA). These questions and comments are contained in Appendix A of the draft review report of the BRP PRA issued in January of 1982 by EG&G-Idaho. A summary of the comments is contained in Section 9 of that report.

The approach taken in this document has been to address each BNL comment separately and to assess its relevance to both the PRA report and the conclusions of that report. The conclusion of this work is essentially the same as that stated in Section 9 of the review report; that is, "although this (the release and consequence analysis) portion of the PRA would be more complete by including the items discussed above (in Section 9), the overall results of the consequence analysis would not be significantly affected."

The BNL comments are addressed in Sections 3 of this report. Attachment A is a letter summarizing our position regarding BNL comments on hydrogen combustion and basemat penetration. The overall basis for the conclusions on the effect of these comments is presented in Section 2.

2.0 SUMMARY EVALUATION OF BNL COMMENTS

The BNL comments have focused on two areas: containment failure mode analysis and health consequence analysis. Some introductory remarks are appropriate before summarizing the effect of these comments on the results of the PRA. The first remark is on the nature of the health consequences resulting from severe accident sequences at BRP. The analyses of health effects performed for the five release categories demonstrated that acute fatalities in the public at large will not occur under any but the most adverse environmental conditions combined with the most severe release assumptions and an absence of evacuation. This observation has been used to respond to several of the comments made by BNL in Section 3 of Appendix A of the review report.

The second remark is on the effect of the various release categories on the predicted CCDF of latent fatalities for BRP. Analysis of both the probabilities and consequences of the release categories has revealed that only BRP-3 contributes significantly to the risk from BRP. Other categories are either too improbable or have too small a set of associated health consequences to be important. This observation was also utilized in the responses to BNL comments contained below and in Section 3 of this report.

2.1 CONTAINMENT FAILURE MODE ANALYSIS

The most serious of the questions raised by BNL were on the containment failure mode analysis. This section will address the two failure modes (hydrogen burning and basemat penetration) which were the focus of BNL questions.

2.1.1 Hydrogen Burning

Analyses reported in Chapter 5 of the BRP PRA main report and in Appendix IV of the same report have suggested that containment failure by hydrogen burning is impossible. This conclusion was questioned by BNL. Therefore, a simple analysis was performed to assess the importance of the hydrogen combustion failure mode should it be possible. It should be noted that the analysis presented in Attachment A supports the conclusions stated in the PRA report that insufficient hydrogen can be produced in BRP to lead to containment failure caused by hydrogen combustion. Factors which contribute to assignment of an extremely low probability for this failure mode include:

a) Containment failure pressures can only be reached if rapid burning of hydrogen equivalent to that which would result from oxidation of 130% of all the in-core zirconium is assumed. It is also necessary that essentially no suspended water be present in the containment atmosphere at the time when burning occurs. Analysis of containment pressure under this condition is presented in the BNL comments.

b) As discussed in Note 14 on page 90 of the BRP PRA main report, basemat penetration by molten fuel which might increase the amount of hydrogen present in containment atmosphere is not possible unless the primary system blows down outside containment, in which case the releases are as severe as those expected to result from hydrogen burning and containment failure.

c) The BRP containment is sufficiently large and the primary system water inventory is sufficiently small such that no steam inerting of the containment is possible.

d) The analysis in Attachment A supports the conclusion that insufficient hydrogen will be produced to cause containment failure by rapid combustion.

Given these factors, the possibility of containment failure by hydrogen burning is considered to be exceedingly low. For the sake of argument, this probability has been conservatively judged to be 0.01 given an accident sequence which produces significant core damage. This value is sufficiently high as to be above argument. The radionuclide releases associated with containment failure by hydrogen burn at BRP would be expected to be on the order of the release fractions for a PWR given the same containment failure mode. These were defined in the Reactor Safety Study to be those associated with the PWR-1 release category. These release fractions are shown on Table 2.1 together with a number of releases associated with conditions which were analyzed for BRP. As shown, for nearly all isotope groups, the radionuclide release fractions for BRP-1 are equal to or higher than those for the PWR-1 release category. Therefore, for purposes of analysis, the BRP-1 release category can be used to depict the effect of hypothesized hydrogen burning leading to containment failure.

The total probability of all release categories except those associated with containment isolation failure (BRP-1 and BRP-3) is approximately 6.0×10^{-4} per year. Therefore, the estimated probability of releases resulting from containment failure by hydrogen burning is approximately 6.0×10^{-6} per year. Table 2.2 shows the effect of this probability being associated with release category BRP-1 on the predicted latent fatality CCDF for BRP. This table was developed using data reported in Table 6.2 of the BRP PRA main report. As shown, the CCDF is not significantly affected by the incorporation of a release category which depicts the hydrogen combustion failure mode.

2.1.2 Basemat Penetration Failure Mode

In the Reactor Safety Study, the basemat penetration containment failure mode was depicted by release categories PWR-6 and PWR-7. The releases associated with these categories together with the releases for release category BRP-5 are shown in Table 2.3. As shown in this comparison, the radionuclide release fraction estimated for category BRP-5 (with a probability equal to 5.9 x 10^{-4} per year) are between those for the two PWR release categories in which containment failure occurred by basemat penetration. Because of the long time required for basemat penetration by molten fuel in BRP (if it were to occur), the releases would be expected to be closer to PWR-7. Therefore, it can be argued that release category BRP-5 is, because of inherent conservatisms in the way in which it was defined, the equivalent both in probability and in severity of releases of a release category which might characterize the effect of basemat penetration. Again, the analysis reported in Attachment A supports the conclusion that molten fuel will not penetrate the basemat.

It should also be noted that, as in the Reactor Safety Study, the contribution of release category BRP-5 to overall risk is insignificant.

TABLE 2.1 COMPARISON OF RADIONUCLIDE RELEASES FOR PWR-1 WITH VARIOUS BRP RELEASE CATEGORIES

FRACTION OF CORE INVENTORY RELEASED

RELEASE CATEGORY DESIGNATION	Xe-Kr	I org.	I_2-Br	Cs-Rb	Te	Ba-Sr	Ru	La
BRP-1	9.0×10^{-1}	7.0×10^{-3}	9.0×10^{-1}	8.1×10^{-1}	1.5×10^{-1}	1.0×10^{-1}	3.0×10^{-2}	3.0×10^{-3}
BRP-3	8.9×10^{-1}	6.9×10^{-3}	8.3×10^{-2}	3.0×10^{-1}	1.4×10^{-1}	3.9×10^{-2}	1.2×10^{-1}	1.7×10^{-3}
PWR-1	9.0×10^{-1}	6.0×10^{-3}	7.0×10^{-1}	4.0×10^{-1}	4.0×10^{-1}	5.0×10^{-2}	4.0×10^{-1}	3.0×10^{-3}
PWR-2	9.0×10^{-1}	7.0×10^{-3}	7.0×10^{-1}	5.0×10^{-1}	3.0×10^{-1}	6.0×10^{-2}	2.0×10^{-2}	4.0×10^{-3}

TABLE 2.2 EFFECT OF THE ADDITION OF A RELEASE CATEGORY WHICH DEPICTS CONTAINMENT FAILURE BY HYDROGEN BURNING ON THE BRP LATENT FATALITY CCDF

MAGNITUDE OF LATENT FATALITIES	PROBABILITY OF EFFECT BEING CAUSED BY RELEASE CATEGORY		PERCENTAGE INCREASE RESULTING FROM HYDROGEN BURN RELEASE CATEGORY
	BRP - 3 (1)	BRP - 1 (2)	
1.0	3.62×10^{-4}	5.93×10^{-6}	1.6
2.0	3.54×10^{-4}	5.88×10^{-6}	1.7
3.0	3.44×10^{-4}	5.78×10^{-6}	1.7
5.0	3.23×10^{-4}	5.56×10^{-6}	1.7
7.0	3.09×10^{-4}	5.32×10^{-6}	1.7
10.0	2.80×10^{-4}	5.02×10^{-6}	1.8
20.0	2.07×10^{-4}	4.21×10^{-6}	2.0
30.0	1.65×10^{-4}	3.50×10^{-6}	2.1
50.0	1.16×10^{-4}	2.77×10^{-6}	2.4
70.0	8.73×10^{-5}	2.26×10^{-6}	2.6
100.0	6.85×10^{-5}	1.72×10^{-6}	2.5
200.0	2.79×10^{-5}	9.90×10^{-7}	3.5
300.0	1.05×10^{-5}	5.88×10^{-7}	5.6
500.0	1.12×10^{-6}	2.04×10^{-7}	18.2
700.0	0	4.14×10^{-8}	-
1000.0	0	0	-

(1) From the BRP PRA, the probability of this dominant release category is 3.7×10^{-4}.

(2) From the analysis in this report, the probability of this release category being caused by hydrogen burning is 6.0×10^{-6}.

TABLE 2.3 COMPARISON OF RADIONUCLIDE RELEASES FOR RELEASE CATEGORIES BRP-5, PWR-6, and PWR-7

FRACTION OF CORE INVENTORY RELEASED

RELEASE CATEGORY DESIGNATION	Xe-Kr	I org.	I_2-Br	Cs-Rb	Te	Ba-Sr	Ru	La
BRP-5	0.9	2.7×10^{-4}	4.0×10^{-4}	4.5×10^{-4}	2.6×10^{-4}	5.7×10^{-5}	1.5×10^{-4}	3.4×10^{-6}
PWR-6	3×10^{-1}	2×10^{-3}	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR-7	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}

2.2 RADIONUCLIDE RELEASE AND CONSEQUENCE ANALYSIS

All of the BNL comments on the BRP release and consequence analysis contained in Section 3 of Appendix A of the EG&G-Idaho review report have been addressed in Section 3 of this report. As in Section 2.1 of this report, the comments and their responses do not affect the results or the insights gained in the Big Rock Point PRA.

3.0 RESPONSE TO QUESTIONS ON RADIONUCLIDE TRANSPORT AND CONSEQUENCE ANALYSIS

This section addresses comments made in Section 3 of Appendix A of the EG&G-Idaho review of the Big Rock Point PRA. Each response refers to the section in the review report which it addresses. For convenience both the question and the response are included.

3.1 COMMENTS ON INPUT CALCULATIONAL TECHNIQUES AND OUTPUTS

COMMENT 3.1.1 In carrying out an analysis of the radionuclide transport within containment and the consequences suffered within the environment, the CORRAL and CRAC codes are respectively used. These codes are presumably used as part of the overall RACAP package which also includes all the phenomenology preceding the CORRAL calculation. Although the connection between CORRAL and CRAC is automatically taken care of within RECAP, some explanation as to how this is done is required since some assumptions are implicit in this step. The primary assumption deals with the mode of release of fission products from the containment building to the environment. The CORRAL code computes a continuous release of fission products, while the CRAC code can only handle a simple puff release. The conversion of a continuous release in time, to a single puff implies an approximation which warrants further discussion. Besides the fission product release fractions, the character of release and energy of the release are affected by the approximations made in the above-mentioned conversion. A further parameter which is required as input to CRAC, and which is not clearly defined in the report, is the height of the release.

RESPONSE 3.1.1

At the time the BRP PRA was carried out, transfer of CORRAL output to CRAC input was not automated but was done by hand. From a graph of activity release as a function of time for the eight isotope groups representative values of release time and release duration were chosen for each CORRAL case. These release curves are shown in Appendix V of the PRA report, while the release times and durations are shown in Table 5.4 of the main report and in Table 3.1 of this response.

- o The CRAC code assumes a release duration of 30 minutes. For release durations greater than 30 minutes a cloud expansion factor is used to account for the additional expansion during the release period. This issue is addressed further in the response to comment 3.1.5.

- o The energy of the release, which affects the buoyancy of the cloud, is input data to the LEAKAGE subgroup in the CRAC code. These data can be obtained from the CRAC runs on file at CPCo. A summary of these energy releases is reported in Table 3.1.

- o The height of release is also input data to the LEAKAGE subgroup and can be obtained from the CRAC computer runs on file at CPCo. This information is also summarized in Table 3.1.

- o It should be noted that because of its high probability and high consequences relative to other release categories, the BRP-3 release category is the only one which influences the BRP consequence curves.

- Finally, because of uncertainties in parameters which had the potential to affect calculated health consequences, a number of sensitivity studies were performed as part of the study. Parameters studied included:

  a) Evacuation model;
  b) Release energy;
  c) Containment isolation modification.

  These sensitivity studies, which are reported in Section 6.3 of the main report, have concluded that the resultant health consequences are insensitive to evacuation rate and energy release accompanying the release of radionuclides.

COMMENT 3.1.2 The initial isotope inventories were determined using the ORIGEN code which requires a neutron spectrum in energy as input. Since, in a BWR the spectrum can vary depending on the coolant void fraction, what method was used to determine a spectrum which was valid both as a function of void fraction and burnup? A discussion of this method should be included. This point is particularly important since the claim is made that the sharp drop off of the Complementary Cumulation Distribution Function (CCDF) for latent effects, relative to that for the Surry plant is because of the greatly reduced fission product inventory. A clear understanding of the basis for this inventory is, thus, necessary.

RESPONSE 3.1.2

Typically neutron spectral effects on fission product inventory lead to a small variation. These variations are not significant in the calculation of public health risk where they would be obscured by other uncertainties. The principal point is that the BRP inventory is much less than the Surry inventory because the BRP reactor operates at 240 MWth and Surry at 2364 MWth. Thus, the BRP fission product inventory would be about one-tenth that of Surry.

COMMENT 3.1.3 Population data used in the CRAC code for the surrounding area should be based on a projection of what it will be at some time in the future. This time is customarily taken as the mid-point between the present time and the expected end of plant life. This should be used rather than the data based on a 1979 census.

RESPONSE 3.1.3

The population data used for BRP were based on 1970 census information. An estimate of the projected population to the year 1991 (mid-point between present and end of plant life) can be made by using the estimated growth rate value of 9% per decade used in WASH-1400, Appendix VI. Thus, from 1970 to 1991 the population could be expected to be about 20% greater than that used in the BRP analysis. This is a small change well within the error bounds of the overall analysis and would not noticeably change the results. Further, the population in the state of Michigan is decreasing at present, and projections over the next twenty years would not be expected to be accurate and might not even reflect the correct trend. Finally, since no acute fatalities were predicted in the analysis of the BRP release categories, local population and its variations with time or season would not affect the overall study results.

COMMENT 3.1.4 When considering the sequence 11 fission product release category (release of noble gases to turbine building available via tortuous routes) volatile fission products such as organic iodine should have been included. In all likelihood, volatile compounds such as organic iodine would be able to follow noble gases out of the containment building, regardless of how tortuous the path. It might also be possible for small quantities of tellurium oxide and ruthenium oxide to escape the same way.

RESPONSE 3.1.4

Release sequence 11 was a non-mechanistic release of noble gases only. The use made of this sequence was to increase the quantity of noble gases in release category BRP-5 from a small calculated value to 90%. Since this category had an insignificant effect on risk, the precise characteristics of sequence 11 are not expected to influence the overall results.

COMMENT 3.1.5 Release sequences 3, 4, 5, 6, 7 and 8 all have comparatively long release duration times. This time is used in CRAC to account for the plume expansion and is merely an adjustment to the puff release model to account for lack of a more realistic model. It has been suggested that this time not exceed 8 hours, since beyond that time the adjustment becomes inappropriate. All the above-mentioned sequences have release duration times longer than 8 hours. They should be redefined with the smaller release duration time in mind.

RESPONSE 3.1.5

The release duration is used to calculate the cloud expansion factor which is a function of release duration to the one-third power (i.e., not strongly sensitive to changes in release duration). Release sequences 3 through 7 have a maximum release duration of 19 hours. The calculated cloud expansion factor for 8 hours is 2.52, for 19 hours it is 3.36, i.e., about a 30% difference. Spreading the cloud out 30% more may reduce the cloud concentration by 30% but it will cover a 30% larger area and therefore affect a greater number of people. The overall effect being no significant difference in total man-rem exposure. Since latent fatalities are proportional to man-rem exposure no differences in study results are expected. Release sequence 8 had a release duration of 228 hours. The expansion factor for 228 hours is a factor of 3 greater than that for 8 hours. This sequence represents a significant contribution to the BRP-5 release category. However, in assigning a release duration to BRP-5, the characteristics of other contributing sequenced determined that the release duration should be 13.5 hours (see Table 3.1 in this report). Thus, no significant effect is expected. Furthermore, the BRP-3 release category, which totally dominates predicted risk from the.plant, has a release duration of 8 hours, which is consistent with the guidelines given for use of the CRAC code.

COMMENT 3.1.6 The eleven release categories are further reduced down to five categories which are used as input to CRAC. From Table 5.5 of Reference 1 describing the five categories (BRP1-BRP5) it is not clear what the time, duration and height of release are for these composite categories. Furthermore, it is not clear how these quantities are combined to give the values used in the composite releases. How are release sequences 1, 3, 6, 8, 11 combined to give BRP5? For these sequences, the release times vary from one hour to 12 hours, the duration varies from 4 hours to 228 hours and the heights are not specified.

RESPONSE 3.1.6

Table 3.1 in this report provides the release times, warning times, release heights, and release energies for the five composite release categories. Comparison of this Table with the information presented in Table 5.4 of the main report indicates that for composite release categories BRP-3, the shortest warning time and release duration were selected from the composite cases. For release category BRP-5, the shortest warning time and a mid-range release duration were selected. Sensitivity studies on evacuation time (see BRP PRA main report Section 6.3) and the discussion in response to BNL comment 3.1.5 presented above lead to the conclusion that overall results of the consequence analysis are not significantly affected by the specific values of these parameters selected.

COMMENT 3.1.7 The particle deposition velocity is one of the factors which determines how rapidly the aerosols deposit on the ground. It is not clear what value was used in this analysis.

RESPONSE 3.1.7

The particle deposition velocity is input data to the ISOTOPE Subgroup of the CRAC code and is available in the CRAC runs of file with CPCo. Values used are 10^{-2} meters per second for solid fission products and 0 for noble gases.

COMMENT 3.1.8 In presenting the results only the CCDF for latent effects are considered. In other Probabilistic Risk Assessments, other health effects such as thyroid cancer, acute fatalities, injuries (excluding cancer), cancer fatalities (other than those from thyroid cancer) and whole body man-rem are included. Furthermore, contrary to the assumptions made in the document, it is felt that economic consequences should also be included. The latter category should include the standard quantities computed in CRAC, the cost of replacement power, and the cost of the plant.

RESPONSE 3.1.8

It was the judgement of Consumers Power Company that the objectives of the Big Rock Point PRA could be best satisfied by focusing attentions on two representative measures of accident consequences: acute fatalities (a threshold phenomenon) and latent fatalities (a continuous phenomenon). These measures provided an adequate basis of comparison between BRP risks and those associated with other nuclear plants. In the absence of some criterion against which to compare the various measures of accident consequences potentially caused by BRP, it was felt that calculations of these measures would serve no useful purpose.

3.2 ADDITIONAL DISCUSSION

In keeping with discussions held with representatives of the NRC and BNL, comments made in Section 3.2 of Appendix A of the review report are addressed here to the degree supportable by the existing information base.

COMMENT 3.2.1 In certain accident scenarios, it was assumed that the containment is slowly pressurized, over the period of several days, and that for these cases primarily noble gases and organic iodine escape, the remaining fission products having been either removed (by sprays) or settled (aerosols). However, the possibility exists that during the failure of the building, the aerosols could be re-suspended and emitted from the building as part of the plume. This point should be addressed and an explanation given as to why it is unlikely.

RESPONSE 3.2.1

The majority of the fission products not released will have been removed by sprays or settle onto surfaces wetted by sprays and condensing steam. The resuspension factor for particles either in pools of water or on wetted surfaces is on the order of 10^{-6} or lower. Except for the Lanthanum group, this would only be a small fraction of the initial releases of the various radionuclide groups. The fraction of the Lanthanum group resuspended could be on the order of that for the initial release, however, this does not represent a major contributor to the overall dose consequences.

COMMENT 3.2.2 In order to compensate for lack of understanding of the physical phenomena, a sensitivity study should be carried to determine the sensitivity of the consequences of interest to changes in the fission product release fraction. This study would answer questions raised in Section 2 (Review Report, Appendix A) related to the appropriateness of the assumed containment failure modes. It would also indicate to which fission product group the consequence was particularly sensitive.

RESPONSE 3.2.2

The release fractions for release categories BRP-1, BRP-2, and BRP-3 are not significantly different from the WASH-1400 category PWR-2. WASH-1400 discusses the contributions of the various fission product groups to the overall consequences. Repeating sensitivity studies already reported in WASH-1400 for similar releases would probably not lead to results significantly different from those in WASH-1400 and therefore this analysis is not expected to produce additional information of value.

COMMENT 3.2.3 At some point in the report there should be a discussion of the effect of a steam/aerosol mixture leaving the containment. Upon leaving, the steam will condense forming a water droplet aerosol, which will interact with the fission product bearing aerosol. The possibility of this occurring and its consequences on the plume dynamics should be discussed.

RESPONSE 3.2.3

At present, no data on this postulated phenomenon and its effect on plume dynamics have been located. Sandia, where most of the NRC funded consequence sensitivity studies have been performed, was not aware of work in this area. The occurrence of this phenomenon would be expected to affect the behavior of the plume very close to the plant, and therefore, have an insignificant or reducing affect on the health consequences to the public at large. This hypothesis is reinforced by the observation that, except as noted in Section 2.0, no acute fatalities to the surrounding population have been predicted for the BRP release categories. The principal implications of the postulated phenomenon would be to the habitability of the site in the region influenced by the plume, and to the economic consequences of accidents postulated to produce severe radionuclide releases from the containment. It should be noted that releases from the containment will travel to the atmosphere either up the stack, to the pipe tunnel and out the blowout panel, out the radwaste vents, or to the turbine building and then either out the louvers or out the stack. These paths are either long with relatively cold surfaces, include large volumes/low pressure drops to the outside (turbine building), or follow torturous paths (to radwaste vaults or electrical penetration). All of these factors will tend to condense the steam, scrub the particulates and allow for deposition. This deposition will tend to increase doses inside the plant, which CPCo has considered, and minimize the contamination outside the plant. The contamination outside the plant is only a concern from a decontamination standpoint. Most of the personnel activity after the release is inside the plant with the exception of ingress and egress, which can be accomplished rapidly with the small staff which will be present.

COMMENT 3.2.4 In carrying out the calculation of fission product behavior within the containment building, the CORRAL code is used. This code does not allow for the effect of radioactive decay. This omission does not allow for the formation of daughter products which might be in different fission product groups and which are thus treated differently. A discussion of this effect and its implications should be included at the appropriate point within the report.

RESPONSE 3.2.4

Daughters that would have formed prior to release time are generated in the CRAC code and become part of the inventory released to the atmosphere. This means however, that some parent nuclides with a high decontamination factor are calculated to be removed from the source prior to release when in reality they would have decayed into daughter nuclides with a low decontamination factor and be available for release at the time of containment failure. On the other hand, some parent nuclides with a low decontamination factor are calculated to be available for release at the time of containment failure, then the CRAC code adjusts these to account for daughters which have formed prior to containment failure. Thus, some daughters with high decontamination factors are included in the inventory of released products to the atmosphere when in reality they would have formed prior to containment failure and been removed from the inventory.

The current method of handling radioactive decay will underestimate the activity release of daughter nuclides with low decontamination factors and will overestimate the activity release of daughter nuclides with high decontamination factors. Because there are more daughter nuclides, from either type parent nuclide, that have high decontamination factors, the overall release of activity is expected to be conservatively overestimated.

COMMENT 3.2.5 The deposition of the released fission products on large bodies of water such as the Great Lakes has not been addressed. Health and economic consequences of depositing all the aerosols and halogens on the surface of a lake would be significantly different from other consequences which have been discussed to date. This difference in consequences warrants further discussion of this mechanism for environmental contamination.

RESPONSE 3.2.5

An estimate of consequences of fission product deposition on the Great Lakes could be made by using work done by Sandia: NUREG/CR-1596, "Consequences from Liquid Pathways After a Reactor Meltdown Accident", June 1981. This report looks at releases via groundwater to the Great Lakes and its consequences. CRAC models milk and food chain consequences of ground deposited activity but not water deposited activity. However, since existing analysis of the health consequences associated with contamination of the food chain have indicated that the consequences associated with this path are much less significant than those associated with airborn releases from containment, Consumers Power Company considers the expenditure of the necessary analytical effort to be unjustified. Furthermore, in the event that meteorological conditions during and after an accident lead to the expectation that deposition on Lake Michigan might result, a thorough monitoring program would likely be put in place to assure that the effect of any such deposition on public health is insignificant. It should also be noted that a release over Lake Michigan will not lead to whole body exposures due to noble gases. The path of the release to the atmosphere will tend to scrub out particulates and some semi-volatile isotopes. The dilution of the nuclides deposited on the lake is very large (800 for releases to the discharge canal and much greater for deposition further out on the lake). Since the nearest intakes for drinking water are at Charlevoix which is 4.6 miles away, dilution effects will be much greater. Also, ground water would not be affected since the gradiant is toward the lake.

COMMENT 3.2.6 The formation and retention of CsI has received a great deal of attention in the literature. This issue should be addressed. In addition to CsI, the oxidation of tellurium and ruthenium during such an event should be discussed. In addition to these points, the likely dose-response characteristics of such compounds should be discussed.

RESPONSE 3.2.6

NUREG-0771, "Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions", June 1981, discusses the impact of cesium iodide vs. elemental iodine and concludes there is no significant difference in accident consequences due to different chemical forms released. Since this is a field which is evolving rapidly at present, Consumers Power Company considered any expenditure of effort to be non-productive at this time.

COMMENT 3.2.7 Some consequence codes (e.g., CRAC-IT) claim to be able to model the effects of the surrounding terrain and variable directional plumes. A discussion of how important these effects are for the site of interest, and what error is incurred by not including them should be included.

RESPONSE 3.2.7

The CRAC-2 code with a unidirectional plume model has been compared with the CRAC-IT code with a variable direction plume model. The comparison study was for the Indian Point site. The results show no significant difference.

Because of the flat terrain in the area of the BRP site, the wind direction is likely to be more persistent over greater distances making the need for a variable plume model of less importance.

The variable plume model would be of more benefit in tracking a plume near a site where local terrain leads to changes in plume direction and where wind data from multiple towers are available.

COMMENT 3.2.8 A limited sensitivity analysis in addition to the one carried out and reported in the document (energy of plume, evacuation model), and the one suggested above regarding fission product release categories, should be included. Such a study would be useful in identifying parameters which affect the consequences most severely. Such a study might also be useful in formulating mitigating actions to reduce the consequences of an accident.

RESPONSE 3.2.8

Had the BRP PRA produced results which showed a significant number of acute fatalities for several types of accident sequences, a sensitivity study such as that suggested above would certainly be justified. However, no acute fatalities were predicted for the BRP release categories. Therefore, CPCo effort has been focused on analysis which better characterizes the environment on site and the effect of this environment on plant staff.

COMMENT 3.2.9 The failure of the basemat due to a core melt-through has been omitted based on mechanistic grounds. However, in the event that such an event should occur, methods of interdicting the contamination should be discussed. The discussion should include the rock formations into which the contamination would deposit itself and the possible ground water pathways available for transporting it.

RESPONSE 3.2.9

Because of the much lower than average power level of the BRP core, it was shown that basemat penetration was not likely to occur. Even in the unlikely event that it did occur, WASH-1400 results for larger cores shows groundwater pathways not to be a significant part of the overall accident consequences. Therefore, CPCo considers the suggested analysis to be unnecessary to satisfy the objectives of the PRA. Should a severe accident occur at BRP, time would be available for analysis and planning to support interdiction activities.

TABLE 3.1 SUMMARY OF RADIONUCLIDE RELEASE PARAMETERS

RELEASE CATEGORY	SEQUENCE NUMBERS INCLUDED(1)	WARNING TIME (HRS)	RELEASE DURATION (HRS)	RELEASE HEIGHT (FT)	RELEASE ENERGY (CAL/SEC)
BRP-1	5	1.0	10.5	0	0
BRP-2	10	0	0.667	0	0
BRP-3	2, 4, 7	1.0	8.0	0	0
BRP-4	9	240	0	0	0
BRP-5	1,3,6,8,11	1.0	13.5	0	0

(1) See Table 5.4 of the BRP PRA Main Report.

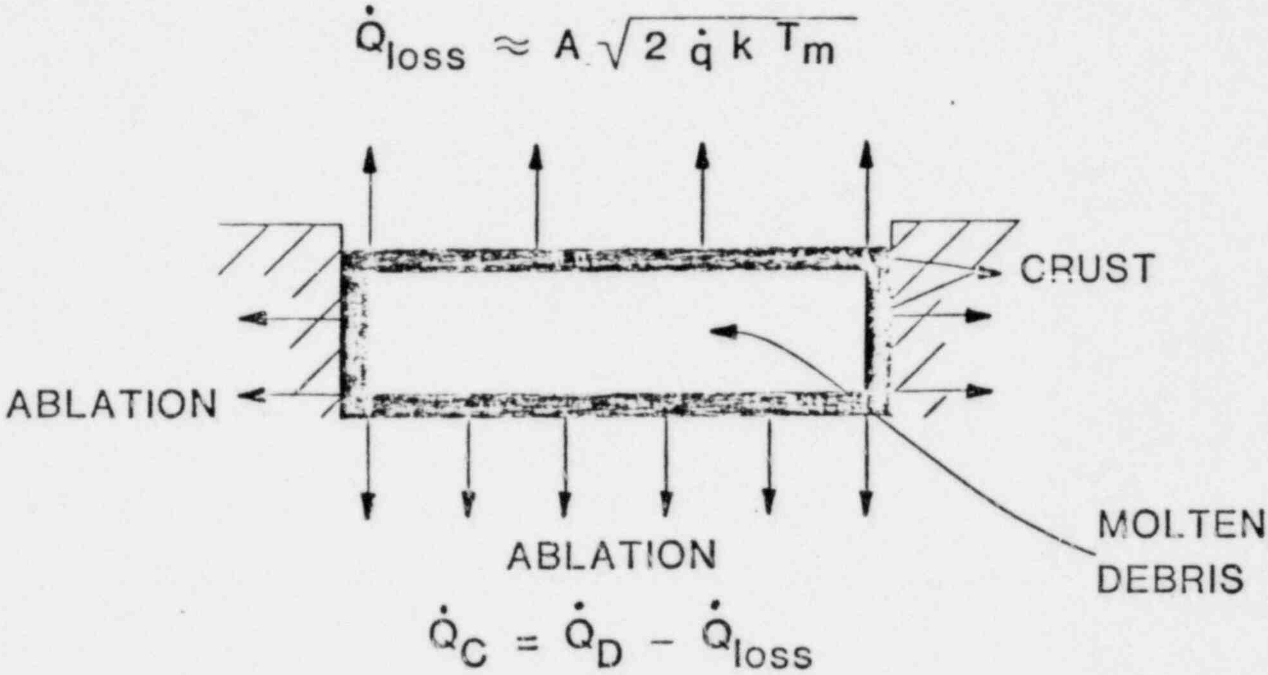


Fig. 1

MODEL FOR CONCRETE THERMAL ATTACK

ATTACHMENT A

BIG ROCK POINT

Evaluation of Concrete Attack

To determine the influence and overall significance of establishing a permanently coolable debris bed in the Big Rock Point system given a severe core damage accident with material released from the reactor pressure vessel, the following analysis has been carried out to determine the extent of concrete attack for various fractions of the total core material. In this evaluation, it is assumed that this fraction of core material is accumulated within the valve pit and initially attacks the concrete as a result of the initial liquid superheat within the melt and the long range quasi-steady concrete attack resulting from the decay power and reaction heat within the melt. A configuration like that shown in Fig. 1 was assumed and the concrete attack was divided into two different segments. The first is that initially rapid attack due to the molten material when the core debris is substantially above its freezing temperature. This was modeled as an instantaneous attack that was only limited by the sensible energy within the melt. In this evaluation, the reaction heat was neglected for simplicity, but would not substantially alter the conclusions of the overall analysis. The pertinent energy balance is given by

$$\Delta x = \frac{m_F c_F (T_F - T_{F,m})}{\rho_c [c_c (T_{c,m} - T_c) + L] A_1}$$

where Δx is the amount of concrete attack, m_F, c_F, T_F, and $T_{F,m}$ are the mass, specific heat, initial temperature, and melting point of the degraded core material. In addition, ρ_c, c_c, T_c, $T_{c,m}$, and L are the density, specific heat, initial temperature, melting temperature, and effective latent heat for the concrete substrate. Also, A_1 is the effective area of this initial attack and is the combination of the horizontal area of the bottom of the pit and the attack area on the sides. This area is evaluated as

$$A1 = \pi r_c^2 + 2\pi r_c h$$

where r_c and h are the radius of the cavity and the boilup height of the fluid pool. For these parametric studies, the boilup pool was assumed to have an internal void fraction determined by the Zuber-Findlay correlation

$$U = 1.53 \sqrt[4]{\frac{g\sigma}{\rho_f}} \frac{\alpha}{1 - \alpha}$$

where U is the superficial velocity of gas through the pool. Therefore, the pool height can be evaluated by

$$h = \frac{V_F}{(1 - \alpha)\pi r_c^2}$$

where V_F is the volume of fuel and concrete ablated assumed within the valve pit.

Carrying out the calculations for the initial concrete attack due to the superheat within the melt, results in typically a few cm of concrete ablation. Once the superheat has been removed from the material and solidification begins, simple conduction analyses show that debris cannot solidify in such a confined configuration since the internal energy generation cannot be conducted to the material boundaries with the total debris bed in a solid state. This analysis only applies to conditions with no water added to the valve pit. Consequently, as the core material attempts to solidify, its inability to conduct the energy generated to the boundary requires that the central region of the material remain sufficiently fluid to provide the necessary circulation for achieving the energy transfer to the boundary. At the boundary itself, substantial crusts of debris can be formed in direct contact with the concrete and a crust over the top of the debris can also be produced as a result of the upward heat losses. To analyze this combined cooling and ablation process, a configuration like that illustrated in Fig. 1 is assumed. As the material approaches its freezing, the convective heat removal from the pool to the crust becomes minimal, and the upward heat loss by the upper crust can be approximated by that energy generated within the crust alone. For the sake of this simple parametric analysis, the upward heat loss is approximated by a conduction equation with internal heat generation assuming that the outer temperature of the crust is negligible compared to the melting temperature of the debris. This can be expressed by the relationship

$$\dot{Q}_{loss} = A \sqrt{2\dot{q}kT_{F,m}}$$

where the volumetric heat generation rate ($\dot{q}$) is evaluated by dividing 2/3 of the decay power at a specific time by the volume of the fuel (V_F). The 2/3 value is used to represent the loss of noble gases and volatiles from the fuel by the time that the material has melted and been released from the reactor pressure vessel.

Ablation velocities (U) can be evaluated from the expression

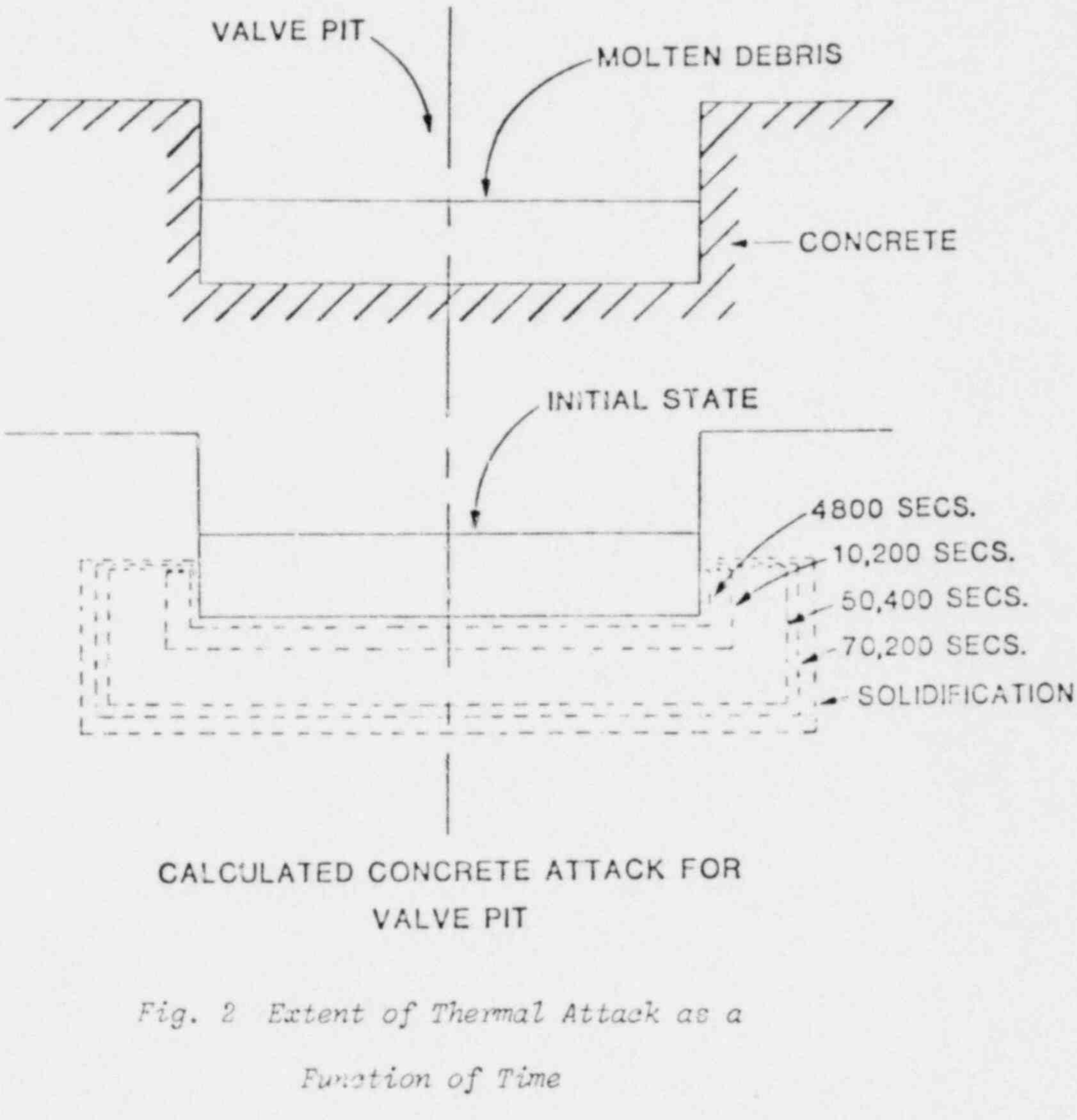
$$U = \frac{q/A)_c}{\rho_c[c_c(T_{c,m} - T_c) + L]}$$

where the heat flux to the concrete $[q/A)_c]$ is determined by subtracting the upward heat loss from the decay power generated in the debris plus the reaction heat liberated as a result of steam flow through the molten pool and reaction with metallic components contained within the system. For these parametric calculations, the reaction heat within the melt was limited by the rate of steam released from the concrete and that release fraction was assumed to completely react with molten species within the melt, the zirconium oxidation reaction was used (heat of formation 600,000 kJ/kg-mole).

The concrete attack was calculated to occur on the side walls and the lower horizontal surface with the upward losses being governed by the conduction through a heat generating media and the release of high temperature gas bubbling through the melt. As the concrete was ablated, the molten concrete products were added to the volume of the debris and the volumetric energy generation rate was reduced correspondingly. This principally affects the upward heat loss from the debris pool since the crust becomes thicker with a smaller energy generation rate.

Calculations for the penetration of the molten debris into the valve pit were carried out assuming that half of the core material was released into the valve pit and initiated the attack with the remaining half left on the floor of the control rod drive room as a result of deflection off of various pipes, valves, pumps, etc. in this CRD room. In the analysis, the additional heat sinks and heat transport paths provided by the pipes, valves and pumps within the valve pit are ignored; a conservative assumption since energy removed by these materials would tend to freeze the pool faster than calculated in this analysis.

The overall progression of the core debris is depicted in Fig. 2 showing the general outlines of the molten mass as a function of time and also indicating the final position where solidification occurred. As illustrated, the original attack is rapid but quickly decays and equilibrates to a frozen condition after 102,600 sec (28.5 hrs) with 0.53 m of downward attack. The hydrogen evolved by this process can be determined from the water released from the concrete and results in an accumulation within the containment of $\sim$ 3%. Since the material available for oxidation during this concrete attack is generally the unreacted zircaloy from the in-vessel processes, the hydrogen accumulation within the containment would generally be less than 7%, i.e. less



CALCULATED CONCRETE ATTACK FOR VALVE PIT

Fig. 2 Extent of Thermal Attack as a Function of Time than the global combustion limit. If additional metal is provided by melting of steel components inside of the reactor pressure vessel, the pressure vessel wall, or additional steel from the components within the valve pit, the reaction would be limited by the extent of concrete attack as considered herein. In addition, as the melt becomes more viscous, gases released from the concrete will begin to bypass the melt and not oxidize metallic constituents. This bypass was neglected in the above analysis, a conservative feature of the approach. In summary, the maximum extent of hydrogen generation including the in-vessel production and that resulting from a non-coolable debris bed assumption is equal or less than oxidation of all the zircaloy within the core.

As a result, the assessment made in the Big Rock Point Probabilistic Risk Assessment is not sensitive to the assumption of a permanently coolable debris bed and could withstand substantial concrete attack without resulting in additional risk to containment failure and public health and safety. This analysis has employed various assumptions, some very conservative and others somewhat optimistic, but the combination of the two provides a reasonable assessment of concrete thermal attack for PRA assessments.