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NUREG-0956

DRAFT

BMI-2104
Volume VII

Radionuclide Release Under Specific LWR Accident Conditions

Volume VII
Response to Peer Review Comments

Prepared by
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February 1985

Prepared for
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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ACKNOWLEDGMENTS

Battelle's Columbus Laboratories wishes to acknowledge and express appreciation for the computer codes made available for this program by Sandia National Laboratory, Battelle's Pacific Northwest Laboratories, and the Kernforschungszentrum Karlsruhe, and for the computations and consultation provided by Sandia and the Oak Ridge National Laboratories. Further, members of the Peer Review Group have contributed significantly to this effort by providing comments, suggestions, and information on various reactor systems design.

The support of the U.S. Nuclear Regulatory Commission is gratefully acknowledged, as is the untiring leadership of the NRC staff, particularly Mel Silberberg, Mike Jankowski, Chris Ryder, and Ralph Meyer.

The work of Chris Ryder in reviewing and summarizing notes, letters, and meeting transcripts was a crucial link in the preparation of this volume.

The diligent efforts of many Battelle staff members contributed to the preparation of this report. The following list identifies those staff making major contributions:

RJ Avers
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RADIONUCLIDE RELEASE UNDER SPECIFIC
LWR ACCIDENT CONDITIONS -- VOLUME VII
RESPONSE TO PEER REVIEW COMMENTS

by

J. A. Gieseke, P. Cybulskis, R. S. Denning,
M. R. Kuhlman and K. W. Lee

February 8, 1985

INTRODUCTION

This is the seventh volume of a report on "Radionuclide Release Under Specific LWR Accident Conditions". The first six volumes of this report are concerned with analyses leading to source term predictions for five nuclear power plants, three PWR's and two BWR's (Surry, Peach Bottom, Grand Gulf, Sequoyah, and Zion), with a selection of accident sequences chosen for each plant. As a means for making the procedures and results of these analyses visible to the technical community and to provide a forum for periodic review, Peer Review Meetings were held by the U.S. Nuclear Regulatory Commission. This volume represents a response to the questions and comments which resulted from the review meetings.

THE PEER REVIEW PROCESS

Peer Review Group and Meetings

Periodic Peer Review Meetings were held to review progress on the analyses and to discuss revisions and improvements in the analysis procedures. A group of scientists with diverse technical specialties was selected by the U.S. NRC to serve as the official Peer Review Group although the meetings were open with usually about 50 observers in attendance. Both the review group and observers asked questions and provided comments, and the comments included in this volume came from both groups. The Peer Review Group members are listed in Table 1.

TABLE 1. INVITED PEER REVIEW GROUP MEMBERS

D. H. Walker	Offshore Power Systems
R. Vogel	Electric Power Research Institute
R. Ritzman	Science Applications, Inc.
R. Hilliard	Hanford Engineering Development Lab
C. Johnson	Argonne National Laboratory
D. Cooper	Harvard University
A. W. Castleman	Penn State University
D. Rowe	Rowe Associates
W. Kastenberg	University of California (LA)
A. Reynolds	University of Virginia
L. Zumwalt	North Carolina State University
S. Levy	Levy Associates
D. Torgerson	Atomic Energy of Canada Limited
R. Potter	United Kingdom Atomic Energy Authority

The procedure for the review process was for Battelle to provide in advance of the meeting draft copies of completed portions the reports, BMI-2104 Volumes I through VI. These draft copies formed the bases for the discussions which were in the form of presentations by Battelle staff. Described in the presentations were the technical procedures and the results for analyses completed up to that time. Battelle's presentations were supported and augmented by presentations from laboratories which had developed several of the codes and which had provided technical input for report preparation. These laboratories included Sandia National Laboratory, Oak Ridge National Laboratory, and Battelle's Pacific Northwest Laboratories. During and after these presentations, questions and comments were received from the Peer Review Group and the observers. In addition to a transcription of the meeting which provided a record of questions and comments, reviewers sent written comments to the NRC after each meeting.

The preparation of Volumes I through VI was an iterative process with some improvements in analysis procedures made during the course of the study. This iterative nature is evident in Table 2 which provides the dates and major topics of discussion for each of the Peer Review Meetings. Revisions in procedures resulted from data becoming available, Peer Review Group comments, and improvements identified by the Battelle staff performing the analyses. Questions and comments from each of these meetings were collected and combined by the U.S. NRC staff into the list forming the basis for this volume.

Description of Comments and Questions

As noted previously, questions and comments were compiled from transcripts of the meetings and from written material submitted to the NRC. The questions ranged quite widely in technical area encompassing such diverse subjects as reactor operation, detailed aerosol physics, and chemistry. Many of the questions were intended for clarification only and were answered during the meetings. However those questions for clarification that implied a general misunderstanding of the analysis

TABLE 2. PEER REVIEW MEETINGS -- DATE, TOPICS

First Peer Review Meeting (January 25-26, 1983)

Topics: General Approach on Source Term Issues
Description of NUREG-0956
Discussion of BMI-2104, Volume I
Discussion of the Surry Analysis

Second Peer Review Meeting (May 24-25, 1983)

Topics: Summary of January Comments, Changes
BWR Peach Bottom Analysis
BWR Grand Gulf Analysis
SPARC Code, GE Model -- Suppression Pool

Third Peer Review Meeting (July 28-29, 1983)

Topics: BWR Mark I Peach Bottom Plant -- Conclude
BWR Mark III Grand Gulf Plant -- Conclude
Ice Condenser Sequoyah Plant
ICEDF Model -- Fission Product Scrubbing by Ice Condenser
Comparison of Models for Suppression Pool -- GE Model, SPARC (NRC), SUPRA (EPRI)
Summary of Code Validation Report
Fission Product and Aerosol Production During Core/Concrete Interaction -- VANESA, Experiments
Chemical Interactions in Primary System -- Theory, Experiments
Uncertainty Analysis from BMI-2104 Reports

Fourth Peer Review Meeting (October 12-13, 1983)

Topics: Calculations for PWR Ice Condenser Plant -- Sequoyah
Recalculated Surry Analysis
Calculations for PWR Zion Plant
Coolant Pools in VANESA Model and Ex-vessel Source Term
Status of Code Validation
Analysis of the Steam Explosion Source Term
Uncertainty/Sensitivity Analysis -- Update

Fifth Peer Review Meeting (January 26-27, 1984)

Topics: Calculations for PWR Ice Condenser Plant -- Sequoyah
Recalculated Surry Analysis
Calculations for PWR Zion Plant

procedures were included for response in this volume. Other questions were based on a technical assumption or opinion and therefore are of more general interest. These have also been considered here.

During the first Peer Review Meeting, the Battelle staff identified and discussed a number of important assumptions required to perform this study and listed major uncertainties associated with various aspects of the analyses. These assumptions and uncertainties were also highlighted in BMI-2104 Volume I and provided a list of issues that the Battelle staff believed to be important and unresolved at the time of the study. Although these were, in effect, comments on the study, they were offered by those performing the study and are not included here because essentially all of these issues reappeared later in the questions and comments from reviewers.

This volume employs an outline based on technical areas or topics. Questions falling under each topic are presented and a response to the question follows in the text. The listed questions or comments are in many cases an amalgam of several individual questions on the same issue. The NRC staff working with the meeting transcripts and written comments, combined individual comments or questions where appropriate to form a new comprehensive question containing the sense of the original questions. This new question was then included on the list for responses.

The following portions of this report volume consist of questions or comments organized by major technical issues. Each comment or question is followed by the Battelle response.

QUESTIONS/COMMENTS AND RESPONSES

DECAY HEAT

Hardware and Transport

Comment: Hardware surfaces should be warmed by the decay heat from fission products. The heat should create convection currents and modify the deposition of both aerosols and vapor fission products. The

ay heat may also melt pipes. Deposited fission products may revaporize and form aerosols. The aerosol may then deposit to start the evaporization-condensation process again. This process may occur repeatedly.

Response: The influence of fission product decay heat on transport and deposition in the RCS has been identified in the BMI-2104 report as an area where considerable uncertainty exists. The influence of this decay heat is ignored in the analyses presented. Preliminary estimates were made by the Battelle staff and presented at the Peer Review Meetings which suggested a possibility for significant effects from decay heating. Work subsequently performed by several organizations has been presented informally in various forums and has indicated a variety of possible outcomes when this heat source is accounted for in different manners. No analysis presented however has attempted to account for chemical reactions between the deposited species and the surface material or co-deposited materials -- which could change the vapor pressures involved in the revaporization; nor have the recirculation currents driven by the heated structures been accounted for except to assume that such currents mix the gas within a control volume. The results obtained so far have demonstrated sensitivity of the results to the mass of structure which is assumed to be heated by the fission products, and the portion of the decay heat assumed to be imparted to the surface. Different assumptions can lead to predictions of pipe melting or of complete redistribution of the deposited material. Truth probably lies somewhere between these two extremes, but more detailed knowledge and better analytical tools than those used so far are required to reduce the uncertainty associated with this potentially important phenomenon.

Comment: To determine the sensitivity of the predicted retention by the reactor coolant system, Battelle estimated the heat load from the fission products and calculated how this would modify deposition. According to the calculations, decay heat reduces the deposition of vapors but not of aerosols in the reactor coolant system. The models underestimate the fission product release because too much credit is given for retention by the reactor coolant system.

Response: Fission product decay heat is expected to reduce the net deposition of vapors in the RCS via the mechanism of revaporization (see previous Comment). It is not anticipated that structures could reach temperatures high enough to vaporize the relatively involatile materials which constitute the aerosol material so a reduction in aerosol retention via this mechanism is not to be expected. Any thermophoretic forces associated with the fission product heating are probably quite small in comparison with the gravitational force which is responsible for most of the aerosol retention in the RCS, so no modification of aerosol retention can be anticipated from this source either.

Ice Condenser Performance

Comment: The decay heat from fission products would likely accelerate the ice melting during an accident. The availability of ice to suppress steam would then be reduced. Ice melting from decay heat is currently not in the models. The models underestimate the consequences of an accident.

Response: The rate of ice melting is determined by the steam and gas flows that are input to the ice condenser. Any fission products carried into the ice condenser are expected to be washed off the ice by the liquid water resulting from steam condensation and melting of the ice. The decay heating of these fission products is included in the energy balance for the containment sump into which the liquid effluent from the ice condenser would flow. It is unlikely that any fission products would adhere to the ice condenser surfaces.

Heat Capacity

Comment: The heat capacity of the reactor coolant system and the containment have not been accounted for. Not only will these structures serve as a heat sink but also as a fission product sink.

Response: The heat capacities of the reactor coolant system as well as those of the containment structures are included in the thermal hydraulic analyses.

FISSION PRODUCT RELEASE

Tellurium

Comment: Models have tellurium leaving a core in the elemental form. Compounds of tellurium, such as gaseous H_2Te, are not considered.

Response: Chemical forms for fission products as they leave the core region are not well known. This results from a paucity of experimental data and from complications in thermodynamic estimates because large numbers of materials are present. Simple estimates based on free energies of formation for single elements in a steam-hydrogen mixture are most often cited for suggesting predominant forms. Several such estimates were available and used in selecting the elemental form for tellurium. The chemical form for tellurium is not a simple case however.

It is possible to estimate other potentially significant forms for Te. A very likely form for Te at the point of release is SnTe which should be of the same level of importance as elemental Te. The form H_2Te is only marginally stable at any temperature. At a hydrogen pressure of about one atmosphere and at any temperature, H_2Te is likely to be of little significance. At temperatures below 1000 K and at hydrogen pressures of 100 atmospheres H_2Te is expected to be of about the same abundance as elemental Te as vapor species but significantly less importance than SnTe. It is obvious that the relative abundance of any chemical form of tellurium will depend on specific conditions, but conditions for most sequences do not favor H_2Te as a predominant form.

In the analyses performed, an attempt was made to account for an important aspect of Te chemistry, namely the reaction with Sn in the Zircaloy cladding. This was crudely modeled using fits of release rates to empirical data. Experimental programs are in place to provide additional data which will permit improved treatments of Te release from the core and its subsequent transport.

Iodine

Comment: Models have iodine leaving the core as molecular iodine. However, due to the chemical properties of iodine, the iodine species leaving the core is unlikely to be molecular iodine. Cesium iodide is a more likely species to be leaving a core.

Response: The CORSOR code considers I and Cs to be released from the melting core independently. Based upon thermochemical analyses performed earlier and reported in Volume I of BMI-2104, it appears that CsI is the preferred chemical form for the I for most temperatures and gas compositions within the envelope of interest. For this reason, in all analyses presented in Volumes II-VI, the I released is combined with the Cs to form CsI, and the remaining Cs (~90 percent) is converted to CsOH. These materials are then input to the RCS transport and deposition code.

In some analyses, the I was followed through the RCS using TRAP-MELT as if it were present in its molecular form in addition to its treatment as CsI. (This is clearly aphysical -- but was done so that the I could serve as a tracer of the transport of essentially inert species through the RCS.) The reported iodine results, however, are based solely on the analyses of the CsI behavior for the accident sequences examined.

Silver

Comment: Little is known about the release of silver. Silver may serve as a source of condensation nuclei onto which aerosols form.

Response: There is considerable uncertainty regarding the behavior of control rod silver in PWR accidents. For low pressure scenarios -- such as the pipe break sequences -- there appears to be the possibility of bursting of the control rods with ejection of molten alloy. For the high pressure sequences it appears possible that the control rods may melt and candle, with the molten alloy relocating to lower regions of the core and possibly refreezing in cooler regions -- at least

temporarily. In any scenario, however, the cadmium (which represents about 5 percent of the control rod alloy mass) is expected to be released and form an aerosol which may act as a seed aerosol for other more volatile species to condense upon. The release of the control rod silver estimated in BMI-2104 is based upon limited experimental results and is now believed to be an overestimate of the total mass of silver released. Current research efforts in the U.S., U.K., and FRG are investigating the vaporization and release of Ag-alloy control rods under accident conditions. The role which silver aerosol would play in RCS transport and deposition is principally that of a fission product scavenger, since by adding aerosol mass to the system one increases the rate of agglomeration and growth, which results in enhanced fallout of particles in the RCS. With respect to providing condensation nuclei, silver's presence would be superfluous -- given the amount of other materials whose release is much less uncertain.

AEROSOL DEPOSITION IN THE RCS

General

Comment: Aerosols deposit in the pipes of a reactor coolant system. Little is known about how the aerosols accumulate and less is known about how the aerosols resuspend. A calculation of the volume of fission product material rather than the mass of fission product material may be necessary to determine the significance of plugging. The models fail to accurately predict flow through pipes because plugging from aerosol deposits is not considered. Further, resuspension mechanisms are neglected in the models.

Response: The deposition of aerosols by a number of mechanisms is predicted for the RCS. The distribution of these deposits is predicted to the extent that gravitational settling and other wall deposits are known separately in each control volume. This knowledge of deposited amounts is probably sufficient to assess the importance of plugging. However, it was assumed that plugging was not significant and that the

deposited materials did not affect flow through the various control volumes.

It is expected that much of the deposited material will be in a liquid or molten state under RCS conditions. If this is the case, it is to be expected that some relocation of deposits will occur and plugging will not occur. An example of the extent of deposition relative to flow area can be seen for the V sequence as described in Volume V where a deposit of about 50 g per cm of pipe length is calculated assuming no relocation. This deposit amounts to less than 10 percent of the flow area and would not be expected to be a significant effect. If relocation occurs, of course, the effect is even less. Based on considerations such as these, it was believed appropriate to neglect the effects of deposits on flow.

Concerning mechanical resuspension, the analyses were based on the assumption that either this would not occur or if it did occur, there would be a negligible effect on subsequent aerosol behavior. The time periods at which resuspension is a possible concern are at the time of vessel melt-through in a high pressure sequence and when molten fuel materials slump into water in the lower plenum resulting in rapid vapor generation. The only other case of possible significance would be during flow in the ECC piping in the V sequence. If the deposits are liquid as expected, any resuspension would be as drops and considering the conditions, the drop sizes would be large (probably >10 μm) even in the most severe cases. The impact of large drops on aerosol behavior in the containment would be negligible. Similarly, for dry deposits (if they could exist), resuspended particles are generally quite large. There remains some possibility for dry deposits to form some smaller particles on resuspension. However, the RCS conditions (temperature) at melt-through are not expected to give dry deposits.

Deposition/Resuspension

Comment: When aerosols pass through pipes at high speeds, turbulent deposition and resuspension should occur. These phenomena do

not seem to be adequately modeled. During post accident conditions, the turbulent flow through pipes may not be the same as turbulent flow during normal conditions.

Response: Turbulent deposition is a well known mechanism both experimentally and theoretically while the resuspension mechanism of deposited particles is not yet fully understood. Many experimental measurements show that the particle deposition velocity normalized by the fluid friction velocity ranges from 10^{-5} to 10^{-1}. For particles larger than about 1 μm, the deposition is primarily due to the particle inertia and for particles of small sizes deposition is primarily due to diffusion under turbulent flow conditions. The TRAP-MELT code contains turbulent particle deposition correlations given by Wells and Chamberlain (Brit. J. Appl. Phys., 18, p 1793, 1967), Liu and Agarwal (J. Aerosol Sci., 5, pp 145-155, 1974), and Gieseke, et al (NUREG/CR-1264, BMI-2041, 1980), and these mechanisms are automatically activated whenever the Reynolds number exceeds 2300. As the comment points out, there will be situations where complicated flow patterns may exist for various geometries and under certain accident events. While it is certainly possible to model complex flows and the corresponding deposition mechanism, lack of geometrical details and uncertainties in input for thermal hydraulic models suggest that a practical approach is to use generalized geometries and flow patterns.

Comment: Deposition in the reactor coolant system is not well understood. Moisture or bulk water in the pipes will influence deposition. The models do not adequately account for Stefan flow, electrostatic interactions, and turbulent reentrainment. A consistent methodology needs to be developed; general-space dependent forms of aerosol equations are needed; unambiguous deposition coefficients are needed; recent turbulent deposition correlations should be used. Deposition rate expressions and correlations seem to be ad hoc.

Response: It is true that deposition in the RCS is not well understood. This unfortunate fact arises from a lack of adequate specification of many of the important independent parameters which govern

the deposition of fission products and a paucity of data for vapor and aerosol behavior within geometries and for thermal hydraulic conditions of interest. Moisture or bulk water would indeed influence deposition if it were present. In general the RCS flow path is superheated and dry in the period of fission product release in a core meltdown scenario. (There is a possibility that water may be present in the lowest portions of the steam generators in a PWR for a pressurized RCS if the secondary side is able to remove sufficient heat continuously or in the pressurizer of a PWR in accidents involving loss of heat removal.) For Stefan flow to be significant in the RCS, the condensation of large amounts of vapor is required. Water is the only vapor which appears to be present in sufficient quantity, but it cannot condense in the RCS for any of the sequences examined. Electrostatic interactions have not been treated in the analyses. If the RCS atmosphere is bipolar ionized due to the intense radiation present, the effect on aerosol coagulation will be negligible. Electrostatic deposition on grounded surfaces would be enhanced only if one postulates particles which can maintain a high charge in a highly bipolar ionized atmosphere. Turbulent deposition and reentrainment are indeed complicated phenomena. The approach taken in this study was discussed in the previous two responses.

General statements regarding the need for consistent methodology and unambiguous deposition coefficients are, of course, agreed upon. The correlations used for aerosol turbulent deposition in this study represent the state of the art. The heat and mass transfer coefficients generated in the TRAP-MELT code should be improved to better reflect the various flow regimes and perhaps entrance zone effects.

Steam Generator

Comment: During a severe accident, the steam generators may remain as an effective heat sink. Significant amounts of steam may condense within the steam generators and remove fission products. These phenomena may be a significant mechanism for removing fission products.

Response: The effect of the steam generators on fission product retention within the primary system is included in the analyses of those sequences in which the fission product transport path is through the steam generators. The effect on fission product retention has been found to be accident sequence dependent and can be quite significant.

Comment: The calculated heat transfer coefficient for a steam generator is erroneous. The calculation is based on normal flow through a steam generator. During an accident, when one side of the steam generator may be stagnant, the heat transfer will be much less than during normal operation. The predicted steam generator heat transfer rate should not depend on the core heat transfer model.

Response: Steam generator heat transfer is based on normal operating conditions only for the purpose of determining the timing of the dryout of the secondary side. Under conditions of effective thermal coupling between the primary and secondary sides of the steam generator, i.e., when there is liquid on both sides, the exact heat transfer coefficients utilized are not particularly important; the overall heat loads during accident conditions are at decay heating levels and are much lower than the capacity of the steam generators. When one or the other side of the steam generator dries out, the heat transfer capability will be drastically reduced; this is taken into account in the analyses. In the evaluation of fission product transport through the steam generators, their heat transfer performance is explicitly evaluated taking into account the flow velocities and regimes, the surface area, and temperature differences as a function of time.

SUPPRESSION POOLS

Particle Deposition

Comment: The SPARC code models scrubbing of aerosols in the suppression pool by employing an assumed spherical bubble shape to obtain deposition velocities which are then corrected to account for the expected

elliptical shape. The initial use of an elliptical shape to develop the deposition velocities is a preferred approach. The assumption of spherical bubbles with subsequent corrections to account for the expected elliptical shapes as part of the deposition velocity predictions may lead to low estimates of aerosol retention (low DF's) for particles larger than a few tenths of a micron diameter.

Response: It is agreed that more thorough theoretical analysis would be preferred to the approach taken in the SPARC code. The use of a spherical form was based on the availability of theoretical treatments for this case and the more detailed approach seemed to be more than was needed to obtain an adequate estimate of scrubbing. It should be noted that scrubbing efficiencies are high enough, even with the approach taken, that the source term is dominated by other factors such as pool bypass rather than by the scrubbing process itself. Since experimental scrubbing data are becoming available, it is expected that differences in analytical treatments will be able to be resolved by comparison with data.

Bubble Movement

Comment: The bubble density in a suppression pool may be high enough to cause bubbles to interact, such as by foaming.

Response: The bubble density in the suppression pool may be high enough to cause interactions between the individual bubbles. Foaming, however, would seem to be an unlikely consequence in the absence of high levels of surfactants in the suppression pool water. A more likely form of interaction is through enhanced bubble rise velocity, called swarm rise velocity, which reduces the residence time of the bubbles in the pool. The SPARC code as used in BMI-2104 uses an expression for bubble rise which assumes the existence of a swarm rather than individual bubbles for its calculation of residence time.

Comment: Surfactants may alter the movement of bubbles. Currently, it is assumed that the bubbles move independently.

Response: Surfactants, if present at sufficient concentration in the suppression pool, could influence the bubble dynamics. It should be kept in mind that the suppression pool serves as a potential source of emergency feedwater for the RCS and is therefore kept scrupulously clean at operating plants. The composition of this water is not expected to include significant surfactants since the water has been processed through the plant's ion exchange columns. In addition to the continual cleaning, efforts are made such as using epoxy paint over metal liners which aid in keeping the water free of extraneous materials which could act as surfactants.

Flashing

Comment: During an ATWS, the temperature of the water in the suppression pool may rise above the normal boiling point. Given this condition, the suppression pool would flash if the containment failed and depressurized. Flashing during a core meltdown might impair the capability of the suppression pool to remove fission products. A sequence has not been defined to account for flashing in the suppression pool.

Response: In accident sequences such as BWR transients with failure to scram, we do find that the suppression pool is well above the boiling point following containment failure. It is, in fact, the overheating of the suppression pool that leads to containment failure in such sequences. The partial flashing of the suppression pool as the containment pressure drops is taken into account in the analyses. Typically the suppression pool has cooled to normal saturation conditions by the time of core melting and fission product release. Such saturated conditions have been utilized in the assessment of fission product scrubbing for these sequences. It is possible that under some circumstances the suppression pool would still be undergoing continuing flashing or boiloff during the time of fission product release. Such flashing could further impair the fission product removal by the pool. We do not at present have a means of quantifying the degree of such further impairment of fission product scrubbing by flashing pools.

Solubility

Comment: The solubility of carbon dioxide in a suppression pool was not modeled. The water in a suppression pool should absorb at least some carbon dioxide to reduce the pressure in a containment. Without considering carbon dioxide solubility, the source term models are conservative.

Response: The solubility of carbon dioxide in a suppression pool was not considered in the BMI-2104 analyses. In response to this comment we have assessed the possible impact on containment pressure of such gas dissolution. The specific example considered is the Grand Gulf TQUV sequence in which the containment was predicted to fail due to long term buildup of noncondensibles, including carbon dioxide and carbon monoxide. Just prior to the predicted time of containment failure in this sequence, the containment atmosphere is predicted to contain approximiately 174,000 lb of carbon dioxide and 106,000 lb of carbon monoxide. The suppression pool is being cooled and contains 11,563,000 lb of water at 114 F. The various gas species are not uniformly distributed within the containment. At the conditions predicted just prior to containment failure, we estimate that about 13,200 lb of CO_2 and 1,040 lb of CO could be dissolved in the suppression pool. These quantities correspond to about 8 and 1 percent, respectively, of the total CO_2 and CO. The dissolution of these fractions of the total gases available would have a small impact on the prediction of containment pressure response. It may also be noted that some fraction of the dissolved gases would be released upon containment depressurization. The effect of such gas release on the fission products in the pool is not know.

AEROSOL DEPOSITION IN A CONTAINMENT

Diffusiophoresis

Comment: The significance of diffusiophoresis has not been determined. At first, this phenomenon was not modeled. Refined models

included diffusiophoresis. However, the consensus among the code developers is that diffusiophoresis is insignificant for two reasons: (1) the steam in a containment should condense before aerosols enter the containment and (2) the surface/volume ratio of a containment is small. Significance of diffusiophoresis has not been rigorously determined.

Response: Whether diffusiophoresis is important or not obviously depends upon prevailing thermodynamic conditions. As the comment indicates, significant steam condensation takes place before fission products enter the containment.

A limited number of calculations showed that diffusiophoresis is not as significant as previously thought partially because thermal hydraulic conditions exhibit a prolonged superheated atmosphere in the containment and partially because the flux due to diffusiophoresis toward walls competes with that onto particles due to condensation of steam. Nevertheless, diffusiophoresis was included in all calculations of aerosol deposition in containments using predicted steam condensation rates as a basis. It is agreed that diffusiophoresis certainly needs further experimental verification.

Scrubbing

Comment: Water drops falling through the atmosphere of a containment can scrub the air of fission products and the mechanisms are well understood. However, the distribution of water drop sizes from a spray system has not been rigorously determined. Therefore, the modeling of scrubbing by spray water drops seems to be unnecessarily inadequate.

Response: Water drop sizes are not uniform in practice as the comment points out. The drops will attain a certain size distribution which will depend upon the type of sprays and the operating water feed rate. The effects on predicted scrubbing efficiency of assuming a distribution of drop sizes instead of a single drop size has not been determined in this study. However, scrubbing by a polydisperse spray can be closely approximated if the proper mean size for the spray drop size distribution is used. To accomplish this, the number median

drop diameter was used in the analysis. This choice is based on consideration of the principle that scrubbing rate is roughly proportional to the inverse of drop size for a given spray water feed rate.

Fog

Comment: The impact of the fog model on the calculated source terms is unclear. Instead, a model based on the carryover of water with steam might be preferable. The validity of the fog model is questionable.

Response: It should be noted first that the fog model is activated only if the supersaturation ratio exceeds a critical value (on the order of 3 or 4) depending upon the containment temperature and second that the supersaturation ratio normally does not reach these values in most accident sequences. We have examined the effects of the fog model upon the overall fission product concentration before the model was activated and found that there was very little difference. With the fog model included, steam condensation forms an enormous number of very tiny water droplets that subsequently undergo intense coagulation among themseleves and with existing aerosol particles. The time required for these water droplets to grow and coagulate with aerosol particles was found to be extremely short. Within the fog model, the usual condensation mechanism in which steam condenses directly onto particles is still operating. The results of utilizing this model showed very little difference in calculated size distributions of fission products.

INTERACTION OF AEROSOL MECHANISMS

Comment: Some aerosol mechanisms, such as gravity settling and impaction, are independent of one another in a thermal hydraulic sense. Other mechanisms, such as thermophoresis, diffusiophoresis, and turbulent deposition, operate on an interaction of heat, mass, and momentum transfers. The interaction is neglected from the modeling.

Response: It is true that thermophoresis, diffusiophoresis, and turbulent deposition depend significantly upon thermal hydraulic

conditions and in fact they could also influence or interact with heat, mass, and momentum transfer. Theoretical understanding and experimental validation on interaction of thermophoresis/diffusiophoresis with thermal hydraulics are not complete at this time. Further, the importance of the possibility is vaguely known. Interested parties may read a recent article by Loyalka (Progress in Nuclear Energy, Vol. 12, pp 1-56). What can interact with thermal hydraulic conditions to a greater extent is the steam condensation onto particles primarily due to latent heat. In our analysis, the effects of condensation and evaporation on the atmosphere thermal hydraulic conditions are treated in MARCH calculations. Thus, whenever the supersaturation condition is met, excess steam above the saturation vapor pressure condenses. Calculations showed that the difference between this type treatment and more rigorous methods where condensation onto particles and thermal hydraulic conditions are directly interacted is rather small for typical accident sequences. The interaction of aerosol deposition and thermal hydraulic conditions can be very important for the primary system when decay heat of fission products is taken into account. This is addressed elsewhere.

Comment: Aerosols of cesium hydroxide and cesium iodide are treated as being separate and independent. Actually a single aerosol composed of varying amounts of each compound will likely exist.

Response: This comment is probably prompted by an artifact of the presentation of the results of the analyses. The aerosol particles are considered to be composed of the low volatility materials released from the core (termed "Aerosol") as well as whatever amounts of CsI and CsOH are predicted to condense on the particle surfaces. The tables and figures presented then sum the CsI mass, whatever form it is found in, and present masses retained in various portions of the RCS and the total released fraction from the core. The same is done for CsOH and Te.

Comment: Coagulation coefficients may be altered by orders of magnitude when ambient conditions are changed by small amounts. This observation is attributed to a strong synergistic relationship among the aerosol mechanisms (thermophoresis, diffusiophoresis, etc.).

Response: This comment is largely similar to the comment appearing first in this section in which interaction of thermophoresis/diffusiophoresis with ambient conditions is discussed. The effects of altered ambient conditions on coagulation rates may not be accounted for until the issue of the mentioned interaction is fully resolved among scientists. The synergistic effects of various individual aerosol behavior mechanisms may or may not be significant and the current state of the technology has not advanced far enough to resolve the question. Again, interested individuals may reference the article cited previously.

STEAM CONDENSATION

Adiabatic Gas Expansion

Comment: When a containment fails, the rapid pressure drop should lead to an adiabatic gas expansion. The adiabatic expansion should enhance the condensation of steam and lead to enhanced fission product removal. This phenomenon is not modeled.

Response: Gas expansion following containment failure is modeled in our analyses and is treated as an isenthalpic process. Depending somewhat on the initial conditions, such an expansion typically does not result in condensation. An adiabatic isentropic expansion of steam, on the other hand, would be expected to be more likely to lead to condensation. We believe that the isenthalpic expansion is a more appropriate approximation to the real process than the more idealized adiabatic isentropic expansion.

Particles

Comment: Steam condensation can occur on both aerosols and on walls and is a significant way to remove fission products. Condensation readily occurs on particles as small as 0.6 μm. If the particles are hygroscopic, then water can condense on the particles even if the atmosphere is unsaturated. The steam condensation phenomenon is not however

well understood for two reasons: (1) the affinity of aerosols for water is unknown and (2) a quantitative description of a boundary layer at a wall is unknown.

Response: We agree that the treatment of steam condensation phenomena in the source term analyses can be improved further. The affinity of aerosol particles for water or the solubility of particles reduces the required saturation ratio and as a result, condensation can be greatly enhanced. Theoretically, the effects of soluble particles on condensation can be treated by accommodating Vant Hoff's factor in the condensation rate expression.

While some of the fission products have a high solubility, the data base for the actual Vant Hoff factor, the surface tension, and the solution density for each species (which vary with the solution concentration) are not currently available. It should be recognized that each soluble species is modeled to be a constituent of the aerosol particles and is subject to various aerosol behavior mechanisms rather than modeling each species to form separate particles. The fraction of soluble substances in the aerosol particles released during the melt release period is typically less than 10 percent and those released from the core-concrete interaction contain less than 1 percent of soluble substances. It is then concluded that while the particle solubility can enhance condensation, the effects would not be as significant as that the condensation rate equation alone would indicate.

On the boundary layer particle concentration profile at the containment walls, it is agreed that the quantitative magnitude of the boundary layer thickness is not well treated. In the past, it was common to supply the value of the boundary layer thickness to the computer code by users as an input rather than rigorously calculating it using boundary layer theory. In our analysis, it was assumed that aerosol particles regardless of their size follow the steam flux toward walls and the particle deposition rate was assumed to be the same as the rate at which steam condenses onto walls. This approach eliminates the need for "guessing" boundary layer thickness.

Comment: Mason's equation is used to describe condensation and evaporation; this equation, when used to describe heat and mass transfer, is invalid for small particle sizes.

Response: As the comment suggests, Mason's equation is not adequate for the gas slip regime or the transition regime where the particle sizes are comparable to the mean free path length for the gas molecules. There exist many theories and suggested models which account for these slip or Knudsen number effects in their treatment of condensation. Generally the suggested treatments involve matching the flux equations such as those suggested by Fuchs and Sutugin (Topics in Current Research, Vol. 2, Edited by Hidy and Brock, Pergamon Press) and many other investigations. As particle size becomes even smaller, molecular kinetics should be applied of course.

The current analysis indicates that typically the mass median diameter of particles in the containment ranges from 1 to 15 μm. The mean free path length of gas molecules in the atmosphere on the other hand is a few hundredths of a micron. As a result, the corresponding Knudsen numbers are quite small. Further, condensation to large particles dominates regardless of the correction for the small size effects. It is believed therefore that correction to Mason's equation for small particles would not result in any significant changes to the calculated source terms.

Nucleation

Comment: Homogeneous nucleation is described using old equations. Nevertheless, in recent years, many advanced equations describing nucleation have been developed. The equations in the current source term models are not being related on the basis of appropriateness.

Response: The fog model was incorporated in this study to provide a convenient means for modeling formation of fog droplets in the containment atmosphere. The incorporation was not intended to resolve the physics of the nucleation phenomenon microscopically. Controversy on validity of various nucleation theories may be referred to Katz (J. of

Chem. Phys., p 4733, 1970). Resolution of a most accurate and sound model among numerous theories by comparing various models with very carefully controlled experiments is clearly beyond the scope of this study. Our calculations showed that thermohydraulic conditions were such that the incorporated fog model is usually not activated during most of the accident period. Even if the model is activated, the effects on the fission products release are minimal thus justifying the use of a simple model in the containment analyses.

In the RCS, the correct prediction of the rate of homogeneous nucleation is irrelevant to the solution of the aerosol dynamics in a severe accident situation. The mass release of low volatility species from the melting core proceeds at a high rate and in an environment to which homogeneous nucleation theory is not applicable due to the highly ionized atmosphere. This neglects the fact that the precise composition of the particles nucleating is unknown, but is expected to be highly variable depending upon local temperatures, inventories, and gas flow rates.

In any situation where high rates of release of condensible vapors occur, the influence upon particle size of either the critical size for or the rate of nucleation is not detectable after only a very short time period compared with the time scale of interest in RCS accident analyses.

MIXING OF CONTAINMENT ATMOSPHERE

Upward Draft

Comment: Convection down the walls of a containment is likely to produce an upward draft in the center of the containment. The draft may hinder the gravity settling of aerosols. This assumes that the atmosphere in a containment behaves as a homogeneous mass.

Response: It is generally known that there would be natural convection in the containment. While this convection coupled with local air turbulence helps aerosol particles to be well mixed throughout the

containment space, the questions have arisen of whether incomplete mixing may occur and whether the particle concentration may be location dependent. In the past, we performed calculations using three separate and imaginary containment compartments (i.e., the lower part of the containment, the center of the containment, and the region near the containment wall) and incorporated several convection flow rates among these three compartments. The calculated results indicated that this treatment did not alter noticeably the long-term aerosol behavior, thereby supporting the well-mixed assumption. Similar bounding calculations were performed by assuming a 90/10 split in aerosol mass sources between each side of a containment separated into two halves by a vertical wall. The results indicated no overall change in leaked mass, again supporting a lack of sensitivity to mixing assumptions and thereby suggesting the adequacy of the well-mixed assumption.

Composition

Comment: The thermal hydraulic conditions in a containment during post-accident conditions are uncertain. One thought considers the atmosphere in a containment to be homogeneous; the atmosphere behaves as a single volume. Another thought considers the atmosphere to be heterogeneous; the atmosphere behaves as many small volumes. Because the containment is significant in attenuating the release of fission products, the thermal hydraulic conditions should be thoroughly understood. The modeling of the thermal hydraulic behavior of a containment atmosphere is inadequate.

Response: The degree of compartmentalization of the containment in the analyses is a question that requires further consideration; it is, however, a question that probably cannot be answered uniquely for all cases. As a minimum, the required compartmentalization will be a function of containment design. Where clear physical barriers exist, subdivision into compartments is appropriate. Where "compartments" are connected by large unobstructed flow areas, the need for such compartmentalization is not at all clear. In the BMI-2104 analyses we have

utilized the minimum compartmentalization consistent with a meaningful description of the physical layout of the plant.

For analyses that assume a high degree of subdivision of the containment, a number of additional boundary conditions and/or assumptions are required; these additional information needs are frequently not uniquely definable and may be arbitrarily selected. In a multiple compartment containment analysis, it will be necessary to specify the location of the point of release of fission products from the primary system. If the point of release is a relief valve, its location is known; if the point of release is a break in the system, its specific location is not known. Similarly, the location of the containment break must be specified. If the point of release of fission products from the primary system is close to the containment failure the predicted results could be much different than if the two are remote from each other.

CORIUM COMPOSITION AND VAPORIZATION

Sequence

Comment: The composition of corium should depend somewhat on the type of sequence that occurs. That is due to the rate at which the UO_2 and structural materials melt and mix. The corium composition ultimately influences the fission product release.

Response: The composition of the corium does indeed vary with the accident sequence and has been taken into account in our analyses. Among the most obvious factors that varies is the degree of oxidation of the Zircaloy. Also, the amount of structural material in the core debris can vary from sequence to sequence. And, of course, the extent of fission product release as a function of time is evaluated separately for each sequence.

Relative Volatility

Comment: Both the fuel and the control rods contain chemical species having a range of volatilities. Some of the species are significant in themselves because they are fission products. Other species are significant because they serve as condensation nuclei. Chemical reaction may influence the release of fission products. Alloys that form from fission products and cladding may influence fission product releases. The vapors from a degraded core will likely have a composition that differs from the core (including control rods). The differential release of fission products may lead to different types of aerosols.

Response: The physical and chemical form of the materials subject to release from the melting core-control rod-structural material has been recognized as one of the principal uncertainties in the analyses contained in the six volumes of the report. However, the procedures employed in this study do provide for a differential release of fission products and control rod materials. The release rate coefficients are derived to the extent possible from experimental data including in-pile, out-of-pile with fuel segments, and out-of-pile with simulants. It is to be recognized that the data base is not adequate to support a mechanistic treatment and additional experimental work is being performed under NRC sponsorship.

MORPHOLOGY OF CORE DEBRIS

Initial Melt

Comment: In the MARCH code, the calculations for core nodes dropping into the bottom head assumes a spherical shape for core debris. The diameter of the sphere is an input variable. The number of spheres is calculated using the sphere volume and the node volume. Each sphere has an assumed morphology; the center contains all uranium oxide and may contain some zirconium and/or zirconium oxide; presumably the shells contain only zirconium and zirconium oxide. The cladding may melt before the UO_2 melts.

Response: The MARCH code provides a number of options for evaluating the interaction of the core debris with water in the bottom of the reactor vessel. If the core debris is assumed to fragment upon contact with water, as has been assumed in our analyses, the debris particle size and configuration must be input. The assumed spherical particles can be specified to be homogeneous in composition or consist of several shells with specified compositions. The key factor beside the particle size is the location in the particles of the unreacted zirconium. In the BMI-2104 analyses we have typically assumed homogeneous particles one-half an inch in diameter. In selected cases we have assumed that the metallic zirconium is on the outside of the particles. The latter configuration leads to the prediction of somewhat greater reaction of the zirconium during the interaction in the vessel head.

While the melting point of the cladding is obviously lower than that of the uranium dioxide fuel, it is not at all clear that the cladding will run off and fall into the vessel bottom as it melts. Dissolution of the uranium dioxide fuel by the melting cladding has been experimentally observed to take place over a range of temperatures. The liquidus temperature for the materials of interest has been observed to vary with the composition and extent of zirconium oxidation. In the MARCH analyses we have utilized an effective melting point for the core materials that is between the melting point of the Zircaloy cladding and that of the uranium dioxide fuel.

High Pressure Ejection

Comment: A significant aerosol source term may arise from a high pressure ejection of molten fuel from a reactor vessel. D. Powers conducted experiments where simulants of molten fuel were ejected from a 1" orifice under 600 psi. A large amount of aerosols was generated. This phenomenon of aerosol formation is not modeled. It appears to be a very significant source of aerosols. Atomization has been given little attention.

Response: The potential aerosol source term arising from a hypothetical high pressure ejection of melt into the containment has not been addressed in these reports. This ejection is theoretically possible only for the sequences in which the RCS remains at high pressure to the time of core slumping. The generation of a large amount of aerosol particulate mass at the time of bottom head failure could lead to significantly enhanced scavenging of fission products previously suspended in the containment atmosphere. Depending on the timing of containment failure, there could be a significant reduction in the source term for some of the sequences analyzed in this report or could lead to enhanced release for fission products still remaining in the debris. Given the high degree of uncertainty regarding the probability of occurrence of high pressure melt ejection and the paucity of usable data regarding the aerosol generated in experimental melt ejections, there does not yet appear to be a supportable basis for incorporating this possible phenomenon into best-estimate analyses of accident sequences.

Sweeping

Comment: When the bottom head of a reactor pressure vessel melts, the core debris will be released into the reactor cavity and the containment building. Depending on the characteristics of the vessel failure, the core debris may remain in the cavity or be swept into the containment.

Response: It is recognized that under certain circumstances the core debris may be swept out of the reactor cavity upon failure of the vessel head. The likelihood of such sweepout would depend on the accident sequence, mode of vessel head failure, and design of the cavity. Whether such debris sweepout would make the consequences more or less severe is not clear. Dispersal of the debris into the containment may enhance the coolability of the debris and decrease the possibility of concrete attack. At the same time, finely dispersed debris in the containment atmosphere could lead to direct heating and chemical reactions with the atmosphere with resulting higher pressure loads. Also,

as noted in the previous response, dispersal of the debris may lead to reduced releases by scavenging of previously airborne fission products or to enhanced release of fission products still remaining in the debris. Current analytical capabilities do not extend into the treatment of the interactions between dispersed core debris and the containment atmosphere.

Comment: Rapid quenching of the debris may cause a steam pulse that reentrains aerosol particles.

Response: Steam production from the interaction of the core debris with water is routinely included in the analysis of accident sequences where such interactions are deemed appropriate. The resulting steam generation typically leads to fairly rapid containment pressure rises followed by substantial steam condensation on containment structures. The rates of pressure increase are not so rapid as to indicate any concern with aerosol resuspension.

Steam Explosion

Comment: Experiments done at the Brookhaven National Laboratory on core-water interaction show a series of steam explosions occurring. The steam explosions tend to disperse the core debris, making the debris coolable. But experiments done at the Sandia National Laboratory show no such explosions. Hence, the debris is not dispersed and may remain uncoolable. The core-water phenomenon is poorly understood and is not well modeled.

Response: It is agreed that the interactions between corium and water are not well understood. For this reason as well as others, a somewhat parametric approach was taken in the BMI-2104 analyses, considering the occurrence of various events for some of the accident sequences. Whether explosions take place when corium comes into contact with water cannot be predicted with certainty. The same is true for the character and disposition of the debris following such interactions. The models used to describe coolability of debris beds are based on experimental results and are believed to be quite appropriate for use in these

analyses; they do, however, require certain key input parameters such as the debris particle size.

With regard to debris coolability, it should be kept in mind that long term coolability can only take place if the debris bed is continuously supplied with cooling water. In many of the accident sequences of interest there is only a finite quantity of water that comes into contact with the debris; after such a finite amount of water is evaporated, the debris will reheat regardless of the character of the debris.

MORPHOLOGY OF AEROSOL PARTICLES

Comment: The changes in the composition of aerosol and vapors when hydrogen burns has not been determined. One fission product that could be affected is cesium iodide. Also, heat of reactions are unaccounted for.

Response: The change in vapor composition in the containment due to hydrogen burning is accounted for, insofar as reduction in H_2 concentration and increase in H_2O is concerned. Other chemical changes in the vapor are not accounted for, nor is vaporization of aerosol particles and potential changes in their composition computed, although there is the possibility of liberation of iodine in this process. This represents a shortcoming in these analyses and the possible impact on the released mass fractions has not been assessed.

Comment: The composition of the aerosols is considered to be constant.

Response: This was addressed in the preceding response. As clearly specified in the BMI-2104 reports, chemical properties of aerosol particles are not considered to change during transport. The composition of each fission product species on particles is kept track of over the entire period of accident time although the composition of each species at a given time is assumed to be constant regardless of the size class. The effects of the assumption that the aerosol composition is constant for each particle size class or source term depends upon the size

distribution of the source particles for each species. Recognizing that currently there exists no firm data base for speculating the size distribution of the aerosols formed in the core for each species, the constant composition assumption is justified. Even if such size distributions become known, the overall effects of the constant composition assumption on the final source terms have to be evaluated by comparing the two methods.

DATA

Extrapolating Data

Comment: Small-scale experiments may not represent large-scale phenomena.

Response: This is a concern which is raised when experiments performed at one set of conditions are used to describe what occurs at another. Perhaps the areas of greatest concern are the experiments concerned with release from fuel, and the flow fields expected to be obtained in the RCS under accident conditions. There is the possibility that phenomena which are unimportant or do not occur under the experimental conditions may become significant under accident conditions. The best precaution against a mismatch between experimental and actual mechanisms of importance is to perform a careful analysis of the full scale system (including the use of dimensionless groups) to be assured that the same mechanisms are expected to be operable in both cases. If the same mechanisms are operable and of measurable importance in both the experiment and full scale, then use of small scale experiments is warranted. Experiments at more than one scale will also help protect against mismatches.

Comment: Fission product release estimates are based on experiments using spent irradiated fuel.

Response: Fission product release estimates are based on experiments using spent irradiated fuel as well as simulants. These estimates have significant uncertainties associated with them due to a

variety of factors which the experiments performed do not fully address. These include composition of the carrier gas, rate of heating and time at temperature, extent of irradiation of the fuel, applied pressure, and the surface to mass ratio of the fuel. The importance of the uncertainty in the estimated release rates is noted in the discussion of the analyses and this area remains as one requiring further experimental investigation.

Comment: The postulated corium-concrete-water interaction is based on metal-concrete-water interaction experiments. Concern is that, even with the experimental basis, the postulated interactions may be inaccurate.

Response: The corium-concrete-water interaction models are based on observations with a variety of materials, real as well as simulants. Experiments have been conducted with oxidic as well as metallic melts. Nevertheless, there is substantial uncertainty in the prediction of such interactions. While the overall quantities of concrete eroded are believed to be reasonably predicted, the rates of downward versus sideward penetration of concrete by the corium are still uncertain. Also there is some uncertainty regarding the nature of the gaseous effluents that leave the corium-concrete interaction zone. The methods used in this study are those which correlate existing data as well as is currently possible and recognize that experiments are under way to expand the data base.

Scarcity

Comment: Data are scarce. Data are needed on subjects such as deposition velocities and high temperature thermodynamics.

Response: It is true that data are scarce for some of the phenomena being simulated. The available data on deposition velocities permit better than order of magnitude estimates to be made for Te, CsOH, and CsI. More precise data for Te would be especially useful, including better definition of its chemical form in the RCS. High temperature thermodynamic data will remain incomplete for the foreseeable future, so

any analyses performed will be forced to rely on the available data and estimate the potential influence of the unknowns on the results obtained.

CONTAINMENT LOADS FROM HYDROGEN

Mode of Burn

Comment: Two modes of hydrogen burning have been proposed for BWR's. In the first mode, hydrogen accumulates in a containment, ignites, and explodes. In the second mode, hydrogen passes through a suppression pool and burns as a diffusion flame. The correct mode must be determined to accurately predict a pressure spike in a containment.

Response: It is not clear that the mode of hydrogen burning can be uniquely defined. The mode of hydrogen burning will depend on the rate of its generation and release from the primary system, the atmosphere composition at the point of release, the availability and nature of ignition sources, etc. From the point of view of containment pressurization, the accumulation and coherent burning of hydrogen would result in the most severe pressurization. Standing diffusion flames may represent the most severe conditions from the point of view of equipment survival. Since it was the intent of this study to examine fission product transport for a variety of containment failure modes, some cases were examined which provided the maximum challenge to containment integrity. The BMI-2104 analyses are intended to represent a credible scenario within the range of accident uncertainties. The likelihoods of various loads and containment responses have been examined by the NRC's Containment Loads and Containment Performance Working Groups.

Extent of Overpressure

Comment: The AE sequence (LOCA) in the Peach Bottom analysis has 40 percent of the Zircaloy in the vessel reacting with water. Enough hydrogen is produced to overpressurize the containment and cause a containment failure. That a 40 percent Zircaloy reaction could produce enough

hydrogen to cause a containment failure is questionable. This calculation is conservative.

The models for hydrogen generation appear to predict an excessive amount of hydrogen. This may be caused by the following:

- An inadequate modeling of the Zircaloy channels
- A failure to recognize depleted steam or
- A single node used to represent the lower reactor plenum.

The excessive amount of hydrogen that is predicted implies an early containment failure; this is conservative.

Response: The predicted timing of containment failure in the Peach Bottom AE sequence is the result of a series of interrelated events associated with the start of slumping of the core into the vessel bottom head. Since the water level in this sequence is below the core as the result of the initial blowdown, there is very little reaction of the cladding during core heatup and initial onset of melting. At the onset of core slumping onto the support structures, much of the core is at very high temperatures but the cladding is largely unoxidized. The interaction of the slumped core debris with the water in the bottom of the reactor vessel produces substantial amounts of steam which flows past the overheated but yet unslumped core regions and results in significant cladding oxidation. The hydrogen and any unreacted steam are heated to high temperatures in the core region. Since this sequence has a large break in the primary piping, the hot gases are released to the containment with little cooling or holdup in the primary system. The rapid introduction of the steam and hydrogen into the drywell results in the transfer of a large fraction of the initial noncondensibles as well as hydrogen into the wetwell air space. With most of the noncondensibles in the wetwell, the containment pressure is predicted to reach the assumed failure level before significant condensation in the drywell and opening of the vacuum breakers can take place. The prediction of the early containment failure for this sequence does make the predicted results conservative. If the containment survives the pressure transient associated with core slumping, failure due to long term buildup of noncondensibles

would be predicted to take place much later in time, and the associated consequences would be less severe.

Comment: The volume of core Zircaloy simulated by the MARCH 2 code accounts not only for the cladding but also for the channel boxes. The extent to which Zircaloy and steam react depends both on the amount of Zircaloy and the surface area; this is important because the steam-Zircaloy reaction produces hydrogen that pressurizes a containment. The modeling of core Zircaloy is inaccurate.

Response: In the analyses presented in BMI-2104 the Zircaloy contained in the BWR channel boxes was included by artificially increasing the thickness of the cladding. While the reaction rate is indeed a function of surface area as well as temperature and steam availability, we have found the latter to be the controlling parameter under degraded core cooling conditions. The latest version of MARCH 2, which was not fully operational at the time of the BMI-2104 analyses, includes provisions for the explicit modeling of the BWR channel boxes and control blades. While our experience with this model is not extensive, analyses that we have conducted indicate that the explicit modeling of the channel boxes yields results that are substantially similar to those obtained using the above approximation. The reason why the added surface area makes little difference is the fact that the extent of reaction is steam limited. It may also be noted that with the channel box model the temperatures of the latter are found to follow quite closely the temperatures of the fuel rods.

Comment: In the Sequoyah analysis, the S_2HF sequences describes a series of hydrogen burns challenging the containment. The last burn after the ice is melted causes a containment failure. The severity of the simulated hydrogen burns is questionable. Nearly all of the zirconium should be oxidized; the depleted oxygen should limit hydrogen burning. The challenge to the containment may be overestimated.

Response: The MARCH analyses take into account the depletion of oxygen in the containment atmosphere as hydrogen burning takes place.

The zirconium is oxidized by reaction with water vapor. In the case of the Sequoyah S_2HF sequence, the series of hydrogen burns taking place relatively early in the sequence involve relatively small amounts of hydrogen and oxygen; thus substantial oxygen is still available. The reason for the prediction of the large pressure rise and containment failure from the last burn is the prediction that the burn will propagate to the upper compartment of the containment. The earlier burns which did not challenge containment integrity were confined to the lower compartment.

CESIUM IODIDE

Solubility

Comment: In the Surry (PWR) analysis, cesium iodide is considered to be undissolved in the primary system. Given the volume of water in the primary system, cesium iodide should be dissolved.

Response: The MARCH code analyses indicate that the RCS is dry during the core melting period of the accident, at least along the pathway from the core to the containment. There is no doubt that if the CsI contacts liquid water that it will dissolve. There is also some possibility that this dissolution will occur in the process of heteromolecular condensation in the RCS atmosphere since as a solute, CsI will lower the vapor pressure of the solvent water. The extent of this vapor pressure lowering would have to be quite dramatic in order for the small concentrations of CsI vapor to join with the water vapor to form a solution droplet at the conditions which characterize the RCS atmosphere during core melting. Data concerning the vapor pressure of H_2O over solutions of various composition at various temperatures were not located during these analyses. Such information would be required to quantitatively assess the possibility of liquid solution of CsI in H_2O forming under the conditions of interest. Recent experimental evidence tends to suggest that CsOH may well be present in liquid (solution) form even if only slight amounts of water vapor are present, and even at the elevated

temperatures which are of interest here. This information has only recently become available and was not factored into the analyses presented in Volumes I through VI. This phenomenon would perhaps be reflected in increased masses for the individual particles due to accretion of water vapor within the limits imposed by vapor pressure considerations, but this requires careful analysis to suggest the likely effect upon aerosol behavior in the RCS.

Volatility

Comment: In the Surry (PWR) analysis, the report states on page 6-31 that iodide is present as nonvolatile cesium iodide. However, cesium iodide is relatively volatile.

Response: Compared to I_2, CsI is relatively nonvolatile. Nevertheless, in the RCS analyses reported, the Cs and I release rates are computed using expressions based on experimental measurements of these species releases and the condensation of CsI in the RCS is calculated according to measured vapor pressures determined for this species. So the treatment given this material is consistent with its physical properties, but the statement referred to here may have misled the reader not familiar with the details of the analyses.

Form

Comment: The chemical form of iodine was considered to be entirely I_2. This was changed to CsI. Other forms of iodine, such as iodomethane, have not been considered. Little is known about the chemistry of iodine.

Response: In the reactor coolant system the iodine is considered to be present only as CsI. This is consistent with results of chemical equilibrium calculations performed for the range of conditions anticipated under accident conditions, specifically the H:O ratios and Cs:I ratios and system temperatures. In the containment, as discussed in Volume I of the report, it is reasonable to estimate that 0.05 percent

of the containment inventory is maintained airborne as volatile forms of iodine (excluding particles) until the time of containment failure. After containment failure a release of these volatile forms to the containment atmosphere is taken to be at a rate of 2×10^{-7} fraction per hour.

BORON CARBIDE

Comment: Boron carbide is omitted from consideration in the simulated BWR core inventory. This is questionable for two reasons:

(1) Because control rod boron can be oxidized to volatile boron compounds, boron carbide can contribute to the aerosol mass.

(2) Boron compounds can react with radionuclides; boron oxide reacts with cesium iodide to form cesium borates and iodine.

Boron compounds may have a significant influence on the chemical reactions of radionuclides such as cesium and iodine.

Response: In a high temperature steam environment, B_4C can apparently be oxidized by the steam to form a high boiling point (2130 K) oxide B_2O_3. This oxide may, in turn, react with the steam to form boric acids which would be more volatile and possibly react with CsI to liberate the I from this species. This is now recognized as an area in which detailed high temperature chemistry data are needed, and such a program should be undertaken to resolve the possibility that much of the iodine in a BWR may be transported as HI and therefore not be subject to significant retention in the RCS. The extent to which B_4C may contribute to the aerosol mass loading should also be addressed in such a study, although this is perceived to be a less immediate problem.

CORE-CONCRETE INTERACTION

General

Comment: The postulated interaction of a molten core, concrete, and water may be optimistic because

(1) The proposed scrubbing efficiency is larger than that used for calculation on the suppression pool at Peach Bottom or Grand Gulf

(2) Film boiling should occur at the molten corium-water interface

(3) Bubble sizes should be described by a Taylor instability calculation taking into account the changes in surface tension from impurities in the water.

Response: The generation rate and subsequent scrubbing of fission products during a core-concrete interaction is one of the highly complex phenomena, and the comment is indeed relevant. The model used in the analysis for describing interaction of a molten core, concrete, and water may not be sufficiently rigorous and uncertainties in the model are believed to be large due to a number of simplifying assumptions. A limited number of calculations performed incorporating varied chemistry model features and thermodynamic conditions have also shown that the calculated results can change significantly. It is expected that as various tests which are being conducted at Sandia, Brookhaven, and in Europe progress and the results become available, the present thermal hydraulic and chemical process models will be greatly improved. The scrubbing efficiency depends upon many factors including the size distribution of aerosol particles, the assumed bubble size, and thermal hydraulic conditions. The film boiling phenomenon and the gas bubble size can also be important in determining the overall amount of the aerosol released to the containment from the core-concrete interaction. The effects of the film boiling phenomenon on the particle size and the scrubbing of fission products are not well understood and applicability of Taylor's instability calculation to water above the core-concrete interaction situation has not been validated. For this reason, the analysis

utilizes the values for the bubble size and the particle size distribution that were observed from recent tests rather than attempting to resolve rigorously the exact mechanism.

Tellurium

Comment: Current experiments indicate how tellurium reacts as it is released from fuel and how it reacts during a core-concrete interaction. Reactions of tellurium and concrete have not been studied. Tellurium-concrete reactions may have a significant influence on tellurium release.

Response: The authors do not believe that there is currently general agreement regarding the reactions Te undergoes as it is released from fuel. It is true that recent experiments have addressed this item, but it is not clear whether the experimental results reported at this time are directly applicable to the current analysis. This is an area requiring further experimental investigation, and it is agreed that Te-concrete (as well as Te-Zr and others) reactions may significantly influence the fate of the Te in LWR accidents.

Decomposition Products

Comment: Carbon dioxide is a decomposition product of the core-concrete interaction. The carbon monoxide and hydrogen also should be listed as decomposition products.

Response: Carbon monoxide and hydrogen as well as carbon dioxide are possible decomposition products of concrete and are considered in our analyses.

RADIOLYTIC EFFECTS

Reactions

Comment: The fate of cesium after the decay of iodine in CsI is unaccounted for. The mobility of cesium may be much different than predicted because a significant fraction of cesium may be liberated from iodide by isotope decay.

Response: The BMI-2104 analyses do not include the effect of radionuclide transmutation on the transport behavior of radionuclide groups. In the case of cesium daughter products which have decayed from iodine, an explicit treatment of the decay chain would only lead to different results if in an intermediate stage of decay the nuclide were to spend significant time as a xenon isotope. The predicted release rates from fuel for cesium, iodine, and xenon species are essentially the same. The predominant chemical forms of iodine (CsI) and cesium (CsOH and CsI) assumed in the BMI-2104 analyses also have very similar transport behavior. Xenon, on the other hand, is assumed to be transported without attenuation. Only the chain $I^{135} \rightarrow Xe^{135} \rightarrow Cs^{135}$ appears to have a xenon isotope with a reasonably long half-life. The resulting activity of Cs^{135} is not high enough, however, to represent a significant health hazard.

Transmutating

Comment: Isotopes transmutate into different fission products. For long-lived isotopes, the contribution of transmutation is insignificant. For short-lived isotopes, the gain and loss of chemical species may be significant.

Response: Radionuclide transmutation is not treated explicitly in the BMI-2104 analyses. In the ex-plant consequence analyses (CRAC-2) that are being performed for the NRC's source term integrating study, radionuclide decay is accounted for in the composition of each group. Two types of errors can be introduced by ignoring radionuclide decay

during in-plant transport. First, chemical processes are a function of the total mass of an element present. Although the mass of each element can change as a function of time due to radioactive decay, in general the bulk of the mass present is represented by stable or long-lived isotopes. The potentially more important error arises from decay chains in which the parent and daughter have substantially different release or transport behavior. The potential error introduced by not treating the decay chains explicitly for in-plant transport has been investigated in the QUEST study. The chain $Te^{132} \rightarrow I^{132}$ was identified as one in which an explicit treatment could affect the results. Since the treatment of decay chains is well within the state of technology, it is planned that this will be taken into account in future improvements in source term methodology and, specifically, will be included in the next generation of fission product transport codes developed under NRC sponsorship.

Charging

Comment: Inside a post-accident containment, radiation fields are likely to exist. The radiation fields should charge aerosols. Though the entire aerosol would be electrically neutral, the positive and negative ions would have different mobilities. Thus, an electrical charge should modify the behavior of an aerosol. This phenomenon has been given inadequate attention.

Response: The effects of radioactivity upon aerosol charging have been examined previously by Battelle (Reed, et al, J. of Aerosol Sci., p 457, 1977; BMI-NUREG 1943 report). The charge distribution on the aerosol particles was shown to depend upon the activity of radiological species comprising particles and the aerosol concentration. The results showed that ion production rate from the particles themselves is in general sufficient to discharge the aerosol particles and additional radiation sources (gases) would increase the discharge rate. Further, the mean number of charges on particles decreases with increasing particle concentration. The calculations indicated that the average number of charges on the particles is quite low. Therefore, we feel that the

difference in the mobilities of negative and positive ions is not likely to be significant enough to affect the aerosol behavior to a significant extent.

Nucleation

Comment: Ionic nucleation could occur in the reactor coolant system (or in the containment).

Response: This mechanism is likely to be the dominant contributor to the formation of new particles in the region of the core. Condensation of vapors onto the critical nuclei is by far the dominant means of formation of the aerosol particle mass. For the very involatile species which constitute the majority of the aerosol mass, the condensation is expected to be essentially complete very near the point of release of the vapor from the melting core. For this and other reasons given above, nucleation is not modeled in the RCS aerosol simulations.

HYDRATED INORGANIC MOLECULES

Comment: At high temperatures and pressures, an inorganic molecule may be hydrated. Because the hydrated molecule has a lower vapor pressure than the unhydrated molecule, the hydrated molecule is more mobile than the unhydrated molecule. The mobility of some fission products is underestimated because the hydrating phenomenon is unaccounted for.

Response: There are two mechanisms which can lead to mass transport enhancement in a steam environment. First, at high steam pressures and high temperatures there will be transport by a solvation route. Synthetic quartz is the most well known process utilizing this principle, but the process can cause transport of other species equally as well. Solubility would be a strong function of both temperature and steam pressure and while this process may enhance mass transport in the vapor phase during core melting, it is not obvious that it would enhance release from the RCS because as the steam blows down to containment conditions deposition of these solvated species could occur.

Another factor however is that a great number of gases of stable hydroxides can form. One notable case would be for barium. The reaction $BaO(c) + H_2O(g) = Ba(OH)_2$ is a very important reaction. Literature values for the vapor pressure of BaO show great discrepancies, and these discrepancies are likely caused by trace moisture forming the gaseous hydroxide as a competing vaporization process. Uranium oxide itself forms a gaseous hydroxide which has been observed as $UO_2(OH)_2$.

To date, although these types of reactions may lead to more transport of certain species, they have not been factored into release computations. This is partly because the data base is so limited and partially because there are believed by the authors to be second-order effects.

STRUCTURE OF MODELS

Number and Arrangement of Control Volumes

Comment: The predicted source terms are influenced by the number of control volumes and/or nodal points used to mathematically represent various parts of a plant; i.e., core reactor coolant system, containment.

The control volumes are connected in series to represent the reactor systems. No parallel connections are made. Though the natural circulation within a control volume can be adequately represented, the natural circulation in a reactor system (many control volumes) cannot be easily represented.

Response: The nodalization of systems in numerical simulations is a standard area of concern for all fields. For codes which permit very fine nodalization, the approach typically followed to address this concern is to use successively finer nodalization (more nodes) until there is no further significant impact on the solution obtained. This is, however, not possible for the analyses under discussion here. In some limited instances, the RCS was nodalized differently (e.g., a single long pipe for a V sequence versus five pipe segments in series; one volume

for the entire upper plenum versus four interconnected volumes) and the results of TRAP-MELT calculations compared. Small changes were effected in the results obtained, but these did not alter the retention of fission products in a significant way. One reason for the lack of significant effects in these results can be readily postulated. Namely, one must be able to supply the boundary conditions for the problem being solved at each of the grid points used. That is, one must be able to supply to the RCS transport and deposition code the temperatures and flow rates at each node if nodalization is to be observed to have a significant impact on the results. This is not likely to present much difficulty in the case of a straight pipe such as in the V sequence. But specification of finer detail of flow and temperature fields within the RPV remains as an extremely difficult proposition. It should be noted that work in this area is currently under way, both in terms of highly detailed simulation of the thermal hydraulics for a limited set of accident sequences in a fixed vessel and in terms of improving the adequacy of the coarser formulations of the problem as used in the analyses in this report. This requires solution of the natural circulation and convection currents which may exist between the various geometrical volumes in the RPV and, indeed, throughout the entire RCS in some instances. One further point which should be made in this connection is that it is not currently obvious what the implications of modeling improvements along these lines are for the ultimate retention of fission products in the RCS.

REPRESENTATIVENESS OF MODELS

Vessel Dome

Comment: The upper dome region in the reactor vessel of the Surry reactor is isolated from most of the flow in the vessel. The same may not be true at other plants.

Response: We agree that the relative isolation of the upper dome region from the rest of the vessel in the Surry reactor is not necessarily representative of all PWR designs. If the reactor vessel

design was such as to allow mixing of flow up into the upper dome region, the effect would be to reduce temperatures in the remainder of the upper plenum and would probably lead to increased fission product retention.

Equipment

Comment: Fire protection systems are accounted for only for Mark I reactors.

Response: Fire protection systems, if activated, could contribute to fission product removal wherever they may be encountered. In the context of the BMI-2104 analyses, fire protection systems appeared to have the most potential impact in the secondary containment of the Mark I BWR design. The effect of fire protection systems has been examined in the ORNL SASA program. In the BMI-2104 analyses no credit was taken for the effect of sprays on source term radiation because of major uncertainties regarding the magnitude and distribution of spray flow.

Concrete

Comment: The composition of the concrete in a reactor cavity determines the extent to which many important reactions occur. One such reaction is the conversion of carbonates to carbon dioxide. When the concrete composition is unknown, the limestone content is assigned a value of 80 percent. This leads to 36 percent carbon dioxide in the concrete. In the Grand Gulf plant, the reactor cavity concrete is 50 percent to 60 percent carbonates. This leads to about 22 percent carbon dioxide in the concrete. There the codes overpredict the amount of CO_2 generated by a core melt to challenge a containment. The predicted consequences are overestimates.

Response: In our analyses we have tried to use the best information available to us. In case of concrete, the actual compositions are frequently not known or available to us; in such cases we have assumed default composition for representative generic types of concrete aggregates. In case of the Grand Gulf analyses, if the actual composition of

the concrete contains less carbonate than we have assumed, the effect on the predicted results would be minor and affect only the TQUV sequence. In the latter, containment failure was predicted to take place due to long term buildup of noncondensibles. With less carbonate in the concrete the time of containment failure could be delayed beyond the time predicted in our analyses. In this sequence however, all the fission products pass through the suppression pool and are quite effectively scrubbed before they are available for release to the environment. Thus substantial variations in the predicted time of containment failure would have a relatively minor effect on the predicted source terms. In the other sequences considered in our analyses for the Grand Gulf design, concrete decomposition does not contribute to containment failure, thus changes in the concrete composition would not lead to large differences in the predicted consequences.

CONSISTENCY OF MODELS

Comment: Some portions of the models are detailed and have a firm scientific basis. Other portions are based on assumptions. This is analogous to measuring with a yardstick and a micrometer. The predicted consequences are only as good as the weakest significant assumption.

Response: It is recognized that not all portions of all models are developed or validated to the same degree. The intent of this study was to use state-of-the-art procedures and to demonstrate that a more mechanistic approach to source term predictions was indeed possible at this time. In fact, the identification of less well developed portions of models was an important result of this study and has helped focus research efforts.

Through the use of parametric analyses in this study and in the uncertainty analyses (QUEST) being performed at Sandia, evaluations of the impact of assumptions and uncertainties have been and are being performed. Such studies have helped to quantify source term variations resulting from assumptions and less well developed models.

Comment: Some portions of the models and some assumptions are "conservative", yet the source term estimates are considered "best estimates". For example:

(1) The dry pathway assumption appears to be invalid considering the amount of water in a reactor system.

(2) The containment sprays are unaccounted for in the analyses of the Grand Gulf station.

(3) The instantaneous vessel failure assumption is unrealistic.

(4) Interactions of a molten core with the lower vessel internals are ignored.

The assumptions need to be justified.

Response: There was no systematic attempt to choose conservative models or assumptions in performing this study. In fact, it is often difficult, if not impossible, to know whether any specific assumption will lead to higher or lower releases. The effects of the timing of events relative to fission product transport behavior often has an impact that is not expected or understood until one examines the details of calculations after their completion. Therefore, the progress of any accident sequence and the accompanying fission product transport was analyzed using a best estimate for each assumption or best available model for any process. The various examples will be discussed individually.

(1) A dry pathway was not assumed for the RCS but was calculated by the MARCH code. By the time of core melting, most of the water has been boiled away in the various sequences. If any water remains, it is in low points away from the pathways for fission product transport. Where water was predicted to exist in a location of any significance, such as overlying the core debris for a period of time in some sequences, accounting was made for its presence. The containment was calculated to be quite "wet" in most portions of most sequences and steam condensation and fog formation were included in the analyses.

(2) Containment sprays were assumed to operate in sequences where the progression of the accident indicated that they would be triggered. Containment sprays were always assumed to be inoperable after containment failure because of the possibility that the failure process would damage the spray system.

(3) In the BMI-2104 analyses failure was assumed to occur over one calculational time step rather than instantaneously. Corresponding to this, the time for fuel discharge has been estimated to occur over 10's of seconds even with small holes. Even so, failure modes are largely unknown at this time. The variety of sequences studied, including the V sequence and isolation failures which have smaller failure openings in the containment, provide a basis for estimating the effects of alternative failure modes.

(4) Interactions of a molten core with lower vessel internals were included in the MARCH analyses of melt-through. The heating and melting of these lower structures were included in the thermal analyses and the resulting molten steel was included in defining the melt composition reaching the concrete basemat.

ALTERNATE SEQUENCES FROM PIPE BREAKS

Comment: The location of a pipe break will determine both the release pathway and the fission product removal mechanisms acting along the pathway. Hence, the point at which a pipe breaks has an influence on the consequences of a reactor accident. In some cases, the influence may be minor; that is, a break anywhere along a section of a pipe may give rise to similar pathways. In other cases, the influence may be significant; that is, different break points lead to different pathways. The sequence definitions inadequately account for a variety of break points.

Response: We agree that the location of the break in the primary piping can influence the consequences of the accident. This can be seen by comparison of the results for PWR hot leg and cold leg break

cases. The latter would typically benefit from steam generator retention of fission products, whereas the former would not. Obviously it has not been possible for us to investigate all possible break locations; we have tried to do sufficient analyses to gain insight on the influence on such factors as break location.

Comment: In the Surry PWR, a break in a large diameter pipe of the RHR system may occur in a section of the pipe in a water tight compartment. The compartment would flood. The fission products exiting the break would pass through about 3 ft of water. Fission products would be partially scrubbed by the water.

Response: We have considered this variation of the V sequence for the Surry design and the results including consideration of fission product scrubbing by water if the break location is submerged are given in Volume V of BMI-2104. The water scrubbing under the conditions assumed resulted in a reduction of the predicted environmental releases by a factor of about five.

ALTERNATE SEQUENCE FROM OPERATOR ACTION

Comment: By the definition of a particular sequence, a specific combination of events is assumed. The events represent the most likely behavior of the reactor hardware with little account of human factors. Human factors include operator intervention and operating procedures. An operator may relieve excess pressure in a drywell by venting the drywell pressure through the standby gas treatment system. The sequences define the most likely behavior of unattended hardware.

Response: The accident sequences as we have analyzed them have been defined on the basis of various risk assessments. These sequences do not exclude operator intervention; the operator actions considered are generally those that are clearly indicated. Specific examples of operator action include switchover from the injection to the recirculation modes of safety system operation, and actuation of the primary system depressurization system. The sequence definitions clearly

do not consider operator actions for which there are no prescribed procedures or ones that may involve extraordinary measures. In the particular example cited in the comment, it is not clear that venting of the containment in the presence of high radioactivity would be a reasonable expected operator action or that drywell venting capability available would be sufficient to prevent containment pressurization.

ALTERNATE SEQUENCES FROM PATHWAYS

Comment: The annular space in the reactor cavity may become a fission product pathway when the steel containment fails at some location and the shield building fails at a different location. A significant pathway arises where fission products could be removed. Furthermore, the reactor cavity is vented through an air cleaning system. Thus the shield building need not fail for a pathway to arise through the reactor cavity annulus. These alternate pathways are not accounted for in any sequence though they are significant.

Response: Clearly a variety of paths can be postulated by which fission products are released to the environment. The significance of any particular transport path cannot be a priori determined; it must be evaluated in the context of a particular design and accident sequence. We have not attempted to assess all possible release paths.

Comment: The Mark III containment design is less vulnerable to an inadvertent bypass of the suppression pool than the Mark I and Mark II designs. However, in Mark III containment, conduits communicate to the suppression pool; these conduits are potential bypasses. Because a suppression pool removes significant amounts of fission products, a bypass has adverse consequences. The definitions of the sequences neglect the bypasses, hence, the computer codes fall short of predicting some important consequences.

Response: In response to comments of this type, we have explicitly considered two cases of suppression pool bypass for the Mark III BWR design. The first case was based on nominal leakage between the

drywell and the outer containment, and the second was based on the assumption of a stuck open vacuum breaker. The results of these analyses are given in Volume III of BMI-2104.

Comment: An ice condenser may be inadvertently bypassed because of misaligned valves or vents. The alternate routes are not considered in any of the sequences.

Response: The ice condenser is basically a passive device with the ice condenser doors being opened by pressure buildup in the lower compartment. There is no alignment or positioning of valves or vents associated with the operation of the ice condenser; thus it is difficult to envision major bypass and loss of the ice condenser function during an accident. Some leakage between the upper and lower compartments without passing through the ice condenser is possible, but would not have a major effect on the predicted behavior. In some additional analyses conducted subsequent to BMI-2104, we have investigated the consequences of containment isolation failures; in the case of the ice condenser the isolation failure was assumed to be located in the lower compartment and the leakage through this opening did not go through the ice condenser.

Comment: In many of the BWR and PWR sequences, the fission product release pathways begin at the core and pass through the upper plenum. However, eddy currents may produce a pathway through the lower plenum, especially when the core slumps and melts through the vessel. The release pathways through the lower plenum are not taken into account.

Response: During initial core heatup and melting, the lower plenum of the reactor vessel will be filled with water and thus not present a path for fission product transport. After the core has slumped into the bottom head there could be a release path from the lower head to other parts of the primary system and to the containment. The significance of such paths is not clear. In a BWR large break loss-of-coolant-accident, for example, the lower plenum may represent a more direct path out the break and into the containment. In PWR transient

sequences, on the other hand, the path through the lower plenum may permit some of the fission products to be released to cooler portions of the primary system. The significance of the pathway through the lower plenum would, of course, depend on how long it was available. If the head fails shortly after core collapse, they may not be important; if there is a long period of time between core collapse and vessel head failure, considerable movement of fission products could take place through this path. These alternate pathways were not considered in our analyses.

Comment: In some PWR containment designs, the reactor cavity may communicate with the containment. Water accumulating on the floor of a containment may find its way into the cavity. When the core melts through the vessel, a large volume of water will turn into steam and raise the containment pressure enough to cause an early containment failure. The possibility of such a pathway and its consequences have been given little attention.

Response: The relationship between the containment sump and the reactor cavity is one of the key aspects of the containment description and is addressed in some detail in the preparation of the input for the thermal hydraulic analyses. The availability and quantity of water in the reactor cavity is one of the factors that affects containment overpressurization as well as the likelihood of attaining a coolable debris configuration. Such considerations are discussed in some length in the Sequoyah analyses in Volume IV of BMI-2104, as well as being included in the analyses for the other designs.

ALTERNATIVE SEQUENCES FROM EQUIPMENT PERFORMANCE

Comment: The electrical penetration of a containment or a drywell may fail before the concrete structure fails; the penetration seals and insulation may degrade under intense heat, radiation, and pressure. A release path through failed penetrations may be either diffuse or concentrated, depending on how many penetrations fail. The

release path would likely pass through the auxiliary building because that is where most of the penetrations lead. The analyses have focused on the failure of the containment structure and neglected a failure of the penetrations.

Response: The BMI-2104 analyses focused on the failure of the containment structure as the comment notes. Additional analyses performed subsequently have addressed the possible failure of containment penetrations and the leak-before-break containment response model. The results of these later analyses indicate the expected effects with higher leak flows leading to higher source terms.

Comment: Steam and fission products are dispersed throughout the ice baskets of an ice condenser by turbulent flow. No structures force the steam and fission products through the stacks of baskets. Thus, a major fraction of the steam and fission products would flow through the ice condenser but around the ice baskets. The ice condenser would be effectively bypassed even though steam and fission products enter the volume of the ice bed.

Response: The mentioned flow pattern may in fact occur in the ice bed. The cross-sectional area ratio and the corresponding flow split between open areas and the ice baskets have not been accounted for in our analyses. It is true that the mechanical scrubbing mechanisms such as diffusion, inertial impaction, and turbulent deposition considered in our analyses would be adversely affected if the above is considered. However, these mechanisms are generally weak. On the other hand, the diffusiophoresis mechanism which is found to be much more important than the above mechanical deposition in the ice bed may not be influenced significantly by the mentioned bypass possibility.

Comment: In some sequences a failure of the containment building leads to a failure of jet pumps. Some of the pumps may survive a containment failure and continue to operate. Often an extensive failure of a safety system is assumed even when at least a portion of the safety

system should remain functional. These assumptions lead to a conservative source term prediction.

Response: Containment failure is assumed to lead to the failure of the safety system pumps when such failure leads to the flashing of the sump from which these pumps are taking suction. These pumps are not designed to operate for prolonged periods in the cavitating mode. Should the pumps survive the containment depressurization and sump flashing, then the sequence would probably not involve core damage. This possibility is recognized.

Comment: When containment recirculation flow fails, reverse flow through the ice beds may occur. This is not considered in the sequence.

Response: Ice condensers are equipped with one-way doors to prevent reverse flow through them. These doors would have to fail and the pressure in the upper containment would have to be greater than that in the lower compartment to lead to reverse flow through the ice bed. This combination of eventualities has not been considered in our analyses.

SELECTION OF CORE MELT MODELS

Comment: Three scenarios of a core melt have been developed:

(1) Coherent drop -- the core instantaneously slumps as one unit when 75 percent of the core is liquefied.

(2) Downward gradual slump model -- radial sections of a core are calculated to melt independently. Upper nodal regions in a radial section heat lower nodal regions. The melt progresses downward.

(3) Upward gradual slump model -- radial sections of a core are calculated to melt independently. Lower nodal regions in a radial section heat upper nodal regions. The melt progresses upward.

The three scenarios reflect an evolving process of developing the computer models. The coherent drop model is no longer used. Instead, the gradual slump models are used. However, these models were developed more on

intuition than on facts. Thus, even the gradual slump models are arbitrary.

Response: Our use of the meltdown models and related options in the MARCH code have evolved over a period in time and attempt to reflect the current understanding of core meltdown behavior. The gradual slumping model, used in the BMI-2104 analyses, represents the latest knowledge about the behavior of a degraded core.

Comment: In the modeling of a core meltdown, a gradual slump is assumed. However, no check is made to ensure that a melted core can be physically accommodated in a lower node (i.e., one that has not yet melted).

Response: The meltdown models utilized are not mechanistic representations of fuel movement, but rather simulations of energy redistribution in a core as it heats up and melts. As core nodes reach melting and continue to heat, the temperature of the melt would tend to rise above the melting point. The modeling assumes that the energy above that required to keep the region at the melting point can mix with the yet unmolten nodes in the immediate vicinity as if by a combination of molten fuel motion, convection, and conduction. Depending on the overall energy balance at any point in time, the molten region can grow, stay constant, or decrease. Melt progression does not require or imply movement of the entire molten region into lower nodes.

Comment: Current models have a core melting homogeneously. Cladding melts at a lower temperature than UO_2. The models do not appear to consider a failure of the core structural material prior to the melting of fuel. A collapse of the core may influence the melting process.

Response: The melting point of the cladding is indeed lower than that of the fuel. It has been observed, however, that the melting cladding will dissolve some of the uranium dioxide fuel. The liquidus temperature for the $Zr\text{-}UO_2$ system has been observed to vary with the local composition and the extent of zirconium oxidation. In our analyses we have utilized an effective melting point for the core materials that

is between the melting point of the cladding and that of the uranium dioxide fuel.

The MARCH modeling does permit structural failure of support structures and attendant core collapse before complete melting of the fuel. Core support failure can be the result of overheating of the core barrel due to thermal radiation or overheating of the lower support structures by slumped fuel. The former would be more applicable to PWR's, while the latter could conceivably be predicted for both PWR's and BWR's. While failure of core support structures may be predicted prior to complete core melting, they would not be expected in the absence of appreciable core melting. Early failure of core support structures can indeed influence the subsequent melting process. Such situations were encountered in some of the Mark I BWR sequences treated in BMI-2104, though it is typically not the case.

Comment: In the modeling of a core meltdown, a gradual slump is assumed. No geometric changes are assumed when accounting for surface areas and flow areas. The modeling of a gradual slump is inaccurate or incomplete.

Response: The meltdown model and metal-water reaction options utilized in our analyses reflect our understanding of the expected behavior of the core under the conditions expected in these accidents. This is by no means an exact representation of how the core may behave. A variety of alternate descriptions may also be plausible. What we have attempted to provide is a self consistent and physically realizable description; a high degree of uncertainty in any such description is acknowledged.

As the fuel in the core begins to lose its initial geometry, a wide spectrum of effects can take place. The most obvious perhaps is blockage of flow channels. It is not clear, however, that such blockage need be total and coherent across an entire core region. By the time fuel melting is reached, there will be substantial oxidation of the cladding; melting and movement of the cladding can expose fresh unreacted surfaces to steam as well as leading to the mixing of such cladding with

the fuel. If any runoff of the molten cladding down the fuel rods does take place, this runoff will probably be unreacted cladding, offering the potential for locally enhanced oxidation ahead of the melt front. Our experience indicates that variations in many of the individual modeling assumptions available do not have a major effect on the overall accident progression.

SELECTION OF SEQUENCES

Comment: The criteria for selecting sequences should be reexamined. Examples show that the selection criteria are inconsistent and incomplete.

- No small break LOCA sequences are considered for Mark I BWR's.
- No LOCA sequences are considered for Mark III BWR's.
- No emergency safety feature power failures are considered for Mark I BWR's or Mark III BWR's.
- Including the Zion plan in source terms analysis raises the issue of including external event initiators into all analysis.
- Operator actions may play a significant role in initiating and propagating sequences.
- A V sequence is not considered for Zion because the risk is assumed to be low.
- A "complete" range of sequences should be clearly defined.
- Some sequences are selected on the basis of risk while other sequences are selected because they involve plant-specific features.

The selection criteria result in analyses that are incomplete and that cannot be compared.

Response: The scope of our effort did not provide for the complete development of the risk profile for each of the plant designs considered. Thus we have had to be selective in the number and type of sequences that could be considered in detail. We have attempted in our selection to provide a perspective covering a variety of initiating

events, reactor designs, and safety feature availabilities. We agree that a more complete set of sequences would be desirable and could contribute to a better understanding of the phenomena involved.

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