containment pressure was high enough, some of the airborne radioactive material was released to the ground where it was scrubbed (DF of 1000 except for noble gases) before escaping to the air. In the most probable core meltdown sequences in the pressurized water reactor above-ground failure was predicted to be unlikely. (It is not clear from where the authors' $<2\%$ came.) For sequences in which above-ground failure of the containment is averted, the release of radioactive material to the environment is much less (by a factor of 10^4 to 10^5) than for the more severe accidents. In fact, using indices of relative hazard for different radionuclides, the predicted consequences of the most likely core meltdown sequences in WASH-1400 are less than the actual off-site population exposure of the Three Mile Island (TMI) Unit 2 accident. In contrast, the Mark I boiling water reactor (BWR) containment design was predicted to have an atmospheric failure for all core meltdown sequences. Again, a broad range of potential consequences was obtained in WASH-1400, ranging from little deposition for direct releases to the atmosphere to substantial retention due to the effectiveness of the suppression pool in scrubbing fission products and because of the credit taken in the analysis for deposition external to the primary containment along the pathway of the release to the environment.

7. The authors base much of their argument on the history of reactor accidents and the results of destructive experiments. They fail to take note of the fact that neither the reactor designs nor the scenarios in these accidents and experiments resemble the accident sequences that have been found to be important in WASH-1400 as well as a number of subsequent risk studies. The reader should recognize that the types of accidents at issue (those predicted to dominate risk) are believed to be very rare events (e.g., 1:200 000 reactor years); statistically they represent only a small fraction of all possible core meltdown accidents. In these events, combinations of failures of engineered safety features are predicted to result in early above-ground failure of the containment building. Despite their very low probability, but because of their potentially high consequences, these sequences were predicted to dominate public risk in WASH-1400 as well as other more recent studies. The magnitude of possible retention mechanisms must be evaluated for the specific conditions expected in these sequences. Accidents that have occurred in 400 reactor years of LWR operation have had little similarity to the behavior expected in these rare events. In particular, the TMI accident is quite unlike the risk-dominant accidents of WASH-1400 and a direct comparison is inappropriate.

8. Table I of Ref. 1 misrepresents the WASH-1400 assumptions concerning fission product release to the environment. Specific examples of apparent misunderstandings are:

a. Significant washout of released fission products by water in the primary system was indeed considered when the release path was through water, particularly in the BWR analyses.

b. Deposition of released radioactivity within the containment as well as external to it, where appropriate, was specifically considered; deposition external to the primary containment was found to significantly reduce releases to the environment in many BWR accident sequences.

c. Fission product removal due to flow through suppression pools was explicitly considered.

d. Aerosol behavior was evaluated by means of the CORRAL code, which is based on large-scale containment experiments.

9. No basis is given to support the iodine attenuation factors assumed in Table III of Ref. 1.

As stated at the outset, we agree that more effort is needed in obtaining a better understanding of fission product behavior in reactor accidents to serve as the basis of safety judgments as well as improved risk assessments. While risk assessments should be conducted as realistically as possible, care must be taken that the assumed realism can be well supported. Due note must be taken of the uncertainties associated with the prediction of both the probabilities and the consequences of reactor accidents. Since these uncertainties are large, the formulation of safety judgments must err on the side of safety by taking into account all possible outcomes at some high level of confidence. A "realistic" or "best estimate" evaluation would, after all, underpredict reactor accident consequences much of the time. Only if the uncertainties associated with such a "best estimate" or "realistic" evaluation are small, would such an approach be acceptable. The concept of "upper limit of possible attenuation factors" as used in Ref. 1 has meaning only insofar as it may help to define the range of uncertainties. Clearly, a safety judgment or meaningful risk assessment cannot be made on the basis of possible attenuation factors that may, in fact, be unavailable most of the time.

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REPLY TO "COMMENTS ON 'REALISTIC ESTIMATES OF THE CONSEQUENCES OF NUCLEAR ACCIDENTS' "

Cybulskis et al.[1] call us[2] to task for leaving "the impression that there is a body of evidence that indicates that fission product release estimates are greatly overestimated." That was exactly our intention. All the empirical evidence to date suggests that the predicted consequences of reactor accidents are too high. On the other hand, there is no accident or *integral* experiment that shows the computer models they advocate give accurate results.

It appears from reading their last paragraph that they did not appreciate a major point of our paper. Calculations of the consequences of reactor accidents are now being used for many things such as the setting of evacuation policies. An evacuation itself encompasses a considerable risk—of death, injury, economic loss, and emotional suffering to the general public. In such a circumstance, a calculation is not conservative if it overestimates the direct consequences of the accident. Such an overestimate may lead to the wrong actions, rather than actions consistent with the risks involved.

We believe that the bulk of the evidence indicates that the fission product release estimates for major light water reactor accidents are grossly overstated in WASH-1400 (Ref. 2). This is due to several reasons—oversimplification of the actual geometry of the plant, the simplified modeling of the accidents, and the neglecting of various physical and chemical phenomena.

This last reason has been especially underestimated in past studies. In particular, the assumption that there is no attenuation of fission product aerosols in the primary system leads to extremely large conservatisms. The first of these conservatisms results from assuming the instantaneous injection and mixing of the fission products with the gas in the containment building. This dilutes the aerosol concentration to the point where large diameter agglomerates (>100 μm) would not form. The second conservatism, which is more important, is the neglecting of the process itself of intense agglomeration to large aerosol sizes. This occurs within seconds when the aerosol concentration is high (>100 g/m^3) (see Fig. 1). Agglomeration reduces the aerosol source term by at least an order of magnitude prior to injection into the containment.

The concentration that should be used for this type of analysis is the local concentration in a heterogeneous zone close to the source where the aerosol first condenses. It is not the uniformly mixed volume of the reactor vessel or containment building. (This class of phenomena can, for example, be observed in the vapor phase burning of sodium vapor above a sodium pool. Here the burning zone is ~1 mm thick immediately above the pool surface. Under these circumstances, >75% of the sodium aerosol grow to large sizes and fall back into the pool. The "released fraction" of aerosols amounts to <25% of the sodium burned and has a log normal size distribution with a 2- to 3-μm mass median diameter and a σ = 1.8 to 2.0.)

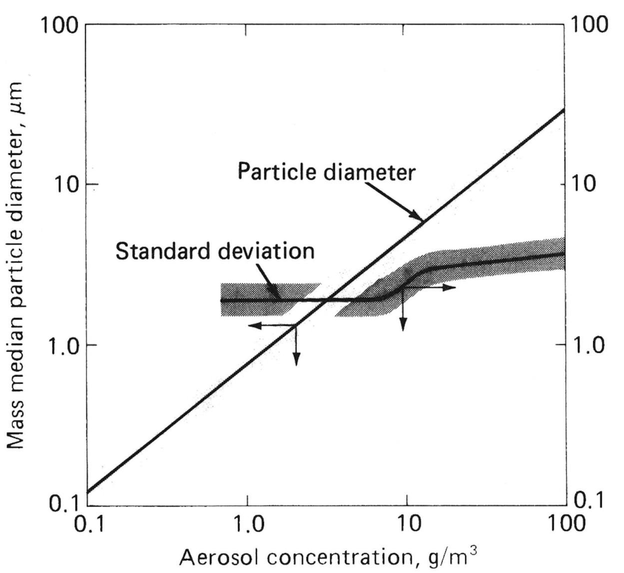

Fig. 1. Aerosol concentration, g/m^3 (Ref. 4).

Cybulskis et al.[1] claim that the attenuation factors we spoke of in our paper may be "unavailable most of the time." We disagree. Things like heat capacity, solubility of fission products, or the behavior of aerosols are natural phenomena. They are not dependent on a particular accident scenario. They are always available as attenuation factors.

The uncertainties in fission product release are indeed large, as Cybulskis et al. note. However, the error is not symmetric around the calculated values; it is highly improbable that the consequences are underestimated by more than a few percent, yet it is highly likely that they are overestimated by several orders of magnitude. To produce realistic estimates, it is not enough to validate one computer code against another, or to rely on sensitivity studies to test for modeling shortcomings.

We would like to comment in some detail on their specific points (numbering system is the same as in Ref. 1):

1. We have examined the top contributors to risk in WASH-1400 and identified "areas of conservatism" which lead to overestimates, by at least a factor of 2, in the results (see Fig. 2). There are five or more such areas for *each dominant sequence*. This results in much more than an order of magnitude difference in the calculated risk.

2. We agree that core meltdown accidents must be considered mechanistically. However, we disagree with Cybulskis et al. in the sequence and timing of the mechanistic models. Table I gives a set of assumptions that we believe are more realistic than those used in WASH-1400. If these assumptions were to be used, much lower consequences would be calculated. In particular, when the path from the core region to containment is filled with superheated steam, the so-called "dry" accident, other attenuation phenomena become important:

   a. Temperatures in the upper region of the pressure vessel will be considerably lower than in the core, allowing condensation of fission products, if not the steam.

   b. Fission products that do not condense in the primary system will be transported into a lower compartment, not the main compartment.

   c. The water originally in the primary system will be condensing at this location due to the heat capacity of the building, giving rise to wet and steamy conditions.

The NAUA code and similar aerosol codes have been used incorrectly to calculate the risk dominant reactor sequences: the high aerosol concentrations (~1000 g/m^3) that would exist in the primary system were not analyzed. The assumption is made that the aerosol starts out as finely divided, micron size particulates, uniformly distributed throughout containment. As a result, rapid agglomeration to large particle size is not calculated.

Also, Table II lists a number of fission product reactions not explicitly treated in the Reactor Safety Study.[3] Reactions like these are the consequence of quasi-static and nonequilibrium thermodynamic reactions. The REDOX

Area of Conservatism	Accident Sequence				
	PWR			BWR	
	V	TMLB′	S_2C	TW	TC
Lack of fission product retention in primary system	•	•	•	•	
No fission product deposition in containment leak passages		•	•	•	•
No fission product trapping in saturated water pools	•	•		•	
No fission product retention by auxiliary buildings	•	•	•	•	•
Total release of "volatile" fission products from the fuel	•	•	•	•	•
Uninhibited fuel oxidation and ruthenium release in steam explosions		•	•	•	•
Iodine assumed I_2 rather than CsI		•	•	•	•
Incomplete aerosol behavior modeling	•	•	•	•	•
Puff discharges upon containment overpressure failure		•	•		•

Fig. 2. WASH-1400 conservatisms impacting consequences for dominant accident sequences (Ref. 5).

TABLE I

Important Timing Considerations in Reactor Accidents

- Major loss of water from primary system precedes fuel failure—wet and steamy containment
- Fuel melting
  - Releases bulk of volatile fission products
  - Must precede *by some time* penetration of the pressure vessel
  - Aerosol agglomeration and iodine reactions occur inside pressure vessel
- Density of fission product aerosols should be based on the free volume of the pressure vessel—not the containment building
- Fuel melting will not start until fast blowdown stage is over
  - Not appropriate to use the speed of escaping steam/water to calculate fission product transport into containment

potential, pH conditions, and the resulting species can be estimated from properly constructed Pourbaix diagrams. There is good evidence for the existence of these species in most reactor accidents, in addition to CsI.

3. The discussion in Cybulskis et al. on the CORRAL code is incomplete. The code is based on only part of the results obtained in the Containment Systems Experiments (CSE). It does not recognize, for instance, that the most important attenuation effect observed in the CSE was in the aerosol generators where the concentration was high.[6] The CORRAL code was developed only on data from the injection chamber of the CSE vessel. Two other connected chambers existed. In those chambers, the aerosol concentration was always several orders of magnitude lower than the initial concentrations in the chamber into which the aerosols were injected.[7] The CORRAL code was never benchmarked on this multicompartment data. Moreover, the CSE was a relatively low concentration experiment, and therefore did not cover the relevant range where mass effects are important. Hilliard and Postma[6] and Parker and Creek[8] have recently published their assessments of some of the conservatisms in the source term.

4. As for containment failures, one must differentiate between the design pressure and the failure pressure, which is usually substantially higher. The example of attenuation through cracks in a failed containment building was meant to show that an additional attenuation factor, not accounted for in WASH-1400, would be operative in most of the likely containment failure modes. On the other end of the spectrum is the so-called "gross containment" failure. Even here large attenuation phenomena would be working because condensation due to heat capacity is always present. This was seen in the Gravel Gertie tests.[9] Even when the containment structure was completely destroyed, aerosols were almost entirely trapped in the debris. In tests done in 1967 at General Atomics on a one-fourth-scale model of the Fort St. Vrain containment building, failure occurred at roughly two and one-half times the design pressure of 600 psi. The building cracked; the steel liner opened up and vented; the crack contracted once pressure was relieved. Catastrophic failure of the containment building did not occur.

5. It is generally agreed today that a metal-water interaction will not result in sufficient thermal-to-mechanical energy conversion to rupture the pressure vessel.[10] This completely changes the pressurized water reactor I release category, which assumes a steam explosion of sufficient energy to fail the pressure vessel and containment building. Many attenuation processes will have already occurred prior to any gross rupture of the pressure vessel. Moreover, the dynamics of core meltthrough are usually determined by the MARCH code. This code may be good for determining general trends but is not accurate for determining specific timing. A core melt is so complex, and the existence of pertinent data so tenuous, that MARCH's ability to reproduce the physical phenomena we call core melt is in real dispute.

TABLE II

Possible Fission Product Reactions

Degraded Core Accident

Reaction Process	Product
Iodine/Cesium	
Iodine with cesium in fuel	Cesium iodide
Cesium iodide with water	Dissolved cesium iodide
Dissolved cesium iodide with oxygen from air	Iodine
Iodine with water	Hypoiodous acid
Iodine with organic material (i.e., paints)	Organic iodides
Iodine with metals in reactor building	Nonvolatile iodides
Iodine with dust and dirt	Nonvolatile iodides
Gravitational settling of solid iodides	Nonvolatile iodides
Adsorption/plate out of airborne iodides on surfaces	Nonvolatile iodides
Filtration of airborne particulates	Immobilized iodides
Removal of nonvolatile iodides by water scrubbing	Iodide solutions
Tellurium/Cesium	
Tellurium with cesium in fuel	Cesium telluride
Plate out of cesium telluride in fuel	Adsorbed cesium telluride
Cesium telluride with water	Cesium-tellurium solution
Precipitation of tellurium from solution	Solid tellurium
Oxidation of tellurium (solution) by air	Nonvolatile tellurium
Particulate Fission Product	
Particulate becomes airborne after fuel clad rupture	Airborne particulate
Airborne particulate settles out due to gravity	Plated/adsorbed material
Airborne particulate scrubbed out by water	Water suspension or solution of fission products

6. We don't believe most of this discussion is contrary to what we stated in Ref. 2. The <2% above-ground failures for containment buildings were referenced in Ref. 29 in our original paper.

7. The phenomena we talked about in Ref. 2 apply over a wide spectrum of reactor designs and accidents. There is no reason to believe that any accident will negate the laws of chemistry and aerosol physics. If the consequence models have not yet satisfactorily reproduced those (perhaps far simpler) reactor accidents that have occurred, how can they correctly predict rare, high consequence ones? A code that cannot interpolate correctly should not be used to extrapolate. An attempt to reproduce the SL-1 accident, using WASH-1400-type assumptions and the CORRAL code, gave disappointing and exceedingly high results.[11] Cybulskis et al.[1] have already stated that "the uncertainties in predicting fission product release from containment are quite large." More importantly, these uncertainties have never been quantified adequately. It is a truism that any future major reactor accident will be unlike any one modeled before, dominant sequence or not. The Three Mile Island Unit 2 accident is a highly relevant accident: it conclusively demonstrated that the currently assumed release fractions are wrong. The critique by Cybulskis et al. strongly suggests an approach based on analysis and calculations rather than direct observations. We believe empirical data should be the basis for analyses.

8. a. No water or surface absorption of volatilized species along the primary system transport path in any emergency core cooling injections failure sequence was considered in WASH-1400. Fission product scrubbing in boiling water reactor suppression pools was included in WASH-1400; however, the pool is not part of the primary system.

b. Reference 3 assumed no retention of any species by auxiliary buildings or structures outside the containment. Also neglected were particulate agglomeration and particle deposition on walls and surfaces in containment. There was only partial modeling of steam condensation effects; any fission product release on a containment rupture was treated as an instantaneous percentage loss of the airborne contents, directly to the atmosphere without depletion.

c. See 8a.

d. See points 2 and 3 for the discussion relative to the CORRAL code. It is also necessary to consider, when qualifying aerosol codes, the condition under which the experimental data were obtained. The CSE data were obtained at low concentration (<0.01 g/m^3) and small particle size. The CORRAL-1 correlation is based on these low concentrations. High concentration (100 to 1000 g/m^3) aerosols are assumed to exist in the reactor pressure vessel following a core meltdown accident. To be considered validated, a code should be compared against high concentration data taken in the first few seconds after the start of the experiment with instrumentation capable of handling particle sizes >10 μm. This has not generally been the case. However, data obtained in the High Temperature/Concentration Aerosol (HTCA) tests[12] in the late 1970s showed that >80% of an aerosol does grow to large size within the first 10 s under such conditions (see Figs. 3 and 4). This type of behavior is shown by some recent calculations using HAA-3b (Ref. 4) and QUICK (Ref. 13). For example, QUICK calculations have shown that for one of the dominant reactor accident sequences, the TMLB′, 99% or more of the initial aerosol mass should be retained in the primary system (Ref. 14), since the residence times will be long and the aerosol concentrations will be high.

9. Table III in our original paper was developed by R. L. Ritzman, the principal author of Appendix VII, "The Release of Radioactivity in Reactor Accidents" of Ref. 3.

In conclusion, we still feel that consequence models produce useless and misleading estimates when they are developed without using experimental evidence to demonstrate their validity.

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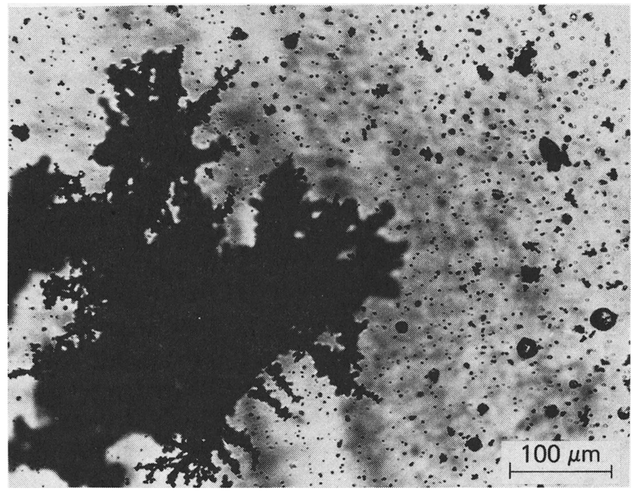

Fig. 3. Aerosol collected from HTCA Test 13 showing a bimodal aerosol distribution (Ref. 12).

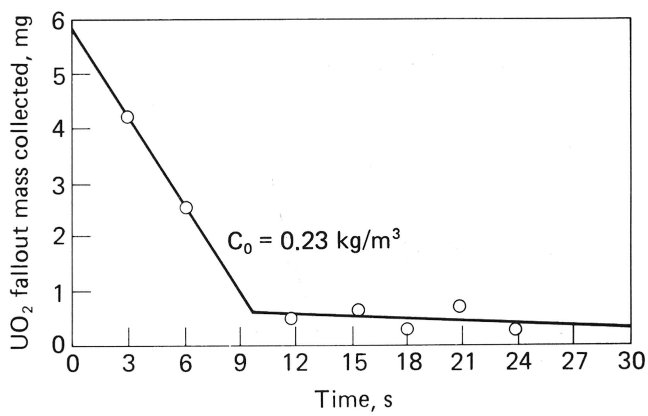

Fig. 4. The HTCA experiments at the Hanford Engineering Development Laboratory (Ref. 12).

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