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New York Power Authority

April 12, 1985
NuD&A-RED-85-06

Dr. N. Palladino
Chairman
U.S. Nuclear Regulatory Commission
1717 H Street, N.W.
Washington, D.C. 20555

Dear Dr. Palladino:

On April 3, 1985 the New York Power Authority made a source term presentation before the five commissioners. At that time the Commissioners requested some additional information, which is attached. I've also included copies of this reply for Mr. Dircks, Mr. Minogue, and Dr. Denwood Ross.

Sincerely,

Rich E. Deem

Richard E. Deem

Attached
jmt

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New York Power
Authority

On April 3, 1985 the Commission heard presentations by various groups on the subject of source terms. The New York Power Authority (NYPA) was represented by Mr. Richard Deem.

During Mr. Deem's presentation, the Commissioners requested additional information supporting NYPA's suggestions that (1) It was timely to re-examine containment leak rate testing requirements and (2) That the American Physical Society's concern about failure to isolate the containment could be addressed by keeping the frequency of large pre-existing holes to less than 10^{-3}/yr (once in a thousand years). Additionally, the Commission issued a general invitation for thoughts on the subject of source term closure.

Attached are NYPA's responses to these requests and a copy of the NYPA April 3, 1985 presentation.

LEAK RATE TESTING AND LARGE PRE-EXISTING HOLES

During the Indian Point hearing both the NRC staff and the licensees concluded that the two risk dominant sequences for the Indian Point plants were the interfacing systems LOCA and the containment slow

overpressurization accident. For example, at a leak rate of 50% of the containment volume/day, 1.9×10^{-6} of the original inventory of tellurium in the reactor core would escape to the environment. These figures are conservative in that they do not take credit for various fission product removal mechanisms that could take place between the point of release at the containment boundary and the environment, e.g., trapping of fission products in adjoining buildings. All containment penetrations lead to the auxiliary building. The various leak rates listed in Table I correspond to different sized holes, as listed. These holes were assumed to exist prior to and throughout the accident. Table II lists the mean number of early fatalities for different leak rates, assuming that no protective actions, such as evacuation or sheltering, have been taken.

As shown in Table I, a leak rate of 50%/day corresponds to a hole size of about 2.5 square inches. Measurements made in a number of integrated leak rate tests* (ILRT) show that containment leak areas are 2.5 square inches or less. Larger containment openings are possible, such as open air locks, but they would not likely to be detected as part of the ILRT process because such potential openings are secured prior to testing. From this, one may conclude that the observed ILRT leakage is in the range of hole sizes that do not lead to a significant number of offsite early fatalities, should there be a core melt, total loss of containment heat removal capabilities, and no protective actions taken.

* See Draft Final Report "Reliability Analysis of Containment

The above analyses examine leak rates and pre-existing holes from a consequence perspective only. As stated in the NYPA April 3, 1985 presentation, one can, and should, examine the issue of large pre-existing holes from a risk perspective as well. Pre-existing holes, of any size, <u>are not risk relevant</u> if they occur less frequently than 10^{-3}/yr (once per thousand years). This statement is based on the observation that pre-existing holes and degraded core accidents are independent events. If a nuclear plant has a core melt frequency of 10^{-4}/yr and meets a <u>criterion</u> of not havinq a large pre-existing hole more frequently than 10^{-3}/yr, the combination of core melt and a containment with a large pre-existing hole is 10^{-7}/yr, once in ten million years.* As described below, it is our view that accidents with a mean frequency of 10^{-7}/yr or less are not risk relevant.

*It was also pointed out on April 3, 1985 that many containments, like Indian Point, have spray systems which should work at least nine out of ten times, should there be a core melt. If the containment spray system does work, radioactivity releases, even with a pre-existing hole, would be very small. For such plants, the combination of core melt accident, a failed spray system, and a large pre-existing hole is 10^{-8}/yr, assuming that the criterion of not having a pre-existing hole more frequently than 10^{-3}/yr is met.

During the Indian Point hearing the risks from a PWR Release Category 2 source term was analyzed. The mean frequency of this accident was $4.8x10^{-7}$/yr. It was found that this postulated large relèase of radioactivity represented but one part in 75,000 of the non-nuclear early fatality risk at Indian Point. Since the source term for a core melt with a large pre-existing hole is less than the PWR Release Category 2 source term,* and since its frequency of 10^{-7}/yr is less than $4.8x10^{-7}$/yr, both the consequences and frequency of the core melt/pre-existing hole accident are less than the accident analyzed in the hearing and found to be a small risk contributor. It is estimated that an accident with a mean frequency of $1.0x10^{-7}$yr and a source term corresponding to a 25 square inch hole, as shown in Table I, would represent less than one part in five million of the non-nuclear early fatality risks at Indian Point. Therefore, if the frequency of a large pre-existing hole is kept to 10^{-3}/yr or less, core melt/pre-existing hole accidents are not risk relevant.

The arguments given above on low consequences and low risks suggest that a shift in emphasis in containment leak testing is warranted. Because of the limited potential consequences of the smaller containment openings, should there be a degraded core event, less frequent containment leak rate testing and higher technical specification leak rate limits are suggested. Offsetting this relaxation could be more emphasis on preventing large pre-existing

* Table I points out that the source term, even with a 25 square inch pre-existing hole, is smaller than a WASH-1400 PWR Release Category 2 source term, except for the noble goals.

holes, such as open airlocks, by keeping their frequency to less than 10^{-3}/yr. Since many of these potentially large openings are not tested during integrated leak rate tests, the use of interlocks, plant isolation procedures, and the like should be emphasized.

SOURCE TERM CLOSURE

In our view the process of source term closure is best defined when it is treated in the larger context of risk analysis. By doing this one ties together in a cohesive way the issues of uncertainty, emergency planning, safety goals, accidents frequency, and source terms.

Attached is a block diagram, figure 1, that could be used as guidance for determining the need for additional source term research and related regulatory activities. Each block in the diagram is assigned a number and there are comments associated with each block.

BLOCK 1

If the mean frequency of an accident scenario is 10^{-7}/yr or less, it is not risk relevant. Justification for this was given earlier in this transmittal. This limit of 10^{-7}/yr. is applicable to any sequence, for any plant at any site.

BLOCK 2

Accidents that do not lead to severe core damage, e.g., large LOCA with ECCS working, are not risk significant.

BLOCK 3

Severe core damage accidents are not risk relevant with an intact containment.

BLOCK 4

Severe core damaqe accidents where the containment retains its integrity longer than the time required to achieve low fission product airborne concentrations, are not risk relevant. For large dry PWR's very low fission product concentrations are achieved in the containment in about 6 to 8 hours*, whereas containment integrity should exist for at least 24 hours.

*For the containment slow overpressurization accident.

BLOCK 5

Is there an accident sequence of sufficient frequency and source term magnitude that it is an important risk contributor? If so, are there economical hardware or procedural changes that could be made that would be more cost effective than further source term research?

BLOCK 6

Having implemented all cost effective hardware and procedural changes, is the residual risk worthy of further source term research? The answer should be no, unless the residual risk, including its uncertainty, exceeds the NRC's preliminary safety goals.*

BLOCK 7

Further research should only go forward if the residual risk, including its uncertainty, exceeds the preliminary safety goals and if such research would be cost effective.

As stated earlier, source term closure should be considered within the larger context of what constitutes acceptable risks. Assuming that acceptability is defined as meeting the NRC's preliminary safety goals* with a sufficient margin to offset uncertainty, then

*For early and latent fatality risks

It is possible to tie together emergency planning and source term closure. Less reliance need be placed on emergency planning to achieve a certain level of risk if source terms are sufficiently small. The corollary to the above statement is that more effective emergency planning means less reliance on smaller source terms to achieve a certain level of risk. The NRC's preliminary safety goal for the early fatality risk can be comfortably met today by utilizing a graded emergency response, <u>without any source term reduction</u>. Very modest source term reductions, factors of 2 to 3 lower than WASH-1400 values, in combination with a graded response result in <u>zero</u> calculated early fatalities.** Consequently, no further source term research is needed to achieve a very low or zero early fatality risk, if the graded response is implemented.

In summary, the key to a focused source term closure effort is to establish what is an acceptable risk. Calculations can then be made to determine what combination of plant design features, emergency planning processes, and source terms most efficiently meets the acceptable risk criterion, with a sufficient margin to offset uncertainty.

**Based on independent analyses performed at the NRC and the DOE.

TABLE I

FRACTION OF FISSION PRODUCT INVENTORY RELEASED TO ENVIRONMENT

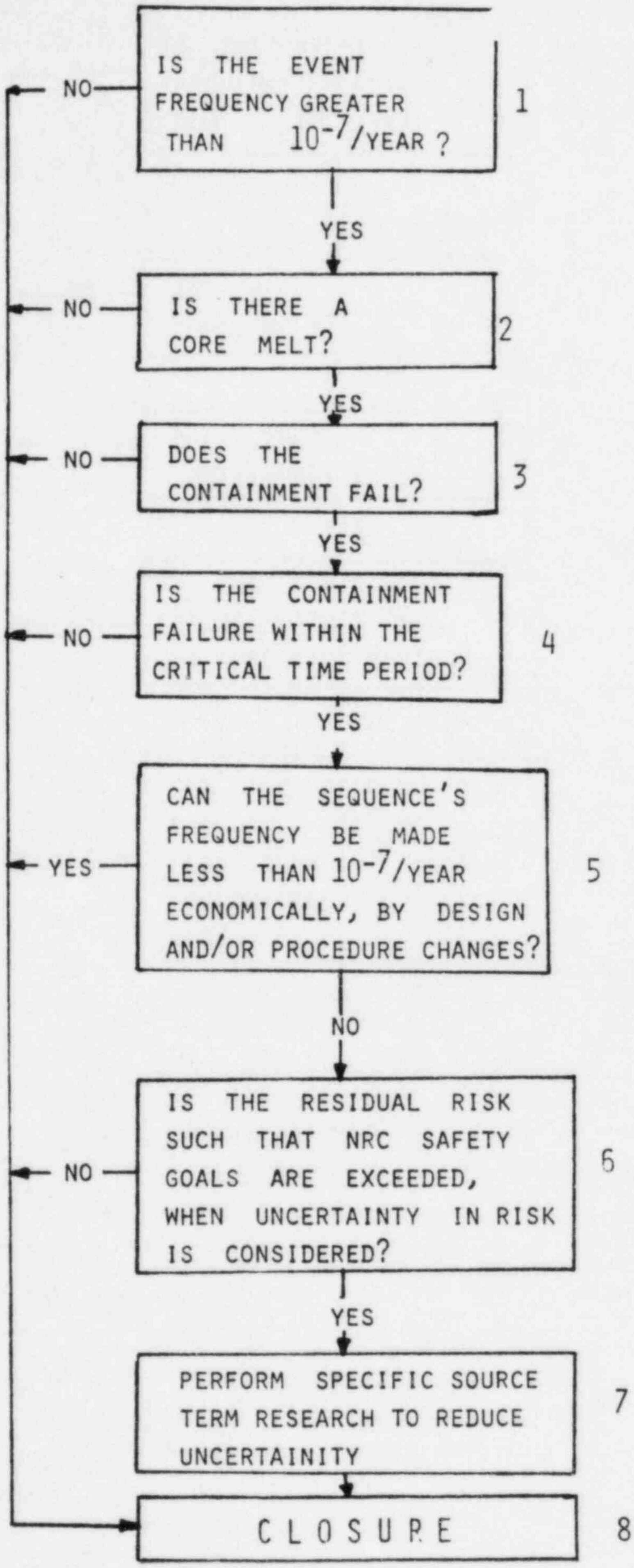
Fission Product Specie	Leak Rate, % per day						WASH-1400 PWR-2 Release Category 2
	.1	1.0	10.0	50	150	5000	
CsI	2×10^{-5}	3×10^{-4}	3×10^{-3}	1.6×10^{-2}	5×10^{-2}	1.4×10^{-1}	7×10^{-1}
CsOH	1.7×10^{-5}	2.7×10^{-4}	2.7×10^{-3}	1.5×10^{-2}	4.5×10^{-1}	1.2×10^{-1}	5×10^{-1}
Te	6.5×10^{-9}	3.5×10^{-8}	3.5×10^{-7}	1.9×10^{-6}	6.1×10^{-6}	1.7×10^{-5}	3×10^{-1}
Ba, Sr	3.1×10^{-6}	2.9×10^{-5}	2.9×10^{-4}	1.6×10^{-3}	5.2×10^{-3}	1.5×10^{-2}	6×10^{-2}
Ru	1.5×10^{-6}	1.2×10^{-6}	6.8×10^{-6}	3.5×10^{-5}	1.1×10^{-4}	2.9×10^{-4}	2×10^{-2}
La	-	2.5×10^{-6}	5.9×10^{-6}	2.3×10^{-5}	2.3×10^{-5}	1.8×10^{-4}	4×10^{-3}
Noble Gases	1.0	1.0	1.0	1.0	1.0	1.0	0.9
Hole Size, Square Inches		.05	0.50	2.50	8.0	25.0	-

TABLE II

CONSEQUENCE RESULTS
ASSUMPTION: NO EVACUATION
EARLY FATALITIES

Leak Rate	Mean Number Of Early Fatalities
0.1%	0
1%	0
10%	1
50%	3.52
150%	39
5000%	314
WASH-1400	3650

FIGURE ONE


IS THE EVENT FREQUENCY GREATER THAN 10^{-7}/YEAR ?
1
NO
YES
IS THERE A CORE MELT?
2
NO
YES
DOES THE CONTAINMENT FAIL?
3
NO
YES
IS THE CONTAINMENT FAILURE WITHIN THE CRITICAL TIME PERIOD?
4
NO
YES
CAN THE SEQUENCE'S FREQUENCY BE MADE LESS THAN 10^{-7}/YEAR ECONOMICALLY, BY DESIGN AND/OR PROCEDURE CHANGES?
5
YES
NO
IS THE RESIDUAL RISK SUCH THAT NRC SAFETY GOALS ARE EXCEEDED, WHEN UNCERTAINTY IN RISK IS CONSIDERED?
6
NO
YES
PERFORM SPECIFIC SOURCE TERM RESEARCH TO REDUCE UNCERTAININITY
7
CLOSURE
8

NEW YORK POWER AUTHORITY
PRESENTATION BEFORE T NUCLEAR
REGULATORY COMMISSIONERS ON
SOURCE TERMS

My name is Richard Deem and I represent the New York Power Authority. We thank the Commissioners for this opportunity to make a presentation on source terms.

The Power Authority's first major effort in the source term area occurred during the Indian Point hearing where we, and Consolidated Edison of New York, sponsored source term testimony by Drs. Walton Rodger and William Stratton. Drs. Rodger and Stratton examined the two risk dominant sequences identified in our Probabilistic Risk Assessment Study, IPPSS, (Indian Point Probabilistic Safety Study) and gave us their best engineering judgements on what more realistic, yet still conservative, source terms might be. IPPSS had utilized source terms that resembled WASH-1400 values. Using the Rodger - Stratton estimates, we recalculated the risks from Indian Point Units 2 and 3, thereby enabling one to observe how various calculated public health and economic risks would be reduced by use of these smaller source terms. Additionally, some source term sensitivity studies were made and presented during the hearing.

Sometime after completing the written testimony it became possible to use more refined computer programs, rather than engineering judgement, to analyze source terms. The Power Authority then began an in-house effort to develop its own source term analytical capability. This in-house effort, made in conjunction with Risk Management Associates, was supplemental to our participation as an IDCOR member, yet differed from the IDCOR effort in that our family of computer programs had as its basis the same computer programs that the NRC staff was developing. To the best of my knowledge, no other utility has developed a source term analysis capability based on the NRC sponsored work.

Because we chose this particular set of computer programs we were, in some ways, a forerunner to the American Physical Society's review of the BMI-2104 work and we too saw a number of deficiencies and limitations. For example, when we first started using the BMI programs, we discovered that they did not account for re-vaporization of various radionuclide species. We soon improved our programs to account for this phenomenon. We also recognized that each of the computer programs we started with was designed to calculate separate aspects of a complex nuclear accident. Simplistic methods might then be used to join the output of these separate programs but with a loss in accuracy and understanding, for in an actual accident many of these separate calculated phenomena would be changing and interacting simultaneously. We realized that to achieve an accurate understanding of a degraded core event it was necessary to couple these programs together so that they all interacted and ran on the same time clock.

Therefore, a- each timestep our compute package recalculates every important pa meter according to the ch..ges that have gone on throughout all of the computer programs.

This distinctive characteristic of our computer package was praised in the APS report, page 179 "...Thus an interactive feedback is implied that usually is not performed. However, the New York Power Authority Group (Deem and Bieniarz, 1984) have written a vectorized code that integrates the functions of MARCH, MERGE, CORSOR, and TRAP-MELT, for the specialized conditions of the Indian Point Reactors. This approach is a considerable improvement and needs extension."

We thank the APS for this very positive recognition and would only add that since that time when the APS made its survey, we have extended our analytical capability so that it now can be used for all PWR's and BWR's.

Time does not permit me to identify today all of the other modifications we have made to the original NRC-sponsored computer package, however, our work has been examined by other groups as part of the Power Authority's commitment to the peer review process. As pointed out by the APS and others, peer review is an essential ingredient in arriving at an acceptable source term analytical capability. Toward that end we have sought out critiques of our work by other knowledgeable people. We have given papers at source term conferences, made lengthy presentations before the Advisory Committee on Reactor Safeguards, the NRC staff, and the APS. We have worked closely with a number of national laboratories, particularly Oak Ridge, to benefit from their knowledge. We have openly and widely distributed our reports and results, both here and abroad. Additionally, we have, ourselves, acted as peer reviewers of other people's efforts - thereby learning from this process too. I have drawn a number of comparisons between the APS review and the Power Authority effort which show similarity in technical approach and similarity in emphasis on the scientific process of peer review. Another point of agreement is that the APS report presents a generally favorable review of source term technology and we concur in these conclusions.

There are, however, some differences between the APS and Power Authority perspectives. Most of these differences can be attributed to timing. Almost all of the APS review effort was concluded in the summer of 1984. This is a rapidly evolving field and there have been many advances since last summer. For example, all of our work on our Mark I BWR came after the APS concluded its review and started to write its report. If one applies the knowledge we have today, a number of APS concerns appear to be diminished or resolved. To illustrate this I am going to discuss how our work addresses four APS concerns:

(1) The need to examine additional accident sequences (VIII.C.1)

(2) Failure to isolate the containment (VIII.B.8)

(3) Core - concrete interactions, and (VIII.A.2)

(4) Uncertainty (VIII.E.2)

First, with regard to examination of additional sequences, the Indian Point Probabilistic Risk Assessment, using WASH -1400 style source terms, calculated that some 98% of the latent fatality and offsite economic risks were due to degraded core events which lead to a slow overpressurization of the containment. We found that the fission product concentrations in the containment atmosphere dropped to very low values some 6 to 8 hours after accident initiation. The time to achieve these very low concentrations, 6 to 8 hours, is much less than the over 24 hours need to overpressurize the containment. We examined a number of different accident sequences and explored the importance of different analytical models. We looked at large breaks and small ones, various particle settling models, with and without the generation of non-volatile aerosols, the effects of two types of piping insulation and, although the early stages of these analyses differed somewhat, approximately the same low concentrations were achieved in about the same time frame. In the large break LOCA case, with no electrical power, all of the votatile radioactive material was initially airborne, and the time it took to reach low concentrations was very similar. These observations have a particular relevance to APS conclusion VIII.C.1 where it was recommended that another iteration be made in the process of selecting accident sequences. For the large dry PWR containment, or for any containment where the time necessary to achieve low fission product concentration is less than the time at which the containment might fail, all accident sequences must produce similar results.

Futhermore, the APS, in its conclusion VIII.B.9, stated that "The diversity of various government, industrial and foreign groups engaged in source term research makes it unlikely that important phenomena will be left unconsidered."

APS conclusion VIII.B.9, above, coupled with the observation that all accident sequences produce similar results when the fission product settling time is less than the containment failure time, indicates that the study of additional sequences is unlikely to result in significant changes in the calculated risks.The peer review process that we have been engaged in is also beneficial in shedding some light on APS concern VIII.B.8, failure to isolate the containment. During our source term presentation before the ACRS in July, 1984, we were requested to examine the relationship between containment leak rate and consequences for the PWR slow overpressurization accident. We found that containment leak rates far in excess of our present technical specification limit only

produce minc increases in public healt ;ffects, assuming a degraded coι event. This observation may be combined with some historical data recently gathered by the NRC on the frequency of different sized leak rates for BWR's and PWR's. Almost all the data correspond to leak rates that are in the low consequence range of our analyses. Only very large openings could be of concern, however, the probability of such a large pre-existing hole is small.

Two inferences can be drawn from the above:

A. It is timely to re-examine containment leak rate testing requirements, and

B. That, if the probability of a large pre-existing hole is less frequent than 10-3/yr, (once per thousand years) it is not risk relevant.

The latter statement is based on the observation that the occurence of pre-existing holes are events that are independent of degraded core events. In the Indian Point hearing it was shown that even prompt releases of WASH-1400 style source terms can result in small public risks if the frequency of such an accident is sufficiently small. In this hearing it was shown that a large release of radioactivity at a frequency of 4.8 x 10-7/yr - about once in two million years - results in risks that are a small fraction of the corresponding non-nuclear risks. Smaller frequencies than 4.8 x 10-7 /yr can be expected for the combination of a core melt and the independent occurrence of a large pre-existing hole. At a degraded core event frequency of 10-4/yr, the combined frequency is 10-7/yr. If there are containment sprays, as there are at Indian Point and elsewhere, this system should operate at least nine out of ten times during a degraded core event. Only very small source terms are possible with a functional spray system. The combination of a degraded core event, a failed spray system, and a pre-existing large hole in the containment is about 10-8/yr (once in 100 million years) or smaller, when the frequency of the pre-existing hole does not exceed 10-3/yr.

Further, even with a pre-existing hole, the source term under these conditions would be smaller than the source term used in the Indian Point PRA. Thus both the frequency and the consequence of a large pre-existing hole accident are smaller than the accident analyzed for Indian Point and found to be a small contributor to the public's overall risks. The APS concern about failure to isolate can be conservatively addressed by keeping the frequency of a large pre-existing hole to less than 10-3/yr.

It is our conclusion that provided containment isolation is not lost more frequently than 10-3/yr, source term technology is now sufficient for these sequences. Generalizing on the above, the Power Authority believes that <u>both</u> elements of risk (i.e., frequency and consequences), must be considered. Source terms for

rare events that are not risk relevant are unimportant to the regulatory process. More specifically, events whose mean frequency is 10-7/yr or less are not risk relevant and further source term efforts for such events is not warranted by reasons of public health and safety.

I would now like to discuss core-concrete interactions. In addition to the Indian Point 3 plant, the Power Authority operates the FitzPatrick Plant, a BWR with a Mark I containment. Since some analyses by others have indicated that Mark I containments may have important core-concrete interactions that could lead to larger source terms, we have begun to apply our source term analytical package to the important accident sequences for this type of plant. The Electric Power Research Institute has taken an interest in our work and is co-funding this activity. Although this effort is still on-going, a number of results that bear on the core-concrete interaction problem are now available. We have learned that the time it takes to fail the Mark I containment is usually longer than others have stated. This is because, for a number of sequences, the volatile fission products are channeled into the suppression pool via the steam relief valve system. These volatile fission products carry with them a large fraction of the decay heat. With this portion of the decay heat going into the suppression pool water, there is far less energy available to heat the drywell air. Consequently drywell heat-up rates are lower and failure pressures take longer to be attained. As in the case of the large, dry PWR this extended time-to-failure should result in lower source terms. Even if there were a significant amount of lanthanides released from a core-concrete interaction, their airborne concentration would be reduced during the longer time period that the containment remained intact.

We have also found that by more precisely modeling the operation of various plant systems, some accident sequences do not lead to a core melt, as was previously analyzed by other. Naturally there is no core concerete interaction if the core does not melt.

It also appears that in some cases earlier analyses over predicted the temperature of the molten core. The amount of lanthanides driven off by a core-concrete interaction is very temperature sensitive and overpredictions of the molten core temperature will lead to over-estimates of lanthanide production. We found and have corrected this computer programming deficiency.

The Power Authority expects to complete its source term analyses of the FitzPatrick plant in a few months. It seems likely that plants with Mark I containments will be shown to have risk profiles comparable to other designs.

Perhaps the issue of uncertainty has generated more interest than any other statement by the APS. In going forward on the source term issue it is important to keep in mind that what constitutes resolution of an uncertainty issue can be different for physicists

than for regulators. The physicist may be concerned with knowing a particular val to a high degree of preci on. The regulator needs to have a technical basis upon which a decision can be made, even though there is uncertainty in the source term numbers. It is helpful to recall that large margins can off-set uncertainty, permitting the decision-making process to proceed. For example, suppose that a regulatory decision was based upon staying below the threshold of early fatalities. With the exception of the noble gases, environmental releases were only one part in a hundred thousand, or less of the radionuclide inventory in the reactor core. Such small releases to the environment, even assuming no evacuation, would be about a factor of 1000 (one tenth of one percent) below the level that might lead to early fatalities.

Thus large dry PWR containments, for this class of accidents, have large margins below the early fatality threshold both in their very small release fractions and in the time available to achieve such small values. This large margin should be enough to offset uncertainty.

Whenever the margin is larger than the uncertainty, decision -making can readily go forward.

To summarize these thoughts:

1. Much of the APS report is positive.

2. Since the APS review was completed, many of its concerns have been addressed.

3. A search for additional accident sequences does not appear to be warranted.

4. Failure to isolate containment is best handled by limiting the probability of having a large preexisting hole.

5. It is timely to re-examine containment leak rate testing requirements.

6. Core-concrete interactions in BWR Mark I plants appear to be less risk relevant than thought before.

7. Decisions can be made even when there is uncertainty. The decision-making process is eased when consideration is given to the margins available.

8. Source term technology for overpressurization accidents in large dry PWR containments is now sufficient for regulatory decision making.

9. Regulatory decisions should be made on the basis of risk, (i.e., the consideration of the combination of probabilities and consequences.) In any case, events less frequent than 10^{-7}/yr are not risk relevant.

10. It is important to identify those regulatory decisions than can be made in the short term as distinguished from those that require confirmatory research. Regulatory action can and should commence on the short term items. As a licensee, we feel it is important to get started.

Thank you.