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COMBINED ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE ON GESSAR II
AND
RELIABILITY AND PROBABILISTIC ASSESSMENT
GESSAR II FDA REVIEW

LOS ANGELES, CALIFORNIA

OCTOBER 20, 1984

REPORTER'S TRANSCRIPT OF PROCEEDINGS

DAVID OKRENT, Chairman of the Subcommittees

JACK EBERSOLE, ACRS Member

LOS ANGELES, CALIFORNIA; SATURDAY, OCTOBER 20, 1984; 8:30 A.M.

MR. OKRENT: The meeting will now come to order. This is a combined meeting of the Advisory Committee on Reactor Safeguards Subcommittee on Limerick Units 1 and 2 and Reliability and Probabilistic Assessment.

I'm David Okrent and I'll be acting as subcommittee chairman.

The other ACSR member present today is Mr. Ebersole.

We also have in attendance ACRS consultants Dr. Powers, Dr. Davis, Mr. Garcia, Dr. Pomeroy, Dr. Trifunac.

The purpose of the meeting is to continue the subcommittee review of the Limerick PRA/SARA and the application of Philadelphia Electric Company for a license to operate the Limerick station. Dr. Savio and Dr. Seth are also here for the ACRS.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the Federal Register on Tuesday, October 9, 1984.

A transcript of the meeting is being kept. And it will be made available as stated in the Federal Register Notice. It is requested that each speaker first identify himself or herself and speak with sufficient clarity and volume so that he or she can be readily heard.

We have received no written statements or requests for time to make statements from members of the public.

The original proposed agenda showed us beginning at 8:30, which we did. And adjourning about 5:30, which I plan to, within a half an hour.

I'm going to propose a modification at this time in the agenda, taking item 7 on seismic questions and making it item 3. I'll let the staff tell me whether the 30 minutes allotted is adequate to cover this matter or whether they think it would be wise now to save more time for their presentation and whatever questions there may be in this area. In which case we will shorten some other thing appropriately.

MR. MARTIN: We think it is adequate at this point.

MR. OKRENT: I'll mentally allow twice the time.

I'll ask the ACSR consultants to participate during the discussion in the usual fashion, making sure that they ask such questions as they think they wish to hear about. I will be interested in the opinions of the ACSR consultants on subjects which relate to their particular areas of interest as much as possible today. So that I have a reasonable feeling for what further questions they have if there are things that they think are seriously incomplete. And also what their assessment is, although there will be a written report before November 1.

MR. SAVIO: Yes, sir, scheduled for November 2.

MR. OKRENT: At present the review of Limerick is scheduled to be heard by the full committee on November 2. But, as I say, I'll be interested in knowing your principal opinions and also, if there are things that you believe are matters where indeed more information is relevant to decision-making or to help the decision-making, we need to hear these during the agenda at the end each section, if that's possible.

Other than that I have no comments.

Mr. Ebersole, do you have any comments?

MR. EBERSOLE: I have just a brief one. In our discussions yesterday on GESSAR II, a topic came up that bothered me a little bit. I think one of our members and several others expressed a belief that in this PRA approach to accident analysis it seemed to be entirely reasonable to allow small windows through which one could go to total disaster. And I guess I'm in disagreement with that. I like to close all the small windows if I can see them at all. So I don't know whether we will see any of those in this analysis here or not, but I'm going to look for them. I don't like small windows the other side of which is disaster, whether it is PRA or just deterministic logic.

MR. OKRENT: You remind me that there was one topic which we did discuss during the GESSAR review, which has been discussed at prior times, which I don't think we raised at the earlier Limerick review. This relates to the following: In the normal deterministic review with regard to piping which is not seismically qualified, I believe the custom is to postulate that these pipes break one at a time to look at what their consequences on environmental, et cetera, may be.

The question that came up is when one looks at things probabilistically, when one talks about earthquakes not only like the SSC but more severe. Some likelihood is introduced of the failure of more than one of these non-seismically possibly non-pedigreed pipes. And we are interested in understanding what the situation is at Limerick with regard to the existence of such pipes. Was there multiple failure study in the PRA/SARA? And if not, what can you tell us about it? If so, where should we look to find it?

In any event, we would like to hear about that during this day at the appropriate point.

Let's see. A moment ago I was told that General Electric was awaiting arrival of their slides. Are they here?

A VOICE: They weren't three minutes ago, but I will check again.

(Discussion held off the record.)

MR. OKRENT: If not, we will reverse the order and let the staff go first.

(Discussion held off the record.)

MR. BOYER: I would think that would be the prudent thing to do.

MR. OKRENT: So I assume the staff has its view-graphs or slides with it?

Good.

All right. So we will go to 2-B before 2-A, and I think the staff wanted to make some introductory comments, brief ones, before beginning the technical discussion.

MR. MARTIN: Bob Martin, NRC staff.

I just wanted to provide more information regarding the severe accident related issues, which were heard in the hearing process this past spring. A partial initial decision was reached by the atomic safety and licensing board on those issues in late August of this year.

Recently I have obtained a copy of the intervenor appeal of that PID to the appeal board. To summarize several of the items in the intervenor's appeal just briefly, they are contending that, for instance, the board's exclusion of staff identified mitigation design alternatives from the licensing process and environmental review violates the commission regulations. That's just a caption of that one point of theirs. Another one would be that the board improperly excluded from the environmental review the risk from sabotage as one of the topics we cover in the discussion of uncertainties in the PRA.

And there are several others. I believe those are the most pertinent ones with respect to our discussions today. I just provide that for information purposes.

MR. OKRENT: Thank you.

If you can loan us a copy of the document, maybe we can peruse it during the day.

Can we start, then, with the NRC presentation on accident progression?

MR. MARTIN: Trevor Pratt will begin with our presentation.

MR. PRATT: What I would like to do is start really with the conclusions. I'm going to tell you what the conclusions are of the last review up front.

(Slide one shown.)

MR. PRATT: I think what I got from the last meeting was that you would like us to take you through a walk-through of the core meltdown phenomenology of the containment failures mode and part of the release and how uncertainties in those calculations would influence overall consequence analysis.

And what I intend to do is to go through -- we have got quite a few slides to try to demonstrate the following four points.

We believe because of the way the calculations were construed for use in the final environmental statement for the Limerick generating station that the source-term calculations are much closer to the upper bound risk estimates than to the lower boundary. And this is a constraint as to the way the calculations were performed.

I'm going to walk through the calculations and do -- the use of the word uncertainty here is perhaps giving it more credit than it deserves. It is really rather upper bound calculations to show the impact of uncertainties regarding containment building failure modes, the timing of containment failure source terms that, if we go to very upper bound calculations, that the risk and this factor applies both to the early fatalities and the long term damage indices.

I would also like to go through some of the new methods and give some indication of the impact of these methods, and the results will not be numerical. We don't have numerical results for limerick specifically at this time. What I would like to do is to go through and give you an indication of how they think things may change, based upon the calculations that we have already performed and the documents that are available.

By new methods we are not really talking about how just the actual methods program, but also the $200,000,000 staff research program is feeding into all this work. I think what we will show is that the longer term damage indices the latent cancer fatalities person-REM and so on. There is a very good potential for a significant sort of potential. There is less potential, although it still exists, for the shorter term damage indices. As we go through and build up the risk profile you will see why we will make that distinction later on.

MR. OKRENT: Before you move ahead, is there going to be another speaker who goes into detail on the phenomenology, the physics of what is going on in a core melt.

MR. PRATT: I'm it.

MR. OKRENT: Well, please, I'm interested in getting the best physical understanding, chemical understanding any others, of what you think transpires for a sufficiently wide range of scenarios that when you are done I have a rather good idea of what you think goes on in the Mark II like Limerick for a broad spectrum of postulated accidents.

MR. PRATT: Right.

MR. OKRENT: I'm more interested in that than some numerical numbers at the moment.

MR. PRATT: Okay, but I think that what is important, one of the questions that was asked at the last meeting was a set of tables put up by Frank Coffman showing risk and the numbers were low and the question was how robust are those calculations. What I would really like to do is get across to you that the particular set of calculations performed here are somewhat conservative and I have to do the numerical walking through to at least demonstrate that to you.

I will certainly indicate and go through our best estimate of how the core will progess in terms of failure modes conditions, probabilities performance of the building, where we think the core freeze is going to go. But for these particular set of calculations, the sensitivity of those assumptions is not large because of the way of calculations were performed.

MR. OKRENT: Well, again, just to make it clear, I want to understand the physical behavior and also which phenomena are largely assumed to go a certain way. So I have an improved basis for thinking about and I would hope to have the benefit of your insight as well as others as to what different paths, for example, which is physically possible, would lead to a marked change in the conclusions, if there are any, that need such consideration.

In other words, sometimes one assumes some path or failure mode is very small like pressure vessel failure, typical, right?

MR. PRATT: Right.

MR. EBERSOLE: I hope your analysis will go through the progression of events that lead to core degredation in some reasonable step-wise fashion anticipating that at any time in the course of that progression cooling will be recovered and will go down a new path, whatever it may be.

I don't know what the direction of your progressive states is, but I hope it is reasonable enough to say, yes, this is what might happen. I might have partial damage, partial melt, interception of that process, or go clear on. And if you don't, it is -- it can't be real.

MR. PRATT: The sounds like you are answering the question before --

That you will not see in the assessment I'm going to give you this morning.

MR. EBERSOLE: You take one big step to melt.

MR. PRATT: We make the assumption -- this is built into the front end analysis.

MR. EBERSOLE: That's the defined point of beginning?

MR. PRATT: Yes, that's right. So I can take you through a very detailed --

MR. EBERSOLE: No, don't do that. We will find that other piece somewhere else.

MR. PRATT: What I'm trying to do is emphasize to you the phenomenological aspects. But I still would like to make the point with the numbers. Bear with me a little bit because I think that is important in terms of the uses that we put this particular PRA to as opposed to the type of things you will be hearing for GESSAR II where the calculations performed were different. There is a marked difference in that these calculations were performed and I would like to point that out to you.

(Slide two shown.)

MR. PRATT: The next slide -- let me move through this one fairly quickly.

This is a layout of the issues I would like to cover with you. I would like to first go through the methods that we used and try to identify those clearly and give you some indication of the accident classes.

Failure modes looked at and the risk perspective part of it give some idea as to where all of the contributions to the numbers that Frank gave at the last meeting are coming from. So you can focus on the areas of importance in terms of phenomenology and then walk through in great detail class one and class four sequences and then give you some indication of impact of the new methods.

(Slide three shown.)

MR. PRATT: What I've tried to show here is when we talk in terms of source terms there are two distinct areas of interest that go into it. One of course is the timing and the duration and the energy associated with the release. And this is really coming from the containment loads and performance calculations.

The other side of course is the quantity efficient product release and the Limerick analysis the way we analyzed the calculation was to do improved calculations regarding containment loan and containment form with Limerick specific features, relative to the RSS methodology.

We looked at the Mark II containment. However, we did not take credit for all of the methods that are going on on the left-hand side of that viewgraph. So what this does is it builds into the calculation rather conservative estimates in the amount of fission products that would be released.

This then feeds into the calculations as somewhat of a conservativism that makes it relatively insensitive to assumptions. Bear this in mind. GESSAR takes credit for all of this picture, very clearly we did not in our evaluation of the Limerick facility as input to the FP containment.

(Slide four shown.)

MR. PRATT: In terms of WASH-1400 methods the boil code was used to model the core degredation within the primary system. The hand calculations are used for containment response. We had specified fission product release fractions for different phases of core meltdown and we use those specified source terms for our calculations.

We did not model nor did we take credit for fission product degredation or retention within the primary system.

MR. POWERS: It seems to me that by neglecting that position in the primary system you are attributing that as a mitigation method which would make your source terms to high upper bounds. It seems to me there could be another factor there that might change that conclusion.

If you were to deposit fission products in the primary system and they subsequently will be vaporized then rather than coming in when the suppression pools were not saturated and decontamination factors were very high, they would come out later. That would change that conclusion from this being a conservative calculation to it being a rather optimistic calculation.

MR. PRATT: It is an important point. We looked at this in great detail when we went into GESSAR when we were using the new methods and found readmission to be a very important point.

You will still find that the calculations performed were extremely robust because there was very little credit given for -- not just only the deposition in the primary system but also pool DF were very low in certain areas and no credit at all taken for the situation where the pool was saturated, for example.

So there are a lot of other non-conservativisms built into the calculation that do tend to make it high. I think your point is very important -- this is when some of the uncertainty we get later on becomes very important.

When you look at something like GESSAR there where you have taken credit for the calculations with that kind of asumption one has to look at it very carefully. So in this particular case it really isn't going to change our calculations, but I think it is is very important point.

(Slide five shown.)

MR. PRATT: I've already mentioned this. Namely, that for a saturated pool we neglected pool scrubbing and used a factor of 100 if the pool was saturated. This is the WASH-1400 methods and of course the corral code was used.

MR. POWERS: Do all releases of fission products in these calculations --

MR. BOYER: Could you speak up, please.

MR. POWERS: Do all releases of fission products coming from the drywell pass through the suppression pool get out to containment?

MR. PRATT: No. Every sequence -- we have got some horrendous releases here. Embarrassingly high.

(Slide six shown.)

MR. PRATT: Just to focus you very quickly, in the LGS-PRA they used the INCOR code with independent analysis. I'll show you some of the differences between the predictions of that code and what we used which was MARCH 1.1 and the LGS-PRA used -- plus the independent analyses.

(Slide seven shown.)

MR. PRATT: This is really taken straight from the PRA and gives you a brief indication of what the INCORE code does. It uses boil which is common with MARCH. It uses a thing called PVMELT, a pressure vessel melt through. You will see later on this tended to give rather long times for penetration of the vessel relative to what MARCH would give. That could impact the warning time. And INTER which is core concrete interaction.

And all of those mass and energy flows coming from the various stages of core meltdown were fed to CONTEMPT-LT to give you the pressure temperature conditions in the containment building.

(Slide eight shown.)

MR. PRATT: Next viewgraph -- I think we can move along. This really equates subroutine to subroutine in terms of two codes. I don't think we need to get into that in too great detail.

(Slide nine shown.)

MR. PRATT: In terms of the LGS-PRA methods that were actually used in terms of calculating source terms, they were based on the reactor safety study methods but there were slight differences from that approach.

There was a petitioning of the melt released between the drywell and the pool. In other words a certain fraction of the melt release was released directly to the drywell in vessel failure. Also for saturated pool a decontamination factor 10 as opposed to one so there was a reduction for certain sequences where the pool would be saturated.

10 percent oxidation release for all failure modes. In other words, it is an assumption that 10 percent of the core debris did pass through the diphragm floor and into the water and contributed to an oxidation release. And for pumps one and three sequences when the containment building depressurized there was a 15 percent pool flushing.

The first BNL review that we did which was in 3028 was on a similar approach to that. There was differences and we documented them in that document. We also calculated -- we have gone through several calculations and source terms for Limerick. This was the first set in 3028.

We did additional calculations in the memorandum that we sent to the contract monitor in which we calculated the decontamination factors of the pools different from WASH-1400 methods to assess that.

I'll go on and describe some of the later calculations that were performed.

(Slide ten shown.)

MR. PRATT: When the SARA document was issued, which is the document dealing with external events at Limerick, the calculations there were somewhat different from our RSS method even further. There was an attempt made to release the fission products in accordance with the trends in NUREG-0772. Ex-vessel vaporization release based on the difference between NUREG-0772 and the RSS predictions.

I did not go to the new ASTPO methods. And then the fission product transport calculations, some of them were based on corral calculations with some hand -- just for some of the volume further on down.

(Slide 11 shown.)

MR. PRATT: As we started to review SARA we were requested by the NRC to recalculate all of the source terms that we had done for not only the internal events but also the external events for input to the final environment statement for the Limerick generating station. And we then were requested to perform the calculations in a very strict manner, following exactly RSS methods.

So we redid and went through 28 calculations for all the various failure modes and we were very strict in our interpretation of the methods. So it will be to these source term calculations which we used that I'll be referring to today and I'll take you through those calculations.

The source terms were documented in the BNL report 33835.

(Slide 12 shown.)

MR. PRATT: Give me some feedback here. Do you really want me to go through all of this or are you familiar with the various classes that we used in the Limerick PRA?

MR. OKRENT: I think you should identify what each class is because as you go from PRA to PRA, everybody has his own philosophy.

MR. PRATT: That's principally where they are here. Every time you look at a PRA you have a different set of judgements. Class one sequences, this range is very similar to the GESSAR arrangement that you discussed the last two days. Class one sequences are basically transients small break LOCA's with loss of coolant makeup.

In terms of what is of interest to me is analyzing the consequences there, the containment building is assumed to be intact at the time of core melting and relatively low pressure. So we have a melt down into an intact containment building. We follow the progression from that point.

Class two are assumed to fail the containment first and melting core down into a failed containment so the assumption of the failure location can be quite important to those.

Class three, when you look at it in terms of containment response it is very similar to class one sequences. You're melting the core into an intact containment. The major difference is the pool is saturated for these sequences so you have different pool DF.

Class four, that's very similar to the class two sequence pressurization, fails containment building first and you then melt the core down to the failed containment building. The source terms calculations are very large because we are failing the containment first and melting the core into an open containment and not taking credit for primary system retention or saturated pool scrubbing.

Very high source terms the way we have calculated them.

MR. OKRENT: Where do you cover the equivalent to the class B event in this picture?

MR. PRATT: The equivalent of class B leakage something along those lines?

MR. OKRENT: A failure between high pressure and low pressure systems, yes.

MR. PRATT: Most of the bypass sequences were put into -- it was a very low probability in the PRA in terms of what the actual number was. I'm trying to recall.

They weren't explicitly calculated in terms of MARCH and corral. MARCH and corral is really a containment response code and if you want to model a long pipe you are going to have to tell it what to do so you can get any answer you want. You have to be careful using that particular suite of codes to model that sequence.

If you recall from WASH-1400 the interfacing system LOCA was not modeled specifically as MARCH and corral, it was an amalgam of failure modes which was bended and we adopted the same procedure. We bended it to what we considered to be a severe release so it was not expicitly calculated. It was put into a severe release.

Again, I think -- we have the same situation, the same problems with GESSAR because when you try to use a new methods you can't use MARCH and corral. Because the important retention mechanism that's for those calculations is primary system retention. You can tell the code anything you want. So we tend to bend them into rather conservative releases.

(Slide 13 shown.)

MR. PRATT: These are the two additional classes that were generated as a result of the SARA document. The class one IS sequence is really a seismically induced sequence leading to principally a failure of the RHR suction lines.

And what we find is a draining of the suppression pool prior to core melt, to the level of the RHR suction lines, so that the ex-quenchers are submerged and the down covers are uncovered so that any fission products released from the drywell through the ventilation release would completely bypass the pool. Whereas anything released down the SRV would enter the sump pool due to the small deductions of the pool.

The class S sequences are various in terms of consequences. We have a seismically induced sequence, most of the probabilities coming from the seismic event. Also thrown in there is random reactor pressure vessel failure. It is a smaller probability than the seismic event.

You really have a massive failure of the primary system equivalent to a large break LOCA and failure of the containment building at the same time.

Again you are melting down into a depressurized system in the containment building and again the source calculations that we performed for that sequence are again rather conservative.

MR. DAVIES: Dr. Pratt, in some of the Mark III designs, as I understand it, there is a returning water storage tank dump feature. If the suppression pool level is below a certain value then the RWST dump into the pool. Does this plant have that feature?

MR. PRATT: I'm not sure. We didn't take that calculation for that.

MR. DAVIES: That could make a big difference for some of these sequences.

MR. PRATT: We didn't take credit for that.

MR. BOYER: We don't have an automatic feature along that line. We can feed the returning water storage tank in through the storage system so we have a path but we don't have an automatic feature along that line.

MR. OKRENT: It is not a seismically qualified path, is it?

MR. BOYER: No.

MR. DAVIES: I wonder if it would be a significant effect on the risk if you had an automatic RWST dump. I guess that's probably not been looked at.

MR. PRATT: If the dump was sufficient to reflood and emerge the down cover IS sequence it would help that sequence quite a bit. For the S sequence of course it wouldn't. You have got automatic failure of containment and the primary system high up. You will see the relative contributions of the IS and the S sequences. These two sequences essentially dominate the early fatality.

MR. DAVIES: I was thinking more of this feature for a class which you showed on the previous slide which are not seismically induced. Containment failure before core melt where the boil off of the suppression pool. Then you get this additional source in there which could be very effective it seems to me in delaying the release and scrubbing the material.

MR. PRATT: That's a good point.

MR. EBERSOLE: Is the loss of water here due to drainage or to overpressure?

MR. PRATT: For which sequence.

MR. EBERSOLE: Class IS.

MR. PRATT: Drainage.

MR. EBERSOLE: The drain phenomenon is due to the profile of the system, the vertical profile; is that right.

MR. PRATT: Do to the severing of the pipe.

MR. EBERSOLE: Severing of the pipe which happens to be below second level. There is not a deliberate feature to avoid that sort of thing. It is simply a gravity drain. It would be enhanced in leakage by overpressure, wouldn't it.

MR. PRATT: Yes. By the time we are melting the core we have assumed that the whole thing is drained.

MR. EBERSOLE: Well, even the addition of more water would not stop the drainage process, would it? It would just come late?

MR. PRATT: I see what you are saying.

MR. EBERSOLE: What could you have done except forestall disaster.

MR. DAVIES: Timing is very important of course if you are talking about evacuation and --

MR. EBERSOLE: I was thinking you wouldn't have much incremental time.

MR. DAVIES: There is a lot of water in the --

MR. ROSENTHAL: For the class two sequences it is a long time containment failure, so I don't think that we have a sensitivity to the details of moving RWST water in the containments. We are talking about over data failure.

For the class four sequences we have been ATWS and you postulate that you are dumping 20 to 30 percent of the core power into the wetwell. You have to move a lot of water in and take a lot of steam out in order to dramatically change the progression in the class four sequence, so although the dumping the RWST is an interesting concept, I don't think it would significantly change the picture that we are prepared to present today.

(Slide 14 shown.)

MR. PRATT: The next viewgraph I'm going through is rather busy. I'll go through it rather quickly. It is simply to give you an indication of the relative frequency of the various classes as identified at Limerick PRA.

MR. OKRENT: When are you going to show a picture of a containment building and go through some scenarios that resemble the alphabet soup, as it were, that we have on the left-hand side?

MR. PRATT: It is there. It is about four or five --

MR. OKRENT: Is it possible to give the physical picture of some of these before you --

MR. PRATT: Sure.

MR. OKRENT: -- interrupt the flow --

MR. PRATT: Reasons why I would like to give you perspective is so that when we talk about the uncertainties you can get a feeling of how important in these particular accident sequences those uncertainties are.

I'm trying to give you the risk profile, show you what is important, take you through the sequences, show how uncertain they can be and do the sensitivity study with you to show you how it impacts risk. That's why I'm following this progression. We have three hours and that's why I figure I had a bit of time to go through this.

MR. OKRENT: Go ahead.

MR. PRATT: The main point of this viewgraph is to show you that we have -- this is a slightly different representation of the sequences. We have grouped them into frequencies that are considered to be regional disasters and the significance of that is the evacuation model that we assumed is extremely conservative. And someone who is here from the staff could address some of the consequence analysis if you are interested in it.

But the evacuation model there is extremely conservative. Very little movement of the people seems to be interrupted as a result of the initiating event whereas the non-regional disasters you have a different evacuation model.

So that's one of the purposes for showing this and one can see some of the seismic events do have significant probabilities under the regional disaster category which do impact the evacuation model.

What was that G level we assumed?

MR. ROSENTHAL: Point four.

MR. PRATT: Anything above point four we assummed the evacuation model would be significantly effected.

MR. DAVIES: I notice in the SARA document they assumed .6-G as the break point. Is there a big difference in those two values in terms of sensitivity?

MR. ACHARYA: My name is Acharya from the NRC staff.

When the staff reviewed as to what should be the proper assignment of the z-pad to differ the very severe earthquake from the non-severe one as far as off-site damage is concerned, our experts thought that the .4-G is more appropriate than the .61-G that was chosen by the other experts.

MR. DAVIES: Thank you.

MR. OKRENT: Did anybody analyze what the over passes are likely to be able to withstand, for example?

MR. ACHARYA: We did think along that line to assess as to what could be the possible seismic load that the load system can take. It was not very possible to come up with that.

Simply we relied upon the damage descriptions that were provided for the modified Mercalli intensity scale. That was also primarily referenced the use by the applicant and we went by the description as to what kind of a off-site damage this could be caused by various MMI earthquakes. We did not do a separate off-site analysis.

MR. EBERSOLE: It seems that in your approach to this problem you were taking the seismic event as a phenomena. And then you aimed your sights at whatever it took to get rid of the suppression pool water and you found a place which was the pipe failure, right?

MR. PRATT: Go through that again.

MR. EBERSOLE: I say you picked the seismic event as the physical event that challenged the integrity and then you set about finding a place where you could challenge the integrity of the suppression pool and you found one, you thought.

Is this plant privileged to have the feature that's currently called in GESSAR II the UPPS system?

MR. PRATT: No.

MR. EBERSOLE: So you are not accounting for that? Had you accounted for it, I might have found a way home, because there is still a way to cool the core, there is still a way to evaporatively discharge the suppression pool. But you don't have it here?

MR. PRATT: No.

MR. OKRENT: Go ahead.

MR. PRATT: In terms of viewgraphs, perhaps as you would like to get to the picture we could spare you the risk perspective.

MR. OKRENT: I don't want to interrupt the flow of your talk. It might have tramatic experiences for --

MR. PRATT: I'll focus in on the table that I really want you to see and we will -- the main point of the next two or three graphs was to distinguish between what we call early damages and the long term damages and that they are coming from different sources than the short term damages that are very sensitive to, for example, to the quantity of fission product released and the evacuation times and so on.

The longer term damages are very insensitive to that. If we could just show you, flip over the next two or three viewgraphs and go to this extremely busy table --

(Slide 15 shown.)

MR. PRATT: -- this is taken straight directly from the final environmental statement for the Limerick generating station and shows -- a selection of the source term failure modes along the top of the thing. The various consequences, conditional on the accident actually occurring, there is no probability of the failure mode of the accident into this.

If you look, for example, at item six total person-rems relatively, little insensitivity to the evacuation model is assumed. There are three different evacuation models. Whereas if you go to the early fatalities one does see a rather large sensitivity.

What I'm going to be doing is using these numbers which are conditional upon the failure mode occurring to show you the importance of uncertainty in terms of containment failure and so on, so I'll be using these numbers together with the probability of accident sequences and the various condition probabilities of the failure modes to be assumed to show you the sensitivity of the results to those uncertainties. That's really the point of putting this in.

(Slide 16 shown.)

MR. PRATT: The main point of this -- this is the viewgraph that was put out by Mr. Coffman at the last meeting and the question was as to how robust are these particular calculations. Ones that I'm going to be looking at --

(Slide 17 shown.)

MR. PRATT: -- I'm going to be looking at the calculations for the entire region. So we will be dealing with early fatalities of .005 for reactor year and total person-rems of around about 1,000 person-rems for the reactor year.

Relative to those numbers I'm going to be looking at in terms of sensitivities of the various phenomenology. Again, this is taken directly from the final environmental statement.

Well, I would --

(Slide 18 shown.)

MR. PRATT: -- what I've done here is breakdown the 5 -- actually .49, we rounded it to 5 acute fatalities per reactor year and show the contributions from the various classes.

One could see by far the largest contributions is coming from the external events, 96 percent, and most of those coming from the IS and the S classes, the IS being one in which we severed the RHR suction lines so you have got about 60 percent of the .005 acute fatalities coming from that particular damage.

MR. DAVIES: On these results, I notice in NUREG-3028 you make the statement that part of the change in consequences is due to your revisions in the CRAC code. In fact, it says in there a factor of three increase was noted by just revisions of the CRAC code having nothing to do with any assumptions in the source terms or the front end part. Is that the case in these comparisons also?

MR. PRATT: Yes. In fact, I have a -- do we have the other viewgraph?

Let me the -- calculations that --

(Slide 19 shown.)

MR. PRATT: -- I think this illustrates what you are saying. The first acute fatalities normalized assuming an accident had occurred. The BNL calculations there is a factor of three, the next column over.

We then revised some of the calculations, set a different value and used the NRC side model and the numbers we used and the LGFP and also the final environment of statement during the final column so you can see that the factor of three is similar qualifying the end results there.

MR. DAVIES: Thank you.

(Slide 20 shown.)

MR. PRATT: The purpose of this particular slide is to show for the longer term damages the person-rem calculations -- the contributors change. Most of it comes from the internal initiator events because they are the high frequency events and the person-rem are absolutely dominated by internal events mostly coming from class one sequences, the largest contributions with some contributions from the other classes so that there is reversal --

In a lot of these sequences one has good evacuation so that the early fatalities that are are not important for these sequences and but they are the more frequent sequences and they are important to the person-rem calculation.

(Slide 21 shown.)

MR. POWERS: Does this comparison also suggest that were this set of accidents that understanding the more slowly decaying isotopes and their release is more important than the more radioactive ID and whatnot, because of your evacuation model ID becomes relatively unimportant for these but the thing WASH-1400 does not treat very well becomes very important?

MR. PRATT: That's true.

I would like now to get into the discussion of the progression of the accident sequence.

The sequence definition on this particular case is that the reactor -- this is a -- excuse me. I'm going over it rather quickly here.

(Slide 22 shown.)

MR. PRATT: What I would like to do is talk about class one sequences go through a sequence definition core melt phenomenology, containment failure codes, and looking at the impact of uncertainties Class one sequences and discussing the sequence definition to start with.

(Slide 23 shown.)

MR. PRATT: The important sequence definition points from the point of view of our assessment is that reactor is scramed, the coolant makeup fails from the start of the sequence of the surface of the transients in that field into this particular grouping would go on for a little bit longer, but we are assuming for the purposes of our calculation that it fails initially for about 60 percent of these class one sequences the system would be at high pressure, which is one of the questions that Dana Powers asked at the last meeting. So a large chunk of these classes of sequences are at high pressure during the --

MR. EBERSOLE: Explain the third, because it suggests that the ADSS system is the culprit. I've long wondered about the electricomagnetically driven ADSS air power systems. What part of that didn't deliver? Was it electrical function? Where was the principal contributer to failure in the ADSS system?

MR. ROSENTHAL: It was manual failure to depressurize the system, not hardware. People. And the procedures have been -- new emergency procedure guidelines would instruct the operators to depressurize the utilities sensitized to it, they have trained their operators to do it. In the PRA though at the time the PRA was done it was perceived to be dominated by human failure.

MR. EBERSOLE: You must have read the --

MR. ROSENTHAL: Not at the time the PRA was done.

MR. DAVIES: The logic has been changed. It doesn't require a manual actuation, is that correct?

MR. ROSENTHAL: But that was not taking credit for the PRA which was done sometime prior -- remember, we are always working with documents were the work starts some years ago.

MR. EBERSOLE: There was a long grinding flap about what you should call ADSS and it was decided it would be a low level. It was automated. Has that changed?

MR. ROSENTHAL: Let me remained you even with automatic ADSS system the operator can sit there and defeat depressurization. We have been telling people that human factor contributions are important.

MR. EBERSOLE: Insert himself in that 90 second delay.

MR. ROSENTHAL: He can.

MR. EBERSOLE: Is that where this took place, you stuck himself in there and blocked it?

MR. DAEBELER: No. Initially during the -- in performance of probabilistic risk assessment it was taken into account that the permissive on high pressure was included. In fact that is not the case now. It is ADSS actuates on low level only so that the reason for the high values in -- lack of initiation of ADSS are due to operator action manually initiating the SRV's in this particular sequence.

MR. EBERSOLE: This is due to the failure of the automatic mode.

MR. DAEBLER: This is not due to failure of the automatic mode. It is failure of the operator.

MR. EBERSOLE: The automatic mode has failed, he now fails to depressurize manually.

MR. DAEBLER: No. In the sequences you can get low level without high drywell pressure. And in those cases, then it was anticipated that the operator would actuate the SRV.

MR. EBERSOLE: But that high pressure has been eliminated.

MR. DAEBLER: That's correct, but that is not taken into account in the PRA.

MR. EBERSOLE: So you get a through line leak and you don't get high pressure so the operator didn't pick it up.

MR. DAEBLER: Correct.

MR. EBERSOLE: Thank you.

(Slide 24 shown.)

MR. PRATT: Going on to the fourth point, the core damage will begin with containment assumed to be intact except for those small fractions where we had a bypass and suppression pool is subcooled out of core melt.

MR. EBERSOLE: Am I correct in saying -- well, wait a minute. As you were. No question.

MR. PRATT: What I've tried to put out here is some of the issues that I think we should be addressing as we go through the phenomenology. This is a kind of a shopping list that I thought got from the last meeting.

The release of the core materials from the primary system, we have shown that there is a certain fraction. Whether it is 60 percent or not of the class one sequences that will be at high pressure as a core meltdown occurs and the importance of whether or not we have a high pressure ejection or low pressure ejection can be important in terms of containment building failure mode calculation or to whether or not we are dealing with a local or gross vessel failure.

Calculations performed by the INCOR code and MARCH code assume relatively slow going through of the vessel which takes a long time and assumes a rather large failure. Indications in particular in appendix H.

PRA indicates that local failures could occur. So for a high pressure case in which you have a low failure of vessel head there is of course the Sandler experiment that shows that core debris could form very fine particles which would directly heat the containment building. That's a concern and something that we should address in the uncertainty.

MR. EBERSOLE: If you are talking about a core melt and the presence of high pressure, are you talking like 1100 psi.

MR. PRATT: Thereabouts.

MR. EBERSOLE: Wouldn't it be a fact of life that core melt would proceed to degrade the lower part vessel first by gravity effects and there would be some discharge from the bottom which could be anything you care to postulate. That's where the rod structure is, after all. And there would be --

MR. PRATT: By the bottom vessel you mean the containment building?

MR. EBERSOLE: No. The reactor vessel.

MR. PRATT: That is where it is coming out.

MR. EBERSOLE: An imaginative person would say you have the beginning of rocket take off. But I'm not going to be that way. I'll say instead you leaked the pressure off from the bottom in a moderated way and you don't blow the head off. That's not hot yet. For heaven's sake.

MR. PRATT: You don't blow the head off.

MR. EBERSOLE: The bottom is the one that's going to get the non-benign treatment of the hot core.

MR. PRATT: That's not what we are suggesting. If I did, I apologize. The concern was as the core debris piles up in the bottom of the vessel the codes that we have will predict the blow of the wall assuming there is no penetration and that could take a couple hours to melt through the core.

MR. EBERSOLE: Assuming there is penetration. There is two hundred odd penetrations.

MR. PRATT: And the point we are making is that there was a writeup in appendix H of the PRA and has been well documented in the IPPSS and EPPSS. Now people are coming around to believing that the wells are going to give way and these things are going to move out and that you will have the high pressure ejection of the molten core materials out of the bottom of the vessel.

I'm sorry. That's the phenomena I'm speaking about.

MR. EBERSOLE: That's the one I wanted to hear.

MR. PRATT: I'm not an expert -- you have the expert sitting at your table. I can tell you what the effects will be on the containment response and how risk will change when we take those types of things into account. But that's precisely the phenomena I'm dealing with.

For those situations where we have the core debris coming out the bottom of the reactor vessel, the next set of items are also of interest. Whether or not we are dealing with high temperature melt or a relatively low temperature.

We are not sure of that range of temperatures. If of course we are dealing with depressurized sequence that comes out and drops on to the diaphragm floor and if it is molten then it could spread. If it is solidified it may hold there and have to reheat before it can spread.

Also relative quantities of steel Zircaloy fuel and also the fraction of metals oxidized can be important.

MR. EBERSOLE: What if the floor is wet or covered with water.

MR. PRATT: The next item water supply to the core debris, extremely important. And also plant specific considerations.

I would like to go through that a little bit. And look at some drawings and go through some of that stuff.

MR. OKRENT: Just a brief question.

I on occasion see reference to the possibility that pipe failure will occur before vessel failure due to heating. Where do you factor that in to your issue?

MR. PRATT: In fact, a lot of people would like to see high pressure ejection go away and that's one of the ways we are investigating trying to get rid of it. I would think it would be a beneficial situation because you would be depressurizing the primary system somewhat remote from the core and you would have have this phenomenon that you seem at Sandler. And I think it is highly likely if you look at some of the calculations that one does.

Most of the calculations that I've seen look at convected problems that might be set up in the primary system and you get beyond the kill point of the steel, which would imply there would be some degredation.

There was a calculation done in which they coupled together the MARCH corral merge trap meld combination and found when you take into account the heating component the fission products that are ultimately in the primary system you find this effect occurs even more.

So it is important and I think it would be beneficial in terms of depressurizing the primary system prior to this rather energetic blow down of the core debris.

MR. EBERSOLE: Let me make it a little more entertaining.

At the time that you got the core in this melt progression at the bottom and challenging the integrity of the bottom of the vessel, all at once I successfully manage to unload tons of water on top of this mess. What do I have then?

MR. PRATT: Under those circumstances that it would not be cooled.

MR. EBERSOLE: I'm talking about the mode of vessel failure. Up to that point it had no pressure in it to speak of.

MR. PRATT: If you go with the assumption that the local failure will occur rather rapidly then you get in with a very short period of time and you have virtually no effect at all. If you are talking about a situation where assuming these penetration don't come away and they are breaking the vessel head over many hours then it is important.

MR. EBERSOLE: If I've obtained a state where the bottom of the vessel is weakened by thermal effect and then I successfully introduce large amounts of water won't I catastrophically fail the bottom section of the vessel?

MR. PRATT: Sure, it might do that.

MR. EBERSOLE: That would lead to some spectacular consequences.

MR. PRATT: Not anymore spectacular than --

MR. BOYER: We will be covering some of that aspect in our presentation and we can do it at any time so if you wanted to go back and forth --

MR. OKRENT: Well, I suspect we best stay with the staff presentation, but if you have specific points of clarification that you think would be useful, please feel free to make them during the staff presentation and I'll ask that they exercise the same prerogative.

MR. BOYER: Fine.

(Slide 25 shown.)

MR. PRATT: For the Mark II containment, and some of these slides are somewhat general. I've cut and pasted a good deal of the presentations I've given over the last couple years or so. And the wetwell under drywell are directly above each other and the diaphragm floor is really the separation between the two.

MR. OKRENT: Show a picture.

MR. PRATT: I will show a picture.

MR. OKRENT: I know you have it engraved in your mind but it may be --

MR. PRATT: Not a very good one, as a matter of fact. Never mind.

I think the implication -- what we are talking about is core debris coming out of the bottom of the vessel and hitting the diaphragm floor here. The viewgraph points out that for the Mark II designs there is going to be limited water availability on this floor. It is going to be limited to the height of the downcome above the floor.

So the initial water interactions unless you have restoration of ECC will be limited and the whole progression of the subsequent interactions depends on how fast this core debris gets through the floor into the suppression pool, except for the high pressure situation where this whole thing is blowing around here in rather violent fashion.

So we would look at and have looked at sensitivity studies for the depressurized case in which we assume the core moves down, sits on this region, some of it may drip down through the downcomers into the suppression pool.

The interactions with the floor and the pressure temperature history is a function of core degredation beyond that point. There are in this region only a couple of drains so that the drainage down through here would not expect to be that large.

The next viewgraph I think is interesting from a general perspective point of view. Let me show you this one.

(Slide 26 shown.)

MR. DAVIES: Before you leave that, I think this is not a correct representation of all Mark II's. Show them for example --

MR. PRATT: Next viewgraph.

MR. DAVIES: How much did you assume bypassed the diagram and got directly into the suppression pool?

MR. PRATT: We will get to the best estimate calculations. Our uncertainty analysis took extremes.

MR. OKRENT: How is the vessel supported in Limerick?

MR. PRATT: On this thing.

MR. OKRENT: With a skirt?

MR. PRATT: Yes.

MR. ROSENTHAL: Could we take a moment on the direct heating. I would like to note that -- if I'm jumping ahead, stop me.

Number one, we have an inerted containment here rather than a non-inerted containment. That has to affect your perception of what is going to burn.

If you have your mental picture of Zion and its instruments, tubes and tunnels, in your head, you compare it to the layout on the slide here, I think you would conclude that the -- that the dynamics of high pressure ejection of propelling material into an upper atmosphere are surely different here from a geometric standpoint seem less likely.

MR. OKRENT: What is the nature of that seemingly solid cylindrical pedestal which the vessel is sitting on? Are there any openings in it?

MR. PRATT: Yes. There is a man-way, person-way here. On the next viewgraph -- let me show you the differences.

(Slide 27 shown.)

MR. PRATT: This is really Pete's question regarding the different configurations.

Limerick is over here and this is taken from an RDA document, just to give RDA a bit of a plug there.

Diaphragm floor, as I said, nothing really here except drains. That doesn't occur here -- as I recall when I visited the site it is flat. There isn't something to step over. It is flat. So there would be a tendency for the core debris in this particular configuration to flow outwards. We don't anticipate a great deal of it going down at this point.

If, however, you look at Shoreham, there are downcomers two feet diameter right under the vessel. So there you would expect core debris to pass down into the water below.

MR. EBERSOLE: As I look above the diphragm floor I see a relatively standard design. Below it I see freestyle. Evidently anybody can build anything he wants to build. Presumably if they were building ten others they would all be different. When one gets as far out in this realm of imagination as we now are do you see any configuration that is better than an other that could be deliberately used without any penalty?

MR. PRATT: You would be getting into a quite a wide area of discussion. Some people would prefer to have extensive core concrete interactions, such as the Germans. They enjoy that. They like it. They don't want water near the core debris.

MR. EBERSOLE: Turn it off and dry it up permanently.

MR. PRATT: They will constrain things to be as dry as they can. In this country the intent is to try to put water into the core and quench it and form a coolable degree bed and terminate it.

MR. EBERSOLE: As soon as we can.

MR. PRATT: That's generally been the philosophy. If you can form a coolable degree bed you prevent all of the nasty vaporizations and so on.

MR. EBERSOLE: Wouldn't this lead to some configuration that would slowly and progressively dump the core into the water?

MR. PRATT: That's what you want. This might be a little bit sudden. I don't know. I would have to look at the calculations. We are just reviewing the Shoreham PRA now.

I don't want to leap ahead of myself in terms of what our conclusions would be there. But certainly I would think that one would like to get the core degree into a coolable degree bed as quickly as possible.

In this particular case one could imagine extensive core concrete interactions at this time and when you look at the phenomenology on this the way we got the upper and lower bound uncertainty at Limerick was to one assume that rather non-mechanistically that all of the core debris got into the water very fast, intimately mixed with just a very small fraction of this water and failed containment at the point. That was an upper bound calculation on risk.

If however you make the assumption that it spreads and gets down there slowly, becomes coolable, then you fail many, many hours, days into the accident, consequences go right down. So that retaining the core debris right here generating a lot of non-condensable gas and failing within two hours is sort of an intermediate assumption. It tends to be rather a conservative use of the RSS methods.

MR. EBERSOLE: All right. I presume you are doing the analysis for all three designs?

MR. PRATT: Not today. Next time when you want to hear about Shoreham.

MR. OKRENT: Maybe you are. I'm not sure.

MR. PRATT: This is rather like the Swedish. Zimmer -- the Swedish design -- a lot of concrete around.

MR. POWERS: You explained through that man-way or doorway that is depicted on the Limerick, what is on the drywell wall opposite that? I'm thinking of things being spread not just blown out, but ordinary splash or more energetic splash. What does that impinge upon?

MR. PRATT: I'm going to have to get help there. I remember when we went around the site, the plant, looking at it and -- in this region -- I'm not sure.

Could someone help me as to exactly what faces the door when you are looking at it?

MR. BOYER: Exactly which door are you talking about now?

MR. PRATT: From this region.

MR. BOYER: That's just into the floor region and it is clear in front of it there are downcomers of course in the floor.

MR. PRATT: On the wall?

MR. BOYER: Well, there is piping between the reactor vessel support and the containment wall, of course.

MR. POWERS: What I'm looking for are in fact penetrations that might be suspect to damage and give you a mechanism to bypass the suppression pool as far as --

MR. BOYER: No penetration at that load level that I can recall. Most of the penetrations are up higher. You are talking about in the wall there are penetrations in the floor. Pipes are passed through the floor and extend above the floor something like 18 inches or so. And have a cap about a foot above that 18-inch section, the 18-inch section being solid pipe and the area above that being open, just enough to support the cap. So the water would collect on the floor until it overflowed into the vent pipes.

MR. EBERSOLE: These differences that we saw in these three pictures -- I made a somewhat synical comment that they might have been made by a draftsman as a personal choice.

MR. BOYER: At this time these containments were being designed there were different design groups doing the work and it was individual design efforts which resulted in the variation, but they were designed by different groups and people were looking at different aspects of the diphragm floor and penetrations through it, the support for the reactor vessel and other things like that and I think they were probably given more consideration than the ultimate accident you are talking about now as being a primary design factor.

MR. EBERSOLE: Thank you.

MR. POWERS: Can you tell us what the flow pathways up around the vessel look like at this plant.

MR. PRATT: There is an opening at this level as I remember.

MR. BOYER: There are gradings, of course, at various levels all the way up, about every ten feet in height there is a grading -- you can walk completely around the building and, of course, there is an extensive amount of piping throughout the whole containment, and piping supports and snuffers and whatnot.

MR. POWERS: What I'm interested in is the availability of flow paths directly from the diaphragm cavity up around to the vessel.

MR. DIEDERICH: Through the annulus between the vessel and the biological shield?

MR. POWERS: Exactly.

MR. DIEDERICH: There is a flow path for things to go through there, up through the area underneath the drywell.

MR. POWERS: Rough estimate on floor area?

MR. BOYER: This diagram does not show the drywell head which is bolted on to the lining of the containment and is present -- is a hemispherical head somewhat similar to the reactor vessel head, but larger diameter. And exists in that space that is shown above the reactor vessel head.

MR. PRATT: This region.

MR. BOYER: Right.

MR. DIEDERICH: The dimension of the annulus is in the order of six to eight inches.

(Slide 28 shown.)

MR. PRATT: In terms of our best estimate calculations, which is what was asked of us, what we assumed was that most of the core debris would be retained in the diaphragm floor. And we calculated the buildup of pressure temperature history as a result of non-condensable gas generation during core concrete interactions.

And the assumption was made that the containment building would fail at the point of 70 centimeters penetration of the diaphragm floor or whichever came first whether or not we reached pressure of about 140 psig.

Later on what I'll do with you is go through some of the recent work that has been going on in the containment loads and containment performance group to show the sensitivity of this assumption to those types of calculations. We assumed that the combustion of the condensable gases of course was prevented because of the melting except for a very small conditional probability. And that early containment failure is the result of, say, steam explosions or direct containment heating. So it would be of a relatively low probability.

MR. EBERSOLE: I think I asked about the inerting. This containment is inerted?

MR. BOYER: Yes.

MR. EBERSOLE: How did you strike a balance between the risk of death of operators that have to go in versus the inerting process?

MR. BOYER: We inert within 24 hours of going to power so that we have the option of checking things just when we are pressurized before we go to power and for making inspection just prior to coming off.

We can start inerting when we are coming down. We can check the containment prior to coming for any leaks or something of that nature, other than we lost a battle on the need for inerting and the relative convenience of getting operators in there.

MR. EBERSOLE: Do you send people in with air packs?

MR. BOYER: No. We shutdown and deinert and go in.

MR. EBERSOLE: I guess we mentioned it before, don't you anticipate the horrendous maintenance down time because you can't get in and fix some feature?

MR. BOYER: It hasn't occurred. We don't have much electrical equipment in there. Mostly it is brought out through transducers and whatnot outside so that hasn't been a problem at Peach Bottom in ten years experience.

MR. EBERSOLE: Do you have that same transducer location here?

MR. BOYER: Yes.

MR. EBERSOLE: You have excess flow checks.

MR. BOYER: Yes.

MR. EBERSOLE: Thank you.

MR. POWERS: When you say this is a best estimate that the diaphragm floor doesn't fail it is really just an assumption here?

MR. PRATT: Does not fail?

MR. POWERS: That's right, the debris retained there rather than having the floor promptly fail, that's really just an assumption, it is not in fact a best estimate based on some analysis of the floor?

MR. PRATT: Based --

MR. POWERS: I can imagine one going through an analysis of debris dropping and breaking the floor just from the impulse to it, thermal stress causing it to fracture rather than just blatant random analysis like that, if that was not done --

MR. PRATT: I can remember you being asked to do that for the containment loads. So what was your conclusion?

MR. ROSENTHAL: I would like to note when we looked at that floor it looked like there was more rebar in the floor than concrete in that concrete floor. And I believe --

MR. POWERS: I'll share with the committee my results. Indeed the floor would not fail under a depressurized injection of the melt. But it should in fact be squirted out under pressure and it would fail promptly.

MR. PRATT: That's right. I think what we are saying is that the best estimate calculations were really principally based on depressurized sequence and one must deal with the high pressure sequences in terms of sensitivity studies.

MR. DAVIES: With regard to the steam explosion I believe your analysis assumed a ten to the minus fourth probability for steam explosion independent of the mechanics of the sequence.

MR. PRATT: That was almost a God-given NRC decision there.

MR. DAVIES: That was what I want to question. I've seen other numbers like ten to the minus two for low pressure interaction, ten tc the minus four for high pressure.

MR. PRATT: We went through a number -- in 3028 we looked at different numbers and assessed the impact there. For the purposes of doing the final environmental statement we made that assumption. I can show you the sensitivity of that assumption in the final assessment.

It was done -- if you like to consolidate all of the various thing together -- so we had a picture for the final environmental statement so that's where that number came from. Results are not terribly sensitive to whether or not it is ten to the minus four or two. It is not that sensitive to one.

MR. DAVIES: Thank you.

(Slide 29 shown.)

MR. PRATT: This is a pressure history which is taken directly from 3028 and I'm putting it up really to show you the various energetics for a particular sequence.

In 3028 we looked at a wide variation of input assumptions in terms of how much the core might spread whether or not it would be a high or low temperature and you can get different tracings to this based on those assumptions.

The important thing, though, is to look at the time from core melt beginning, which is the time when we assumed the warning would occur, to the point of the release, at this point, so it is that difference in time that goes into warning time for the CRAC analysis.

So you are interested more in how many hours there is in here.

If we walked through this, I've shown in the dotted line the Limerick PRA calculations which were done using the INCOR codes and we of course used MARCH. I think the traces are rather similar. They are displaced in time principally by the use of this vessel failure model which does give you rather longer times to fail the vessel than MARCH does. You can see the difference in here, core melt beginning round about the same time, slumping round about the same time for both codes but our head failing in this time frame, whereas this is where the head failed in PV melt.

The subsequent interactions become somewhat similar. This was an intermodel and this was also an intermodel. For this particular assumption we assumed that the containment failed after we had penetrated 70 centimeters of the flow. There is still a way of failing as a result of overpressure.

MR. DAVIES: The question on your melt begins -- the BNL calculations show a lower start of the melt, but in your report I thought you said that you had revised the MARCH calculation to account for a 20 percent increase in decay heat early in time.

MR. PRATT: I love the way you set my next viewgraphs up.

MR. DAVIES: Why do you start earlier than the other calculation which has the old MARCH?

MR. PRATT: Because this was the old calculation.

MR. DAVIES: Which one?

MR. PRATT: This was in 3028. I will show you how it changed when we did the calculation --

MR. OKRENT: Would you mind repeating what you said about your assumption that failure occurred with 70 centimeter concrete penetration, although you were not overpressurized?

MR. PRATT: Yes. The assumption was made -- this is something that we kind of followed from the PRA -- was that if you failed the diaphragm floor that even if although the pressure was not at the point where you predict failure it would give. It was the lateral support of the floor that was lost. That's not true of all Mark II designs. Some of them are freestanding designs and wouldn't fail quite the same.

MR. OKRENT: And the two suare foot hole in the drywell, this is --

MR. PRATT: You have to deal with that in terms of sensitivity studies and if you get up to about two feet and much beyond it is a depressurization in how that goes into crack and beyond that you could make it seven foot, eight foot and you were not going to get much difference in terms of concentration. Make it smaller than that and it is a bigger factor.

MR. ROSENTHAL: I just wanted to make the point that the initial sharp pressure risers are not significantly larger than the design pressure containment. 150 minutes. The challenge to the ultimate containment pressure is out at six hours or so. And the rate of pressure increases -- it is not too fast so I don't think that we are too sensitive to the details of the containment fail at a 130 or 140 or 120 psi. It is far less important than the fact that it is believed to survive for several hours.

MR. PRATT: I think that's right. The important thing is this assumption here. We have this spike as predicted by MARCH in there. PECO's analysis put a dotted line that this is a scenario of uncertainty and this is how you have to deal with the phenomena of vessel failure in terms of PRA.

You really don't know -- this is predicting it doesn't fail for one particular calculation. One could envision other calculations where you might run into trouble. You have to allow for the possibility of failure earlier in a probabilistic way.

MR. EBERSOLE: That diagram has a problem I mentioned awhile ago, I can't see why you put up there, "head failure" when it is bottom failure that should occur.

MR. PRATT: That is bottom failure, yes.

MR. POWERS: The spike at RPV failure is just depressurization of the vessel or does that include interaction with this 18 inches of water on the floor?

MR. PRATT: It does, just a small amount in this particular case.

MR. POWERS: Small amount as in 18 inches. The suggestion was that the water could stand up to 18 inches high on this floor.

MR. PRATT: I would have to go back and look at the -- this was a couple of years back. There was an allowance there, what depth I'm not sure. No more than 18 -- you can't stretch your imagination that far.

This was kind of a best estimate type of picture in which you could do sensitivity studies and vary this time in here, which would affect the warning time. But then you would have to look at how you quantify this in the containment entry.

The assumption was made during this time period because we were at high pressure and high tempature that there is a potential for leakage into the containment building. In the PRA 50-50 split was assumed between the scenarios, that 50 percent of the time a leakage would occur sufficiently large to prevent this overpressure sequence from occurring and 50 percent of the time it would go on and fail as a result of either the going through the floor or overpressurization. So that's built into the assumption --

MR. EBERSOLE: Can you sort of parametrically curve your code or do whatever is necessary to hypothesize after the head failure, the bottom head failure, that you now introduce a physical phenomenon which from zero but gradually increasing increasing and get the molten fuel into the water and increasing rates?

MR. PRATT: Yes.

MR. EBERSOLE: I think it is going to grossly affect your story.

MR. PRATT: Yes. In NUREG-3028 we looked at the two extreme calculations assuming that, one in which the core degree went down and interacted and produced a steam spike that would fail right here. This spike went right up.

As far as containment we ananlyzed that and that was our upper bound estimate. And the other one was that it went in in such a controlled manner it doesn't fail. It took many, many hours. So that's an important point and we did look at that.

(Slide 30 shown.)

MR. PRATT: To follow from Pete's introduction to this next viewgraph, this was a calculation we performed for the final environmental statement that did use the new decay heat curve that's in MARCH 2 and one can see that things do move forward in time.

Again, the point of the start of core melt is kind of the assumptions regarding the primary system boil out so that's why there is a slight difference in there, but it does speed the process up a bit. So these are the times that we assumed in the final environmental statement for the source term calculations.

MR. OKRENT: I missed something. You said a new decay heat curve, which differed in what way?

MR. PRATT: It tried to reflect the ultimate decay in the calculation so it did. It was about --

MR. DAVIES: 20 percent for the first hour.

MR. PRATT: Right, increase in the decay heat.

MR. OKRENT: That's enough.

MR. PRATT: So the question that was asked in terms of the containment failure modes are really whether you have early versus late containment failures and whether or not it is leakage versus overpressurization and the failure location whether it is in the drywell or in the wetwell below the pool to allow it to drain and also the effectiveness of the standby gas treatment.

(Slide 31 shown.)

MR. PRATT: This is the containment event tree in the PRA and they will probably be going through it in rather more detail.

Those questions that I mentioned are asked on the top. And one could look at it fairly quickly. Leakage versus overpressurization is 50-50 split. 50 percent of the time leakage prevents overpressure failure. Of that 50 percent of the time of the overpressurization failure mode you would assume the failure in the drywell versus wetwell.

50 percent of that went in the wetwell 10 percent of the time it would delay the water line which would drain the water.

MR. OKRENT: These are all subjective?

MR. PRATT: Yes, based on structural analysis that showed that the crack would occur round about the center of the diaphragm area and propagate upwards and downwards and that's where the 50-50 split came from. So the chance of getting down into the suppression pool and draining that was given a lower probability. But you are right in the sense that one must look at this and vary these numbers around to see the sensitivity.

MR. OKRENT: Well, let's say the estimate of whether or not you would leak sufficiently to prevent over pressure, what is the technical basis for whatever the number is that's used?

MR. PRATT: There was an estimate done by the -- I forget the AE for PECO which really didn't identify areas where you would get leakage. There was a subjective judgment made that leakage would occur.

What I can do is to go on later on and talk about the work of the containment loads and containment performance people who are looking specifically at this problem and their results are tending to support the fact that you would have leakage before failure in these particular designs simply because you have enormous temperatures in the drywell and the pressurization rates are rather slow.

So that if we can get over the initial vessel failure time the long term harsh environment in the drywell would tend to result in leakage rather than a gross containment failure.

MR. HUGES: I'm Gene Huges. I'll be making a presentation for Philadelphia Electric or part of it shortly.

When Bechtel did the analysis of containment capability we also had Chicago Bridge & Iron do an analysis of the head and attempted to determine other leak paths. We looked at penetrations. We met with a number of different individuals to try to determine if there was a leak point or weak point in the design.

We came to the conclusion that up to the 140 psig value in the development we really weren't sure whether we would have leakage or not. We weren't able to identify a clear path. The capability appeared to be there but there was large uncertainty so the value of 50-50 split was really a judgment call based on inability to determine an exact split.

MR. OKRENT: Let's see. When it leaks it then goes up through the filtering system, I assume?

MR. PRATT: A certain fraction of the time. Effectiveness of the standby gas treatment, so not all of the time was it assumed to be operative.

MR. OKRENT: Yesterday we were hearing about a theory in which leaks plug. Are the leaks you expect you might get here the plugable type or the non-plugable type?

MR. PRATT: No credit taken. This is old land, not new land.

MR. OKRENT: I'm asking Philadelphia Electric at the moment, I guess.

MR. HENRY: Bob Henry. I'll be making part of the Philadelphia Electric presentation. Dr. Okrent just requested whether the leaks were plugged by aerosol.

MR. OKRENT: Yes.

MR. HENRY: Typically you would not expect to find the large concentrations of aresol at this plant that you might find at a larger Mark III system.

MR. OKRENT: That should be obvious to me?

MR. HENRY: Will a simple yes suffice? As we will be talking about later in our estimates using current knowledge we would not expect to have extensive concrete tack in this plant which is something over and above what was done four years ago.

In the Mark III system which I believe you probably heard about yesterday these types of systems, these kinds of accident sequences you would expect to have a significant amount of concrete tack.

MR. ROSENTHAL: May I you address the leakage a little bit?

We are talking about an inerted containment here. So one knows prior to the event that you don't have a lot -- what one does have is containment isolation and that's a very nice feature is that the insurance of that you take credit for so you don't expect to have many linear feet of small leaks and be able to maintain operation.

MR. OKRENT: It is the leaks caused by the higher temperature and the --

MR. ROSENTHAL: I think -- well, we were worried about small cracks and whatnot or openings and would they plug. It is a preexisting -- in one set of considerations so I don't think we have a preexisting leak rate here and then worry about the plugging.

Now, one has the consequential inducement and you have a pressure phenomenon and a time and temperature phenomenon and the indication from the containment performance group was that the --

Well, in this case the pressures aren't excessive so that one worries about time and temperature and that's a long time to fail. It is some time and we are postulating containment failure due to over pressure before one would expect the time-temperature failures. So I think we are in good shape.

MR. DAVIES: Trevor, back on the SGTS for a minute.

In appendix D of the LGS-PRA it states that the operation of the SGTS under severe accident conditions is very important to risk. In other words, it has a very important effect on the source term. I notice you assign a 90 percent probability for successful operation of the SGTS. Now, I haven't looked at that system in this plant but in other plants there is a real question whether the system can operate under these conditions because filters aren't designed for this kind of loading. Furthermore they are not designed for the kinds of aerosols.

Where did the 90 percent figure come from? It seems a little optimistic to me.

MR. PRATT: That was the applicant's number and I think they will address where that came from.

MR. DAVIES: I'll wait, then. Thank you.

MR. PRATT: What I would like to do now is just -- this is taken from our input to the final environmental statement and these are our subjective judgments as to how the various failure modes will go together. We are assuming that this is very similar to the applicant's analysis.

Round about the quarter percent of the time it will be a drywell failure and wetwell failure. Rather a small fraction of the time it will be below the water level of the drain and this is a steam explosion failure mode. Hydrogen burn and there is an equal split between leakage which would allow standby gas treatment systems to work and that which it wouldn't.

Bear these in mind. Later on I would like you to see the sensitivity of the overall risk to these assumptions.

MR. OKRENT: I'm missing something. I thought I heard Mr. Rosenthal a moment ago suggest that you expect failure to occur before leakage could keep the pressure from going up. Did I missinterpret what he said?

MR. ROSENTHAL: What was done here.

MR. PRATT: Right.

MR. ROSENTHAL: Then either now or at the end of the presentation what the new information from the --

MR. PRATT: You are in somewhat an involving process here. This was sometime ago and in a rather subjective decision. It is more a statement of ignorance than knowledge, to be absolutely honest. And what we are trying to do is to better define. The latest results I have from the containment performance group is that some of the upper bound calculations that these higher pressures the head would lift and you would release the pressure through there so they wouldn't be dealing with a catastrophic failure.

MR. OKRENT: Which head?

MR. PRATT: Drywell. This is very preliminary work so I don't have the numbers on it. That's why I really didn't want to go through. We were just about to put those areas --

MR. OKRENT: You already have about a 50-50 chance of leakage of one kind or another.

MR. PRATT: Built into this analysis, yes. But I'm saying that's a decision made sometime ago and I'll show you the sensitivity of the results to that assumption.

For example, if the standby gas treatment system doesn't work this is a very severe release and will give you significantly higher consequences than drywell failure that occurs later. So leakage can work for you and against you.

MR. OKRENT: How robust is the .025 for wetwell?

MR. PRATT: This one?

MR. OKRENT: Yes.

MR. PRATT: This number is not robust, but the risk estimates are extremely robust. Because for the class one sequences this is my worst failure mode rather than this -- this is failing many hours after vessel failure. So most of my fission products are in the pool.

If I have got to get out and even if the crack is down below and the water draining -- so that the consequences here are not very large. For the class four accidents sequences this is a very important effect.

In our analysis, again not important because we use the DF of one. I don't care where I put it I have horrendously high results. Analysis done by PECO is a factor of ten difference between these failure modes and this one.

MR. EBERSOLE: You said if the standby gas stream system didn't work. Am I wrong in misunderstanding here how it is designed? I didn't think the gas treatment system could even begin to tolerate the local heat of this degree. That will not work. It will burn up.

MR. PRATT: I think what we are dealing with in this particular case where it does work, for example, would be a situation which you were dealing with very low leakage, just enough --

MR. EBERSOLE: Struck with a balance on the capability on the standby gas treatment heat treatment capability.

MR. PRATT: Wait for the applicant.

MR. EBERSOLE: That's one of the necessary criteria. It can just take so much and then it burns up?

MR. PRATT: That's right.

MR. BURNS: As Trevor said, I think that the question of SGTS effectiveness is very sequence dependent to the particular containment event tree that he put up was for a case where you may not -- you maybe rising in pressure in the containment and not have a particularly large load of fission products in the containment. SGTS provides a leakage pathway out of the reactor building for any very small, very small leakages out of the containment.

One of the things that was not taken into account in the Limerick design is that there is a recirculation system within the reactor building which has additional filters so that any leakage from the containment into the reactor building would have to go through the safety related recirculation system and also then to the SGTS.

So there are several filters in the reactor building. SGTS is that buzz word for a pathway out of the reactor building only dealt with very, very small leakages in which the fission product loading was very very small.

MR. EBERSOLE: That seemed to be a contradiction. In effect you are saying --

MR. BURNS: The effect of including SGTS in our method of assessments of risk is at most a factor of and only lateness.

MR. EBERSOLE: Is that because it failed a factor of two in the standby?

MR. PRATT: On a conditional value you can see the difference. I'll show you a difference in assuming it occurred. What he is talking about is overall risk perspective. I don't care --

MR. DAVIES: Appendix D of the PRA says it is very important but I don't recall it being a number there.

MR. PRATT: It is an important mechanism if it works relative to if it doesn't. But if you fold it into the overall risk assessment it is not a large effect.

MR. EBERSOLE: My problem is he says it only will take a small load yet apparently it doesn't work. That's what bothers me.

MR. BURNS: I'm saying there is no larger factor, at least in the risk assessment that we did.

MR. EBERSOLE: Thank you.

MR. PRATT: In running through some of the numbers and how they go together, try to bring out that type of point what is important and what isn't.

MR. POWERS: Your SE failure mode is -- I presume that's for steam explosion but in fact it only means rapid over pressurization due to fuel coolant interactions.

MR. PRATT: That was the steam explosion release taken right from WASH-1400. We didn't reanalyze that.

MR. POWERS: That whole failure mode probability there excluded possibility of promptly putting the core melt into the water?

MR. PRATT: Yes.

MR. POWERS: That turns into two assumptions, that the diaphragm floor does not promptly fail nor does melt flow across the floor and come down the downcomers.

MR. PRATT: Yes, but what I would like to do is show you sensitivity of the risk to that .01.

MR. DAVIES: Did you assume no containment sprays for any of these scenarios?

MR. PRATT: That's right.

MR. DAVIES: Notice that Philadelphia Electric intends tests of the system every 30 days and it could be reliably effective.

MR. PRATT: I think it is a built-in conservatism in the PRA but certain --

MR. DAVIES: That's certainly true, but the risk mix might be considerably different, that is the sequences which are important might change quite a bit.

MR. ROSENTHAL: Station blackout is still a dominant contributor. Surely the equipment is there and the staff did ask PECO to agree to testing that equipment and they agreed to do that. A step in the right direction. But it just won't change the risk profile very much as long as station blackout in seismic events has the picture that they now do.

MR. DAVIES: I would agree unless you consider recovery of off-site power after core melt.

MR. ROSENTHAL: I think it is in the right direction. They are going to do the right thing.

MR. OKRENT: By the way, are we asking you so many questions that we will not get through the material you think is important to cover?

MR. PRATT: I've lost track of time.

MR. OKRENT: In about an hour 15 minutes.

MR. PRATT: I think we have pretty well gone over most of the --

MR. OKRENT: If we are asking more questions than that, let me know, okay?

MR. PRATT: Let me run back very quickly over the fission product spray release calculations.

As I said, they were based on reactor safety study methods. For this particular case the pool was subcooled so we did assume decontamination factor of 100. MARCH, 1.1 and corral.

(Slide 32 shown.)

MR. PRATT: For class one sequences this is the way we analyzed it, principally initially the mount release goes down into the suppression pool. There is -- because for certain sequences we are at high pressure other sequences somewhat intermediate and low pressure there would be some fraction of the melt release flowing from the reactor vessel directly into the drywell region.

Of course, the core degree and vaporization release is directed to the drywell and the code does calculate the mixing assumed to be homogeneously mixed of the fission products in the drywell down through the downcomers and back again.

Whatever is needed to keep these things in balance throughout the course of accident sequence. So that if you have a number of hours after vessel failure, for example, you will get certain fractions of these fission products even coming in the vaporization release. You will see the pool will be scrubbed to some extent.

For this particular case, for our best estimate there would be a failure in the drywell and wetwell or in this region below the water level. And again as this occurs some hours after vessel failure this is actually the worst failure location in terms of fission product release because you bring down whatever happens to be in the drywell through the pool and out. For the other failure locations it tends to be somewhat low.

I don't think the calculations that we performed -- this is for 3028, calculations we redid for the NRC for final environmental statement are terribly different in terms of overall fission product release calculations. They are round about the same.

No credit was taken for primary system retention whatsoever and the only decontamination was within the pool. We didn't calculate any aerosol accumulation and settling in the auxiliary building for these sequences. No credit was taken for that. For the other sequence where we had leakage. For example, again there was -- seemed to be a bypass of the pool for the fraction of the releases that did not go down into the pool.

I don't know whether you want to spend any more time --

MR. EBERSOLE: Are there any penetrations at all in that floor where the molten core is sitting?

MR. PRATT: Penetration in the floor?

MR. EBERSOLE: Yes.

MR. PRATT: Yes, indeed, there are four-inch diameter drains. They lead down into a tank which is situated here.

MR. EBERSOLE: What did you do with them? Artificially plug them up?

MR. PRATT: No. Built into the calculation is the assumption that you lose a certain fraction of this core debris through that type of mechanism, 10 percent, for example, and then you would freeze the local area and you couldn't flow through.

MR. EBERSOLE: On the way down.

MR. PRATT: Yes, well, mechanistically the assumption. You can postulate anything you want.

MR. EBERSOLE: Is the design of that floor sufficient to comfortably carry the new load on it.

MR. PRATT: Again, Dana asked the question earlier and his conclusion was --

MR. EBERSOLE: Solid steel.

MR. PRATT: It would hold it for the cases where you would depressurize the vessel failure. If you had this high pressure ejection it would give way.

(Recess taken.)

MR. OKRENT: The meeting will reconvene.

Mr. Pratt --

(Slide 33 shown.)

MR. PRATT: What I would like to go into now is the impact of all of this in terms of uncertainties. And this really, again, is a shopping list of the concerns that I think we have all talked about over the last hour or so. The high pressure melt ejection and potential of direct heating of the containment atmosphere.

Jack mentioned, of course, we don't have an inert atmosphere, so the oxidizing might be smaller, but, nevertheless, a lot of heat is being dumped directly into the containment atmosphere. Steam explosions and potential, which is really a potential for other containment failure, failure of primary system during the core melt. This is the point that you brought up.

Also consider the failure location. Is it leaking or is it not? What I would like to do is go through and assess how those uncertainties change the risk numbers, so I'm going to have to go back to those and do a little bit of arithmetic and walk you through those tables a little bit to show you. But I think it is worth doing.

If you recall, these earlier viewgraphs that I gave you, where I showed that the total person/REM 685 of it was coming from the class one sequence.

(Slide 34 shown.)

MR. PRATT: And the acute fatalities is

essentially no contribution to this number, very small quantity. What I'm going to be doing is I'm working relative to those numbers to show you the impact on the risk.

(Slide 35 shown.)

MR. PRATT: If we go to the person/REM calculations, 685 person/REM, these are the conditional probabilities that we spent a bit of time discussing on the previous viewgraph. The conditional probability of a failure in the drywell, wet well and so on.

Steam explosion point zero zero one down in here. The total class frequency is given on the top, eight point three minus five.

In order to get this number, you would multiply the conditional probability by the class frequency by the conditional mean person/REM. That's why I gave you those -- those tables from the final environmental statement. So these are the person/REM for this particular failure mode, multiplied by that conditional probability by the frequency to give you the number here.

So you can see by looking at it, quite a lot of it is coming from leakage without standby fast treatment systems operating.

A little bit from failure in the drywell. These are lower consequences. This is five minus -- five times ten to the five person/REM. To give you feeling for the differences between the consequences of the calculations.

MR. OKRENT: By the way, what is the 95 percent person/REM that corresponded to one of those that has an exponent of seven?

MR. PRATT: I think we would have to delve into the --

MR. OKRENT: About.

MR. PRATT: Could you say the question again.

MR. OKRENT: What we see there are the mean conditional person/REM. And if you wanted to ask yourself, what is the magnitude of one of the larger events in the spectrum that's used to calculate the mean, I just arbitrarily said the 95 percent. I'm interested in any confidence level. I don't mean the absolutely least probable.

MR. ACHARYA: Details mean person/REM. Of course the conditional here -- conditional upon the -- concerns of the accident.

MR. OKRENT: I understand.

MR. ACHARYA: And whatever probability distribution that will be on the consequence magnitude, that will be coming from the various conditions during which the accident will take place.

Now, is mean such as ten million could have a distribution at the high end of consequence value, something like about 100 times more, but the probability will be something like ten to the minus four or so.

MR. OKRENT: Are you saying, though, at around the 95 percent point it is --

MR. ACHARYA: I've not looked at the probability in terms of the percentile. But the probability distribution that I'm imagining, of course, the person/REM of magnitude is like one in a thousand will have conditional probability. Almost any will result in 100 thousand person/REMS. But higher person/REM, we have to go for unusual conditions and load probabilities.

That magnitude a person/REM which will be 100 times the number that is quoted there like ten million will have in the probability of that will be something like one in one thousand, one in ten thousand.

MR. OKRENT: Okay.

MR. DAEBELER: One of the conclusions I get from that term is that the SGTS operability is extremely important because class one accidents based on a previous slide are the dominant person/REM contributors.

MR. PRATT: To show you the -- well, in terms of the overall risk profile here, it isn't important and I think that was the point. What I'm going to do to show you the range is to assume that every time we have core melt, we have steam explosion failure. Put that equal to one and everything else equals zero, and then pick the best condition, put that equal to one and put everything else to a zero, and I'll show you the effect of risk.

I think if you do put this equal to one that's as far as I could stretch my imagination with these things. I don't think we could do much worse than that. That gives you kind of upper bound -- not really an uncertainty analysis. It is an upper bound on the uncertainty associated with the core melt phenomenon. That's what --

MR. OKRENT: It doesn't give us any clue as to how valid your base case is.

MR. PRATT: It gives you a feeling, to me, that the calculations that we have here are conservative because we are very, very close to that upper bound. In fact, we are far away from it. I can see if I go and look at the improved calculations, the significant reduction in these numbers which would mean that my best estimate would go a lot further away upper bond.

MR. OKRENT: What is a factor of four from the upper bound?

MR. PRATT: If I take this and put it equal to one, run through the calculation, the overall person/REM will go up by a factor of four, no more.

MR. OKRENT: Well, I can do that in my head.

MR. PRATT: Right.

MR. OKRENT: But there is something built into that like a frequency -- in your heading and so forth and so on.

MR. PRATT: There is frequency built into this?

MR. OKRENT: Into the person/REM number.

MR. PRATT: Here? No.

MR. ACHARYA: In the last column.

MR. PRATT: Yes. But what I'm saying this is the one to look at.

MR. OKRENT: Those are big numbers and, some of them, and we just talked about how they are means and they are numbers, factor of ten or more are bigger, larger --

Continue.

MR. PRATT: [illegible] n really going to go through the arithmetic that you said that you have done in your head already. The point I'm trying to make here is that uncertainties of the nature that we have been talking about, steam explosion failures, failure earlier on, the most it could do is change this by a factor of four on conditional values. Right?

So that if I look at these failure modes, which I know I've got built-in conservative systems. I don't calculate primary system retention which I know exists. Pool DF are rather modest aerosol agglomeration in the auxiliary building is ignored. My conclusion would be these things, because I haven't taken the specific processes into account, my person/REM calculations which are coming from here, are very conservative. That's the whole point. I think it is a very important point. You are going to be looking at numbers that don't look anything like this on GESSAR. So your uncertainties become rather large in comparison to this type of number.

MR. OKRENT: Go ahead. Continue.

(Slide 36 shown.)

MR. PRATT: We can do the same thing for early fatalities and here you will see sensitivity, because we have got a lot of zero's. We don't predict early fatalities for these failure modes. We only predict early fatalities for a steam explosion release and, of course, we give that a very low probability.

So the contribution of this -- if you recall the number was point zero zero five as being the early fatalities per reactor year, this is a very small contribution coming from class one sequences.

MR. OKRENT: Again, what is the range on early fatalities where you show point five, for example?

MR. PRATT: I think the same answer --

MR. OKRENT: Is it a factor of 100?

MR. ACHARYA: See, I cannot answer that offhand.

All but we did not pull out --

The one, the last one, where you have mean conditional fatality is point five, that could not be very large even when one would look at the high end of the consequence peculiar -- But I doubt you could reach something like ten or so. That's my guess because you said, I have not looked at the probability distribution of the early fatalities for the early runs. From experience, I can say the one which gave a mean value, that is mean -- Over all the conditions, you could not get in the high end of the consequence spectrum more than ten or so.

On the other hand, when you have got two hundred, it could be something like a thousand.

MR. OKRENT: You mean a factor of a thousand?

MR. ACHARYA: No.

MR. OKRENT: Two hundred could go to one thousand?

MR. ACHARYA: Could go to a thousand. Not one thousand times.

MR. OKRENT: I'm a little suspicious of your upper limits, but let's let it go.

MR. ACHARYA: I have a slide to show the probability distribution of early fatality which, of course, has accident probability, but you can have some information on the -- which is independent of the probability.

(Slide 37 shown.)

MR. OKRENT: By the way, let me give you a request or a warning or whatever.

When you are all done giving the presentation, I would like to hear from you, your best judgment, on where everything you have told us might go wrong and the risk would be substantially larger than you are showing. Do you understand the question?

MR. PRATT: Yes.

MR. OKRENT: All right.

MR. PRATT: The next two slides really are just the arithmetic that you said you have already done in your head. Let me just mention the upper bound calculation. What I did was took the core melt frequency and assume that we had every time a class one sequence occurred, a failure. I don't want to dwell on it, but you can go to the final table. I think this answers your question. This is about as bad as I'm prepared to go.

(Slide 38 shown.)

MR. PRATT: In terms of uncertainty associated --

MR. OKRENT: My question wasn't related to only the uncertainties in release or so forth. It was a general question.

MR. PRATT: Well, I won't address that general question. I don't think I should. Parts of the other -- that go into that question were reviewed by other people.

I can tell you as far as I would go and take it with them as far as they would go.

MR. OKRENT: But my question isn't how large can the number get, because --

MR. PRATT: The point is this number is not that large.

MR. OKRENT: Well --

MR. PRATT: The point of the last meeting was that when Frank Coffman put that number up somebody in the committee said, "Those numbers are small." How robust are they? I'll tell you that I wouldn't dare multiply them by more than four from my perspective. You can get other people to multiply by higher numbers, fine. But that's it.

My best estimate to too close to the upper bound, and as I improve my methods, I'm going to come away. I'll tell you how far I can come away and give you indications in which area.

MR. OKRENT: Go ahead.

MR. PRATT: Well, I think that's the point. I've taken for the class one sequences, which is virtually the core melt frequency, the limit, and put it equal to steam explosion release. And if I look at increase for class one early fatalities, it goes from small number to a very large number.

But again as we are being dominated in early fatalities by seismic event, we have virtually no evacuation. The overall increase in fatalities is not large. The same with person/REM. This is dominating 700 out of thousand. I put it up high as I can go. Again, that's the factor. So really I feel very comfortable and this is the point at which you would put these numbers to use.

The numbers are useful in going into the hearing board on a final environmental statement and saying that we are high, but we don't see we can go up pretty much higher in this particular area. Useful in that regard. And the numbers are relatively small and acceptable, and that's a very useful point.

What you would do with these numbers is quite another matter, because I know I have built-in conservatism. And that's a hard question to answer.

That's all I have in class one. If you have no more questions I can move into class four.

Again, we will have the same general run-through.

(Slide 39 shown.)

MR. PRATT: When I gave this viewgraph, certain people didn't receive it too well, because I said that containment response calculations are rather straightforward. Here, if you recall, we are dumping decay heat into a suppression pool and boiling water so my feeling that this wasn't very exciting from somebody calculating containment response. Oakridge, of course -- look at various stratification in the pool and think that is important.

But I think most of the uncertainty for this sequence is really related to the systems to hydralics and neutronics, what level of power and so. It is a bad release. I mean release fractions for this calculation were dreadfully high. We can't go any higher than that.

The containment fails initially, core melts into a failed containment building. The pool is saturated, and we don't take any credit for pool scrubbing.

(Slide 40 shown.)

MR. PRATT: The next curve is really taken again straight from 3028 and shows a comparison between the Limerick and the BNL analysis and we are predicting within less than 50 minutes than the containment building would have failed as a result of normal pressurization and melting down into a failed containment building.

MR. DIEDERICH: What power level did you assume, 30 percent?

MR. PRATT: Yes.

MR. DIEDERICH: I noticed in the LGS PRA, they assumed 30 percent for the portion of core that is covered, and then decay heat for that portion that is uncovered. I did not understand how they could establish a level.

Did you look at that and see if there was a significant difference in the amount of power arriving at the suppression pool?

MR. PRATT: It wasn't a large factor.

MR. DAEBELER: Thank you.

MR. PRATT: In terms of containment event three, again I don't think I want to spend time going through that one because this is a situation where we are failing a containment building first. Really the only question that one would ask is the distribution of failure location, whether or not it is in the drywell, wetwell or the wetwell below the water line.

Again this is important for the analysis that was done by PICO because they took a decontamination factor of ten. If the failure location allowed the downcomers and the quenchers to be submerged so they got significantly lower source terms than we did.

In our calculations we didn't take credit for source terms for all of these failure modes are very high.

(Slide 41 shown.)

MR. PRATT: In this particular calculation, this really shows the paths taken by the station products, if we assume that the failure location is below the pool level and drains. So the melt release would go down the pipes and directly out and the vaporization down the downcomers. Again without pool scrubbing.

I don't think I really need to get into the tables. I think I can pass through that one rather quickly with you. Just get to the bottom line and impact on risk.

(Slide 42 shown.)

MR. PRATT: This goes back to the original tables that I showed you, the source terms calculations -- and we can see them from your handout -- are extremely conservative. Very high. There is obviously a great deal of potential for lower source terms, if we go to some of the newer methods with saturated pool scrubbing. You would see them go down for all failure modes except the one where we drain the pool. Of course, we didn't take credit --

Again the class four source terms really only contributed 12 percent to early fatalities. I think 88 percent is coming from side events which are very severe really with no evacuation. So I really don't see too much sensitivity. I cannot go up. If I go down, I'm only going to take 12 percent away.

Again on person/REM that is a small contribution, so this is basically my conclusion.

I think before I go into the next part which was the new methods you were thinking about -- changing the order.

MR. ACHARYA: Consequence analysis, that might help understanding as what went on.

MR. ROSENTHAL: While Mr. Acharya is getting prepared, it is our intent that Mr. Pratt address the remainder of his presentation on newer methods following Mr. Acharya.

(Slide 43 shown.)

MR. ACHARYA: This cartoon shows the type of emergency response that in the calculations. But those accidents which were initiated by causes other than earthquakes emergency response modes and that's depicted here in this diagram and this one here.

The assumption was that within ten miles of the -- which is the exposure emergency, that would be evacuation out ten miles from all those areas that will come under the -- outside of ten-mile zone there will be some form of emergency response, depending upon whether the dose level will be high there or not.

For the evacuation parameters, we did not have the site specific information available to us, but that was a preliminary study done on behalf of the applicant by -- which did not take into account the various emergency notification systems that would be in place and that would not give us information as to what could be the values of -- value of one of parameters that would go to the assumption, namely, delay time before evacuation.

This delay time before evacuation should incorporate how people are notified to take emergency action in the first place.

So the course we took to get some numbers for the delay time before evacuation was by looking at what was done for Indian Point site, which is another site, and that was some good evacuation time estimate study that were available to us which we had used in the Indian Point hearing.

The reason we used the Indian Point study that the -- the site may not be too different in the way of population density and in the way of people are responding for the evacuation.

So we picked up a two-hour delay time before evacuation. Ten minutes will be the time that would be -- That following the warning sounded by the reactor operator, people in charge of the decision, decide upon what to do in the way of emergency response, take about ten minutes to come to the conclusion what to do about it.

Given the decision that the people be evacuated, the notification system would notify most of the people in the ten mile in a matter of 15 minutes because that is part of the emergency plan of the site we are shooting be for in the guidelines provided.

So on top of receiving the notification, people will take some amount of time to prepare before they will be the evacuation routes by automobiles. That's called preparation time. And again the study for Indian Point was some assistance. And we picked up a people time of 90 minutes, people preparation time. All of these three components came to about two hours.

That's what was done in the DS APS.

Now, we happened to see -- Then came the applicant's analysis study that came in May and the results of which we did not use in the -- that study is -- otherwise, a review of that will be done by the emergency plan branch.

We don't have much involvement, but let me tell you the preliminary results -- my preliminary reading of that is that the numbers that is established for the delay before evacuation is not too different from what we assumed in the DS APS.

For instance the -- these things point out that the delay before evacuation will not be substantially more than two hours and of that two hours the people's preparation time, the best value is about 90 minutes. That's a spread -- the people's preparation 30 minutes to 150 minutes and average is 90 minutes like we assumed. And this average tends to cover most of the people's preparation time.

MR. OKRENT: Could you give me just an order of magnitude PL, or what the effect would be if there were no evacuation.

MR. ACHARYA: I have got that assessment, so if you just -- that is coming generally here.

My second one, alternate, is that evacuation would not take place. Then, as one of my slides will show, perhaps the early fatality number would go up by a factor of up to four, and so from the accidents initiated by causes other than earthquakes, because perceivable earthquakes will be different emergency responses. And the site conditions -- the last one here which is the most pessimistic one that was used for the earthquake situation, that drives the early fatality number very much and when the probability, that controls the risk probability.

So what is the sensitivity of early fatality to this different degrees of response, I have that in a few minutes.

MR. OKRENT: By the way, we should finish the NRC presentation by around noon, but no later. Keep in mind what you want to present.

MR. ACHARYA: The next component was of the evacuation parameters, the speed for evacuation. The speed before evacuation, that is used for the consequence analysis purpose. It is kind of a gross type of estimate, and that is normally obtained by assuming how long the people will take to empty a ten-mile zone and so the time taken by -- ten-mile radius divided by the time to empty the zone, that gives roughly the effective radial evacuation speed.

So the earlier study by NUS -- was not very good as far as the delay time estimates was concerned because it did not take into consideration the notification system that the component, the other component of time, namely the travel time, okay, that parameter we took from there because the study did look to the system, and few other elements that go to that kind of calculations though. So that was not reviewed.

We still said that in the absence of anything else, let us assume -- well, just for the timing take that number and do the sensitivity analysis.

On the basis of that we determined that the speed of evacuation will be about two point five miles per hour. Now comes the study, that points out that there are two similar conclusions, that is, the total evacuation time according to this is not larger than six hours, of which two hours is delay time. So four hours is left and that gives us two point five miles per hour.

So whatever we assumed, more or less it is kind of consistent with the new study. But I said the new study has not been thoroughly reviewed.

Outside of the ten-mile zone, we assume that there will be some extension of speed to ten miles, and people from very highly contaminated grounds, such as the projected grounds over the next seven days, it won't matter -- because that's about the level (Unintelligible) -- And then study some sensitivity. We assumed this one as an alternative to that is where we do not assume the evacuation -- suppose it could not take place because of bad weather conditions (Unintelligible) and too late in the decision rather the people will be left where they are and they will not (Unintelligible) --

Here the assumption was that all people will be relocated, however, six hours after the plume has left the air. There is all those areas that are contaminated by the plume. The reason we took the six hours here is that six hours also happens to be the time of evacuation in case the evacuation would have taken place.

Failing that -- (Unintelligible). At least this would be done as a minimum. So the time to relocate should not be much higher than six hours after the plume has passed. And beyond the ten miles, the same type of location from the hot spots was assumed.

Now from those accidents for which the site was very badly damaged by a seismic event, we assumed that none of these modes may be operative. And others, we assume that the situation will be quite adversely affected by the seismic situation and people might be quite confused and at a lost and --

Later on, when people have taken stock of agencies, at least people in very highly contaminated areas will be relocated elsewhere, and we assume about 24 hours later that will happen.

So these are the emergency assumptions, and using these assumptions the next page that you have in the handout gives the parameters how the evacuation was treated, all of this kind of stuff.

(Slide 44 shown.)

MR. ACHARYA: Here you get the complete list of the source numbers, seven release categories that were identified by BNL.

(Slide 45 shown.)

MR. ACHARYA: Of these categories, we threw away some of them because they had low probability and we thought they will not (Unintelligible). Now here in this table you will see that the probabilities with triple stars here (Unintelligible), and we made a preliminary comparison as to how much and we found that they are (Unintelligible) -- they don't (Unintelligible) -- so we discarded them from any further consequence analysis, so eventually we landed with table with risk categories for the consequence analysis (Unintelligible).

(Slide 46 shown.)

MR. ACHARYA: You have already seen this perhaps -- (Unintelligible). Each of these accidents, they were evaluated under three different emergency response modes. The ones that you see here, blanks, either -- they are not evaluated under the assumption of very pessimistic off-site emergency conditions for the reason that the probability was less than ten over minus nine, or else that particular accident sequence was not initiated by seismic -- (Unintelligible).

(Slide 47 shown.)

MR. ACHARYA: When this conditional mean values of the consequences were combined with the probabilities of the accidents, only those ones will have the (Unintelligible) -- probability distribution included in the ABS, and one is an example of that here.

Now, this is early fatality. It has three components, the (Unintelligible) -- delay relocation because that was initiated by seismic event, and the one that is with circle, that was from other than seismic event, and then finally the one that's (Unintelligible) -- the sum of the two.

Now, from here you can see the -- of the consequence magnitude, something -- something like this is -- (Unintelligible). Non-seismic initiated accidents, and the other ones, it is something like 20 thousand.

Now, we evaluated our early fatality under two assumptions. Now, this is one in which we have the supporting medical treatment available for the people who are so exposed.

(Slide 48 shown.)

MR. ACHARYA: The next slide shows again the early fatality estimates assuming the supporting medical treatment was -- assuming that the supporting medical treatment was not provided. Now, so here is an indication as to what were the circumstances that drive the early fatalities up.

MR. OKRENT: Is that cross-over real?

MR. ACHARYA: Well --

MR. OKRENT: At the low end?

MR. ACHARYA: Yes. The reason is there is -- at the low end, it is driven by the probability of the accidents, and you know -- you will see from the one of the tables that I showed before, the probability for the seismic events is very small -- quite small -- compared to the internal events.

So that is what is showing here. The internal higher than the external ones.

MR. EBERSOLE: Could you maybe clarify for me what is the nature of the doses that permit improvement by medical treatment? Give me some characterization.

MR. ACHARYA: That is a very difficult thing to say. Actually anybody who has a dose exposure above the supposed threshold, assumed threshold for fatality should be a candidate for the supporting medical treatment.

MR. EBERSOLE: I was getting around to what sort of relative contributions are you talking about, direct radiation, ingested dose, which is subject to some --

MR. ACHARYA: The medical treatment that we are talking about is for exposure to the bone marrow. Now, the previous high exposure to the bone marrow will vary badly.

MR. EBERSOLE: By direct radiation.

MR. ACHARYA: By direct radiation from the cloud and from the ground contamination and inhalation from the ground will provide a very small contribution to this. So mostly direct radiation, yes.

(Slide 49 shown.)

MR. ACHARYA: In the APS there are all kinds of -- (Unintelligible). I did not intend to take time in showing all of them, so what we did was that we took all the CCDS that were shown in the APS and we took the top curve, like I had shown here for each (Unintelligible) -- the took total CCDS and simply read off the value (Unintelligible) -- and this page shows that. So you see, as you go down the probability level that's the first column indicates the consequence magnitude.

And the reason that the consequence magnitude goes higher here, that's because of the contribution of the metallurgical effects.

(Slide 50 shown.)

MR. ACHARYA: This you had already seen. This is nothing but the area underneath the CCDS. This is where the risk is and you have two columns here that is 50 miles.

(Slide 51 shown.)

MR. ACHARYA: Just to give you a flavor as to what contributions from the seismic and non-seismic to the mean, this one is showing that, that is, as you see we took the early fatality, the last column here is dominated by the seismic and this dominance is here also, but when you come to the (Unintelligible) -- and for the other ones it is -- the risk is dominated by internal events.

MR. ROSENTHAL: On this slide right here, if you wouldn't mind putting it back up, there was the issue of the seismic contribution to total health effects, and I think the point there is that with the -- from severe earthquakes late relocation dominating it says that seismic events beyond point four G Are the dominant contribution to early fatalities, and that seismic events --

Can you draw a conclusion about seismic events less than point four G relative to the total risk?

MR. ACHARYA: The seismic contribution to the health effects is very small. That will not show up in comparison to the internal events.

MR. ROSENTHAL: But the point then is that the plant's design basis is point one five G and up to about point four G that seismic is not a dominant contributor to early fatalities as modeled. And it is only for those seismic events which are beyond point four G for which you inhibit evacuation that seismic then becomes the dominant contributor to early fatalities?

MR. DIEDERICH: On that conclusion if you assumed no evacuation for seismic events less than point four G, would you have the same conclusion? What I'm concerned about is you are going to lose power in the region for events much less than point four G and my concern is you may not be able to have effective evaluation without any power.

MR. ACHARYA: You are going to give the differential treatment to all those accidents that were initiated by low seismic and the internal ones? Okay? Now, we have not done that way. But, however, it is very easy to do that from the big table of conditional mean values that I showed. Now mix them in different way.

This perhaps tells me -- this column here is evaluated without the assumption of evacuation rather with the (Unintelligible) -- people were hussled away from the ten mile six months after the plume passed.

Now, the numbers (Unintelligible) -- that's the factor by which that showed in the previous table will go up.

If you did not evacuate, or if you did this -- now, however, if you come to the total, the total is essentially unchanged. It was five times eight minus three before, and it is six times minus three, because this column is the same. This was the big one over this one.

MR. OKRENT: May I ask a general question: In view of the very low risks that the staff is showing for early fatalities and for societal risks, the environmental statements, and you are telling us here today, Mr. Pratt's argument that you can't see really how they can get much larger, why didn't the staff recommend doing away with the emergency preparedness?

MR. ACHARYA: We meet some situations. For instance, that are situations like seismic, I cannot do much, but if the accident is not from the seismic cause, but from other causes, the risk may be small, but the conditions upon the accident -- you talk about the number there (Unintelligible).

MR. OKRENT: You are giving me mean values of the risks to look at, in general, although I agree you showed a couple of distributions. And those mean values, in fact, are small and you are saying they are small. And I agree that those are small numbers. And so again, I'm asking what is the justification for emergency preparedness at all when you calculate such a low risk from the plant?

MR. ACHARYA: Well, from the risk of the -- well, I will talk about this at some later time, since this is the comparison I would like to show, that is, for example, this is the population exposure.

MR. OKRENT: I've seen those already.

MR. ACHARYA: You haven't seen this maybe.

MR. OKRENT: I looked ahead, but I knew it before I looked.

MR. ACHARYA: That concludes my presentation.

MR. OKRENT: Maybe someone else on the staff will answer my question.

MR. ROSENTHAL: I think we do intend to publish NUREG-0956 and after publication of NUREG-0956 -- in the APS review source terms, we intend to prepare a sequence of white papers, of second key papers to the commission and so sometime next spring in connection with those, both NUREG-0956 and second key papers on the potential uses of that information would be the planned time to come to the ACRS.

MR. OKRENT: You mean with the question about the need for emergency preparedness or what?

MR. ROSENTHAL: That would be the subject, potential subject of one of the text of white papers that we have identified that we will have to write.

MR. OKRENT: I'll be interested in seeing it.

MR. DIEDERICH: A brief question. Can you put that last slide on again?

I've been concerned recently that use of the 50-mile radius might be misleading and the reason for that is that we are now showing lower source terms for these accidents than were originally calculated in one fourteen hundred for the category one and two accidents. What that should mean is that as you go out further from the site the doses become less and less very quickly. So that by using 50 miles you are picking up a lot of population that would not be exposed and you are making the comparison show --

In other words, you are not comparing the people exposed to the accident if you go out to 50 miles. In other words, what would the numbers look like if you went, say, to ten miles? Would you get a more significant ratio of consequences to all others?

MR. ACHARYA: That we have not calculated because, as you know, there are certain things that are called proposed safety goals where they have a 50 mile number. However, there are lots of tables in the APS where the societal risk is shown as a function of distance, that is, every ring -- like a heavy band of radius -- all the ranges they are tabulated in the APS. So if at any point in time one stopped at any particular (Unintelligible) -- Ten, or 15 miles but also the -- (Unintelligible).

MR. DIEDERICH: You see what I am saying. 700 person/REMS may be out to ten miles, and after ten miles that number stays the same, whereas the eight hundred thousand goes up very quickly as the radius goes up and you may not have a valid comparison. That's my only point.

MR. ACHARYA: I suspect that that might happen provided the source number come down and then what happens when one has to strike at that point of time an appropriate distance for --

MR. DIEDERICH: Then you have to compare the exposed population.

MR. ACHARYA: This 700 is not picked up much closer to the 50 miles because you find if we go to the (Unintelligible) -- It could be something like 1300 or 1400 person/REMS so we have rounded (Unintelligible) -- significant figure. We show one thousand but actually it is about 1300 or so. So substantial amounts in this case also comes from outside 50 miles so the comparison may not be (Unintelligible) --

MR. OKRENT: We better move along because we are going to have to finish at 12 o'clock with the staff presentation.

MR. ROSENTHAL: While Trevor is getting ready, I would like to make a point. If you look at the class four releases that were modeled in the FES, one is numerical values of the release fractions, one is hard pressed to believe intersystem LOCA could give you larger release fractions and those class four events would have a higher -- have a finite probability, and the intersystem LOCA may well have a far lower probability, so I don't think it would necessarily change the total risk profile.

MR. OKRENT: I was only trying to understand where it fit in the classification picture.

By the way, Mr. Rosenthal, I earlier asked Mr. Pratt a question. I'll address the same one to you. You can think about it while he is talking.

If there were to be some serious -- I'll call it -- omission or flaw or oversight in this analysis such that the real risks were 100 times or more greater, what would your candidates, your leading candidates for trouble spots be? I'll be interested in hearing.

MR. PRATT: My response was specifically to those areas that I was reviewing.

MR. OKRENT: Well, he has got an overall responsibility here, so I'm giving him the full picture.

MR. PRATT: I know it.

I apologize in a way for the presentation. It is going to be rather qualitative. We don't have numbers. BCL, our present -- well, probably not at present, on Monday -- are calculating new systems for certain --

MR. OKRENT: They don't work weekends.

MR. PRATT: We certainly do at Brookhaven, but I'm not sure what they did. All night and so on.

They are doing the calculations for selected sequences, and the idea is to help in transferring the technology over to Brookhaven. So we are familiar with the trends but we don't have them for Mark II specific configurations and I can tell you trends. I can't give you real numbers.

Basically the -- they are being addressed in terms of these various activities under the accidents source term program. Specifically BMI 2104, and I believe that the calculations done for Limerick will form an additional volume in this type of reporting procedure.

Also we are all involved, just about everybody in the country, in terms of containment loads working group and we have had many meetings to try to define better containment loads to run through our analysis, which is then fed into the containment performance working group.

At Brookhaven we have the dual role of taking the loads from the containment performance working group and feeding them into the performance working group who then try to estimate leakage paths and so on.

So there is a good deal of interrelations which is attempting to define -- in fact, this document was used by us extensively in our GESSAR review because earlier on in our GESSAR review we did not have the applicable codes, so we had to work with MARCH, INCOR and tell it what to do, and this was the basis of doing some of that telling.

This is a rather large set of viewgraphs.

(Slide 52 shown.)

MR. PRATT: It will give you a flavor as to how things have changed in terms of the various stages. We have got WASH-1400 here and the newer methods and I've really gone through this and explained where we differed. Core leak contamination has certainly calculated a number rather different than one. But it is different and of course it does calculate increased agglomeration of settling of aerosols relative to what corral would have calculated.

Of course, GESSAR would give you a sequence dependent release of the fission products as opposed to just the specified values. And of course Vanessa would calculate (Inaudible).

MR. OKRENT: Spark is developed in which laboratory?

MR. ROSENTHAL: Battelle Northwest with Mr. Posma.

MR. PRATT: We have gone through Brookhaven several versions of most of these codes. I forget what number we are up to.

So principally that's the system of codes that we are working with.

(Slide 53 shown.)

MR. PRATT: As I see them the impact of the new methods.

In that the relation, we are talking about the timing and the chemical probabilities primary system retention and remission. We did not calculate retention, therefore, if there is any disadvantages in terms of remission. But as we move into the new methods, we are going to have to look at that fairly carefully and make sure particularly for some these boiler sequences at a later time.

Generally what we are talking about is primary system retention and we have some values. We kind of agonized over whether to go through tables with you as to what the primary system retention might be. Probably the time to share that with you would be during GESSAR when we could go through and show you some of the calculations we have there.

MR. OKRENT: Fine.

MR. PRATT: This is Vanessa and different from the calculations that we did in WASH-1400. As I mentioned, I think there is a potential there for more releases than one would have calculated.

In terms of fission product release, I mention now it would tend to give you increased agglomeration and settle -- of course, suppression pool would, if we calculate for saturated pools, the decontamination factor would again reduce the source terms.

Those are the general trends of new methods. I don't know quite how much more detail to go into on that.

MR. OKRENT: Objectives, I'm not so interested in as findings. Do you have some major findings you can tell me about?

MR. PRATT: I don't know how major they are, but they have findings. I'll pass over the objectives and the approach.

The definition -- what the containment load tried to do was to look and develop standard problems for the various six containment designs that we had identified: BWR/1, two and three, large drive, setup and ice condensor, so there are six different types, and for the Mark I, this is actually Mark I and II, the focus really was on the pressure temperature response during core concrete interactions. We thought that was an important thing to look at. Particularly concerned with the sort of confined space in the drywell.

So the issue really to be addressed was the pressure temperature response and the mode of containment failure. In terms of working group were to take this pressure temperature histories and try to decide how the containment building would perform. So we identified a standard problem and looked at sensitivity studies in terms of the initial conditions that the core debris would be in. So we looked at various temperatures, various masses of steel and so on in the mix and looked at the concrete, different concrete types, and assessed impact.

(Slide 54 shown.)

MR. PRATT: In terms of the calculation methods, this is really a presentation that I gave at the joint NRC -- who was involved in the calculations -- to give a flavor. BNL was involved and we worked with MARCH one point one B, which is a version of MARCH developed by Oakridge that has channel box models and so on.

So it is not really that much different in terms of in-vessel interactions. It is more improved in-vessel progression type thing.

We also can look at MARCH one point one and MARCH two and we replaced the intercept routine, which is a more improved concrete interaction model. I believe there is a presentation in the light water meeting next week (Inaudible).

In a slightly different way there is linking and there is difference in the results as a result of the way it was mixed BCL. MARCH two with what they called modified intercept. It is not the same intercept in the Mark I point one (Inaudible). It is somewhat modified and we spent a lot of time identifying differences in the codes.

(Slide 55 shown).

MR. OKRENT: You have got about three minutes.

(Slide 56 shown.)

MR. PRATT: Just to prove it was a cooperative effort.

This is really the differences in the results. Mark II, the spread wasn't that great. This is trying to make the same initial conditions, just differences in the modeling. One can see this is pressure against time after the point of vessel failure. And this is sort of spread in modeling and principally due to the assumptions regarding up heat transfer and degassing walls and so on.

MR. POWERS: The higher the curve, the more degassing. The higher consequence to the cases where degases from the concrete was occurring, is there any structural consideration (Inaudible).

MR. PRATT: No.

MR. POWERS: I'm thinking in particular of the vessel support structure failing and causing the vessel to drop.

MR. PRATT: No. The only consideration is the point where we penetrate 70 centimetes of floor up until that point --

MR. POWERS: None of the overhead concrete structural consequences of overheated and degassed --

MR. PRATT: Yes. Degassing certainly, but not degradation.

MR. POWERS: Not structural consequences.

MR. PRATT: No. In fact this model, Sandia model, does the degassing of the concrete, but it does not consider the structural implications of that degassing.

MR. POWERS: I have heard that some of the degassing models do so with -- (Inaudible).

MR. PRATT: We were a little --

MR. EBERSOLE: How do you level out at the two hundred pounds?

MR. PRATT: In our particular case, this was a calculation that we performed at Brookhaven. What you find is we -- as I say -- you saw three different calculations that we did there at Brookhaven. This is one where we artifically turned off the heat transmitter, because we thought the aerosol would blanket the upward radiation and spread it out over quite a large area.

So what happened is at later times the thing froze and you just slowed down the penetration rate. That's really what it was.

And if you look at other models which transfer more heat up then you will find that the pressure temperature goes up higher. So this is a lower bound calculation of pressurization rates. We did other calculations where we did transfer heat up.

So the aim was to try to represent a spectrum of possible responses and models so you could get a flavor for that. You will find lower pressurization with much higher temperatures, again depending on the model that one would use.

So this is one area where your assumptions regarding upward and downward heat transfer and how the core spreads is quite sensitive.

MR. OKRENT: You have a minute.

MR. PRATT: You can look at the drafts in there, and just quickly the observations of the group --

(Slide 57 shown.)

MR. PRATT: This is something very new. Literally this week I got the new leakage areas that would suggest that this leakage really shouldn't be sealed. It is a head, would prevent over pressurization and gross failure of the -- so I think -- Let me see.

(Slide 58 shown.)

MR. PRATT: I will define what I mean by the first and second category. The first category are failure modes in which the containment building is held together, and you were bumping it up for a period of time. In our estimate at Limerick, I think that then you would tend to get lower source terms and I've gone over the reasons why I think you would.

We have got -- actually, I didn't put it under here, but the primary system retention is one method. Containment ESF, pool scrubbing and so.

All of these would tend to make these categories somewhat lower than we predicted.

The second category I still believe are uncertain. There we are talking about situations in which the containment building has failed or bypassed. These are the ones with the seismic events and there you are going to really be primarily relying on primary system retention. But there is a question of re-emmision and just how much of it is retained is quite an open question. So I think there is a possibility for reducing source terms but it is rather more difficult. So I'm tending to think of these in two distinct categories.

MR. OKRENT: Thank you.

Do you want to try to respond to my question, Mr. Rosenthal, as to where, if we are missing something, it might most likely be?

MR. ROSENTHAL: I have two areas, one mitigation, one prevention, both of which would affect the frequency of release rather than the magnitude of the release. And I say that in terms of magnitude because I think that, as Mr. Pratt has tried to show, that we have used very large magnitude releases, as you can obtain by studying the release -- tables of release fractions on the mitigation side.

We have assumed 1 percent of the time the plant would be de-inerted with a failure due to hydrogen 1 percent of the time. That leads to early containment failure, which we qualitatively know has got to be worse than late containment failure when there is time to either recover or do something or run away. So the very pragmatic issue would be to assure that in fact the containment is meant to be run inerted over the life of the plant.

And I think that here we have 1 percent not inerted and in comparison a steam explosion at ten to the minus four and much more concerned about operation of the plant at a few percent inerted versus 1 percent inerted and a bigger difference. And if I take my minus four and make that ten to the minus two --

It is a somewhat pragmatic issue but that behavior bothers me.

On the front end side, that is not my area of responsibilities but we have some people here.

I'm bothered that we have a plant for diesel and station blackout is still the dominant contributor to core melt. If you have an adversion to core melt, rather than an adversion to relative risk and you have to ask why does that station blackout still have that large contributor and I think you have to -- is it real or is it an artifact of the PRA and I haven't heard that area pursued.

MR. OKRENT: Thank you. Your comments are interesting.

MR. DAEBELER: I was just going to say I suspect it is because of common cause failures if you use, for example, a beta factor method you gain very little going beyond two redundant trains or pieces of equipment. If you were to go to a different type of emergency electrical generation you would gain, I think, some -- that doesn't say that the model is valid but that's what PRA uses and that's the consequence of it.

MR. POWERS: On the category labeled "steam explosion failure" I had thought I was under the understanding that was not a rule, steam explosion, but rather just an over-pressurization in the analysis we were presented earlier.

MR. ROSENTHAL: I think the point we have associated ten to the minus four conditional probability of a horrendous release fraction given core melt, no matter how you got there.

MR. POWERS: Well, it seemed to me there was -- analyses were directed to assume that you would not get melt down into that water in the plant promptly after vessel failure. And that fell into a steam explosion type of analysis. I mean, that was an assumption upon which the analysis was carried out.

MR. ROSENTHAL: Let me know within the containment load working group effort, we did ask Mr. Cordini (Phonetic spelling) to look at the feasibility of a coherent movement of corium into the pool by the downcomers and the potential for rapid loads to include steam explosions. And I am under the impression, and perhaps you can help me with this one, that his conclusion was that that did not seem likely.

MR. POWERS: Perhaps for steam explosion, but I cannot remember what he had to say but just over-pressurizing --

MR. ROSENTHAL: We worried over this. There is a question of communication in the water region below the diaphragm floor, and as Trevor Pratt said earlier, the upper bound calculation risk estimate in 30 -- NUREG CR 30 20 A is when you don't have communication and hence you can get steam over-pressurization failure -- so we worried about the thermohydraulics with respect to rapid steam reduction below the diaphragm floor.

We worried with the best talent we had about steam explosions in that same region and so when they asked what am I concerned with, I brought up a very much more pragmatic issue and that was the operation of the plant.

MR. BOYER: I think you pointed to two areas. And I would say that one of the benefits of the effort of the PRA and the examination into the potential modes of care failure and probabilities and whatnot has been to provide to the operating force and the engineering force of the utilities the areas which are more important than others in preventing the release of fission products to effect the health and safety of the public and provided areas in which we can increase our attention and educate the operators, has led to the development of trip procedures, has led to the development of suppression pool venting procedures, or containment venting procedures, and last ditch cooling methods.

We have gone down the line of emergency preparedness for events happening in the plant to a much greater degree than we had done before and therefore I think we have certainly further decreased the probability of these events reaching a serious magnitude. So I think that's one of the things that's come out and will continue to benefit from the studies of this type.

MR. OKRENT: I'm going to recess for lunch in a moment and we are using lunch time -- I'll warn you because I want to be back at one, whenever we leave -- I'm going to pose a question to Mr. Pratt and anyone else who is so inclined, that this in fact Mr. Trifunac raised to me, namely: Is there any way in which an aftershock occurring one to 60 minutes or whatever after the original earthquake which presumably was severe enough to start you down the road to a core melt, is there any way in which an aftershock could perturb what you have been analyzing significantly and change any of your thinking?

I'll just leave it as a question to mull over during lunch and we should reconvene at one o'clock.

(Noon recess from 12:00 P.M. to 1:00 P.M.)

MR. OKRENT: The meeting will reconvene.

MR. BOYER: In the beginning, this afternoon I think it might be appropriate for the nuclear industry to recognize the death of Sol Levine for a moment. This was

MR. OKRENT: The meeting will reconvene.

MR. BOYER: In the beginning, this afternoon I think it might be appropriate for the nuclear industry to recognize the death of Saul Levine for a moment. This was a sad event for the nuclear industry in total. Saul has worked with the Atomic Energy Commission, with the Nuclear Regulatory Commission and as a consultant for -- to the industry, and he has been associated with the probabilistic analysis work and WASH-1400 for a number of years. I think he was a good friend and to all of us and thought it might just be appropriate to recognize his passing.

We will begin then this afternoon with Gene Hughes, who at the time of the PRA or Limerick was being conducted was with SAI, and he was the responsible person in charge of the probabilistic analysis work. He will be followed by Robert Henry, who did a lot of the analysis work for it.

MR. OKRENT: I do intend to allocate 90 minutes to this topic as the agenda shows. So if people are asking you too many questions, brush them away. Brush away the bad questions, not the good questions.

MR. BOYER: I'll pass that to Gene and see that he gets done in 90 minutes.

MR. HUGHES: I wish I had the luxury of deciding which were the good and which were the bad.

I m going to skip over several of the slides in the handout, because I think the material has been covered.

The presentation that I have this afternoon addresses itself to the inplant physics analysis. It is in response to the request made at the last meeting. I'm going to talk a little bit about the general approach, a little bit about the methods, perhaps less than I would have if we had gone first.

So then I have taken class one and class four. I have a number of charts of walk through of those events what is transpiring. Then I want to include that with the fission product source term.

I want to point out that the analysis that I'm going to be describing was performed in 1980, using the code package available at the time, the RACAT package developed by EPRI. There have been a number phenomenological advancements since, and I will not describe those in great detail. Bob Henry will talk about some of those, but please feel free to ask.

(Slide 59 shown.)

MR. HUGHES: In order to move along let me skip through accident bending. You know there were six different types of accidents. The physical processes were models with the INCOR computer code. I think you are familiar with that code.

The containment structural evaluation I am going to spend a few moments on, because I think that may be, if not new information, certainly germane to what would come. The containment event tree I want to spend a few minutes on. Then division product transport will come really after the discussion of class one and class four.

(Slide 60 shown.)

MR. HUGHES: So without further delay let me put up a chart that we can describe the Mark II containment we are looking at. First of all, let me point out that in comparison to the chart that you saw this morning, the drywell head was described and that is shown in the upper region here.

The analysis of containment capability was one of the early activities that had to be undertaken. We were in touch with Bechtel Power Corporation. They performed the analysis using finite element methods. They looked at analysis two ways: One was a rather simplistic model. The other was more complex.

They concluded that building could withstand between 120 and 160 p.s.i.g. as a lower limit; that is, one calculation set at least 120, the other set at least 160.

They pursued the analysis to determine where onset of yield would occur as their failure criteria, but not necessarily inconsistent with other containment studies that have been done.

When we faced the reality that we had to select a single number, the number we selected was 140 p.s.i.g. which represents the best judgment of those involved that that was an appropriate limit below which over pressure failure probably wouldn't occur with a high confidence.

The other criterion involved the diaphragm core. They also had undergoing in the analysis at some point, as you will see shortly, corium in contact with the diaphragm floor, diaphragm floor penetration beginning to occur and weakning.

The 70 centimeter criterion penetration was based on the location of rebar in the floor. The floor is a structural member with the containment itself and the analyses in fact showed that minor growth associated with pressurization was contained by the floor.

So we weren't sure exactly what the mechanism might be, but we felt failure of the floor would be sufficient to terminate the analysis in time and assume the containment was ruptured.

The other phenomenon that we will talk about has already been alluded to. That involves the fact that there are drain holes here. These drain holes were recognized in the analysis as commented during the molten core phase of accident, when the core is outside the vessel. We did assume ten percent progressed into the suppression pool.

Again, I'll be going through this in detail momentarily with the class one and class four.

The other thing we looked at was the possibility of leak, as I commented this morning, we were unable to identify a weak point for the containment. We felt leak was possible. We concluded that a 50-50 split with leak was about the best that we could come up with. So we included the possibility of leak.

The next thing that came out of this study from Bechtel was a look at the overall deflection that might occur where the stresses were in an effort to define the location and type of failure that might occur. We were particularly interested because of fission product path that might exist with failures either in the drywell or the wetwell or failures very low that would drain the pool.

The Bechtel analysis pointed to a high stress point in the wetwell region about the mid height just above the water level and suggested it would progress upward rather than downward due to the way of the building is configured.

They also suggested that the drywell was not that far behind. So we concluded that the best thing to do was to put roughly 10 percent of the failure probability in the wetwell region and then split the remainder between the drywell and wetwell air space.

Recent conversations with Bechtel suggest that in fact the 10 percent for the wetwell region may be slightly high. The other split may be about right.

Certainly this is an area where we do not have detailed mechanistic calculations, but judgments are the best we can make.

With that chart fresh in mind, let me put up the containment event tree.

(Slide 61 shown.)

MR. GARCIA: Gene, I've got a question: You indicated a 10 percent split for the wetwell area with the remaining 90 percent split for the drywell area in the wetwell and the remainder of the drywell. That's not way the event tree is indicated. Would you can explain the difference.

MR. HUGHES: Let me put the event tree up.

The numbers that I was quoting a moment ago were approximate. The actual numbers shown here. If you follow through the containment event tree, let me point out -- I'll come back to class four -- there is a different number there. I was really speaking of class one.

If you follow through the containment event tree to the point of exactly where you are going to end up with failure -- excuse me. I've over here.

(Slide 62 shown.)

MR. HUGHES: Let me point up here.

We are coming through the containment event tree to hydrogen, to leak sufficient, to containment failure pressure and then get into the location. As you can see here we split the location 50-50 between the drywell and the wetwell. And then for the wetwell we split it 90-10. So this resulted in .45 and .05 and instead of the 50-50. But the numbers were approximate as shown.

MR. GARCIA: Right.

MR. HUGHES: Okay?

MR. GARCIA: Yes.

MR. HUGHES: Let me proceed through the containment event tree from left to right. And first point out the top path here there is a very small probability that if the core melt degredation accident occurs the containment failure would not occur.

That very, very small probability is probably one of the more strong conservatisms in the study and I think you will hear more about that shortly. Certainly a higher likelihood than we gave it that the containment would not fail for this type of event.

We assumed, however, that the containment would almost always fail. We looked at rapid overpressure in vessel and in containment. The number that had been used in WASH-1400 was ten to the minus two. We reduced it to ten to the minus three and split it between the wetwell and the pressure vessel.

The containment overpressure failure there may be conservative but it was a number we felt was about right at the time.

The hydrogen number of .01 is developed from the amount of time that the containment is deinerted. As indicated this morning 99 percent of time the containment is inerted. We assumed that in the 1 percent that it was not we would have hydrogen present. It would then burn and we would either receive a burning overpressure failure, relatively gentle sort of like a drywell failure, or we would have a failure rapid due to the detonation. And we assumed the ten to one split here.

This ten to one split was also judgmentally derived. It included factors associated with the steam that might be present during such an event, that it was our judgment that that was about the right split.

In the event that we move over to the possibility of containment leak, you will notice that we had two different numbers per leak. We often talk about the 50-50 split. That was for class one, two and three.

For class four, where we have a more rapid pressurization, the likelihood of leak sufficient to preclude the event was very small, four times ten to the minus four.

The numbers to the right then show the progression of such a leak through standby gas treatment system success or failure. And we did take credit for standby gas treatment system for those cases in which the leak would occur through the reactor building.

Then proceeding to containment overpressure, again almost all cases did lead to overpressure failure. Here we had the split between the drywell wetwell and then the subsequent split between the water volume and the air space.

Now, I'm going to skip over a number of charts that deal with the analysis process.

(Slide 63 shown.)

MR. HUGHES: And move to corral with the comment that the analysis portion and the charts that I'm skipping over largely indicate that the analysis was very similar to WASH-1400. However, it was enhanced by computer methods and codes that had been developed.

What's shown here is the corral portion of the analysis, where the results of the in core compartment flows, pressure, temperatures, et cetera, were used to analyze fission product transport and release.

Here you will notice the fission process or removal process, rather, and on the right the fission product sources that were involved in the various release paths.

The gap and melt release occurring in the vessel, then proceeding into the drywell and wetwell regions with natural deposition and some pool scrubbing pool.

Pool scrubbing, by the way, was only associated with those portions of the flow that went through the pool. The portions that bypassed the pool, et cetera, were assumed to be released directly.

I might comment that the natural deposition removal process did not include substantial removal in the reactor vessel played out and the like were not included in the upper vessel region. We assume that the material was driven to the pool and that may be a conservatism although I think you have heard some discussion of the pros and cons of that.

I guess I've got a couple more charts before going through class one.

(Slide 64 shown.)

MR. HUGHES: Just observing that the natural deposition played out and gravitational settling, there was gravity and deposition in the structures, there was none in the reactor pressure vessel. This was not a modern state of the art technology. It was an aerosol, not quite the way we would have it today.

The suppression pool scrubbing was only included where it was effective and where the flows went through the pool. Standby gas treatment system filtration was included for those cases that had leaks that were felt to be small enough to be handled and of course the molten fuel on the diaphragm floor at containment failure did contain some fission products.

The numbers shown here are the decontamination factors. I won't go into these in detail except to say that there is a sizeable body of opinion that they are quite conservative and could be much larger than numbers shown.

So let me go to class one.

(Slide 65 shown.)

MR. HUGHES: What I'm now going to do is with a couple charts walk through the major steps and then put up the time line. I will then move from the time line to some time slices as we proceed through the event and look at various flow paths, et cetera.

First of all, to restate class one, we are talking about TQUV-TQUX sort of sequence transient event occurs, scram occurs, event coolant makeup is assumed to fail. We proceed to steam through the relief valves.

(Slide 66 shown.)

MR. HUGHES: Since we have no coolant makeup to this particular case, as we steam through the relief values we remove inventory. We then proceed to release the gap material followed by the melt releases. These releases are scrubbed in the suppression pool because we are connected to the pool through the safety relief valve lines. This is not 100 percent scrubbing but those materials that are driven through by the pressure flows are scrubbed.

The vessel then fails the reactor vessel with the containment intact. We then have the corium moving to the diaphragm floor. We assume 10 percent of it is involved in the oxidation release. 90 percent of it sits on the floor and proceeds through a vaporization type release.

Part of the vaporization release is scrubbed in the pool and that's the portion associated with the pressures driving flow through the downcomers.

MR. POWERS: The 10 percent oxidation release, that's an assumption?

MR. HUGHES: Yes.

MR. POWERS: The holes in the diaphragm floor are like four inches in diameter?

MR. HUGHES: Four inch pipes and larger concrete, about ten inch diameter.

I think a discussion of the sensitivity to that assumption and some of the more recent thinking will be provided by Dr. Henry shortly.

(Slide 67 shown.)

MR. HUGHES: At this point we are building pressure and we are interacting with the diaphragm floor and the two things we are looking at for possible containment failure. Containment fails several hours after vessel failure. Again the time line will be up shortly.

For this particular sequence I'm going to walk through -- I'm looking primarily at the three gamma type release paths. The drywell, the wetwell above the pool and the wetwell below the pool. The suppression pool is subcooled for this event and we did include resuspension of scrubbed fission products at containment failure.

This was an addition of fission products release at containment failure equal to 15 percent of those that had been removed. We took credit for removal but when we terminated -- when we reached containment failure we assumed some flashing of the pool and 15 percent resuspension.

MR. POWERS: 15 percent is just an assumption?

MR. HUGHES: Yes. It was an assumption for this particular case. It may be slightly conservative. For this case you will see the temperatures are not that high.

(Slide 68 shown.)

MR. HUGHES: Here we have the time line for the event in terms of comparisons of this. With some of the Brookhaven calculations, they tended to get a melt a little later. Their time for the RPV attack is a little shorter and the result is their release is a little earlier. Times are roughly close. They are not dramatically different.

Here we are looking at the MSIV close reactor scram, core uncovery is at about half hour. Starter core melt is when we get the coolant level below about a third of the active core height.

Core slump is assumed to occur when we get 80 percent of the core melted. It slumps into the lower head, it is the lower portion of the RPV. Suppression pool temperature a little later is about 150 degrees F. The RPV bottom head failure occurs as we get core concrete interaction and at that point we assume the 10 percent oxidation release occurs.

We then proceed to build up pressure. We get to containment failure at about six and a half hours and then the resuspension occurs about then.

Let me start to walk through this event with about seven or eight charts.

(Slide 69 shown.)

MR. HUGHES: First let me point out on this one the numbers are a little tough to read so I'll try to repeat them.

The reactor drywell and wetwell are slightly pressurized at the beginning of the event. That's the way they normally operate. In this particular case we are looking at essentially the beginning of the event. The flows had not yet begun through the relief valve lines to the suppression pool. We have had a transient occur and we are now beginning the process.

If I move ahead in time to 50 minutes, we get to the point that having failed ejection we are begining to remove inventory. And as we begin to remove inventory we get the water level below the top of active fuel. We have not yet at this point caused core melting to occur. We have got our flow paths set up, however.

You will notice that the pressure in the upper portion of the reactor vessel has increased to the safety relief valve set point. The flows are proceeding through the safety relief valve lines.

By the way, this cartoon shows this as if it's right on the floor. It is actually about four feet above the floor.

At this point we are having flows, the steam is being quenched in the pool. Our temperature is 120 degrees Fahrenheit and the pool pressure is slightly increased at that point.

(Slide 70 shown.)

MR. HUGHES: Then we move forward in time to 1.3 hours. We get to the point that core melt initiation is beginning. Here we still have the same release path, vessel is still intact, we are driving the fission products that are released through the safety relief valve line into the suppression pool. The suppression pool is heating up. It is now at 147 degrees F. Pressure is increasing slightly again to 17 psi in the wetwell.

Then move forward again.

(Slide 71 shown.)

MR. HUGHES: We get to two-and-a-half hours. Here we are looking at core slump and vessel head attack. At this point the core material is in the lower portion of the reactor vessel. The degredation of the reactor vessel is beginning. As the steel is melted the molten steel rises above and the attack continues. The suppression pool has been heated to 153 degrees F. Our flow path is still the same.

MR. DAVIES: What are those two circular objects below the diaphragm floor?

MR. HUGHES: Those are drain tanks associated with the drain lines.

MR. DAVIES: I thought they might be crystal balls.

MR. HUGHES: I thought they looked like something else.

MR. BOYER: About 1,000 gallon tank, I would say. Each of the floor drain and equipment drain.

(Slide 72 shown.)

MR. HUGHES: The next phase or the next step of particular interest is the one where we have the pressure vessel failure occur in the PRA. This was an area where we did take advantage of some work done by Bob Henry. One of the questions that was asked was did we do any separate effects or any analyses outside of the code package to convince ourselves that it made sense or it was about right. This was one particular area where the code package has in it an artifice that assumes instantaneous time zero vessel rupture that results in fairly large pressure increases which we felt were unrealistic.

So we talked to Bob Henry. He did some analysis. The fact that we had penetrations in the lower portion of the vessel leads to most likely failures of another type a progressive type of failure. So we proceeded to include that type of effect and this is shown in the PRA as an area of difficulties continuity. We weren't quite sure what -- we knew the model we had just wasn't really correct.

At this point we have the diaphragm floor with the molten material in contact with it. We did look at two cases here. One had the material largely confined in the area beneath the vessel. The other had the material spread over the floor.

It turned out that in timing the attack of the diaphragm floor was more controlling than pressure and this resulted in more rapid attack. So we used in this case as the base case which was slightly more conservative.

Here you can see that at this point we have filled the drywell with the material. We no longer have the path through the relief valve. We are now pushing any material that delta P will drive into the the suppression pool through the downcomers, but of course this is not quite as efficient as it was with the relief valves. Less of the material will get down and there is mixing in this region.

Temperature is increasing, the pressure is increasing, and we do have at this point the 10 percent oxidation occurring in the suppression pool.

MR. POWERS: In your analyses you were using something akin to the core concrete interaction?

MR. HUGHES: Yes.

MR. POWERS: Does that include degasing the concrete above the melt?

MR. HUGHES: I believe the answer is no. But let me repeat the question and Ms. Mendoza can assist me with the answer. The question is the code includes the core concrete interaction occurring at the interface between the core and the concrete. The heating that's associated with that event in the drywell region can cause some gas go or off gasing to occur.

Does this code include that type of capability or does it analyze that? I think the answer is no.

MS. MENDOZA: I didn't quite get the question.

MR. HUGHES: Rephrase the question.

MR. OKRENT: I thought you stated his question quite clearly.

MS. MENDOZA: Except for the decomposition at the interface, we did not account for any degasing other than the debris and the concrete interface.

MR. POWERS: Thank you.

I guess one of the questions that comes up with your familiarity with the plant is are there any areas that you think the analysis would be affected by this radiant heat being focused either on the concrete directly above the core debris or any structures along the annulus or whatnot?

MR. HUGHES: You have asked a very complex question and to answer it based on judgment. I'm not sure would be fair.

I'm not aware of anything that's in there that would cause a problem and I certainly will yield the floor to anyone else that's with us who has looked at it.

I think the modeling we were doing at the time was fairly simplistic in this regard and we have not re-visited it to think about the --

MR. POWERS: I'm strictly asking for a judgment call based on that, being more familiar with the plant than I am.

MR. HUGHES: I can't think of anything down there that would get me excited.

MR. POWERS: The only thing that I can think of is that your vessel is supported on something that sticks out over an area of an intense radiation. If that vessel were to drop does it pull anything loose or damage anything?

MR. HUGHES: You are looking at that portion there, and, to be quite candid, I'm not sure what the effects would be. I don't know. I understand the question, but I don't know the answer.

MR. BOYER: There is a path for natural ventilation and circulation through that, going from the opening at the floor level, which is about the size of a normal door. And the opening at a higher level, underneath the reactor vessel, which is the access for removal of control rod drives, so that again is another half a door. At least, you are able to get through it without just bending your head.

And there are some other openings where the control rod piping comes out, so that is partially -- two openings that are partially filled. But there is -- I would say a reasonable amount of natural ventilation that would be occurring through there, even assuming debris builds up like that, which seems a little improbable if it was -- if it was molten. It would still be hot but in a lump form to remain in that position.

MR. HUGHES: I think part of the answer also is we do have the rapid over-pressurization failure-type case included. So it would have to be some kind of mechanism that would likely change our view of what the likelihood of such an event was. But we are really speculating, I think.

(Slide 73 shown.)

MR. HUGHES: The containment failure occurs and what I've shown here is the failure in the drywell region. The other two types of failures that I also mentioned I've shown in red associated with the air space above the pool and with the pool draining itself.

In the next case I show one in the wetwell for class four.

Here we calculate a release through a break that's assumed to occur in the containment and we proceed to release the products that are being driven off by the molten core through the vaporization without benefit of depletion through the pool.

We do include the flashing of the pool at this point and release of the fision products associated with that. 15 percent of those that were previously entrained.

That completes TQUV.

(Slide 74 shown.)

MR. HUGHES: And I will now do the same for class four.

Class four is an isolation type of event followed by reactor failure to scram, coolant injection continues and the power is held up. We assume the power was at 30 percent for all nodes that were covered. This was an artifice. The power would probably be below due to reducing flow, but at the time we did the analysis it seemed like the best judgment we had available to us. So we have power at 30 percent. We have coolant injection continuing. We had the reactor vessel isolated. We are steaming through relief valves to the suppression pool. The suppression pool is heating up. Containment failure occurs rapidly compared to the other cases.

Upon containment failure we assumed that injection was lost. We assumed injection was lost with a probability of one. This is probably another conservatism in the PRA. Once coolant injection fails we proceeded to core melt into a failed containment. Releases occur then in an open containment.

(Slide 75 shown.)

MR. DAVIES: On the previous slide the 30 percent power level, that applies, as I read the PRA only to the portion of the core that's covered.

MR. HUGHES: That's correct.

MR. DAVIES: If that's a small portion it could be a very small power level overall.

MR. HUGHES: At the time before we have the failure of the containment, we are driving the power with, I think, RCIC and HPCI both operating. So that's the flow that we would have for the type of case we thought was reasonable and that tends to drive the flow up so that gives us the 30 percent power. Once we get to the containment failure and then we begin to boil dry, that's when we begin to drop the power level out. So it is not sitting there for substantial periods of time largely dry with nothing occurring. We have the thing generally covered with the 30 percent power applying to most of the core until we get containment failure.

(Slide 76 shown.)

MR. HUGHES: This is the time line for this type of event. Again it begins with MSIV closure. No scram occurring. Here we reach containment over pressure, failure. 140 psig comes in because we do not have the diaphragm floor attack. Core melt initiation occurs thereafter, due to the loss of the ability to inject.

Core slump occurs about an hour later. Around four hours we get the RPV bottom head failure, core concrete interaction vaporization phase beginning and the 10 percent oxidation release.

Around seven hours we have the diphragm floor penetration but that again is only academic interest.

I'll walk through the event as before.

(Slide 77 shown.)

MR. HUGHES: We start at zero plus minutes. Here we are very similar to class one at the initiating event. MSIV closure. We do not yet have flow through the relief valve lines but we will get it shortly. Power suppression pool is 95. Drywell and wetwell slightly pressurized.

(Slide 78 shown.)

MR. HUGHES: We go out to around ten minutes. We have set up higher pressure. Water level is beginning to come down. Or excuse me. Water level is coming down because of HPCI and RCIC. Not because of releases.

Here we have flow through the safety relief valve lines. The pressure is at 16 so it has increased very slightly but the temperature has gone up to 170 degrees F in the pool. So we are putting substantial heat into the pool in a rapid period.

(Slide 79 shown.)

MR. HUGHES: We go out to 40 minutes. Here we have the containment at ultimate pressure. The difference is 156, 155 between the drywell and wetwell. Both are approximately at 140 psig. This is where we assume containment failure begins due to the pressurization. Here we have the pool heated around 362 degrees F and we still have the flow coming through the same way but we are about to fail containment.

The next chart is only a couple of time steps later.

(Slide 80 shown.)

MR. HUGHES: Here you can see the temperature has dropped in the suppression pool slightly. Pressure has begun to drop in the wetwell and we begin to come down.

Again, in terms of the release path we looked at drywell, wetwell region above the pool and wetwell region below the pool. For these charts I've shown the release path for the wetwell region above the pool.

MR. POWERS: Your analyses don't depend on assuming any size of break here? You just make an assumption?

MR. HUGHES: There is an assumed break size for the flow rates out of the building. And for the subsequent analysis with corral, I'm trying to recall the number -- three square feet.

MR. POWERS: That was not -- your results aren't very sensitive to that?

MR. HUGHES: I don't recall doing sensitivity studies, but I shouldn't think so.

MS. MENDOZA: Pretty much -- you have depressurized at the time that the core melt and release from the fuel occurs and so the driving force really would be not dependent on the pressure and sensitivity on the --

MR. HUGHES: We come out to 45 minutes you see the pressure is coming down.

(Slide 81 shown.)

MR. HUGHES: Pool temperature is coming down. The release is continuing to occur. Come out to 1.2 hours.

(Slide 82 shown.)

MR. HUGHES: I've come still further down and at this point I'm beginning to initiate core melt.

Here I've had the break. I'm removing the material. I'm boiling off -- so boil off is occurring. The level is coming down and I'm now beginning to melt the core.

The next chart shows the --

(Slide 83 shown.)

MR. HUGHES: -- 2.2 hour time at which 80 percent core melt has occurred. Core debris slumps to the lower head. We begin to attack the lower head region. Throughout this period of time we have had our flow path still through the relief valves to the suppression pool because we still haven't broken the reactor pressure vessel.

(Slide 84 shown.)

MR. HUGHES: At around four hours we do achieve vessel rupture. We then move into the diaphragm floor attack. At this point the containment is intact. We have again 10 percent of the material going through an oxidation release. Rapid release of that material. We have 90 percent of corium on the diaphragm floor going through vaporization. The vaporization occurs and the material is released.

Here there is a difference in terms of where the failure is. For the vaporization effects here with the failure as shown the material is driven through the pool and out the break.

Once we get the material through the diphragm floor then, of course, we have the rest of the material released.

If this break -- for the case where the break is above we have this material driven off directly without benefit of the suppression pool. So those were modeled separately.

MR. EBERSOLE: Which was the pressure in the RPV when the bottom came out?

MR. HUGHES: 150 psi.

MR. EBERSOLE: Did the bottom come out suddenly or progressively.

MR. HUGHES: In this analysis we took benefit of some work by Dr. Henry. The code would calculate an instantaneous vessel failure due to pre-collapse but with the many penetrations it was viewed that a more gradual release would occur.

MR. EBERSOLE: It would then come out in enlarging steams?

MR. HUGHES: Yes.

MR. EBERSOLE: Wouldn't it plow a hole through that immediate floor, just chew it straight through.

MR. HUGHES: I think I'm going to defer your question to Dr. Henry's presentation that will be coming up shortly.

MR. EBERSOLE: All right.

(Slide 85 shown.)

MR. HUGHES: The next chart is just a little later in time, three hours. Here we do have the diphragm floor failure and for that period of time we have been taking the vaporization release through the break by whichever path is applicable.

(Slide 86 shown.)

MR. HUGHES: That completes the description of class one and class four as they were analyzed.

I want to talk for a few minutes about the radionuclide release categories and the results that came from the analysis. The flow paths, the pressures, the quantities of flow, the timing was fed into the corral code.

Corral was used to calculate the fission product released. Source terms were compared for different classes of accidents and different release paths and we went through a binning process. Binning reduced the number of cases that we would have to run for off-site effects.

What you see here are the results of that binning. The 11 different radionuclide release categories -- for those who count charts -- the leaks count twice because there are two different classes.

The oxidation release is generally the steam hydrogen explosion type rapid release cases. The class one, two, three, overpressure were grouped into overpressure release or OPREL. Other cases are as shown.

The bottom three are associated with either random reactor vessel failure or seismically induced reactor vessel failure and seismic events.

I particularly wanted to point out the numbers associated with the release fraction for iodine, helium, telurium. In particular I want to point out class four gamma prime prime for which the release fractions are quite high.

I think in terms of the conservatism of the analysis and one of the questions of what types of things might drive it worse this certainly indicates that we did include cases of substantial postulated release.

If you also look above you will see oxidation release and a class four gamma are important. If you look at a class four gamma prime, you can see in that the effects of the suppression pool since this was a failure in the wetwell region above the water level for which we had sustained clean up through the suppression pool.

This, of course, is only part of the story. The other part relates to the timing.

(Slide 87 shown.)

MR. HUGHES: And I'm a little outside the area of phenomenology but I just wanted to present this to complete the picture and then we will proceed to Dr. Henry.

What this shows are the various parameters associated with the releases as they were included and you will see that again the class four gamma prime prime is a rapid case being rapid and having large fractions released makes it important to the consequence analysis.

(Slide 88 shown.)

MR. HUGHES: If you take all of those and rack up what did we conclude or where did we come out, this takes the frequency of each of the different release categories and plots that as bars. It shows the severity of the type of release in the terms of the effects of they would have from left to right. More severe being more toward the left. Less severe being more toward the right. And this is not a one, two, three ranking. These are general where things fall.

As you can see, the more severe type consequence cases are indeed less likely. The more likely cases are indeed less severe.

This is somewhat reassuring but this was really done recently based on results some time ago.

(Slide 89 shown.)

MR. HUGHES: I've got about three more slides that relate to phenomenology but indirectly. So if you can bear with me.

This slide goes through things that are not included. The first item is current phenomenology. I told Bob Henry I would have a chart with his name on it and I do, so he will cover that.

Containment sprays have been identified as a means which could be used to arrest the progression of an event or to clean up some of the fission products. We did not include the containment sprays in the analysis.

RHR operation during core degredation, indeed system operation during core degredation to cool the suppression pool to take other steps to stop an event were not included.

I noted the comment by Mr. Ebersole earlier today that this type of thing is important. We certainly agree. But we didn't have the information that we could put into the study.

The venting of containment has been talked about. Venting of containment is included in the procedures, training is being done with -- it has been done. The plant can do it. This has been evolving over the last four years. It is not in the PRA and it is certainly a conservatism in what is there.

The low pressure injection for ATWS would include the possibility that if we do have ATWS event with the core having minimal flow to it we might be able to extend the event in time and reduce the consequences or we might be able to arrest it altogether with very low flows from low pressure systems. We didn't include that type of thing at all.

The external water sources that were assumed not available or actually were not analyzed, they were not included as shown. These exist, the CRD, for example, is there, but we didn't take credit for it.

ADS on low level only is a recent modification. At the time the PRA was done ADS was initiated with low level coupled with high drywell pressure coupled with low pressure pumps running. In this case the ADS has been modified and the drywell permissive removed. That obviously is not credit as it only recently occurred.

Core concrete attack, we are aware of development in this area that suggests the type of attack and penetration depth and gases released may have been overstated in our analysis, but we have not redone the analysis.

Primary system retention, major area where we did not remove material in the primary system. I think Bob Henry may speak to that but that's an area where we certainly move the material either into the air space or into the suppression pool. We did have resuppression from the pool with the 15 percent flashing.

Decontamination factors, suppression pool scrubbing, and the fact that numbers significantly larger than ten and 100 are discussed and talked about. Where that will settle out, I don't know, but it may be another conservatism.

Non-procedural operator intervention errors is a possible non-conservativism. This is an area we identified a week ago one where, while we do have this type of error in transient initiating frequency, we certainly can't and don't claim completeness in covering all of these types of failures.

(Slide 90 shown.)

MR. HUGES: The last chart includes a reminder that the study was done four years ago. I think at the time we did it we were at or near the state of the art. But of course it has progressed substantially and Dr. Henry will discuss that.

We have had some advances in understanding that were associated with the study. We did reduce the steam explosion probability. The decontamination factors were increased, but perhaps not as much as they might be today.

Mark II containment failure pressure had not been determined before the study. Fission product retention in the reactor building was included.

The RPV failure mode was studied by Bob Henry and is in appendix H and added something to the understanding of that mechanism of failure and we struggled with the issue of molten core debris disposition.

As indicated earlier we ended up with the judgment but I think the thought process had some value.

Source terms are comparable to WASH-1400. The accident sequence frequencies are also comparable, but certainly lower. I think our view of the study, those of us that were involved in it, is that it stood up fairly well. It is probably conservative.

You have been patient. I know Bob Henry is ready to talk about more recent phenomenology so let me turn the floor over to Bob.

MR. HENRY: As Gene just stated, codes used at the time were state of the art. They were coupled with -- they still had some regions that had to be addressed by separate evaluations to stand along calculations to be plugged into the codes, to guide the assumptions. I would like to briefly go through a couple of those, as we walk through some accident progression, also give you a feel of what we feel the net result would be of work that's been done since this study was done approximately four years ago.

(Slide 91 shown.)

MR. HENRY: The conclusions we came to -- which I would like to put up first, and the third one is the one I would like to draw your attention to.

First off, the study itself, the calculations did not include the influence of control rod drive flow. It included some of the sequences which were initially judged to be core melt sequences would indeed not even cover the core. Not a large fraction of them but it was a conservatism in the study. It was not available in the code package at the time.

As a result of some of these other issues we will talk about some of the core melt sequences which would release material from the vessel could indeed achieve a stable state in the containment and would not result in containment failure.

But it is coupled with this one. It says failure sequences are very long and the time of containment failure is very long and principally the containment would be failed due to steam overpressure, which means debris is largely going to end up -- a large fraction of debris degree would end up in the suppression pool.

So this is where we will spend quite a bit of time over the next half hour or so, by your schedule.

Is that about right?

MR. OKRENT: We have 35 minutes.

MR. HENRY: Assumptions which are in the Limerick PRA, the first two, no CRD flow. RCIC was assumed to be insufficient to cool under an ATWS state. Large quantities of core material were required to fail the vessel at the time. It was an assumption of the code that one had to accumulate 80 percent of the molten debris before the vessel would fail. This was one of the things which guided our judgment. It was an overstatement.

We certainly don't feel that's the case now. You don't have anywhere near that amount of material before the vessel would be threatened thermally.

Given this the assumption from the hand calculations said in -- they forgot to take this off the slide -- debris would be spread over the floor before you could calculate any other failure of the diaphragm. As a result the calculations were carried out with the debris on the diaphragm floor.

If that's the final state, of course, you have a concrete attack there is no influence of suppression pool cooling. It has been stated a couple of times already, there was no primary system retention.

(Slide 92 shown.)

MR. HENRY: Current understanding -- modeling certainly says the CRD is most important. RCIC could potentially be important until the CST is depleted or until some other arrangements are made.

Also I want to take about 20 percent in BWR core tc result in thermal attack of the vessel. Look at some specific geometric considerations, but this is what would govern the failure of vessel under current understanding and also then the resulting accident progression.

MR. DAVIES: On the CRD thing, I haven't looked specifically at Limerick but in other plants I believe that's about 80 gpm per train. Two trains give you about 160 gpm which is not sufficient to remove decay heat.

Is this plant different?

MR. HENRY: The number you quoted is without scram. With scram it is about 120 gpm, depressurized 180 gpm and that is sufficient to remove about 1 percent --

MR. DAVIES: Pretty close call.

MR. HENRY: The only reason I put this up, if you were to for instance analyze Browns Ferry with the code package at the time that would have been a core melt sequence, but it wasn't.

(Slide 93 shown.)

MR. HENRY: Again current modeling would lead us to the conclusions we come to in a second. Debris would be distributed over the pedestal floor, however, about 90 percent would end up in the suppression pool, as a result of the specific design of the floor drains also the equipment drains which are inside the pedestal and possibly even failures of the downcomers which are immediately outside of the passage between the pedestal and drywell region.

Fission products in the current codes are released mechanistically. Major fraction would get deposited within the primary system. Natural circulation both within the primary system within the containment and the subsequent heat up of the primary system determined the ultimate fission product distribution and, depending upon some very specific features, in this particular plant you find you can get substantial retention of the primary system permanently.

MR. POWERS: If you put 90 percent of the fuel into the water draining through the drain and whatnot would that increase by factor of nine the oxidation release that was used in the original state? Based on 10 percent.

MR. HENRY: If you made the same assumptions which were made in the study, that would be a logical conclusion. But I think when you look at the circulation that results within the pool you come to the conclusion that virtually all of of it would again be retained in the pool because most of the quenching gets carried on way down into the --

MR. POWERS: So you would allow release to occur. It would just be trapped by the overlying water?

MR. HENRY: If the release is mechanistically true -- and you know as well as I do that that's a debate between a lot of chemists -- but making that assumption, if you did it the way it was done in the study then the real conclusion would be that it would follow straightforward.

On the other hand this is the quenching, as we will get to, that actually occurs deep within the water so you would expect it also would be trapped.

(Slide 94 shown.)

MR. OKRENT: Did you consider whether the vessel would fail above the core?

MR. HENRY: I'll get to that in just a second.

(Slide 95 shown.)

MR. HENRY: In fact, right now.

Influences of natural circulation of the primary system, and this is something that attention has been brought to as a result of TMI and as a result of a lot of additional -- particularly the PWR systems, whether you could have substantial natural circulation from the core into the upper plenum to heat it up.

You will not have this in your package, but let me just refer to it to identify exactly what we are talking about.

(Slide 96 shown.)

MR. HENRY: Once oxidation begins you get fairly high temperatures in the core region. The question is can you get natural circulation from here to here to bring all this steam back into the core to sustain the oxidation process, because if you can't the oxidation process is merely limited by how much water you boil off inside the core region.

Well, within BWR you find that the geometry itself provides a natural impediment to circulation of that large mass of steam back into the core because the separators themselves really provide you with effectively a flow dike at this location here, because the gases inside of the stand pipes are hotter than surrounding it. So the potential for this is to rise, not to fall back down in.

The natural circulation itself is really limited to the steam you have inside the overall bypass region before you go into the separators. When you do the balance that's a very small amount of additional steam compared to what you are pouring off here.

MR. EBERSOLE: I believe you were the gentleman who was going to describe how this 1100 pound pressure in this liquid fuel is going to emanate from those control rod drives.

MR. HENRY: We will get to that.

MR. EBERSOLE: You are going to get to that?

MR. OKRENT: Let's see.

In what you just answered were you arguing that the top of the vessel will not get hot?

MR. HENRY: Yes, sir, you get very little energy transfer up here and the mass of this material is more than sufficient to keep these temperatures well within the range where structural integrity would be maintained. It is a much different consideration than what one would have in the PWR system where there is no limitation on circulation.

As a result the oxidation is limited by the steam starvation which comes from the water depletion within the core.

(Slide 97 shown.)

MR. HENRY: Natural circulation of bypass can give you some addition but it is really a second order effect.

Following vessel failure natural circulation of the primary system is the thing which can determine where fission products can ultimately be deposited, perhaps revaporized and redeposited.

Natural circulation within the primary system is indeed most important to look at for the mechanistic progression of the accidents.

MR. POWERS: Following vessel failure natural circulation in the primary system from where to where? What is the distance? Show me the hot spot, Bob.

(Slide 98 shown.)

MR. HENRY: Typically after vessel failure you find a hot spot here, hot spot here, because fuel is still here. There is a reasonable amount of volatile fission products upwards of 15 percent of the decay power in it. So once the vessel has failed here there is always the path for colder gases to fall down through the annular region, flow being controlled by the forces of jet pumps and circulating this way, then.

There is also a hole in the bottom of the vessel. You have to consider whether this flow is coming in or going out if the containment is depressurizing.

MR. POWERS: Even if your picture of the core meltdown process were wrong and you had a very homogeneous core meltdown pressure you still have a natural circulation which is the hot spot after the separaters.

MR. HENRY: After the vessel fails to lose its water block down here, that's correct.

(Slide 99 shown.)

MR. HENRY: Vessel failure mechanisms, a couple of things enter into specific consideration. One, the core plate which locates assemblies laterally at the bottom is not designed to take a large weight of material. Once you begin to -- that plate can be loaded, a simple beam calculation, say, about 30 tons of material, assuming that the plenum is not heated at all, is sufficient to start failing that which would allow the debris to fall down in between the assemblies into the lower region.

(Slide 100 shown.)

MR. HENRY: So under that configuration which is about 15 percent of the core material, this plate sitting in here would allow the debris to fall down in between these control rod drive guide tubes.

Another potential failure mechanism is through the in-core penetraion tubes which go up through the core.

(Slide 101 shown.)

MR. HENRY: They essentially have containment pressure inside of them and are about one-and-a-half inches ID. Typically like 47 of the probes available and five at the most will be occupied. Both of these will provide localized failures.

Just for your own reference I gave you a drawing that's not a particularly good viewgraph, but two pages later --

(Slide 102 shown.)

MR. HENRY: These are what the penetrations would look like. This is a CRD penetration from guide tube with a limited penetration weld here and the same thing is true for the in-core penetration. It is a limited depth weld which is maybe something like an inch or so. At the most it would be about six inches.

So this ID is one-and-a-half inches and this is the containment atmosphere up through the core. This type of failure, if it were to melt up in the core and the melt has super heat, super heat would be the temperature above its melting point, of something like 100 to 200 degrees Centigrade, that it has sufficient thermal energy to flow all the way out into this region and establish failure to the pressure boundary.

If this for whatever reason were to be plugged and the material finally fails the beam that I was telling you about it would affect the beam calculations on the core plate, debris would fall down into the lower plenum region, of which there are all these limited depth welds to be penetrated and just the initial flow of the debris could fail these welds in the range of time frames by analysis of 15 to like 60 seconds. And if you have the vessel at 1,000 psi or 1100 psi --

(Slide 103 shown.)

MR. HENRY: -- it gives you a feel for the timing, and you have something like 20 percent of the material down here, which is about 40 tons of failure to single penetration, the blow out of that material and the subsequent ablation that it also would do to the vessel, the discharge of that material would occur at about two seconds.

If you were to go back to the study which assumed 80 percent of the materials in the lower plenum that discharge takes about six seconds. It is a very rapid thing because it can ablate the hole and make the hole larger.

So the time frame we used at the time was 80 percent of the material could come out in about six seconds.

I'll get back to that time frame in just a minute when we talk about the integrity of the diaphragm floor.

MR. OKRENT: Where is liquid water if at all in the vessel during this time?

MR. HENRY: I didn't hear the first part of your question.

MR. OKRENT: Where is liquid water if at all in the vessel during the phenomena you have just been describing?

MR. HENRY: Depending upon sequence. Let's take a high pressure sequence, then the lower plenum is usually filled with water. And the water should be considered to be in two separate regions. The first is this region outside of the control rod drive tubes which is intimately coupled with the shroud region through the jet pumps.

The second region is inside the guides tubes which was only coupled up to the rest of the vessel up through the path that the debris is coming into. So if the material tries to fall down, if it tries to go into these tubes and tries to vaporize the water in there the steam has to flow out the same path the debris is coming in.

On the other hand, if it goes between the tubes -- these are like eleven inch tubes on twelve inch centers -- it can go down through that central star-shaped region, then that water can merely be displaced up into the surrounding shroud rod region.

Of course we do that balance and the second one easily wins out. There is water here but the debris just falls right through the water because it is seven or eight times heavier.

It can also freeze on the control rod drive tubes. As it goes down that gives you some steam generation that goes back up into the vessels, but that freezing is insufficient to prevent it from coming all the way down to the bottom.

Do you have another question?

MR. EBERSOLE: If there were no water at all and then I managed to super heat this material, just prior to its ejection from the nozzles and then unfortunately I think, I suddenly found a source of water, could I have a rather catastrophic explosion pressuized in the vessel and would catastrohically take it apart?

MR. HENRY: No. The velocity of that evaluation as well as experiments which are certainly not to this scale or anything, but you remember TMI, for instance, had a very rapid refill of the core as well when the core was grossly overheated. Whether or not there is water in here and the debris comes down through it or the debris is down here and you turn the water on, they both have roughly the same type of steam generation rates. To give you a feel of it, the kind of rate you might anticipate would be 200 megawatts of steam.

I just put these pictures together to give you some rough idea of where things would be distributed at key points in time. We can go through these fairly quickly or delay them as long as you like --

(Slide 104 shown.)

MR. HENRY: -- as long as we stay on the schedule.

Just before we fail the vessel we are looking at something perhaps with 20 percent of the material. Maybe upwards of like 40 tons is in the lower part of the vessel and remaining water -- the water -- I mean here is the water which is outside the control rod drive tube so it is intimately coupled with any failure location.

About 80 percent is still up here in the core being overheated and is also continuing to oxidize because as a result of the steam generation you put some steam back up through the vessel which those things that were steam starved can now also continue through their oxidation to some degree.

So now when the material fails we first have to look at the size of the failure you have in the vessel and the rate at which this material comes out. And as Gene Huges was saying earlier, of course we had 80 percent of the material here in the study by assumption that was the fundamental point of starting the containment analysis.

MR. POWERS: Does the failure of that configuration of water over 20 percent of the core endanger that energetic interaction a clearcut call -- isn't there a substantial amount of debate based on experiments at Brookhaven National Laboratories and at other national laboratories that would lead one to believe that there is potential for an explosion in the configuration?

MR. HENRY: I'm not saying this is benign. It is steaming at a rate of about 200 megawatts. That's not -- that's a lot of steam generation. I think your question is directed towards whether or not you are going to have an explosive interaction. For this particular one, as you know, the argument on steam explosions has gone on for years.

MR. POWERS: I don't want to get into it either.

MR. HENRY: Let me point to what I think is the simplest answer for this particular case, that is the kind -- the area that you have available for interaction in BWR is very limited because you can't get material down inside of regions where the only path of the water and steam to go out is back up through that same zone.

So you are really talking about that region between these large control rod drive tubes, which as I said are eleven inches in diameter on twelve inch centers and you are talking about the cusp that's in between them.

So it is very difficult in the BWR to get large masses of material coherently interacting to do a substantial amount of damage. To me when I look at the bottom of the BWR that's as far as I had to go to say there is no way you can get explosive interaction in the BWR.

Other people don't have the same insights that I do so they don't come to the same conclusions.

MR. POWERS: It is fair to say there have been no explosion tests that I'm familiar with in which that kind of obstruction has been present. How long are those assemblies going to be remaining?

MR. HENRY: These assemblies, depending upon on what the sequence is, if it is high pressure sequence they are full of water. If it is a low pressure sequence like ADS then you could have flushed it out.

I believe there is still a meeting on the 27th and 28th of November to discuss steam explosions. This is one of the key points which I intend to make at the meeting. If you do these tests, especially the pressures that represent reactor systems, the structure is a key part of how the reactor system would respond. BWR has very extensive lower plenum structure.

MR. EBERSOLE: Is it true generally if you pour water on molten steel that it simply freezes at the top and the water boils off the top?

MR. HENRY: If you pour water on the top of molten steel you can certainly get explosions. One of the problems in the steel foundry industry is that they don't see it every time. They will do it 100 times and it won't happen. Generally you won't get an explosive interaction. People get very confident and then when they have one a lot of people get burned. But you can certainly have one.

(Slide 105 shown.)

MR. HENRY: I would like to move on to something to a case that's just after the vessel fails -- now, and before we get into it, I would like to show you a few of the flow paths that are considered in analyses carried out since the time of the study.

I believe Gene Hughes talked earlier about the gas flow paths and also Mr. Boyer mentioned there is a control rod transfer door through the pedestal at this location. There is also four control rod drive windows, two on each side, which just have the hydraulic lines coming through, giving you an effective area, they are about 90 percent open, 80 to 90 percent open. It is one door passageway here out on to the drywell floor. There is no step in that particular system.

MR. OKRENT: When you say "door," do you mean doorway or do you mean door?

MR. HENRY: This is a doorway here. There is no door. This is a passage way here.

So these are the regions which determine the natural circulation flow between here and here. They also are the regions which determine where gas flows would be exhausted if the primary system were blowing down into you. So that's part of the things we have to consider.

One of the things I think is different here than -- and perhaps I should point it out now -- is that with the parallel passageways the calculations that you do on dispersal are somewhat different now because the path available for just gas flow up high that doesn't have to carry heavy debris with it can be favored against the low path down here. So it is something where you have to look at parallel paths and the influence on dispersal.

MR. POWERS: Based on some of our experiments at somewhat higher pressures, about 1500 psi, that the debris has enough kenetic velocity to follow those higher pathways.

MR. HENRY: You can indeed get material to splash up here. When you do the analysis you have to look at the flow split between the regions and a sustained entrainment which is a typical two phase flow evaluation.

We also have the possibility of having gas, liquid, and solids into the downcomers and the one I would like to talk about now are the drains that have been alluded to several times during the day, because this diaphragm integrity here, while it was assumed at the time that this remained -- the integrity was maintained, we only came to that conclusion because we started with a base assumption that we had 80 percent of material up here which was molten at the time and that discharge took place in something like five seconds.

The time to fail these drains is in the range of 15 to 60 seconds. So the conclusion we came to at the time, if you have 80 percent of the material it will be distributed on the floor.

On the other hand, as we have talked about, it is difficult to see where you would ever get to a point where you would have 80 percent accumulated before you failed the vessel. So then the design of these drains and their integrity become much more of an issue.

Let me first talk a little bit about debris dispersal.

(Slide 106 shown.)

MR. HENRY: There are a couple of features of the accident, especially for Mark II type systems, that are more influential. The debris dispersal requires that you have a reasonable amount of pressure in the primary system at the time. Something greater than 150 psi or so.

When we get to the higher pressure sequences, especially those where we are at 1100 psi, the lower plenum is full of water. And this is determined also by the accident definition because those things that run a long time also have a lot of water just by CRD injection.

When we flash the water after vessel failure, remember we have the debris coming out then the water comes out. And then the gases can come out as well. I'll show you an example for this. 50 to 90 percent of what you assume to be finely particulated to go into the atmosphere is going to go to the pool.

If you assume you have finely particulated debris you would expect that debris to pretty much follow the gas flow especially if it is going to stay around long enough to try to exchange heat with the containment. If you don't have finely particulated debris particles then you also have something like about 50 to 70 tons of water which is coming out in the process of the flashing which is going to go with the larger size of debris.

In other words, if you don't have finely particulated debris there is going to be a lot of water available. If you do assume this is finely particulated it's going to also have a hard time staying out of the suppression pool.

We can look at that, I think, in a fairly straightforward way. I gave you an example here that assumes that we start off at about --

(Slide 107 shown.)

MR. HENRY: -- 1,000 psi seven MPA --

You have to excuse me. I work in the SI system.

The gas volume available at the time of failure is about 500 cubic meters. Something like maybe 80 percent of the primary system volume. We will assume it has an average gas temperature of about 800 kelvin. That's a realistic number. It could be as high as 900 or so but in the average it is not too much different than that.

If you do that calculation that says you have got something over 500 moles of gas and if you assume it is half steam half hydrogen this translates into 250 moles of hydrogen or 500 kilograms, about 1,000 pounds. So that's more than any accident that we would certainly calculate for the oxidation. If anything, you will get even more steam than what I'm assuming.

MR. POWERS: I thought we were steam starved on our Zircaloy reaction at this time.

MR. HENRY: This is a hand calculation. I merely made an assumption here that this is half steam half hydrogen. Typically if you carry through on an analysis for the boildown you will find that this is in the range of maybe ten to 15 percent and this is then 85 to 95 percent steam.

Just for the sake of this -- because -- by looking at how much steam you are going to produce you will have to blow it down and you are also going to flash. You can then look at the ultimate distribution of the things that you are going to put into gaseous form.

The saturated water volume we deal with is roughly 100 cubic meters roughly below the core and that's what you are going to blow down and it's something like 74 tons. So the steam that's going to be formed by flashing as you depressurize this to one atmosphere you get about one fourth coming off at steam. That says the amount you are going to flash is about 1100 moles and compare that to something like 250 we have with hydrogen at that point in time.

(Slide 108 shown.)

MR. HENRY: Of course as it tries to pressurize that gas and vapor goes to the suppression pool. So the fraction of noncondensables that you have in this example is 20 percent. And the fraction that you have of condensable is 80 percent, of course.

So that says about 80 percent of gas flow goes through the suppression pool. Just very crude numbers. So also that tells you if you postulated if you have finely particulated debris in the blowdown time, which is about ten seconds, then all that material is also going to go to the suppression pool.

So if you postulate that you have very finely fragmented material to heat the gases and the inerted containment atmosphere only about 20 percent of that could indeed even try to heat the atmosphere directly. If it were fine enough to stay with the gas it is going to go to the pool.

MR. OKRENT: You say "go to the pool," but that takes some time. How much time does it take to heat and how much time does it take to flow?

MR. HENRY: Let me give it to you slightly differently. You are not so much worried about actually heating the gases themselves. You are worried about whether that energy can be transferred to any equipment around. So the time for this flow, which is at most ten seconds, is short when compared to the total response time of the structural members of any equipment that you are concerned about in the drywell, like the drywell spray systems.

MR. OKRENT: You are saying the depressurizers are cooled down even if it gets hot.

MR. HENRY: Even if it gets hot -- I don't think you would catch the fact that it got hot because there is so much water coming after that and also all that material is really going to the pool.

MR. OKRENT: By the way, did I miss it, is there some chance for further generation of steam from this water that didn't flash meeting some hot fuel?

MR. HENRY: I did not say. You didn't miss it.

There is a chance that would certainly be calculated at a complete set of codes. As the water comes out, hits the debris, it would start generating more steam.

All that really does is that ends up going to the pool which even reduce this even more. But you can indeed generate more steam.

Another key point to that is that there could be debris that doesn't go to the pool because your downcomers are 18 inches up off the floor. Then also the evaluation, this is what you are just driving at, if you have 74 tons coming out you have got about 55 tons of water left and that has to pressurize to saturation. So this is the water that's only going to be going on the floor. So the time difference is a couple of seconds. Very fast compared to the failure time.

(Slide 109 shown.)

MR. HENRY: Just after the vessel has failed, in a rough perspective we consider having about 80 percent of the material up in the vessel. It now has a hole in it. Have maybe 10 percent left on the floor which could have been either quenched as a result of water coming on top of it. It's also exchanging heat directly with the concrete when it first comes out.

Debris could have also gone into the suppression pool through the downcomers. And debris could have gone into the suppression pool through these vents. And this is the part I'd like to come back to now, because the failure time for these and their specific design is a very important part of how this particular containment would respond.

MR. OKRENT: Is this whole story really dependent on the conclusions on whether it is 20, 50 or 80 percent that's in the bottom when it fails?

MR. HENRY: I guess what we heard this morning, depending on how you use the analyses you would say no. If you are asking me if the total accident progression and the time of containment failure is depending upon it, yes. For this system if it does not go into the pool then you won't really change much from what was done four years ago.

(Slide 110 shown.)

MR. HENRY: I'm not sure how well these will show up here. You may be better off looking on the copies you have in front of you.

This is what the floor drains look like, of which there are two in the pedestal region and something like eight, I believe, on the drywell floor. The four inch pipe is here and they sit in a hole which is approximately a foot in diameter. Diaphragm integrity for the floor drains is represented by this one inch annular piece of steel that's welded in place. So the debris comes out and covers this. This is the thing which is actually the diaphragm integrity.

For the equipment drains a similar type --

(Slide 111 shown.)

MR. HENRY: -- configuration is used which sticks up above the floor. And again there are two inside the pedestal and, I believe, something like eight out in the drywell floor. This represents the diaphragm integrity. This is the wetwell suppression pool atmosphere here and this is the drywell atmosphere here. This is a blowup of it here.

If this has failed then the debris can go directly into the suppression pool as a result of this impass. This was looked at at the time and it is part of appendix H to the Limerick study.

And again, given the initial condition that we have 80 percent of the material, it got distributed in a time frame, order of magnitude ten seconds, which is comparable to the time it takes to melt this through here. Again it is like 15 seconds or so to a thermal analysis.

So our conclusion at the time was material was going to get distributed on the drywell floor. We will then have concrete attack and that's the way we should ananlyze it.

As a result of looking at more details in the accident progression details of the vessel geometry, we come to the conclusion that vessel would fail maybe with 15 to 20 percent of the melt so the rest of at the time comes out over the next five to ten hours. So in essence what we are then looking at is the potential for debris to go directly into the suppression pool.

(Slide 112 shown.)

MR. HENRY: You don't have this in your handout, one tank which is for equipment drains and another for floor drains. There are two floor drains, two equipment drains that come out of the pedestal region. These are the ones which I think would really control this whole evaluation because both of those -- actual steel plate is -- three quarters to one inch.

As soon as that melts directly it seals the suppression pool and any debris coming out of the vessel can flow into there except for what can be stablely (sic) frozen on the surface of the pedestal floor.

So getting back to the question that you just asked, are you sensitive to it? No. But that's also yes if you are looking at total accident progression, but that next amount of material is coming out over several hours and there should be no impediment for this going directly into the pool as a result of that.

MR. POWERS: Do drains that are going to carry all our gas flow -- the outer ones -- are those similar in diameter and whatnot as those within the central floor?

MR. HENRY: The gas flow during the blowdown? I don't think this is fatal.

MR. POWERS: What I'm asking is are they similar in diameter on your downcomers?

MR. HENRY: Drains are four inch pipes. They go directly into tanks. This tank you should look at as being drywell atmosphere. It is only like a 1,000 gallons, the gas flow as shown earlier going in through these 80 downcomers.

MR. POWERS: There are 80 of them?

MR. BOYER: 84.

MR. HENRY: How much time?

MR. OKRENT: Five minutes.

MR. HENRY: Once we have a bypass of the diaphragm floor then a couple of things should be involved in the evaluation. One is how -- of this debris falling into the pool. You need to evaluate the split of energy between what is formed as steam and what goes in to increase sensible heat to the pool. If all of it goes to steam, of course, then you would overpressurize the containment, but that's not physically correct.

Also the natural circulation flows through these compartments must also be looked at because that's how all these heat sinks come into play over the progression of the accident.

Since we are running a little short on time, I gave you an example for the quenching model which has been developed since that time to look at the physics which have been associated with Mark II. Let me just give you a very quick runthrough without going through the example, and tell you what pieces of physics are important in this case.

As debris comes into the pool and tries to quench it generates steam. As a result of it trying to fall into the pool you have heavy debris in the pool. So you have a balance between this highly voided region, because the steam wants to escape upward, which tries to set up circulation flow of liquid through here and this very heavy material sitting in the interaction zone.

So you look at a balance. To get an idea how big this interaction zone is you would make this as deep as required to set up the circulation flow which gets the terminal velocity of all these particles.

As you go through the calculations you find you are actually independent of what that particle size is. The net results of the calculation says if the pool is saturated, of course, everything goes into steam formation. If the pool is subcooled approximately 50 degrees Centigrade nothing goes into steam formation. And someplace in between you get a whole spectrum of how much goes in.

But that should be part of the overall accident progression and is in the more current models. As debris comes into the pool you actually calculate the split of how much goes out as steam to pressurize the containment and how much of that energy by quenching goes directly in to increase the sensible heat in the pool due to the natural circulation of liquid in the pool around the regions where it is trying to come in.

This is where you would also get some retention if you actually formed a oxidation release. This is where it is also getting trapped, because this is where the circulation is substantial.

MR. POWERS: Does this model also include the formation of hydrogen? I think that would be have very important for this kind of a scenario and especially for BWR reactors where you have got so much Zirconium available to participate in the energetic interaction of water.

MR. HENRY: This particular one I gave you in the calculations doesn't have noncondensable gases in it. The only way you could get substantial hydrogen formation to compete with the volumetric formation of steam is if you made this very, very, small particulate. Since this material is only dribbling into the pool its size should be determined by its capillary size as following through the region up above the pool. That, of course, is in the range of centimeters for this high surface tension material.

MR. POWERS: Wasn't there quite a little bit of work being down on dropping the droplets of that dimension roughly into water that shows there is a substantial amount of hydrogen formation without decay of heat?

MR. HENRY: This model has been compared to experiments where oxidation has occurred. I gave it to you, one for stainless steel, one for copper, in your handout.

MR. POWERS: Neither one of which were very energetic interaction with water.

MR. HENRY: Molten stainless steel is very rapid oxidation. More rapid than Zircaloy.

MR. POWERS: It is not self-propagating is the problem.

MR. HENRY: You can't have everything.

MR. EBERSOLE: If you drop a substantial amount of this material into water you do get an explosive reaction.

MR. HENRY: You can get localized explosions, but generally speaking people have experienced that when this is dropped into very deep pools that if the explosion is there you can't observe it.

(Slide 113 shown.)

MR. HENRY: As this goes into a pool the kind of depth penetration you have is maybe a meter. It's like one meter out of seven meters. For these very high temperatures found experience says it has to penetrate to some wall that can have pours of water and water inpours that can be heated up and exploded in order to provide trigger to initiate the propagation.

MR. EBERSOLE: Isn't it disbursed by the violence of steam formation and establishes very good transfer that it is distributed all over?

MR. HENRY: In order to get it down to regions where you can get very rapid oxidation you would be talking about something smaller than a millimeter. Typical sizes of this, as it just pours in over like five hours if it is set by the capallary size, as it falls through here is several centimeters.

The information I've given you in the back for experiments I believe one is stainless steel one is --

Yes, these were experiments --

(Slide 114 shown.)

MR. HENRY: -- that were just carried for other reasons, but it is hard to find these in the literature to actually tell how much went into the pool, how much was actually released as steam. Along with the stainless steel in the water in the two kilogram experiment the water was 80 degrees Centigrade so subcooled 20 degrees.

You would calculate half of it would be formed as steam. When you look at the pressurization they measured it wasn't quite that much. So that the model indeed overestimates how much steam would be formed. UO2 going into water 20 degrees C, say 10 percent would go in as steam. That's a very small amount.

That's in good agreement with what they saw. Copper 40 degrees C -- the people who did that experiment -- this is a long term one which is more typical than what we are looking at here just pouring in two-and-a-half kilograms a second. This is like five minute pour. They saw no steam coming out the top. You calculate that's the critical subcooling, nothing is going to come out.

MR. OKRENT: I think we are going to have to come to a conclusion very quickly.

MR. BOYER: It reminds me of my earlier days in the fossil boilers watching molten ash coming out of slime capped furnaces and I have steamed by arms on some boilers that have formed down when a whole mass of that came down at once, but we never had any problems with explosions down there.

(Slide 115 shown.)

MR. HENRY: Perhaps I can just wind up with this one.

This is what we would then calculate the ultimate distribution to be after everything has come out of the vessel. We have all our natural circulation but we can save that for some other time. As a result of this failure debris principally ends up in the suppression pool.

So the major difference between what you would look at now and what was done four years ago is that you make much more effective use of this major heat sink because the debris ends up in the pool. It did not end isolated up on the diaphragm slab. So this gives you instead of six hours to containment failure it is more like a day to containment failure. Similar kinds of accident definition.

MR. OKRENT: Any more brief questions?

If not, thank you.

We had best move along.

I could ask whether our consultants want to make any particular observations now on the material we have heard so far. Anything especially you can call out? It is not a requirement, but if there are some things you want to point out, this is an opportunity.

MR. POWERS: I just want to point out that it is not entirely clear to me that the evolving technology of either accident progression --

MR. BOYER: Speak into the microphone.

MR. POWERS: It is not at all clear to me --

MR. BOYER: Little closer.

MR. POWERS: -- the evolving technology of either analysis of accident progression or source terms is necessarily demonstrating that using WASH-1400 method has been a conservative approach. I think I see ways of -- from the newer methods in which source terms are actually --

MR. BOYER: We can't hear you.

MR. POWERS: I think I can see how source terms, particularly for those elements that were not extensively released in WASH-1400 analysis, would actually go up in the more modern, yet somewhat speculative new methods of accident analysis and source term analysis.

I don't think that's included in some of the things that Bob Henry pointed out to us which presents a fairly benign story and it doesn't include some of the more recent things that have been done at other laboratories.

But the analyses, for instance, for things like Peach Bottom accident sequences seem to be giving very, very high refractory releases that are not reflected in the --

MR. OKRENT: Is there something different between a Mark I and Mark II for the kinds of scenarios that might involve high rhenium release for Mark I?

MR. POWERS: I don't think their mechanisms built for Mark I give very high rhenium releases either for WASH-1400 type analysis, except there is a steam explosion or the more modern analyses. The high rhenium releases came about only when you have a strong oxidation cusp.

What is similar, I think, between Mark I's and

Mark II's is if you analyze the melt concrete interaction as it was described to us earlier today, when you have melt interacting with the diaphragm floor I think that's very similar to melt interacting with the concrete in Mark I. And I think vaporization releases were observed in the more modern analyses for Mark I's might be transferable, at least in a qualitative sense, to thinking about a Mark II, if we discount the draining mechanisms that were described by Mr. Henry.

MR. DAVIES: I want to make one brief comment that troubled me a little bit.

In going through the analyses, particularly the documentation, I note that a mix of optimistic, conservative, realistic, and arbitrary assumptions are made, depending on the scenario and the sequence and what that could do is mislead someone in terms of which sequences are important and which are not. And I think we must be very careful when we try to determine how to improve things in looking at this mixture of assumptions.

I think that's the main comment I have at this point.

MR. OKRENT: I think we had best begin the discussion on seismic questions now. We will probably have a break in the middle of it.

MR. MARTIN: Bob Martin of the NRC staff.

Some of what we would cover on the subject of the staff's best estimate of the seismic contributions to risk has been covered in various discussions before today. I would like to summarize in an overall fashion a few comments from our documents which reflect the work done individually by Mr. Acharya, Mr. Rosenthal and others of the staff with us here today.

I note that in the Limerick severe accident risk assessment the spectrum of probabilities of seismic induced core melt accident sequences varied over a wide range, several orders of magnitude. However, the mean that is the point of the best estimate probabilities of seismic induced core melt accidents sequences is used in the staff analysis which essentially came from the SARA, are within the range of probabilities developed in SARA and are within a factor of about six of the upper end of the spectrum of probabilities --

MR. OKRENT: You are talking too fast. I can't really understand what you are saying. For example, it sounded to me like you took your mean from their PRA. Is that what you are saying?

MR. MARTIN: For the seismic essentially, yes.

MR. OKRENT: You mean you didn't calculate your own mean seismic core melt frequency?

MR. MARTIN: The mean value of the seismic hazard from the SARA we took essentially from the severe accident risk assessment.

MR. OKRENT: When you use the term seismic hazard are you referring to a curve? I'm trying -- I'm sorry. I really don't know whether I understand what you are telling me. I wish you would give not so abbreviated a summary but -- we asked for a presentation in this area. Is there one?

MR. MARTIN: Perhaps Kelvin Shiu could help us with the seismic hazard information which we took from the SARA.

MR. OKRENT: One of the least well specified matters in the Limerick PRA and SARA, according to what I read in the Brookhaven reports, the place where Brookhaven seemed to have the most open questions, related to seismic, and then there was a big open question concerning seismic hazard curve.

They had a consultant who wasn't in what I would call strong agreement with the approach used by Limerick and the Livermore people's estimate would differ appeciably. So I would like to hear what the staff's best estimate is of the first likelihood of core melt from seismic induced contributors.

And, secondly, the contributions to risk and then what uncertainties in this are and how you have disposed of the things in the BNL report that they let this sort of questions relating to seismic safety -- I don't think that was discussed in any meaningful way at a previous meeting.

MR. MARTIN: The things raised in the BNL report we considered in the manner discussed in the beginning of that report and was discussed in a risk evaluation report.

In other words, various recommendations were made in that report published at that time. We evaluated them and came to the conclusion as stated in the risk evaluation report wherein we discussed the BNL report.

MR. POMEROY: Could you briefly state that conclusion.

MR. MARTIN: Excuse me?

MR. ACHARAYA: The sequences of the accidents initiated by the severe seismic events that were used in the staff analysis, the best estimate values of that essentially came from a SARA.

MR. OKRENT: Why do they come a SARA? I don't understand.

Do you support SARA as being correct in that area?

MR. ACHARAYA: Well, the staff did recognize that there is a substantial amount of uncertainty in the hazard function and analysis, but at the time that the frequencies were used in the environmental statement it was the SARA analysis that was the best available to us --

MR. OKRENT: I really don't care what you use in your environmental statement. I'm asking about your best opinion technically on what the seismic contributions are.

MR. ACHARAYA: I'm trying to tell you that.

After the PRA analysis was complete, came an interim report from Livermore in which their 50 percentile of the medium was somewhat very close to the upper the 95 percentile of the hazard function that was for SARA. But, however, there was the substantial overlap of the Livermore hazard function of the upper percentile and the lower distributions with that of SARA.

Well, the point estimates of the seismically induced severe accidents, the frequency that we have chosen from SARA there were within about a factor of six from the SARA's upper estimates. Now, what that meant in the relation to the Livermore analysis may be -- the best values of the frequency that were used that might be about an order of magnitude or so lower than that of the Livermore study.

So our feeling now is that that is the best estimate frequency values for the seismically induced accidents that have been used in the risk calculations could be a factor of about an order of magnitude low compared to the Livermore's upper estimate. And that's not the only factor. The people who did our review presented analysis give us the impression -- that is we learned a couple of days before we came here, that the --

MR. OKRENT: The which analysis?

MR. ACHARAYA: Fragility -- could be non-conservative staff analysis by a factor of two. Now, taking these both into consideration, both the hazard of the fragility analysis, the best estimate frequencies of seismically induced accidents could be too low compared to the upper estimates by a factor of about 20.

So that is our current feeling that the best estimate values that we used here, in our analysis, could be considered up to a factor of two. And I could put this in a little more perspective as I go along.

In the APS analysis we have stated before we were thinking about this factor of 20. We were thinking we are in fact a factor of six from the SARA upper estimate. So in that context we have here analysis here, which as I see, that the public risk that is risk of early fatality of 11 cancer fatality --

No. Let me go back.

As it showed in some of the earlier the early fatality is dominated by the seismic events. The early fatality from seismic events is about factor of up to 30 higher compared to the internal events -- now, it would jackup the probability of the severe accidents induced by severe seismic by a factor of six. What that could mean that the early fatality that is portrayed could go up by a factor of six but it will -- uniform factor of six -- but now what I have said it is if however our best estimates values could be by a factor of 20 and the risk of early fatality could be increased by a factor of 20 also.

But the other health impacts, the seismic was making about equal contributions compared to the non-seismic accidents.

Now, that is not for the cancer but I think one can do arithmetic just applying a factor of 20 now to get the new risk.

Now, what that would mean is that there is also -- as to what would happen overall risk -- excuse me. Overall uncertainty -- without going through this uncertainty that we picked up later on the from the Livermore study. It appears we have -- but considering the seismic it is really not -- we picked up the seismic from the Indian Point analysis, and compared that with the Limerick analysis. The PRA's for these three plants we considered to be almost similar in quality.

Now, based upon the judgment from time to time -- we said that the risk factor portrayed here could not be exceeded by more than 40 on the high side and perhaps it would not be lower by a factor of lower by 400. So in the background of the conclusion of this type which is already documented in the -- from what we learned now, that is the factor of 20 -- the factor of ten in the hazard analysis and factor of two in the fragility analysis and -- the factor of 20 and that is covered in our statement regarding uncertainty in the IDS.

So this is what the uncertainty in the seismic analysis would do to the risk analysis that you are portrayed here.

MR. OKRENT: What would you now say is your estimate of contributions to core melt frequency from seismically induced events?

MR. ACHARAYA: Well, it could be -- some them, not all of them could be exceeded by 20 -- that's what I said.

MR. OKRENT: What number do you get for frequency if you don't --

MR. ACHARAYA: The best estimate.

MR. OKRENT: Best estimate. All right. Give me your best estimate frequency of core melt analysis in seismically induced events.

MR. ACHARAYA: This is what you have already in the PSR. That would be in the PSR the numbers that are multiplied --

MR. OKRENT: I'm trying to find out what the staff currently thinks is their best estimate. I'm trying to understand why there is this sort of ambiguity in your answer. At Indian Point in fact Sandler wrote a report in which he stated we come up with this difference with what the utility says on these and they gave an estimate, and this difference on fire and this difference on seismic and so forth.

Whether they were right or wrong, they came up with these values and in the end the staff I think to some extent accepted certain of these in their testimony. We are interested, and we indicated before the meeting, we would like to know what the staff's estimate is of the seismic contributions to core melt and risk, given whatever it is that you know and don't know.

MR. COFFMAN: Frank Coffman, from liability risk assessment branch.

I'm not too sure that I can speak for the staff but I can speak from some of the experience with this review on seismic and possibly address your question.

First of all, Brookhaven was not asked to calculate a seismic frequency number in their review of the external event SARA report in contrast with asking them to produce a reassessment as we did in the internal.

MR. OKRENT: Is there some reason, by the way, why they weren't asked to give you a number on external?

MR. COFFMAN: Yes, sir, it was primarily schedule.

MR. OKRENT: Well, I find that a very curious reason for something this important.

MR. COFFMAN: Well, I don't know that it would be fruitful for me to try and explain the ingredients that went into making that decision primarily, though the SARA report was submitted like two years after the internal events PRA was.

The NUREG-1068 summarizes the staff's position and the staff in essence says --

Maybe I shouldn't say the staff position.

It represents what was approved to be reported and that is that the means calculated within the SARA report seemed reasonable and that one should not use a single value but should represent the seismic contributions from seismic risk by a range.

MR. OKRENT: I would be willing for the staff to give me their 5 percent and 95 percent confidence range.

MR. COFFMAN: Do you want the numbers?

MR. OKRENT: I think that's harder to do, but I'm willing to take it.

MR. COFFMAN: The staff did not characterize them as 5 and 95 percent because I think that's sophistry to indicate that there is that level of knowledge.

MR. OKRENT: That's what I was trying to indicate a moment ago by my comment that it is harder to give a 5 or 95 percent in that area that's meaningful, but if you want to give me a range you have to tell me something about what its meaning is.

Let me give you an example from another area. A man comes up to you and says I want you to play this game. It is probably pretty safe. The risk of mortality could be as low as ten to the minus eight, but I have to admit it might be as large as .9.

This is a very wide range and if you just use geometric mean or whatever you want, put a log normal in, any distribution in you want, you will get, you know, a median, a mean somewhere far from .9, but I think you might find that unsatisfactory if all he told you was that -- what I -- the two limits I just gave you. I know I would.

MR. COFFMAN: Sir, I think to represent what I read as the staff's response to your hypothetical game is that the staff is saying, "That's not the best game in town. We don't understand all that goes into that game. So we are going to stick with something we know," because I would like to emphasize this was done in the context of licensing Limerick and it was a different perspective on Limerick.

So I think that gives a framework -- a background for then responding to your question. And I think you wanted to know what the mean and the range on the seismic core damage frequency?

MR. OKRENT: The staff's estimate --

MR. COFFMAN: I can read what the utility thinks. The staff has no independent estimate and the staff considers that the utility estimate was reasonable.

MR. ROSENTHAL: Was adequate.

May I make a few comments?

MR. OKRENT: Yes, you may.

MR. ROSENTHAL: We believe that the plant will survive the SSE, which is the licensing basis for the plant. It surely is appropriate to say how much seismic margin the plant has in this context. What we have told you is that with .15 GESSE the values of .4 or perhaps .6 the plant seems to have a lot of seismic margin.

We have further gone on to say that for seismic events of multiples of the design basis of the plant that should those events occur, then there would be early fatalities that the fraction of the seismic risk dominates total early risk.

MR. ROSENTHAL: Those things we have said and I think there is comfort in those items.

MR. OKRENT: I might note the term "multiples" is not a well defined -- because two is a multiple.

MR. ROSENTHAL: I think that earlier this morning we specifically used the value .4 which was the difference between regional and non-regional disasters. It is less than .4 that the internal events dominated the early fatalities and that for seismic events greater than .4 -- when you consider seismic events in excess of .4 then total early fatalities are a large contributions of seismic.

MR. OKRENT: Excuse me. If I can interrupt a minute, I believe all of that was in the context of the contributions of risk from seismic as presented in the Limerick PRA. It did not represent the separate staff assessment of the seismic.

MR. ROSENTHAL: But looking at the the tails of the distribution of the hazard curve that beyond .4 G aren't going to effect -- about the risk from less than .4 G events and that was in the tails of the distributions that you have the problems.

The next thing is if you have looked at the fragilities no single item were pointed out as the kind of items where you can fix just a few things in the plan you would increase the fragility of the whole plan. That might have been a cause for action.

Now, when you go beyond that in an argument -- we have a value of ten to the minus 21 in the NUREG-1068 for the low frequency range of seismic contributions to the class S. That's another way. What the review --

What Pete was saying was they just didn't know -- that's an absurd number to try to pull more meaning out of.

MR. OKRENT: Does it really say ten to the minus

21?

MR. ROSENTHAL: That's what is printed. But let me say that I'm -- I don't understand the issue that's at hand in the sense that one used the PRA to explore your risk of -- from seismic events, seismic events beyond the SSE. And there are some -- I've stated the conclusions.

Now, do you wish to explore in greater depth the risk for seismic events that are more than three times the SSE and where would that lead us?

MR. OKRENT: Look, we are supposed to be reviewing the PRA and seismic contributions as part of it. We also are supposed to be reviewing and this was called out in the letter just what the safety of this plant is with regard to seismic.

How well do we know it and are there things that either should be studied or considered in some way with regard to its seismic adequacy? Both of those were called out in the ACRS letter as things that needed to be reviewed in connection with the committee review for full power. We are just trying to develop the information.

Now, it seems to me one piece of information that's relevant. Certainly the applicant considered it relevant in doing his PRA, was what is the seismic contributions to core melt and what is it to risk?

And when I read what your reviwers wrote in the seismic area, and what our consultants have written and what Livermore has written -- from what I know -- I suspected that maybe the staff wouldn't in fact conclude that their best estimates was the same as that of the applicant. They would have some differing one and also they might have something to say about the questions which are not concerning the seismic -- Brookhaven raises that are not answered, really, in the 1068 document, I guess you call it.

So right now I'm sort of missing, I guess, some information that I had assumed the staff was going to supply to the subcommittee meeting.

MR. COFFMAN: It was not totally clear what we were going to be addressing until we had met with Leon Rider, Dr. Rider (Phonetic spelling), and he clarified -- he gave us a clarification, which was what we were working on and that is that we would come to give you and give you the information that Dr. Acharya had presented on the effect of seismic on consequences.

And that we didn't explicitly discuss at the meeting but we understood that the staff considered the applicant's estimates to be reasonable. The next step was what about the comments made by the consultants in NUREG-CR393?

Then the reply was as Dr. Acharya explained. And let me just briefly summarize, that if you look at the adjustments to the component fragility that were discussed, that has the effect of increasing the core damage frequency from seismic contributors by a factor of two. And that if you look at the contributions that the Livermore hazard function -- new estimate of hazard function which is draft -- would have that it was like a factor of six to ten, somewhere in there, that both of these were within the uncertainty range reported in 1068 and therefore we felt like that the conclusions in 1068 were not that sensitive to this new information, however, limited those conclusions were.

MR. OKRENT: I would suggest we take a ten minute break and we will reconvene on the subject.

(Recess taken.)

MR. OKRENT: We mentioned at the beginning of the day we would like to understand whether in the design of Limerick with regard to piping that is not seismically qualified and not seismically analyzed, even if it is not called class one, how the behavior of such piping and its possible effects on the course of events given a severe earthquake was treated and designed.

Let me go on and say, it is my understanding that frequently in many reactors, what is done is to postulate any single such line may break and we will see what happens.

I'm asking first, was that what was done for

Limerick or did you assume that, for example, all of the non-qualified seismic lines or many might break and pouring out more water or whatever it is that could accompany it? Which path did you follow, do you recall?

MR. BOYER: Repeat the question, please.

MR. OKRENT: There is a certain amount of piping that is not in class one and may not have been analyzed to see whether it can withstand the line-base earthquake with reasonable stress if it is not seismic as class one.

Ordinarily, it is my understanding, that in the design of a plant the staff asks, or for some reason the utility will have engineers postulate that a single pipe breaks and look at its effects then perhaps postulate another single pipe but now fixing the first one. Another single pipe brakes and looks at its effects and so forth rather than considering the possibility that there is an earthquake -- it is shaking all the pipes and that these are unknown pedigree and unknown capability to resist the earthquake might lead to rupture of more than one, and therefore that the utility might analyze this compound event. I am asking which is the practice followed?

MR. SCLUTHER: I guess we do have a number of non-safety related piping systems inside of safety related areas and, yes, we did postulate failure of those lines on a -- on a single failure basis throughout the plant. But in addition to that, we also evaluatd all non-safety related items to assure ourselves that they would not fail under the safe shutdown earthquake loading and cause damage to related equipment in the area of safety related equipment in the area.

That included maintaining pressure integrity of any liquid lines, non-safety related also. So that if we had a safe shutdown earthquake while there may be considerable damage to these lines, we did evaluate the effects of the earthquake load and did show that they would not fail.

MR. BOYER: Would not affect the safe shutdown?

MR. EBERSOLE: If they did not fail more than one at a time.

MR. BOYER: We examined more than one at time, right?

MR. SCLUTHER: We looked at all the non-safety related--

MR. BOYER: Assumed they could fail. All those in the area of a safety related piece of equipment. In other words --

MR. OKRENT: Well, that's a little bit of a hard answer to interpret in the following way: If you are looking at flooding effects, for example, flooding may well occur distant from the points of the break. And it might be that if a few of these pipes were to break you would have flooding effects that you are not designed for, whereas if one broke, it was included in the design.

I can't tell from your answer whether your look included that sort of what I'll call somewhat distant effect. You could also have environmental effects of high temperature from steam and so forth which are larger if you have more than one type in a region than a single one. Can you help me -- I know you said you looked to determine either that they did not fail or if they did fail they wouldn't hurt safety related equipment nearby.

MR. SCLUTHER: We did postulate a failure of the single leak, okay, just as you mentioned, but for all non-safety related piping and components in a safety related area we also assured ourselves that they would not fail in a manner to affect the safety relayed equipment in an area by looking at the -- for example, the safe shutdown loads on those components and verifying that they would retain at least their pressure boundary, okay, but we did not postulate multiple failures.

MR. OKRENT: Now, there is a slight difficulty that remains, which is the following: You certainly met the staff's deterministic requirements with regard to the SSC, and you met, I think, what is their requirement with regard to looking at one pipe at a time.

As we know, there are some differences of opinion about the frequency of the SSC, but in any event, the range that I suspect is predicted is between like ten to the minus three per year to ten to the minus four per year, roughly. Neither of those are very, very small frequency.

We wouldn't like to go up by one order of magnitude and have automatically not only severe damage to the plant but a large release, for example.

It is certainly not going up a factor from ten to ten to the minus three. So it is not so clear that we know enough about the status of what I'll now call seismic systems interactions,, including not only flooding or environmental effects that could arise from the pipes, but whether equipment mounted above the motor centers, key motor centers, redundant motor centers, you know, is not seismically qualified. And even though you have looked at the SSC, and it didn't reach ultimate, let's say, that twice the SSC, since we dcn't how much margin there was in your look, but if you were going to exceed code we could sort of -- Mr. Kennedy could estimate -- well, it will go up to point seven nine.

But since we don't know what the stresses were when you looked at SSC, right now it is sort of a position where it is hard to tell just what the seismic safety situation is in that regard.

I'm making an observation. I don't know whether there is more information. I assume there is not more information available today. I don't know whether you have more information back at the ranch, as it were, that bears on part of the question, like, for example, the piping. It may well be when you analyzed piping and you got numbers that are not close to ultimate, the loads were small, but --

MR. BOYER: There could be some more information we might have relative to that but it isn't present here with us.

MR. OKRENT: Well, do you want to add something?

MR. EBERSOLE: I have admired the high agree of compartmentalization at Limerick, but every time you buy compartmentalization you also have to fight failures-- (inaudible).

If you admit enough fluids or gases to those rooms you have structural failure of the structures. For instance-- (inaudible). You examined your designs to be sure that fluid releases into the several independent compartments you have got around the plant and, in fact, lead to curious structural damage and provide a coupling mechanism you didn't think was there.

MR. BOYER: I believe we have but I would have to verify that.

MR. EBERSOLE: It is the price you pay for compartmentalization. A lot of equipment that has been seismically tested, and I am talking mostly about relays -- rather not relays but devices such as position switches -- it has been found, and I cannot believe it after all these years, that they have been so tested against shatter and malfunction without the presence of the mechanical load near the set point.

As you know, everything that approches the set point is more nervous to go whenever it is going to go. And so we find, I think, maybe substantial amounts of equipment which if you now test with the actual fluid load or not at zero you find you have malfunctions that you never realized were there.

I think an examination of that needs to be made.

MR. BOYER: Well, I know there has been a lot of work done on that relay shatter type of thing.

MR. EBERSOLE: Relays would apply if you had a part potential on them near the trip point (inaudible). But a pressure switch, for instance, near the trip set point is ready to go anyway.

MR. OKRENT: Well, I suspect that we will want to talk about the seismic areas when we meet with the full committee and there will be the probabilistic kind of questions and what is the status of the comments of your reviews and so forth than there are these deterministic kinds.

MR. BOYER: Right.

MR. OKRENT: Are there other things you want to raise now?

Are there any comments from the consultants, any light you might want to try to shed or questions you might want to raise on the seismic hazard curve aspect?

MR. POMEROY: I would like to see if the staff can help me a little bit.

I think if I understood the gist of the comments that you don't feel that a factor of 20 puts a different perspective on the final result. And if that is a correct interpretation, would you care to comment on a factor of 100 or a factor of one thousand? Where is it that you do think there is some difference?

MR. ROSENTHAL: Mr. Pratt's presentation including table seven damage-state probabilities was handed to you. If you increase seismic by 20 I believe you increase the total core melt by 30 percent.

I would ask you --

MR. ACHARYA: Certainly, factors of uncertainty as 100 or one thousand will certainly make a difference. But the factor of up to 40 that could still make a difference as regards to compliance with some of the criterion that might be later.

One I have in mine is the safety goal that is being -- but as far as the public risk is concerned, the type of risk that we are portraying here increased by a factor of 40 that still will be very low compared to the non-nuclear background risk. And so that's where our picture is.

And as regards to this factor of 40, that may be picked up from the seismic. The correct state of the seismic analysis is such that all that is being seen here in the way of the PI and the role of seismic in this analysis that experts including some of the member of staff believe it is poorly -- as far as the public risk point of view.

We have so stated that in the -- the most significant earthquake damage anywhere within the vicinity of Limerick site be two to 300 years during which we have records are -- (inaudible) 50 millimeters away during an earthquake at Wilmington, Delaware in 1871 whose magnitude can be estimated to be less than five.

We certainly can not exclude from the reasonable assumptions no risk to the public resulting from earthquake induced damage at seismic (inaudible) during its operating life.

MR. OKRENT: Well, in your last answer it sort of seemed to me that at Limerick you are tending to discount, for some reason, the opinions obtained by Livermore from a panel of consultants whereas at other sites you haven't. And it is not completely clear to me what basis you have for discounting --

MR. ACHARYA: This is not what I have done. I'm not a seismologist, but at the same time it has been put in here by some of the staff who are seismic experts. Seismic analysis at Indian Point or Limerick (inaudible).

MR. OKRENT: Excuse me. In fact, as you may or may not know at Indian Point, for whatever reason, the staff chose to discount the USGS's most recent map of expected seismicity around the country, so I find sometimes what looks like a bit of an element of convenience --

I hope I'm not being overly harsh, but I find it hard to tell why, you know, at Sequoia they suggested the country -- they have to be re-evaluated, but when you are doing Indian Point PRA you can discount what USGS says. It is a little curious to me. That's all.

MR. POMEROY: I've another question with regard to 1068.

In 1068 the staff pointed out, I think, in a comparison of Indian Point PRA and this PRA, that the seismic hazard ended up just about the same. That violated the intuition of the reviewer that was writing in 1068. It violates my intuition.

I realize there is nobody here that can directly address that, but it does violate my intuition. But I wonder if you could clarify for me what happens when something like that happens? This is a good use. PRA's comparative evaluation between two different sites.

And I'm curious when the reviewer himself says that this violated intuition; is there any further response to that or is it just simply written down and we go on with what we are doing? Can you clarify that for me at all?

MR. COFFMAN: I'm not sure I can clarify it as much as I can place it in a category where you might be able to get a clarification.

That is that I think these questions -- your two questions so far and the questions concerning non-classification equipment -- non-category of equipment -- that we will simply have to get the appropriate staff reviewer here to address those.

The other aspect is that if one steps back in perspective to the more general conclusions that were made by the staff in 1068, then we do address it. I had planned to cover that, and the next item on the agenda --

MR. OKRENT: Well, it is not going to be the next item, although I know it is listed as such.

MR. COFFMAN: Would you like me to address it now, then?

MR. OKRENT: In you want to respond further to Mr. Pomeroy's question, please do, but let's not move into anything more general.

MR. COFFMAN: We will wait to address your specific question -- the questions you have stated with the appropriate staff reviewer.

MR. POMEROY: Perhaps you could clarify one further question that I have. We now are beginning to develop a suite of the external event PRA's. Is there an ongoing effort within the staff to look comparatively at the different PRA's?

MR. COFFMAN: Let me start from the general. I may have have to proceed down to get to something that is what you are looking for.

There is as a matter of course an item which is an action item on our director that annually he produce a report for the commission whereby we assimilate within the staff intelligence what we are learning from all PRA's, and so that is the routine effort that is going on.

But when it comes to specific seismic sites, comparing them on the plants, then again I would have -- I'm afraid I would have to defer to the staff expert.

MR. ROSENTHAL: There is a PRA reference manual NUREG 1050 that's progressing along. And a compliment of that is a document that's been produced by the division of engineering which addresses on a -- almost a philosophic level what should be done about seismic events beyond the design bases and what should be done with the uncertainty, and that's a big thick document comparable to 1050 that's slowly working its way through.

MR. EBERSOLE: I'll use a model that I once saw.

We have a seismic event, one will probably experience selective failures where the weakest thing fails first, and the order of failure becomes unfortunate. For instance, the condenser neck is fastened to the turbine exhaust with fabric or rubber like thing that's not all that strong it, shakes around and breaks and the turbine circ water pumps continue to run, and I have a prodigious flow of water into the turbine haul. I can't have --

You have already stopped me.

MR. BOYER: Go ahead.

MR. EBERSOLE: Well, anyway, I was going to say the reason you can't stop them is the trip devices for those non-safety grade trip breakers was non-seismic batteries because it was in the industrial set rather than the safety set.

MR. BOYER: Wait a minute. Trip devices are springs.

MR. EBERSOLE: But they are tripped by application of a DC trip signal which wouldn't come on account of the batteries were gone because that was not a safety design AC/DC system, it being in the switch yard and --

MR. BOYER: Could be.

MR. EBERSOLE: That was the picked scenario. You know, the little window I referred to.

MR. BOYER: Actually our basement is designed for flooding.

MR. EBERSOLE: It would not bother you anyway?

MR. BOYER: Right.

MR. EBERSOLE: There are cases where that hasn't been true.

MR. OKRENT: Mr. Rosenthal just gave me a note that we need to hear item number five on the agenda and I would like to call for that now, a little out of turn, with items three and four, but that's the way life is.

MR. ROSENTHAL: Dr. Kastenberg of UCLA and RDA will make a presentation on mitigation options that could be employed at a Mark II facility.

We viewed the studies as generic studies as part of an overall NRC agenda to look at mitigation features both by RES and NRR. The plant -- we needed a sample plant and in fact the plant, Mark II, looked at is Limerick.

MR. KASTENBERG: What I will try to present today along with Phil Hammond of RDA, is, as Jack said, some work on mitigation systems. It is part of a larger project and

I'll give you a little idea of what the larger project is.

(Slide 116 shown.)

MR. KASTENBERG: What I'll try to cover is basically the purpose and objectives of the study, the approach, the philosophy and assumptions that we are using in the study. I'll try to put up front what the principal findings are and discuss a little bit about the containment failure modes and then Phil Hammond will discuss the mitigation systems that we have come up with for Mark II containment and little bit on cost benefit and I'll finish with a little bit on uncertainty.

Basically what I will -- what my role is is to try to -- I hate to use the words for those of you familiar with something happening at UCLA -- I will try to bridge the gap between what Brookhaven presented this morning and what the RDA designers are doing in terms of it mitigation.

(Slide 117 shown.)

MR. KASTENBERG: Basically by mitigation we mean the following, those actions, devises or systems intended to reduce or ameliorate or remove the consequences to the public of a severe accident wherein by definition the core of the reactor has been degraded or has melted. And in practice what this means basically is try to prevent containment failure.

There is one area which is a little hazy in this definition of mitigation and that is for some of the accident sequences in fact containment fails before core melt. And we use the definition of keeping the containment from failing even though it might proceed to core melt itself.

The objective of the overall study --

(Slide 118 shown.)

MR. KASTENBERG: -- is three fold. One, to answer the question is mitigation technically feasible in these reactors? What would it cost? And then last but not least, what would benefits be so that somebody could conceivably do a cost benefit assessment to see if one wanted to add these mitigation systems.

(Slide 119 shown.)

MR. KASTENBERG: By cost benefit we mean some variation of one of the following three --

You do not have this viewgraph. I just pulled it this morning from a recent talk that I gave on value impact. But I want to bring it right here at the beginning.

What Phil Hammond will show you is a device we use to do some initial screening and we calculated some costs and we calculated from Brookhaven work some person/rem averted and we are using this ratio of cost per person/rem averted in terms of a screening or ranking of various mitigation devices.

There are some more sophisticated ways of doing value impact such as using the so-called net benefit method and these complex ratio methods. I believe you may have seen one earlier in the week with respect to GESSAR.

But for what we will show you today we will just be using this first ratio.

MR. EBERSOLE: When you are doing this do you have to strictly stay in the mitigate mode when you are looking at an improvement or can you step back and say, "Uh-huh. I should never have got that way in the first place and I will be back to the prevent mode as part of this exercise and reenforce the whatever it was, that say I never got there anyway"?

MR. KASTENBERG: It is a good point. If we had responsibility for the whole program of risk reduction we would do precisely what you are saying, but as the NRC's program is divided up into different pieces we are only looking at mitigation in this study and only one type of mitigation and that is system and not operator oriented.

MR. EBERSOLE: You are boxed in by administration?

MR. KASTENBERG: This doesn't prevent us from thinking in more general terms, right.

MR. EBERSOLE: I know the problem.

MR. KASTENBERG: Let me mention the approach, just the general approach for the whole program.

(Slide 120 shown.)

MR. KASTENBERG: We are about halfway through the program right now, three fifths of the way through the whole program, basically to survey containments and look at how they might fail in severe accident. That part has been completed to survey mitigation technology. That part is basically complete to design specific systems for three different plants.

And we are just about complete with this part, and the three types of plants that we are looking at were Mark II containments, Mark III containments, and the advanced Westinghouse large dry containment.

We are now getting into the last two aspects of this program to develop cost benefit assessment procedures and then perhaps to explore other types of benefits in this benefits and to outline how the NRC might implement these in decision-making.

(Slide 121 shown.)

MR. KASTENBERG: That's gives you an idea what we show you today fits into the overall program.

The philosophy and assumptions are basically the following four: That is, mitigation should be complete. I'll show you mathematically, hopefully, why we make this argument, but from a physical point of view, basically what we are finding in going through these studies is that you have various threats to containment and in many instances if you do something to prevent one threat to the containment then one of the other ones become dominant.

And until you keep eliminating them to the point where you have exhausted all the available funds in terms of cost benefit assessment, you basically get down to where there is nothing left to mitigate against. You wouldn't want to do it and we will give you some good examples of this a little later on.

Secondly, accident phenomenon must reach a determinant end state, that is, again, just because you have mitigated against an accident by improving containment, that accident is still progressing, getting back to the comment that you made just before.

And unless the operator does something or you can then say than the accident is ended it is still there, it is still progressing. So we tried to design a mitigation system so that you know what the end state is.

We are working with the assumption that operator action is not available and, again, we are aware of the fact that other people are looking at this and in the end what one would want to do is trade off systems versus operator actions. But in this project, we are not looking at what the operator could do to intervene in an accident.

Last but not least we are trying to design all of these under the assumption of electric power is not available. That is, the normal electric power is not available.

(Slide 122 shown.)

MR. KASTENBERG: Now, before we get into some of the assessment and some of the design that we have come up with, I want to go back to the three questions that I raised because I think it is important as we go through this to see -- give you the conclusions up front, so to speak. Everybody seems to be doing that and we will do that also.

I think what you will find is that the answer to the first question is yes, that mitigation strictly with systems is technically feasible, that is, you will find that we know how to design systems to cope with various accidents. We know how to improve the containment, we think, to cope with the environment that would be in the containment.

They think you can build it and test and that these things would work when called upon to work. We think that with good engineering practice in fact you can cost these things out. You may be off by a factor of 2, perhaps, but there are engineering costing procedures and in fact you can cost these systems out and Phil will show you what some of these systems cost.

Those of you who have been sitting through two-and-a-half days of risk assessment recognize when you get down to this it is very, very difficult to determine just what those benefits would be.

I thought Pete said it very, very well at the end of Bob Henry's talk when he said, "I've heard best estimate. I've heard conservative. I've heard engineering judgment. I have heard assumption," and so. When you look at the spectrum of PRA's that have been done and someone asks you to give -- quantify what the benefits are, it is very, very, very difficult to do that.

So that's really the bottom line so far of our study. The first two the answer is yes, the last one we are in a difficult situation.

MR. DAVIES: Bill, in your previous slide you said you were going to assume no electric power available. And I am not sure what that means. Are you talking about off-site power, DC power?

MR. KASTENBERG: Talking about both off-site and on-site power and that for some of the systems that we are looking at, we are going to show you that you would might want to add a dedicated diesel for example for that particular mitigation system. We are not going to rely on anything in the plant as it is constructed.

MR. DAVIES: I just have a quick problem with that.

Some action sequence in fact the most risk dominant have nothing to do with loss of off-site power. You would have it available and you might be unduly penalizing yourself --

MR. KASTENBERG: That's true.

MR. EBERSOLE: In order to say you are engaged in frequently the mitigating mode at what physical state or point of beginning into damage, because you haven't prevented it, where do you start? Have you already melted the core?

MR. KASTENBERG: In all of these cases we are looking at the point where the core has melted and in most cases you have penetrated the vessel, just as a frame of reference.

MR. EBERSOLE: They gave you a hard job.

MR. KASTENBERG: Although I made a rather strong statement that you can design these systems, you can cost them out, but it is difficult to determine what the benefit would be. Nonetheless, to try to screen a number of mitigation options and to try to rank them, we did go ahead and try to estimate benefits. And we did try to estimate the benefits using the Brookhaven review and in this case, the Brookhaven review for Limerick, and to give you an idea where the numbers come from --

(Slide 123 shown.)

MR. KASTENBERG: -- and to show basically how we arrived at benefit -- I'll just bore you with this standard equation -- we get the frequency of the various containment class from the Brookhaven review. We get the conditional probabilities for the containment failure modes. And we get the consequences of interest, in this case person/rem, all of these from Brookhaven and then if you take these double sums you can get the risk for each consequence of interest and for us, as I mentioned for the screening procedure, we used man/rem --

(Slide 124 shown.)

MR. KASTENBERG: -- basically to get a feeling for -- to get a feeling for the risk reduction, hence the benefit. We are trying to eliminate some of the various -- we are trying to eliminate some of the various containment failure modes. And for complete mitigation basically what you are doing is eliminating all of the P's to where the only P that would be left would be the P of no failure.

And the reason that you do that, of course going back one viewgraph in your packet, is that the sum of the conditional probabilities of containment failure for each containment failure class have to equal one.

So if you are going to eliminate all failure modes the only P that's left is the P for no failure.

Now, in practice we don't actually do that. For some of the containment failure modes the cost benefit is so small that you wouldn't advocate trying to build a device to protect against that.

MR. OKRENT: The cost benefit is so large?

MR. KASTENBERG: So small.

MR. OKRENT: Let it go. Forget I said that.

MR. KASTENBERG: Then just if one wanted to try to reproduce our results, I did include two tables. I don't want to go through them in great detail but I did want to include the tables.

These are -- these tables from -- basically from the Brookhaven report, review of the Limerick PRA, and the class frequencies are on the top of each column and then the conditional probability of each containment failure mode for those containment failure classes are shown underneath them. And as I mentioned before each one of these has to sum to one.

And again this concept of complete mitigation with all of the uncertainties, if you mitigated against one of these, for example, this probability would just shift to one of the other P sub I's. Again within the context of cost effectiveness, if you prevented against, say, gamma mode it would only show up that -- this probability would have to be apportioned amongst the other containment failure modes.

(Slide 125 shown.)

MR. KASTENBERG: Last but not least, just to set the stage for the systems that we would be looking at and what their benefit might be, these are the consequences again from the Brookhaven review, and for the numbers that we will show you we are using the man/rem out to 50 miles, person/rem out to 50 miles.

(Slide 126 shown.)

MR. KASTENBERG: As a preliminary to the actual design, we had to try to create a sequence of matrixes, such as this one, which shows you what the contributions to risk would be for the various containment failure modes because you want to look and see where you want to start and design. And this is an example of the process that we went through.

And then what will happen is certain things will jump out at you as more important ones to consider as it has been brought up both by PECO and by Brookhaven. It is the most -- largest contributions to risk in terms of population are the class one sequence overpressurization failures and that's where one would want to start the design.

For the screening process people at RDA, what they did was went ahead and did a number of designs and then as each feature in the design eliminates the containment failure mode they would take credit for that man/rem and make ratios of dollars per man/rem and use that for their initial rank and that's basically how we went ahead and did this study.

(Slide 127 shown.)

MR. KASTENBERG: Accident end states that we started with were the following: We wanted to look at mitigating against steam generation, against in-vessel hydrogen generation, against containment concrete decomposition, ex-vessel steam pressurization, ex-vessel steam explosions, ex-vessel hydrogen generation and residual heat load.

After listening to the presentations this morning by Brookhaven and by PECO, some of these might jump out. Why did you even bother when the consensus is that the probability is small?

Again, for a first cut, you want to take a look at all of these and either want to eliminate those because the probabilities are so small or because the cost benefit is just not there.

I'm going to turn the floor over to Phil and Phil will show you some of the systems that have been designed to cope with these various threats to containment and then show you what the cost benefit might look like in terms of the Brookhaven numbers for systems to cope with these accident incidents.

MR. HAMMOND: There are different objectives in these studies than in most of those that you have been exposed to, namely, trying to determine what the actual risks at the end of a accident. We are trying to find out whether there is a defensible way to make a mitigation system which has to be defended against all the uncertainties and probabilities and other ways of handling severe accidents.

So we have indeed taken a very conservative approach. We wanted to eliminate all of the uncertainties because when you are making a policy decision that is mitigation worthwhile you can't have it subject to attack because you didn't study this method of phenomena well enough.

So what we have done in most cases is to force the uncertain phenomenon into a given path, place where the core might melt and spread all over the membrane or some other thing might happen or the droplets may fall into the water in the right sides or they may not. We have introduced extra systems at high cost to force it to go in a way that we can understand.

Now, that doesn't mean that there shouldn't be further efforts to understand these modes, these phenomena. It's just that we had to take an assumption in order to get an answer in a reasonable time. So we have taken what you might call the brute force approach. Wherever there is uncertainty in the phenomenon we forced it to go in an understandable way.

There are different accident pathways, as you are well aware, but it is quite clear that when -- once a severe accident has gotten underway and all the normal safety systems have failed, the core is about to melt, and there is no AC power, and you can't count on any operators being there, or doing the right things, things that end up in that system begin to converge. There really aren't as many ways in which the failed core can end up. There are separate ways we have forced it to go to a known way.

The end states represented here are a rather complete list and you will see that what these really act to be -- have to be studied, functions that had to be performed by mitigation. We had to remove heat, residual heat from the core material. We had to control where the core melt residue ends up and we have to be able to handle overpressure of the containment.

Well, these many pathways begin to converge into a few end states but we have no way of determining which end state is the one we have to handle in any given day, so our system has to handle the envelope of these end states.

That's rather an extreme assumption especially as there is no power around and no operators. But the object here again I must say not to recommend the best fix for Limerick or any other particular plant. We are trying to test the consequences of a policy of adopting mitigation as a way of handling severe accidents.

So I just want you to realize there is lots of cheaper and easier ways to do what we have assumed here, but reality is that indeed the operator could make certain actions if he were properly trained.

We have no data available to us now that shows he is properly trained. We are not even sure what he should do. At Three Mile Island he would have been better off if he had put on his hat and gone home. There are certainly ways to use existing equipment that's now within the plant for venting and for other functions.

You are talking about installing a new system to do it on top of what is there. And I realize that's unnecessarily expensive but we have no way to split the difference.

So we are setting the upper bracket and forcing phenomena into the known path. Costing methods we are using are conventional. We are not using safety grade equipment for mitigation. This was by assignment from the NRC.

We don't feel when you are talking about the far end of the tail that the residual risk after the design basis accident that is necessary to use safety grade equipment. We are using standard industrial high grade industrial equipment. The installation method, there is no paper trail so that the costs are not as high as they might be otherwise.

We have used very generous 1984 construction costs, however, based on consultants and data that's available to us. For instance, the cost per man day for a worker in the mitigation installation would be one thousand twenty dollars a day including overheads and insurance and all our costs and the concrete basic 600 dollars a yard except when put on very special ways.

MR. OKRENT: Sounds like it is going to be built by consultants.

MR. HAMMOND: Well, see, most of the equipment is standard equipment, pumps, pipes, filters, which are easy to get -- pick up the phone and get a quote on. That's what we did.

We also costed in three ways depending on the status of the plant, one where the plant is still on the drawing board so it costs very little to add a pit there or pump there or pipe there. Second one is the plant is already pretty much finished but not contaminated. You would have to modify it but you wouldn't have to shut it down. And the third way is retrofit.

I will have to say for Limerick the costs -- the costs represent the retrofit mode. However, including after it is radioactive -- we did not include a cost for replacement power and the reason is we think it can be timed in the normal shutdowns. If it wasn't done with the crash basis it could be done in the refueling of shutdowns. But that remains to be seen.

So there are some better ways probably of accomplishing what we are showing here but we wanted to remove uncertainties wherever we could and that's one step that we took for that purpose.

Now, our findings are, as Bill said, that it is indeed technically feasible and that the operation equipment can be made essentially passive such that it operates because it is there and it functions because it is there, not depending on a very complex system and the cost is determinable.

I'm going to come back to the benefits a little bit later.

Three functions we have to cover are heat removal, core control and venting.

(Slide 128 shown.)

MR. HAMMOND: This represents the dedicated heat removal system for the pool and the sprays. There is a spray system and heat removal system in the pool. It gets cooling water from an external on-site source, whatever the heat source is.

Take it through heat exchanger and return it to the heat dump. The diesel engines are non-electric. They are started by a pressure signal and don't use any electric power. The pump starts -- you realize that when the -- heat exchanger there is a signal that comes off the fact that there is pressure in this line that opens the isolation valves both inside and outside the containment so that the isolation is maintained all the time until this pump starts and then the pipes are opened.

Similarly, the discharge from the containment water, the pool water is cooling in this heat exchanger and then goes into the wetwell spray area. So we have cooling containment by spraying and heat exchanger from the pool.

(Slide 129 shown.)

MR. HAMMOND: That shows the sprays from the pumps and they are installed with the values on the outside and a spray nozzle on the inside. There is a minimum of interference with the internal structure.

I wish I had time to go more into the details on that but it isn't really worth it.

(Slide 130 shown.)

MR. HAMMOND: Now into controlling the core, we made quite a bit of study of the core spreading out on the membrane and leaking through all these valves and fuses.

MR. BOYER: Is that a header on those sprays or are they individual spray nozzles?

MR. HAMMOND: There is a header.

MR. BOYER: Containment, you are probably aware, has a number of points secured to it running up and down along the walls of the containment which would interfere with the header. It would be not impossible. You wouldn't have it along the walls anyway?

MR. HAMMOND: It is schematic. I'm not sure that's where the headers would be.

In forcing the core to go to a known position, we looked at spreading it out on the membrane and decided that all of this is a novel idea and it may work and it might not work, and it is still a matter of strong discussion as you have heard here today.

So we have ruled out the membrane as a cooling thing and have taken two other approaches, either one of which would work well.

In this case we have plugged up the openings in the pedestal up above there and put in the necessary bevels to ensure that when the vessel breaks loose, the material does come down. And where he was talking about the drain melting through, well, we put in our fusible plug, you might say, so that indeed, in very short order that material would come down into this pit area here.

And then we don't want a steam explosion at that point, although some would say it wouldn't happen, we couldn't live with the uncertainty, so we have excluded the water from this area except for a couple of feet.

And we have water wall tubes installed around here so that these walls will not be attacked, and we have a pebble bed that is barely covered with water. The material falls on there and at that time this opening is fused by the heat that's present and the -- then it slowly floods in.

That's one method in which we use the room under the pedestal there for a core catchment.

MR. GARCIA: One question related to water wall tubes. What kind of materials --

MR. HAMMOND: Steel.

MR. GARCIA: What kind of --

MR. HAMMOND: As long as they have water inside they won't be attacked, because they are acting like a rising film of --

MR. BOYER: Acting like a boiler tube?

MR. GARCIA: From the pool itself.

MR. HAMMOND: From the pool itself. They just come squirting up there.

MR. GARCIA: Thank you.

MR. HAMMOND: That's just a temporary mode anyway, to prevent spreading out. It will soon be quenched once the water gets up there. It is a transient problem you have there.

Indeed that whole thing needs further study. It has a few uncertainties, but it is quite cheap and probably defensible.

(Slide 131 shown.)

MR. HAMMOND: We do have a more wool-plated version in which you would install a dry crucible with a water jacket on it below the base mat. And it is coupled with the pool water, and the pumps are now down here because they are not easy to put them out of the way here.

In this case the pedestal area is kept dry. And these are seals, fusible seals that prevent water from being in this any time until the heavy core material lands on it, displaces the water, and then it melts through and dumps it onto the next seal, and then the water comes through and floats up. And then that melts, and pretty soon the core material gets down to here, where it is capable of being cooled indefinitely, and actually well out of the way.

This system has a very nice feature: It is very easy to clean up and recover from the accident.

But this one costs more unless you have a new plant for backfitting.

Well, in the case of the conventional methods of mitigation, we also have to have another system, that I don't have a sketch for, but it is a standard. When the ATWS event occurs, there is steam filling the building that will quickly fail the containment. But that's not contaminated. The core is not melted yet.

So we would propose a very large steam vent, a reclosing relief valve that would vent off this high pressure steam and then reclose permanently when the ATWS event is over. The core boils dry and the ATWS event quits. Then the system would then switch to a filter, much slower flow filter, so there would be relief of slow overpressure through a gravel -- very large gravel vent filter.

Now, then, in the course of doing that we discovered another option --

(Slide 132 shown.)

MR. HAMMOND: -- that combined some of these features. I should say when we have the ATWS venting system, we also have to have a subpressure relief valve so that when you turn on the sprays the pressure goes negative, and you can't let the condenser fail from underpressure either. So it gets pretty complicated.

In looking at that we decided it might be better just to keep the containment always at zero pressure.

Build a big filter that's essentially vented to the atmosphere at all times, connect it to the pool, and let the ATWS steam go out that way and feed back that way, and let the -- all the overpressure events essentially not be overpressure events because there is no way to put a pressure on a container. It is always connected to the atmosphere.

This means building a filter larger than anyone else has considered except possibly the Swedish one. Indeed, we have looked carefully at the experimental work they did. We have also looked at the work that went into the filters that are present on the Savannah River evaporators -- reactors -- and I personally worked with condensers and filters in krypton and xenon when I was at Los Alamos.

MR. BOYER: Is this non-safety grade?

MR. HAMMOND: Yes, this would be non-safety grade. This is just a gravel filter with a charcoal filter on top of it. No paper filters.

But it is very large, something like eight feet in diameter, the duct.

MR. EBERSOLE: Is it a wet filter?

MR. HAMMOND: Wet filter, yes.

MR. DAVIES: Does it get the gases, or do they get released?

MR. HAMMOND: No. In the case of this continuously vented system -- talking about two filters. One which has a sort of pop valve on it. The overpressure releases it into a filter. That one does not -- hold the heavy gases. They would go out.

In this one we would propose to keep this whole filter chilled to minus 80 Fahrenheit at all times until the accident occurs. It takes a 60 horsepower motor to do that, but once the accident occurs there is so much stored coal there it is passive. You don't have to have power during the accident.

MR. EBERSOLE: Is it chemically treated, the fluid.

MR. HAMMOND: The charcoal has different grades that are chemically treated, but the gravel is just gravel.

MR. EBERSOLE: With water.

MR. HAMMOND: Dry gravel. It is chilled to minus 80.

MR. EBERSOLE: I thought it was wet.

MR. HAMMOND: It gets wet when the event occurs, but only part way up and we have carefully -- we graded the rock so that we end up with still dry charcoal and not even krypton-xenon gets out. If it doesn't warm up you have to have power within two or three days or it will begin to warm up. If your don't get power in two or three days you would have to seal off these openings here and force it to warm up back into the building.

This is a very quick runthrough and I realize that it is going to stir up more questions then I have time to give answers, but I better get to what the --

(Slide 133 shown.)

MR. HAMMOND: What we end up with is kind of a menu of mitigation systems. These having to do with heat removal, dedicated surface-type cooling that I showed you, the one that's underground for cooling the pool. Drywell sprays, external feed or internal feed, that's the two different versions I showed you.

Core control either by the base plant gravel bed or the dry crucible underneath. Pressure control by -- in every case we have assumed the 3A fix is already there and yet that still leaves more than 1 percent residual total risk from ATWS. And since we are assuming we are going to take care of everything over 1 percent residual risk we still had to put in the vents. If we could get a 4A version to get that down we could leave all that out and save quite a bit of money. This ATWS clean steam vent represents a fair amount of money.

Then the filter vent is shown here. In some cases we showed a large hydrogen recombiner because although the system is normally inerted, after you have vented out for an ATWS and sucked air back in, you have a flammable possibility. So we had to consider that possibility. We don't think it is cost effective.

Then there is the large vacuum breaker and those -- now these different options represent different combinations of these choices. And the one that I'll spend more time is number I here, which represents almost complete mitigation. That doesn't mean one necessarily would do it that way, but we wanted to analyze that case.

(Slide 134 shown.)

MR. HAMMOND: Here it is again compared with the low pressure system that I mentioned where we used a filter.

Now, these costs are in the thousands of dollars and I would emphasize they are, we think, conservative but they do not include replacement power costs nor any multiplier because it happens to be at a nuclear plant. It is based on industrial grade equipment.

The interesting thing is that their benefits -- since we are essentially taking care of all of the contingent risk, residual risk, comes out for the low pressure system at about $400.00 per man/rem and 230 per man/rem and for the 500 mile radius, where the conventional system where you have high pressure containment that's vented when you have an accident is slightly higher.

I don't think these differences are really significant but at least it shows that it's certainly worth studying this low pressure system a little further.

It has another set of benefits very hard to quantify at that point because with this system, low pressure open containment you don't need to run the plant inerted -- which tell me how much it is, some kind of operation benefit for running the plant not inerted as compared to running it inerted. And we did not take that into account.

So coming back to the last question --

(Slide 135 shown.)

MR. HAMMOND: -- we think it is possible to design systems to handle the full envelope of events from a severe accident and we think we can get a fair handle on their costs. We think it is possible to do. And that the benefits are, as Bill says, somewhat uncertain, but in the direction of a reasonable number.

I wanted to tell you what these numbers actually are, in case you can't read them.

(Slide 136 shown.)

MR. HAMMOND: It looks like low pressure system the total cost is about $10,000,000 for this venting filter system, including the core catcher.

And 13 for the conventional method.

Any questions?

MR. ROSENTHAL: I would.

MR. OKRENT: Go ahead, Jack --

MR. HAMMOND: We want to come back and talk about the uncertainties a little more.

MR. ROSENTHAL: One need not do the whole list. For instance, you could take heat removal and coolant spread, top two items, at 2.7 million and eight hundred some odd thousand and get a considerable benefit. The way the work has been done it is like a Chinese menu, you can pick and choose. Mitigation philosophy that one can settle for less than that.

This is a mitigation study by contract, because we were told that we should spend proper emphasis on mitigation. There is plenty of other work on prevention going on.

Next, I sat at some ACRS meeting and we discussed filter vents on large dry containments. And we were criticized, one, for not having good costs estimates. And two, for over estimating the costs. So we have tried -- we have asked our contractors to come up with costs without the stack of QA and EQ and paper trail to try to minimize those costs. The costs will go up if you want those items. We recognize that. But we are trying to give a favorable ratio.

The other thing was that we have not in the past been able to actually have traceable cost numbers and this contract now gives us the facts, cubic yards of concrete, estimates of what it would cost and so puts us in a much more credible position to discuss potential costs than the staff has been over the last few years.

MR. HAMMOND: Our report has detailed breakdowns on the cost of each of these items including the time it takes to install it. We want to talk about these --

MR. GARCIA: Since earthquakes provide at least 10 percent of the risk of any of the PRA's that have recently been done has any consideration been given to the additional cost that would be incurred to make these mitigation systems in category one type structure?

MR. HAMMOND: Yes -- two parts to the answer. The answer is yes, we have assumed these have to be at least up to the earthquake level of the rest of the plant, it has to survive. And I don't think that's really a passive system. There is really not that much of a problem.

Second part of it is we have not included very much of the benefits from external events.

Is that right?

We don't have very much benefit from external events so the benefits available, if we did include those, would go up. In other words, our dollars per man/rem would go way down.

MR. BOYER: I don't get what you said. Benefits from external events?

MR. HAMMOND: We are taking about -- we are studying the benefits in terms of man/rem. We included external events there would be more man/rems because there would be higher probability of accidents. We left those out.

MR. KASTENBERG: I guess I was going to talk a little bit about uncertainty and also give a summary, but a summary has been given nicely by Jack. But I did want to emphasize a few other things before we leave this viewgraph.

One thing I wanted to emphasize is that some of the designs that you saw, some of the pictures are really conceptual. We showed this one other place. We showed that large tower and everyone said that will fall down in an earthquake. You could put that underground. It doesn't mean the filter system has to be standing there. It is just a visual picture.

Secondly, obviously the plant as built has pool cooling and has containment sprays and one of the features that Phil and Jim duly came up with, which I think is unique, is that they have this idea of the direct drive diesel. There is no electricity involved. Those direct drive diesels are directly driving the pumps. And that's a different focus than what you have in the plant as it is.

In some of the other instances if you were to look at the report you will see that throughout the report. Many of the functions that we work with are functions that are in the plant but these are to work in a severe accident when those functions do not work.

I wanted to emphasize that the pictures are schematic. They give you an idea what the functions are supposed to do.

I wanted to mention on the uncertainty in the latest stages of this project we are to look at developing value impact measures and you heard much today on uncertainties with regards to phenomenon. But I have found and others have found that in trying to do the value impact there are other uncertainties as well which make the cost benefit analysis difficult to do.

One of the ones that we are concerned with in trying to show this bottom line result, for example, is the fact that in the code that everybody uses, the CRAC code, there is an assumption in there among other assumptions which drives the person/rem. And that has to do with interdiction.

Interdiction itself is a mitigation strategy and it is assumed that all of these codes that you will interdict land if the whole body dose to a person moving back on that land after evacuation is 25 rem or less. We don't know that's what the number would be, yet that is a mitigation philosophy. And it is already built in to the person/rem that's shown on these tables.

If you change that parameter in the code and made it 50 rem, the man/rem would go up, the interdiction costs would go down.

If the NRC or the public demanded that you interdicted down to 5 rem, the interdiction costs would go way up and the man/rem would go way down. So somehow we have to get that in to -- any value impact study of mitigation has to deal with all the other factors in the computer codes and that's one of the reasons why trying to come up with a cost benefit description for someone is so difficult.

It is not only the phenomena that bring in the uncertainties as we have heard today, but many, many other parameters that are in the code, and so the kind of analysis that we are doing in terms of trying to arrive at a decision of whether to go or no-go and do more work is only one input into whatever the decision-making process would be on any of these plants. I wanted to make that clear right from the beginning.

MR. OKRENT: Questions or comments on this entire presentation?

When did you say the reports are to be submitted?

MR. KASTENBERG: Let's see, on the first three parts of the project, the first two reports are in draft form and the NRC has them. That's just a summary of containment and containment failure modes and then a summary of mitigation features, a second draft report that the NRC has.

They are writing the third report which includes the Limerick work, but we sent the NRC the chapter on Limerick. And I had understood that the staff had sent the ACRS that chapter on Limerick.

MR. HAMMOND: I think you already have this.

MR. OKRENT: I'll have to check. I don't remember having seen it, but the ACRS may have.

Dr. Savio doesn't remember either, but if you saw his office --

MR. ROSENTHAL: We have provided the RDA reports under FOIA to various people, as a courtesy to the ACSR without FOIA's and we have also provided them to interveners in the hearing board. If you would put up your slide where you have this list of mitigation features and cost I would like to point out that the costs do not include discounting.

(Slide 137 shown.)

MR. BOYER: What do you mean by discounting there?

MR. HAMMOND: They don't include any discounting over the life of the plant. They are a one-time investment.

We don't discount the risks either.

MR. ROSENTHAL: That the person/rem reverted over the 30 or 40 year life is not discounted, but is just an amount times $1,000 and summed over the life of the plant. And the last item is that these values are based on the work in NUREG-CR3020A which is the methodology that was described earlier this morning.

MR. OKRENT: Let's see. Did you say where you stand on your other two containment types?

MR. KASTENBERG: Well, the Mark III is represented by the GESSAR PRA. The work is completed and Phil told me this morning that the figures are being drawn and the report would be complete within a month.

MR. OKRENT: You said an advanced containment?

MR. KASTENBERG: We are looking conceptually at the Westinghouse advance reactor. But there is no PRA for that and we really don't have the full design on it. So it is more of a qualitative design.

MR. HAMMOND: We won't be able to get very much benefit, because there is no risk assessments. But we will be able to assume something that showed what the mitigation equipment was.

MR. EBERSOLE: I can see now the advantage of being constrained to mitigate, at least you develop ideas.

MR. HAMMOND: I think it is important to realize that in any real case you use a combination of prevention, operator action, and mitigation.

MR. EBERSOLE: Is the scope of it just to the Limerick-type design, this study of yours?

MR. HAMMOND: It's all the Mark II's.

MR. EBERSOLE: What about the PWR's?

MR. HAMMOND: Yes. The first tasks we surveyed all the types of containments there are and indicated what type of system you would have to have, what functions, but we did not detail designs. So that's all five there are in the country.

In the second set we surveyed all the kinds of mitigation equipment that has ever been dreamed up by somebody.

And the third one we got the three specific plants, one of which is PWR.

MR. OKRENT: Any other comments on this now?

Thank you. We will move on with the agenda.

We have not yet done items three, four or six. And I suspect that if we devoted not more than ten minutes to each of items three, four and six we can get in the most important comments.

MR. BOYER: We can do ours in ten minutes.

MR. OKRENT: Why don't we try, then, ten minutes on item four and ten minutes on item six and we will then let the NRC go out in a blaze of glory on item three.

MR. DAEBELER: What I've indicated in the charts --

(Slide 138 shown.)

MR. DAEBLELER: -- is conclusions and insights gained from performing the probabilistic risk assessment and the severe accident risk assessments for Limerick. We have given some of this previously and I'll just hit a couple of highlights.

I oriented towards three areas including a brief review of the overall results, some plant specific conclusions that we can draw by looking at the sequences, and some programatic insights that were gained.

Briefly skipping a chart and going to the next one just to give a -- again using care that there is some differences between methods, et cetera. But looking at core damage frequency and comparing the point estimates of some other facilities with the overall results of Limerick with the point estimate, upper estimate, and lower estimate as compared to the safety goal.

(Slide 139 shown.)

MR. DAEBLELER: From this we can conclude that the estimated core damage frequency at Limerick is below the safety goal and similar to other PRA's.

(Slide 140 shown.)

MR. DAEBLELER: Skipping the next slide and going to early fatality risk, here in looking at the total man-caused risk and total natural risk hazard surrounding Limerick and comparing that to the Limerick results as well as WASH-1400. Joining the upper and lower estimates again, giving a boundary to the results.

(Slide 141 shown.)

MR. DAEBLELER: I'm not sure this one we showed exactly this way previously, but it gives an idea on risk in terms of the annual individual risk. Again looking at U.S. averages, safety goal, and the upper and lower estimates for both the latent cancer fatalities and the early fatalities. And here we see factors, as an example here, of seven and in the case of the early fatalities 200. In the case of the latent fatalities of the upper estimates below the safety goal.

From this, then, we make the following conclusion: That the risk due to the operation of Limerick is much less than other risks. It is less than proposed safety goal and comparable to reactor safety study. And we do believe that Limerick does not represent a disproportionate risk to the public.

(Slide 142 shown.)

MR. DAEBLELER: The study was performed with the purpose of estimating potential risk contribution to the public due to Limerick operation. It was also to respond to an NRC request. It also was done in order to comply with NEPA requirements for the environmental reports. The results are that the risk is less than the proposed safety goal and comparable to reactor safety study, as I said. And thus we believe again that the PRA/SARA results verify the adequacy of the design of the Limerick plant.

(Slide 143 shown.)

MR. DAEBLELER: Let me briefly show this.

We begin now to get into some plant specific areas and look at the contributions of the internal and external events. Here we see the internal events, seismic and fires. And we can see the relative contributions of those to core melt frequency. And again we see the large uncertainty in the seismic area discussed previously.

(Slide 144 shown.)

MR. DAEBLELER: The dominant core damage sequences. Here we see that they come from loss of off-site power contributing on a point estimate basis approximately 25 percent of the total.

A loss of feedwater, TQUX being about 15 percent and then we have the seismic loss of off-site power. Again these are points estimates in the relative contributions and we can see that no one sequence clearly overwhelmingly dominates.

(Slide 145 shown.)

MR. DAEBLELER: In conclusion then on core damage frequencies and relative importance, we find they were dominated by internal events. Earthquake and fires are lesser contributors. No single sequence dominates the core damage frequency. That a reduction in the frequency of that sequence would cause a substantial reduction in the core damage frequency. After review of the sequences that no single system or function is so important that reduction in its likelihood or failure would cause substantial reduction in core damage frequencies.

MR. OKRENT: Would you remind me: Except for the seismic part, which we have talked about, with regard to systems interactions would say you have done a study that's comparable in depth to what Indian Point has done recently or the review LER's did which were applicable or something like the one where they did sort of what I would call a mini-Indian Point kind of review? They did a walk down but only a small fraction of the effort of Indian Point.

MR. DAEBLELER: First, I might say in the performance of the PRA some system interaction work was conducted. As far as specific tasks directed at that, that was not performed but we have done some other things.

MR. BOYER: I think the sum of the various parts that we did is about equivalent to what Indian Point had done. I would have to do some varification of that, but that's my impression, that that is the situation.

MR. OKRENT: All right. Well, for when next we meet, why don't you see -- take a better look at what they have done recently because they have done a substantial amount.

MR. EBERSOLE: I thought it was almost characteristic of suppression pool designs like this one to find that the focus of failure was pretty much converged on inability to get the heat out of the suppression pool because of AC containment failure. You don't find this to be the case?

MR. DAEBLELER: No. Our TW sequence is talking about that sequence --

MR. EBERSOLE: The sequence that ultimately overheats the suppression pool water?

MR. DAEBLELER: No. We find that to be lower in our particular case.

MR. EBERSOLE: And why would you be significantly different from the standard finding? Like GESSAR?

MR. DAEBLELER: Well, in the case of GESSAR I'm not sure, again, what all the plant configurations look like relative to the plant configurations for Limerick. That would have an impact on the situation. It would be on recognizing the importance of such things as the RHR system, RHR service water system kinds of considerations. I can't relate -- I don't know the relationship between our system and GESSAR, the system.

MR. EBERSOLE: All right.

MR. OKRENT: Does BNL have any comment on Mr. Ebersole's question?

MR. SHIU: My recollection is that GESSAR only has two RHR loops versus Limerick has four loops. And that's a major difference in terms --

MR. OKRENT: Four 50 percent loops or --

MR. SHIU: No. They were able to maintain suppression pool cooling with one of the four loops where as GESSAR needs one of the two.

MR. OKRENT: Four 100 percent loops.

MR. EBERSOLE: But that's still an AC power dependency and I thought that was a major subject to which --

MR. SHIU: The AC power, loss of AC power is treated in the loss of off-site power. And this shows as a dominant sequence in Limerick as well as GESSAR. However, for other transients such as MSIV or turbine trip you have to assure yourself of the availability of off-site power at that point.

Maybe I'm not answering your question.

MR. EBERSOLE: I'm saying that the progressive failure of AC power is what is putting GESSAR in its worst problems, off-site as well on-site. And I was trying to find out why it didn't do it here irrespective of how many RHR pumps it's got.

MR. SHIU: GE has two -- three diesel generators. Limerick has four. Both the RHR are tied to only two diesels. The third diesels only provide power to the AGTS system. That is again a major difference.

MR. EBERSOLE: So it is the third and fourth diesels that help out here?

MR. SHIU: Yes.

MR. EBERSOLE: Right. Thank you.

MR. DAEBLELER: Again, then, going to risk and looking at early fatality risk and we find here that fires really don't contribute to that. We see the distribution of effects of internal initiators as well uncertainties on the seismic initiating events.

(Slide 146 shown.)

(Slide 147 shown.)

MR. DAEBLELER: The latent fatality situation is somewhat different than that in that it is not quite the degree of uncertainty on the seismic and the fires do -- although lesser contributors, do have a contributions.

(Slide 148 shown.)

MR. DAEBLELER: From those results, then, we can draw these conclusions: That the seismic initiated accidents are a major contribution if you consider the hypothesis that a large magnitude earthquake occurs in the plant region.

Looked at the other way is that the upper estimate for the seismic events is larger than for the internal initiators, but the low estimate is negligible.

Now, except for these seismic considerations then the internal initiated events cause the major contributions. In terms of latent risk, the internal initiated events are still a major contributor. However, as we saw on the chart before, seismic also contributes with the upper estimate about the equivalent to the internals and fire is a lesser contributor.

(Slide 149 shown.)

MR. DAEBELER: We looked through at these and broke down the internal and seismic initiators to look at early risk and found that the early risk is primarily due to the ATWS sequences. There is a lesser contribution, but a contribution from vessel failure, that is random vessel failure we are talking about. But again we looked at sequences, and no single sequence dominated that risk.

In the cases of the seismic, the early risk again was due primarily to the vessel support failure at the high accelerations. In terms of latent risk we found that the internal sequences are the same as those affecting core damage frequency, and therefore no single sequence dominates.

From the seismic viewpoint we found there was distributing -- risk was distributing between three types of sequences. And they are the loss of off-site power, again with loss of all AC, reactor building failure, and the vessel support. And again we found because it was distributed between those three sequences, no single sequence dominated.

(Slide 150 shown.)

MR. DAEBELER: Then we quickly surveyed the -- quickly -- We surveyed the various sequences and looked through the overall effort of the PRA and came up with the following list of functions important to core damage and risk.

Some of these in one case may be important to core damage and in other cases important to risk. But kind of gives an overview of listing of systems that do turn out to be important to these features and we have divided them into internal initiators, seismic initiators, and fire initiators.

And a number of these have been discussed previously, including the use of the power conversion system, depressurization and the high pressure systems. And the availability of AC power, which we have just gone through, and which includes diesel reliability and battery life.

Considerations: ATWS prevention and mitigation. In the case of seismic initiators, again we talk about the availability of AC power. The RPV supports we talked about. And the low seismic accelerations, the resetting of control circuitry. We also found that training in fire prevention and mitigation of fires is being of importance.

MR. DAVIES: Excuse me.

MR. DAEBELER: Yes.

MR. DAVIES: I thought your conclusion most recently is that AC power is not required for HPCI and RCIC room cooling.

MR. DAEBELER: That is correct. Really what we are saying here it influences that -- we have to make sure we have cooling and have the procedures for that. That's the importance that we are talking with there by opening the doors.

MR. DAVIES: Thank you.

(Slide 151 shown.)

MR. DAEBELER: This has been previously shown, and I don't know if you really want to review it again, but I can just briefly show and I'll be glad to discuss any of these if you wanted to.

But the PRA was an evolving situation and with interaction with the systems engineers and the PRA practitioners, we found that the PRA did influence the installation and the design of these five areas.

(Slide 152 shown.)

MR. DAEBELER: Likewise, in going through the effort -- again this is a repeat of a slide that was shown at the last meeting or the one before last -- of those areas that we confirmed through the PRA that these various features were desirable.

MR. DAVIES: Excuse me.

MR. DAEBELER: Yes.

MR. DAVIES: I think that part of the desirability of those features depends on the assumptions in the PRA. Depending on your common cost failure model, you can eliminate the desirability of four diesels. And I think it is important that that be recognized, that not everyone would agree that four diesels is that much better than two.

MR. BOYER: Not much better than what?

MR. DAVIES: Than two diesels. If the failure factor is like point one, then four doesn't help you any more than two.

MR. DAEBELER: Right.

MR. DAVIES: That's a subject of debate right now.

MR. EBERSOLE: Are those four diesels supplied with water from two cooling systems? And are they cross-tied?

MR. DAEBELER: Yes. Emergency service water system.

MR. EBERSOLE: That's an automatically started system?

MR. DAEBELER: Yes. Yes.

MR. BOYER: You have probably more depth in that than I do, but I inherently would rather have four diesels than two.

MR. EBERSOLE: Yes. Sometimes I think I would rather have them on radiators rather than service pipes.

MR. DAEBELER: Then there were some procedures we also thought were influenced in including this HPCI RCIC room cooling that we mentioned previously along with the venting, the containment spray that we talked about, the importance of reestablishing the power conversion system, as well as resetting the control circuitry. And these things were emphasized, the importance was emphasized.

(Slide 153 shown.)

(Slide 154 shown.)

MR. OKRENT: Venting reminds me, where does Limerick stand on venting of containment? Is it part of --

MR. DAEBELER: The EPG's, right, yes, our trip procedures.

MR. BOYER: If you remember, Dave gave the report.

MR. OKRENT: All right. I had to get back in the right month. Okay.

MR. DAEBELER: The last item is I think of interest, and these are some of the insights we gained by performing the PRA's are, if you will, of a programatic nature.

We do believe that the PRA process enhances the understanding of the plant. And I think this is a very important aspect, both in design and in the operation. We realize that -- and we should consider that due to uncertainties in modeling and data, it is really best to use PRA in looking at alternates.

Recognizing these inherent uncertainties is critical in evaluating any potential plant changes because potential fixes may have significantly more or less benefits than point estimates might initially indicate. We do believe that in evaluating alternates, estimates of core damage in particular resulting from internal initiators can be very important inputs to those decisions.

That's all I have. If there are no questions I'll turn it over to Mr. Diederich.

MR. OKRENT: Let's go.

MR. DIEDERICH: Because of our commitments to the staff and to the ACSR in the past and because of our interest in plant safety, we plan to continue to use our PRA in support of Limerick operations.

We initiated a study -- with a consultant of course -- as to how to best go about this. The goals of our study were to establish something within our existing organization. Not an appendage, but something that fit in with our mode of operation. We wanted to establish the technical bases thac we would use in our ongoing use of the PRA and to go through a well-planned, phased implementation effort.

MR. OKRENT: How many people who work for Philadelphia Electric were either full-time or at least half-time participants in the PRA or SARA?

MR. DIEDERICH: Let me explain to you how we supported PRA/SARA.

We had several people were involved indirectly dealing with our consultants. They were the -- our PRA/SARA people. These are about three of those. Maybe three-and-a-half.

But in addition to that, while the process was going on of developing the PRA, our systems engineers reviewed all the fault trees for their particular systems. And when operator actions were called on, the plant staff reviewed those areas of the fault trees. So that although it looks like a PRA thing with PRA people, it was actually a combination of plant operating staff, and the insights got to be spread throughout. That's the kind of thing we wanted to keep going in our ongoing use, because that's where the real benefits come in.

(Slide 155 shown.)

MR. DIEDERICH: We have decided that the way -- one way to accomplish this is to establish a PRA maintenance and use group. This group will be comprised primarily of engineers to document the original design bases, to update the PRA both for an original plant base line and on a periodic basis thereafter, to evaluate modifications to the plant, to evaluate changes to the technical specification, to maintain and use the computer codes that go along with the PRA, to do data analyses of failure rates and things of that nature, to provide PRA training to others, and to perform miscellaneous studies and analyses as requested by others.

(Slide 156 shown.)

MR. DIEDERICH: This would be housed in our engineering department, but a lot of our value is to be gained in our operating branches. And to act as our field PRA arm we planned to use our independent safety engineering group, which is in our operating department, formed as a result of one of the TMI action plan items.

Part of their job is to continually evaluate operating experiences, not just at Limerick, but also at other plants, our own Peach Bottom and plants throughout the industry.

They will identify those particular areas where PRA is valuable in assessing the operating experience. They will also assure that PRA results are reflected in operating procedures, maintenance activities and the training of operating and maintenance personnel.

(Slide 157 shown.)

MR. DIEDERICH: Putting this in a little picture, we end up with, as our PRA organization, the PRA maintenance and use group continuing to interact heavily with the system engineers.

The PRA maintenance and use group interacting with the independent safety engineering group, and the independent safety engineering group be the eyes and ears, and I guess mouth, spreading the word to plant operations.

MR. OKRENT: Has PRA affected your maintenance practice?

MR. DIEDERICH: Has PRA affected our maintenance practice? I would say that it has not, to a large extent.

MR. BOYER: Well, with regard to taking equipment out of service, we have had a practice in the PRA to show that this would be desirable, say, to do maintenance on the diesels during plant outages rather than during the time that the plant is in service. And we have -- that philosophy follows to other essential class one systems as well.

So from that standpoint, when we do our maintenance work, you would use a systems interaction or, I guess -- I guess it would be more PRA type logic.

In the need for spares and availability of -- to be able to repair equipment in a short time frame to reduce down time would come from somewhat of a PRA logical approach. In other words, what part is important? What part should we have?

So to that extent I think it would affect -- I would say it would affect maintenance procedures, maintenance policy, more than the actual operation or conduct of the performance of a maintenance worker in his skills. That's a sort of given, I think, that either way.

MR. OKRENT: You may already do it, but I would assume somehow you would try to adjust your maintenance to minimize the chance of common mode errors like happened on 10-11, I guess it was, or like happened what happened on BWR.

MR. DIEDERICH: In general that's our practice already. We didn't identify anything particular from our PRA study, but that is the type of thing which we could investigate further once we have established the expertise in our maintenance and use group.

MR. EBERSOLE: May I ask, in looking at your tech specs, it has been proposed a long time in the past that tech specs should be matrixed to fit the plant condition, whether down or low power or whatever, so that you would not inadvertently enter into a degree of disablement or degradation of reliability not consistent with the plant condition.

A case in point would be Turkey Point, where they simultaneously degraded and stopped both coal and overpressurization protection devices at the precise time they were entering the coal pressurization condition. It would never have occurred if they had been at full power to start doing repair work on the systems.

So are you going to move toward an arrangement where your work in fact will be carefully keyed by matrix conditions to the plant operating condition?

MR. BOYER: Whether it is done by matrix or not, on that example you stated, I don't think -- it would have occurred because the type of analysis we do about taking a piece of equipment -- before we take a piece of equipment out of service would look at those considerations. I think that's -- under the gun -- under the heading of generally good operating practice and good maintenance practice and reliability of the plant.

The scheduling of equipment out of service and the effect of that piece of equipment out on the operability of the plant is something we always look at.

MR. EBERSOLE: On your diagram up here you run down to system engineers. Is there a sub-group of multiple systems and integral set of systems below just independent system engineers? Do you have system engineers that have some sort of control over system integration?

MR. DIEDERICH: Well, our organization is basically functional, by discipline and subdiscipline within that.

MR. EBERSOLE: I'm trying to find out how the systems are interrelated to each other.

MR. DIEDERICH: There aren't that many of us, and there are sufficient interfaces between the systems that our system engineers get to know fairly well not only just their systems, but those that interface with them.

And although I could never call it a formal program, we are all there together on the same floor, all working on the same problems, and many times someone will solve the problem of his friend rather than just his own.

MR. DAEBELER: I might add, there is a specific director by management of cross-training of engineers. They may be specialists on a given system for a year, year and a half, or something of that sort, and then they will move to another system.

MR. EBERSOLE: Thank you.

MR. OKRENT: Better move along. We're both asking too many questions.

MR. DIEDERICH: Yes

(Slide 158 shown.)

MR. DIEDERICH: For the technical bases, we tempted to define the scope of our PRA, the basis for measurement of the goodness of things, and the level of detail that we would go to.

(Slide 159 shown.)

MR. DIEDERICH: For our scope we decided to concentrate on internal initiators. And primarily exclude external initiators, seismic and fire, and accident effects.

We planned to make periodic evaluations of the need to go deeper into these as things become clearer, but as we have heard right in this room and several weeks before, those things aren't clear, and we are not going to have a giant effort, so we would like to be most effective with the personnel we have.

(Slide 160 shown.)

MR. DIEDERICH: For our unit of measure we are using core damage frequency, primarily. We considered using other things in the consequences area such as risk to population or an average individual, or hypothetical individual, some quantity of plant release. But once you get past core damage frequency, we get into the phenomenology which is less clear at the moment and we feel it would be taking our resources to attempt to clarify things which are perhaps not clarifiable.

We do, however, have all the models used in our PRA on our computer and for those sequences where things require containment response to make comparisons, we will have the ability to do but it will not be part of our normal course of events.

(Slide 161 shown.)

MR. DIEDERICH: For level of detail we plan to start out with exactly what we have in our present scope and explain -- expand the detail as needed by applications. I can see that as we go into actual using these, comparing modification effects, the level of details is going to continue to expand. Our periodic update of the PRA will roll in all the new modeling that we have done and allow us to get the most benefit out of it.

(Slide 162 shown.)

MR. DIEDERICH: For scheduling our implementation, we believe that training and staffing our initial organization will take approximately six months -- that's organizing to get organized -- and then following that as long as 18 months will be necessary to baseline and document the PRA.

It is not that the documentation does not exist, mind you. But it exists in a large number of cardboard boxes. And to be using this on an ongoing basis it is unacceptable to go leafing through cardboard boxes for every bit of information.

(Slide 163 shown.)

MR. DIEDERICH: We believe that the result of our efforts will be a PRA which has been integrated into our organization and of doing business. We think that the results will be reflected in the modifications we make, in our plant operations, in our maintenance programs and in our training programs of maintenance and operating personnel. We believe our PRA will be maintained up to date, and we also plan periodic reevaluations of our program to assure that we are getting the most out of them on an effectiveness basis.

(Slide 164 shown.)

MR. COFFMAN: The purpose of this presentation is to describe an assessment for the robustness and the conclusions drown from the Limerick review of the Limerick PRA and SARA, and I will approach this assessment by one, restating selective conclusions from the NUREG to describing what seems to be the source of uncertainty to each conclusion. Two, which each conclusion is sensitive, and three, appraising the soundness of each conclusion, if possible.

One of the conclusions was the review of the dominant accident sequences found no instances of non-compliance with the deterministic regulatory requirements. This conclusion is based upon an analysis on cut sets of front line systems in the leading accident sequences.

Cut sets were determined after adding the support system dependencies included Brookhaven discovered dependencies. The source of uncertainty which this conclusion appears most sensitive is in the completeness of fault tree modeling of support systems dependencies.

The system structure of Limerick was not, quote, "completely modeled" in PRA, and because hidden dependencies have the potential to transcend many levels in the fault tree, the discovery of a hidden dependency can change the order of leading accident sequences.

So specifically the dependence of safety related equipment upon equipment not required to be qualified for larger earthquakes may be missing from the fault trees.

The Limerick PRA did not completely model the dependencies of either the reactor protection system or control systems. Limerick PRA did model dependencies such as the heat removal function upon the heat exchangers service service water discharge headers and the spatial dependence of the ADS upon the undesirable location of the gas supplied to the ADS.

Brookhaven added functional dependencies like the HPCI pump lubrication, dependence upon the suppression pool temperature, and the dependencies added by Brookhaven increased the estimated core damage frequency, but not dramatically.

Although the PRA did not completely model the systems structure at Limerick, the second conclusion that we made appears sound. However, this conclusion may not necessarily apply to the entire list of sequences because you only looked at the leading sequences, and the conclusion nay not necessarily apply to the evaluation of sequential failures among elements.

Evaluation of sequential failures is very difficult using fault trees because it requires modification to the success criteria in time steps.

One of the other conclusions was that operation of Limerick does not seem to possess a disproportionate share of the societal risk compared with plants which are also located in areas of high population density. This conclusion is closely associated with the bottom line and is cumulatively sensitive to all major sources of uncertainty.

They have been pretty well described. This meeting kind of covered that. I don't know that I need to mention any of those. Maybe there is an added one that was not covered, or at least I wasn't cognizant when it was covered, and that is that other plants in high population density sites are older than Limerick. Therefore, there was no assessments of aging -- there may be an assessment of aging that wasn't included.

One of the other conclusions was that the dominant -- Oh, I'm sorry. I'm going to come back to that, that collusion in just a second, to summarize, but I want to step through just quickly two more conclusions.

Another conclusion in the NUREG was that the dominant contributors to the core damage frequency are the transients and loca events.

Let me summarize. The PRA showed external events contributed 38 percent to the core damage frequency. Staff review showed that the external events contributed 10 percent. But if you even use the reviews high values for seismic and high value for fire core damage frequencies, the external events still only contributed 34 percent.

I don't know of any other sources of uncertainty that haven't already been identified, that that conclusion is sensitive to.

And then there was a conclusion that in recognition of the substantial uncertainties in the PRA and SARA it appears reasonable and prudent that the applicant establish and implement a safety assurance program. This is an interesting conclusion in the sense that it is consistent with the defense in-depth philosophy of the regulations that itself was probably motivated in view of uncertainties, yet this process could challenge the past implementation of this philosophy that resulted in the pursuit of conservatisms in separate plant features.

The PRA integrates experience and judgment, and by its ongoing use it could provide a basis to determine the totality of conservatisms from among the intended conservatisms on separate plant features.

That may not be a clear way to say it, but in essence it gives us the opportunity to balance competing risk.

In summary, allow me to contrast the uncertainties in the Limerick PRA with the risk from other PRA's and the risks from other hazards. The staff has reviewed PRA's from three other plants, principally because they too were in high population density sites. These are the PRA's of the Indian Point units two and three and the Zion plant.

Estimated uncertainty on the risk results from these four PRA's is about 40 times greater than the spread among the estimates of the expected risk for those PRA's. Therefore, we came to the conclusion that the risks at Limerick are well within the spectrum of risks calculated for other high population density sites.

By using the high values for sequence class frequencies shown in table four of NUREG 1068, in combination with the largest values for the conditional mean accident consequences shown in Table K-1 of NUREG 0974, which was -- Table K-1 was presented at least twice today -- which was the FES -- one can estimate a pessimistic low probability risk for Limerick.

You need to remember that the conditional mean accident consequences are themselves judged to contain a significantly pessimistic bias.

The resultant pessimistic low probability estimate of the latent cancer fatalities, including thyroid cancers, is about zero point six per year of plant operation. Place that estimate in contrast with the range from two to 25 fatalities per year calculated for a comparable one thousand megawatt electric coal-fired plant. Society has tolerated consequences well beyond my pessimistic estimates without lasting effects.

Regardless, primary objectives of the PRA review gave priority to each effort to continue the improvement in public health and safety associated with the operation of Limerick considering the uncertainties and limitations of the PRA.

That concludes what I felt I had time to say.

MR. OKRENT: I think the hour is late and there don't seem to be any -- Do I see a hand?

MR. BOYER: I just wanted to clarify -- I have three items left over that you would like to hear next time, which I presume is November 2nd. And before we conclude we ought to highlight those.

MR. OKRENT: What were those three items?

MR. BOYER: One was the extent of the consideration of failure of non-class one systems on the actions of safety systems. That was the discussion we had with Jesse Ebersole. And some clarification of how we treated that.

Second was a comparison of system interaction considerations between Limerick and Indian Point.

The third related to the area of seismic hazards and further consideration of the comparison of Indian Point and Limerick. That was your discussion with the staff. I think one of their people who is not here today may be able to shed some light on that.

MR. OKRENT: Well, I think we will have those three items on the agenda somehow for the next meeting.

MR. BOYER: We will be prepared to address those.

MR. OKRENT: It may not be exactly in the context worded, especially the last one. The seismic one is a little broader, but in any event, we will try to develop a tentative agenda for the meeting with the full committee as soon as we can. We have to integrate the outcome of two different subcommittee meetings, actually, as you know, and discuss how best the committee might spend its time. The committee may think differently.

MR. BOYER: Right. We will be working --

MR. OKRENT: We will try to get at least a fairly good idea of the agenda to you as soon as we can. Sometime this week. Not this week, since this week is nearly over.

MR. ROSENTHAL: We thought -- and at least I was wrong -- that we would be adequately responsive to your concerns over seismic as expressed in the ACSR letter. It would be use useful to us to have as clear a definition, perhaps more than an agenda item -- a paragraph to insure that we can be responsive to you.

MR. OKRENT: I'll try.

Let me ask our consultants to make sure that they get us their reports by a week from Monday. They should arrive at Mr. Savio's office in -- I don't know how long it takes to get through the NRC.

(Discussion held off the record.)

MR. OKRENT: And also if Messrs. Trifunac and Pomeroy have some comments to add in the area of the seismic, please.

Let's see. Are there any other points that are vital?

If not, the hearing -- I'll thank everybody for what I found to be an interesting meeting and I'll adjourn it.

(The hearing was adjourned at 5:45 p.m.)

CERTIFICATE OF OFFICIAL REPORTER

This is to certify that the attached proceedings before the UNITED STATES NUCLEAR REGULATORY COMMISSION in the matter of:

NAME OF PROCEEDING: GESSAR II AND RELIABILITY & PROBILISTIC ASSESSMENT(ACRS)

DOCKET NO.: NONE

PLACE: LOS-ANGELES, CA

DATE: FRIDAY, OCTOBER 19, 1984

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission.

(Sigt). Mary Ellen Tearney
(TYPED)

Official Reporter

Reporter's Affiliation

3-1

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LIMERICK CONTAINMENT BEHAVIOR DURING CORE MELTDOWN ACCIDENTS

PRESENTED BY

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OCTOBER 20, 1984

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

OBJECTIVES

AN IN-DEPTH WALK-THROUGH OF LIMERICK CONTAINMENT FAILURE MODES, SOURCE-TERMS AND CONSEQUENCES WHICH WILL DEMONSTRATE THAT:

- SOURCE-TERM CALCULATIONS PERFORMED FOR THE LGS-FES ARE MUCH CLOSER TO UPPER BOUND RISK ESTIMATES THAN LOWER BOUND

- UNCERTAINTY LEADING TO HIGHER SOURCE TERMS RESULTS IN LESS THAN FACTOR OF 4 INCREASE IN TOTAL RISK

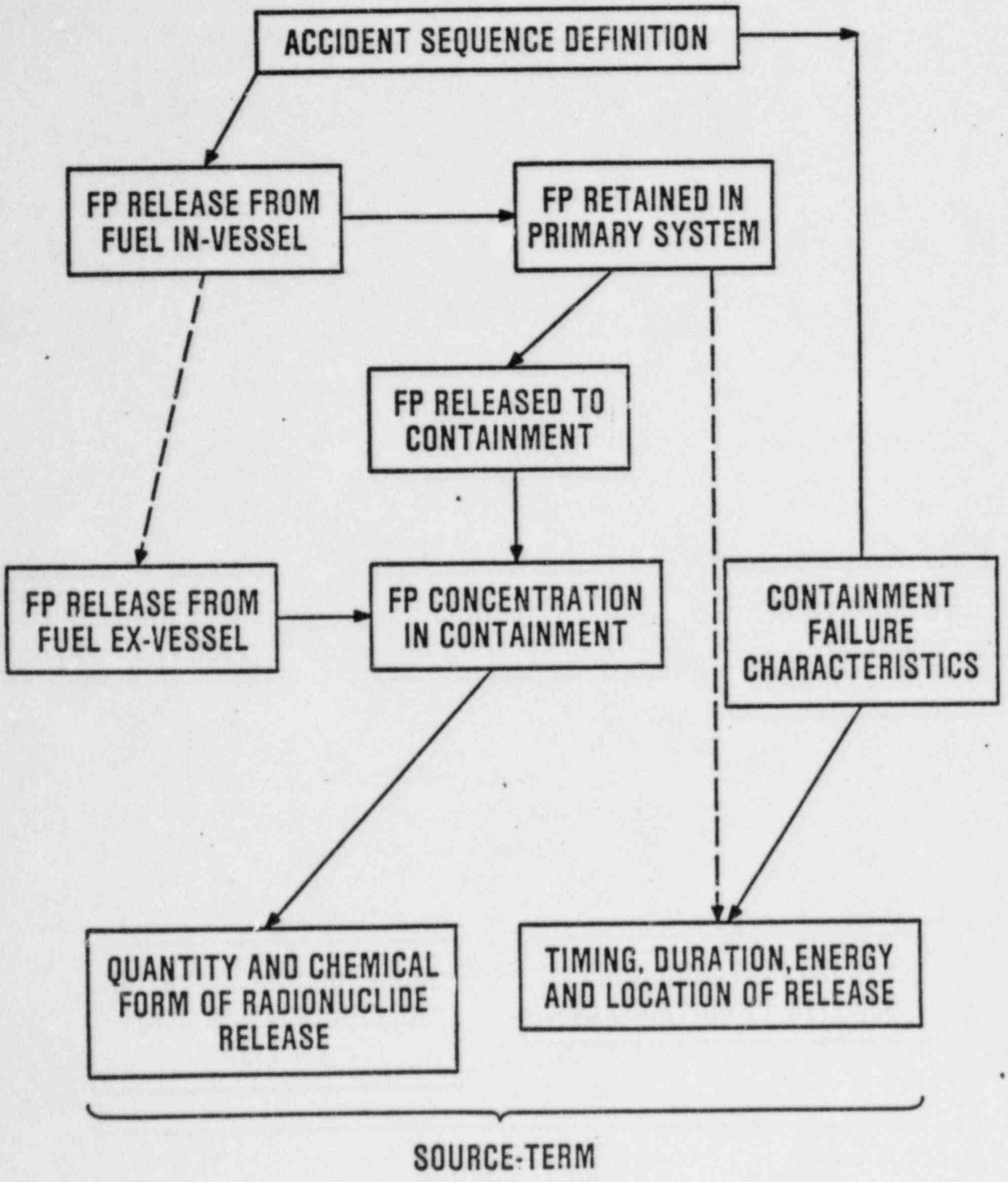
- NEW METHODS HAVE THE POTENTIAL FOR SIGNIFICANTLY REDUCING LONG-TERM DAMAGE INDICES (LATENT FATALITIES, PERSON-REM, ETC.)

- NEW METHODS HAVE LESS POTENTIAL FOR REDUCING SHORT-TERM DAMAGE INDICES (EARLY FATALITIES, ETC.)

OUTLINE

- SOURCE TERM DEFINITION
- METHODS
- ACCIDENT CLASSES
- POTENTIAL FAILURE MODES
- RISK PERSPECTIVE
- CLASS I SEQUENCES
- CLASS IV SEQUENCES
- IMPACT OF NEW METHODS
- SUMMARY AND CONCLUSIONS

SOURCE-TERM DEFINITION



ACCIDENT SEQUENCE DEFINITION

FP RELEASE FROM FUEL IN-VESSEL

FP RETAINED IN PRIMARY SYSTEM

FP RELEASED TO CONTAINMENT

FP RELEASE FROM FUEL EX-VESSEL

FP CONCENTRATION IN CONTAINMENT

CONTAINMENT FAILURE CHARACTERISTICS

QUANTITY AND CHEMICAL FORM OF RADIONUCLIDE RELEASE

TIMING, DURATION, ENERGY AND LOCATION OF RELEASE

SOURCE-TERM

WASH-1400 METHODS

- BOIL CODE USED TO MODEL PRIMARY SYSTEM BEHAVIOR
- HAND CALCULATIONS FOR CONTAINMENT RESPONSE
- SPECIFIED FISSION PRODUCT RELEASE FRACTIONS FOR DIFFERENT PHASES OF CORE MELTDOWN (GAP, MELT, VAPORIZATION, AND OXIDATION
- NO FISSION PRODUCT DEPOSITION MODELED IN PRIMARY SYSTEM

WASH-1400 METHODS (CONT.)

- FISSION PRODUCT DECONTAMINATION FACTORS IN SUPPRESSION POOL:

  DF=1, SATURATED POOL

  DF=100, SUBCOOLED POOL

- CORRAL CODE USED TO PREDICT FISSION PRODUCT TRANSPORT IN CONTAINMENT (AND RELEASE WHEN CONTAINMENT FAILS)

CORE MELT PHENOMENA AND CONTAINMENT RESPONSE METHODS

- LGS-PRA:
  - USE OF INCOR CODE WITH INDEPENDENT ANALYSES
- LGS-SARA:
  - USE OF MARCH 1.1 CODE WITH INDEPENDENT ANALYSES
- BNL REVIEW:
  - USE OF MARCH 1.1 CODE WITH INDEPENDENT ANALYSES

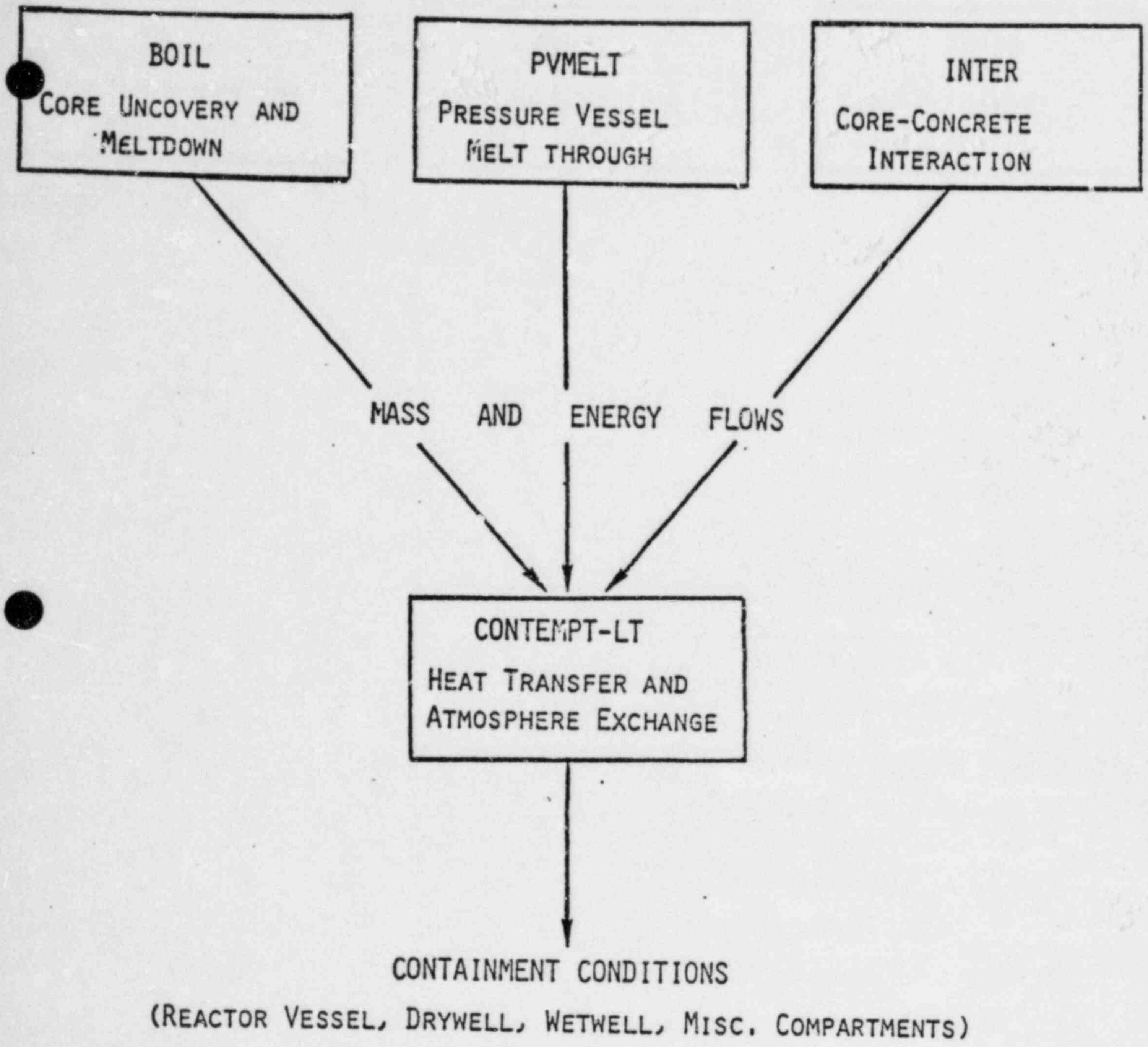


Figure 7.1 Diagrammatic Representation of INCOR Organization.

(Reproduced from the Limerick PRA).

Table 7.1 Comparison of INCOR and MARCH Computer Codes.

Phenomena	Modeled in Subroutine	
	INCOR	MARCH
Rapid primary system depressurization	-	INITIAL
Slower primary system depressurization, core uncovery, and meltdown	BOIL	BOIL
Pressure vessel melt-through	PVMELT	HEAD
Core debris/water interactions in cavity	-	HOTDROP
Core debris/concrete interactions	INTER	INTER
Containment response characteristics	CONTEMPT-LT	MACE

LGS-PRA METHODS

- LGS-PRA INTERNAL EVENTS SOURCE TERM CALCULATIONS BASED ON RSS METHODS BUT WITH SOME DEVIATIONS -
    - PARTITION OF MELT RELEASE BETWEEN DRYWELL AND POOL
    - DF=10 FOR SATURATED POOL
    - 10% OXIDATION RELEASE FOR ALL FAILURE MODES
    - 15% POOL FLASH RELEASE (CLASS I AND III ONLY)
- BNL REVIEW (NUREG/CR-3028) BASED ON SIMILAR APPROACH TO LGS-PRA
- BNL INPUT TO MITCHELL MEMO (DATED AUGUST 1983) BASED ON POOL DFs CALCULATED USING POSTMA POOL SCRUBBING MODEL

LGS-SARA METHODS

- LGS-SARA EXTERNAL EVENTS SOURCE TERM CALCULATIONS DIFFERED FURTHER FROM RSS METHODS
  - IN-VESSEL MELT RELEASE BASED ON TRENDS IN NUREG-0772
  - EX-VESSEL VAPORIZATION RELEASE BASED ON DIFFERENCE BETWEEN NUREG-0772 AND RSS PREDICTIONS
  - FISSION PRODUCT TRANSPORT BASED ON HAND CALCULATIONS, NOT CORRAL CODE

LGS-FES METHODS

- SOURCE TERM CALCULATIONS BASED ON RSS METHODS

- BNL STAFF CALCULATED 28 SOURCE TERMS FOR BOTH INTERNALLY AND EXTERNALLY INITIATED ACCIDENT SEQUENCES

- SOURCE TERMS DOCUMENTED IN BNL-33835

ACCIDENT CLASSES

• CLASS I	CONTAINMENT INTACT AT TIME OF CORE MELT AND AT LOW PRESSURE	TRANSIENTS AND SMALL BREAK LOCAs WITH LOSS OF COOLANT MAKEUP
• CLASS II	CONTAINMENT FAILS PRIOR TO CORE MELT DUE TO OVERPRESSURIZATION	TRANSIENT OF LOCAs INVOLVING LOSS OF HEAT REMOVAL
• CLASS III	CONTAINMENT INTACT AT TIME OF CORE MELT AND AT HIGH PRESSURE	ATWS AND AN INABILITY TO PROVIDE COOLANT MAKE-UP, LOSS OF HEAT REMOVAL
• CLASS IV	CONTAINMENT FAILS PRIOR TO CORE MELT DUE TO OVERPRESSURIZATION	ATWS WITH COOLANT MAKE-UP AND LOSS OF HEAT REMOVAL

ACCIDENT CLASSES (CONT.)

• CLASS IS	CONTAINMENT FAILS PRIOR TO CORE MELT DUE TO SEVERED RHR SUCTION LINE	SEISMICALLY INDUCED SEQUENCE LEADING TO FAILURE OF COOLANT MAKE-UP AND LOSS OF WETWELL INTEGRITY
• CLASS S	PRIMARY SYSTEM AND CONTAINMENT FAIL AT START OF ACCIDENT	SEISMICALLY INDUCED SEQUENCE (ALSO RANDOM RPV FAILURE) LEADING TO FAILURE OF COOLANT MAKE-UP AND LOSS OF PRIMARY SYSTEM AND CONTAINMENT

Table 7 Damage state probabilities

Damage State	Total Probability	Probability Regional Disasters	Probability Non-Regional Disasters
I-S	7.6(-8)*	-	7.6(-8)
I-T	8.31(-5)	2.27(-6)	8.1(-5)
II-T	3.8(-6)	4.0(-8)	3.8(-6)
III-T	3.9(-6)	7.4(-7)	3.2(-6)
IV-A	5.0(-9)	-	5.0(-9)
IV-T	4.2(-7)	9.5(-8)	3.25(-7)
IS-C	1.44(-7)	1.3(-7)	1.4(-8)
IS-$\overline{C}$	1.0(-6)	9.0(-7)	1.0(-7)
S-H2O	5.45(-8)	4.1(-8)	1.35(-8)
S-$\overline{H2O}$	3.83(-7)	3.79(-7)	1.35(-8)

*7.6(-8) = 7.6 × 10^{-8}

POTENTIAL FAILURE MODES

- FAILURE BY PRESSURE OR TEMPERATURE
- BASEMAT PENETRATION
- STEAM EXPLOSIONS
- HYDROGEN BURN INDUCED FAILURES
- FAILURE TO ISOLATE CONTAINMENT BUILDING
- ACCIDENT SEQUENCE BYPASSES CONTAINMENT BUILDING, I.E., MSIV LEAKAGE

RISK PERSPECTIVE

- SHORT-TERM DAMAGE INDICES
- LONG-TERM DAMAGE INDICES
- RISK MEASURES
- RISK DOMINANT SEQUENCES

SHORT-TERM DAMAGE INDICES

- EARLY FATALITIES AND INJURIES
- WITHIN ONE YEAR OF ACCIDENT AND RELATIVELY CLOSE TO REACTOR SITE
- STRONG THRESHOLD EFFECT (OVER 320 REM WITH SUPPORTIVE MEDICAL TREATMENT)
- STRONGLY INFLUENCED BY:
    - TIMING, MAGNITUDE AND DISPERSAL OF FISSION PRODUCT RELEASE
    - EMERGENCY RESPONSE AND/OR SHELTERING OF POPULATION

LONG-TERM DAMAGE INDICES

- DELAYED CANCER FATALITIES, THYROID CANCERS AND PERSON-REM
- OVER 30 YEARS AFTER ACCIDENT AND FOR 500 MILES AROUND SITE
- STRONGLY INFLUENCED BY TOTAL CURIES RELEASED
- RELATIVELY INSENSITIVE TO EMERGENCY RESPONSE OF THE POPULATION

Table K.1 Conditional mean values of societal consequences from individual release categories for three alternative offsite emergency response modes

Consequence Category	Offsite Emergency Response Mode	Release Categories									
		I-T/DW	I-T/WW	I-T/$\overline{WW}$	I-T/SE*	I-T/HB	I-T/$\overline{LGT}$	II-T/WW	III-T/WW	III-T/HB	III-T/LGT
1. Early fatalities with supportive medical treatment (persons)	Evac-Reloc	0	0	0	2(2)**	1(1)	5(-1)	0	0	1(1)	0
	Early Reloc	1(0)	0	0	7(1)	1(1)	1(0)	2(2)	3(1)	1(1)	0
	Late Reloc	3(1)	5(-1)	5(-1)	---	1(2)	5(1)	2(3)	4(2)	2(2)	2(-2)
2. Population receiving in excess of 200 Rems total marrow dose from early exposure (persons)	Evac-Reloc	0	0	0	2(3)	4(2)	4(1)	5(2)	2(3)	4(2)	3(0)
	Early Reloc	1(1)	0	0	1(3)	3(2)	2(1)	2(3)	2(3)	3(2)	0
	Late Reloc	1(2)	3(0)	1(0)	--	1(3)	9(2)	5(3)	7(3)	1(3)	5(0)
3. Early injuries (persons)	Evac-Reloc	4(1)	0	0	3(3)	5(2)	5(1)	6(2)	3(3)	5(2)	5(0)
	Early Reloc	5(1)	1(-2)	2(-2)	3(3)	4(2)	4(1)	2(3)	3(3)	4(2)	8(-1)
	Late Reloc	2(2)	2(0)	1(0)	--	1(3)	6(2)	3(3)	6(3)	1(3)	9(0)
4. Delayed cancer fatalities (excluding thyroid) (persons)	Evac-Reloc	6(2)	1(1)	4(1)	.6(3)	2(3)	1(3)	4(3)	4(3)	2(3)	2(1)
	Early Reloc	6(2)	3(1)	5(1)	6(3)	2(3)	1(3)	4(3)	4(3)	2(3)	3(1)
	Late Reloc	7(2)	3(1)	5(1)	--	2(3)	1(3)	4(3)	4(3)	2(3)	3(1)
5. Delayed thyroid cancer fatalities (persons)	Evac-Reloc	1(2)	2(1)	2(1)	8(2)	6(2)	2(2)	1(3)	9(2)	6(2)	1(1)
	Early Reloc	1(2)	2(1)	2(1)	8(2)	6(2)	2(2)	1(3)	1(3)	6(2)	2(1)
	Late Reloc	2(2)	2(1)	2(1)	--	7(2)	2(2)	1(3)	1(3)	7(2)	2(1)
6. Total person-rems	Evac-Reloc	1(7)	5(5)	8(5)	4(7)	2(7)	2(7)	6(7)	6(7)	2(7)	4(5)
	Early Reloc	1(7)	5(5)	9(5)	4(7)	2(7)	2(7)	6(7)	6(7)	2(7)	5(5)
	Late Reloc	1(7)	5(5)	1(6)	--	2(7)	3(7)	7(7)	7(7)	3(7)	6(5)
7. Cost of offsite mitigation measures (1980 dollars)	Evac-Reloc	3(8)	5(7)	6(7)	2(9)	1(9)	1(9)	4(9)	3(9)	1(9)	1(6)
	Early Reloc	2(8)	2(6)	3(6)	2(9)	1(9)	1(9)	4(9)	3(9)	1(9)	1(6)
	Late Reloc	2(8)	2(6)	3(6)	--	1(9)	1(9)	4(9)	3(9)	1(9)	1(6)
8. Land area for long-term interdiction (m^2)	Evac-Reloc	1(6)	2(4)	3(4)	7(7)	2(7)	3(7)	1(8)	6(7)	2(7)	0
	Early Reloc	1(6)	2(4)	3(4)	7(7)	2(7)	3(7)	1(8)	6(7)	2(7)	0
	Late Reloc	1(6)	2(4)	3(4)	--	2(7)	3(7)	1(8)	6(7)	2(7)	0

*This release category has a probability less than 10^{-9} per reactor-year to be initiated by severe earthquakes; it is not analyzed with Late Reloc mode for its insignificant contribution to risks due to its low probability.

**2(2) = 2 × 10^2 = 200.

***These release categories are initiated by plant internal causes; therefore, the Late Reloc mode does not apply.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table K.1 (Continued)

Consequence Category	Offsite Emergency Response Mode	Release Categories									
		III-T/$\overline{\text{LGT}}$	IV-T/DW	IV-T/WW	IV-T/$\overline{\text{WW}}$	I-S/DW***	IV-A/DW***	IS-C/DW	IS-$\overline{\text{C}}$/DW	S-H2O/$\overline{\text{WW}}$	S-$\overline{\text{H2O}}$/$\overline{\text{WW}}$
1. Early fatalities with supportive medical treatment (persons)	Evac-Reloc	6(-1)	6(2)	5(2)	6(2)	0	7(2)	3(2)	1(2)	0	0
	Early Reloc	1(0)	1(3)	1(3)	1(3)	0	1(3)	7(2)	7(2)	2(2)	6(2)
	Late Reloc	7(1)	4(3)	4(3)	4(3)	--	--	3(3)	3(3)	2(3)	3(3)
2. Population receiving in excess of 200 Rems total marrow dose from early exposure (persons)	Evac-Reloc	5(1)	5(3)	4(3)	4(3)	0	4(3)	2(3)	2(3)	4(2)	4(2)
	Early Reloc	3(1)	6(3)	5(3)	4(3)	5(-1)	5(3)	3(3)	3(3)	1(3)	2(3)
	Late Reloc	1(3)	1(4)	1(4)	1(4)	--	--	9(3)	9(3)	5(3)	8(3)
3. Early injuries (persons)	Evac-Reloc	6(1)	5(3)	4(3)	3(3)	0	3(3)	2(3)	2(3)	5(2)	6(2)
	Early Reloc	4(1)	5(3)	4(3)	4(3)	5(-1)	3(3)	3(3)	3(3)	2(3)	2(3)
	Late Reloc	7(2)	7(3)	6(3)	7(3)	--	--	6(3)	6(3)	3(3)	5(3)
4. Delayed cancer fatalities (excluding thyroid) (persons)	Evac-Reloc	1(3)	5(3)	5(3)	5(3)	2(2)	5(3)	4(3)	4(3)	3(3)	4(3)
	Early Reloc	1(3)	5(3)	5(3)	5(3)	2(2)	5(3)	4(3)	4(3)	3(3)	4(3)
	Late Reloc	1(3)	6(3)	6(3)	6(3)	--	--	4(3)	4(3)	3(3)	4(3)
5. Delayed thyroid cancer fatalities (persons)	Evac-Reloc	2(2)	2(3)	2(3)	2(3)	3(1)	2(3)	9(2)	9(2)	7(2)	1(3)
	Early Reloc	2(2)	2(3)	2(3)	2(3)	3(1)	2(3)	9(2)	1(3)	8(2)	1(3)
	Late Reloc	2(2)	2(3)	2(3)	2(3)	--	--	1(3)	1(3)	8(2)	1(3)
6. Total person-rems	Evac-Reloc	2(7)	8(7)	7(7)	8(7)	3(6)	8(7)	5(7)	5(7)	4(7)	6(7)
	Early Reloc	2(7)	8(7)	8(7)	8(7)	3(6)	8(7)	5(7)	5(7)	5(7)	6(7)
	Late Reloc	3(7)	9(7)	8(7)	9(8)	--	--	6(7)	6(7)	5(7)	7(7)
7. Cost of offsite mitigation measures (1980 dollars)	Evac-Reloc	1(9)	5(9)	5(9)	5(9)	9(7)	5(9)	2(9)	2(9)	2(9)	3(9)
	Early Reloc	1(9)	5(9)	5(9)	5(9)	4(7)	5(9)	2(9)	2(9)	2(9)	3(9)
	Late Reloc	1(9)	5(9)	5(9)	5(9)	--	--	2(9)	2(9)	2(9)	3(9)
8. Land area for long-term interdiction (m^2)	Evac-Reloc	3(7)	1(8)	1(8)	2(8)	3(5)	1(8)	5(7)	6(7)	5(7)	8(7)
	Early Reloc	3(7)	1(8)	1(8)	2(8)	3(5)	1(8)	5(7)	6(7)	5(7)	8(7)
	Late Reloc	3(7)	1(8)	1(8)	2)8)	--	--	5(7)	6(7)	5(7)	8(7)

Table 3 Risk Review of Limerick

Risk Index	PECO1/6/	Review2/6/	Comment
Early fatalities (per plant year of operation)	3.3E-4	5.0E-3	3/, 4/
Latent cancer fatalities (per plant year of operation)	2.8E-2	5.0E-2	4/, 5/,
Person-rems (per plant year of operation)	295	700	

1/Estimates are obtained from Limerick SARA

2/Estimates are obtained from Limerick FES (Table L.1a). See the FES for the uncertainties associated with these estimates.

3/Estimates are based on supportive medical treatment.

4/Estimate are based on crediting those plant modifications which are dicussed in Section 5.

5/Estimates include thyroid cancers.

6/Estimates correspond to "population to 50 miles" case.

Table 5.11h Estimated values of societal risks from severe accidents, per reactor-year

Consequence type		Estimated risk within the 50-mile region	Estimated risk within the entire region
1.	Early fatalities with Supportive medical treatment (persons)	5(-3)*	5(-3)
2.	Early fatalities with minimal medical treatment (persons)	8(-3)	8(-3)
3.	Early injuries (persons)	2(-2)	2(-2)
4.	Latent cancer fatalities (excluding thyroid) (persons)	4(-2)	7(-2)
5.	Latent thyroid cancer fatalities (persons)	1(-2)	1(-2)
6.	Total person-rems	7(2)	1(3)
7a.	Cost of offsite mitigation measures (1980 $)	5(4)	5(4)
7b.	Regional industrial impact costs (1980 $)		5(4)***
7c.	Plant costs (1980 $)	1(5)	
8.	Land area for long-term interdiction (m^2)**	1(3)	1(3)

*5(-3) = 5 x 10^{-3} = .005

**About 2.6 million m^2 equals to 1 mi^2.

***Excludes costs of crop and milk interdiction, which are included in 7a.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

ACCIDENT CLASS CONTRIBUTION TO MEAN EARLY FATALITIES PER REACTOR-YEAR

ACCIDENT CLASS	EARLY FATALITIES	PERCENTAGE OF TOTAL
INITIATED BY EXTERNAL EVENTS:		
CLASS IS	3.0(-3)	61
CLASS S	1.0(-3)	21
CLASS IV	0.4(-3)	8
CLASS III	0.3(-3)	6
INITIATED BY INTERNAL EVENTS:		
CLASS IV	0.2(-3)	4
TOTAL	4.9(-3)	100

ACCIDENT CLASS CONTRIBUTION TO MEAN LATENT FATALITIES PER REACTOR-YEAR

ACCIDENT CLASS	LATENT FATALITIES	PERCENTAGE OF TOTAL
INITIATED BY INTERNAL EVENTS:		
CLASS I	4.2(-2)	63
CLASS II	0.8(-2)	12
CLASS III	0.7(-2)	10
CLASS IV	0.2(-2)	3
INITIATED BY EXTERNAL EVENTS:		
CLASS IS	0.5(-2)	7
CLASS S	0.2(-2)	3
CLASS I	0.1(-2)	2
TOTAL	6.7(-2)	100

ACCIDENT CLASS CONTRIBUTION TO MEAN PERSON-REM PER REACTOR YEAR

ACCIDENT CLASS	PERSON-REM	PERCENTAGE OF TOTAL
INITIATED BY INTERNAL EVENTS:		
CLASS I	685	67
CLASS II	100	10
CLASS III	100	10
CLASS IV	20	2
INITIATED BY EXTERNAL EVENTS:		
CLASS IS	50	5
CLASS S	30	3
CLASS I	30	3
TOTAL	1015	100

CLASS I SEQUENCES

- SEQUENCE DEFINITION
- CORE MELT PHENOMENA
- CONTAINMENT FAILURE MODES
- FISSION PRODUCT RELEASE
- IMPACT OF UNCERTAINTIES ON RISK

SEQUENCE DEFINITION

- REACTOR SCRAMS
- COOLANT MAKE-UP FAILS
- PRIMARY SYSTEM REMAINS AT HIGH PRESSURE FOR APPROXIMATELY 60% OF CLASS I SEQUENCES
- CORE DAMAGE BEGINS WITH CONTAINMENT INTACT
- SUPPRESSION POOL SUBCOOLED

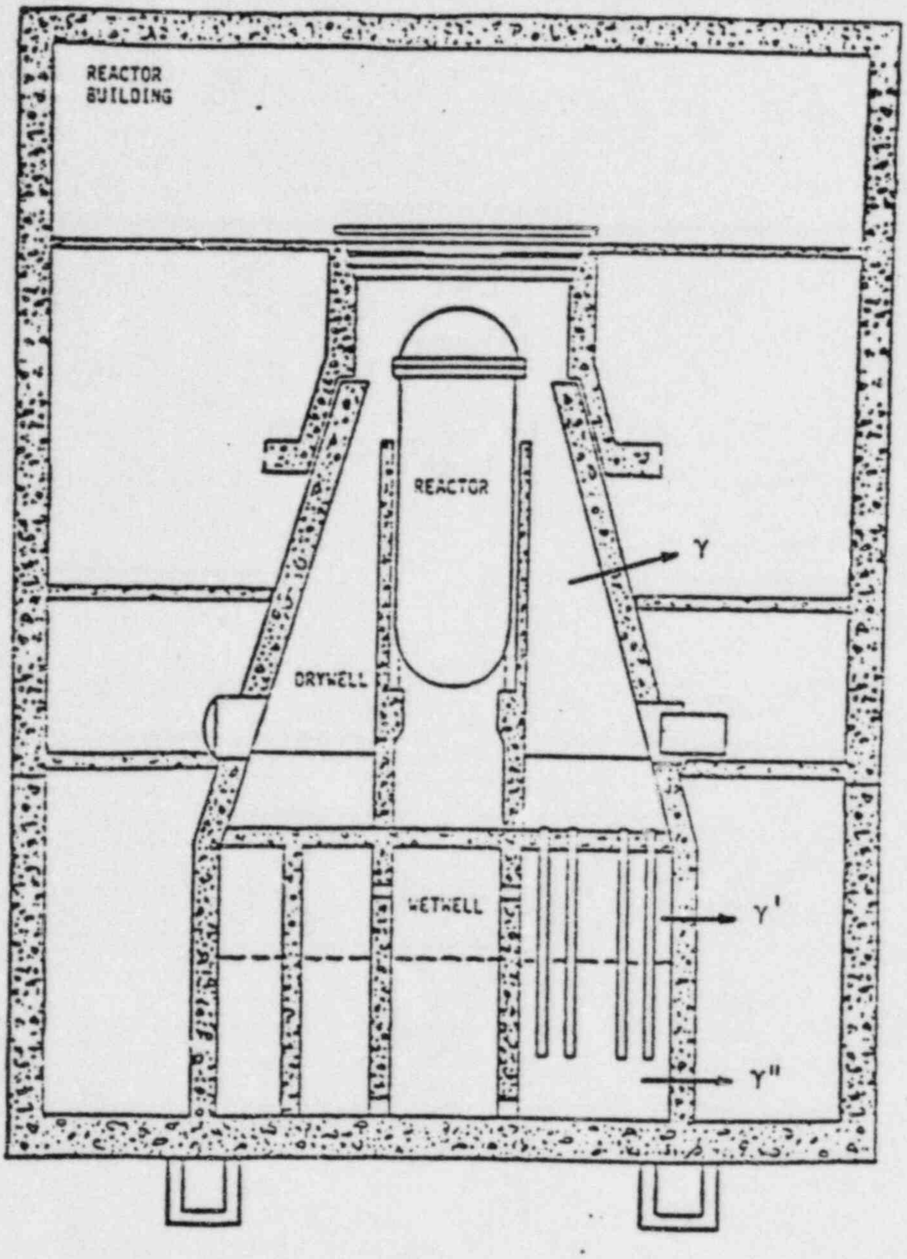
CORE MELT PHENOMENA

ISSUES:

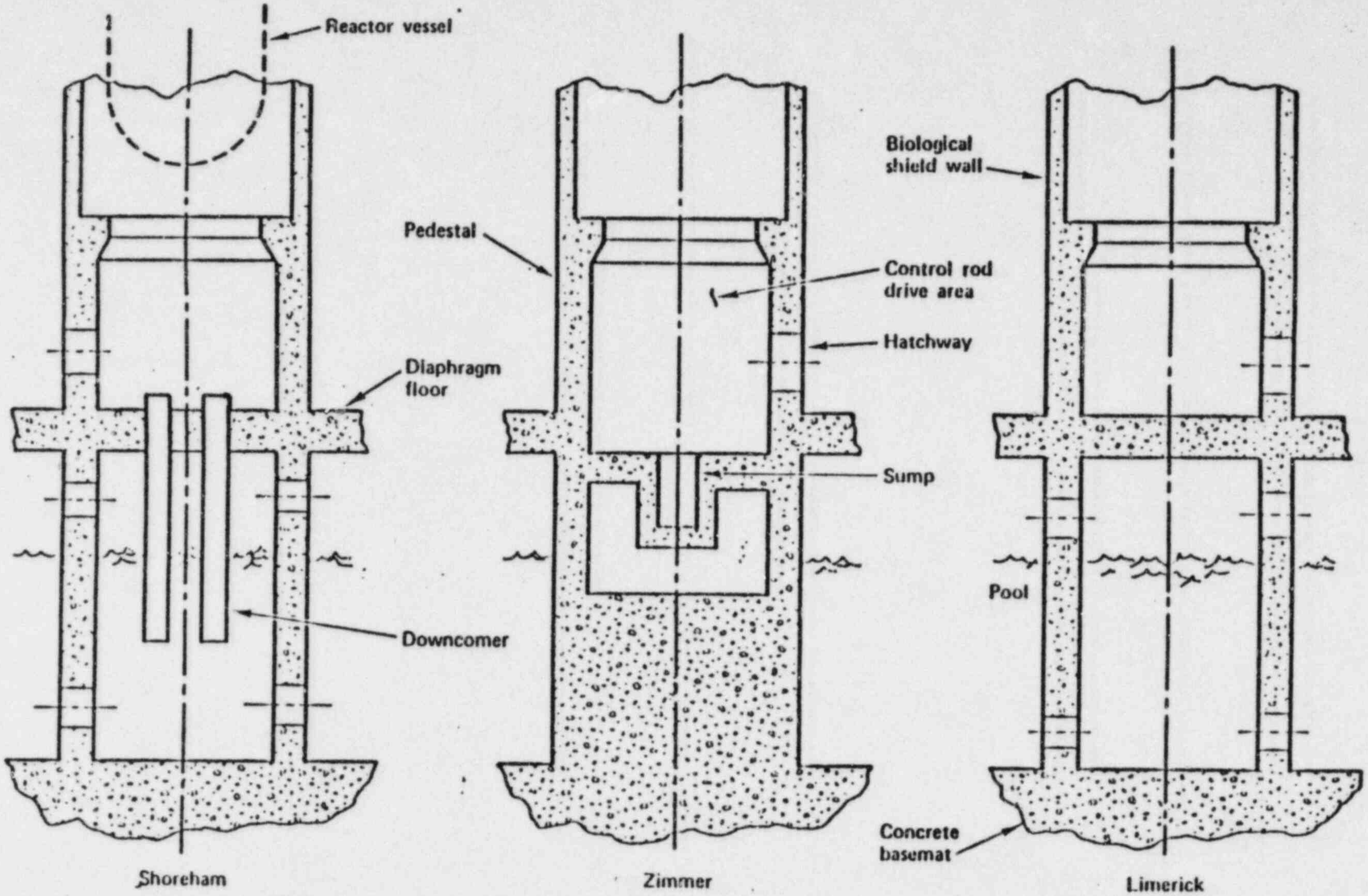
- RELEASE OF CORE MATERIALS FROM PRIMARY SYSTEM:
  - HIGH VS. LOW PRESSURE RELEASE
  - LOCAL VS. GROSS VESSEL FAILURE
- COMPOSITION OF CORE MATERIALS:
  - HIGH TEMPERATURE (MOLTEN) VS. LOWER TEMPERATURE (SLURRY)
  - RELATIVE QUANTITIES OF ZIRCALOY, STEEL, AND FUEL
  - FRACTION OF METALS OXIDIZED
- WATER SUPPLY TO CORE MATERIALS
- PLANT SPECIFIC CONSIDERATIONS

BWR WITH A MARK II CONTAINMENT

- WETWELL (AND SUPPRESSION POOL) DIRECTLY UNDERNEATH DRYWELL (AND REACTOR VESSEL)
- DIAPHRAGM FLOOR SEPARATE WETWELL AND DRYWELL
- WATER CANNOT ACCUMULATE ON DIAPHRAGM FLOOR
- HENCE, INITIAL CORE/WATER INTERACTIONS WILL BE LIMITED
- SUBSEQUENT ACCIDENT PROGRESSION DEPENDS ON HOW CORE MATERIALS PASS THROUGH DIAPHRAGM



BWR WITH A MARK II CONTAINMENT



Shoreham

Zimmer

Limerick

Variations in the Mark II pedestal configuration.

EX-VESSEL CORE MELT PHENOMENA

- BEST ESTIMATE - MOST OF CORE DEBRIS WILL BE RETAINED ON DIAPHRAGM FLOOR

- CONTAINMENT INTEGRITY WILL BE CHALLENGED BY LONG-TERM PRESSURE/TEMPERATURE BUILDUP DURING CORE/CONCRETE INTERACTIONS

- CONTAINMENT FAILS WHEN CORE DEBRIS PENETRATES 70 CM OF DIAPHRAGM FLOOR OR PRESSURE REACHES 140 PSIG

- COMBUSTION OF COMBUSTIBLE GASES PREVENTED BY INERTING

- EARLY CONTAINMENT FAILURE (STEAM EXPLOSIONS, ETC.) CONSIDERED TO BE LOW PROBABILITY

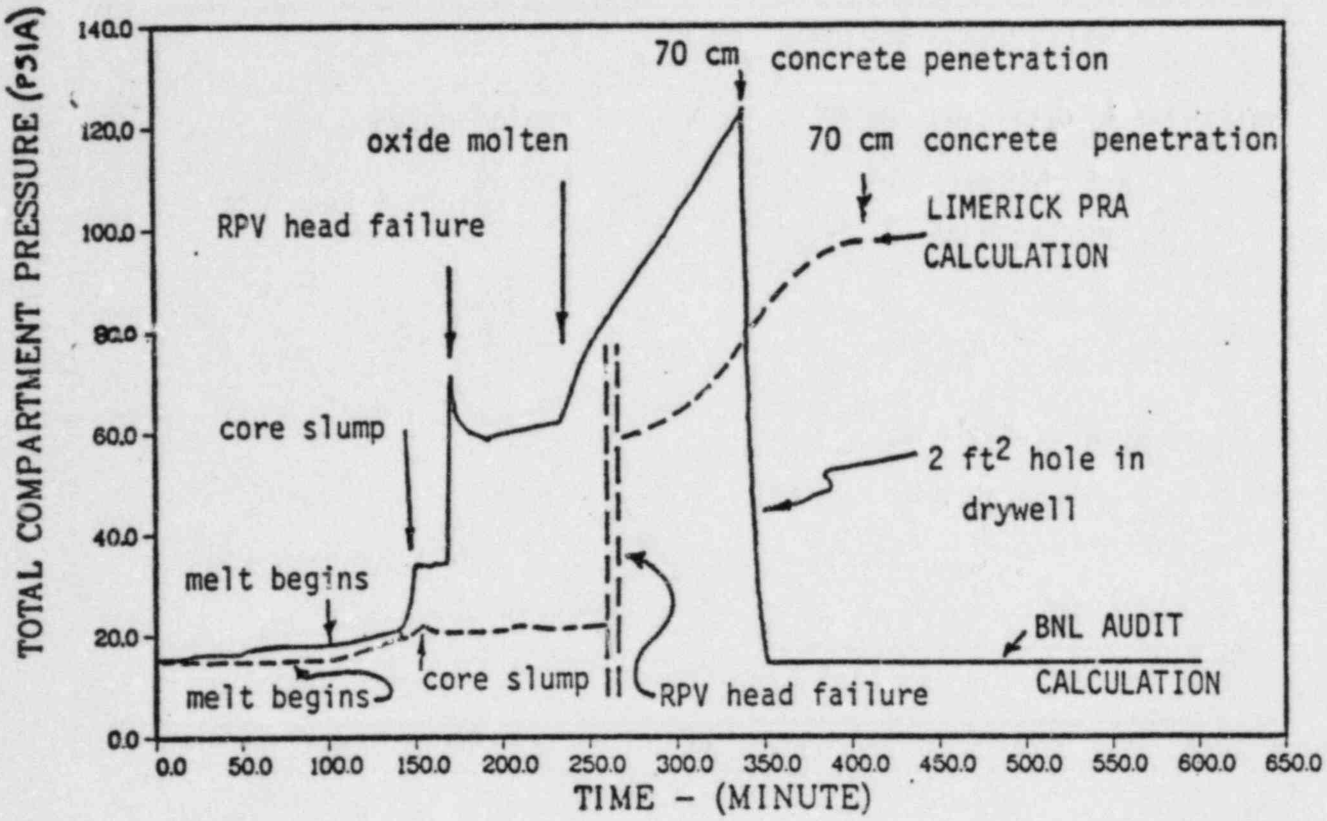


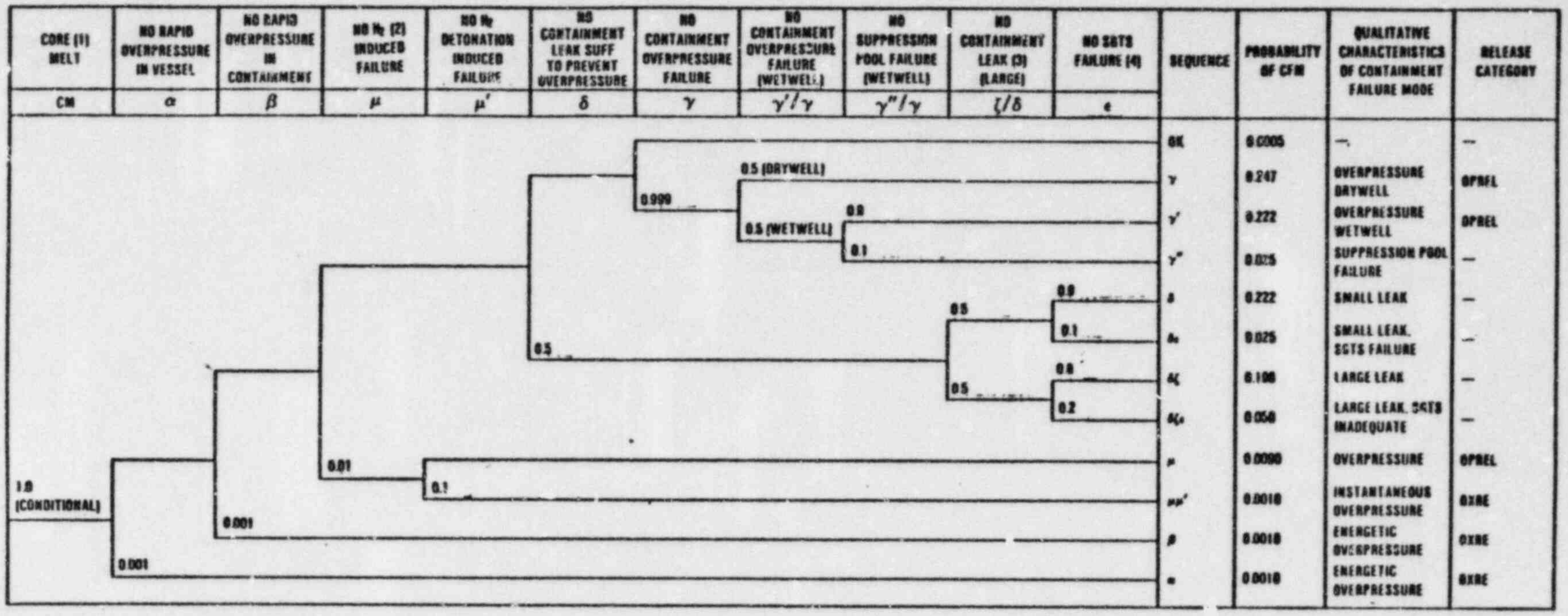
Figure 7.2 Containment Pressure History for Class 1

Table 3.2 Comparison of BNL and Limerick PRA analysis of the Class I sequences (TQUV)

Key Events	Analysis in Limerick PRA	BNL Analysis	
		NUREG/CR-3028	FES Calculation
Start of core melt (hours)	1.3	1.65	1.50
Core slump (hours)	2.5	3.08	2.42
Vessel head failure (hours)	4.3	3.71	2.90
Start of core/concrete interactions (hours)	4.3	3.71	2.90
Time (hours) core debris penetrates 70 cm of diaphragm floor causing collapse of floor and containment failure	6.5	6.12	5.17
Pressure at containment failure (psia)	88	113	118

CONTAINMENT FAILURE MODES

- EARLY VS. LATE CONTAINMENT FAILURE:
  - STEAM EXPLOSIONS
  - H_2 INDUCED FAILURES
- LEAKAGE VS. OVERPRESSURIZATION FAILURE
- FAILURE LOCATION:
  - DRYWELL VS. WETWELL
  - WETWELL ABOVE POOL VS. BELOW POOL
- FOR LEAKAGE FAILURE:
  - EFFECTIVENESS OF SGTS



(1) CONTAINMENT FAILURE MAY HAVE OCCURRED PRIOR TO CORE MELT. IN THOSE CASES (CLASS II AND CLASS IV) THE CONTAINMENT FAILURE MODES ARE ONLY USED AS MECHANISMS FOR RELEASE FRACTION DETERMINATION.

(2) ASSUMES THAT H_2 EXPLOSION IN CONTAINMENT CAUSES OVERPRESSURE FAILURE WITH DIRECT PATHWAY TO OUTSIDE ATMOSPHERE.

(3) LEAKAGE AT 2400 VOLUME PERCENT/DAY.

(4) FAILURE STANDBY GAS TREATMENT SYSTEM.

Figure 6.2 Limerick PRA Containment Event Tree for Classes I, II, and III Event Sequences.

Table 2.7 ' Assignment of conditional probabilities

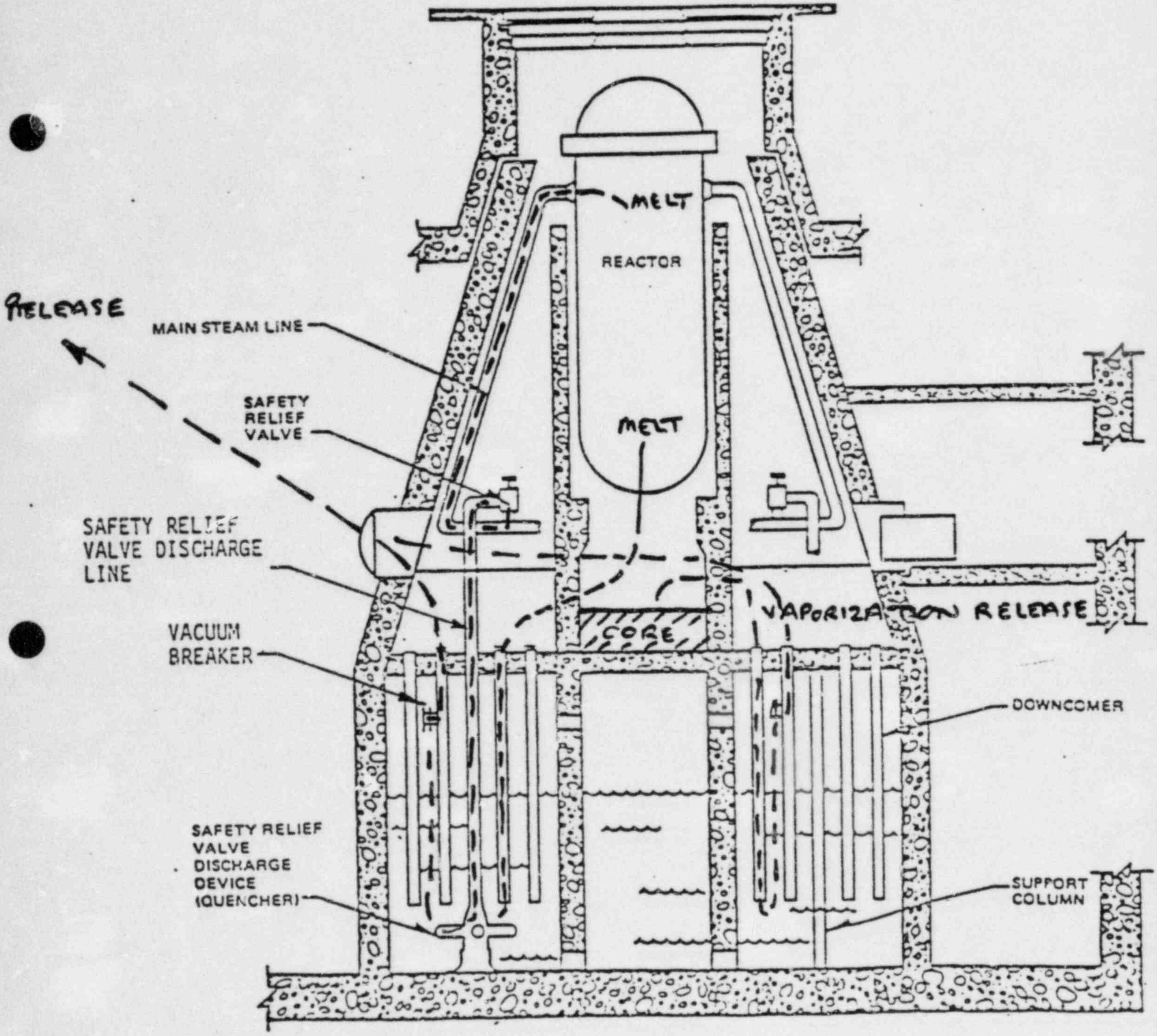
Damage States	Containment Failure Modes and Release Paths							
	DW	WW	$\overline{\text{WW}}$	SE	HB	LGT	$\overline{\text{LGT}}$	No Core Melt
I-S	0.247	0.223	0.025	0.0001	0.01	0.222	0.273	0
I-T	0.247	0.223	0.025	0.0001	0.01	0.222	0.273	0
II-T	0.250	0.225	0.025	0.0001	0	0	0	0.5
III-T	0.247	0.223	0.025	0.0001	0.01	0.222	0.273	0
IV-A	0.500	0.45	0.05	0.0001	0	0	0	0
IV-T	0.500	0.45	0.05	0.0001	0	0	0	0
IS-C	1*	0	0	0.0001	0	0	0	0
IS-$\overline{\text{C}}$	1*	0	0	0.0001	0	0	0	0
S-H20	0	0	1**	0.0001	0	0	0	0
S-$\overline{\text{H20}}$	0	0	1**	0	0	0	0	0

*In the LGS-SARA, this failure mode was considered similar to a drywell (DW) failure mode, however, this should not be interpreted as a failure location in the drywell. Class IS sequences result in failure of the RHR suction lines, which partially drains the suppression pool exposing the downcomers but leaving the quenchers submerged. Thus, the melt release will be scrubbed by the pool (similar to WW failure mode) and the vaporization release will not be scrubbed by the pool (similar to DW failure mode).

**Again, assigning the $\overline{\text{WW}}$ failure mode to Class S sequences relates to the fission product release path (and lack of suppression pool scrubbing) rather than to the failure location.

FISSION PRODUCT RELEASE

- BASED ON RSS METHODS
- GAP, MELT, AND VAPORIZATION RELEASE AS SPECIFIED IN RSS
- SUPPRESSION POOL DECONTAMINATION FACTOR (DF)=100
- THERMAL HYDRAULICS BASED ON MARCH 1.1 CODE
- FISSION PRODUCT TRANSPORT BASED ON CORRAL CODE



CLASS I TRANSIENTS, FAILURE IN DRYWELL

($C_1\gamma$, I-T/DW)

AS MODELED IN LGS-PRA & NUREG/CR-3028

Table 4.2 Fission product release fractions for Class 1

ASSESSMENT	LGS - PRA	NUREG-3028	FES CALCULATION		
FAILURE MODE	$C_1\gamma$	$C_1\gamma$	I-T/DW	I-T/WW	I-T/$\overline{WW}$
OXIDATION RELEASE	Yes	Yes	No	No	No
Xe - Kr	1.0	.939(-1)	1.0	1.0	1.0
Organic Iodine	---	---	6.99(-3)	6.99(-3)	6.99(-3)
I_2	1.1(-1)	9.3(-3)	1.78(-3)	1.48(-4)	2.09(-4)
Cs	9(-2)	2.0(-2)	1.88(-2)	3.11(-4)	9.19(-4)
Te	1.6(-2)	4.6(-2)	8.41(-2)	1.23(-3)	2.16(-3)
Ba	1.0(-2)	1.7(-3)	9.94(-4)	1.91(-5)	8.22(-5)
Ru	3.0(-3)	3.0(-3)	4.95(-3)	7.39(-5)	1.39(-4)
La	3.0(-4)	6.1(-4)	9.89(-4)	1.46(-5)	2.61(-5)
DF for I_2	100	100	100	100	100
DF for Aerosols	100	100	100	100	100
Core Melt Start	1.3	1.75	1.5	1.5	1.5
Core Melt End	2.5	2.43	2.42	2.42	2.42
1st Vap. Release			2.90	2.90	2.90
2nd Vap. Release			3.40	3.40	3.40
Vap. Release End			4.90	4.90	4.90
Containment Fail	6.5	5.23	5.17	5.17	5.17

IMPACT OF UNCERTAINTIES ON RISK

- HIGH PRESSURE MELT EJECTION:
  - POTENTIAL FOR DIRECT CONTAINMENT HEATING AND EARLY FAILURE
- STEAM EXPLOSIONS:
  - POTENTIAL FOR EARLY FAILURE
- FAILURE OF PRIMARY SYSTEM DURING CORE MELT:
  - POOL BYPASS
- CONTAINMENT FAILURE LOCATION AND CHARACTERISTICS
- ASSESS ABOVE UNCERTAINTIES ON TOTAL RISK ESTIMATES

FAILURE MODE CONTRIBUTION TO CLASS I
PERSON-REM PER REACTOR YEAR
CLASS FREQUENCY 8.3(-5)*

FAILURE MODE	CONDITIONAL PROBABILITY	CONDITIONAL MEAN PERSON-REM	PERSON-REM
FAILURE IN DRYWELL	0.247	1(7)	200
FAILURE IN WETWELL	0.223	5(5)	9
FAILURE IN WETWELL WITH LOSS OF POOL	0.025	8(5)	2
STEAM EXPLOSION	0.0001	4(7)	0.3
HYDROGEN FAILURE	0.01	2(7)	17
LEAKAGE WITH SGTS	0.222	4(5)	7
LEAKAGE WITHOUT SGTS	0.273	2(7)	450
TOTAL			685

*8.3(-5) = 8.3×10^{-5}

FAILURE MODE CONTRIBUTION TO CLASS I
EARLY FATALITIES PER REACTOR YEAR
CLASS FREQUENCY 8.3(-5)*

FAILURE MODE	CONDITIONAL PROBABILITY	CONDITIONAL MEAN EARLY FATALITIES	EARLY FATALITIES
FAILURE IN DRYWELL	0.247	0	0
FAILURE IN WETWELL	0.223	0	0
FAILURE IN WETWELL WITH LOSS OF POOL	0.025	0	0
STEAM EXPLOSION	0.0001	200	2(-6)
HYDROGEN FAILURE	0.01	10	8(-6)
LEAKAGE WITH SGTS	0.222	0	0
LEAKAGE WITHOUT SGTS	0.273	0.5	1(-5)
TOTAL			2(-5)

*8.3(-5) = 8.3×10^{-5}

CLASS I UPPER BOUND UNCERTAINTY ESTIMATE

- ASSUME NON-MECHANISTICALLY THAT CONTAINMENT FAILS AT VESSEL FAILURE (STEAM EXPLOSION SOURCE TERM) FOR ALL CLASS I SEQUENCES

- PERSON-REM = $8.3x10^{-5}$ x $4x10^{7}$ = 3300

- EARLY FATALITIES = $8.3x10^{-5}$ x 200 = $1.6x10^{-2}$

CLASS I LOWER BOUND UNCERTAINTY ESTIMATE

- ASSUME CONTAINMENT LEAKAGE PREVENTS OVERPRESSURE FAILURES FOR ALL CLASS I SEQUENCES AND SGTS IS EFFECTIVE

- PERSON-REM = $8.3\times10^{-5} \times 4\times10^{+5} = 33$

- EARLY FATALITIES ARE ZERO

IMPACT OF CLASS I UNCERTAINTY ON RISK PER REACTOR YEAR

RISK INDEX	EARLY FATALITIES		PERSON-REM	
	CLASS I	TOTAL	CLASS I	TOTAL
UPPER BOUND	1.6(-2)	2.1(-2)	3300	3630
LGS-FES	2(-5)	5(-3)	685	1015
LOWER BOUND	ZERO	5(-3)	33	363

CLASS IV SEQUENCES

- SEQUENCE DEFINITION
- CONTAINMENT FAILURE MODES
- FISSION PRODUCT RELEASE
- IMPACT OF UNCERTAINTIES ON RISK

ATWS WITH CONTINUED COOLANT INJECTION

- REACTOR DOES NOT SCRAM
- COOLANT INJECTION CONTINUES
- SUPPRESSION POOL STARTS BOILING
- CONTAINMENT FAILURE OCCURS RAPIDLY DUE TO STEAM PARTIAL PRESSURE
- COOLANT INJECTION FAILS
- CORE MELTS INTO A FAILED CONTAINMENT
- UNCERTAINTY RELATED TO PRIMARY SYSTEM T/H AND NEUTRONICS
- CONTAINMENT RESPONSE CALCULATIONS STRAIGHTFORWARD

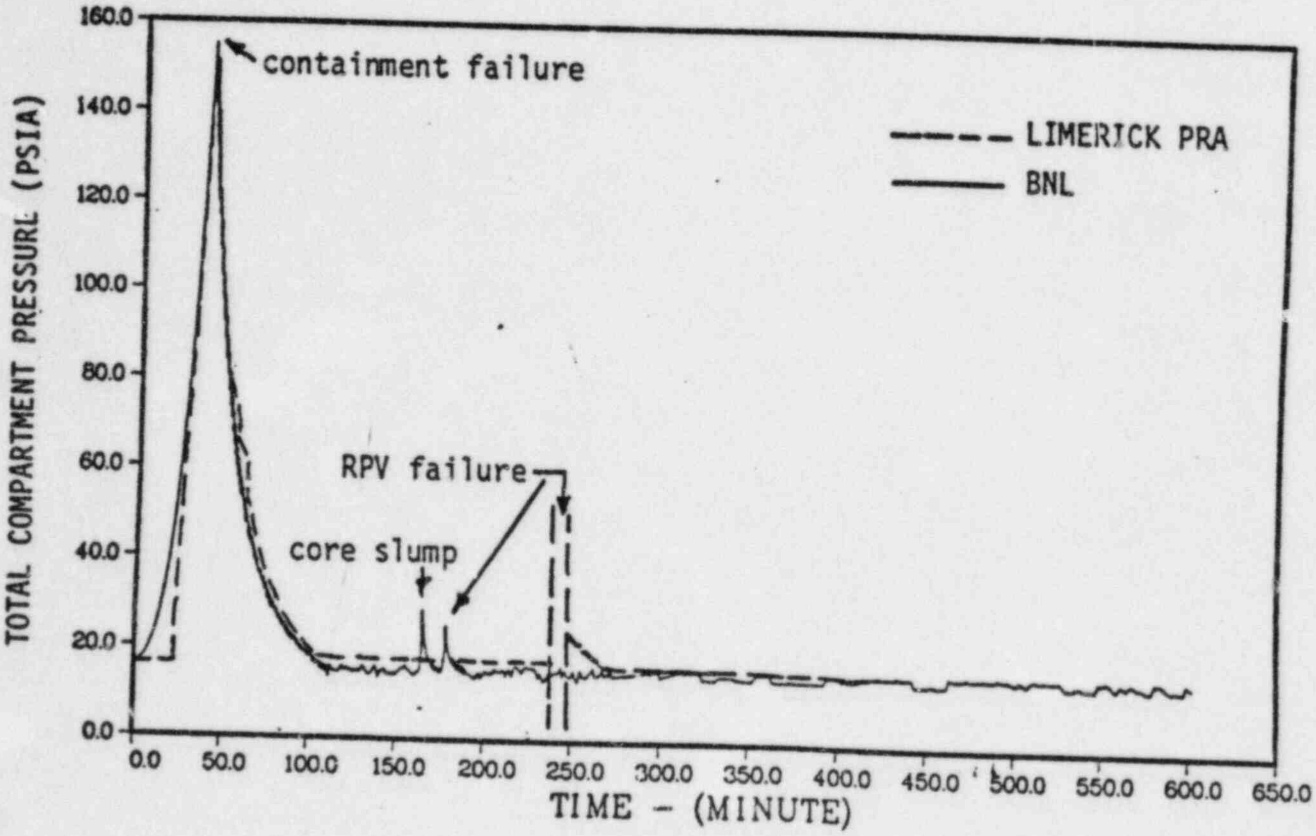
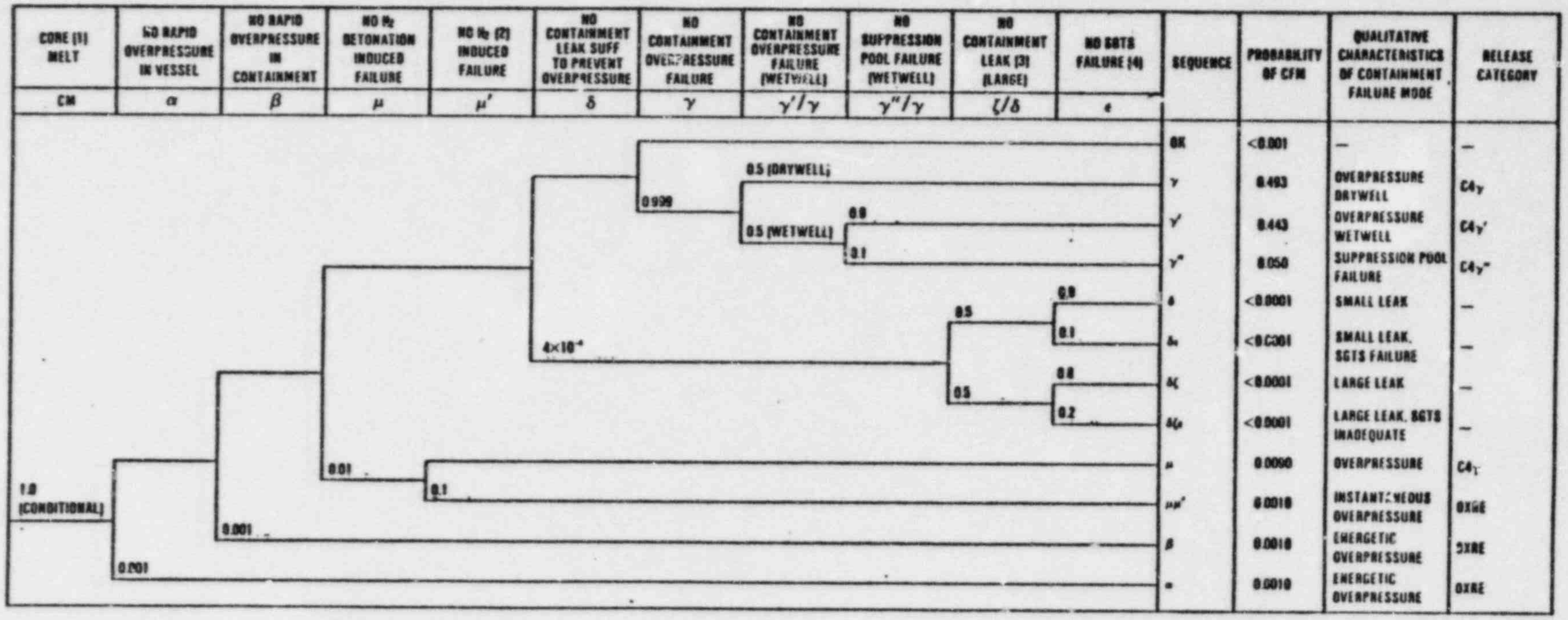


Figure 7.9 Containment Pressure History for Class IV. (Case ATW-IV)



(1) CONTAINMENT FAILURE MAY HAVE OCCURRED PRIOR TO CORE MELT. IN THOSE CASES (CLASS II AND CLASS IV). THE CONTAINMENT FAILURE MODES ARE ONLY USED AS MECHANISMS FOR RELEASE FRACTION DETERMINATION.

(2) ASSUMES THAT H_2 EXPLOSION IN CONTAINMENT CAUSES OVERPRESSURE FAILURE WITH DIRECT PATHWAY TO OUTSIDE ATMOSPHERE.

(3) LEAKAGE AT 2400 VOLUME PERCENT/DAY.

(4) FAILURE STANDBY GAS TREATMENT SYSTEM.

Figure 6.3 Limerick PRA Containment Event Tree for Class IV Event Sequences.

Table 2.7 Assignment of conditional probabilities

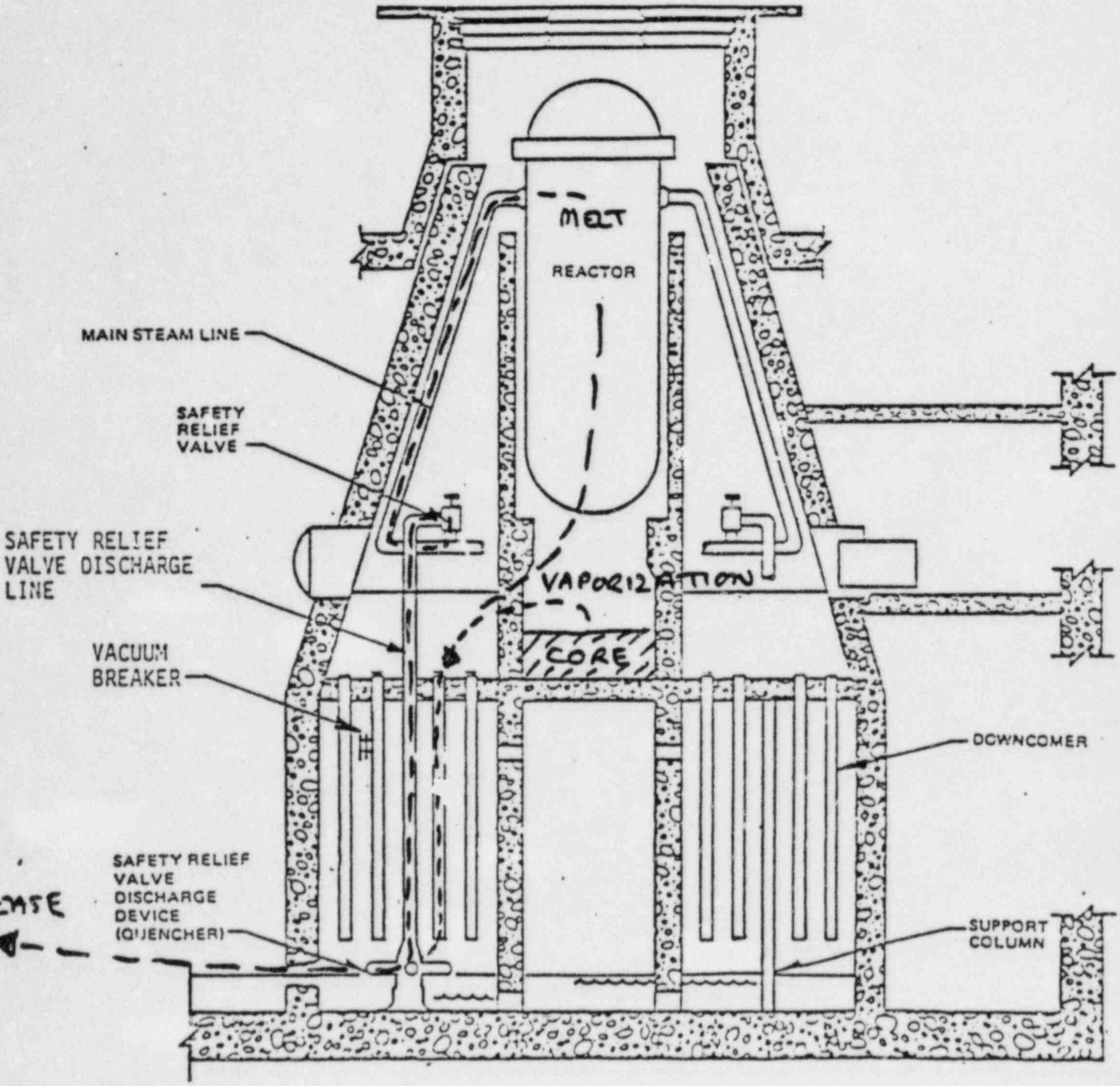
Damage States	Containment Failure Modes and Release Paths							
	DW	WW	$\overline{WW}$	SE	HB	LGT	$\overline{LGT}$	No Core Melt
I-S	0.247	0.223	0.025	0.0001	0.01	0.222	0.273	0
I-T	0.247	0.223	0.025	0.0001	0.01	0.222	0.273	0
II-T	0.250	0.225	0.025	0.0001	0	0	0	0.5
III-T	0.247	0.223	0.025	0.0001	0.01	0.222	0.273	0
IV-A	0.500	0.45	0.05	0.0001	0	0	0	0
IV-C	0.500	0.45	0.05	0.0001	0	0	0	0
IS-C	1*	0	0	0.0001	0	0	0	0
IS-$\overline{C}$	1*	0	0	0.0001	0	0	0	0
S-H2O	0	0	1**	0.0001	0	0	0	0
S-$\overline{H2O}$	0	0	1**	0	0	0	0	0

*In the LGS-SARA, this failure mode was considered similar to a drywell (DW) failure mode, however, this should not be interpreted as a failure location in the drywell. Class IS sequences result in failure of the RHR suction lines, which partially drains the suppression pool exposing the downcomers but leaving the quenchers submerged. Thus, the melt release will be scrubbed by the pool (similar to WW failure mode) and the vaporization release will not be scrubbed by the pool (similar to DW failure mode).

**Again, assigning the $\overline{WW}$ failure mode to Class S sequences relates to the fission product release path (and lack of suppression pool scrubbing) rather than to the failure location.

FISSION PRODUCT RELEASE

- BASED ON RSS METHODS
- GAP, MELT, AND VAPORIZATION RELEASE AS SPECIFIED IN RSS
- SUPPRESSION POOL SATURATED, DF=1:
  - HENCE, FAILURE LOCATION HAS MINOR IMPACT
- THERMAL HYDRAULICS BASED ON MARCH 1.1 CODE
- FISSION PRODUCT TRANSPORT BASED ON CORRAL CODE

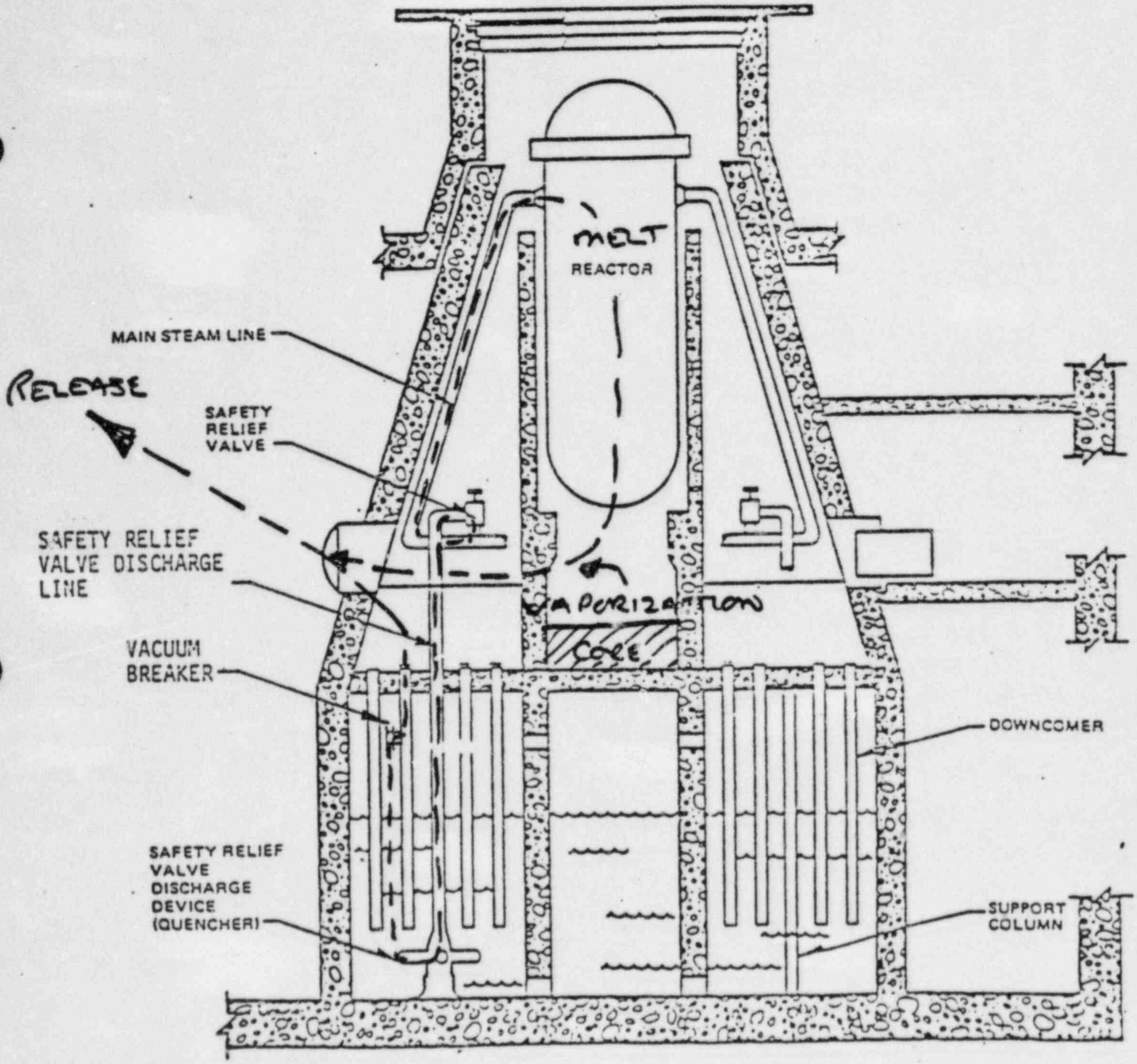


CLASS IV ATWS, FAILURE IN WETWELL

BELOW POOL LEVEL

($C_4\gamma$", IV-T/$\overline{WW}$)

AS MODELED IN LGS-PRA AND NUREG/CR-3028



CLASS IV ATWS WITH LOCA AND DRYWELL FAILURE

(IV-A/DW)

AS MODELED IN LGS-PRA AND NUREG/CR-3028

Table 4.5 Fission product release fractions for Class IV (failure location DW)

ASSESSMENT	LGS - PRA	NUREG-3028	DES CALCULATION	FES CALCULATION
FAILURE MODE	$C_4\gamma$	$C_4\gamma$	IV-T/DW	IV-T/DW
OXIDATION RELEASE	Yes	Yes	Yes	Yes
Xe - Kr	1.0	1.0	9.99(-1)	9.99(-1)
Organic Iodine	---	---	6.99(-3)	6.95(-3)
I_2	2.61(-1)	1.54(-1)	9.39(-1)	4.74(-1)
Cs	2.02(-1)	7.49(-1)	8.61(-1)	4.86(-1)
Te	4.34(-1)	7.47(-1)	8.62(-1)	5.09(-1)
Ba	2.90(-2)	8.60(-2)	9.40(-2)	5.54(-2)
Ru	9.50(-2)	1.10(-1)	1.49(-1)	8.55(-2)
La	5.20(-3)	1.03(-2)	1.15(-2)	6.82(-3)
DF for I_2	10	10	1	1
DF for Aerosols	10	1	1	1
Core Melt Start	1.2	1.25	1.13	1.13
Core Melt End	2.2	2.7	2.20	2.20
1st Vap. Release			2.47	2.47
2nd Vap. Release			2.77	2.77
Vap. Release End			4.47	4.47
Containment Fail	.67	.67	.67	.67

Table 4.7 Fission product release fractions for Class IV

(failure location $\overline{WW}$ below wetwell waterline)

ASSESSMENT	LGS - PRA	NUREG-3028	DES CALCULATION	FES CALCULATION
FAILURE MODE	$C_4\gamma''$	$C_4\gamma''$	IV-T/$\overline{WW}$	IV-T/$\overline{WW}$
OXIDATION RELEASE	Yes	Yes	Yes	Yes
Xe - Kr	1.0	1.0	1.0	9.98(-1)
Organic Iodine	---	---	6.99(-3)	6.95(-3)
I_2	7.30(-1)	7.08(-1)	8.74(-1)	4.68(-1)
Cs	7.0(-1)	7.49(-1)	8.04(-1)	5.18(-1)
Te	5.50(-1)	7.47(-1)	5.82(-1)	4.81(-1)
Ba	9.0(-2)	8.60(-2)	9.60(-2)	5.96(-2)
Ru	1.20(-1)	1.10(-1)	1.38(-1)	8.31(-2)
La	7.0(-3)	1.03(-2)	7.90(-3)	6.51(-3)
DF for I_2	—	—	—	—
DF for Aerosols	—	—	—	—
Core Melt Start	1.2	1.25	1.13	1.13
Core Melt End	2.2	2.7	2.2	2.2
1st Vap. Release			2.47	2.47
2nd Vap. Release			2.77	2.77
Vap. Release End			4.47	4.47
Containment Fail	.67	.67	.67	.67

Table 4.6 Fission product release fractions for Class IV (failure location WW)

ASSESSMENT	LGS - PRA	NUREG-3028	DES CALCULATION	FES CALCULATION
FAILURE MODE	$C_4\gamma'$	$C_4\gamma'$	IV-T/WW	IV-T/WW
OXIDATION RELEASE	Yes	Yes	Yes	Yes
Xe - Kr	1.0	1.0	1.0	9.99(-1)
Organic Iodine	---	---	6.99(-3)	6.95(-3)
I_2	7.0(-2)	9.80(-2)	9.39(-1)	4.61(-1)
Cs	9.0(-2)	7.49(-1)	7.72(-1)	4.81(-1)
Te	2.0(-1)	7.47(-1)	6.88(-1)	4.45(-1)
Ba	1.6(-2)	8.60(-2)	9.0(-2)	5.60(-2)
Ru	8.8(-2)	1.10(-1)	1.19(-1)	7.81(-2)
La	6.0(-3)	1.03(-2)	9.40(-3)	6.03(-3)
DF for I_2	10	10	1	1
DF for Aerosols	10	1	1	1
Core Melt Start	1.2	1.25	1.13	1.13
Core Melt End	2.2	2.7	2.2	2.2
1st Vap. Release			2.47	2.47
2nd Vap. Release			2.77	2.77
Vap. Release End			4.47	4.47
Containment Fail	.67	.67	.67	.67

NRC DEVELOPMENT OF NEW METHODS

- UNCERTAINTIES ASSOCIATED WITH THE SOURCE-TERM DISCUSSED IN NUREG-0772

- ACTIVITIES OF ACCIDENT SOURCE TERM PROGRAM OFFICE (ASTPO):

  - RADIONUCLIDE RELEASE UNDER SPECIFIC LWR ACCIDENT CONDITIONS: BMI-2104

  - CONTAINMENT LOADS WORKING GROUP (CLWG)

  - CONTAINMENT PERFORMANCE WORKING GROUP (CPWG) NUREG-1037 (FOURTH DRAFT)

  - QUANTITATIVE UNCERTAINTY EVALUATION OF SOURCE TERM (QUEST)

IMPACT OF UNCERTAINTIES ON RISK

- CLASS IV SOURCE TERM CALCULATIONS ARE VERY CONSERVATIVE:
  - HENCE, VERY LITTLE POTENTIAL FOR HIGHER SOURCE TERMS
  - MORE POTENTIAL FOR LOWER SOURCE TERMS (SATURATED POOL SCRUBBING)

- CLASS IV SOURCE TERMS CONTRIBUTE:
  - 12% TO EARLY FATALITIES
  - 2% TO PERSON-REM

- HENCE, UNCERTAINTIES IN CLASS IV SOURCE TERMS WILL HAVE RELATIVELY SMALL IMPACT ON OVERALL RISK

COMPARISON OF SOURCE-TERM METHODS

Analysis/Function	"WASH-1400" Methods	Newer Methods
Fission product inventory	ORIGEN computer code. Calculates radionunuclide inventories	Core Inventory based on WASH-1400 scaled to core power
Thermal-hydraulic conditions	Boil code used for primary system behavior, hand calculations for containment	MARCH 2 computer code for primary system and containment. MERGE code for primary system thermal-hydraulics
Fission product release from core material	Specified release fractions for different phases: gap, melt, vaporization and oxidation	CORSOR computer code to predict core release (gap and melt); VANESA to predict "vaporization" release.
Release of fission products to containment	No deposition in primary system	TRAP-MELT computer code to predict holdup in primary system
Fission product attenuation in suppression pool	Decontamination factor of 1 or 100 (pool in saturation or not)	SPARC computer code to calculate decontamination factor
Atmospheric release of fission products	CORRAL computer code to predict fission product release to atmosphere	NAUA-4 computer code to calculate release of fission products from containment

POTENTIAL IMPACT OF NEW METHODS

- IN-VESSEL RELEASE OF FISSION PRODUCTS:
  - TIMING AND CHEMICAL FORM
  - PRIMARY SYSTEM RETENTION AND RE-EMISSION
- EX-VESSEL RELEASE OF FISSION PRODUCTS:
  - TIMING AND CHEMICAL FORM
- FISSION PRODUCT TRANSPORT IN CONTAINMENT AND AUXILIARY BUILDINGS:
  - INCREASED AGGLOMERATION AND SETTLING
  - IMPACT OF SUPPRESSION POOLS

OBJECTIVES OF THE CWG

- TO MECHANISTICALLY MODEL CONTAINMENT BEHAVIOR UNDER SEVERE ACCIDENT CONDITIONS
- TO SYSTEMATICALLY ADDRESS A NUMBER OF STANDARD PROBLEMS APPLICABLE TO REPRESENTATIVE PLANTS FOR THE SIX CONTAINMENT TYPES UNDER CONSIDERATION
- FOR EACH STANDARD PROBLEM THE GROUP WILL:
  - ESTABLISH STANDARD METHODOLOGY WHERE POSSIBLE
  - PROVIDE A BROAD CONSENSUS VIEW OF AREAS WHERE CALCULATIONS CAN BE PERFORMED WITH CONFIDENCE
  - IDENTIFY WHERE UNCERTAINTIES EXIST AND PERFORM SENSITIVITY STUDIES

APPROACH

- STANDARD PROBLEMS SELECTED TO ADDRESS ACCIDENT PHENOMENOLOGY WITH POTENTIALLY SEVERE CONTAINMENT LOADING:
  - SELECTION OF PROBLEMS BASED ON INSIGHTS GAINED FROM EXTENSIVE ANALYSES BY NRC CONTRACTORS AND INDUSTRY
- EACH SAMPLE PROBLEM ANALYZED BY SEVERAL DIFFERENT ORGANIZATIONS
- RESULTS ARE COMPARED IN OPEN FORUM WITH EXTENSIVE PEER REVIEW
- CONTAINMENT LOADS THEN PROVIDED TO CONTAINMENT PERFORMANCE WORKING GROUP (CPWG)

DEFINITION OF STANDARD PROBLEM

- FOCUS OF MARK I AND MARK II STANDARD PROBLEM:
  - PRESSURE/TEMPERATURE RESPONSE DURING CORIUM/CONCRETE INTERACTIONS

- ISSUE TO BE ADDRESSED (BY CPWG):
  - MODE (OVERPRESSURE VS. TEMPERATURE) AND TIMING OF CONTAINMENT FAILURE

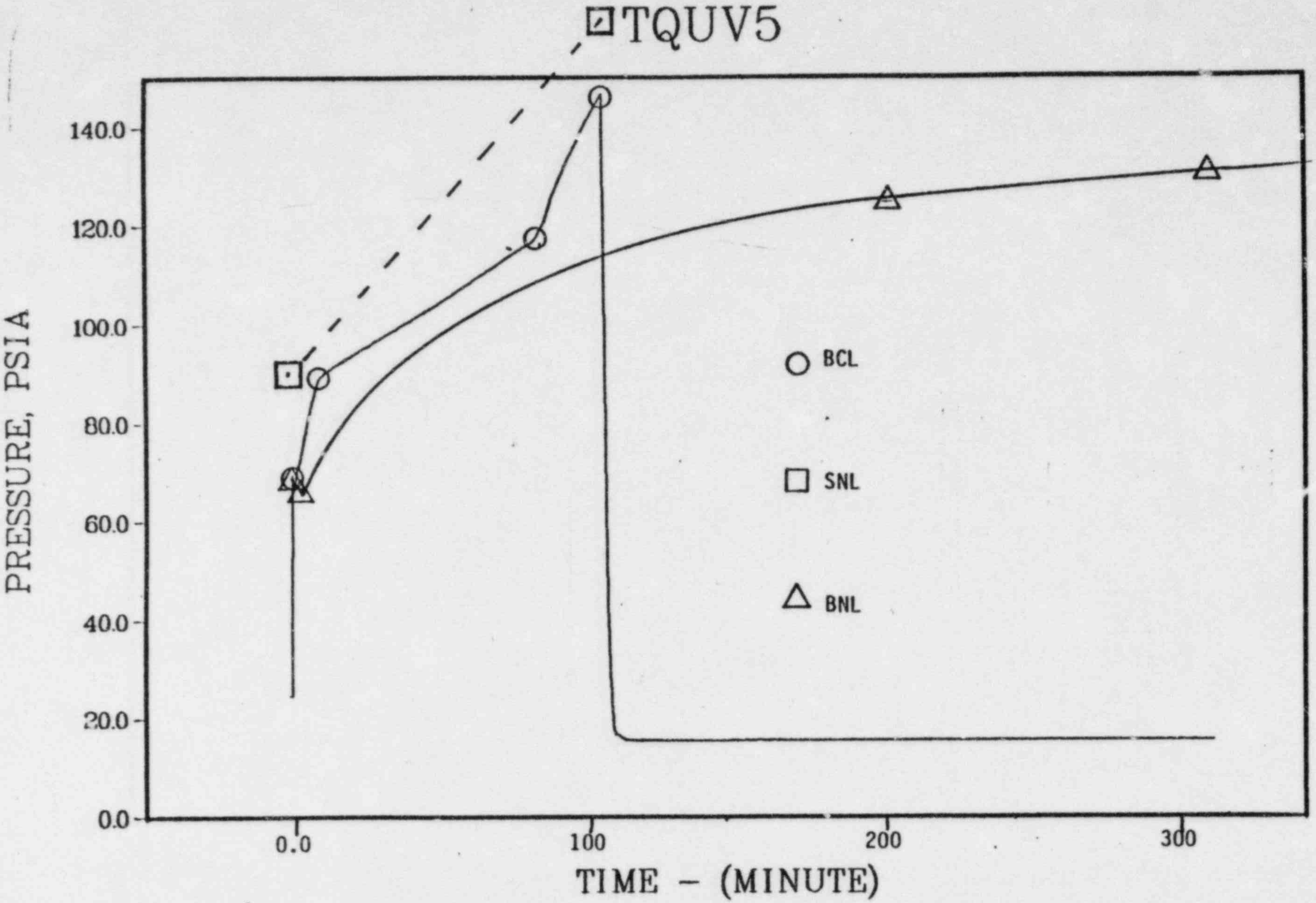
- SENSITIVITY STUDIES:
  - INITIAL CORIUM TEMPERATURE
  - ZIRCONIUM, STEEL, AND UO_2 MASS IN CORIUM
  - METAL OXIDATION IN-VESSEL
  - EX-VESSEL CORIUM DISPERSAL
  - CONCRETE TYPE

CALCULATIONAL METHODS

- ORNL:
    - MARCH1.1B (INTER USED TO MODEL CORIUM/ CONCRETE INTERACTIONS)
- BCL:
    - MARCH2 (WITH MODIFIED INTER)
- BNL/PURDUE:
    - MARCH1.1B (STAND ALONE CORCON MOD 1)
    - MARCH1.1 (STAND ALONE CORCON MOD 1)
    - MARCH2 (STAND ALONE CORCON MOD 1)
- SANDIA:
    - MARCON (MARCH2 LINKED WITH CORCON MOD 2 PLUS OTHER MODIFICATIONS)

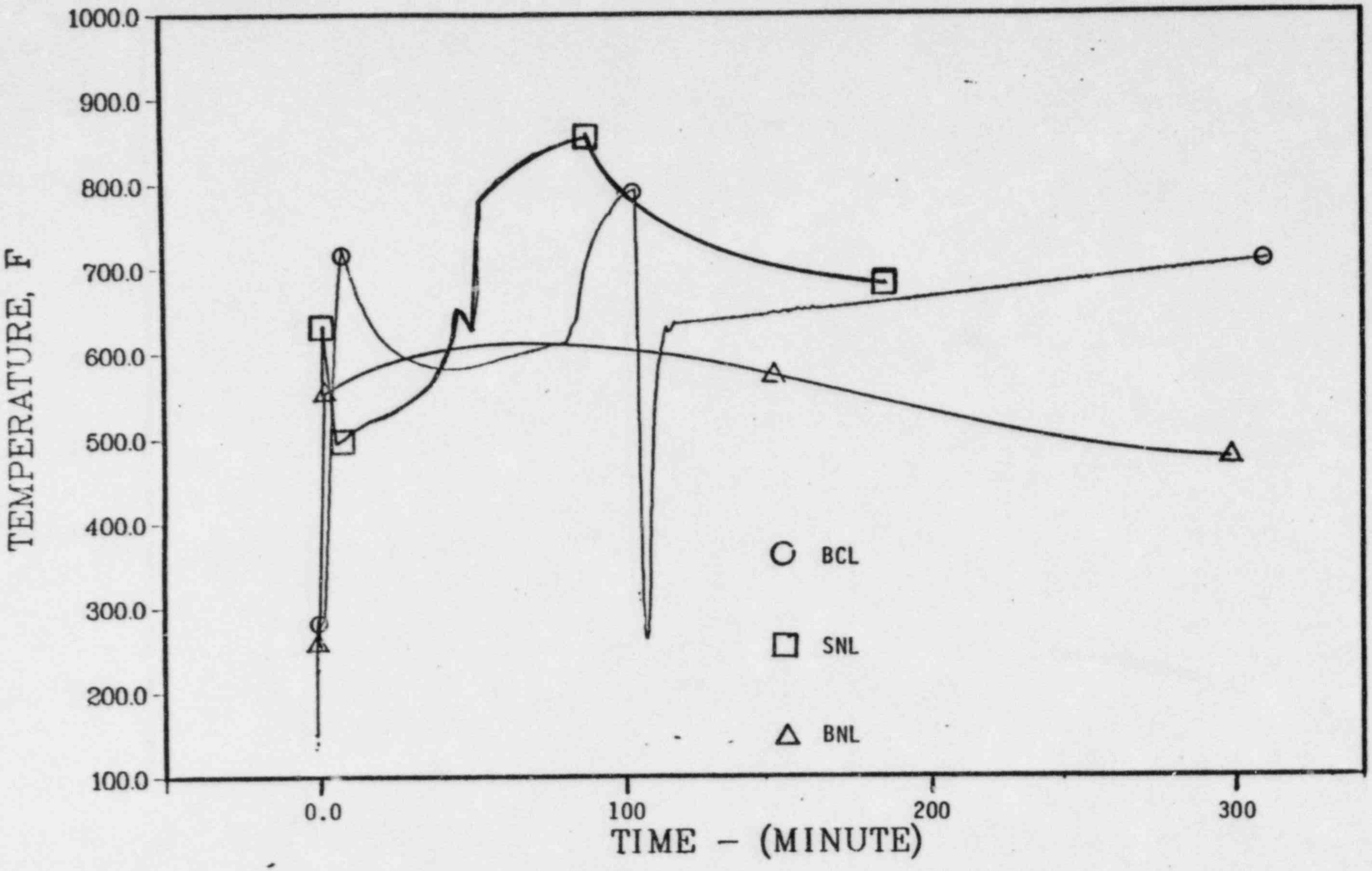
Mark II (TQUV)

	5	5a	5b	5c	5d	6	7	7a	8
Corium Spread (m)	5					3	5		3
Debris Temp (^oF)	4130					2700	4130		2700
Concrete Type	L					L	B		B
Free H_2O (%)	3	6				3	4	8	4
Steel in Corium (lb)	140K		85K			140K	140K		140K
Pool Losses (%)	0			25	50	0	0		0

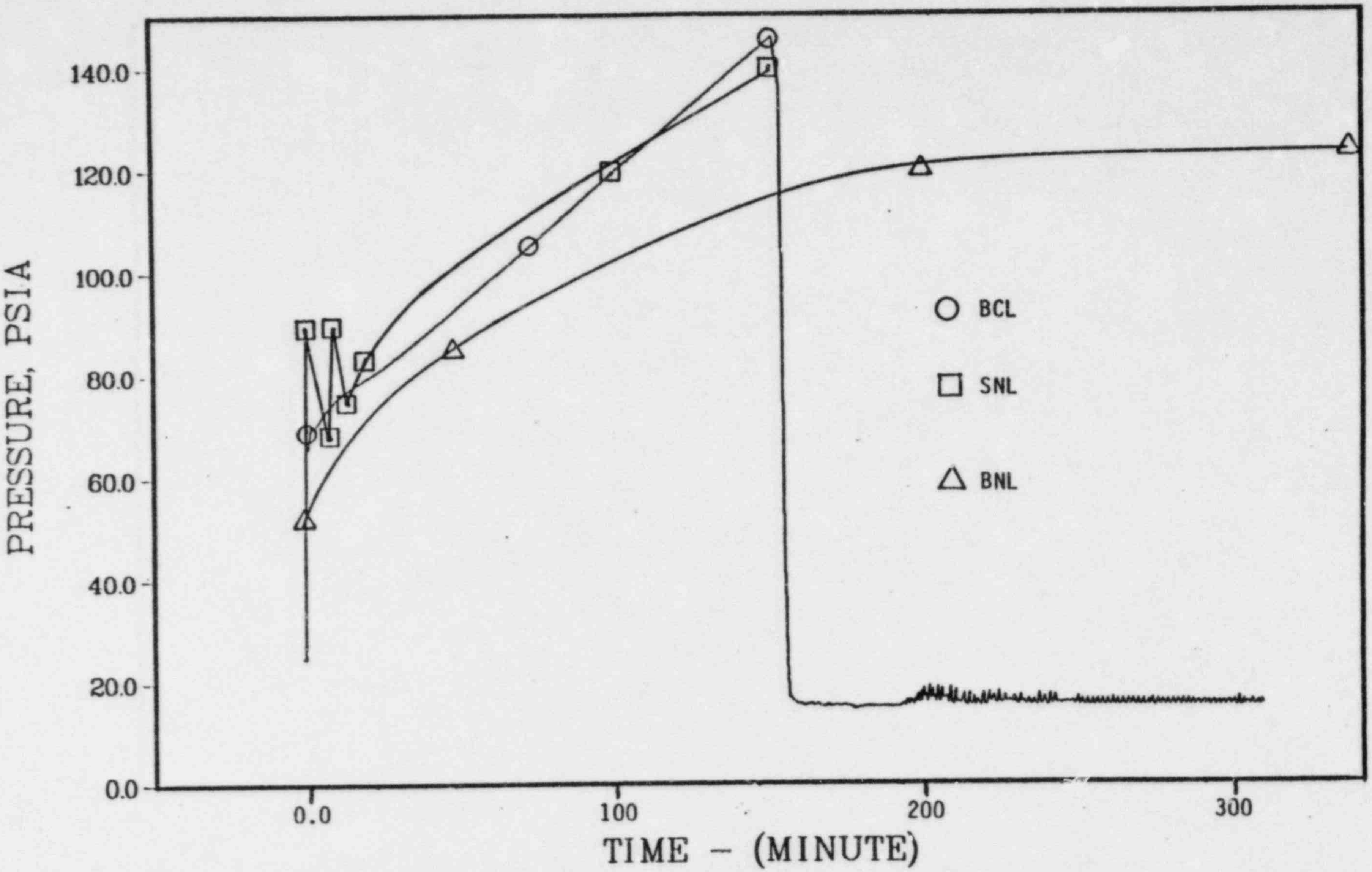

TQUV5
PRESSURE, PSIA
140.0
120.0
100.0
80.0
60.0
40.0
20.0
0.0
0.0
100
200
300
TIME - (MINUTE)
○ BCL
□ SNL
△ BNL

WTP:2.32

TQUV5



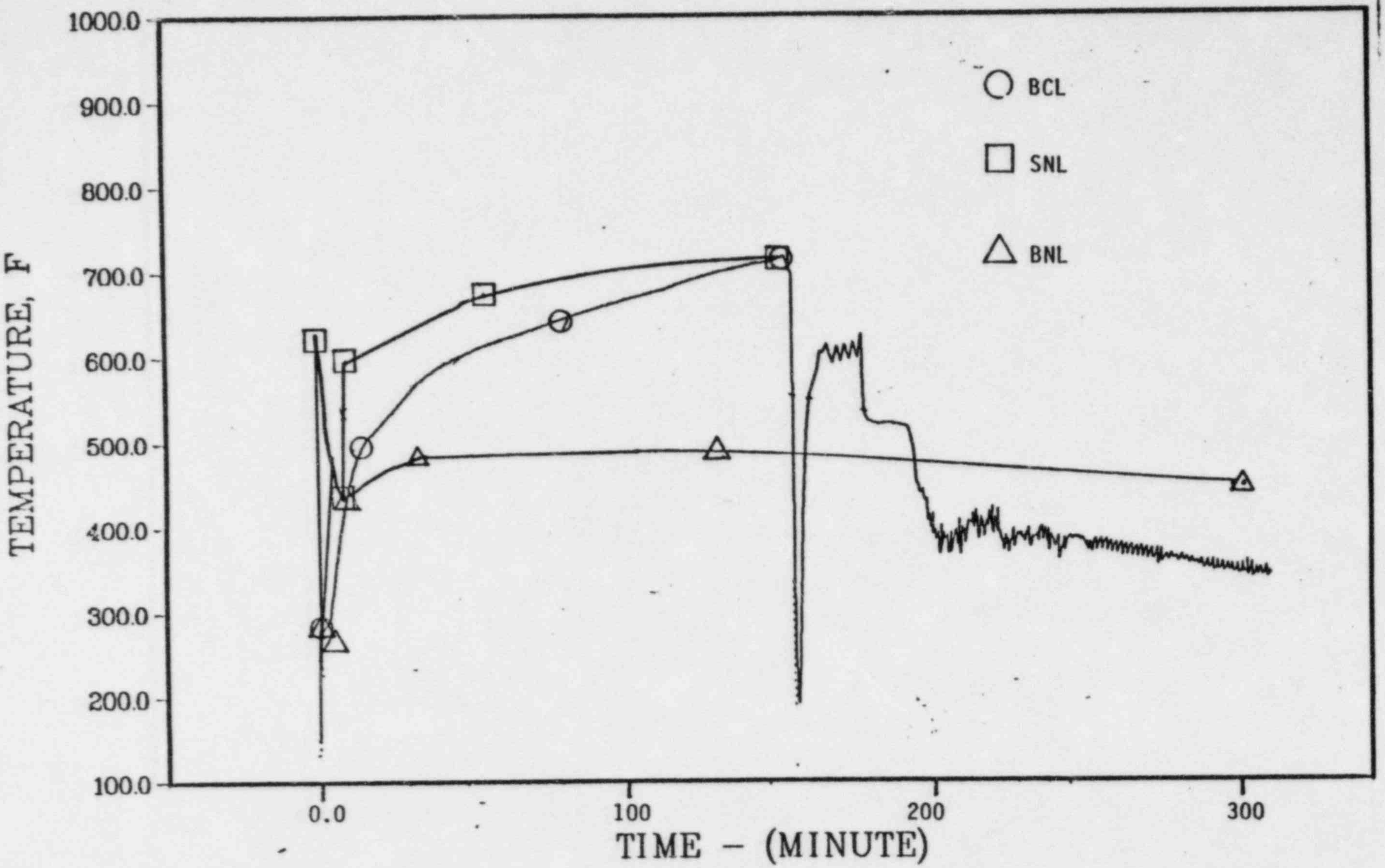
TQUV7



PRESSURE, PSIA
140.0
120.0
100.0
80.0
60.0
40.0
20.0
0.0
0.0
100
200
300
TIME – (MINUTE)
○ BCL
□ SNL
△ BNL

WTP:2.34

TQUV7



OBSERVATIONS REGARDING CLWG AND CPWG ACTIVITIES ON MARK II CONTAINMENTS

- GRADUAL PRESSURE BUILD-UP DURING CORE/CONCRETE INTERACTIONS WITH ELEVATED DRYWELL TEMPERATURES

- DRYWELL ENVIRONMENT COULD RESULT IN SEAL DEGRADATION AND LEAKAGE

- CPWG RESULTS INDICATE SEAL LEAKAGE COULD PREVENT OVERPRESSURIZATION FAILURE

- RESULTS TEND TO SUPPORT ASSUMPTIONS IN LGS-PRA

FAILURE MODES

- FIRST CATEGORY:
    - CONTAINMENT BUILDING IS INITIALLY EFFECTIVE AND THEN FAILS
- SECOND CATEGORY:
    - CONTAINMENT BUILDING FUNCTION IS EITHER BYPASSED OR COMPROMISED

FIRST CATEGORY

- STEAM EXPLOSIONS
- HYDROGEN BURN INDUCED FAILURES
- FAILURE BY OVERPRESSURIZATION
- BASEMAT PENETRATION

SECOND CATEGORY

- FAILURE TO ISOLATE CONTAINMENT BUILDING
- ACCIDENT SEQUENCE BYPASSES CONTAINMENT BUILDING

SUMMARY AND CONCLUSIONS

- SOURCE-TERM CALCULATIONS PERFORMED FOR THE LGS-FES ARE MUCH CLOSER TO UPPER BOUND RISK ESTIMATES THAN LOWER BOUND

- UNCERTAINTY LEADING TO HIGHER SOURCE TERMS RESULTS IN LESS THAN FACTOR OF 4 INCREASE IN TOTAL RISK

- NEW METHODS HAVE THE POTENTIAL FOR SIGNIFICANTLY REDUCING LONG-TERM DAMAGE INDICES (LATENT FATALITIES, PERSON-REM, ETC.)

- NEW METHODS HAVE LESS POTENTIAL FOR REDUCING SHORT-TERM DAMAGE INDICES (EARLY FATALITIES, ETC.)

IMPACT OF NEW SOURCE - TERMS

- FIRST CATEGORY - LOWER SOURCE TERMS
  - CONTAINMENT PERFORMANCE
  - AEROSOL AGGLOMERATION AND SETTLING
  - CONTAINMENT ESFs
- SECOND CATEGORY - UNCERTAIN BUT STILL POTENTIAL FOR SIGNIFICANT SOURCE TERMS
  - PRIMARY SYSTEM RETENTION UNCERTAIN

OUTLINE

I. PURPOSE AND OBJECTIVES

II. APPROACH

III. PHILOSOPHY AND ASSUMPTIONS

IV. PRINCIPAL FINDINGS

V. CONTAINMENT FAILURE MODES

VI. MITIGATION SYSTEMS

V. COST BENEFIT

VI. UNCERTAINTY

MITIGATION

THOSE ACTIONS, DEVICES, OR SYSTEMS INTENDED TO REDUCE, AMELIORATE, OR REMOVE THE CONSEQUENCES TO THE PUBLIC OF A SEVERE ACCIDENT WHEREIN THE CORE OF A REACTOR IS DEGRADED OR MELTED. IN PRACTICE, THIS MEANS KEEPING THE CONTAINMENT FROM FAILING.

OBJECTIVE

- IS MITIGATION TECHNICALLY FEASIBLE?
- WHAT WOULD IT COST?
- WHAT WOULD THE BENEFITS BE?

APPROACH

- SURVEY CONTAINMENTS AND HOW THEY FAIL.
- SURVEY MITIGATION TECHNOLOGY.
- DESIGN SPECIFIC SYSTEMS FOR THREE DIFFERENT PLANTS.
- DEVELOP COST/BENEFIT ASSESSMENT PROCEDURE.
- EXPLORE OTHER TYPES OF BENEFITS AND DISBENEFITS, AND OUTLINE POSSIBLE IMPLEMENTATION STRATEGIES.

PHILOSOPHY AND ASSUMPTIONS

- MITIGATION MUST BE COMPLETE.
- ACCIDENT PHENOMENA MUST REACH A DETERMINATE END STATE.
- OPERATOR ACTION NOT AVAILABLE.
- ELECTRIC POWER NOT AVAILABLE.

$$R^{i} \equiv \sum_{j=1}^{J} \sum_{k=1}^{K} f_{j} P_{jk} C_{ki}$$

f_j = FREQUENCY OF j^{th} CONTAINMENT CLASS FREQUENCY (SEQUENCES).

P_{jk} = CONDITIONAL PROBABILITY OF THE k^{th} CONTAINMENT FAILURE MODE.

C_{ki} = THE i^{th} CONSEQUENCE OF INTEREST FOR THE k^{th} CONTAINMENT FAILURE MODE.

R_i = RISK (i^{th} CONSEQUENCE).

$$R^i = \sum_{j=1}^{J} \sum_{k=1}^{K} f_j P_{jk} C_{ki}$$

- $\sum_{k=1}^{K} P_{jk} \equiv 1 \qquad j=1,2,\ldots J$

- If $P^*_{jk} \equiv 0$ due to mitigation,

Some P_{jk} increase for $k \neq k^* (P'_{jk})$

$$\Delta R^{i}_{k*} = \sum_{j=1}^{J} \sum_{k=1}^{K} f_j (P_{jk} - P'_{jk}) \, C_{ki}$$

$$= \sum_{j=1}^{J} \sum_{k \neq k*}^{K} f_j (P_{jk} - P'_{jk}) \, C_{ki}$$

$$+ \sum_{j=1}^{J} f_j P_{jk*} C_{k*i}$$

$$\text{Benefit} \cong \sum_{k*} R^{i}_{k*} \text{ for complete mitigation}$$

TABLE 3-7. CONDITIONAL PROBABILITY OF CONTAINMENT FAILURE, RELEASE CATEGORY AND CLASS FREQUENCY

Mode of Containment Failure	Class I (9.5×10^{-5} yr^{-1})	Class II (4.1×10^{-6} yr^{-1})	Class III (3.4×10^{-6} yr^{-1})	Class IV (3.0×10^{-7} yr^{-1})
α	0.001 (OXRE)	0.005 (OXRE)	0.001 (OXRE)	0.01 (OXRE)
β,μ'	0.002 (OXRE)	0.05 (OXRE)	0.002 (OXRE)	0.09898 (OXRE)
γ	0.247 (OPREL)	0.2245 (OPREL)	0.247 (OPREL)	0.445 ($C4\gamma$)
γ'	0.1235 (OPREL)	0.1105 (OPREL)	0.1235 (OPREL)	0.2226 ($C4\gamma'$)
γ''	0.1235 (OPREL)	0.1103 (OPREL)	0.1235 (OPREL)	0.2226 ($C4\gamma''$)
δ	0.2223 (none)	0.500 (none)	0.2223 (none)	-- --
δ_+	0.0247 (OPREL)	-- --	0.0247 (OPREL)	-- --
δ_ξ	0.247 (OPREL)	-- --	0.247 (OPREL)	-- --
μ	0.009 (OPREL)	-- --	0.009 (OPREL)	-- --
Total	1.000	1.000	1.000	1.000

Definitions:

OXRE is the oxidation release.

OPREL is the overpressurization release.

$C4\gamma$ is failure of the drywell release for ATWS.

$C4\gamma'$ is failure of the wetwell above the suppression pool release for ATWS.

$C4\gamma''$ is failure of the wetwell below the suppression pool release for ATWS.

TABLE 3-8. CONSEQUENCES FOR EACH RELEASE CATEGORY (BNL REVIEW)

Release Category	Acute* Fatalities	Latent* Fatalities	Man-Rem* (500 miles)	Man-Rem* (50 miles)
OPREL	0	2.2×10^3	1.42×10^7	0.78×10^7
OXRE	97	1.9×10^4	4.90×10^7	2.5×10^7
C4γ	75	1.4×10^4	7.88×10^7	4.7×10^7
C4γ'	69	1.4×10^4	7.86×10^7	5.3×10^7
C4γ"	138	1.3×10^4	7.36×10^7	3.6×10^7

*Based on WASH-1400 source-terms and methodology.

TABLE 3-11. MAN-REM/YEAR (OUT TO 50 MILES) FOR EACH CONTAINMENT FAILURE MODE - INTERNAL INITIATORS (WITH ATWS-3A-FIX)*

Failure Mode	Class I	Class II	Class III	Class IV
α	2.6	0.56	0.09	0.08
β,μ'	5.2	5.6	0.18	0.08
γ	182.5	7.3	6.5	6.3
γ'	93.0	3.6	3.3	3.5
γ''	93.0	3.6	3.3	2.4
δ	--	--	--	--
δ_t	18.5	--	0.67	--
δ_ξ	186.5	--	6.7	--
μ	10.7	--	0.3	--
Total	592.0	20.7	21.1	12.4
Total risk = 646 man-rem/year (50 miles)				

---

*Based on WASH-1400 source terms and methodology.

ACCIDENT END STATES

1. ATWS STEAM GENERATION.
2. IN-VESSEL HYDROGEN GENERATION.
3. CONTAINMENT CONCRETE DECOMPOSITION.
4. EX-VESSEL STEAM PRESSURE RISE WHEN THE HOT CORE DEBRIS ENCOUNTERS WATER.
5. EX-VESSEL STEAM EXPLOSIONS.
6. EX-VESSEL HYDROGEN GENERATION.
7. RESIDUAL HEAT LOAD.

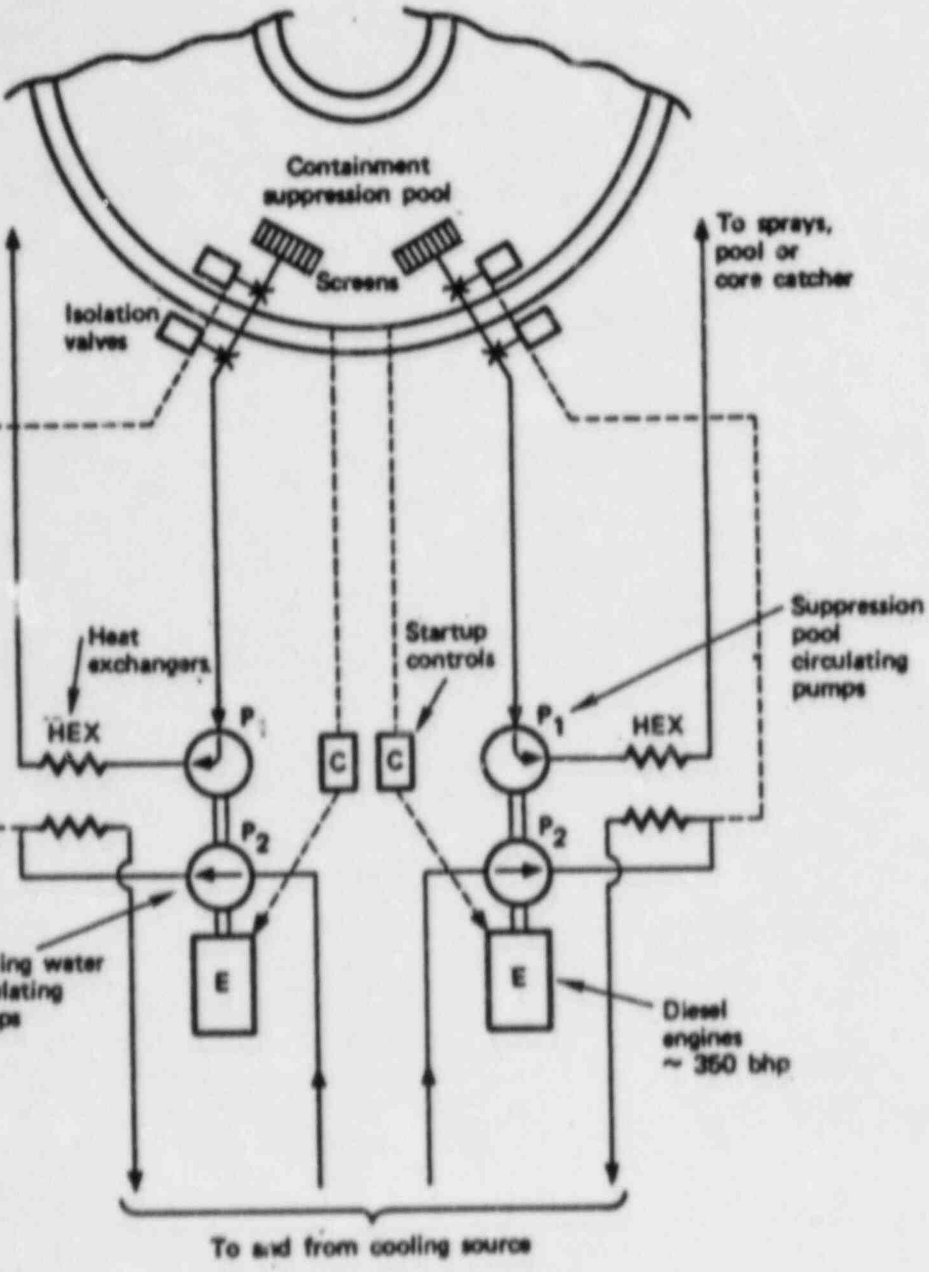


Figure 3.5. Schematic dual heat removal system.

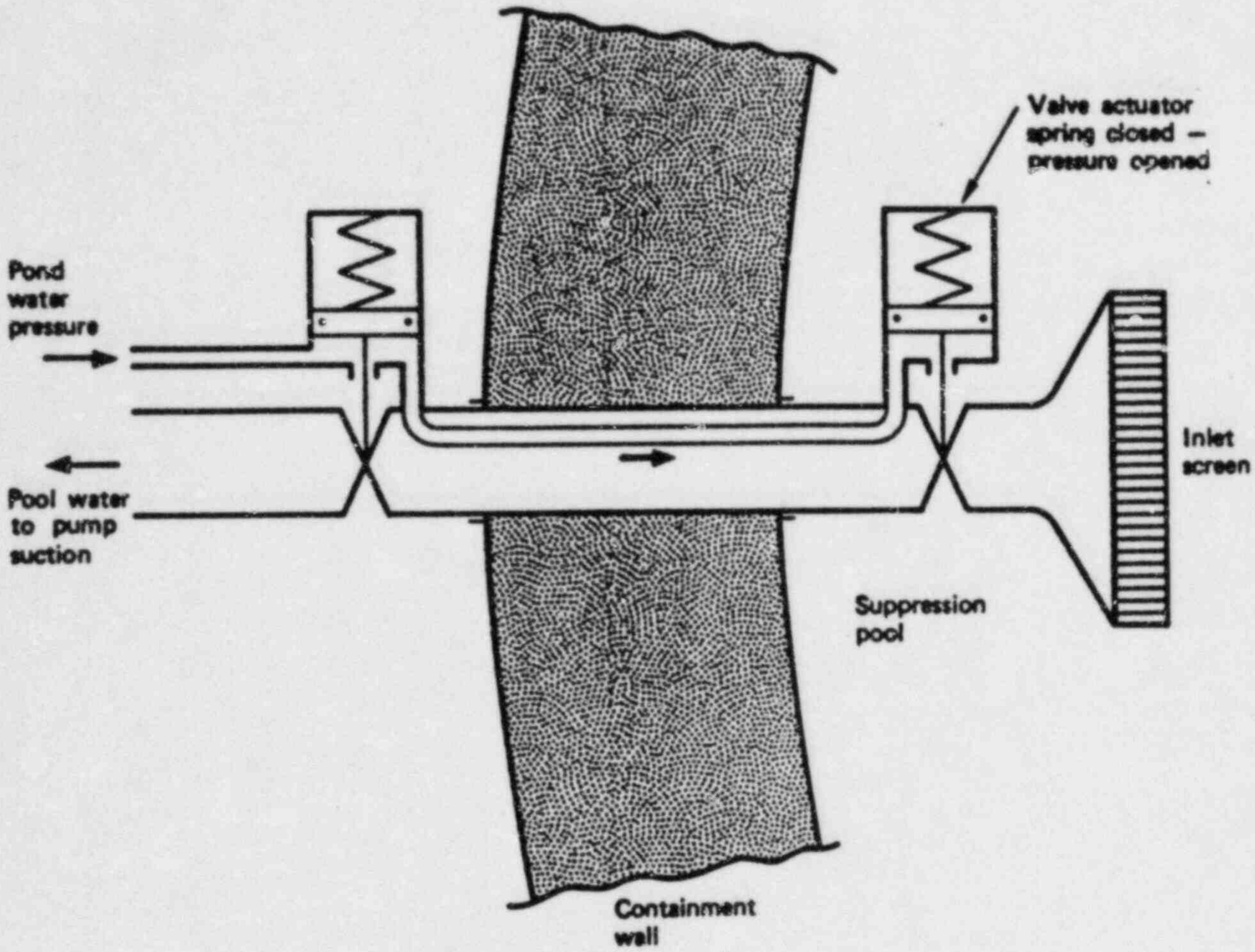


Figure 3.7. Schematic – double isolation valves for intake.

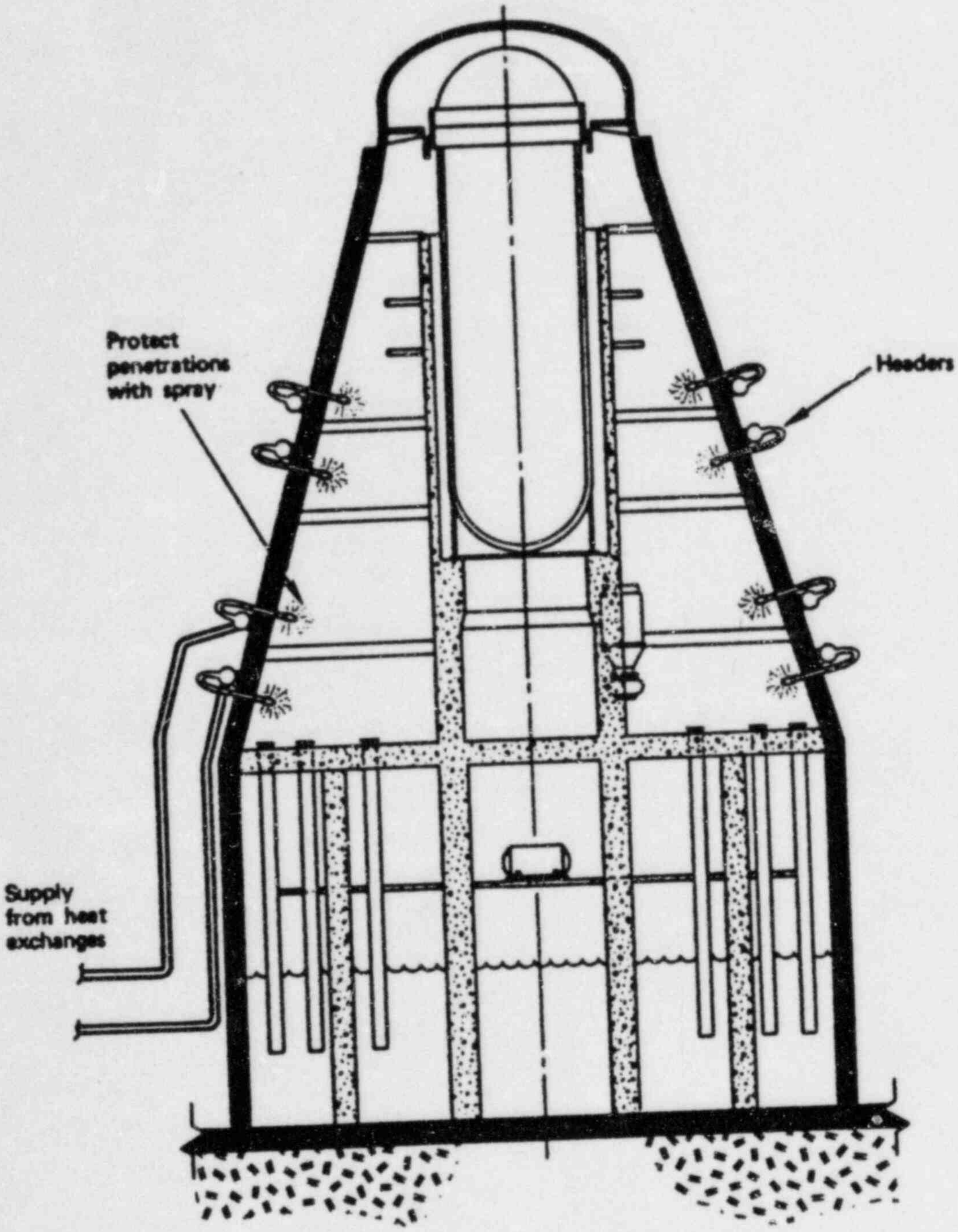
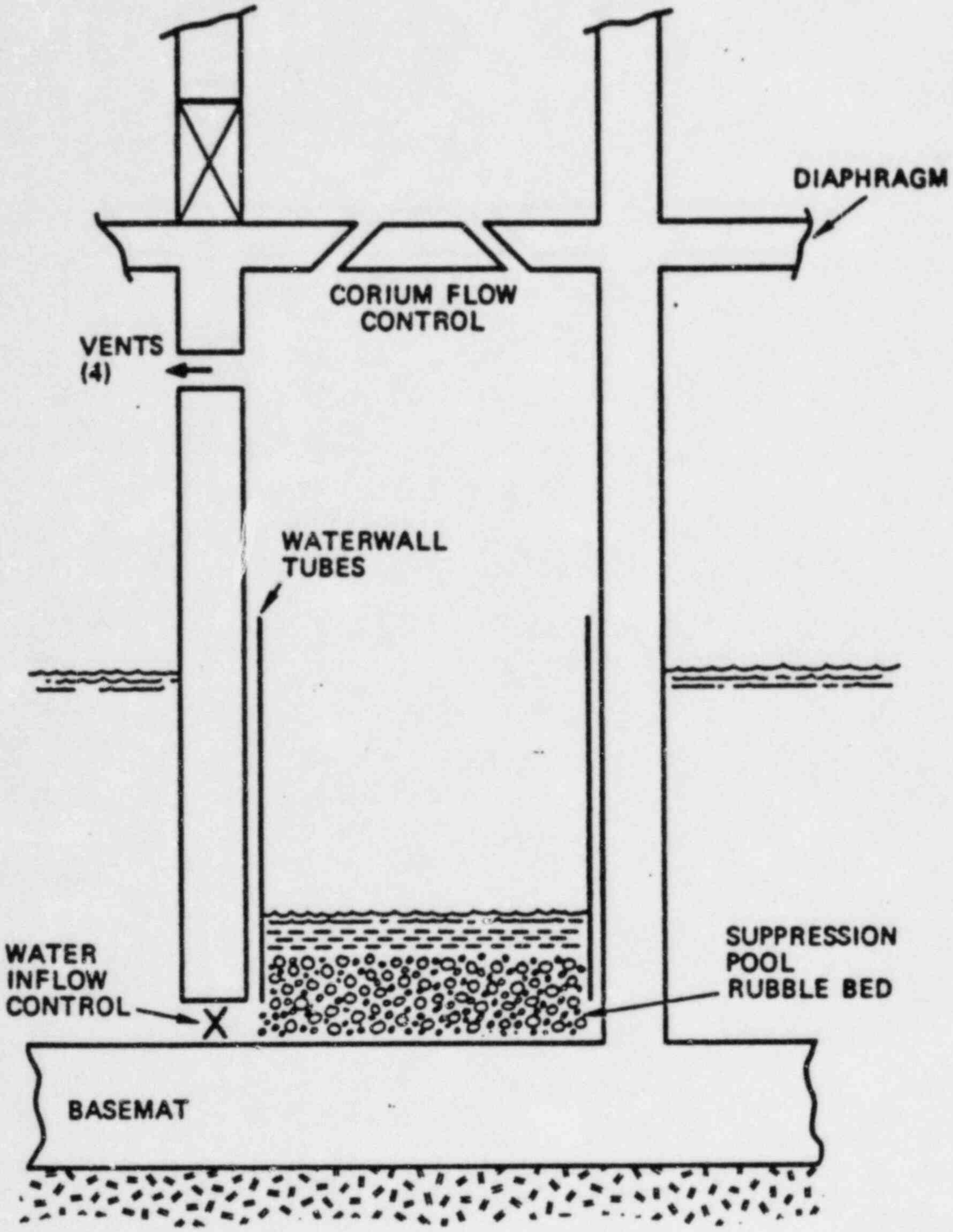
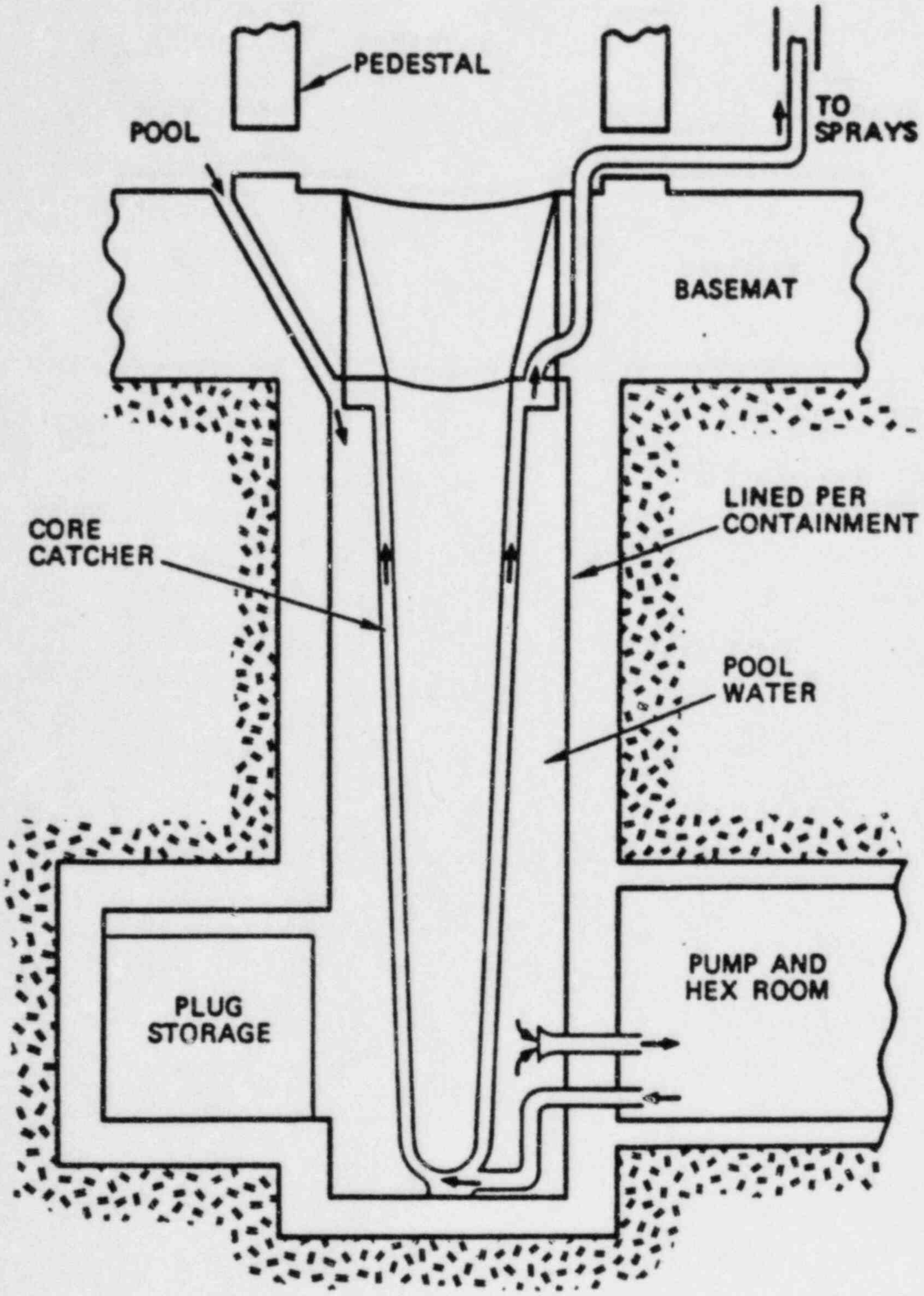


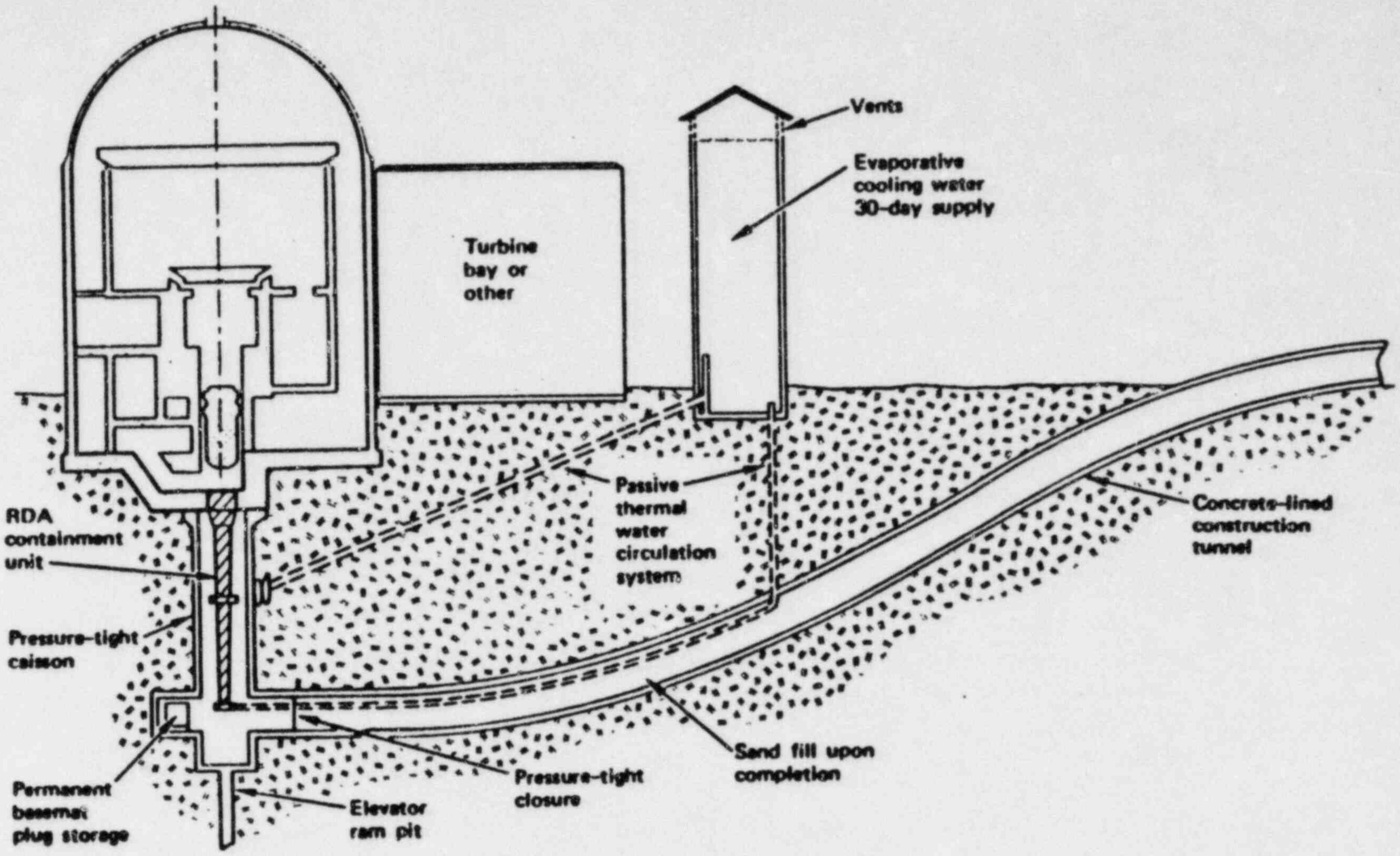
Figure 3-8. Sprays into upper drywell.

SCHEMATIC – RETENTION IN CENTER PEDESTAL



SCHEMATIC – DRY CRUCIBLE





Frontispiece. Overall view of RDA core retention unit at existing reactor.

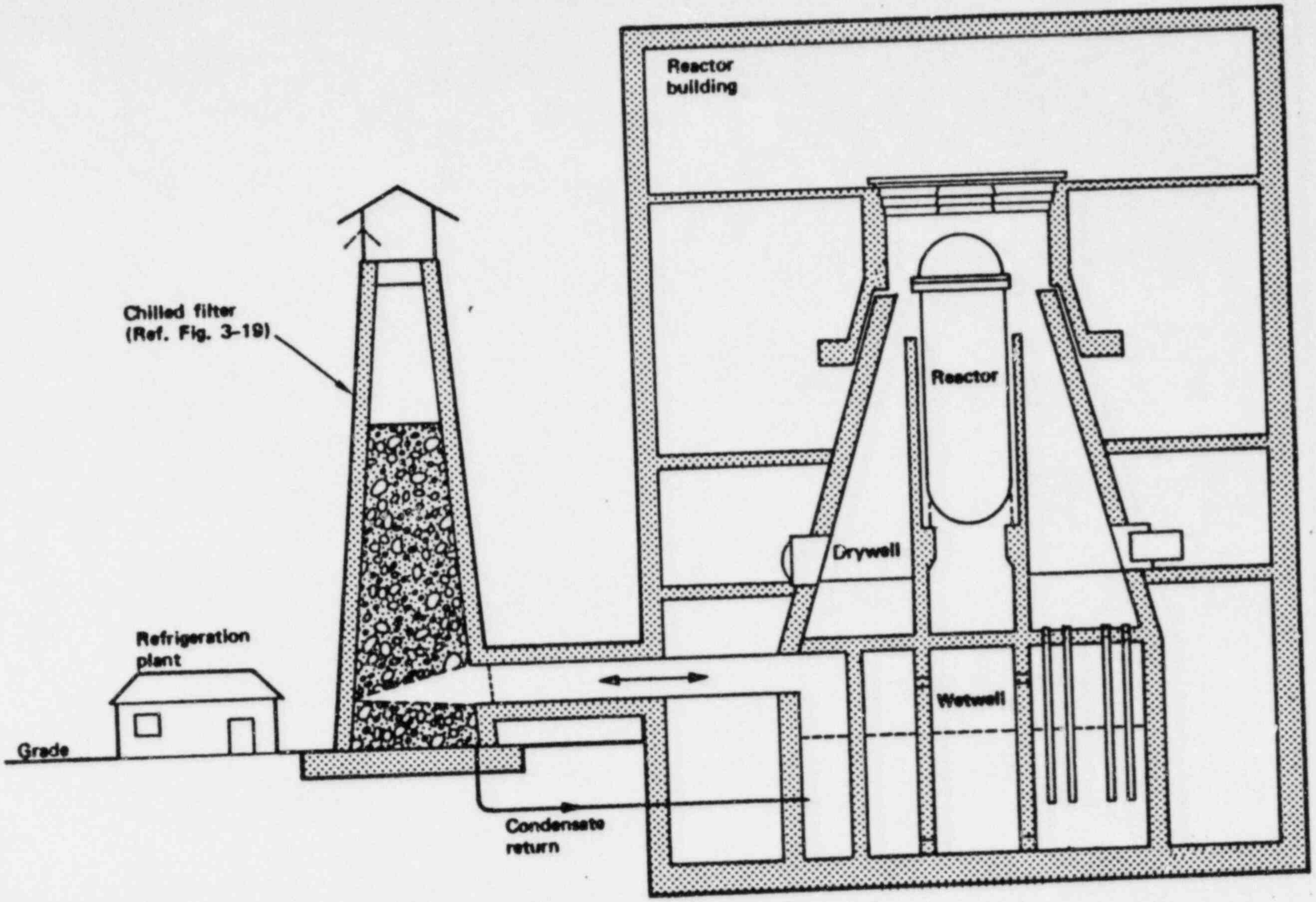


Figure 3-19. Schematic - open Mark II containment with chilled filter.

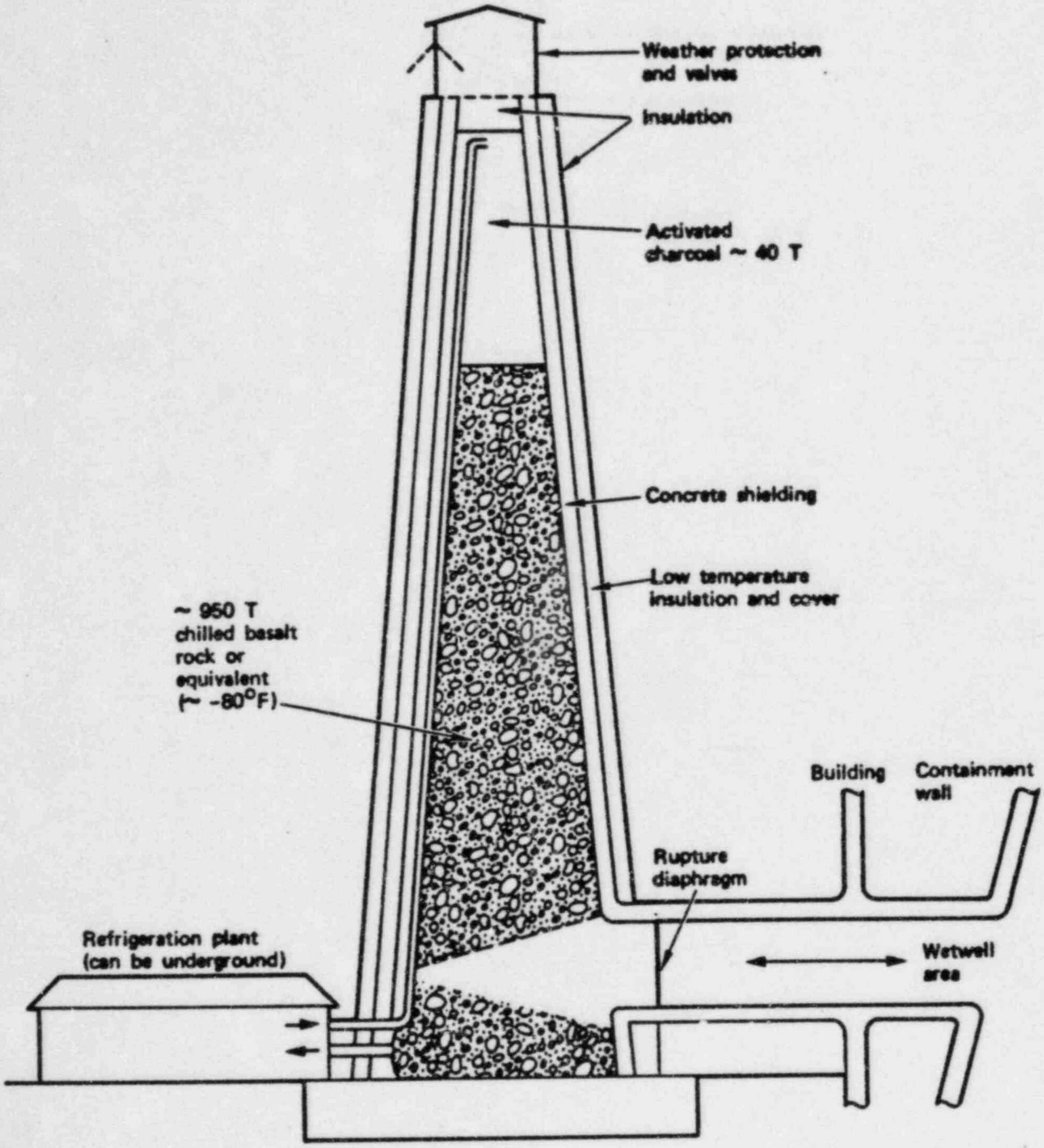


Figure 3-20. Schematic of chilled filter system.

Function		Equipment	A	B	C	D	E	F	G	H	I	J	K
		Options (in $/1000)	Overpressure control ATWS protection	ATWS protection without "3Afix"	Overpressure plus hydrogen control (FVCS)*	Total mitigation with FVCS	Overpressure control only	Overpressure and hydrogen control	Overpressure control + core control	Overpressure, hydrogen and core control	Total mitigation with core control	Total mitigation with core control	Overpressure and core control (no hydrogen)
Heat removal	Pool	Dedicated cooling + Separated + Underground	2770	2770	2770	2770	2770	2770	2770	2770	2770	2530	2530
	Spray	Drywell sprays + External feed + Internal feed					860	860	860	860	860	514	514
Core control		+ Basemat rubble bed + Dry crucible							3445	3445	3445	20,670	20,670
Pressure protection	Over	ATWS "3Afix"	1728	No 1728		1728					1728	1728	
		ATWS clean vent			2785	2785							
		Filtered vent	3573	3573				3573		3573	3573	3573	
		Large H_2 combiner									1336	1336	
	Under	Large breaker	1336	1336	1336	1336							
**Impact – cost in $/1000			9407	9407	6891	8619	3630	7203	7075	10,648	13,712	30,351	23,714
Value or benefit – man-REM averted		50 miles	1068	1893	21,365	22,433	23,739	24,179	23,739	24,179	26,247	26,247	23,739
		500 miles	1856	3232	38,178	40,034	42,420	43,208	42,420	43,208	45,064	45,064	42,420
Impact/value ratio		$/man-REM (50 mi)	8808	4970	322	384	153	298	298	440	543	1202	999
		$/man-REM (500 mi)	5068	2910	180	215	85.6	167	167	246	304	673	559

*Filtered vent containm[illegible] system will provide risk reduction factor of 10

**Based on RSS source terms and methodology

TABLE 3-17. CONTAINMENT MITIGATION - HIGH PRESSURE vs LOW PRESSURE

Function		Equipment	Options in $/1000	
			Low pressure open containment with chilled filter	High pressure containment per Option I
Heat removal	Pool	Dedicated cooling + Seperated + Underground	2770	2770
	Spray	Drywell sprays + External feed + Internal feed	860	860
Core control		+ Basemat rubble bed + Dry crucible	3445	3445
*Pressure protection	Over	ATWS "3Afix"	Yes	Yes
		ATWS clean vent	–	1728
		Filtered vent	–	
		Large H_2 combiner	–	3573
	Under	Large breaker	–	1336
	Both	Chilled filter	2938	–
		Open containment	300	–
Impact – cost in $/1000			10,613	13,712
Value or benefit – man-REM averted		50 miles	25,247	25,247
		500 miles	45,064	45,064
Impact/value ratio		$/man-REM (50 mi)	404	543
		$/man-REM (500 mi)	230	304

*Based on Figure 3-16 conditions

PHILADELPHIA ELECTRIC COMPANY
LIMERICK GENERATING STATION, UNITS 1 AND 2

ACRS SUBCOMMITTEE MEETING

OCTOBER 20, 1984

LOS ANGELES, CALIFORNIA

MEETING AGENDA
AND
APPLICANT'S SLIDE PRESENTATION
FOR OCTOBER 20, 1984

AGENDA FOR THE ACRS COMBINED SUBCOMMITTEE'S
LIMERICK UNITS 1 AND 2
RELIABILITY AND PROBABILISTIC ASSESSMENT

OCTOBER 20, 1984 - LOS ANGELES, CA

1.	Executive Session	D. Okrent	15 Min.	8:30 - 8:45 am
2.	Discussion of In-Containment Analysis (Accident Progression through Containment Failure)			
	A) PECo Presentation	E. A. Hughes R. E. Henry	90 Min.	8:45 - 10:15 am
xxxxxx BREAK xxxxxx			15 Min.	10:15 - 10:30 am
	B) NRC Presentation	NRC Staff	3 Hrs.	10:30 - 1:30 pm
xxxxxx LUNCH xxxxxx			60 Min.	1:30 - 2:30 pm
3.	NRC Assessment of PRA/SARA Uncertainties and Limitations to include a discussion as to how this is used in the decision process	NRC Staff	30 Min.	2:30 - 3:00 pm
4.	PECo Discussion of PRA/SARA Insights into Plant Design and Operations	G. F. Daebeler	30 Min.	3:00 - 3:30 pm
5.	NRC Evaluation of PRA/SARA Insights - To include a critique of the three best prevention/mitigation options which have not already been implemented	NRC Staff	60 Min.	3:30 - 4:30 pm
xxxxxx BREAK xxxxxx			15 Min.	4:30 - 4:45 pm
6.	PECo Discussion of future use of PRA/SARA	A. R. Diederich	15 Min.	4:45 - 5:00 pm
7.	NRC Discussion of Open Seismic Risk Questions - To include a discussion of the NRC's best estimate of the seismic contribution to risk	NRC Staff	30 Min.	5:00 - 5:30 pm

IN-CONTAINMENT ANALYSIS
(ACCIDENT PROGRESSION)

E. A. HUGHES
R. E. HENRY

IN PLANT PHYSICS PRESENTATION

- DESCRIBE LGS PRA METHODOLOGY
    - GENERAL APPROACH
    - METHODS
    - CLASS I CASE
    - CLASS IV CASE
    - FISSION PRODUCT SOURCE TERM
- CONCLUSIONS
- CURRENT METHODOLOGY COMPARISON

IN-CONTAINMENT RADIONUCLIDE TRANSPORT AND RELEASE

ACCIDENT SEQUENCE CLASSES (BINNING)

SEQUENCES MODELED — PHYSICAL PROCESSES

CONTAINMENT STRUCTURAL EVALUATION

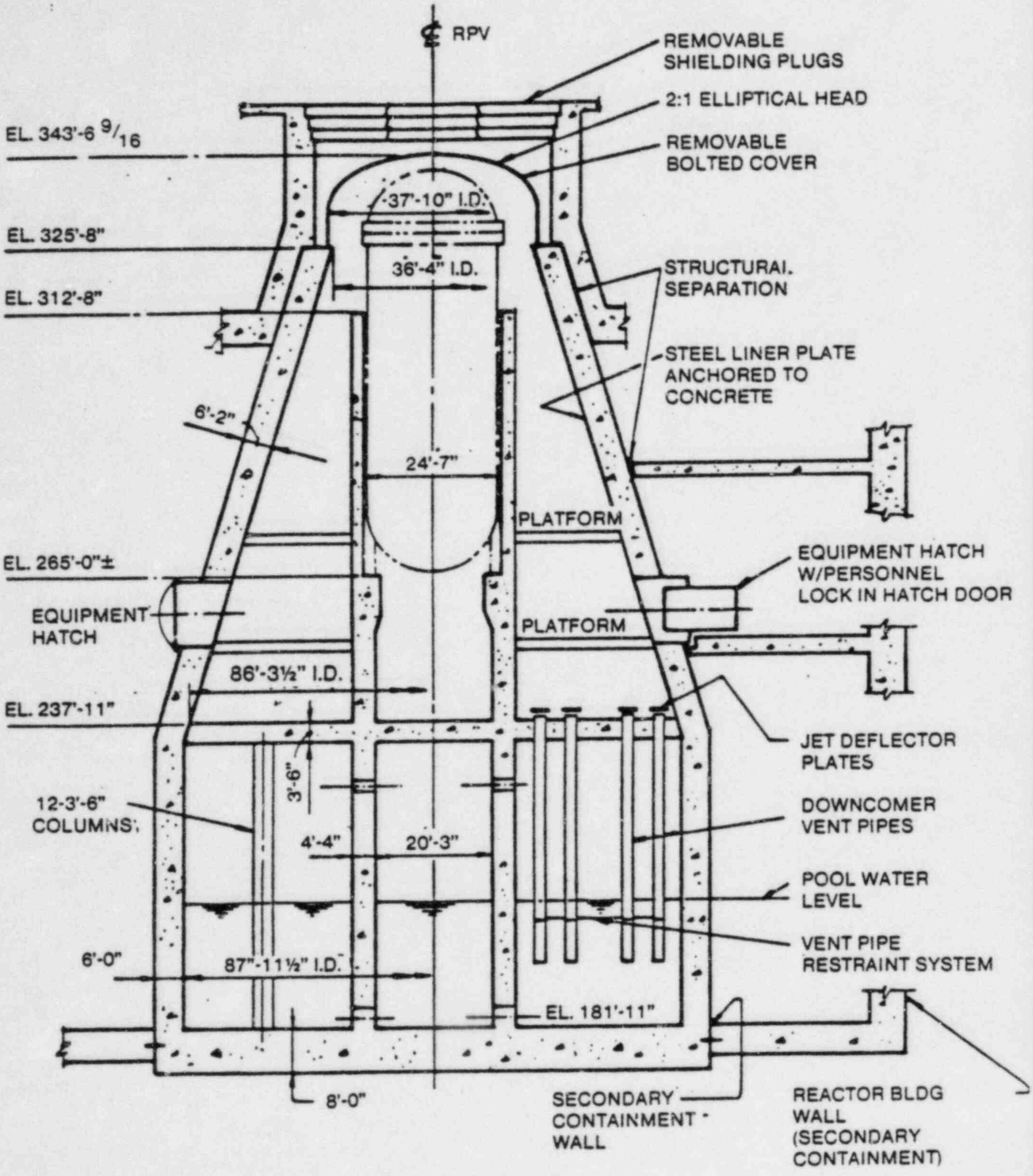
CONTAINMENT EVENT TREE

FISSION PRODUCT TRANSPORT

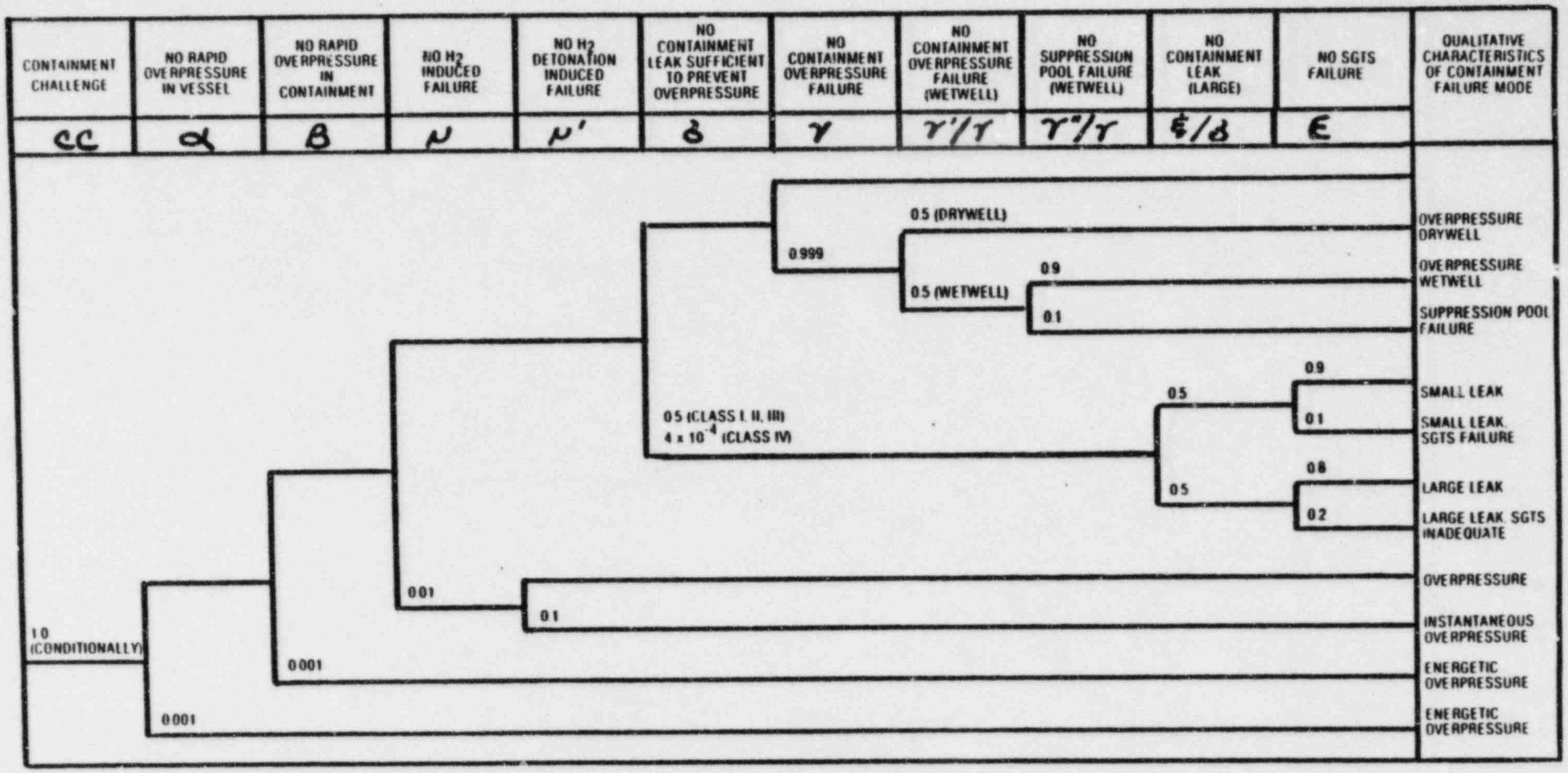
SOURCE TERMS

ACCIDENT SEQUENCE BINS

ACCIDENT CLASS (BIN)	CORE CONDITION	CONTAINMENT CONDITION AT CORE DAMAGE	EXAMPLE
I	• CONTROL RODS INSERTED • DECAY HEAT	INTACT AT LOW PRESSURE	TQUV
II	• CONTROL RODS INSERTED • LONG TERM DECAY HEAT	FAILED	TW
III	• ATWS; LOCA • 30% POWER	INTACT AT HIGH PRESSURE	T_FC_MU
IV	• ATWS • 30% POWER	FAILED	$T_FC_MC_2$
S	• IMMEDIATE CORE UNCOVERY • DECAY HEAT	FAILED	RPV RUPTURE; SEISMIC AND RANDOM
IS	• CONTROL RODS INSERTED • DECAY HEAT	FAILED	SEISMIC REACTOR BUILDING FAILURE

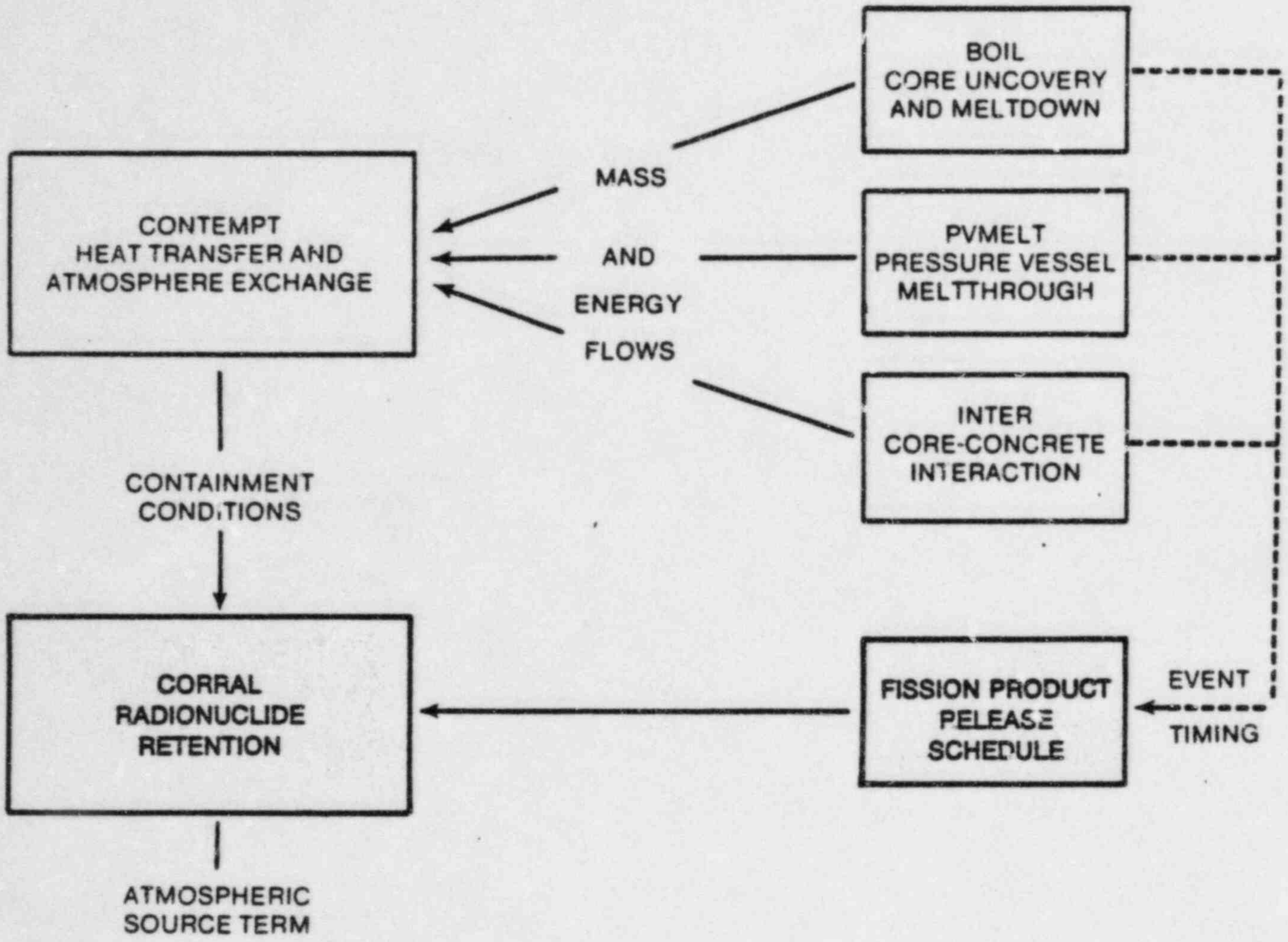


VERTICAL SECTION
CONTAINMENT GENERAL ARRANGEMENT



CONTAINMENT EVENT TREE FOR THE MARK II CONTAINMENT

ACCIDENT PROGRESSION ANALYSIS



BOIL
CORE UNCOVERY
AND MELTDOWN

PVMELT
PRESSURE VESSEL
MELTTHROUGH

INTER
CORE-CONCRETE
INTERACTION

MASS
AND
ENERGY
FLOWS

CONTEMPT
HEAT TRANSFER AND
ATMOSPHERE EXCHANGE

CONTAINMENT
CONDITIONS

EVENT
TIMING

FISSION PRODUCT
RELEASE
SCHEDULE

CORRAL
RADIONUCLIDE
RETENTION

ATMOSPHERIC
SOURCE TERM

PLANT RESPONSE TO PHENOMENA

	RSS METHODOLOGY	LGS METHODOLOGY
Core Meltdown	Boil Code	Boil Code
RPV Melt Through	Previous Analysis Hand Calculations	PV Melt
Steam Explosion	Parametric Analysis Likelihood Estimates	Sandia/ANL New Estimates
Concrete Melt Through	Hand Calculations Scoping Studies	Sandia Inter
Containment Conditions	Hand Calculations Generic Rate Curves	Contempt Coupled with Boil, PV Melt and Inter

RADIOACTIVITY BEHAVIOR AND ESCAPE

- RELEASE MECHANISMS
    - SAME AS WASH 1400
        - GAP
        - GAP
        - MELT
        - OXIDATION
        - VAPORIZATION

        RECIPES FOR EACH BASED ON EXPERIMENTAL DATA
    - RELEASE FROM PRIMARY SYSTEM—ORIGINALLY HAND CALCULATIONS TO GUIDE SIMPLE DF ASSUMPTIONS, NOW INCOR/CORRAL
    - RELEASE FROM CONTAINMENT—ORIGINALLY CORRAL, NOW INCOR/CORRAL

GAP RELEASE COMPONENT

- TIMING - EARLY IN CORE HEATUP PROCESS
- DURATION - SHORT (SECONDS TO MINUTES)
- DRIVING FORCE - FUEL ROD DEPRESSURIZATION AND PRIMARY SYSTEM STEAM FLOW
- PRIMARY SYSTEM RETENTION—NONE
- SPECIES - VOLATILE FISSION PRODUCTS
- CORRAL FORMAT - PUFF RELEASE AT ONE MINUTE AFTER START OF BOILOFF AND CORE UNCOVERY

MELT RELEASE COMPONENT

- TIMING - BEGINS WITH ONSET OF CORE MELTDOWN
- DURATION - ONE TO TWO HOURS LONG
- DRIVING FORCE - THERMALLY ACTIVATED MIGRATION AND VAPORIZATION FROM FUEL COMBINED WITH PRIMARY SYSTEM STEAM FLOW
- PRIMARY SYSTEM RETENTION—NONE
- SPECIES - ALL FISSION PRODUCTS AS EITHER GASEOUS OR PARTICULATE FORMS
- CORRAL FORMAT - TEN EQUALLY SPACED AND SIZED RELEASES COVERING THE CORE MELTDOWN PERIOD

OXIDATION RELEASE COMPONENT

- TIMING - COINCIDENT WITH POSTULATED STEAM EXPLOSION EVENT

- DURATION - VERY SHORT

- DRIVING FORCE - DISPERSED SMALL PARTICLES OF HOT FUEL RELEASE CERTAIN FISSION PRODUCTS AS THE FUEL IS OXIDIZED

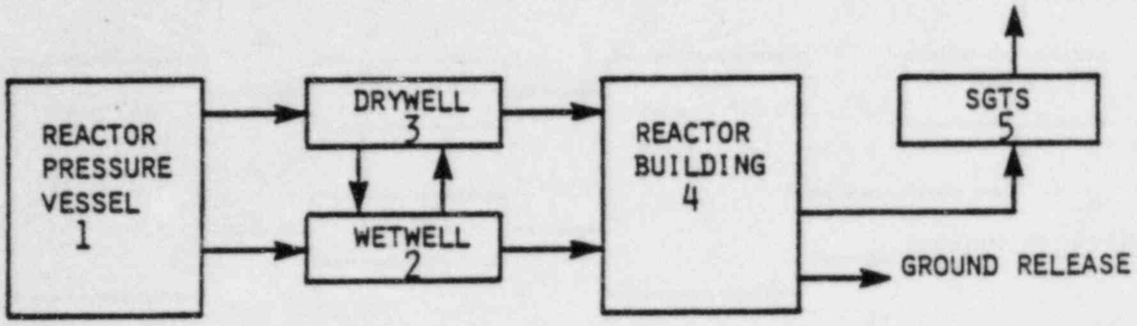
- SPECIES - FISSION PRODUCTS THAT ARE VOLATILE OR THAT FORM VOLATILE OXIDES

- CORRAL FORMAT - PUFF RELEASE AT THE END OF CORE MELTDOWN OR VESSEL FAILURE IF THE STEAM EXPLOSION OCCURS

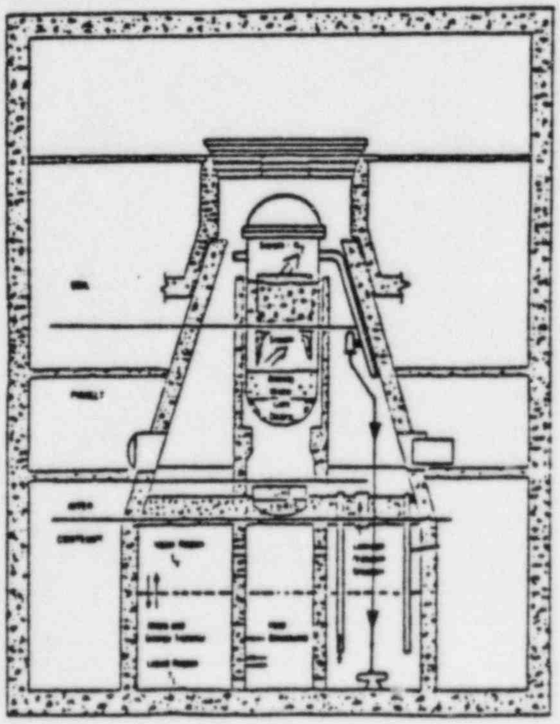
VAPORIZATION RELEASE COMPONENT

- TIMING - LATE; I.E., STARTS WHEN MOLTEN CORE DEBRIS CONTACTS CONCRETE DIAPHRAGM FLOOR UNDERNEATH REACTOR VESSEL
- DURATION - SEVERAL HOURS
- DRIVING FORCE - HIGH DEBRIS TEMPERATURES WITH SPARGING BY CONCRETE DECOMPOSITION GASES
- PRIMARY SYSTEM RETENTION—DOES NOT APPLY
- SPECIES - ALL FISSION PRODUCTS AS EITHER GASEOUS OR PARTICULATE FORMS
- CORRAL FORMAT - TWENTY IMPULSE RELEASES WITH EACH SUCCESSIVE RELEASE AT AN EXPONENTIALLY LOWER VALUE THAN THE FIRST

SCHEMATIC OF CORRAL LGS MODEL



COMPARTMENT	REMOVAL PROCESSES	FISSION PRODUCT SOURCES
1	NONE	GAP AND MELT RELEASES
2	NATURAL DEPOSITION	
	POOL SCRUBBING	STEAM EXPLOSION RELEASES
3	NATURAL DEPOSITION	VAPORIZATION RELEASES
4	NATURAL DEPOSITION, LEAKAGE	
5	HEPA FILTERS	



FISSION PRODUCT RETENTION MECHANISMS CONSIDERED

- NATURAL DEPOSITION PLATEOUT AND GRAVITATIONAL SETTLING
  - IN CONTAINMENT STRUCTURE SURFACES
    - DRYWELL
    - WETWELL CHAMBER
  - REACTOR BUILDING SURFACES
- SUPPRESSION POOL SCRUBBING
- SGTS FILTRATION
- MOLTEN FUEL (QUENCHED OR FROZEN ON DIAPHRAGM FLOOR)

SUPPRESSION POOL SCRUBBING

POOL CONDITION	DECONTAMINATION FACTORS IODINE & PARTICULATES	NOBLE GASES
SUBCOOLED	100.	1.0
SATURATED	10.	1.0
BYPASS	1.0	1.0

CLASS I ACCIDENT SEQUENCES

- TRANSIENT EVENT OCCURS
- REACTOR SCRAM SUCCESSFUL
- COOLANT MAKE-UP FAILS
- STEAM THROUGH RELIEF VALVES

CLASS I ACCIDENT SEQUENCES

- CORE MELT OCCURS
- GAP AND MELT RELEASE SCRUBBED IN SUPPRESSION POOL
- VESSELS FAILS WITH INTACT CONTAINMENT
- 10% OXIDATION RELEASE
- PART OF VAPORIZATION RELEASE SCRUBBED IN SUPPRESSION POOL

CLASS I ACCIDENT SEQUENCES

- CONTAINMENT FAILS SEVERAL HOURS AFTER VESSEL FAILURE
- THREE CONTAINMENT FAILURE LOCATIONS CONSIDERED:
  - DRYWELL
    - NO ADDITIONAL SCRUBBING
  - WETWELL ABOVE POOL
    - ALL RELEASES SCRUBBED
  - WETWELL BELOW POOL
    - NO ADDITIONAL SCRUBBING
- SUPPRESSION POOL SUBCOOLED
- RESUSPENSION OF SCRUBBED FISSION PRODUCTS AT CONTAINMENT FAILURE

ACCIDENT SEQUENCE TIME LINE

LIMERICK PRA

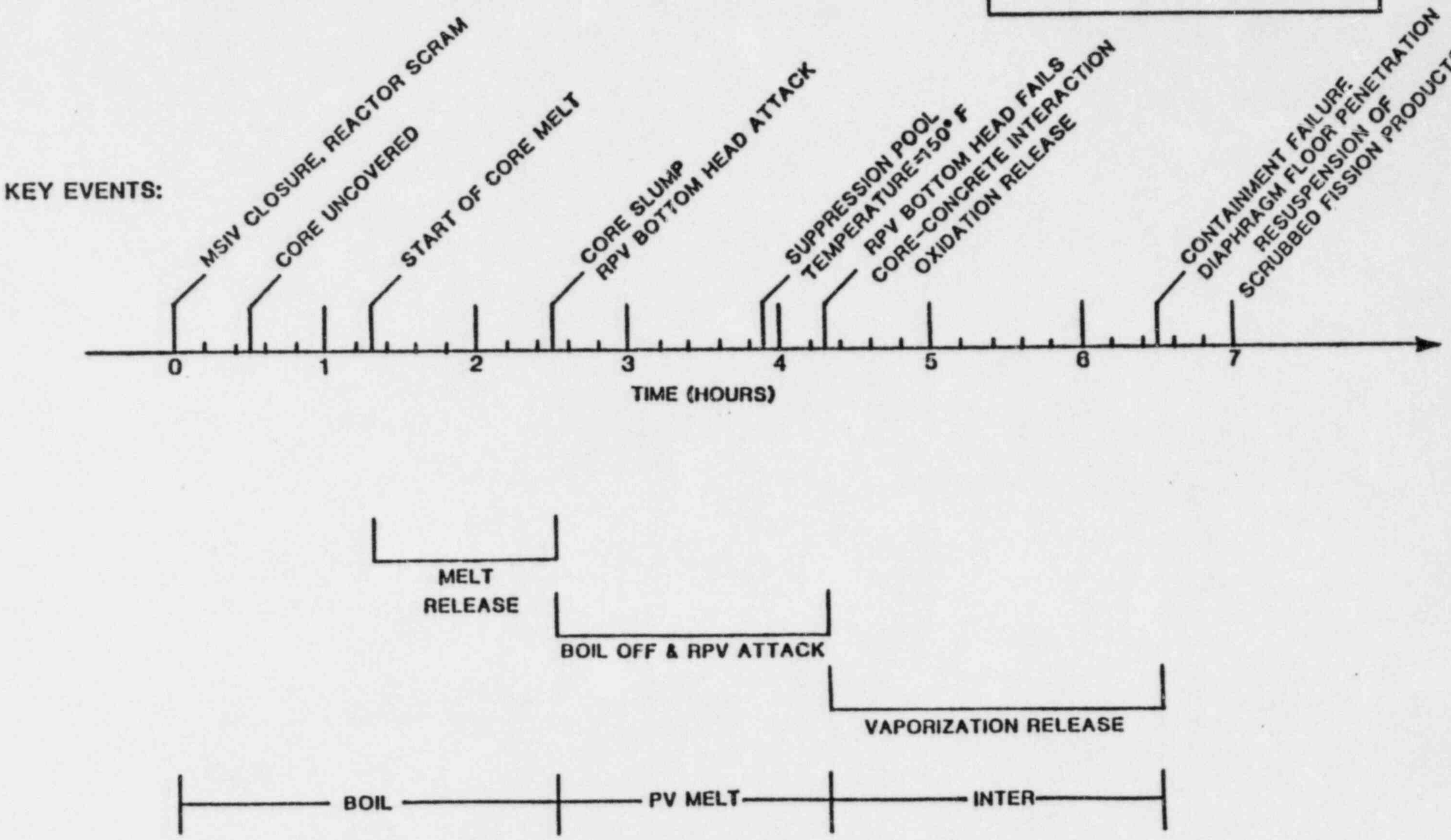
CLASS I

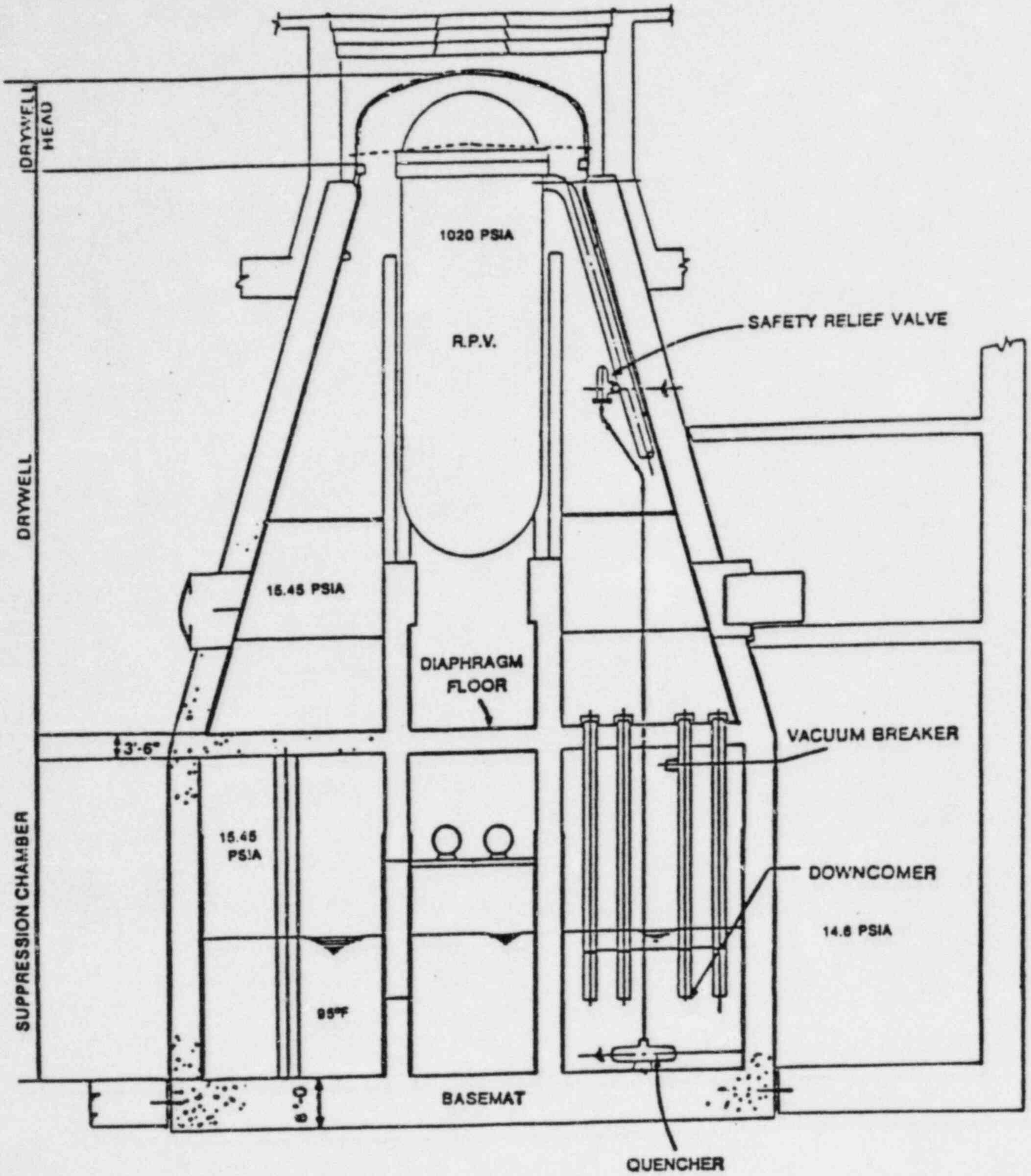
TYPICAL SEQUENCE:

TQUV, TQUX

CONTAINMENT FAILURE MODES:

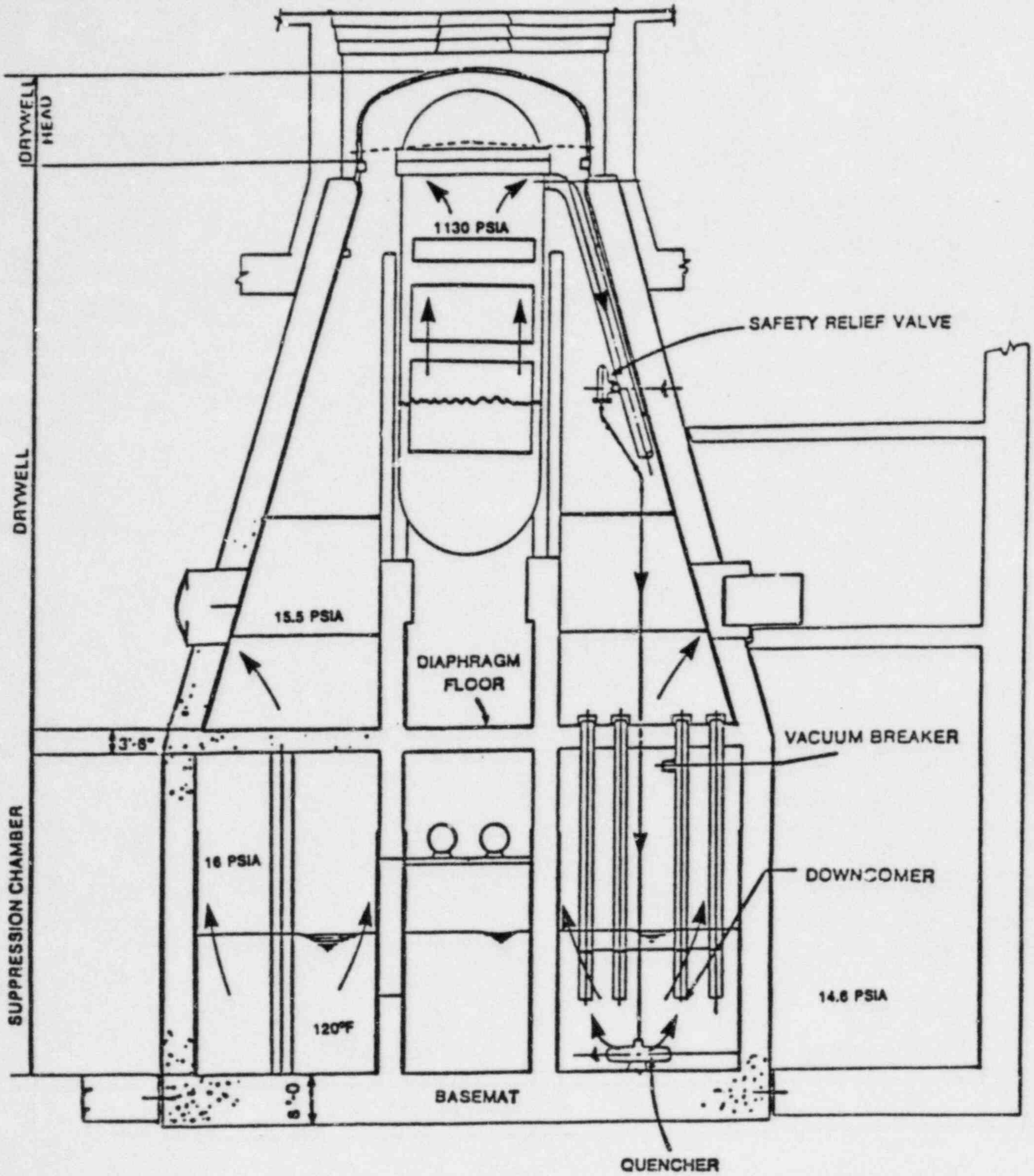
$\gamma, \gamma', \gamma''$





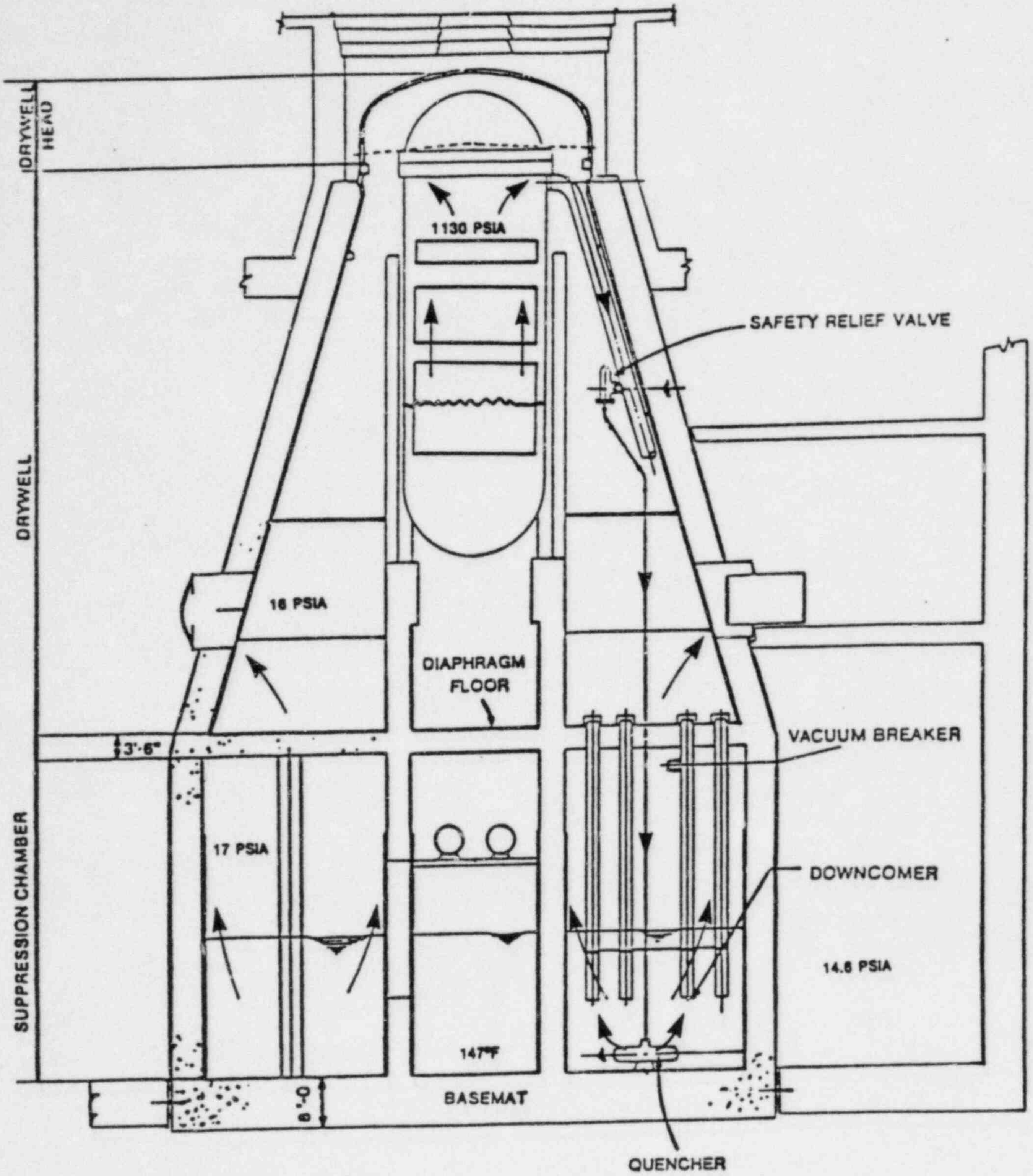
TIME = 0+ MINUTES
LGS CONTAINMENT CONDITIONS
CLASS I ACCIDENT SEQUENCE
INITIAL CONDITIONS

GRAPHICAL REPRESENTATION OF
PHYSICAL PROCESS IN CONTAINMENT



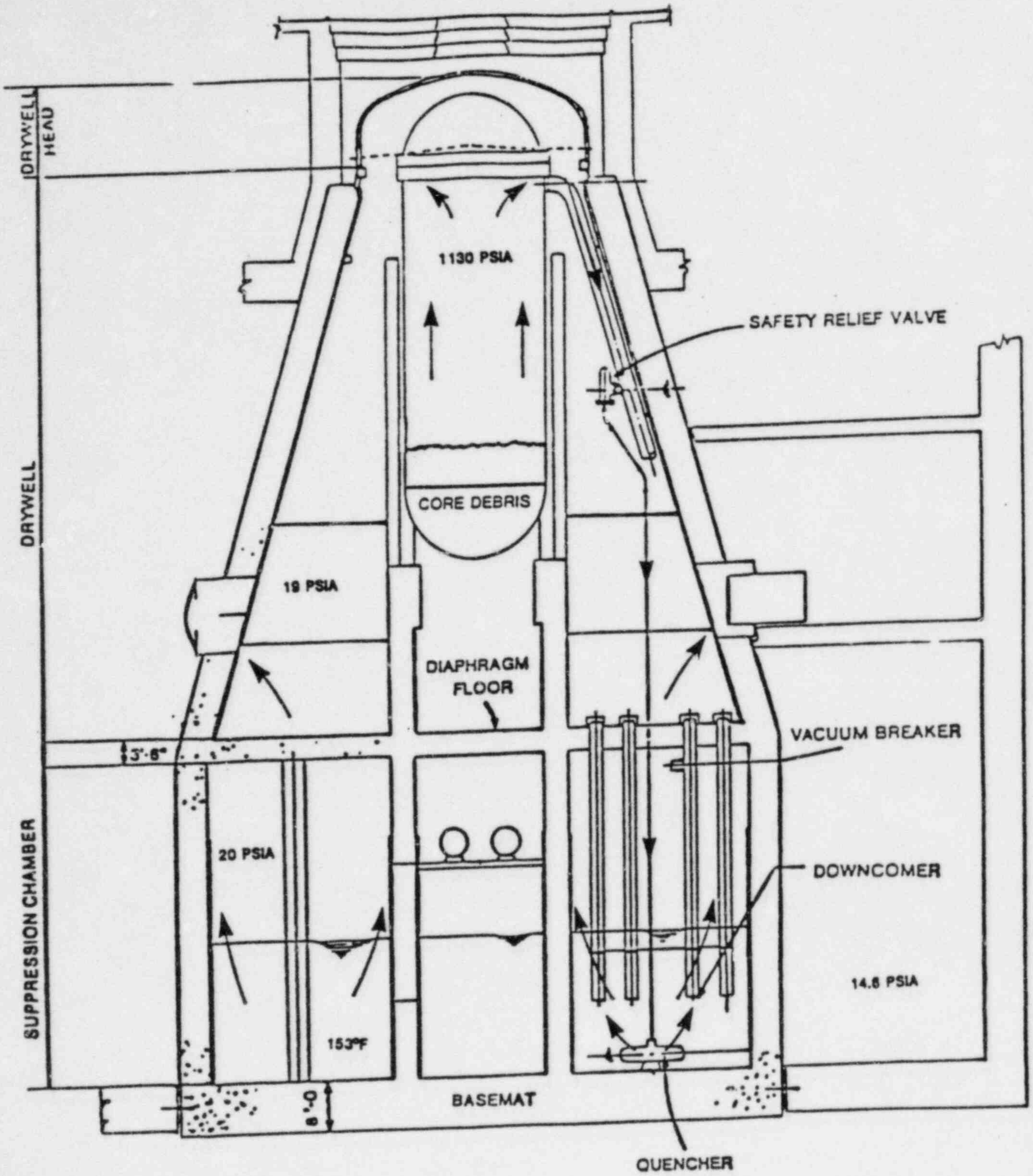
TIME = 50 MINUTES
LGS CONTAINMENT CONDITIONS
CLASS I ACCIDENT SEQUENCE
WATER LEVEL BELOW TAF.

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



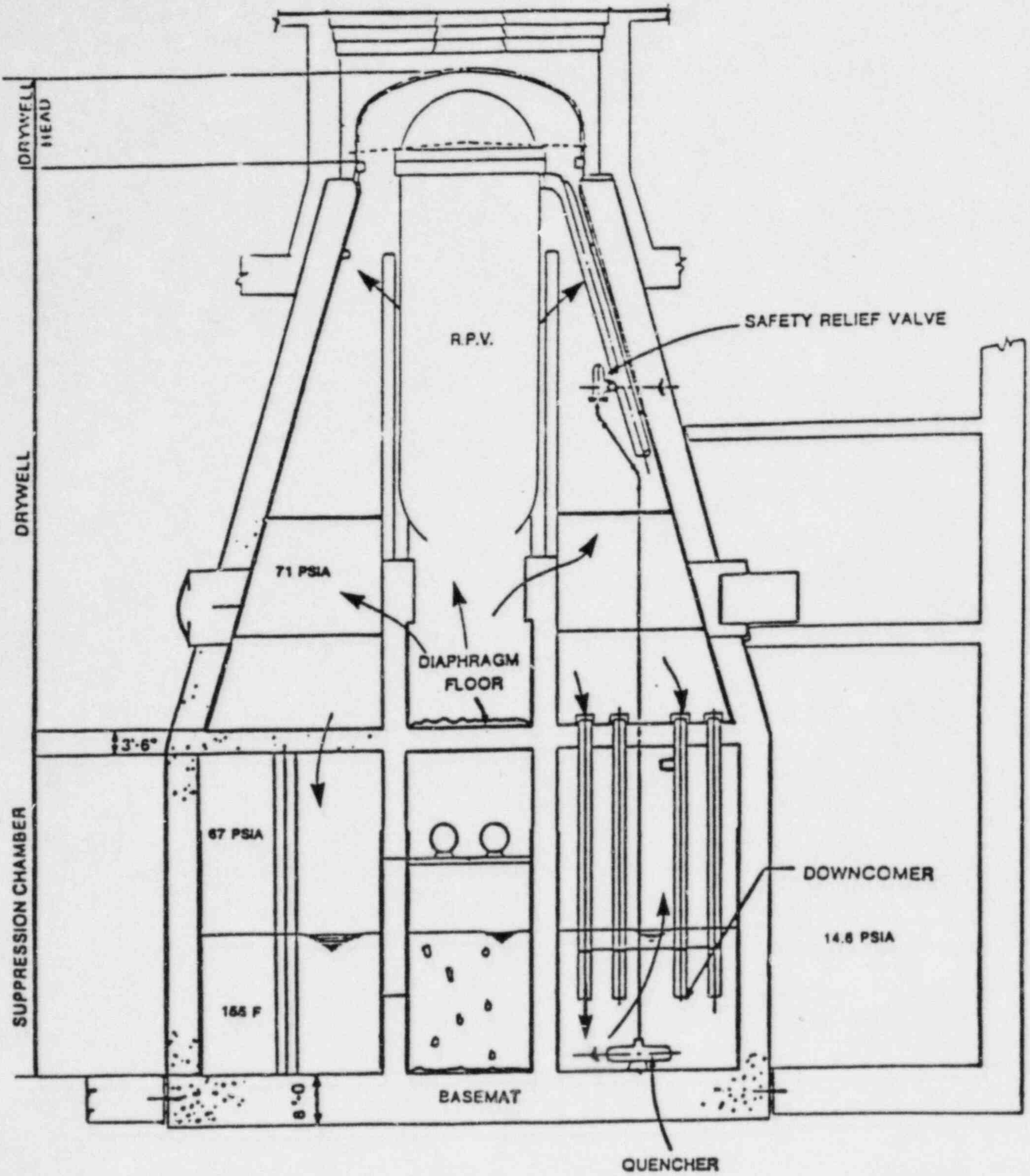
TIME = 1.3 HOURS
LGS CONTAINMENT CONDITIONS
CLASS I ACCIDENT SEQUENCE
CORE MELT INITIATION

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



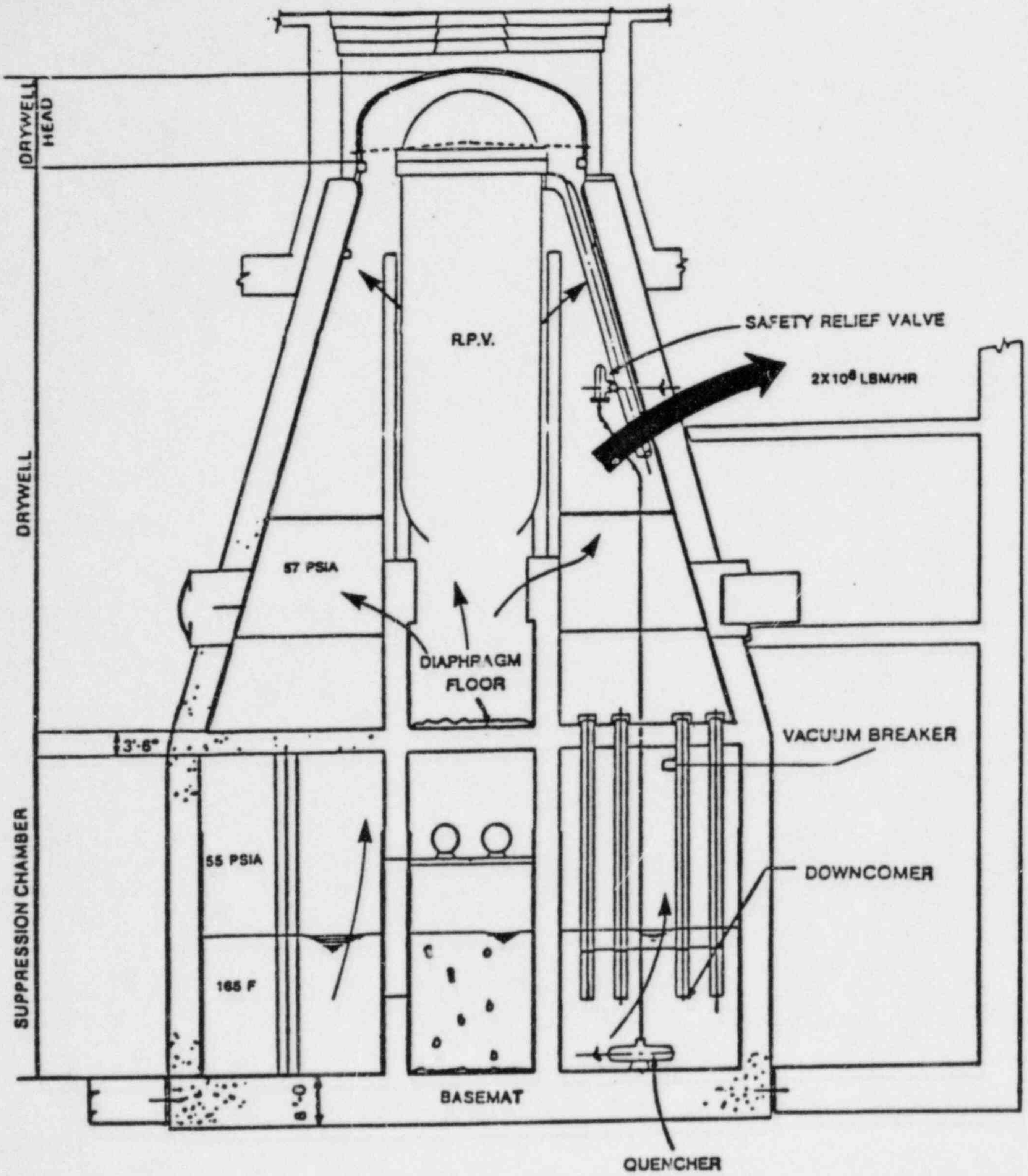
TIME = 2.5 HOURS
LGS CONTAINMENT CONDITIONS
CLASS I ACCIDENT SEQUENCE
CORE SLUMP & VESSEL HEAD ATTACK

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



TIME = 4.3 HOURS
LGS CONTAINMENT CONDITIONS
CLASS I ACCIDENT SEQUENCE
VESSEL RUPTURE AND DIAPHRAGM
FLOOR CONCRETE ATTACK

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



TIME = 6.5 HOURS
LGS CONTAINMENT CONDITIONS
CLASS I ACCIDENT SEQUENCE
DIAPHRAGM FLOOR PENETRATION (70 CM)
AND CONTAINMENT FAILURE

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT

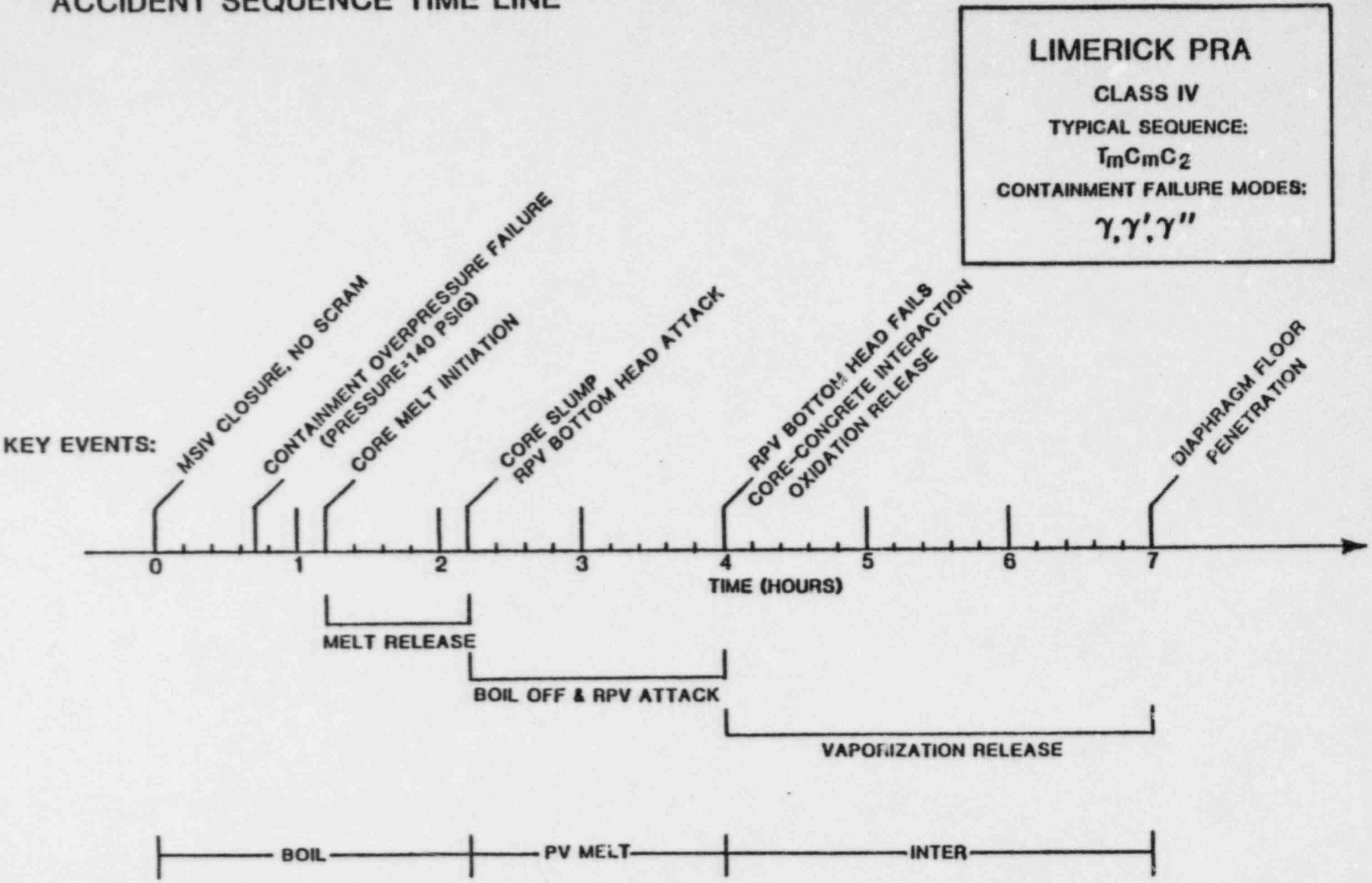
CLASS IV ACCIDENT SEQUENCES

- ISOLATION EVENT OCCURS
- REACTOR DOES NOT SCRAM
- COOLANT INJECTION CONTINUES
- POWER AT 30%

CLASS IV ACCIDENT SEQUENCES

- STEAM THROUGH RELIEF VALVES
- SUPPRESSION POOL SATURATED
- CONTAINMENT FAILURE OCCURS RAPIDLY
- COOLANT INJECTION FAILS
- CORE MELTS INTO A FAILED CONTAINMENT
- RELEASES OCCUR IN OPEN CONTAINMENT

ACCIDENT SEQUENCE TIME LINE



LIMERICK PRA

CLASS IV

TYPICAL SEQUENCE:

$T_m C_m C_2$

CONTAINMENT FAILURE MODES:

$\gamma, \gamma', \gamma''$

KEY EVENTS:

MSIV CLOSURE, NO SCRAM

CONTAINMENT OVERPRESSURE FAILURE (PRESSURE=140 PSIG)

CORE MELT INITIATION

CORE SLUMP
RPV BOTTOM HEAD ATTACK

RPV BOTTOM HEAD FAILS
CORE-CONCRETE INTERACTION
OXIDATION RELEASE

DIAPHRAGM FLOOR PENETRATION

MELT RELEASE

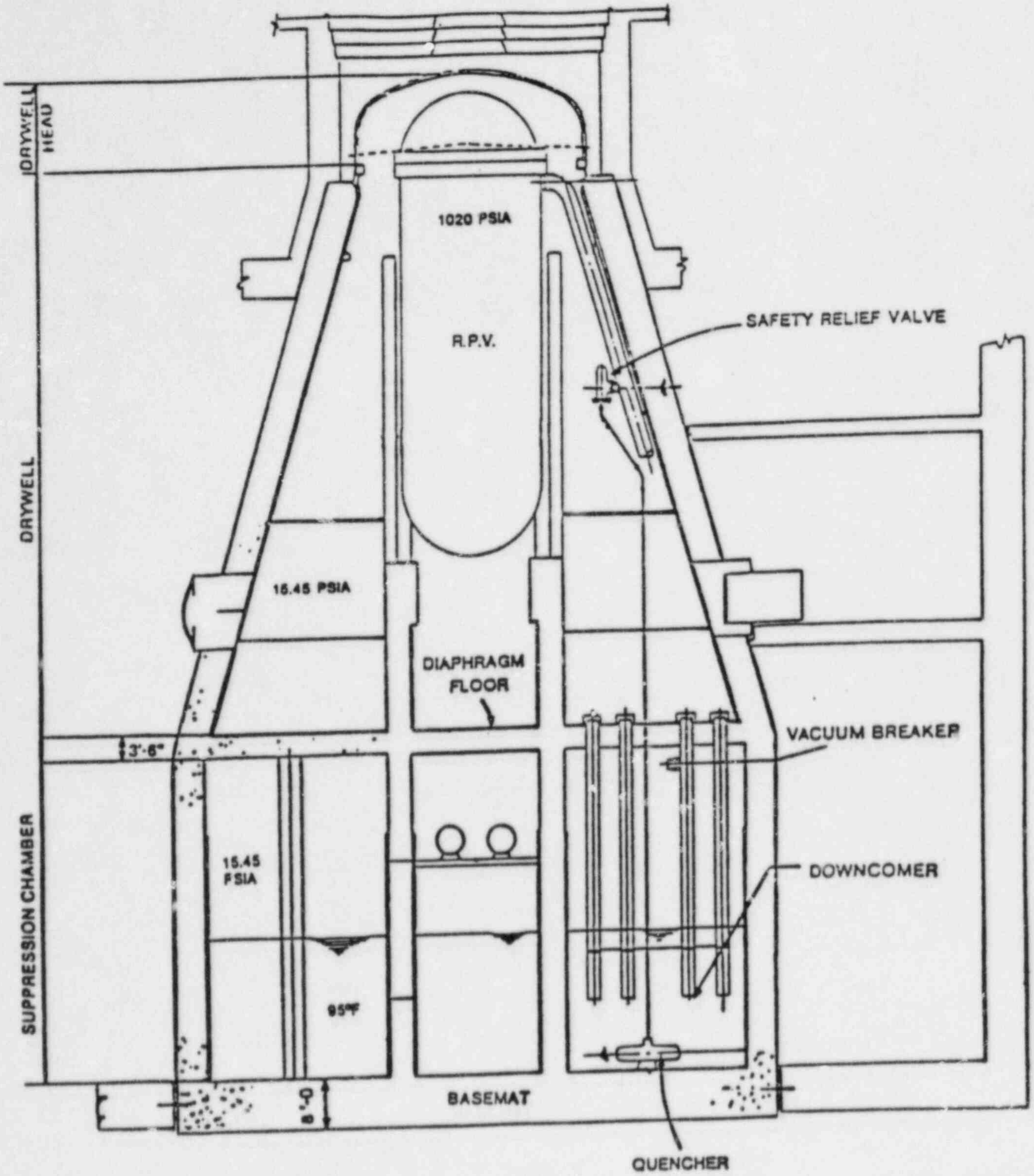
BOIL OFF & RPV ATTACK

VAPORIZATION RELEASE

BOIL

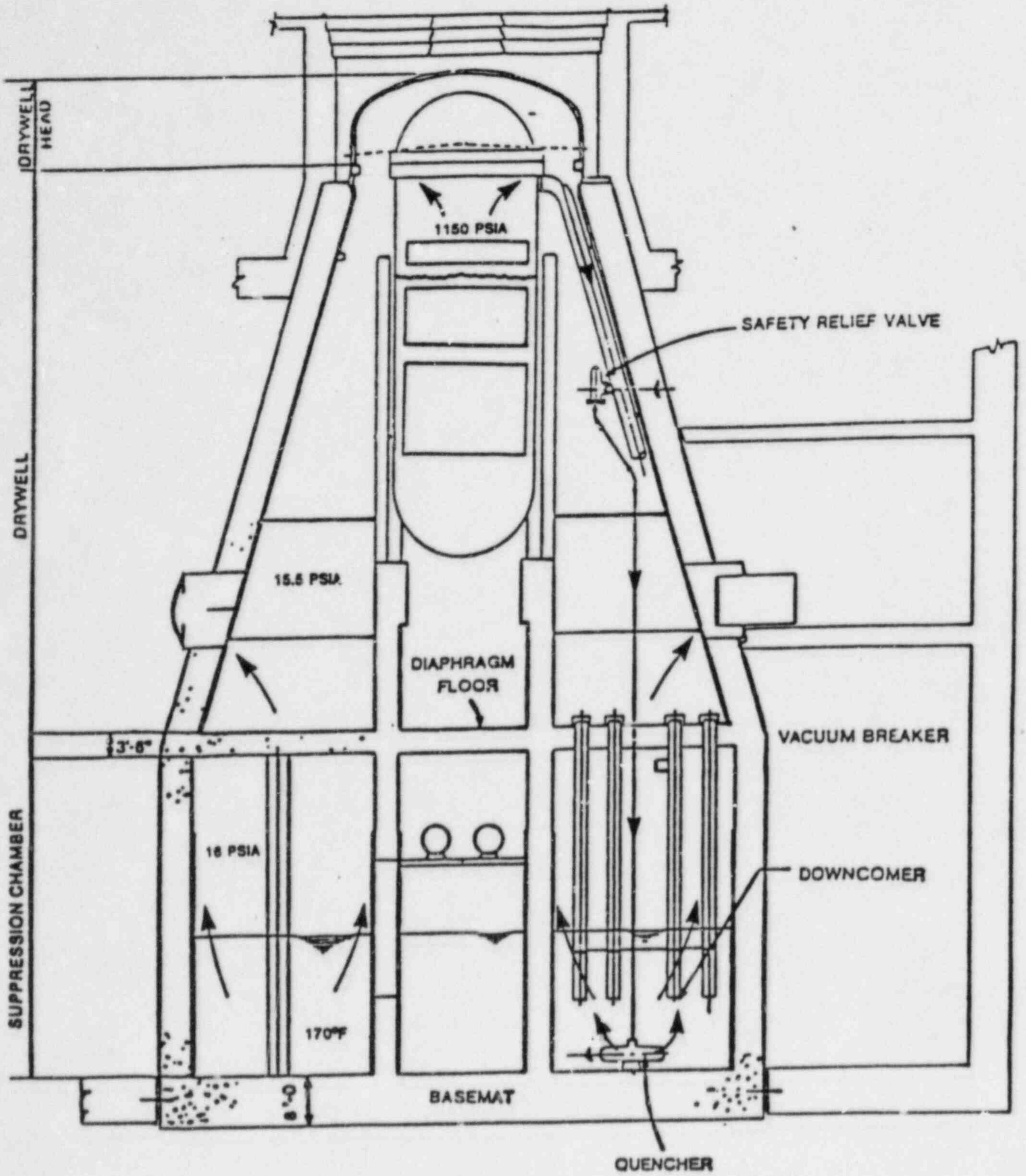
PV MELT

INTER



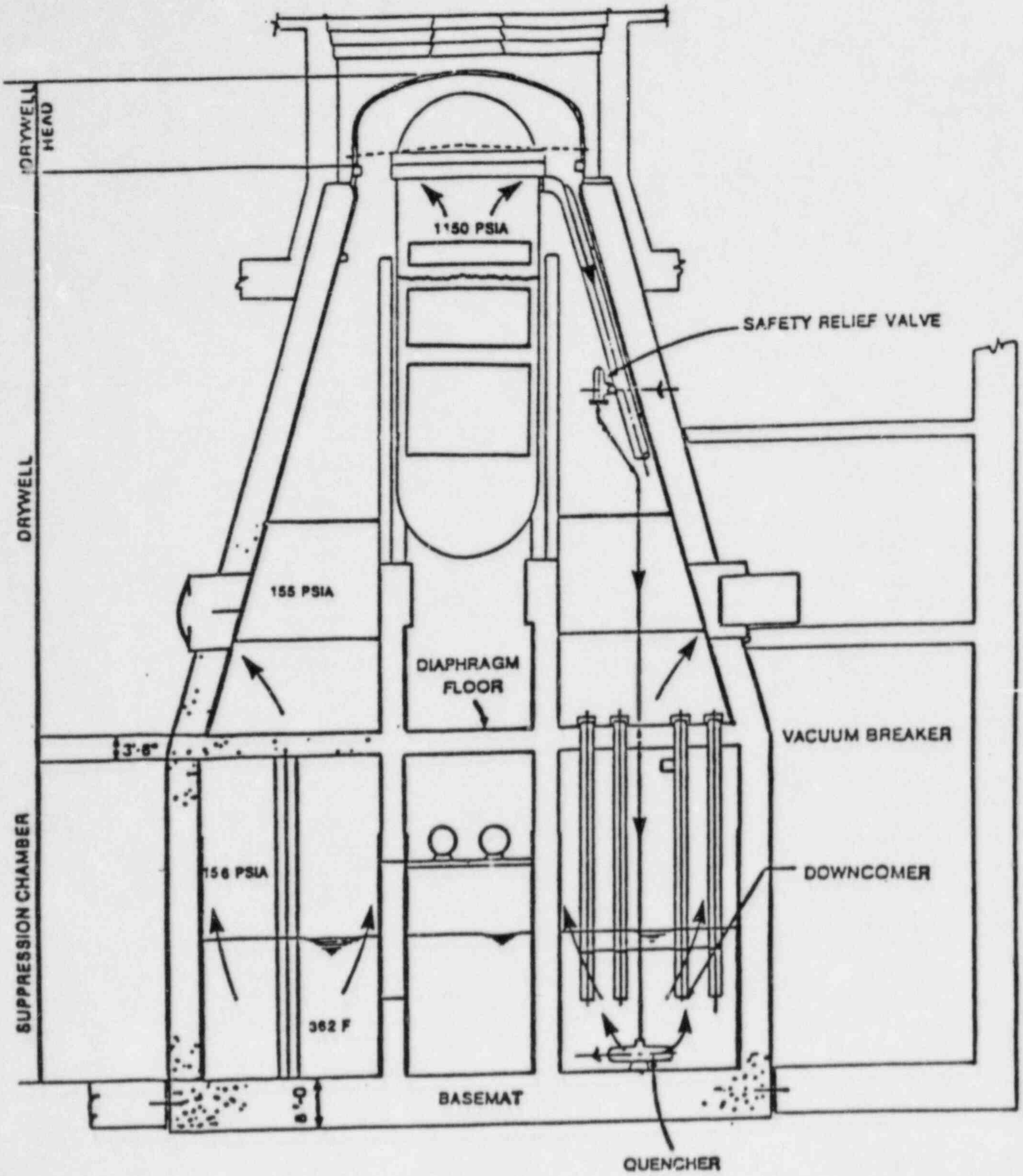
TIME = 0+ MINUTES
LGS CONTAINMENT CONDITIONS
CLASS IV ACCIDENT SEQUENCE
INITIAL CONDITIONS

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



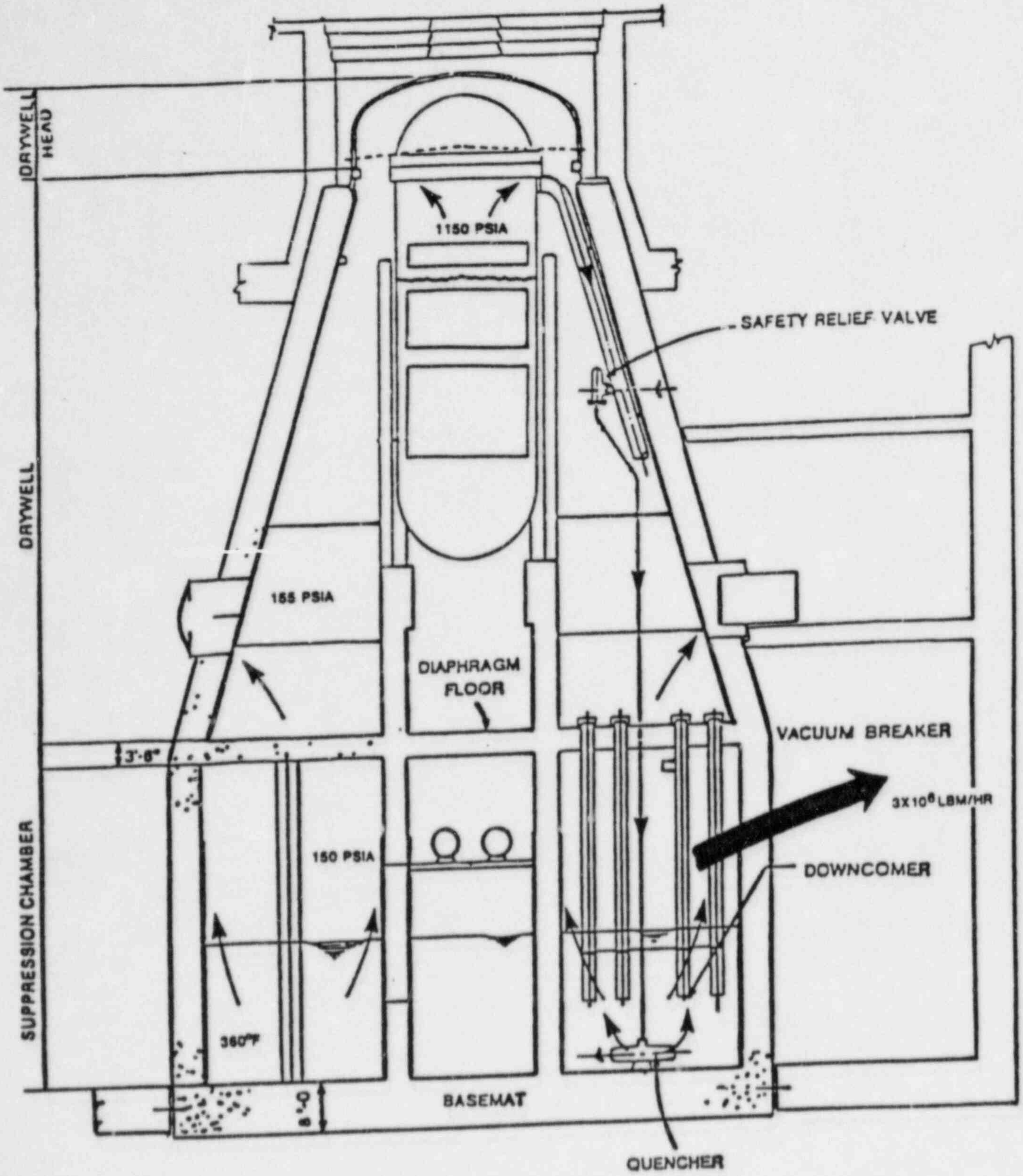
TIME = 10 MINUTES
LGS CONTAINMENT CONDITIONS
CLASS IV ACCIDENT SEQUENCE
CORE AT 30% POWER

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



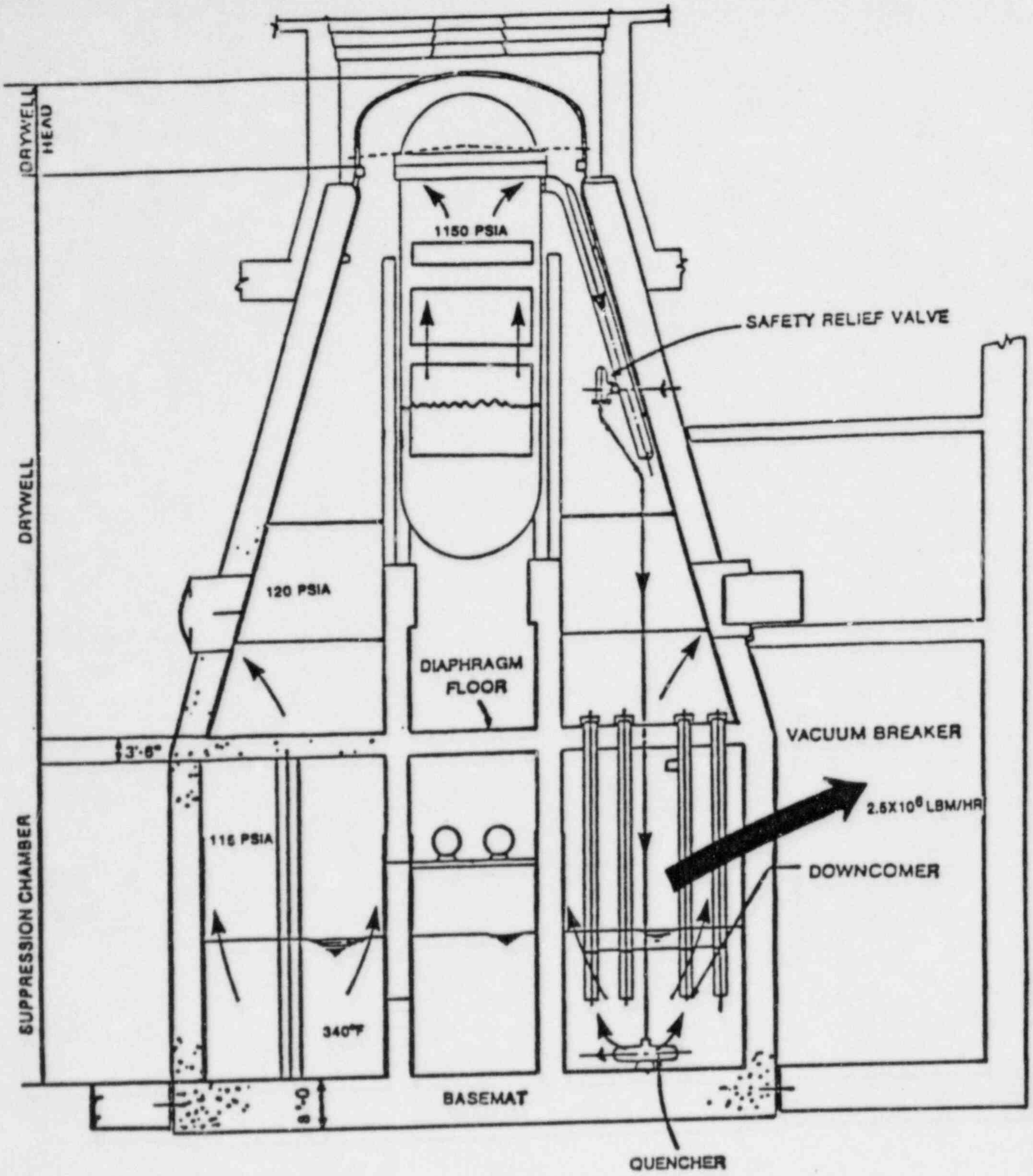
TIME = 40 MINUTES
LGS CONTAINMENT CONDITIONS
CLASS IV ACCIDENT SEQUENCE
CORE AT 30% POWER
CONTAINMENT AT ULTIMATE PRESSURE

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



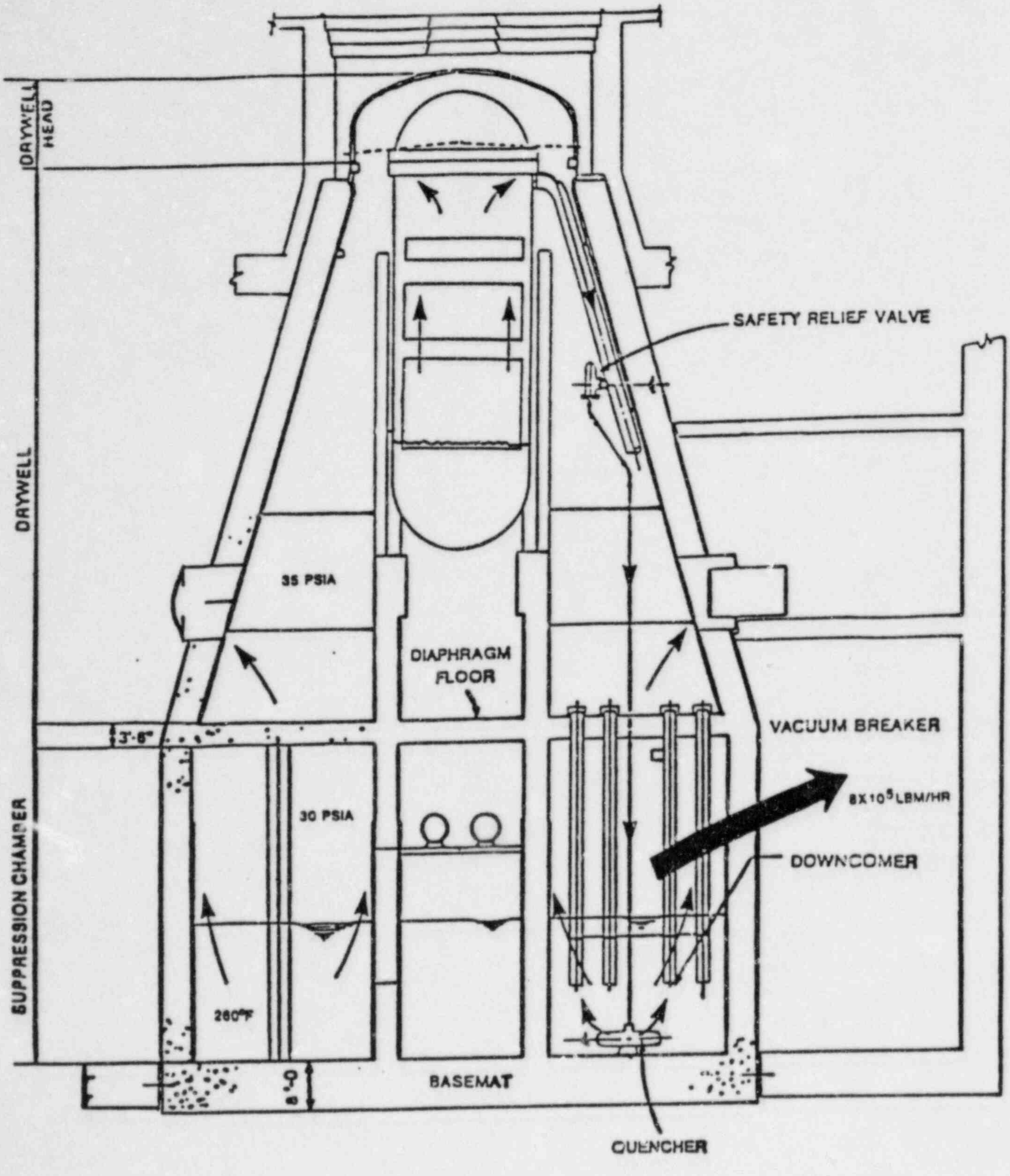
TIME = 40+ MINUTES
LGS CONTAINMENT CONDITIONS
CLASS IV ACCIDENT SEQUENCE
INITIAL CONTAINMENT DEPRESSURIZATION

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



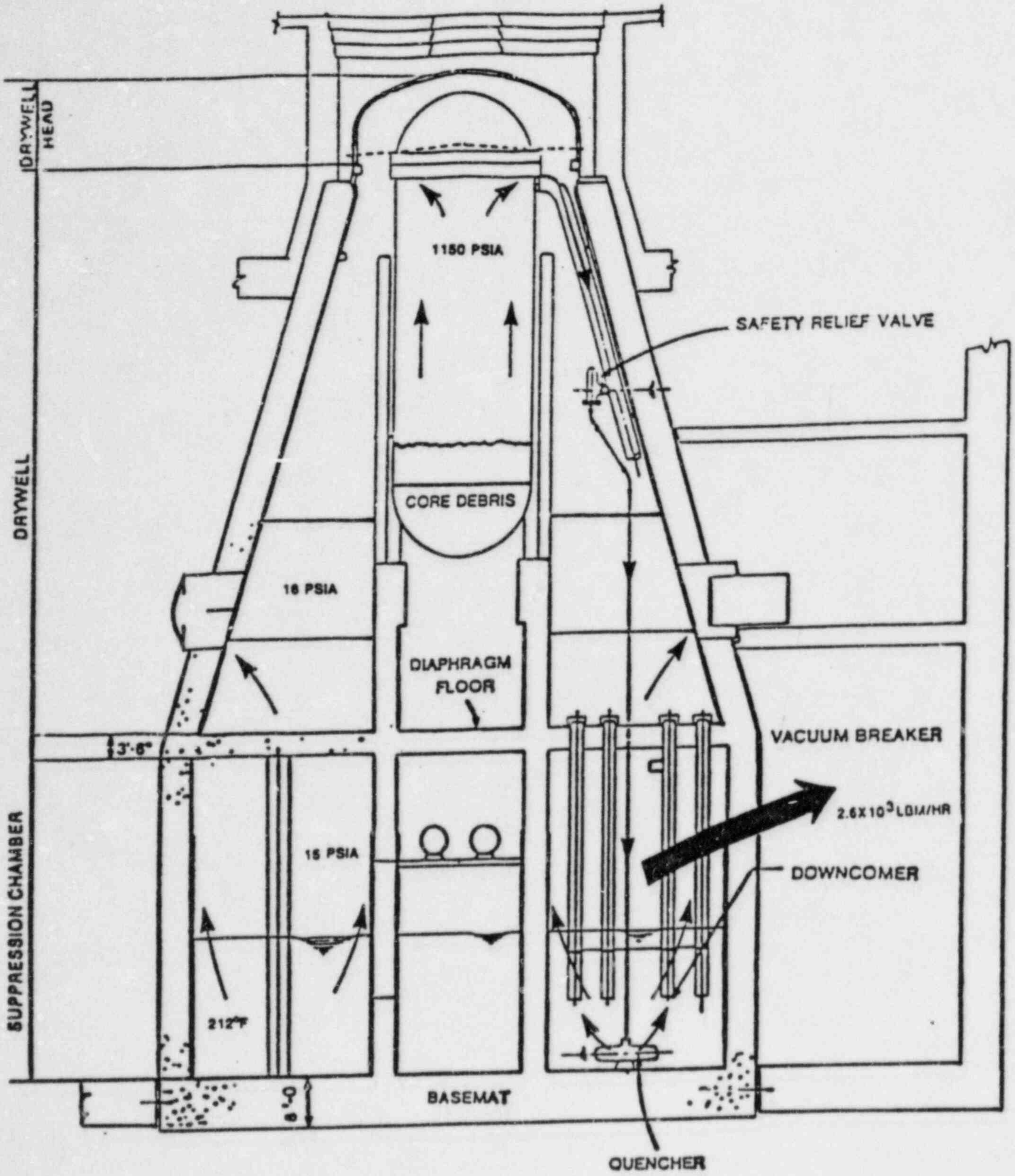
TIME = 45 MINUTES
LGS CONTAINMENT CONDITIONS
CLASS IV ACCIDENT SEQUENCE
LOSS OF CORE COOLANT MAKEUP
AND WATER LEVEL BELOW TAF

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



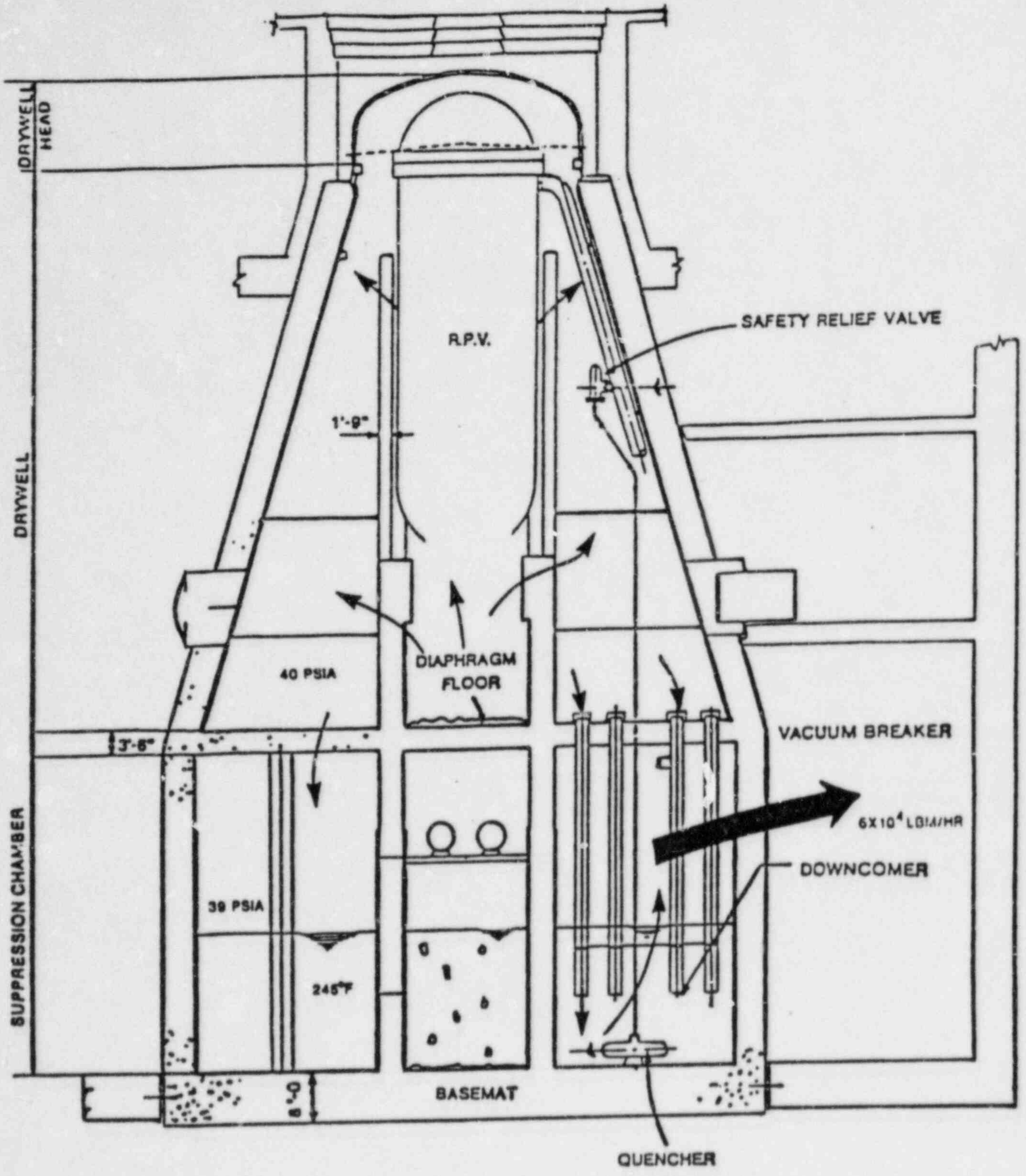
TIME = 1.2 HOURS
LGS CONTAINMENT CONDITIONS
CLASS IV ACCIDENT SEQUENCE
CORE MELT INITIATION

GRAPHICAL REPRESENTATION OF
PHYSICAL PROCESS IN CONTAINMENT



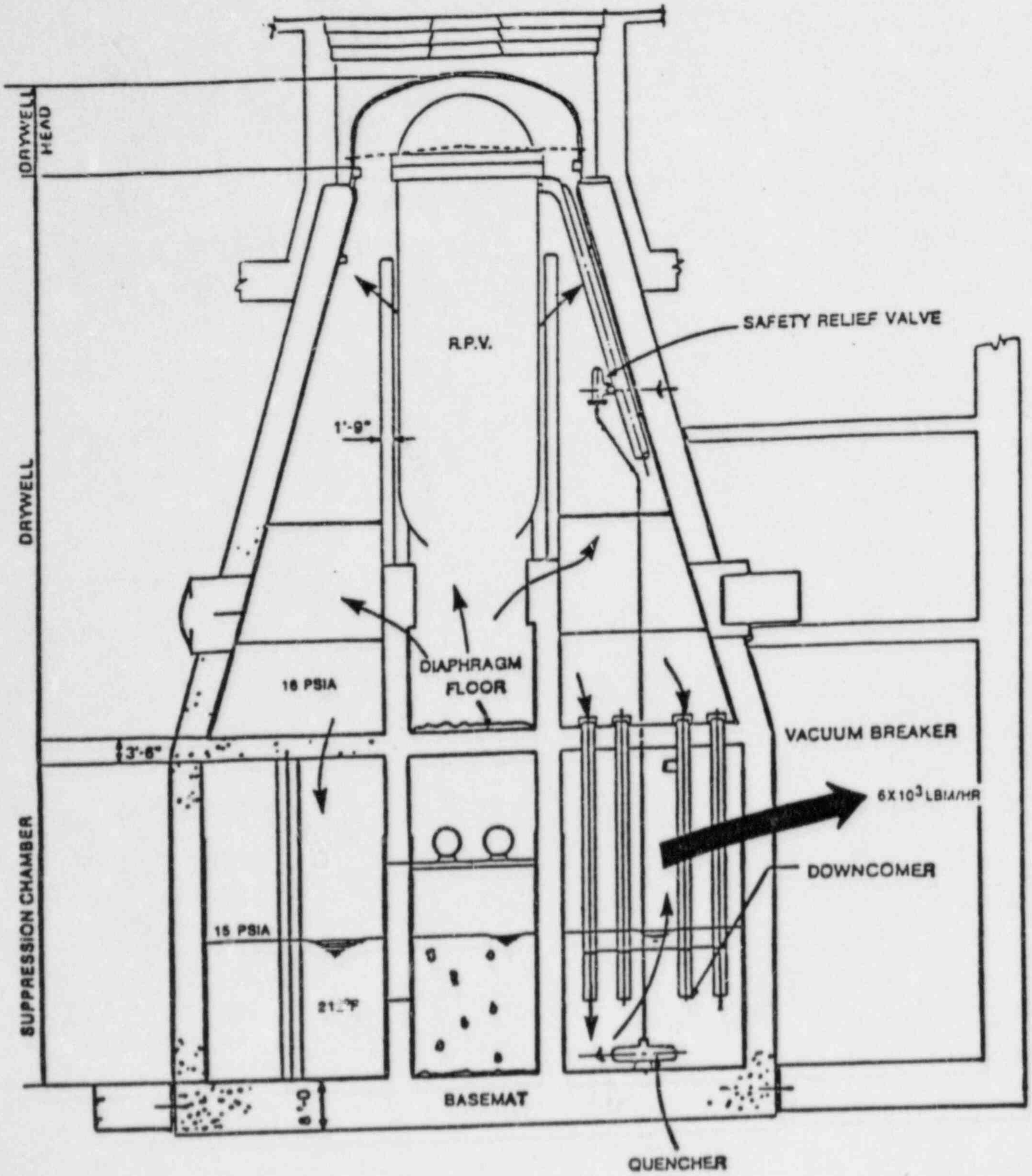
TIME = 2.2 HOURS
LGS CONTAINMENT CONDITIONS
CLASS IV ACCIDENT SEQUENCE
CORE SLUMP AND VESSEL HEAD ATTACK

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



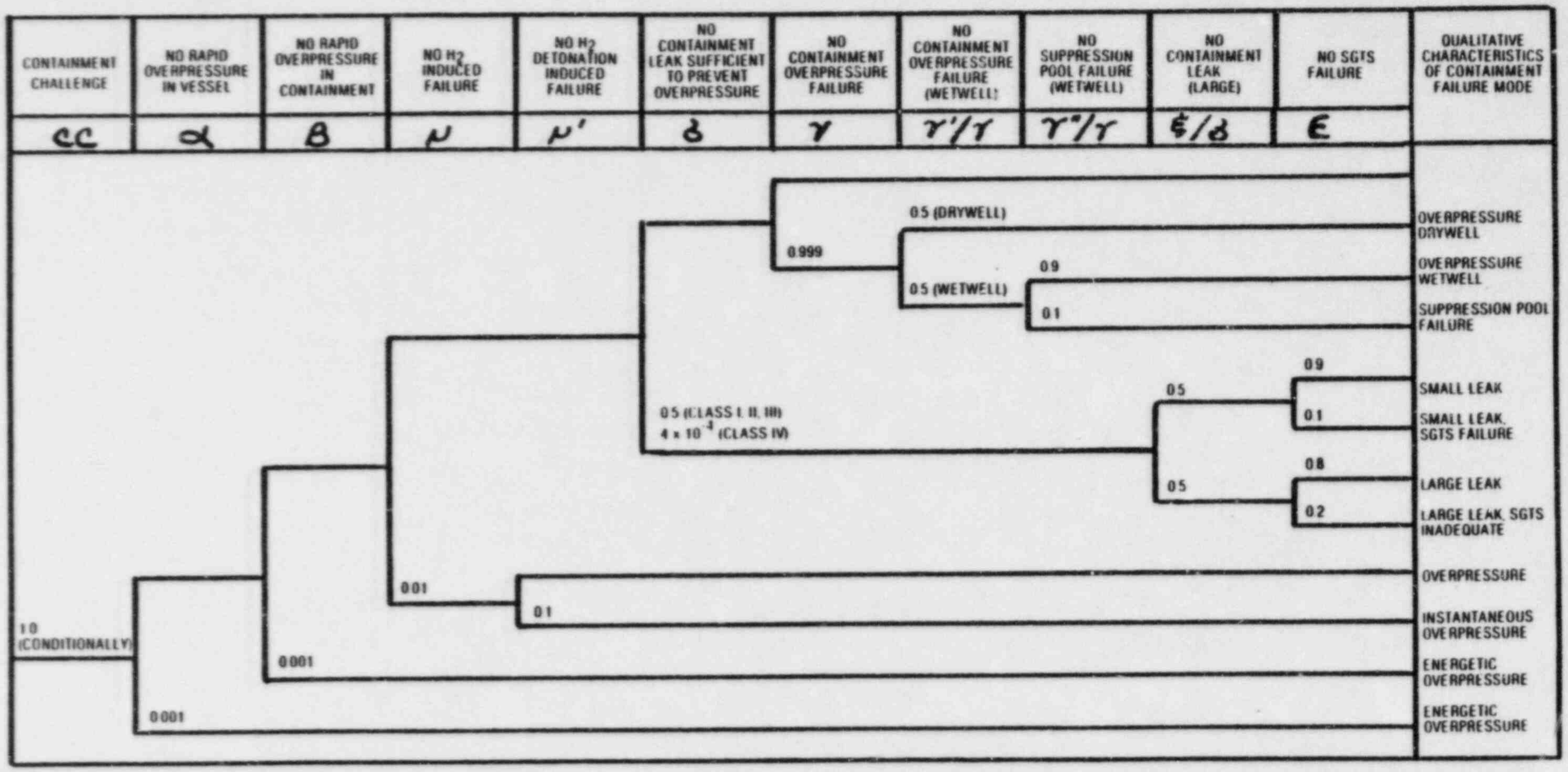
TIME = 4 HOURS
LGS CONTAINMENT CONDITIONS
CLASS IV ACCIDENT SEQUENCE
VESSEL RUPTURE AND DIAPHRAGM
FLOOR CONCRETE ATTACK

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



TIME = 7 HOURS
CLASS IV ACCIDENT SEQUENCE
DIAPHRAGM FLOOR PENETRATION (70 CM)

GRAPHICAL REPRESENTATION OF PHYSICAL PROCESS IN CONTAINMENT



CONTAINMENT EVENT TREE FOR THE MARK II CONTAINMENT

RESULTS OF BINNING:
ELEVEN RADIONUCLIDE RELEASE CATEGORIES

OXRE — ALL STEAM, H_2 EXPLOSIONS FOR ALL CLASSES

OPREL — CLASS 1, 2, 3 OVERPRESSURE
DRYWELL AND WETWELL FAILURE ABOVE POOL

C4γ — CLASS 4 DRYWELL FAILURE

C4γ' — CLASS 4 WETWELL AIRSPACE FAILURE

C4γ" — CLASS 4 POOL FAILURE

LEAKS — WITH OR WITHOUT FILTRATION

C123 γ" — OTHER EVENTS WITH POOL FAILURE

RB (IS) — POOL PARTIALLY DRAINED, SEISMIC FAILURE

VR — SEISMIC, VESSEL RUPTURE (DRY)

VRH20 — SEISMIC, VESSEL RUPTURE (WET)

RELEASE CATEGORY

RADIONUCLIDE RELEASE FRACTIONS

	RADIONUCLIDE RELEASE FRACTION		
Source Term	I	Cs	Te
OXRE	0.20	0.06	0.50
OPREL	0.11	0.09	0.016
C4-γ	0.261	0.202	0.434
C4-γ'·	0.07	0.09	0.20
C4-γ"	0.73	0.70	0.55
C123-γ"	0.13	0.17	0.50
LEAK 1	0.019	0.0098	0.046
LEAK 2	0.0027	0.000098	0.00046
RB	0.05	0.09	0.09
VR	0.1	0.33	0.33
VRH2O	0.5	0.73	0.75

SOURCE-TERM RELEASE CATEGORY CHARACTERISTICS

SOURCE TERM	T_r (hr)	T_d (hr)	T_w (hr)	h (m)	Q (cal/sec)
OXRE	4.0	0.5	3.0	27	8.4×10^{6}
OPREL	7.0	2.0	6.0	27	8.4×10^{6}
C4 γ	1.5	2.0	1.0	27	7.0×10^{4}
C4 γ'	1.5	2.0	1.0	27	7.0×10^{4}
C4 γ''	1.5	2.0	1.0	10	7.0×10^{4}
C123 γ''	7.0	2.0	6.0	10	7.0×10^{4}
LEAK 1	7.0	2.0	6.0	27	7.0×10^{4}
LEAK 2	7.0	2.0	6.0	27	7.0×10^{4}
RB	1.5	3.0	1.5	10	8.4×10^{6}
VR	0.25	3.5	0.25	10	1.4×10^{4}
VRH2O	0.34	0.65	0.34	10	2.0×10^{6}

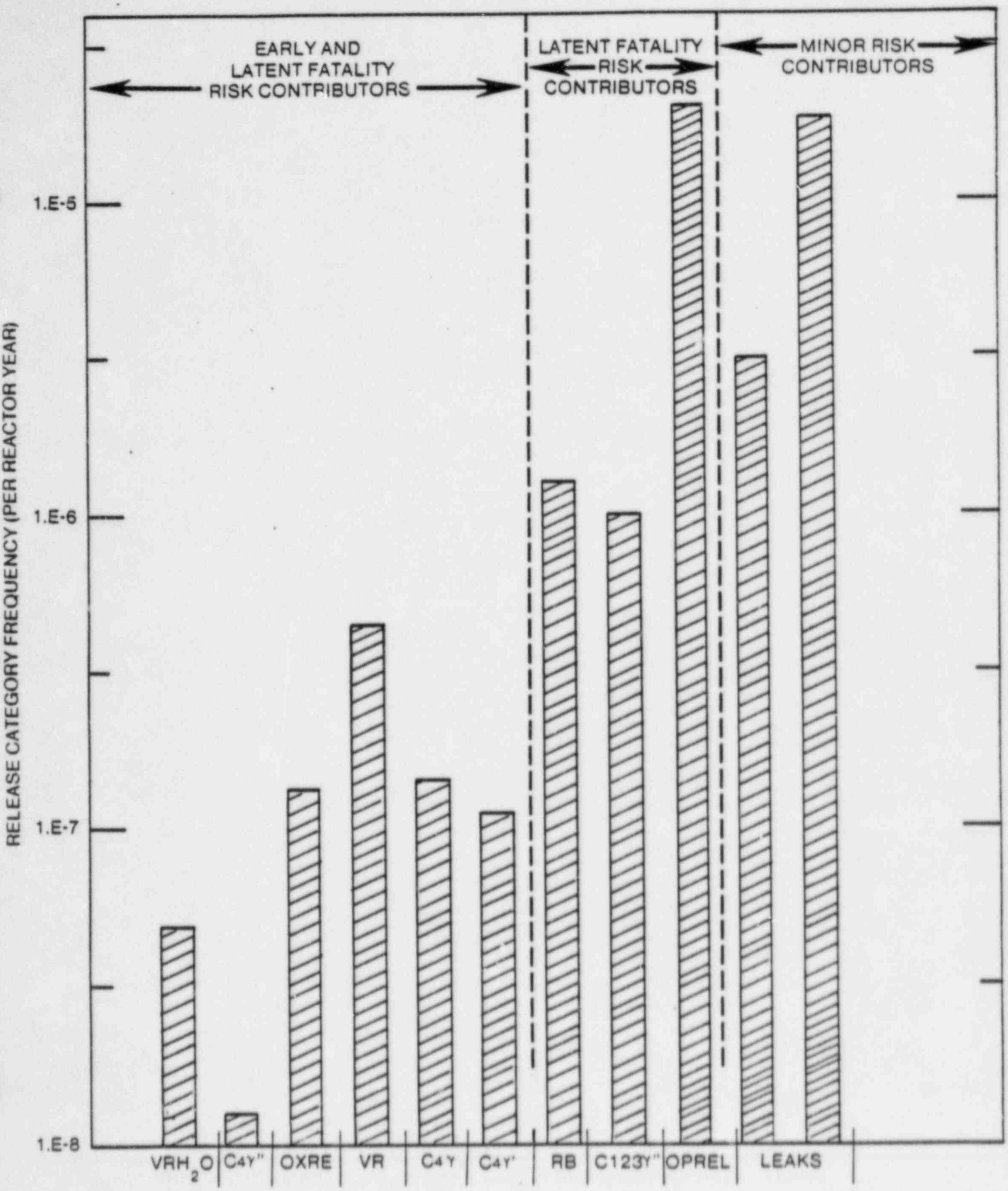
T_r = Time of Release

T_d = Duration of Release

T_w = Warning Time

h = Height of Release

Q = Energy of Release



COMPARISON OF RADIONUCLIDE RELEASE CATEGORY FREQUENCY VERSUS CONSEQUENCE IMPACT

ITEMS NOT INCLUDED

- SOME CURRENT PHENOMENOLOGY—
  BOB HENRY TO COVER
- NO CREDIT FOR CONTAINMENT SPRAYS
- NO RHR OPERATION DURING CORE DEGRADATION
- NO VENTING OF CONTAINMENT
- NO SUCCESSFUL LOW PRESSURE INJECTION FOR ATWS
- NO EXTERNAL WATER SOURCES ASSUMED AVAILABLE

  FOR CORE COOLANT INJECTION

  - SERVICE WATER
  - FIRE PUMPS
  - KEEP FULL SYSTEMS
  - CONDENSATE TRANSFER PUMPS
  - CRD

- NO CREDIT FOR ADS ON LOW LEVEL ONLY
- CORE CONCRETE ATTACK (INTER vs. CORCON)
- PRIMARY SYSTEM RETENTION ASSUMED INEFFECTIVE
- ADDITIONAL DECONTAMINATION FACTOR BENEFIT
- NON-PROCEDURAL OPERATOR INTERVENTION ERRORS
  (ERRORS OF COMMISSION)

SUMMARY OF LGS PRA
IN-PLANT PHYSICS EVALUATION

- METHODOLOGY SIMILAR TO WASH-1400 UPDATED TO "STATE-OF-ART" 1980
- ADVANCES IN UNDERSTANDING:
  - STEAM EXPLOSION PROBABILITY DECREASED
  - DECONTAMINATION FACTORS OF SUPPRESSION POOL INCREASED
  - MARK II CONTAINMENT FAILURE PRESSURE
  - FISSION PRODUCT RETENTION IN REACTOR BUILDING
  - RPV FAILURE MODE
  - MOLTEN CORE DEBRIS DISPOSITION
- SOURCE TERMS COMPARABLE TO WASH-1400
- ACCIDENT SEQUENCE FREQUENCIES COMPARABLE BUT LOWER THAN WASH-1400

INFLUENCE OF ADDITIONAL MODELING

- Some sequences initially judged to result in core melt are coolable with limited or no core damage.
- Some core melt sequences do not result in containment failure.
- In general, containment failure sequences have a considerably longer time interval before containment failure. Containment failure due to overpressure by steam generation.

ASSUMPTIONS IN LIMERICK PRA

- No CRD flow.
- RCIC injection insufficient for cooling the core during an ATWS without SLC injection.
- Large quantities of molten core material (80%) required to fail the vessel.
- Debris distributed over the pedestal and drywell floors. (Concrete attack and failure by diaphragm floor penetration, also suppression pool cooling is not effective.)

cooling is not effective.)

- No primary system retention.

CURRENT MODELING

- CRD included unless prohibited by the sequence definition.
- RCIC injection included until failed or CST depleted.
- About 20% of the core material molten is sufficient to cause vessel failure.

CURRENT MODELING

(Continued)

- Some debris distributed over the pedestal and drywell with most (~ 90% or more) in the suppression pool - flow paths are floor drains in the pedestal and failed downcomers on the containment floor.
- Fission products released mechanistically - major fraction deposited within the primary system.
- Natural circulation and primary system heat up determine the ultimate fission product distribution.

INFLUENCES OF NATURAL CIRCULATION WITHIN THE PRIMARY SYSTEM

- The geometry of a BWR impedes natural circulation flows between the core and the upper plenum (separators and dryers).
- Zircaloy oxidation is generally limited by "steam starvation" and would not be greatly altered by the limited natural circulation of steam possible in a BWR.

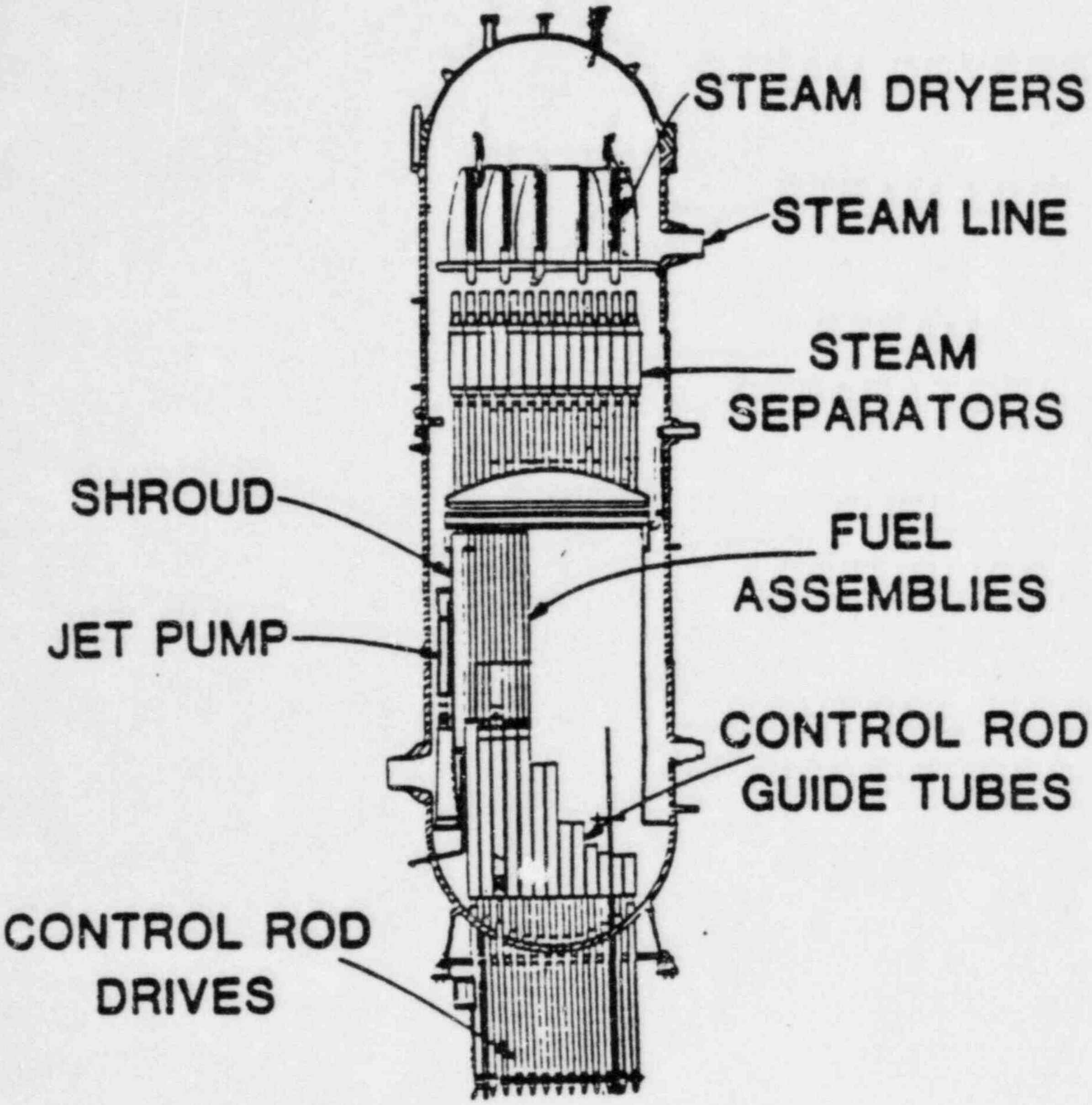
INFLUENCES OF NATURAL CIRCULATION WITHIN THE PRIMARY SYSTEM

(Continued)

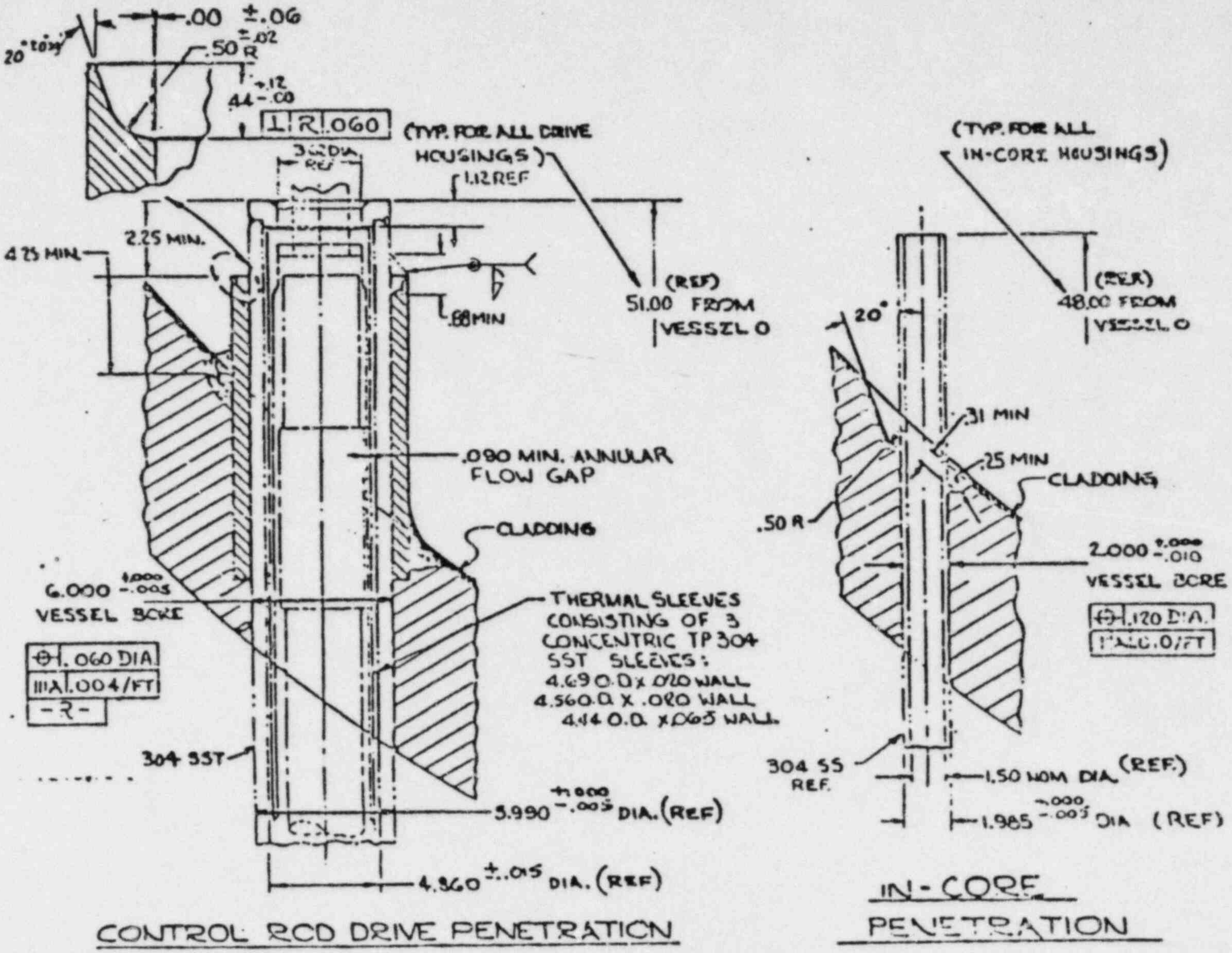
- Natural circulation of steam in the core bypass region of BWRs is difficult to sustain because the energy release within the fuel bundles controls the thermal-hydraulic conditions and causes higher temperature gases to remain at the top of the bundles.
- Following vessel failure, natural circulation within the primary system is the mechanism for transportation of revaporized fission products throughout the primary system.

VESSEL FAILURE MECHANISMS

- Failure of the core plate ~ 30 tonnes of debris would flow into the lower plenum and fail the weld(s) around a penetration(s).
- Melt through of in-core instrument tubes inside the core and discharge molten core materials through the tube.
- Both mechanisms result in a localized failure of the pressure boundary.



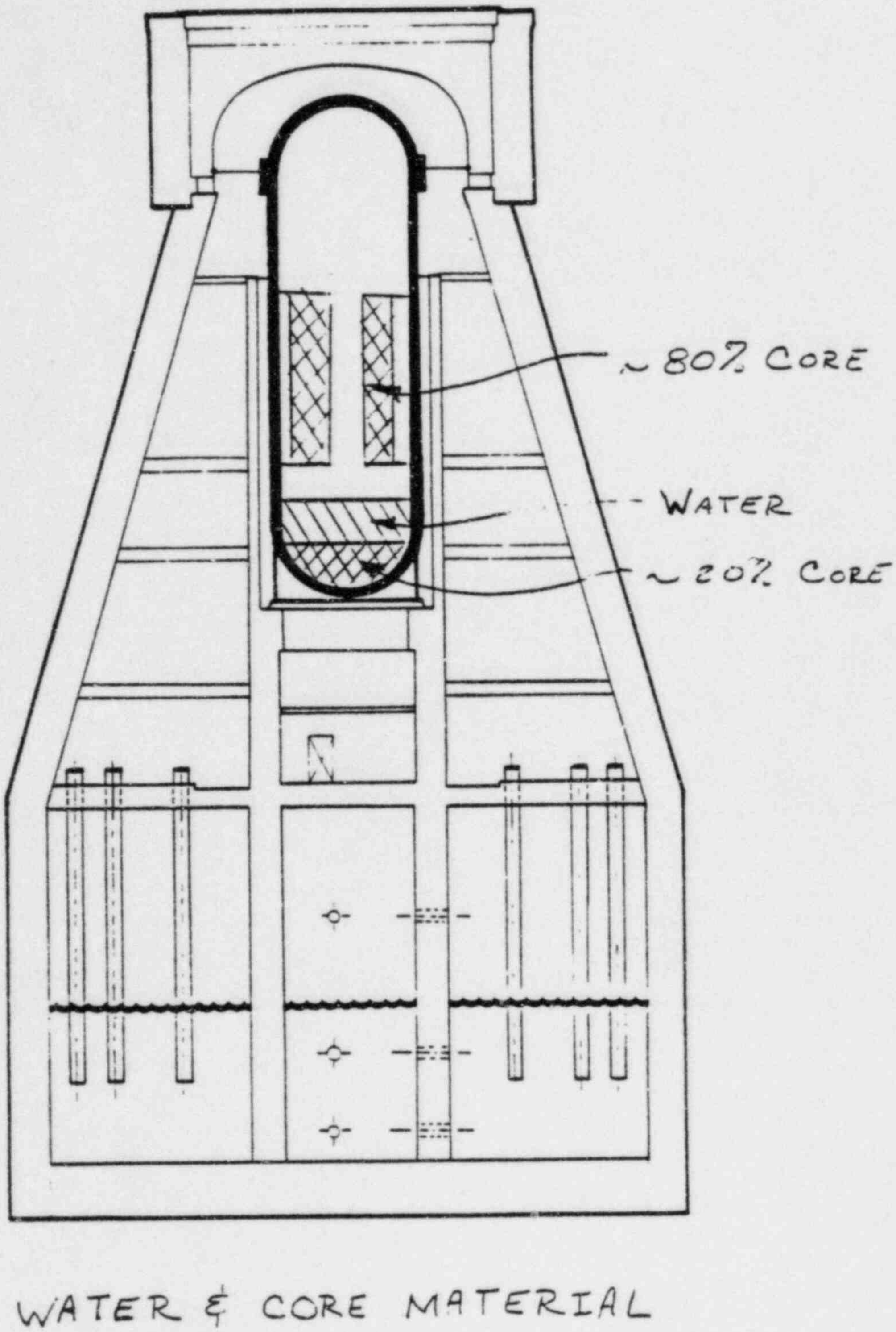
BWR VESSEL CONFIGURATION



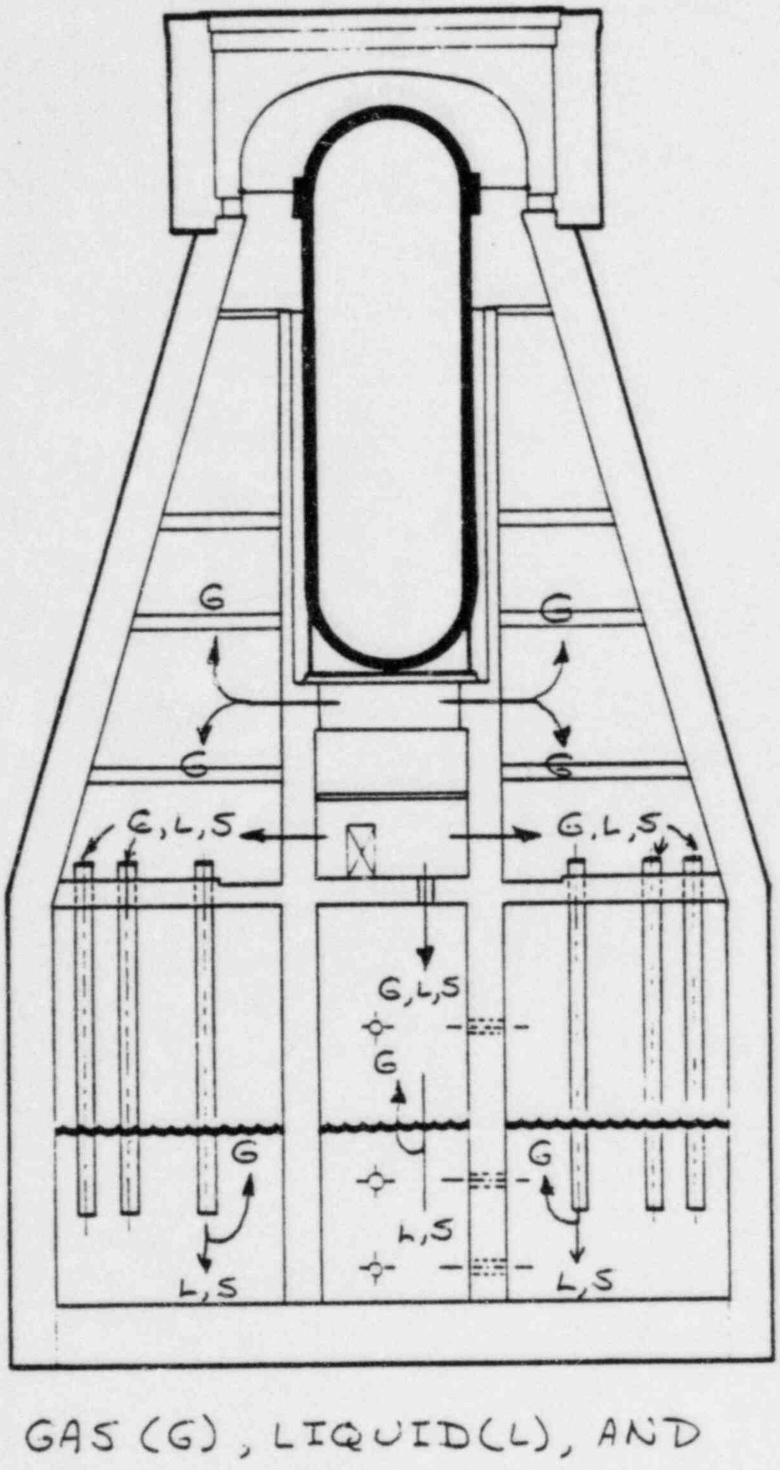
CONTROL ROD DRIVE PENETRATION

IN-CORE PENETRATION

FIG. 3-1 REACTOR VESSEL PENETRATION.



WATER & CORE MATERIAL DISTRIBUTION AT VESSEL FAILURE



GAS (G), LIQUID(L), AND
SOLID(L) FLOW PATHS
AFTER VESSEL FAILURE

DEBRIS DISPERSAL

- Only for significant pressures in the primary system > 1 MPa (150 psia).
- Water would be present in the lower plenum at vessel failure ~ 80 - 100 m^3.
- Flashing of water and noncondensable gas blowdown would dictate that 50 - 90% of the steam and gas would go through the suppression pool.
- Finely particulated core debris would follow the gas flow.
- Larger sizes of core debris would be separated from the gas flow along with the water and would be cooled by the water.

EXAMPLE OF DISPERSAL CONDITIONS

- Pressure – 7 MPa.
- Gas volume – 500 m^3.

  Average gas temperature – 800 K.
- Moles of gas – 526.

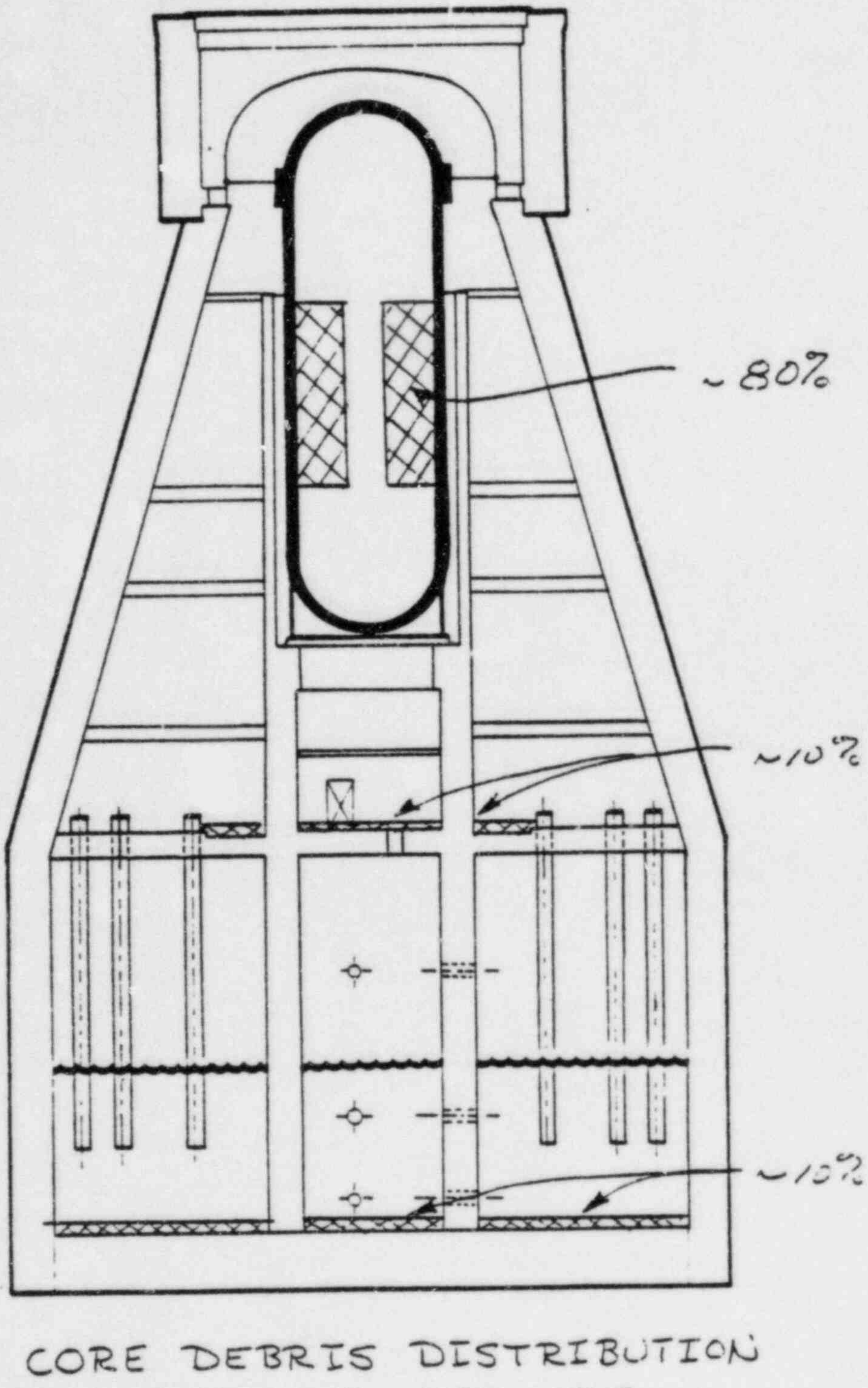
  Assume 50% steam – 50% H_2.
- Saturated water volume – 100 m^3.
- Water mass – 74,000 kg.
- Steam formed by flashing ~ 0.25 (74,000)

  = 19,000 kg.

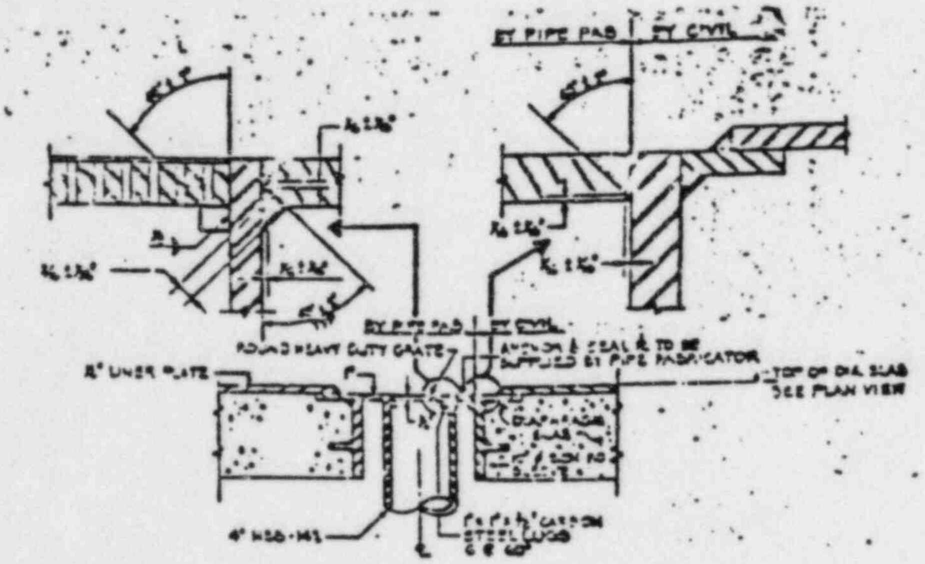
  ~ 1055 moles.

EXAMPLE OF DISPERSAL CONDITIONS
(Continued)

- Fraction of noncondensables $= \frac{263}{1318} = 0.2.$
- About 80% of gas flow goes through the pool, thus a similar fraction of finely particulated debris would go through the pool and be cooled.
- Mass of water on drywell and pedestal floor $\sim$ 55,000 kg – large heat sink.



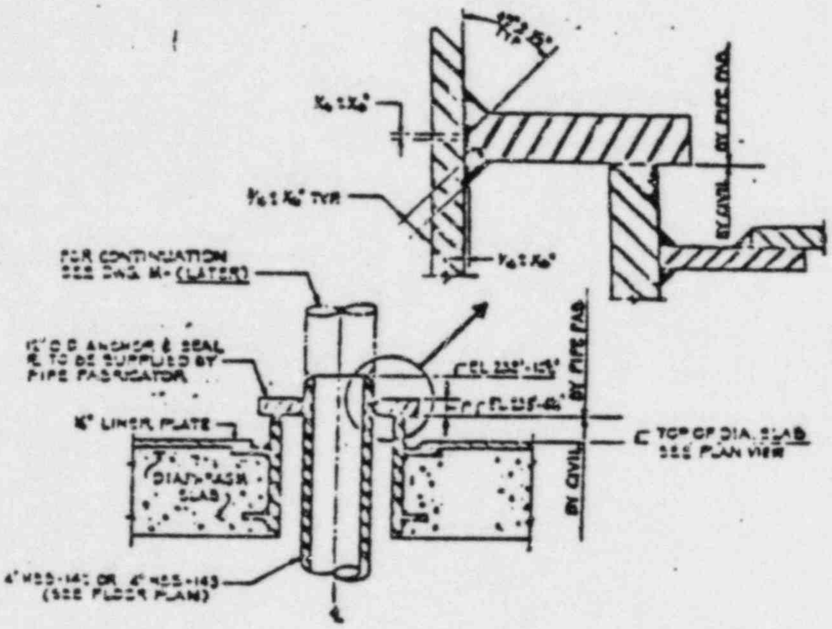
CORE DEBRIS DISTRIBUTION
AFTER VESSEL FAILURE



4" FLOOR DRAIN CONNECTION DETAIL

NO SCALE

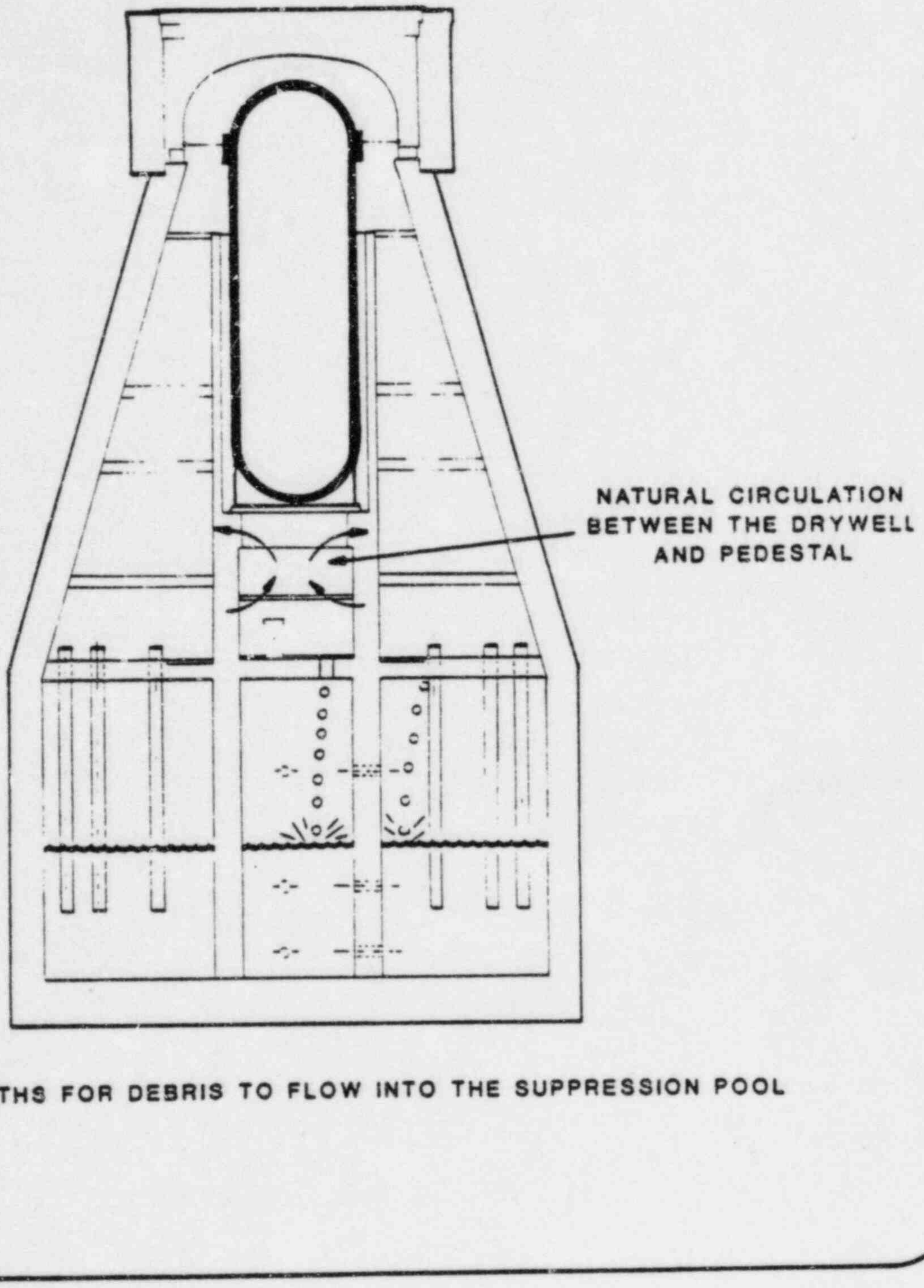
FIG. 5-1



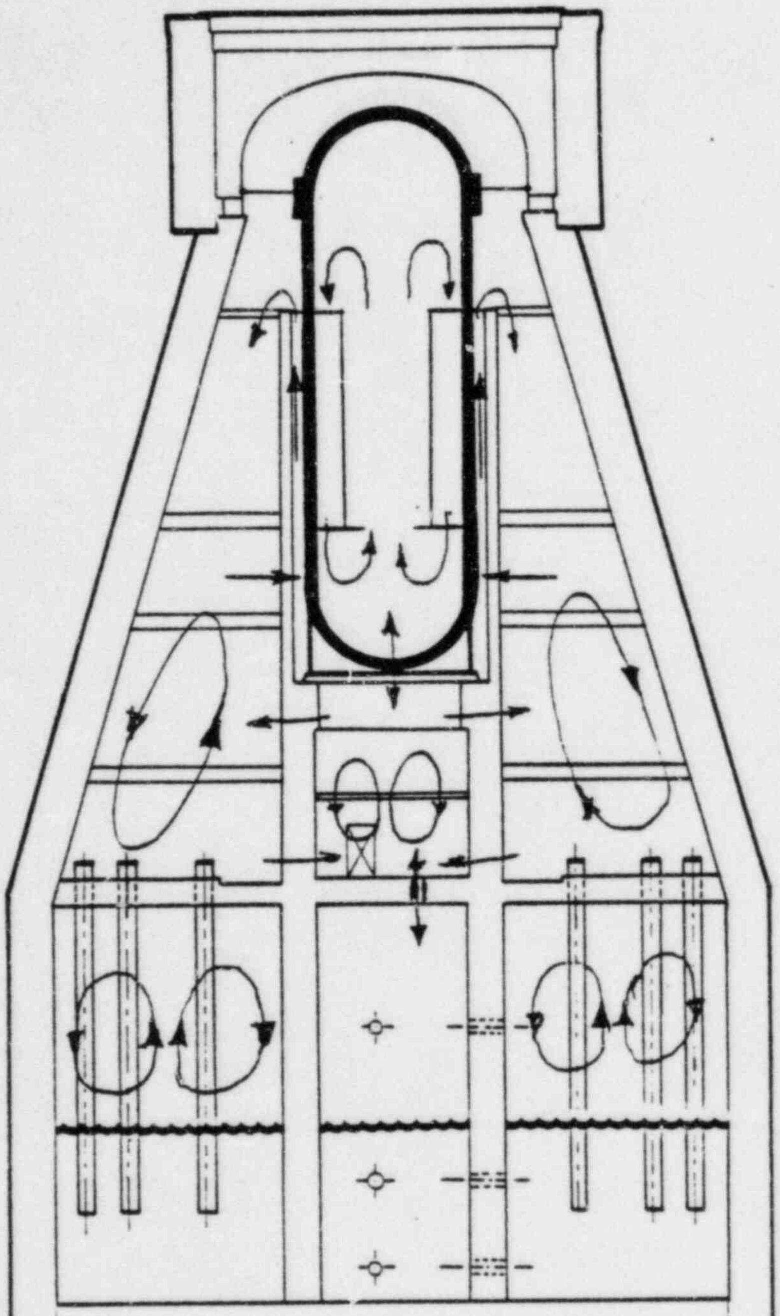
4" EQUIPMENT DRAIN CONNECTION DETAIL ②

NO SCALE

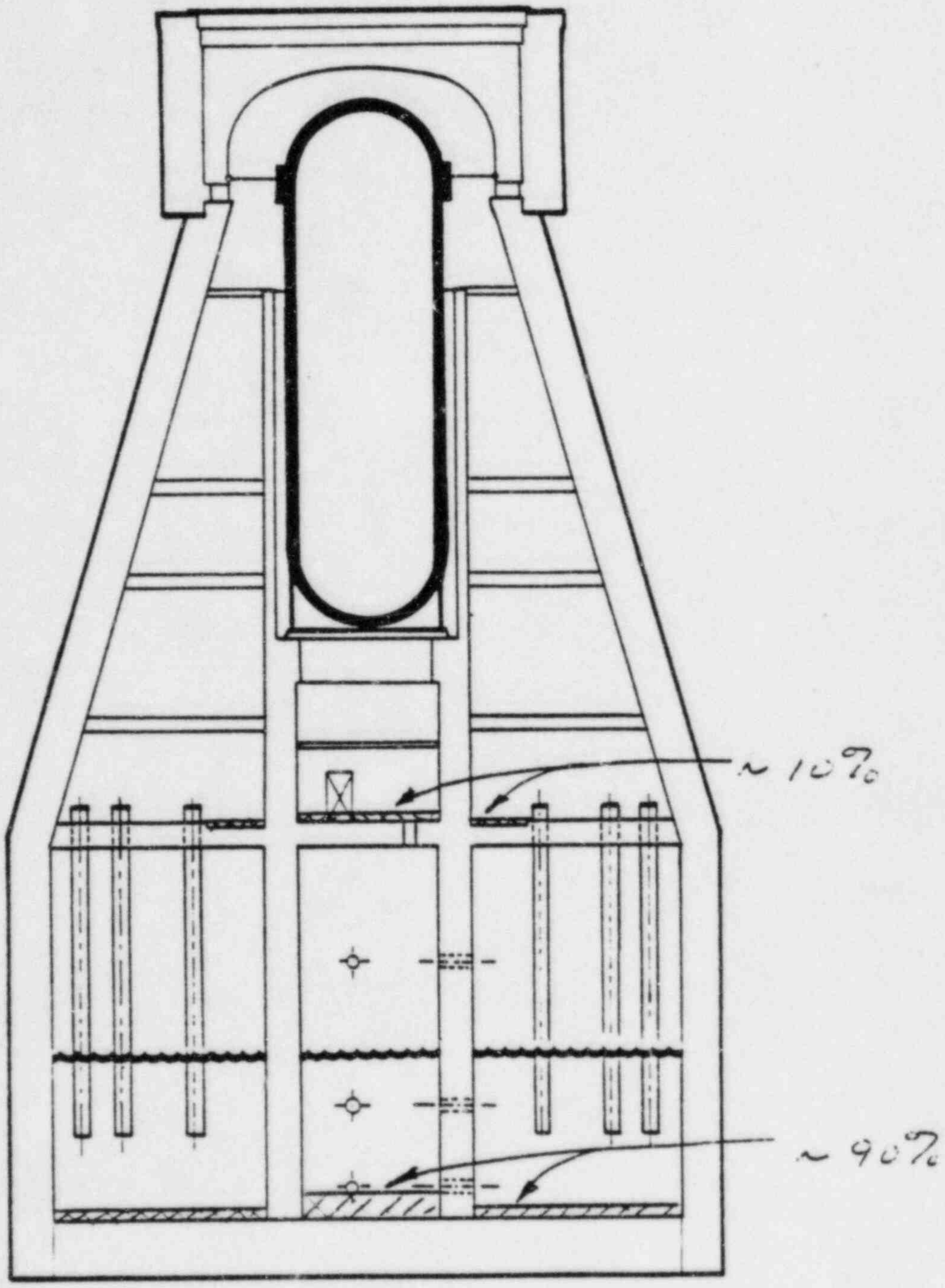
FIG. 5-2



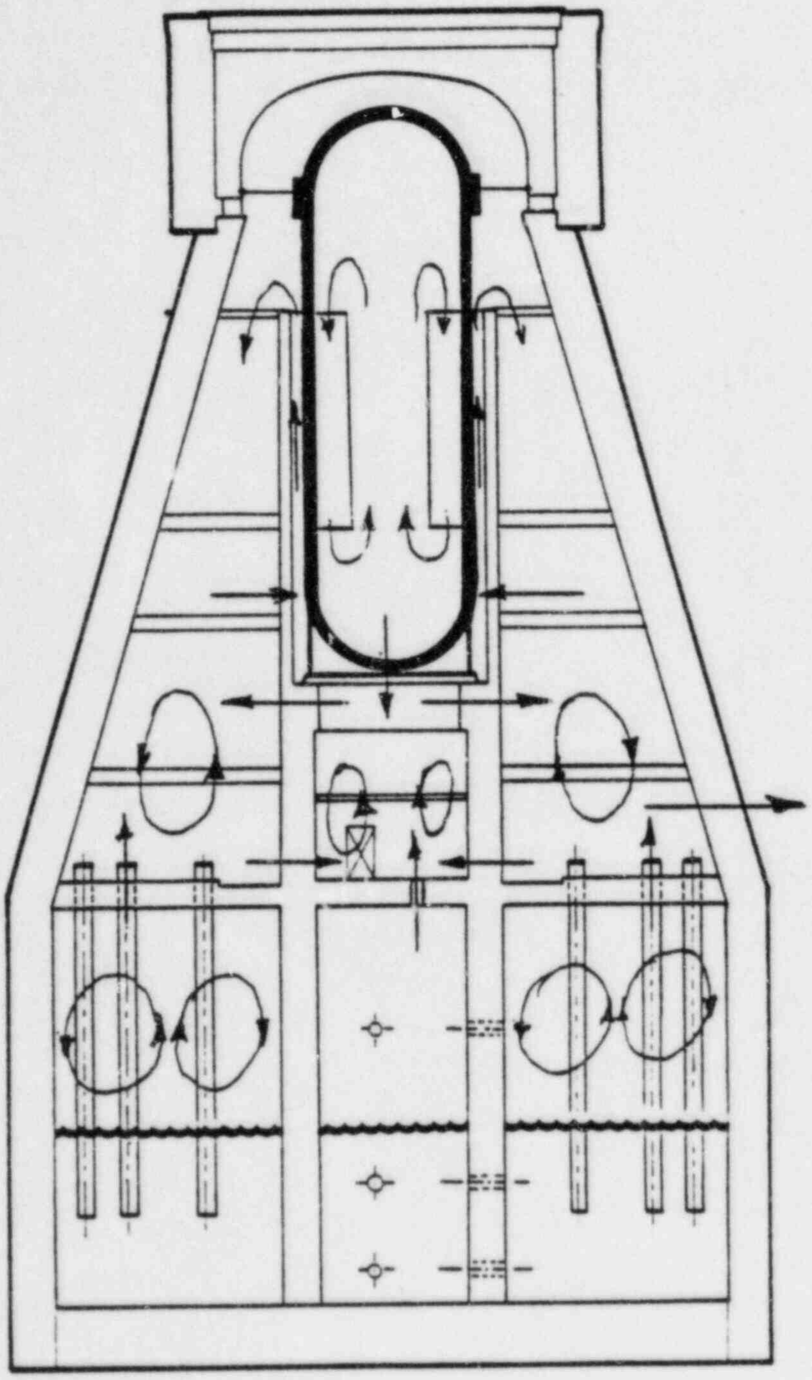
POSSIBLE PATHS FOR DEBRIS TO FLOW INTO THE SUPPRESSION POOL



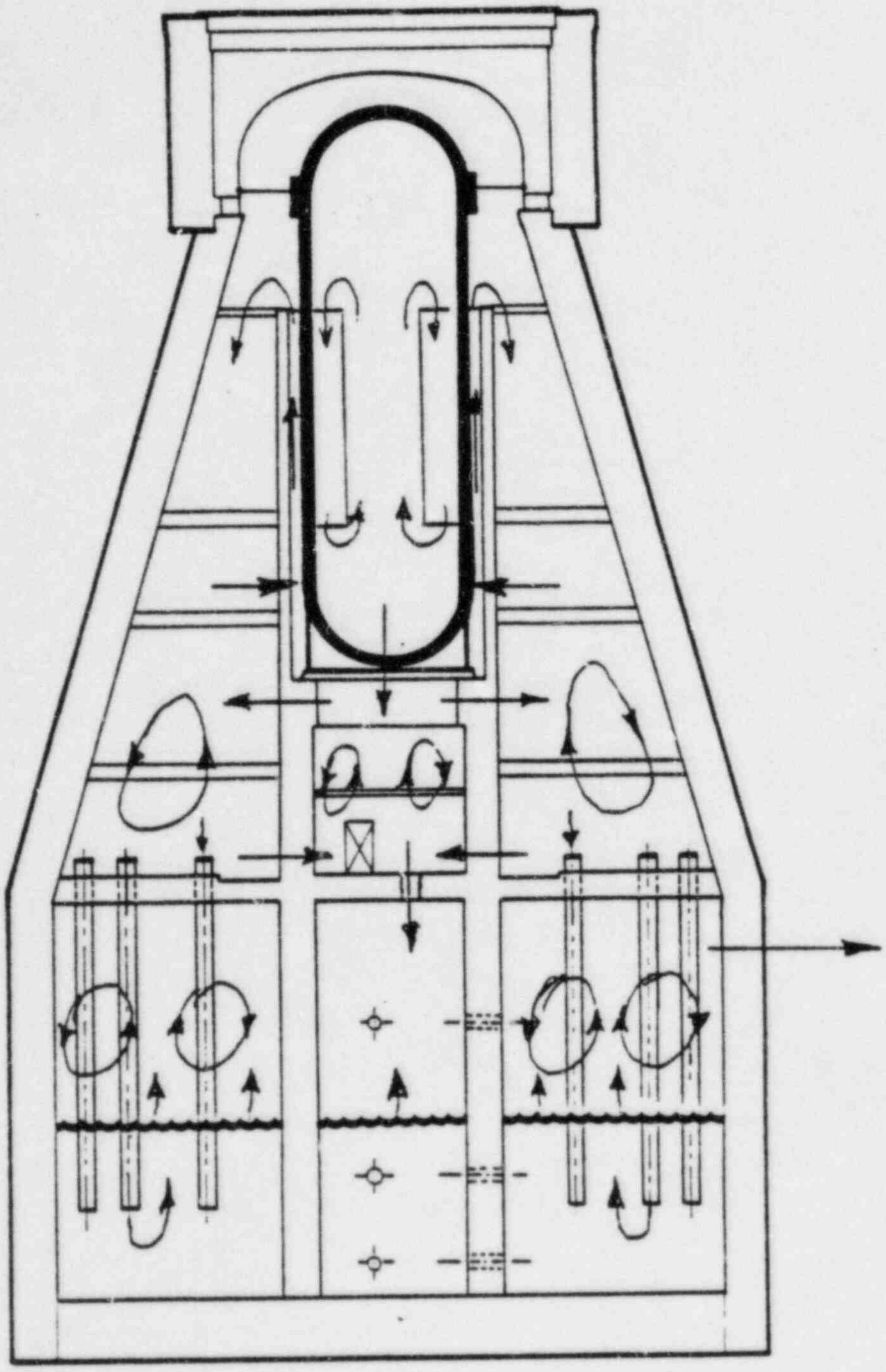
NATURAL CIRCULATION PATHS
AFTER VESSEL FAILURE



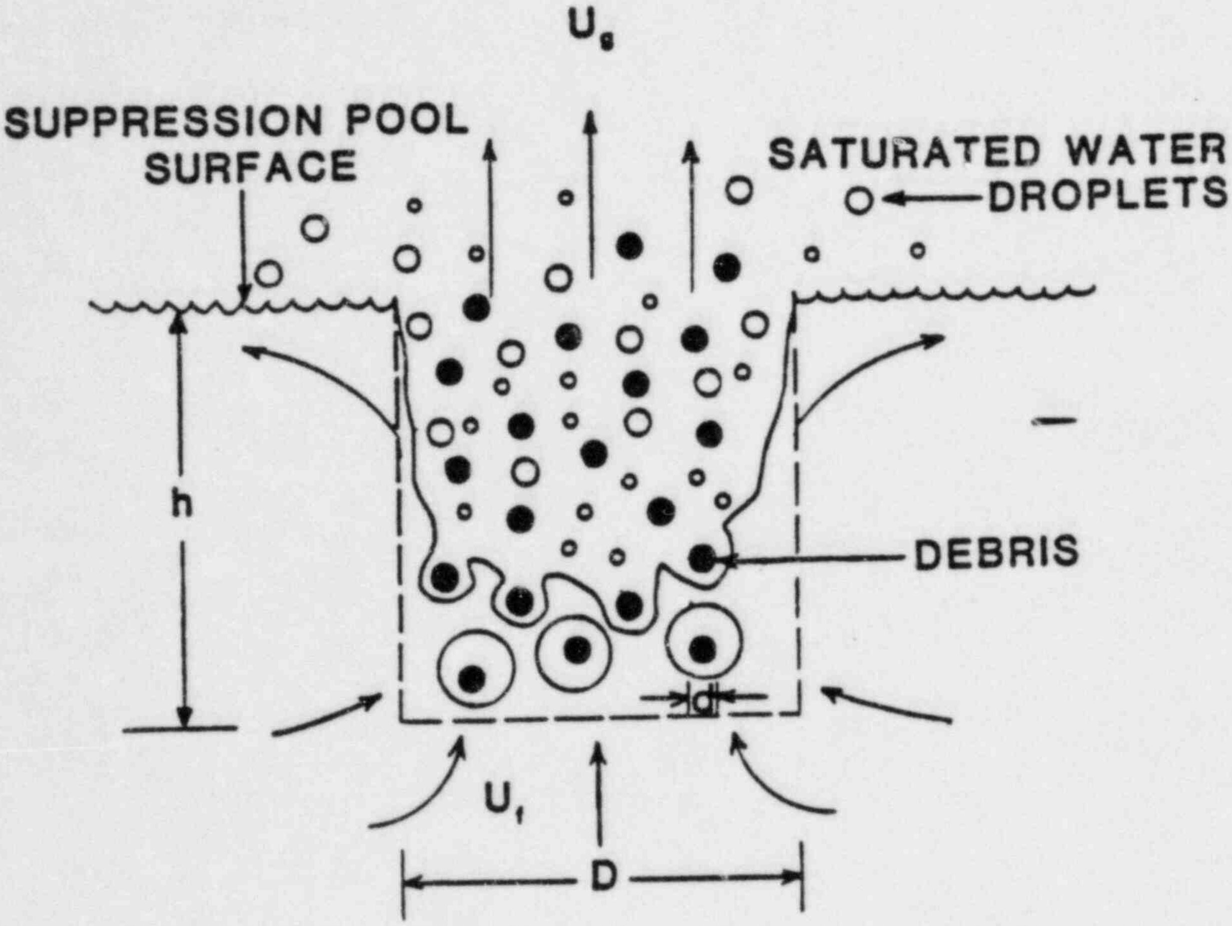
ULTIMATE CORE DEBRIS DISTRIBUTION
(~10 HRS. AFTER VESSEL FAILURE)



GAS FLOWS AFTER A DRYWELL FAILURE



GAS FLOWS AFTER A
WETWELL FAILURE



WATER CIRCULATION
IN THE
INTERACTIVE ZONE DURING QUENCHING

QUENCHING MODEL

$$\dot{Q}_g = \dot{Q}_c - \dot{Q}_\ell \qquad \frac{\dot{Q}_g}{\dot{Q}_c} = 1 - \frac{\dot{Q}_\ell}{\dot{Q}_c}$$

$$\frac{\dot{Q}_g}{\dot{Q}_c} = 1 - \frac{\dot{m}_\ell c_\ell \left[T_{sat} - T_B\right]}{\dot{m}_c c_c \left[T_c - T_{sat}\right]}$$

$$\left[\rho_f - (1 - \overline{\alpha})\, \rho_f\right] gh = \frac{m_c g}{A} = \overline{\alpha}\, \rho_f gh$$

$$\frac{m_c}{A} = \overline{\alpha}\, \rho_f h \qquad m_c = \dot{m}_c \tau$$

$$\tau = \frac{x^2}{\alpha_p} = \frac{1}{\alpha_p}\left(\frac{r_p}{3}\right)^2 = \frac{1}{\alpha_p}\left(\frac{d_p}{6}\right)^2 = \frac{d_p^2}{36\,\alpha_p}$$

$$U = \sqrt{\frac{4}{3}\left(\frac{\rho_p - \rho_f}{\rho_f}\right) g d_p} \qquad h = U \cdot \tau/2$$

$$\frac{m_c}{A} = \overline{\alpha}\, \rho_f U \cdot \tau/2 \qquad A = \frac{2\, m_c}{\overline{\alpha}\, \rho_f U \tau} = \frac{2\, \dot{m}_c}{\overline{\alpha}\, \rho_f U}$$

$$\dot{m}_\ell = \rho_f A U = \frac{2\, \dot{m}_c}{\overline{\alpha}}$$

$$\frac{\dot{Q}_g}{\dot{Q}_c} = 1 - \left(\frac{2}{\overline{\alpha}}\right) \frac{c_\ell\left[T_{sat} - T_B\right]}{c_c \left[T_c - T_{sat}\right]}$$

QUENCHING MODEL

Comparisons with Data

($\overline{\alpha} = 0.5$)

1. Stainless Steel-Water

   $m_c \sim 2$ kg $T = 80°C$

   $\frac{\dot{Q}_g}{\dot{Q}_c} = 0.5$ (Conservative Estimate)

2. UO_2-Water $m_c \sim 2$ kg $T = 20°C$

   $\frac{\dot{Q}_g}{\dot{Q}_c} = 0.1$

3. Copper-Water $\dot{m}_c \sim 2.7$ kg/sec

   $T = 40°C$

   $\frac{\dot{Q}_g}{\dot{Q}_c} = 0$

CONCLUSIONS AND INSIGHTS

G.F. DAEBELER

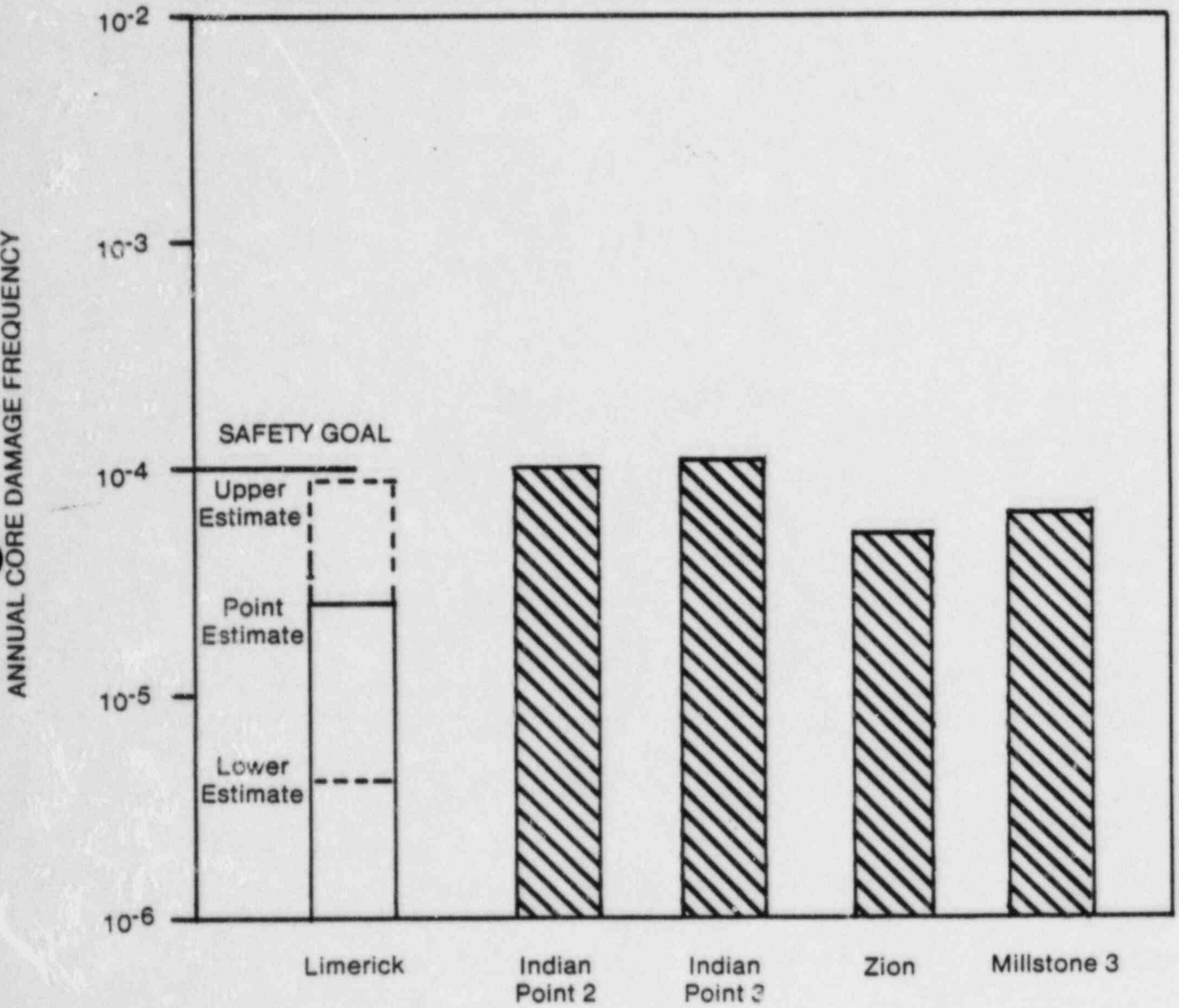
CONCLUSIONS AND INSIGHTS

- OVERALL RESULTS
- PLANT SPECIFIC CONCLUSIONS
- PROGRAMMATIC INSIGHTS

CORE DAMAGE
RESULTS OF PRA/SARA
POINT ESTIMATES

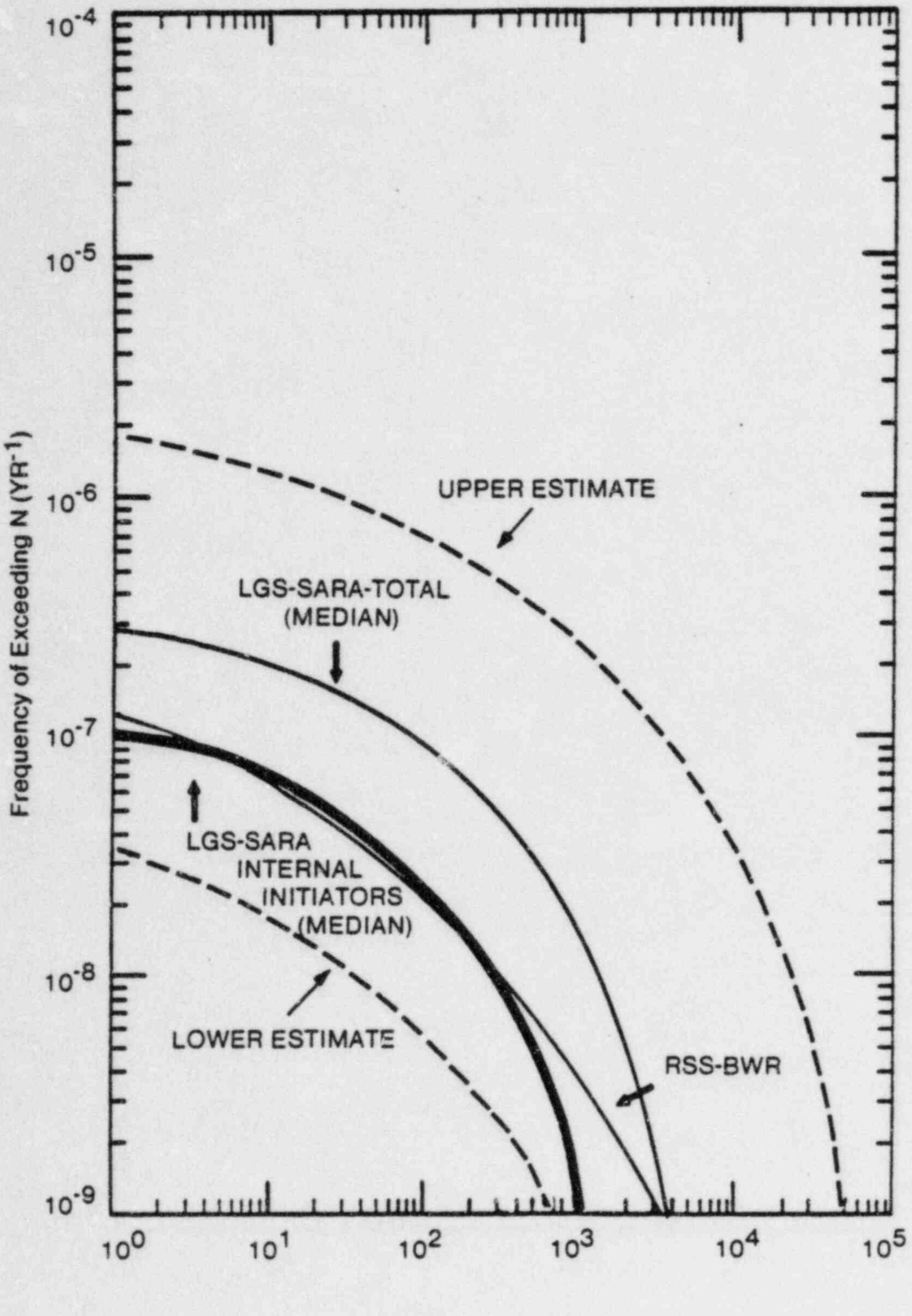
	FREQUENCY OF CORE DAMAGE (PER REACTOR - YEAR)	% OF TOTAL CDF
INTERNAL EVENTS	1.5×10^{-5}	62
EARTHQUAKES	5.7×10^{-6}	24
FIRES	3.4×10^{-6}	14
OTHERS	NEGLIGIBLE	—
TOTAL	2.4×10^{-5}	

COMPARISON OF LIMERICK CDF WITH POINT ESTIMATES OF OTHER PLANTS



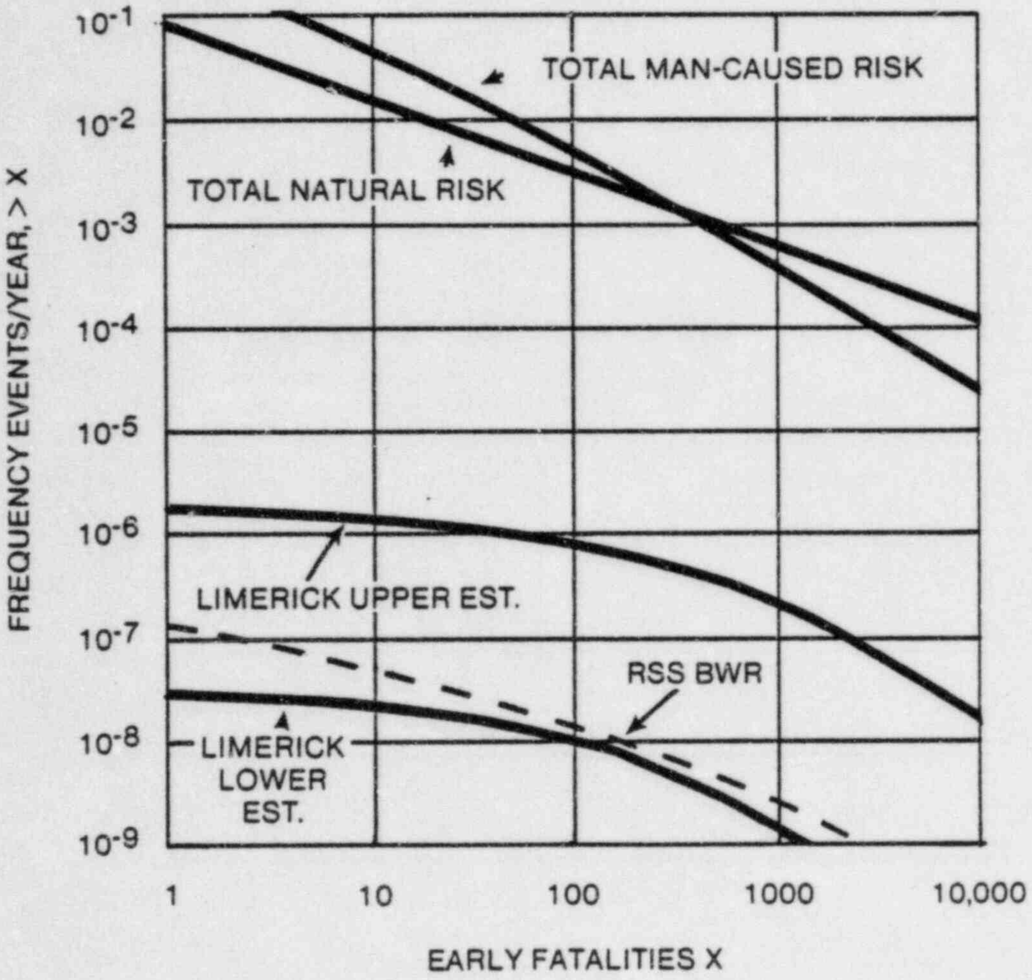
ESTIMATED CORE DAMAGE FREQUENCY AT LIMERICK

- BELOW SAFETY GOAL
- SIMILAR TO OTHER PRA's



CCDFs of acute fatalities-comparison with the Reactor Safety Study.

EARLY FATALITY RISK



ANNUAL INDIVIDUAL RISK

	EARLY FATALITY	LATENT CANCER FATALITY
U.S. Avg.	5×10^{-4} (1)	2×10^{-3}
Safety Goal	5×10^{-7} (2)	2×10^{-6} (3)
LGS Upper	7×10^{-8} (2)	1×10^{-8} (3)
LGS Lower	1×10^{-10} (2)	2×10^{-10} (3)

(1) Accidental Causes

(2) Avg. Within 1 Mile

(3) Avg. Within 50 Miles

RISK DUE TO OPERATION OF LIMERICK

- MUCH LESS THAN OTHER RISKS
- LESS THAN PROPOSED SAFETY GOAL
- COMPARABLE TO REACTOR SAFETY STUDY
- LIMERICK DOES NOT REPRESENT A DISPROPORTIONATE RISK TO THE PUBLIC

SUMMARY

PURPOSE

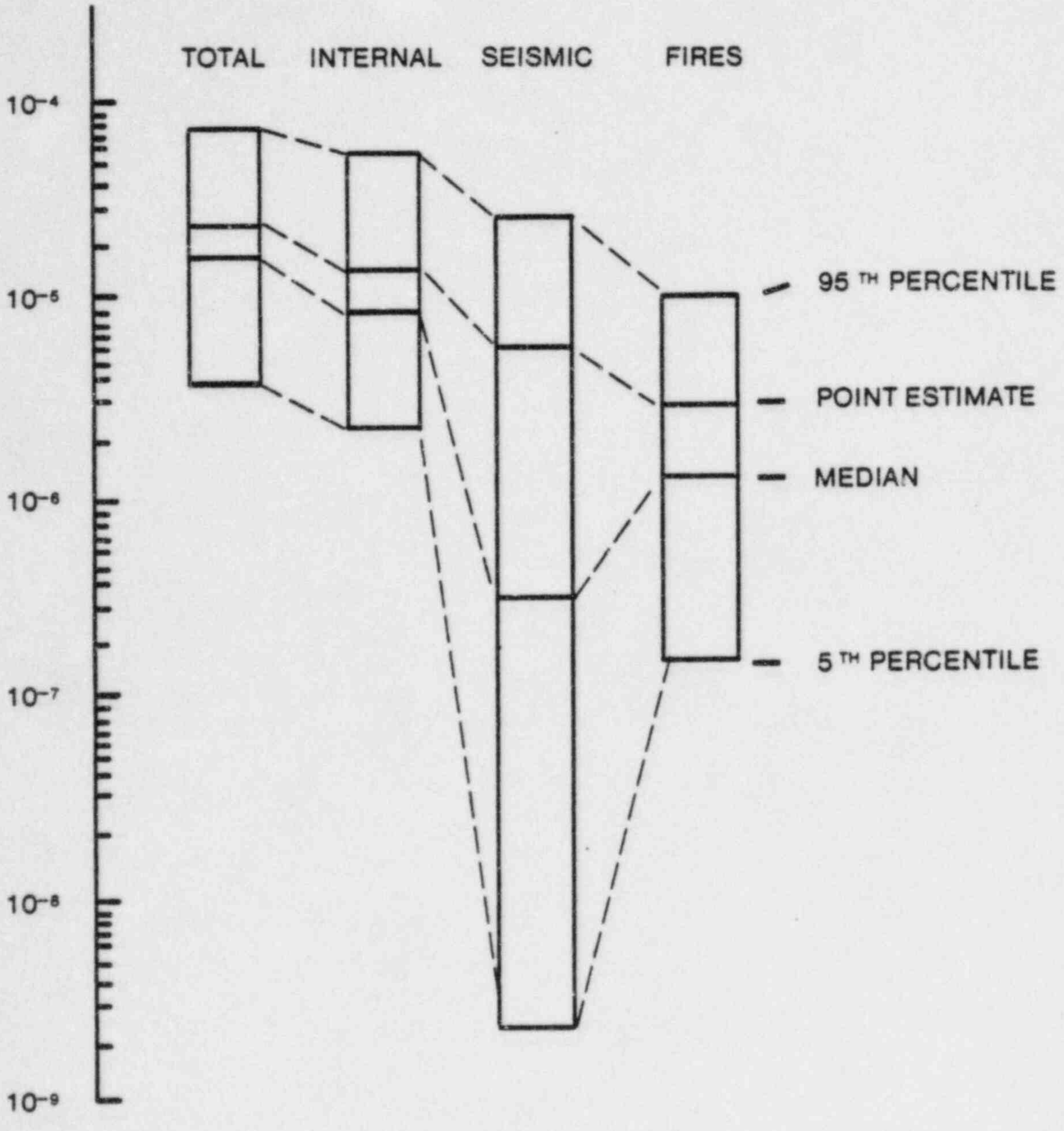
- DEMONSTRATE THE POTENTIAL RISK CONTRIBUTION TO THE PUBLIC DUE TO LIMERICK OPERATION
- RESPOND TO NRC REQUEST

RESULTS

- RISK LESS THAN PROPOSED SAFETY GOAL AND COMPARABLE TO REACTOR SAFETY STUDY
- PRA/SARA RESULTS VERIFY THE ADEQUACY OF THE DESIGN OF THE LIMERICK PLANT

PLANT SPECIFIC CONCLUSIONS

ANNUAL CORE MELT FREQUENCY



ANNUAL
CORE DAMAGE FREQUENCY

	LOWER ESTIMATE	MEDIAN	UPPER ESTIMATE	POINT ESTIMATE
INTERNAL	2.4×10^{-6}	9.2×10^{-6}	6.0×10^{-5}	1.5×10^{-5}
EXTERNAL				
SEISMIC	2.2×10^{-9}	3.3×10^{-7}	2.7×10^{-5}	5.7×10^{-6}
FIRES	1.7×10^{-7}	1.4×10^{-6}	1.2×10^{-5}	3.4×10^{-6}
OTHER		— NEGLIGIBLE —		
TOTAL	4.0×10^{-6}	1.8×10^{-5}	7.8×10^{-5}	2.4×10^{-5}

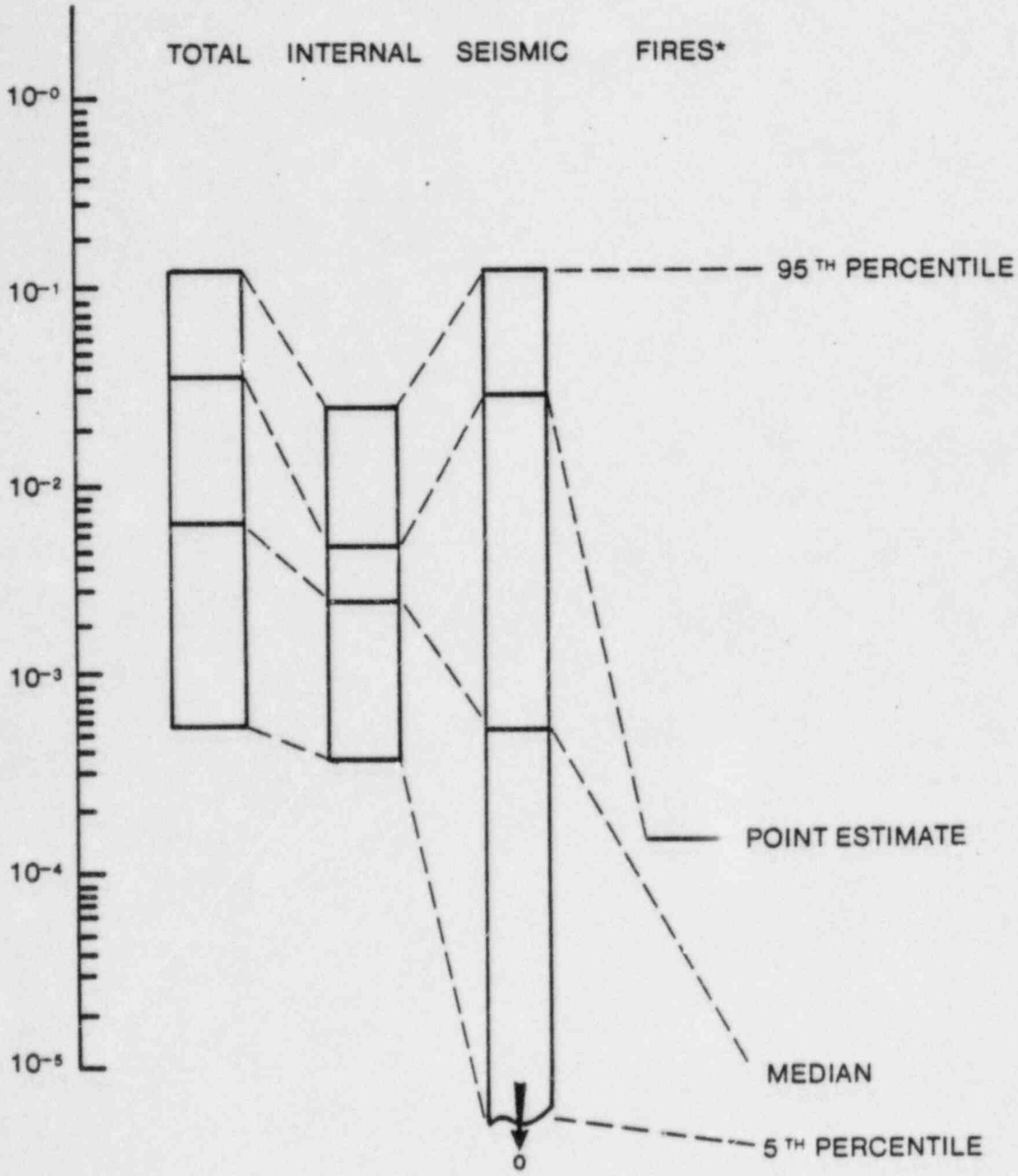
DOMINANT CORE DAMAGE SEQUENCES

DESCRIPTION	DESIGNATION	POINT ESTIMATE	PERCENT OF TOTAL
LOSS OF OFFSITE POWER COMMON CAUSE FAILURE OF ALL DIESELS FAILURE OF HPCI AND RCIC	T_EUV	5.9×10^{-6}	25
LOSS OF FEEDWATER FAILURE TO RESTORE FEEDWATER FAILURE OF HPCI AND RCIC FAILURE OF TIMELY DEPRESSURIZATION	T_FQUX	3.6×10^{-6}	15
SEISMIC LOSS OF OFFSITE POWER SEISMIC FAILURE OF AC/DC BUSES AND SWITCHGEAR	T_SE_SUX	3.2×10^{-6}	13

CORE DAMAGE FREQUENCY (CDF)

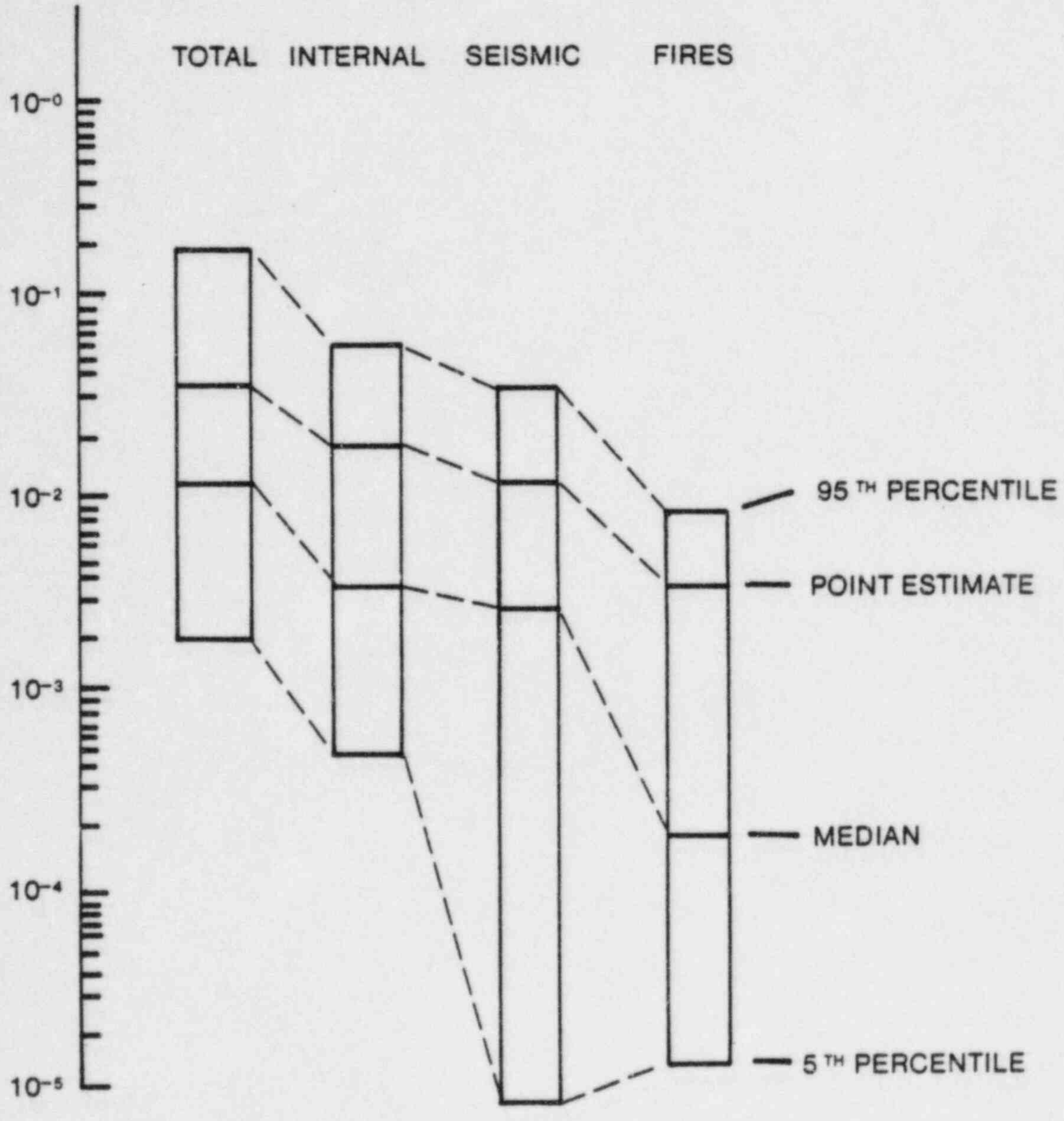
- DOMINATED BY INTERNAL INITIATED EVENTS
- EARTHQUAKES AND FIRES ARE LESSER CONTRIBUTORS
- NO SINGLE SEQUENCE SO DOMINATES CDF THAT A REDUCTION IN ITS FREQUENCY WOULD CAUSE A SUBSTANTIAL REDUCTION IN CDF
- NO SINGLE SYSTEM SO IMPORTANT THAT A REDUCTION IN ITS LIKELIHOOD OF FAILURE WOULD CAUSE A SUBSTANTIAL REDUCTION IN CDF.

EARLY FATALITY RISK



* FIRES DO NOT CONTRIBUTE
TO EARLY FATALITIES

LATENT FATALITY RISK



EARLY RISK

- SEISMIC INITIATED ACCIDENTS ARE A MAJOR CONTRIBUTION FOR THE HYPOTHESIS THAT A LARGE MAGNITUDE EARTHQUAKE OCCURS IN PLANT REGION.
- UPPER ESTIMATE LARGER THAN FOR INTERNAL INITIATED EVENTS
    - LOW ESTIMATE NEGLIGIBLE CONTRIBUTOR
- EXCEPT FOR SEISMIC CONSIDERATIONS, INTERNAL INITIATED EVENTS CAUSE THE MAJOR CONTRIBUTION

LATENT RISK

- INTERNAL INTIATED EVENTS ARE MAJOR CONTRIBUTOR
- SEISMIC ALSO CONTRIBUTES
    - UPPER ESTIMATE ABOUT EQUIVALENT TO INTERNAL
    - LOWER ESTIMATE LESSER CONTRIBUTOR
- FIRE IS A LESSER CONTRIBUTOR

EARLY RISK

- INTERNAL
  - DUE PRIMARILY TO ATWS SEQUENCES
  - LESSER CONTRIBUTION FROM VESSEL FAILURE
  - NO SINGLE SEQUENCE DOMINATES RISK CONTRIBUTION
- SEISMIC
  - DUE PRIMARLY TO VESSEL SUPPORT FAILURE AT HIGH ACCELERATIONS (> 1g)

LATENT RISK

- INTERNAL
  - SAME SEQUENCES AS THOSE AFFECTING CORE DAMAGE FREQUENCY
  - NO SINGLE SEQUENCE DOMINATES
- SEISMIC
  - DISTRIBUTED BETWEEN
    - LOOP AND FAILURE OF ONSITE POWER
    - REACTOR BUILDING FAILURE
    - VESSEL SUPPORT
  - NO SINGLE SEQUENCE DOMINATES

FUNCTIONS IMPORTANT TO CORE DAMAGE AND RISK

INTERNAL INITIATORS

- RECOVERY OF PCS
- DEPRESSURIZATION
- HPCI AND RCIC
- AVAILABILITY OF AC POWER
  - RECOVERY OF OFFSITE POWER
  - DIESEL RELIABILITY
  - BATTERY LIFE
  - HPCI/RCIC ROOM COOLING
- ATWS PREVENTION AND MITIGATION

SEISMIC INITIATORS

- AVAILABILITY OF AC POWER
- RPV SUPPORTS
- RESETTING OF CONTROL CIRCUITRY

FIRE INITIATORS

- TRAINING IN PREVENTION AND MITIGATION OF FIRES

LGS DESIGN FEATURES INFLUENCED BY PRA/SARA

- INSTALLATION OF ALL RHRSW AND ESW PUMPS BY UNIT 1 OPERATION
- STANDBY LIQUID CONTROL SYSTEM
  - ADDITION OF 3rd PUMP
  - ARRANGEMENT OF EQUIPMENT FOR ENHANCED TESTABILITY
  - USE OF REDUNDANT PENETRATIONS FOR INJECTION
  - INJECTION THROUGH CORE SPRAY SPARGER
- ADS AIR SUPPLY:
  - TYPE AND LOCATION OF BACKUP SUPPLIES
  - PHYSICAL ARRANGEMENT OF PIPING & VALVES
  - DESIGN OF SAFETY/NON-SAFETY INTERFACES
  - USE OF DUAL PILOT SOLENOID VALVES
- MSIV AIR SUPPLY IMPROVEMENTS
- FIRE PROPAGATION BARRIERS FOR REACTOR ENCLOSURE EQUIPMENT HATCHES

PRA/SARA CONFIRMS DESIRABILITY OF INCLUSION OF THE FOLLOWING FEATURES

- 4 DIESELS PER UNIT EACH WITH:
  - REDUNDANT AIR START SYSTEMS
  - REDUNDANT ESW SUPPLIES
- 4 SEPARATE ELECTRICAL DIVISIONS
- NUMBER AND ARRANGEMENT OF OFFSITE POWER SOURCES
- ASSIGNMENT OF REDUNDANT COOLING LOADS TO SEPARATE ESW LOOPS
- RHR PUMP DISCHARGE CROSS-TIES
- DESIGN OF ESW/SW INTERFACES
- AUXILIARY STEAM SUPPLIES TO SJAE's
- FLEXIBILITY IN USE OF SPRAY POND AND COOLING TOWERS
- REDUNDANT, SERIES SUPPRESSION POOL/DRYWELL VACUUM BREAKERS
- ESTABLISHMENT OF APPROPRIATE FIRE ZONES

PROCEDURES INFLUENCED

- HPCI/RCIC ROOM COOLING
- CONTAINMENT SPRAY
- VENTING
- REESTABLISH PCS
- RESETTING OF CONTROL CIRCUITRY

PROGRAMMATIC INSIGHTS

- THE PRA PROCESS ENHANCES UNDERSTANDING OF PLANT DESIGN AND OPERATION.
- DUE TO UNCERTAINTIES IN MODELING AND DATA PRA IS BEST USED TO COMPARE ALTERNATIVES.
- RECOGNIZING INHERENT UNCERTAINTIES IS CRITICAL IN EVALUATING POTENTIAL PLANT CHANGES. POTENTIAL FIXES MAY HAVE SIGNIFICANTLY MORE OR LESS BENEFIT THAN POINT ESTIMATES WOULD INDICATE.
- IN EVALUATING ALTERNATES, ESTIMATES OF CORE DAMAGE FREQUENCY RESULTING FROM INTERNAL INITIATORS CAN BE IMPORTANT INPUT.

FUTURE USE
OF
PRA

A. R. DIEDERICH

CONTINUING USE OF PRA

STUDY GOALS

- INTEGRATION WITH ORGANIZATION
- ESTABLISH TECHNICAL BASES
- PLAN IMPLEMENTATION

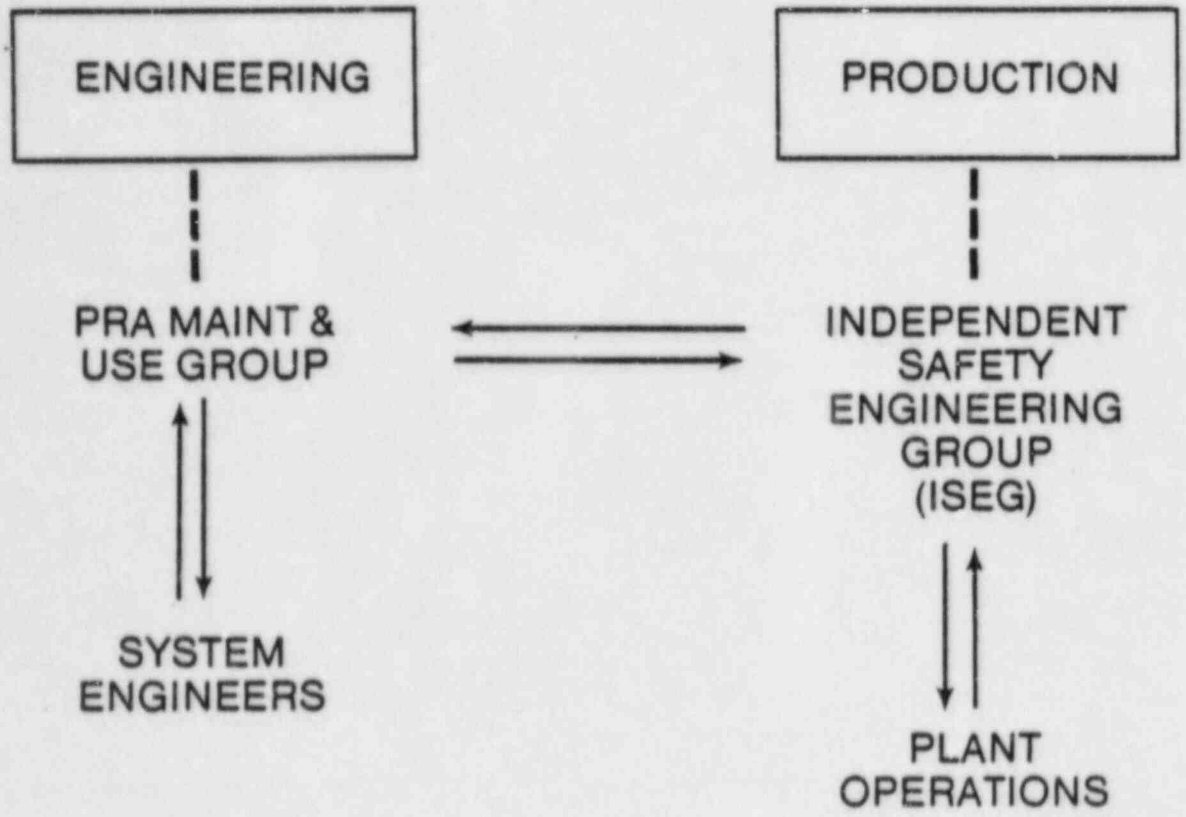
PRA MAINTENANCE & USE GROUP

- DOCUMENT ORIGINAL BASES
- UPDATE PRA
  - BASELINE
  - PERIODIC
- EVALUATE MODIFICATIONS
- EVALUATE TECH SPECS
- MAINTAIN/USE CODES
- DATA ANALYSIS
- PRA TRAINING
- STUDIES/ANALYSES

ISEG

- EVALUATE OPERATING EXPERIENCE
  - LIMERICK
  - OTHERS
- IDENTIFY/REQUEST PRA STUDY
- ASSURE PRA RESULTS REFLECTED IN
  - PROCEDURES
  - MAINTENANCE
  - TRAINING

ORGANIZATION



ENGINEERING

PRODUCTION

PRA MAINT & USE GROUP

INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

SYSTEM ENGINEERS

PLANT OPERATIONS

TECHNICAL BASES

- SCOPE
- MEASURE
- DETAIL

PRA SCOPE

INCLUDED:

- INTERNAL INITIATORS

NOT INCLUDED

- EXTERNAL INITIATORS
- ACCIDENT EFFECTS

PERIODIC EVALUATION
OF MAJOR STUDY UPDATE

MEASURE

CHOICE:

- CORE DAMAGE FREQUENCY

OTHERS CONSIDERED (CONSEQUENCES):

- POPULATION
- INDIVIDUAL
- PLANT RELEASE

DETAIL

- PRESENT PRA LEVEL
- EXPAND DETAIL AS NEEDED BY APPLICATION

IMPLEMENTATION

TRAINING INITIAL ORGANIZING/STAFFING
6 MOS.

BASELINE/DOCUMENT
18 MOS.

RESULT

- PRA INTEGRATED WITH ORGANIZATION
- RESULTS REFLECTED IN
  - MODIFICATIONS
  - OPERATIONS
  - MAINTENANCE
  - TRAINING
- PRA MAINTAINED UP-TO-DATE
- PERIODIC RE-EVALUATION OF BENEFITS

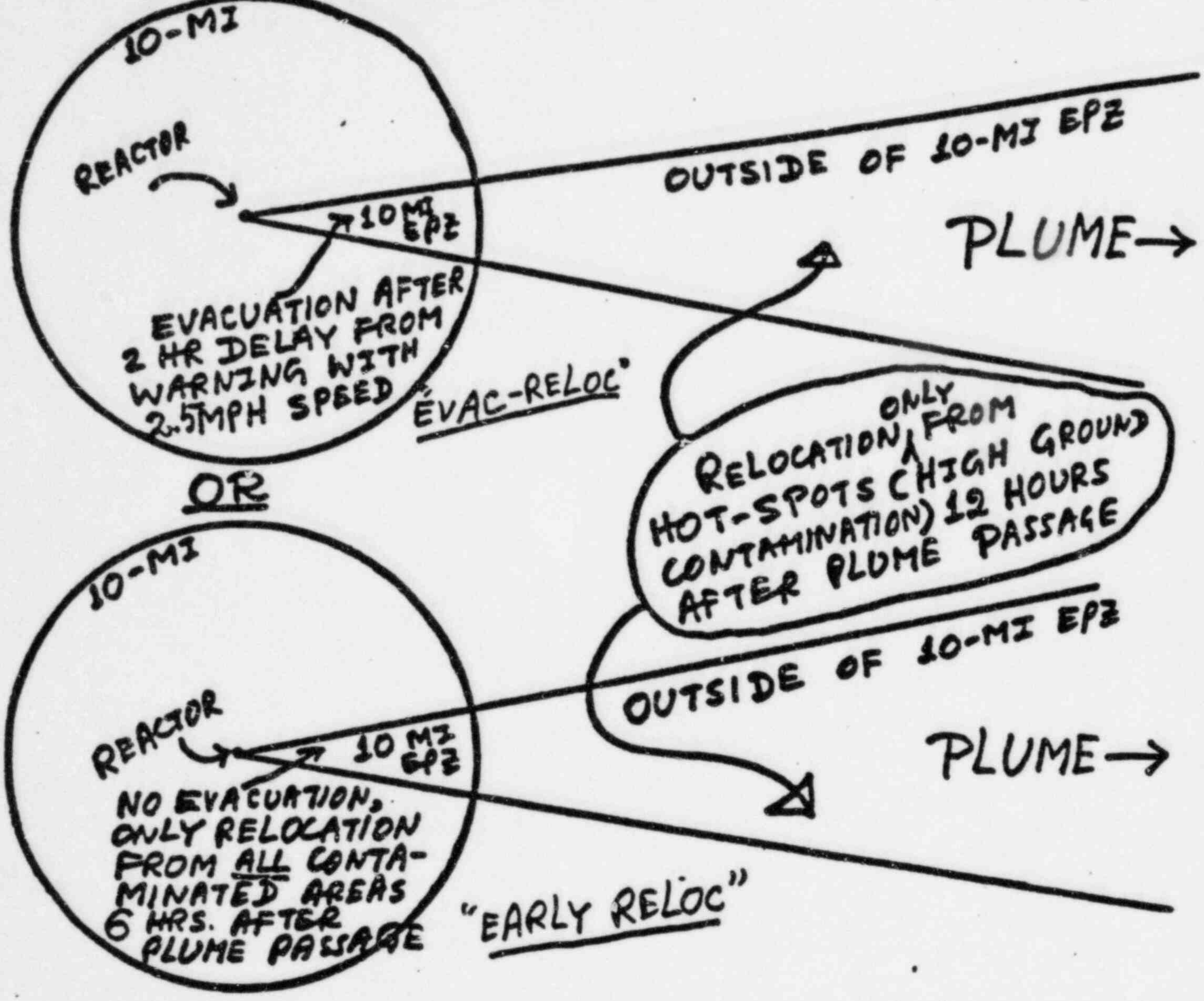
3-2

④

LIMERICK FES
CONSEQUENCE ANALYSIS

EMERGENCY RESPONSE MODES

DURING CONDITIONS OTHER THAN SEVERE EARTHQUAKES



DURING A SEVERE EARTHQUAKE

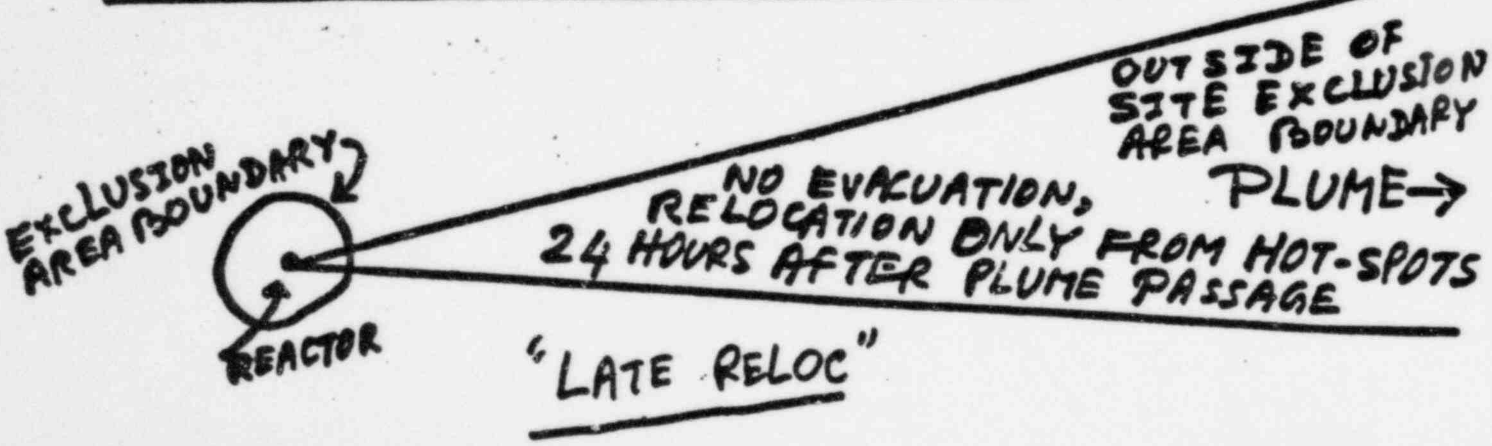


Table 5.11f Emergency response assumptions for each reactor unit

Emergency response set no.*	Evacuation distance (mi)**	Delay time (hr)	Effective evacuation speed (mph)	Effective downwind distance moved*** (mi)	Relocation zone size (mi) Zone		Zone B relocation time (hr)	Zone B relocation dose criterion (bone marrow dose projected for 7 days) (rems)	Shielding protection factor (fraction)	
					A†	B†			During evacuation, plume/ground	Other times, plume/ground
1	10	2	2.5	15	0	>10	12	200	1¶/0.5¶	0.75¶¶/0.33¶¶
2	N/A††	N/A	N/A	N/A	10†††	>10	12	200	N/A	0.75¶¶/0.33¶¶
3	N/A	N/A	N/A	N/A	0	>0	24	200	N/A	1.0¶¶¶/0.5¶¶¶

*Sets 1, 2, and 3 are also identified as Evac-Reloc, Early Reloc, and Late Reloc, respectively, in text, tables, and figures.

**To change miles to km, multiply the values shown by 1.609.

***An artificial parameter used only to represent a realistic path-length for each evacuee over which radiation exposure to the evacuee is calculated in the CRAC code.

†Zone A is the 10-mile plume exposure pathway emergency planning zone; Zone B is the area outside Zone A.

††N/A - Not Applicable.

†††Relocation takes place 6 hours after ground contamination.

¶During evacuation, automobiles are assumed to provide essentially no shielding to gamma rays from the plume and some shielding to gamma rays from the contaminated ground. The selected values of shielding protection factors for the plume and the ground during evacuation are taken from Table VI 11-13 of Appendix VI of WASH-1400.

¶¶At other times than during evaucation, shielding protection factors are the average values representative of normal activities of the people during which some people are indoors and some are outdoors. The selected values of the shielding protection factors for the plume and the ground for this situation are taken from Table VI 11-13 of Appendix VI of WASH-1400.

¶¶¶During an abnormal situation in the site region caused by a external event such as a severe earthquake, it is assumed that many of the buildings may not remain habitable to provide shielding protection to the people against gamma rays from the plume. So, the shielding factor for the plume is taken to be 1. However, the nature of the ground surface is assumed to become altered by debris and possibly mud/slush/water generated from a severe earthquake. So, the ground shielding factor (provided by the altered ground and whatever building structures that would still have remained intact) of 0.5 was selected for this scenario, which is about midway between the values 0.33 for normal situation and 0.7 for an ordinary and uncovered ground surface.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties.

Table 5.11c Summary of the atmospheric release specifications used in consequence analysis for Limerick Units 1 and 2[a]

Release category[b]	Release time (hr)	Release duration (hr)	Warning time for evacuation (hr)	Energy release (10⁶ Btu/hr)	Release height (m)	Fractions of Core Inventory Released							
						Xe-Kr	Organic I[c]	Inorganic I	Cs-Rb	Te-Sb	Ba-Sr	Ru[d]	La[e]
I-T/DW(22)*	5	0.5	4	100	30	1	7(-3)**	2(-3)	2(-2)	8(-2)	1(-3)	5(-3)	1(-3)
I-T/WW(25)	5	0.5	4	100	30	1	7(-3)	1(-4)	3(-4)	1(-3)	2(-5)	7(-5)	1(-5)
I-T/$\overline{WW}$(24)	5	0.5	4	100	30	1	7(-3)	2(-4)	9(-4)	2(-3)	8(-5)	1(-4)	3(-5)
I-T/SE(14)	2	0.5	1	100	30	1	--	1(-1)	1(-1)	4(-1)	1(-2)	4(-1)	2(-3)
I-T/HB(20)	2	0.5	1	100	30	1	--	2(-1)	6(-2)	1(-1)	7(-3)	8(-2)	1(-5)
I-T/LGT(26)***	2	3	0	1	30	0.7	--	3(-3)	1(-4)	5(-4)	2(-5)	3(-5)	6(-6)
I-T/$\overline{LGT}$(18)	2	3	0	1	30	0.7	--	2(-2)	1(-1)	5(-2)	2(-3)	3(-3)	6(-4)
II-T/WW(8)	20	4	5	1	30	1	7(-3)	7(-1)	3(-1)	2(-1)	4(-2)	4(-2)	3(-3)
II-T/SE(14)	30	0.5	7	100	30	1	--	1(-1)	1(-1)	4(-1)	1(-2)	4(-1)	2(-3)
III-T/WW(10)	3	1	2	100	30	1	7(-3)	8(-2)	2(-1)	6(-1)	2(-2)	4(-2)	7(-3)
III-T/SE(5)	2	0.5	1	100	30	1	--	4(-1)	5(-1)	5(-1)	5(-2)	5(-1)	3(-3)
III-T/HB(20)	2	0.5	1	100	30	1	--	2(-1)	6(-2)	1(-1)	7(-3)	8(-2)	1(-5)
III-T/LGT(26)	0.5	4	0	1	30	0.7	--	3(-3)	1(-4)	5(-4)	2(-5)	3(-5)	6(-6)
III-T/$\overline{LGT}$(18)	0.5	4	0	1	30	0.7	--	2(-2)	1(-1)	5(-2)	2(-3)	3(-3)	6(-4)
IV-T/DW(2)	1	3	0.5	1	30	1	7(-3)	5(-1)	5(-1)	5(-1)	6(-2)	9(-2)	7(-3)
IV-T/WW(4)	1	3	0.5	1	30	1	7(-3)	5(-1)	5(-1)	5(-1)	6(-2)	8(-2)	6(-3)
IV-T/$\overline{WW}$(3)	1	3	0.5	1	30	1	7(-3)	5(-1)	5(-1)	5(-1)	6(-2)	9(-2)	7(-3)
IV-T/SE(5)	2	0.5	2	100	30	1	--	4(-1)	4(-1)	5(-1)	5(-2)	5(-1)	3(-3)
I-S/DW(23)	5	0.5	4	100	30	1	7(-3)	3(-3)	5(-3)	3(-3)	6(-4)	3(-4)	4(-4)
IV-A/DW(1)	1	3	0.5	1	30	1	7(-3)	5(-1)	5(-1)	5(-1)	6(-2)	9(-2)	7(-3)
IS-C/DW(13)	0	3	0.4	1	30	1	7(-3)	8(-2)	1(-1)	6(-1)	7(-3)	8(-2)	7(-3)
IS-C/SE(14)	1	0.5	1	100	30	1	--	1(-1)	1(-1)	4(-1)	1(-2)	4(-1)	2(-3)
IS-$\overline{C}$/DW(12)	1	3	1	1	30	1	7(-3)	8(-2)	1(-1)	6(-1)	8(-3)	1(-1)	7(-3)
IS-$\overline{C}$/SE(14)	2	0.5	2	100	30	1	--	1(-1)	1(-1)	4(-1)	1(-2)	4(-1)	2(-3)
S-H2O/$\overline{WW}$(11)	3	5	3	1	30	1	7(-3)	1(-1)	2(-1)	3(-1)	1(-2)	5(-2)	4(-3)
S-H2O/SE(5)	4	0.5	4	100	30	1.	--	4(-1)	4(-1)	5(-1)	5(-2)	5(-1)	3(-3)
S-$\overline{H2O}$/$\overline{WW}$(9)	3	4	3	1	30	1	7(-3)	3(-1)	3(-1)	4(-1)	3(-2)	6(-2)	5(-3)

[a] See Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

[b] See Appendix H for designations and descriptions of the release categories.

[c] Organic iodine is added to inorganic iodine for consequence calculations because organic iodine is likely to be converted to inorganic or particulate forms during environmental transport.

[d] Includes Ru, Rh, Co, Mo, Tc.

[e] Includes Y, La, Zr, Nb, Ce, Pr, Nd, NP, Pu, Am, Cm.

*Number in parentheses indicates relative ranking of the release category according to cesium fraction.

**7(-3) = 7 × 10^{-3} = 0.007.

***This release category is combined with III-T/LGT in consequence analysis.

Table 5.11d Summary of the calculated mean (point estimate) probabilities of atmospheric release categories

Release category	Probability of the release category initiated by internal causes, fires, and low to moderately severe earthquakes (per reactor-year)	Probability of the release category initiated by severe earthquakes (per reactor-year)
I-T/DW	2(-5)*	6(-7)
I-T/WW	2(-5)	5(-7)
I-T/$\overline{\text{WW}}$	2(-6)	6(-8)
I-T/SE	8(-9)	2(-10)***
I-T/HB	8(-7)	2(-8)
I-T/LGT**	2(-5)	5(-7)
I-T/$\overline{\text{LGT}}$	2(-5)	6(-7)
II-T/WW	2(-6)	2(-8)
II-T/SE	4(-10)***	4(-10)***
III-T/WW	2(-6)	4(-7)
III-T/SE	3(-10)***	7(-11)***
III-T/HB	3(-8)	7(-9)
III-T/LGT	7(-7)	2(-7)
III-T/$\overline{\text{LGT}}$	9(-7)	2(-7)
IV-T/DW	2(-7)	5(-8)
IV-T/WW	2(-7)	4(-8)
IV-T/$\overline{\text{WW}}$	2(-8)	5(-9)
IV-T/SE	3(-11)***	1(-11)***
I-S/DW	4(-8)	0
IV-A/DW	5(-9)	0
IS-C/DW	1(-8)	1(-7)
IS-C/SE	1(-12)***	1(-11)***
IS-$\overline{\text{C}}$/DW	1(-7)	9(-7)
IS-$\overline{\text{C}}$/SE	1(-11)***	9(-11)***
S-H20/$\overline{\text{WW}}$	1(-8)	4(-8)
S-H20/SE	1(-12)***	4(-12)***
S-$\overline{\text{H20}}$/$\overline{\text{WW}}$	1(-8)	4(-7)
Total probability per reactor-year	9(-5)	5(-6)

*2(-5) = 2×10^{-5} = .00002

**This release category is combined with III-T/LGT in consequence analysis.

***Any release category with probability less than 10^{-9} per reactor-year is omitted from consequence analysis because of its low probability and insignificant contribution to risks.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table K.1 Conditional mean values of societal consequences from individual release categories for three alternative offsite emergency response modes

Consequence Category	Offsite Emergency Response Mode	Release Categories									
		I-T/DW	I-T/WW	I-T/$\overline{WW}$	I-T/SE*	I-T/HB	I-T/$\overline{LGT}$	II-T/WW	III-T/WW	III-T/HB	III-T/LGT
1. Early fatalities with supportive medical treatment (persons)	Evac-Reloc	0	0	0	2(2)**	1(1)	5(-1)	0	0	1(1)	0
	Early Reloc	1(0)	0	0	7(1)	1(1)	1(0)	2(2)	3(1)	1(1)	0
	Late Reloc	3(1)	5(-1)	5(-1)	---	1(2)	5(1)	2(3)	4(2)	2(2)	2(-2)
2. Population receiving in excess of 200 Rems total marrow dose from early exposure (persons)	Evac-Reloc	0	0	0	2(3)	4(2)	4(1)	5(2)	2(3)	4(2)	3(0)
	Early Reloc	1(1)	0	0	1(3)	3(2)	2(1)	2(3)	2(3)	3(2)	0
	Late Reloc	1(2)	3(0)	1(0)	--	1(3)	9(2)	5(3)	7(3)	1(3)	5(0)
3. Early injuries (persons)	Evac-Reloc	4(1)	0	0	3(3)	5(2)	5(1)	6(2)	3(3)	5(2)	5(0)
	Early Reloc	5(1)	1(-2)	2(-2)	3(3)	4(2)	4(1)	2(3)	3(3)	4(2)	8(-1)
	Late Reloc	2(2)	2(0)	1(0)	--	1(3)	6(2)	3(3)	6(3)	1(3)	9(0)
4. Delayed cancer fatalities (excluding thyroid) (persons)	Evac-Reloc	6(2)	1(1)	4(1)	6(3)	2(3)	1(3)	4(3)	4(3)	2(3)	2(1)
	Early Reloc	6(2)	3(1)	5(1)	6(3)	2(3)	1(3)	4(3)	4(3)	2(3)	3(1)
	Late Reloc	7(2)	3(1)	5(1)	--	2(3)	1(3)	4(3)	4(3)	2(3)	3(1)
5. Delayed thyroid cancer fatalities (persons)	Evac-Reloc	1(2)	2(1)	2(1)	8(2)	6(2)	2(2)	1(3)	9(2)	6(2)	1(1)
	Early Reloc	1(2)	2(1)	2(1)	8(2)	6(2)	2(2)	1(3)	1(3)	6(2)	2(1)
	Late Reloc	2(2)	2(1)	2(1)	--	7(2)	2(2)	1(3)	1(3)	7(2)	2(1)
6. Total person-rems	Evac-Reloc	1(7)	5(5)	8(5)	4(7)	2(7)	2(7)	6(7)	6(7)	2(7)	4(5)
	Early Reloc	1(7)	5(5)	9(5)	4(7)	2(7)	2(7)	6(7)	6(7)	2(7)	5(5)
	Late Reloc	1(7)	5(5)	1(6)	--	2(7)	3(7)	7(7)	7(7)	3(7)	6(5)
7. Cost of offsite mitigation measures (1980 dollars)	Evac-Reloc	3(8)	5(7)	6(7)	2(9)	1(9)	1(9)	4(9)	3(9)	1(9)	1(6)
	Early Reloc	2(8)	2(6)	3(6)	2(9)	1(9)	1(9)	4(9)	3(9)	1(9)	1(6)
	Late Reloc	2(8)	2(6)	3(6)	--	1(9)	1(9)	4(9)	3(9)	1(9)	1(6)
8. Land area for long-term interdiction (m^2)	Evac-Reloc	1(6)	2(4)	3(4)	7(7)	2(7)	3(7)	1(8)	6(7)	2(7)	0
	Early Reloc	1(6)	2(4)	3(4)	7(7)	2(7)	3(7)	1(8)	6(7)	2(7)	0
	Late Reloc	1(6)	2(4)	3(4)	--	2(7)	3(7)	1(8)	6(7)	2(7)	0

*This release category has a probability less than 10^{-9} per reactor-year to be initiated by severe earthquakes; it is not analyzed with Late Reloc mode for its insignificant contribution to risks due to its low probability.

**2(2) = 2 × 10^2 = 200.

***These release categories are initiated by plant internal causes; therefore, the Late Reloc mode does not apply.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table K.1 (Continued)

Consequence Category	Offsite Emergency Response Mode	Release Categories									
		III-T/$\overline{\text{LGT}}$	IV-T/DW	IV-T/WW	IV-T/$\overline{\text{WW}}$	I-S/DW***	IV-A/DW***	IS-C/DW	IS-$\bar{\text{C}}$/DW	S-H20/$\overline{\text{WW}}$	S-$\overline{\text{H20}}$/$\overline{\text{WW}}$
1. Early fatalities with supportive medical treatment (persons)	Evac-Reloc	6(-1)	6(2)	5(2)	6(2)	0	7(2)	3(2)	1(2)	0	0
	Early Reloc	1(0)	1(3)	1(3)	1(3)	0	1(3)	7(2)	7(2)	2(2)	6(2)
	Late Reloc	7(1)	4(3)	4(3)	4(3)	--	--	3(3)	3(3)	2(3)	3(3)
2. Population receiving in excess of 200 Rems total marrow dose from early exposure (persons)	Evac-Reloc	5(1)	5(3)	4(3)	4(3)	0	4(3)	2(3)	2(3)	4(2)	4(2)
	Early Reloc	3(1)	6(3)	5(3)	4(3)	5(-1)	5(3)	3(3)	3(3)	1(3)	2(3)
	Late Reloc	1(3)	1(4)	1(4)	1(4)	--	--	9(3)	9(3)	5(3)	8(3)
3. Early injuries (persons)	Evac-Reloc	6(1)	5(3)	4(3)	3(3)	0	3(3)	2(3)	2(3)	5(2)	6(2)
	Early Reloc	4(1)	5(3)	4(3)	4(3)	5(-1)	3(3)	3(3)	3(3)	2(3)	2(3)
	Late Reloc	7(2)	7(3)	6(3)	7(3)	--	--	6(3)	6(3)	3(3)	5(3)
4. Delayed cancer fatalities (excluding thyroid) (persons)	Evac-Reloc	1(3)	5(3)	5(3)	5(3)	2(2)	5(3)	4(3)	4(3)	3(3)	4(3)
	Early Reloc	1(3)	5(3)	5(3)	5(3)	2(2)	5(3)	4(3)	4(3)	3(3)	4(3)
	Late Reloc	1(3)	6(3)	6(3)	6(3)	--	--	4(3)	4(3)	3(3)	4(3)
5. Delayed thyroid cancer fatalities (persons)	Evac-Reloc	2(2)	2(3)	2(3)	2(3)	3(1)	2(3)	9(2)	9(2)	7(2)	1(3)
	Early Reloc	2(2)	2(3)	2(3)	2(3)	3(1)	2(3)	9(2)	1(3)	8(2)	1(3)
	Late Reloc	2(2)	2(3)	2(3)	2(3)	--	--	1(3)	1(3)	8(2)	1(3)
6. Total person-rems	Evac-Reloc	2(7)	8(7)	7(7)	8(7)	3(6)	8(7)	5(7)	5(7)	4(7)	6(7)
	Early Reloc	2(7)	8(7)	8(7)	8(7)	3(6)	8(7)	5(7)	5(7)	5(7)	6(7)
	Late Reloc	3(7)	9(7)	8(7)	9(8)	--	--	6(7)	6(7)	5(7)	7(7)
7. Cost of offsite mitigation measures (1980 dollars)	Evac-Reloc	1(9)	5(9)	5(9)	5(9)	9(7)	5(9)	2(9)	2(9)	2(9)	3(9)
	Early Reloc	1(9)	5(9)	5(9)	5(9)	4(7)	5(9)	2(9)	2(9)	2(9)	3(9)
	Late Reloc	1(9)	5(9)	5(9)	5(9)	--	--	2(9)	2(9)	2(9)	3(9)
8. Land area for long-term interdiction (m^2)	Evac-Reloc	3(7)	1(8)	1(8)	2(8)	3(5)	1(8)	5(7)	6(7)	5(7)	8(7)
	Early Reloc	3(7)	1(8)	1(8)	2(8)	3(5)	1(8)	5(7)	6(7)	5(7)	8(7)
	Late Reloc	3(7)	1(8)	1(8)	2)8)	--	--	5(7)	6(7)	5(7)	8(7)

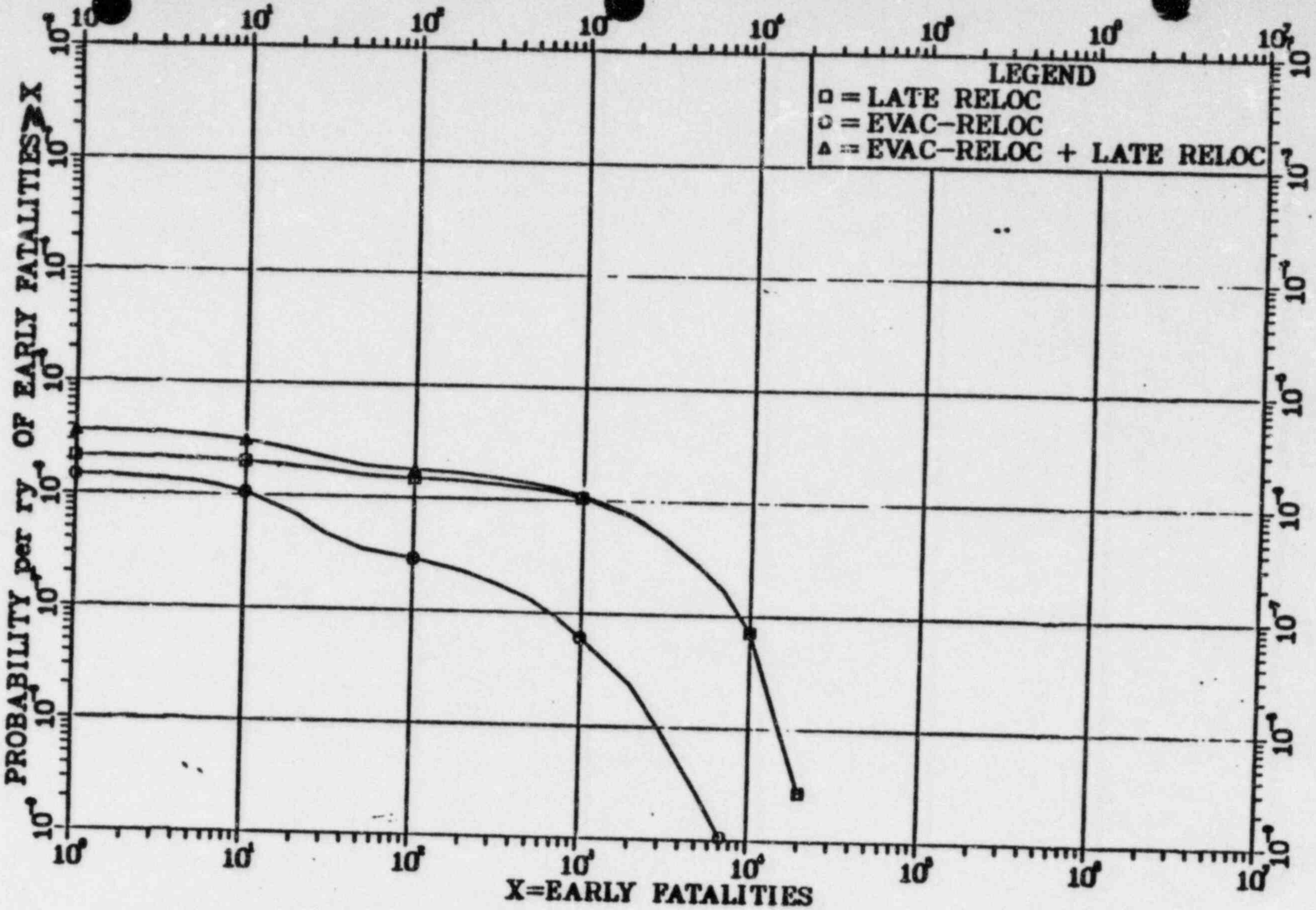


Figure L.10 Probability distribution of early fatality with supportive medical treatment

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

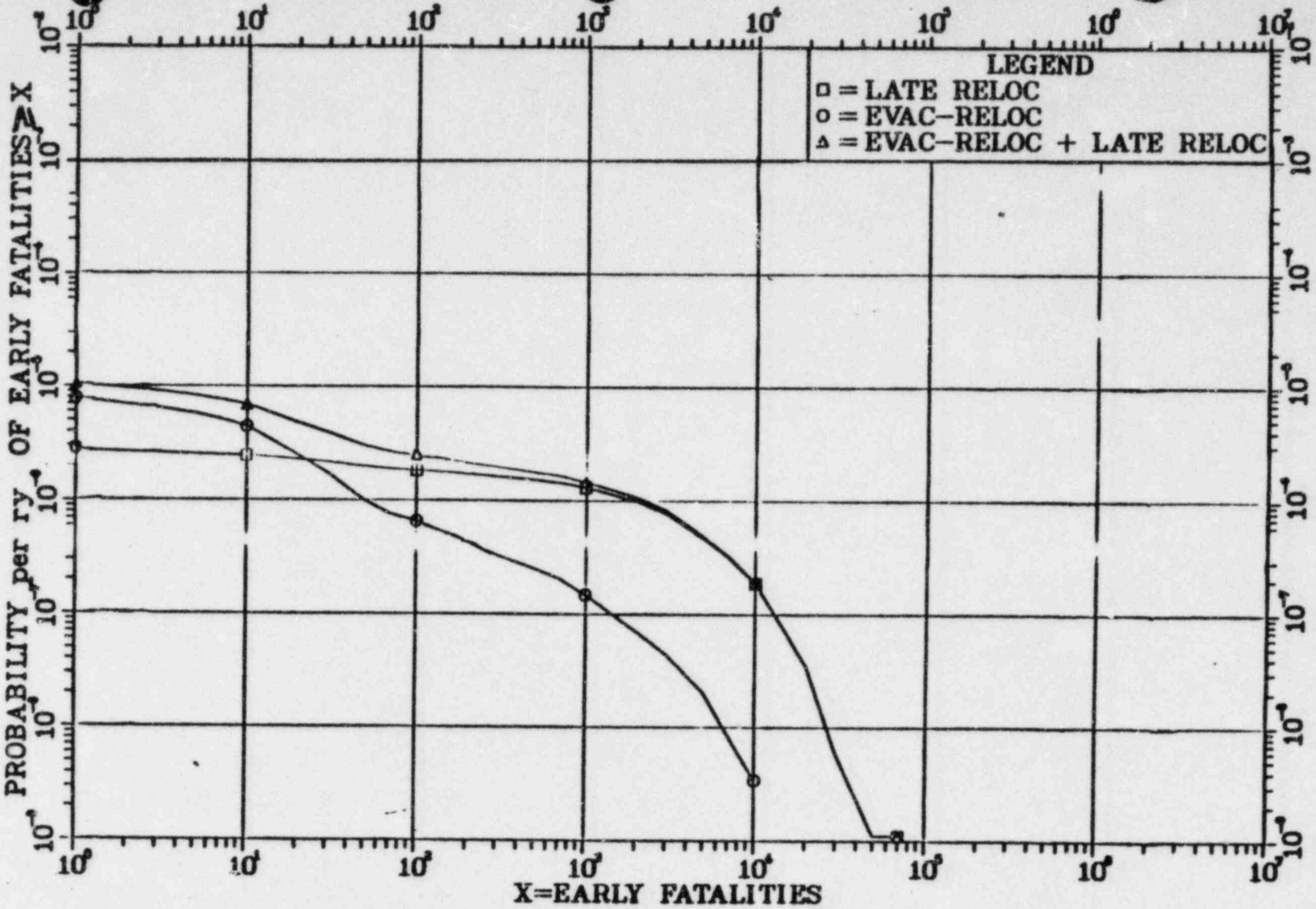


Figure L.11 Probability distribution of early fatalities with minimal medical treatment

NOTE: Please see Section 5.9.4.5(7) for a discussion of uncertainty.

Table 5.11g Summary of environmental impacts and probabilities

Probability of impact per reactor-year	Persons exposed over			Population exposure, whole body (million person-rems)*		Latent cancer fatalities (persons)				Early fatalities (persons)		Early injuries (persons)	Cost of offsite mitigation measures (millions of 1980 $)	Land area for long-term interdiction (millions of m^2)**
	300 rems thyroid dose	200 rems total marrow dose	25 rems whole body dose	50 miles (80 km)	Total	Excluding thyroid		Thyroid		With supportive medical treatment	With minimal medical treatment			
						50 miles (80 km)	Total	50 miles (80 km)	Total					
10^{-4}	0	0	0	0	0	0	0	0	0	0	0	0	0	0
10^{-5}	2(3)	3(1)	2(4)***	2(1)	3(1)	1(3)	2(3)	3(2)	3(2)	0	1(0)	9(1)	1(3)	4(1)
5×10^{-6}	7(3)	2(2)	5(4)	4(1)	5(1)	2(3)	3(3)	4(2)	6(2)	0	2(1)	2(2)	3(3)	7(1)
10^{-6}	4(4)	5(3)	3(5)	7(1)	1(2)	5(3)	7(3)	2(3)	2(3)	1(3)	2(3)	4(3)	6(3)	1(2)
10^{-7}	2(5)	3(4)	1(6)	1(2)	3(2)	1(4)	2(4)	4(3)	4(3)	9(3)	1(4)	3(4)	2(4)	3(2)
10^{-8}	5(5)	2(5)	3(6)	2(2)	5(2)	2(4)	3(4)	6(3)	6(3)	2(4)	3(4)	2(5)	3(4)	7(2)
See Figure	5.4b	5.4b	5.4b	5.4c	5.4c	5.4d	5.4d	5.4d	5.4d	5.4e	5.4e	5.4f	5.4g	5.4h

*About 260 cases of genetic effects may occur in the succeeding generations per million person-rem to the exposed generation.

**About 2.6 million square meters equals 1 square mile.

***2(4) = 2 x 10^4 = 20000.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table 5.11h Estimated values of societal risks from severe accidents, per reactor-year

	Consequence type	Estimated risk within the 50-mile region	Estimated risk within the entire region
1.	Early fatalities with Supportive medical treatment (persons)	5(-3)*	5(-3)
2.	Early fatalities with minimal medical treatment (persons)	8(-3)	8(-3)
3.	Early injuries (persons)	2(-2)	2(-2)
4.	Latent cancer fatalities (excluding thyroid) (persons)	4(-2)	7(-2)
[illegible].	Latent thyroid cancer fatalities (persons)	1(-2)	1(-2)
6.	Total person-rems	7(2)	1(3)
7a.	Cost of offsite mitigation measures (1980 $)	5(4)	5(4)
7b.	Regional industrial impact costs (1980 $)		5(4)***
7c.	Plant costs (1980 $)	1(5)	
8.	Land area for long-term interdiction (m^2)**	1(3)	1(3)

*5(-3) = 5×10^{-3} = .005

**About 2.6 million m^2 equals to 1 mi^2.

***Excludes costs of crop and milk interdiction, which are included in 7a.

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

RISKS OF SURFACE/GROUND WATER CONTAMINATION ARE NOT INCLUDED IN THIS TABLE

Table L.1b Societal risks within the entire region of Limerick site with Evac-Reloc* and Late Reloc* offsite emergency response modes

Consequence type	Risk per reactor-year		
	From causes other than severe earthquakes (Evac-Reloc)	From severe earthquakes (Late Reloc)	Total
1. Early fatalities with supportive medical treatment (persons)	2(-4)**	5(-3)	5(-3)
2. Early fatalities with minimal medical treatment (persons)	7(-4)	8(-3)	8(-3)
3. Early injuries (persons)	1(-2)	1(-2)	2(-2)
4. Latent cancer fatalities (excluding thyroid) (persons)	6(-2)	1(-2)	7(-2)
5. Latent thyroid cancer fatalities (persons)	1(-2)	2(-3)	1(-2)
6. Total person-rems	1(3)	1(2)	1(3)
7. Cost of offsite mitigation measures (1980 dollars)	5(4)	6(3)	5(4)
8. Land area for long-term interdiction (square meters)	1(3)	2(2)	1(3)

*See Section 5.9.4.5(2).
**2(-4) = 2 x 10^{-4} = .0002

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

Table M.1b Societal risks within the entire region of Limerick site with Early Reloc* and Late Reloc* offsite emergency response modes

Consequence type	Risk per reactor-year From causes other than severe earthquakes (Early Reloc)	From severe earthquakes (Late Reloc)	Total
1. Early fatalities with supportive medical treatment (persons)	1(-3)** (4)	5(-3)	6(-3) (1)
2. Early fatalities with minimal medical treatment (persons)	2(-3) (3)	8(-3)	1(-2) (1)
3. Early injuries (persons)	1(-2) (1)	1(-2)	2(-2) (1)
4. Latent cancer fatalities, excluding thyroid (persons)	6(-2) (1)	1(-2)	7(-2) (1)
5. Latent thyroid cancer fatalities (persons)	1(-2) (1)	2(-3)	2(-2) (1)
6. Total person-rems	1(3) (1)	1(2)	1(3) (1)
7. Cost of offsite mitigation measures (1980 dollars)	5(4)	6(3)	5(4)
8. Land area for long-term interdiction (square meters)	1(3)	2(2)	1(3)

*See Section 5.9.4.5(2).
**1(-3) = 1 x 10^{-3} = .001

NOTE: Please see Section 5.9.4.5(7) for discussion of uncertainties. Estimated numbers were rounded to one significant digit only for the purpose of this table.

COMPARISON WITH BACKGROUND RISKS

RISK TYPE	LIMERICK REACTOR ACCIDENTS	BACKGROUND
POPULATION EXPOSURE WITHIN 50 MILES	700 PERSON REMS/RY	800,000 PERSON REMS/YR
LATENT CANCER FATALITIES WITHIN 50 MILES	5×10^{-2} * CASES/RY	10,000/YR *
EARLY FATALITIES WITHIN 1-MI FROM EAB	6×10^{-4} ** CASES/RY WITH MIN. MED. TRTMT.	2/YR **

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* RATIO OF BACKGROUND TO LIMERICK REACTOR ACCIDENTS $\approx 5 \times 10^{-6}$

** RATIO OF BACKGROUND TO LIMERICK REACTOR ACCIDENTS $\approx 3 \times 10^{-4}$