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PUBLIC MEETING

Nuclear Regulatory Commission
One White Flint North
Rockville, Maryland

Wednesday, March 15, 1989

The Commission met in open session, pursuant to notice, at 2:00 p.m., Lando W. Zech, Jr., Chairman, presiding.

COMMISSIONERS PRESENT:

LANDO W. ZECH, JR., Chairman of the Commission
THOMAS M. ROBERTS, Commissioner
KENNETH M. CARR, Commissioner
KENNETH C. ROGERS, Commissioner
JAMES R. CURTISS, Commissioner

STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

SAMUEL J. CHILK, Secretary
MARTIN MALSCH, Deputy General Counsel
DOCTOR DENWOOD ROSS
ERIC BECKFORD
JOSEPH MURPHY
VICTOR STELLO, JR., Executive Director for Operations

P-R-O-C-E-E-D-I-N-G-S

2:05 p.m.

CHAIRMAN ZECH: Good afternoon, ladies and gentlemen.

This is an information briefing this afternoon in which the staff will provide a status report and preliminary results of NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants.

NUREG-1150 was published as a draft for comment in February of 1987. Extensive public comments have been received. In addition, the draft document has been subject to three independent peer reviews and the staff has received comments from the international community.

The staff has been in the process of improving the report to address the comments received. Some of these improvements and the results available to date pertaining to accident frequencies from internal events will be discussed with the Commission today.

In addition, the staff has been requested to discuss the possible uses of the information that has been developed to date. I'd like to emphasize, though, that this briefing will focus on a limited but

very important part of NUREG-1150, that is the aspect of NUREG-1150 that considers severe core damage frequency from internal events.

When the final peer reviews have been completed and NUREG-1150 is published, we expect that it will represent a major advance in the methodology for examining the risks associated with five specific nuclear power plants as well as the uncertainties associated with those risks.

I understand that the slides for the presentation this afternoon on the staff paper SECY 89-058 should be available at the entrance to the meeting room.

Do any of my fellow Commissioners have any opening statements before we begin? If not, Mr. Stello, you may proceed.

MR. STELLO: Thank you, Mr. Chairman.

In the paper that we sent to the Commission on February the 17th it indicated that we would report back to you on our meeting with the ACRS, which I'll do in a moment, and most recently have sent you a paper that complies with the Commission's directions thus far with respect to peer review on March the 14th. And I have a few comments I want to make about those.

What I propose to start with is a broad picture of where we are. We are here today to brief you on the status. We don't need any direction or decision from the Commission at this point. And expect to have the next draft of 1150 ready in the middle of April and would suggest a further meeting of the Commission when that document is ready. At that point, hopefully, and Eric Beckford will explain where we are in getting ready for the peer review process, to only begin then the peer review. But, of course, we have the issue of what do we do with the report while it is being peer reviewed, which are the interim uses. That is the particular question that I went and met with the ACRS to deal with.

You recall you got a letter from them that clearly endorses the peer review and they are satisfied and anxious that that process begin. But they indicated in that letter that they had not yet reached any conclusion with respect to interim uses.

We met with them and suggested that they ought to provide their advice to the Commission with respect to this issue of interim uses. They, as best as I can understand, hope to be able to do that but probably not before their April meeting and possibly even the meeting might be later in May when they

expect to deal with the issue and, hopefully, will communicate to the Commission their views on interim uses.

We will this afternoon tell you what our view of interim uses and why we believe they're proper and appropriate while the peer review process is ongoing.

One other aspect that has come up in the last month or so would suggest for external events, specifically seismic earthquakes, earthquake hazard to nuclear plants, that in looking at that issue we are looking at earthquakes which are extremely severe and at best is understanding we have of low probability. But dealing with that issue we believe needs to be bifurcated in a particular way which we have in our memo of March 14th proposed a particular way we think makes sense to us since talking about earthquakes that are 8, 9, 10 times the SSC, or to say it in a way that's understood perhaps easier, earthquakes of G values that are at levels in excess of 1.0 representing a seismic event of such a magnitude that it's going to require special consideration. And how do you even evaluate the consequences of such severe earthquakes? In some cases --

CHAIRMAN ZECH: What would that be in the

Richter scale?

MR. STELLO: I don't know. Does anybody know that answer? Joe?

MR. BECKFORD: Well, it's approaching 1-G ground acceleration, so it's pretty high.

CHAIRMAN ZECH: I understand that. Can you translate that to a Richter scale event, which is more generally public understanding, I think, than the 1-G?

MR. BECKFORD: We can certainly provide that number. I don't want to provide it off the top of my head. I'm not a seismic expert --

MR. STELLO: About 8? Magnitude of 8?

MR. MURPHY: About 8, yes.

MR. STELLO: Let's get the answer for the record, but if we had to make a judgment, it's magnitude 8 or probably higher.

MR. MURPHY: Eight or higher.

MR. STELLO: Or higher. The point being it is extremely severe earthquakes.

CHAIRMAN ZECH: Give us that, too, when you have a chance to.

MR. STELLO: We will.

MR. BECKFORD: We'll provide it to the Commission.

MR. STELLO: I think in one context it's

somewhat good news, which means you have to have very, very severe earthquakes before you start to predict damage to the facilities. And that's as we would expect because they are very robust facilities and designed to fairly high standards to begin with. But we will deal with that issue when we come back to the Commission in April or May, whenever the report is ready.

COMMISSIONER ROBERTS: I still have to wonder why we're expending time and resources on such a low probability event.

MR. STELLO: I think it would be best if we could hold that until we have the briefing on the subject.

COMMISSIONER ROBERTS: All right.

MR. STELLO: We're not really prepared to do that today. That's a legitimate question.

COMMISSIONER ROBERTS: Of course it is.

MR. STELLO: How far out should you go in evaluating damage? How far out in that spectrum do you go and how do you calculate it?

CHAIRMAN ZECH: All right. Make that part of your presentation when you come next time.

MR. STELLO: We're not really ready to do that today, but rather to suggest it's a unique

problem in setting the charter for the peer review group, which is what we need to do. We've suggested an approach in our March 14th letter because we think it is going to be a rather special problem to deal with. And that was explained briefly in our March 14th memo, but we will have another opportunity to deal with that with the Commission.

With that brief introduction, I'll ask Eric to give you some more specifics on the subjects to concentrate on the status of the peer review and our readiness. We still have some steps to take. And then we'll turn immediately after that to the briefing itself with Joe Murphy.

So Eric?

CHAIRMAN ZECH: All right. Thank you very much. You may proceed.

MR. BECKFORD: Yes. Thank you, Mr. Chairman.

I think what I would add to Mr. Stello's statements on the status of 1150 relates to the peer review panel in the letter dated yesterday. We have advised the Commission how we would propose to compose that panel. Your formal approval will be requested when the report itself is presented, but we wanted to tell you this is our thinking as to who should serve

on that panel.

There's a letter in the Office of the Secretary now in draft form which will go to the General Services Administration to obtain the GSA approval for establishing the peer review panel. It will be a federal advisory committee panel and that requires that approval. We are using the charter which accompanied the letter yesterday on the 1150, because we have to give a charter to the GSA for their approval. So we'll be going ahead on the basis that that charter will have your approval or it will be close enough so it will satisfy that purpose.

Our expectation is that by the time that GSA has considered it and approved it, that that would be about coincident with the time that we would want to get the peer review effort underway, which is after you have received and considered the 1150 and heard the presentations on it. I assume that would be sometime by or after the middle of May.

I think that's all that I need to say. We have been in touch with the proposed members of the peer review and everyone has agreed to serve. Doctor Kautz, proposed chairman, has been in conversation planning when these meeting would take place. So the preliminary work is all taken care of to set that

study in motion.

I think that's all I need to say about that. And I'd propose that we --

COMMISSIONER ROGERS: Could I just ask you--

MR. BECKFORD: Yes.

COMMISSIONER ROGERS: -- do you expect to set a rough timetable for the peer review?

MR. BECKFORD: Well, I have in mind a time that I would like to see it completed, in about eight months from the time that it gets underway. But, of course, you can't predict these things. It seems to me that that is doable over an eight month period. In other words, by the end of the year just after December they should be completed or very close to it.

CHAIRMAN ZECH: All right. Thank you very much.

Mr. Murphy, you may proceed.

MR. MURPHY: Could I have slide 2, please (Slide).

What I would like to do is I will identify the major areas of the improvement that we have made since the staff report was issued two years ago, briefly touch on the current status, present the accident frequency results that we have to date for the internal events, as we said, and then touch on the

prospectives we've gained from the work that we've done at this point.

If I could have the next one, please.

(Slide.)

The major improvements in the accident frequency analysis have come in an updated accident frequency analysis. And the change here has been not so much a reflection of criticisms that we received in the draft, but the fact that the plants have changed with time. And we're now reflecting the plants as they were in March of last year as opposed to what they were when the draft was done.

The changes, in many cases, resulted from the items that were found in the draft report. The utilities saw what we had found, identified a vulnerability they felt existed in their plant, and they fixed it. For this reason, as you see, the numerical results, in general they've gone down. And that's happened largely because the problems that we had identified in the draft report no longer exist.

We have an improved uncertainty analysis. I think this was the major point of criticism that we had received the last time.

We have used decision theoretic expert judgment technique very similar in concept to that

used by EPRA in their seismic work, which was one of the recommendations of both the uncertainty peer review committee as well as the Katzenberg Committee.

And in the plant specific analysis we've done more plant specific failure rate analysis of the data rather than using generic data.

And we are in the midst right now of a concerted effort to have improved documentation because I think one of the biggest problems we had last time was a failure to communicate what our results were. And, hopefully, we're doing a better job this time.

The next side, please. (Slide.) In the work that's still in progress dealing with the accident phenomenology and risk analysis here, because we have utilized expert opinion in a much more formalized manner than we did before, we have many more experts involved to run from a much broader segment of the nuclear community with a significantly larger information database. Effectively the backend, as we call it, of the accident phenomenology and risk analysis have been completely redone. We've made a significant number more of source term code package runs. There have been a significant number of other code runs of the advanced severe accident codes that

provided information to support the decision making process. The expert opinion elicitation process.

In addition to that, we are using an improved consequence analysis code and we have added the consideration of external event for two plants, Surry and Peach Bottom.

There's a substantial amount of new research available to us in the interim between the draft report and this version. Doctor Ross will discuss that.

COMMISSIONER ROGERS: Could I just ask a question before you move on? You say you have an improved uncertainty analysis. Was it possible to get any feeling about what happened to the uncertainties that would have come from such an analysis with the original more limited number of experts versus the expanded number of experts? In other words, did you get any feeling about whether to start to diverge the system or converge it?

MR. MURPHY: I suspect the methodology itself didn't change things too much. By bringing in the expanded pool of experts, in some cases our uncertainty bounds expanded because with a larger group of experts we had more varying views. In some cases where we had people drawn largely from the

national laboratory community, they --

MR. BECKFORD: There were more optimists.

MR. MURPHY: Yes. When we added the industry people, we added more optimists essentially.

COMMISSIONER ROGERS: Added more what?

MR. MURPHY: Optimists so that the uncertainty bounds grew, but they grew in the lower direction.

COMMISSIONER ROGERS: Oh, I see. Yes. Okay. And you would see that be somewhat be in a difference between the mean and the medians, I would imagine, in those distributions then?

MR. MURPHY: Yes, sir. When we come back a month from now, you'll see some fairly substantial differences between the means and the medians.

CHAIRMAN ZECH: All right. Let's proceed, please.

DOCTOR ROSS: At the top of page three of SECY 89-058 there was a paragraph on advances in the science since the last two years, since the draft version of 1150 and the version that we're preparing this spring. A little more detail what we mentioned, there's some detailed codes.

What we've done in the last two years is to couple the detail melt progression codes, which are

focused mostly on the core and to follow the core as it goes down and melts through the reactor vessel, we coupled those codes with our system codes that we previously had used for the loss of coolant accident. And this gave us a more mechanistic better understanding of core melt progression. And in doing so, we got a much better, more detailed, more realistic model for the boiling water reactor.

The second line on this chart, BWR severe damage, we did a detailed melt down experiment in the ACRR, a research reactor at Sandia. We've had an impile test involving what we think was a reasonably typical boiling water reactor geometry, including a control blade and a zircaloy channel box. And this gives us a lot more information on how the core melt progression would occur in the BWR and it aids in our model.

The third bullet we mention some prolink tests in NRU, a research reactor in Canada. And we build in this country test bundles and ship them to NRU and then they run these in their reactor in a degraded cooling mode and getting temperatures eventually up to 2400 degrees C.

The big advantage here is we get a few more pins, radiated pins are included and they are full

length. This gives us important information on zircaloy oxidation and hydrogen generation.

The last item on this slide has to do with hydrogen, the transition from deflagration to detonation and the fact that we're getting some information that as the temperature goes up, the deflagration and detonation limits change and come closer to each other. And since a good number of the reactors that we deal with do not have any noted atmosphere, then the question of hydrogen combustion and its effects have to be considered.

We have several computer codes that we are improving and refining. An important thing that we're getting here is we're getting a lot of information and cooperation from other countries, especially the Federal Republic of Germany.

Let's go to the next slide. (Slide.) The acronym on the first line, DCH, refers to direct containment heating. We have a fairly large facility, one-tenth scale of a containment at Sandia, where we get prototypic materials and inject them at high pressure into this simulated containment environment and observe what happens in a heat transferred chemical reactions.

We have a companion test, referred to the

second bullet on this slide, being done at Brookhaven National Lab. It's smaller scale, is 1/42nd scale. And we're using semblance there of wood, metal and water. And in both cases, both Sandia and Brookhaven, we're trying to get more knowledge, more science on how molten core materials might be dispersed if they melt through the bottom of a pressurized water reactor while the vessel is at high pressure.

One thing that's been suggested at various countries is the way to avoid this phenomena is put in a primary system depressurization mode, which brings up the question is how low do you have to get in pressure before DCH no longer becomes a problem. And that's an important part of our short term research program.

The third bullet, we've done some fairly large, what we think, are prototypic core concrete tests where we get a molten material such as stainless steel or zurcoxide together with quantities of unreacting zirconium. We're talking about 200, 250 kilogram quantities. Poured on concrete of different types of concrete and observe the concrete oblation, gas formation and especially what happens when you put the zire in and then compare the results with our computer codes core con. And through a standard

problem with computer codes developed by other countries.

The last item, and one that's been somewhat a controversy, is the Mark-1 melt spreading. Sometimes we call this a dry well melt through question which can, under some circumstances, fail the steel liner of a BWR Mark-1 containment. We've done experiments both at Sandia and at Brookhaven, more at Brookhaven, in the ten kilogram range. We've had molten lead poured on various flat plates with and without water and at various super heats to see what the regime of these materials are and how would you expect the molten materials to spread to the liner, if at all, and what the effects of water are.

We'll be doing larger experiments once we've done a test matrix here and comparing it with other experiments done at the University of California, Santa Barbara and some experiments done at Sandiea.

Now, this is the research progress roughly in the last two years. We had to freeze the technical input to the version of NUREG-1150 about nine months ago, roughly. At that point in time all of the work was done, the expert opinion had been elicited and we've been doing calculations and analysis since then. Any experimental results since that time won't be in

the report. Of necessity, you have to have a freeze date. This is important as we deal with our peer review because if something new has transpired since then, and we'd expect good results to be coming in more or less uniformly, then whatever that tells us will be something we'll have to consider in the future as code development.

We do want to make it clear to our peer committee and to the Commission we didn't drop the gate and say that's when we froze the technology and now we have the rest of the time is on report writing. That's where we are. And I'm ready to go back to Joe now.

CHAIRMAN ZECH: All right. Thank you very much. You may proceed.

MR. MURPHY: In terms of our current status, the contractor analysis are essentially complete or will be by the end of this week. The computer is turning out answers as we speak.

We need to take the time between now and our next meeting for the final quality assurance quality control check of what they've done and for the staff to understand what they've done and to prepare the final report.

We are committed to timely completion and at

this time we see no reason why we can't have the document ready for distribution to Mr. Stello and the Commission in mid-April. I believe we're still on the calender for a Commission meeting the first week of May.

Before I present the results on the next slide I'd like to discuss some considerations that I ask you to consider as you look at the results.

First is that a single value of the risk is rarely a sole basis for regulatory decision. We will show the distributions, we will show the means and the medians of what I'm about to show you. I think you have to look at the entire range of the information rather than just a single value.

In some cases accident management procedures are in place at these plants, in other cases they are not yet in place. This can make a substantial effect on the risk and can significantly reduce it.

There are substantial differences between these plants and between those and similar plants. The results are highly plant-specific. It would not be appropriate to look at, say, the Peach Bottom results and think they were applicable to all General Electric Mark-1 plants or the same thing true of the sub-atmospheric plants with Surry or any of the

others.

And finally, there's a problem with PRA that we should identify, and that is that the current state of the art of problemistic risk analysis is such that it cannot reflect management influences. And there I mean really the safety culture of the plant except as it is reflected in the plant specific failure rate data. A prolonged long poor maintenance practices will show up as higher failure rates, but a change in the safety ethic of a plant in a relatively short period of time won't be reflected in these analysis. And effectively, other steps have to be taken to, we like to use the phrase, "make the PRA come true." And that is, to make sure the plant is operated in the way that it was assumed to be operated as we did the analysis.

As I mentioned earlier, the results for this briefing are limited to the severe core damage frequency from internal events. We have presented them both as distributions and as bars that identified the means and medians. And we will talk about the relative contributions of the various initiating events to the mean severe core damage frequency.

On the next figure, labeled Figure 1 --

CHAIRMAN ZECH: Page 10. That's it.

MR. MURPHY: -- we show the results from the five plants. As you can see, the distribution on a large scale are relatively symmetrical. The results for the two BWRs are relatively low. I ask you in looking at these not to think just in terms of these five studies, but there have been a large number of PRAs done over the years. While I tell you that they were done to different assumptions and boundary conditions, if you were to take all of the PRAs that have been done and averaged them, you get a number closer to ten to the minus four as the average mean frequency. These plants are lower than that, at least four of the five are. Zion is somewhat higher and we'll discuss that in a second.

On the next figure we show this more in terms to make the individual statistics more apparent. Here the upper and lower ends of the box represent the five and 95 percentiles of the distribution. The plus sign represents the median of the distribution while the circle represents the mean.

Typically there's a factor of two to three difference, in some cases a little bit more between the median and the mean. Again, the mean values for the BWRs are roughly the same and they're a few times ten to the minus X where the PWRs are somewhat higher

than that.

The Zion plant is, I guess, about three times ten to the minus four. There are special considerations with Zion that I would like to discuss next.

On the next viewgraph -- in the dominate accident sequences we've analyzed it on the Zion plant arises from a loss of the component cooling water. And this loss of component cooling water can occur from one of two ways. Either from a common cause failure of the pumps of the system or from a pipe break in the system. The results of the loss of component cooling water is that you lose cooling to the reactor cooling pump seals, which can lead to a seal failure. And in addition, you lose cooling to the high head safety injection pumps which could cause loss to the safety injection pumps. In this case you have a small LOCA with no high head safety injection arising from a single failure.

I would point out in this, however, that about 80 percent of the frequency comes from the component cooling water pipe break frequency that we've used. This was obtained through expert elicitation, but it was a highly uncertain number. We're talking low pressure piping and there's not much

data regarding the failure rate of said piping.

There are new pump seals available -- new O-rings available for the reactor coolant pump seals of Westinghouse pumps. Installation of the new pump seal would essentially eliminate this problem. It would take the leakage rate associated with the loss of cooling from a few hundred GPM and lower it to tens of GPM.

Finally, the Zion charging pumps are used for the high head safety injection. And we have assumed, as stated in the FSAR, that they require component cooling water cooling. They are very similar in design to similar pumps which are used on Sequoyah. And TVA provided us information that they have been able to run those pumps for 24 hours without cooling and they have successfully functioned.

If the same thing were true on Zion, this problem would essentially, again, go away. It would lower the dominate sequence by well over an order of magnitude so that the -- in looking at the Zion results, I think they have to be looked at in light of this prospective that we are not sure of the performance of the charging pumps under this environment and then in addition to that the uncertainty in the analysis and the relative ease of

fixing the O-rings in the reactor coolant pumps.

MR. STELLO: Maybe I ought to make a comment at this point to emphasize the rather dynamic nature of PRA work. Joe has not indicated it, but in a number of these plants while the team was there doing the PRA and reviewing things that procedures were developed on the spot that would just simply eliminate problems. And I now have been informed that Zion, in recognition of the fact and I think you'll see in a little while, that the risk is essentially dominated by this particular single event. And you can decrease the overall core melt frequency by fixing this problem by a factor of ten. And they have, in fact, made some changes to the plant already. So what you are about to see does not now even reflect the plant as it exists where this particular problem has already been addressed by the utility and they, in fact, are going to be doing more which, because of the dynamic nature of the process, the core melt frequency would, in fact, already shift. I guess, nearly in order of magnitude?

MR. MURPHY: Yes. Because of that one change, which is part of the insight you get out of doing this and the company on their own has reflected, as I guess it's fair to say essentially all of the

utilities that were involved in this process have also done, to a large extent the changes that they've already made have been incorporated in the results you see. In this case, there are changes that are being made that already effect this number and this number therefore does not, even as we speak, reflect the actual plant that exists today.

Okay, Joe.

MR. MURPHY: In the next viewgraph, labeled Figure 3, we show the principal contributors to the core damage frequency in the boiling water reactors. You see there's a strong influence of station blackout in the Peach Bottom plant. There's also a significance associated with the anticipated transients without a scram. Relatively low contributions from anything else.

COMMISSIONER ROBERTS: But does this reflect the implementation of the station blackout rule?

MR. MURPHY: Not a complete implementation in that we analyzed the plant as it existed, as I said, last March and there was not complete implementation at that point.

I point out, though, that something has to be dominate. And these numbers are very low, so the fact that you have a tall spike say, on Grand Gulf, it

says that station blackout is the dominate contributor to a very low core damage frequency, is not something to raise a concern over.

We have a similar figure on the next for the PWRs. And here the fact on Zion on the seal LOCA becomes apparent. The point I was making, the thing that was driving it is the tall bar on the right. With the fixes that Mr. Stello refers to, it would be reasonable to expect that that would drop from the range of 25 to 30 times ten to the minus five, down to on the order of three or less. So that that would bring Zion back in the range of the other plants.

Here again we do see an influence of station blackout, varying influence depending on the plants and the specific design features of the plants. And contributions from loss of coolant accidents, and this is primarily, as I'll get to in a minute, from cases where the switch over from the injection mode to the circulation mode of ECCS is a manual operation.

What I'd like to do next on the next few--

COMMISSIONER CARR: Can you tell me why you shifted evidence on the graphs?

MR. MURPHY: Mainly just to --

COMMISSIONER CARR: Just to confuse me, right?

MR. MURPHY: Well, not intentionally. But the intention was just to fill up the graph.

COMMISSIONER CARR: Well, but you've put those minor ones and Sequoyah and Surry down to where they don't look important at all and you've brought them up in Peach Bottom and Grand Gulf. I don't mean the tall ones, but it's the lesser significant ones that you're hiding.

MR. MURPHY: Yes. I would agree that we could give higher credence to the lower numbers here than you should. The key point is that these numbers are low. The size of graph doesn't make that evident.

MR. STELLO: If you can change the scale to make them look the same way, then Zion won't fit on the scale either.

COMMISSIONER CARR: Well, just change them to Zion's scale.

DOCTOR ROSS: We show the pie charts, I think that problem will go away.

MR. STELLO: We'll find more than one way to display them so that it's clear.

CHAIRMAN ZECH: All right. Let's proceed.

MR. MURPHY: In terms of the insights and prospectives we've gained, again in terms particularly of the accident and frequency analysis what 1150 does

is we've added five plants to the compending of PRA information that already exists. And there we have available to us in the NRC, I guess, 25 or so PRAs and there have been more than that done, some of which are not available to us. The insights we've gained from these five studies are added to that information database and has to be tested against that database. If we find something that's vastly different from what's been found before, we needed to go back and look at it carefully to make sure it was valid.

In terms of the boiling water reactors that we study, both of them, as you've seen, were dominated by station blackout and ATWS, but the absolute values were quite small. One of the reasons station blackout was so small at Peach Bottom was that they have very highly reliable diesel generators. They have four diesel generators at the plant, anyone of which can power both units. And historically they have about the best diesel generators that we have seen from a reliability standpoint, from a failure rate standpoint on their diesels.

At the Grand Gulf plant they have an independent diesel that powers the high pressure core spray system. This is on the next viewgraph.

Battery depletion is an important item to

consider in a long-term loss of all AC power. One of the things that can cause problems is if during the period where you're operating on turbine driven pumps, you lose DC power and therefore lose control power and the ability to control the situation you're in. Again, Peach Bottom has a very effective low shedding program so that their batteries will last about 12 hours without battery chargers. Some of the other plants we looked at were as short as four hours.

The failures in the emergency service water system could be very significant at Peach Bottom. A degradation of the system could markedly increase the core damage frequency, but more than an order of magnitude.

We looked not only in terms of what contributes to our assessment of core damage frequency, but also what it's most sensitive to, what feature if it were to increase, cause the highest-- the impact on increasing the core damage frequency. And in this case, the emergency service water system is very important.

And the PWRs we've already talked about the Zion situation. Again, with the exception of that seal LOCA problem, they tend to be dominated with station blackouts and LOCAs. One alternative method

for cooling the core that we have given credit to in loss of coolant accidents has been the feed and bleed mode of cooling. Here we find that if there are block valves, the block valves on the steam generator pressurizer relief valves were closed because of leakage. This can significantly reduce the reliability of that type of a cooling mode.

Station blackout is significant at Surry primarily because this is a plant where for a two unit site there are only two diesel generators, one for each plant and one that's a swing unit between the two plants.

At the Sequoyah plant we found that the-- on the next viewgraph -- early initiation of containment sprays in an ice condenser plant can lead to an early draw down of the refueling water storage tank. To recall the type of situation I'm talking about, the ice condenser plants have a large body of ice found in the annulus of the containment that will condense steam that's generated. And therefore, there are relatively small containment compared to other PWRs and are low pressure design so that the pressure at which sprays are activated in an accident is significantly lower. It's a few PSI at Sequoyah versus 25 PSI at Surry, for instance.

What this does is the sprays come on sooner and they pump a lot of water, so the draw down of the tank occurs faster so the switch over from injection to recirculation happens earlier in time at a time when it's more stressful. And in this plant, it's a manual operation so that the human factor associated with an error in making that transfer is what caused this to be important.

In more general insights that we've gained, is that the operator recovery actions during an accident have had a significant effect on core damage frequency. In many cases, in effect, greater than in a factor of 10 on a given accident. And we have credited such actions where there were procedures to do so or where there was an extremely long length of time for diagnoses.

The symptom based procedures that are available now have had a significant effect in reducing the human error probabilities we've assessed.

We have found that the cross ties --

CHAIRMAN ZECH: I think that's very important. Could you elaborate on that just a little bit because I would agree with your conclusion. Tell us about the differences that perhaps that you saw in the symptom base procedures as opposed to the others?

MR. MURPHY: What we found was that in the past the problem was, of course, that the old style procedures you had to diagnose the sequence you were in and then proceed down the track. So the initial big possibility for human error was misdiagnoses of the type of accident you were in.

CHAIRMAN ZECH: You had to essentially analyze the casualty and the cause of it right then and there and then you proceeded from there?

MR. MURPHY: Yes. In this case you just --

CHAIRMAN ZECH: That's described kind of briefly symptom oriented. I think that's important, you know, to have a brief explanation of that.

MR. MURPHY: We can put it more in the report. But basically here the operator responds to the symptoms that he's receiving in the control room. The way the errors were assessed in our analysis, we looked at the information that was flowing into the control room and what his procedures told him to look for. And then looked at each step of the procedures as he had to operate in using human reliability techniques developed by psychologists, looked at each step in this procedure in terms of the likelihood of an error as he progressed along.

CHAIRMAN ZECH: Well, I'd say it a little

differently. But let me just say what it means to me.

The other procedures we had event oriented. You kind of diagnosed the whole event. You say there's the casualty and that's the reason, and you kind of went backwards and you did things that you thought would take care of the casualty. You had to really diagnose the actual cause and what was going on. And we found out, at least this has been my experience, that that was putting a great burden on the operator, in some cases. And then the symptom oriented process came along.

Now what the symptom oriented process does, in my view, is no matter what kind of an incident you might have in the control room, the operator immediately takes actions. He doesn't have to diagnose exactly what's wrong with the plant. He doesn't have to figure out the cause. But he immediately takes actions one by one that will prevent that accident from becoming a serious accident. Just by following the procedures of the symptoms and addressing them, the actions he'll take will eventually take care of the casualty to the point where he's solved the problem that he may still not have understand exactly what took place. But by so doing, you keep the plant under control.

And so that's very important, in my judgment. And the operators I've talked to at the many plants I've visited are very appreciative of the fact now that whenever they have a problem, they address these procedures, symptom based, and immediately can take actions to keep the plant under control and to get it under control.

So that's a very important, I think, progress. And we've made that since the Three Mile Island accident, as the staff well knows. But I do think that's been significant. And again, I think the operators generally have a great appreciation.

It takes a burden off them. They already have fixed procedures and they know those procedures will keep that plant under control at each step of the procedure. So rather than having to diagnose the whole casualty and know everything about it from the beginning, they have these procedures that control the casualty and keep the casualty under control and eventually keep the plant in a safe condition.

So it's very important, I think. And I appreciate your bringing it up, but I just wanted to emphasize that.

I've been impressed by the difference that's taken place in those casualty procedures. I think

it's a lot more than the psychologists that you might have referred to might be important, but I think from the operator's standpoint, it gives him a great deal of confidence. And the operators I've talked to using these procedures in the simulator, which is the only place you really can practice because, you know, you hope you'll never have these casualties. But on the simulator they can go through all the casualties they can imagine and use the procedures, the exact procedures they'd use and practice on those to the point where they gain confidence that symptoms they're addressing will be taken care of as they proceed through the accident sequence.

So it's a very important step. I think it may not be as fully appreciated as -- at least as I appreciate it, having watched and talked to the operators themselves and recognize the way they appreciate this rather significant change.

Let's proceed.

MR. MURPHY: We found some of the plants have cross-ties between systems and in some cases between units of the same plant. At the Surry plant, for instance, the cross-ties were initially installed for Appendix R considerations for fire prevention to reduce the impact of fires, but they've had a

significant impact on reducing the core damage frequency just associated with internal events in that it significantly increases the redundancy available to the plant. And again, with the symptom based procedures, the operator has many more ways of getting water to the core. We give him credit for that.

We observe in passing that a poorly designed cross-tie can have a very negative effect. And I think this is obvious. It can defeat the redundancy that's intended. But it has had a significant effect and a larger effect than we anticipated as we went into the study.

Finally, we find that in some areas if steam is released into the reactor building during a severe accident it could effect the reliability of equipment located in open areas. But we have found in some of our plants the pumps are located in electrical rooms are sealed in the reactor building. Effectively, therefore, they can continue to work in a severe accident environment even though they weren't particularly designed with that thought in mind when they came up with the sealing. But this essentially obviates a problem that could be important in other plants. And we have found plants that have this feature and it's a very positive one and has a

significant effect on the core damage frequency.

That's a quick summary of the status we have in terms of our insights that we got out of the front end part of the study of the accident frequency analysis.

MR. STELLO: We're through, Mr. Chairman.

CHAIRMAN ZECH: All right. Thank you very much.

Questions from my fellow Commissioners? Commissioner Roberts?

COMMISSIONER ROBERTS: No.

CHAIRMAN ZECH: Commissioner Carr?

COMMISSIONER CARR: That last comment, I thought we EQ'ed equipment so that that steam wasn't going to be a bother. Are you trying to tell me now we got to go back and seal the rooms, too?

MR. MURPHY: No. What I'm saying is that in certain cases the environment that a piece of equipment can be exposed to in a severe accident may exceed the conditions for which the EQ tests were made. Where the rooms have been sealed, this has had a significant effect.

I'm talking about equipment now that is located outside the containment in the reactor building. In a severe accident --

COMMISSIONER CARR: I see.

CHAIRMAN ZECH: Balancing plant equipment?

MR. MURPHY: Yes. Balancing plant equipment.

MR. STELLO: It's not the safety grade equipment for which you have the specific EQ, but rather the balancing plant equipment which can be used to deal with challenges to the plant which do not have those stringent standards that you have on the EQ equipment. For example, inside containment. That EQ standard does not apply to that broad spectrum of equipment.

CHAIRMAN ZECH: Commissioner Rogers?

COMMISSIONER ROGERS: I just wanted to say that I was very pleased with the way the peer review panel has come out and the composition looks excellent. And the preliminary drafts, the charge to the panel looks like -- I want to compliment you on bringing that off because I know it took a great deal to do that. And it looked very good to me.

Just a question. This normally closed block valves on PWRs that related to feed and bleed, was there a PWR done for Waterford?

MR. MURPHY: Not -- I'm not aware of one.

MR. STELLO: No.

DOCTOR ROSS: We have plans as soon as we get through with this set, to do one or two CE plans. But we have --

COMMISSIONER ROGERS: I understand that's the only plant that doesn't have a feed and bleed capability?

DOCTOR ROSS: No. Palo Verde didn't.

COMMISSIONER ROGERS: Palo Verde.

MR. STELLO: CE plants do not have.

COMMISSIONER ROGERS: Okay. All right. Maybe it's just -- have any PRAs been done on those?

MR. STELLO: Palo Verde.

COMMISSIONER ROGERS: No CE plants?

MR. MURPHY: Not recently.

DOCTOR ROSS: Palo Verde has.

MR. STELLO: We don't have them.

DOCTOR ROSS: When the IPE gets done, in essence this level one stuff will be done by everybody. Help is on the way.

COMMISSIONER ROGERS: See how that comes out on that score.

CHAIRMAN ZECH: Have the utilities done PRAs on some of the CE plants? Anybody know?

DOCTOR ROSS: Millstone.

MR. MURPHY: I believe the Millstone -- yes.

CHAIRMAN ZECH: We haven't reviewed it, though, is that what you're telling us?

Does anybody know that's here that can help us out? Yes, please step forward and identify yourself for the reporter.

MR. SHERON: Brian Sheron, in The Office of Research.

Back in 1974 the issue of PORVs for the CE plants came up. There are six CE plants that do not have PORVs. Waterford, Palo Verde I, II and III and--let's see. I think that there's two in California. I can't remember the name. Songs, that's right. San Onofre II and III.

CHAIRMAN ZECH: San Onofre. All right.

MR. SHERON: We did a very extensive study, PRA type of study. It was not a full scope PRA. But we looked at the risks associated with not having PORVs on these plants. We looked at the benefits and the detriments. We did cost benefit analyses.

The results -- there was also a study, a similar study done by the industry. The results were that the staff found that there was a small but positive benefit to putting PORVs on those plants. The industry concluded that there was a small negative benefit. And it came about, as usual, I think on

assumptions about operator action and inaction.

The issue was remanded to generic issue 84 pending resolution of USI A-45. As you know, we subsumed A-45, which was decay removal systems, into the IPE. And right now the RES staff is examining generic issue 84 and determining how we want to proceed on it at this time.

But the issue was examined fairly thoroughly and I can tell you right now that as I understand it, Palo Verde actually had PORVs on site and ready to be installed.

CHAIRMAN ZECH: All right. Thank you very much.

MR. STELLO: Do you happen to know if there are any plants that have a PRA that they submitted? That was the question. Does anybody know?

MR. SHARON: We'll find out and get back to the Commission.

MR. MURPHY: One I would mention is that the NRC sponsored in the late "70s under the IREP study a PRA on Calvert Cliffs. That was the only one I'm aware of.

CHAIRMAN ZECH: And you'll get back to the Commission --

MR. STELLO: We will.

CHAIRMAN ZECH: -- if you would, please. All right.

Commissioner Rogers, anything else?

COMMISSIONER ROGERS: No, that's fine. Thank you.

CHAIRMAN ZECH: Commissioner Curtiss?

COMMISSIONER CURTISS: (No response.)

CHAIRMAN ZECH: As you go along here with the NUREG-1150 study, are some of the insights you're getting being fed back to the designers as well as to the utilities so they'll know what we're doing as we go along?

MR. MURPHY: The main insights that we've developed we've had a constant process of communication with the people who are developing guidance for the IPEs. And so that guidance going forward and to how to do IPEs contains much of what we've found in 1150. Almost on a daily basis.

CHAIRMAN ZECH: Is that getting to the designers, the suppliers, the big companies?

MR. BECKFORD: The documentation was provided to all of the operators and I believe that the manufacturers, the RMs have all seen this.

CHAIRMAN ZECH: Doctor Ross, you have a comment?

DOCTOR ROSS: Yes. Just last Friday I had reason to meet with General Electric and went over what we've been talking about this afternoon and discussed not only some of the insights for their product lines, but also the methodology that they will probably be using as they move forward in responding as a vendor in the IPE resolution.

CHAIRMAN ZECH: Good. Well, I think it's important --

MR. STELLO: I'm sure the utilities, as I've already said, most of them have made significant changes to the core melt frequency by changes that they made to the plants while the process is going on. They're completely familiar with it.

CHAIRMAN ZECH: I think it's important that we keep, you know, the suppliers, the designers informed of what we're doing. And I presume some of the utilities are doing that, but I would ask you to kind of follow through on that to make sure that that's happening.

MR. STELLO: Okay.

CHAIRMAN ZECH: Some of the slides you showed were pretty obvious descriptions, really, of the difference in core melt between the BWRs and the PWRs. And, you know, we've heard you tell us that

before. And could you just briefly discuss that particular situation as you've seen it to date on these five plants?

The slides show what you found, but is there anything more that you could add to what we've seen in these slides?

MR. MURPHY: The main situation we have found in comparing these five plants for the BWRs versus the PWRs is that there are more ways of adding water to the core of a boiling water reactor. Some of these ways are not safety grade ways, but there are more ways and what the symptom based procedures do is they lead you to these other ways to adding water if you can't add using the primary means of getting water in. That seems to have had a big effect.

I think I also have to say that it's obvious also, I think, that a poorly designed BWR could have a core damage frequency higher than that of a well designed PWR if the problem design was in the balance of plant. It was hard to make a generality about one class of plants versus the other. But looking at well designed versions of both plants, there are more ways of getting water into the core of the boiling water reactors.

MR. STELLO: I would hasten to add one

point. We're looking at a picture of core valve frequency. That's only one ingredient to overall risk, the consequences. And I think as you start to look at overall consequences, for which this is one ingredient, that tends to start to equilibrate.

CHAIRMAN ZECH: Well, I think we understand that.

Have you looked at anything in the same light, have you seen or come up with any ways other than the ones you've talked here about, station blackout and other things, that would reduce the magnitude of PWR core melt damage?

MR. MURPHY: I think it would be plant specific, but obviously on Zion the utility is already taking steps to reduce. Surry has two gas turbines on site, large gas turbines. But at present they do not have the procedures or all the equipment necessary to start those from the control room in a station blackout situation. Using those could eventually provide a 40 or 50 megawatt source of electricity on site that would essentially drive station blackout away.

Other plants where we have a rapid decrease of the station batteries, depletion of the station batteries in a blackout situation, here there's -- you

know, I hate to do instant designing on the spot, but looking at the low chinning routine, you have unessential loads on the batteries in such a situation that could be shed. It is something that would be worth doing. And if the situation was bad enough, bringing in a small generator. You know, if we can generate DC power, it would be relatively simple.

In fact, there's a number of things like this that where we use the insights we got from 1150 to prepare the table that was attached to the accident management paper we sent forward.

I'm having trouble remembering the specifics of any other specifics of that, but many of the things on that table derive exactly from looking at these five studies.

CHAIRMAN ZECH: Okay. Fine. Well, one of the things that, you know, I think is pretty clear, at least it is to me I think from the your slides here in this presentation what I understand of the accident situation in general, is that looking at the big picture what you really need if you have a problem is some kind of power and you need water. You have electricity available or power available and if you have water available, you know, it's a huge step in the right direction.

So I agree with you when you say some of the utilities, perhaps, have gas turbines, maybe not even safety graded, but from a practical standpoint I think if I were a utility executive that's what I would be thinking about no matter what NRC said to give me a comfort factor, that I would have power and I would have water if I had a real problem.

And clearly I read in some of the things you're telling us here, as I say to perhaps over simplify some of the technical analysis you've done, but if you have power and if you have water, you're a big step in the right direction, in my judgment.

I'd like, Mr. Stello, perhaps just one final question, to ask you, do you think there are any potential problems for the licensees if we go ahead and give them this second draft of NUREG-1150 to allow them to use it in individual plant examination process while we are still undergoing the final peer review and with the understanding, of course, that we will eventually put out a final report? So in the meantime they'll be using a report that is not final, but I know it's had a considerable amount of review up until now. Could you comment on that just briefly?

MR. STELLO: Mr. Chairman, I feel fairly strongly on this point because I think we have

information about the safety of their plant that in our judgment is the best we know how to do. I think that to deprive these utilities from that information would just be unfair and the wrong thing to do for safety. I think they ought to have it and what they will do with it, they will do.

I think the question of suggesting that we ought to require anything from it is a different matter. But in terms of providing the kind of understanding and insights, I think it's extremely important that they have it.

I would go one step further and I think the information is to the best of our ability to glean from the development of research over the past nearly, I guess, 15 years what all this means and how it comes together, I would go for an even more general availability of the information. I think the proposed interim uses that we have in here encompass doing what I just said.

CHAIRMAN ZECH: Well, at this stage I'll still want to review, you know, the final product when it comes, but at this stage I would agree with you. It seems to me we have an obligation if we have insights that might be valuable to the utilities and it would appear that we might have, as even though we

have not finally produced the product that we essentially, I believe, have somewhat of an obligation to let them know of our concerns at the moment, even though it's not the final decision.

It seems to me that some of the larger issues will clearly be accepted even in the final product. And I don't have too many reservations that it's going to have significant changes, but I think we should recognize and the utilities should recognize that they are not dealing with the final product. And I'm sure you'll make that well known to them. I think they're prepared for that right now.

But I do think I would agree with you, that it seems to me that that's the right approach to take, all recognizing that we have not put out the final product but we've gone so far in this process. So far we've involved the utilities in a IPE program that I believe these insights that we have developed should be made available to the utilities.

Well, let me just say --

MR. STELLO: Mr. Chairman?

CHAIRMAN ZECH: Yes.

MR. STELLO: I should add that remember or recall that I suggested and asked the ACRS that they ought to speak to this issue so you'd have a better

fix.

CHAIRMAN ZECH: Yes. We understand that and we look forward to hearing from the ACRS because we know that they have had other views on it. and we're mindful of that and respectful of that, but we also want, you know, to hear their views before we make a final decision. That's why I qualified my views somewhat. But it does seem to me that there is value, there is merit and obligation, in a sense, to make known what we have learned so far.

But we do want to hear from the ACRS and we'll do that.

Well, let me thank you for the briefing today. I think it's been very helpful. We know it's only part of the 1150. We just talked the internal events today. I think the Commission will look forward to hearing the results of the remaining portions of the analysis being developed, especially those pertaining to the external events. And we'll look forward to hearing that in about a month, I believe. Is that correct?

MR. STELLO: Yes, sir.

CHAIRMAN ZECH: So we know you have more work to do there. And we want you to do this job right. It's awfully important. This is just an

important initiative.

As we know, the Rasmussen report was done some years ago and is still looked on as a major work. This essentially is what you're doing here, I know, in trying to update that with the technology of the '80s and '90s. And so it is a significant undertaking and so it is important to do it right. And so we don't want to rush you or the peer group to the point that you may want to do some modifications. But I do think it's important that we get out what we can and allow the licensees the chance to at least see what we've learned up to this time.

What I'd like to do, a final request of the staff, is when we get the second draft or the final draft of the NUREG-1150 that you're going to put out to the utilities for their use, that we get a copy of that, you know, several weeks at least before we have any Commission meeting on it so we can study it very carefully.

MR. STELLO: We would intend to provide it to the Commission and I would not propose to release it generally until after the Commission meeting.

CHAIRMAN ZECH: Well, I think all of us want to look at it and we need some time, I think.

MR. STELLO: Okay.

CHAIRMAN ZECH: Because it's such a significant document, I would suggest that we have at least two weeks so that we can study it carefully and perhaps make some input that we may want to make.

You want another comment?

MR. STELLO: Well, I have a feeling that my schedule has just been accelerated about another week.

CHAIRMAN ZECH: Well, it probably has. I think that's probably right. But I think you can handle that.

MR. STELLO: I don't think Joe Murphy can, though.

CHAIRMAN ZECH: Yes, he's right here.

MR. STELLO: I don't have any problem with it, but I don't think he can.

CHAIRMAN ZECH: I haven't heard any objections from Mr. Murphy, so --

MR. MURPHY: The meeting is scheduled in May.

CHAIRMAN ZECH: -- I'm sure he can handle it, too. All right.

MR. STELLO: We'll try.

CHAIRMAN ZECH: Do the best you can.

MR. STELLO: Yes, sir.

MR. MURPHY: We had set the 17th of April,

Mr. Chairman.

MR. STELLO: To me.

CHAIRMAN ZECH: Well, we need it two weeks before that, so --

MR. STELLO: What he's basically saying is I don't get to take it home that night and stay up as long as necessary to decide if there's a problem or not. I get one day.

Okay. We will do our best, sir.

CHAIRMAN ZECH: That's all I want to know.

MR. STELLO: Okay.

CHAIRMAN ZECH: We won't ask you to do any more than that.

Do any of my fellow Commissioners have any comments or anything else?

MR. STELLO: We just lost a week is what happened.

CHAIRMAN ZECH: All right. Well, you can handle that.

MR. STELLO: Okay, sir.

CHAIRMAN ZECH: Thank you very much.

We stand adjourned.

(Whereupon, the briefing was adjourned at 3:10 p.m.)

<u>CERTIFICATE OF TRANSCRIBER</u>

This is to certify that the attached events of a meeting of the United States Nuclear Regulatory Commission entitled:

TITLE OF MEETING: PRELIMINARY BRIEFING ON THE STATUS OF NUREG-1150

PLACE OF MEETING: ROCKVILLE, MARYLAND

DATE OF MEETING: MARCH 15, 1989

were transcribed by me. I further certify that said transcription is accurate and complete, to the best of my ability, and that the transcript is a true and accurate record of the foregoing events.

Judy Hadley

Reporter's name: JUDY HADLEY

NUREG-1150

SEVERE ACCIDENT RISKS: AN ASSESSMENT FOR FIVE U.S. NUCLEAR POWER PLANTS

STATUS REPORT AND INTERIM RESULTS

MARCH 15, 1989

TOPICS FOR DISCUSSION

- MAJOR AREAS OF IMPROVEMENT
  - ACCIDENT FREQUENCY ANALYSIS
  - ACCIDENT PHENOMENOLOGY AND RISK
- CURRENT STATUS
- ACCIDENT FREQUENCY RESULTS TO DATE
  (INTERNAL EVENTS ONLY)
- PERSPECTIVES GAINED

MAJOR IMPROVEMENTS
(ACCIDENT FREQUENCY ANALYSIS)

- UPDATED ACCIDENT FREQUENCY ANALYSIS REFLECTING CHANGES IN PLANT DESIGN AND OPERATIONAL PRACTICES SINCE DRAFT

- IMPROVED UNCERTAINTY ANALYSIS USING EXPERT JUDGEMENT, AS RECOMMENDED BY PEER PANELS.

- IMPROVED ANALYSIS OF PLANT-SPECIFIC FAILURE DATA

- IMPROVED DOCUMENTATION

IMPROVEMENTS (CONT.)

(ANALYSIS IN PROGRESS)

- ACCIDENT PHENOMENOLOGY AND RISK ANALYSES
  - COMPLETELY REDONE
  - INCREASED NUMBER OF SOURCE TERM CODE PACKAGE RUNS
  - SUBSTANTIAL SUPPORTING CALCULATIONS AND ANALYSES
  - IMPROVED CONSEQUENCE ANALYSIS
  - VASTLY IMPROVED EXPERT ELICITATION

- CONSIDERATION OF EXTERNAL EVENTS FOR TWO PLANTS

NEW RESEARCH AVAILABLE

(POST DRAFT NUREG-1150)

- IN-VESSEL ANALYSES WITH DETAILED CODES (MELPROG/TRAC, SCDAP/RELAP5, MELCOR, BWRSAR)
- BWR SEVERE DAMAGE TEST (DF-4) IN ACRR
- FULL LENGTH COOLANT BOILDOWN TESTS (FLHT-4 & 5) IN NRU
- HYDROGEN DDT AND HIGH TEMPERATURE DETONATION LIMITS.

RESEARCH AVAILABLE (CONT.)

- DCH TESTS (DCH 2&3) IN LARGE SURTSEY FACILITY.
- SMALL SCALE CAVITY DISPERSAL TESTS AT BNL.
- CORE-CONCRETE TESTS WITH SUSTAINED HEATING (SURC 3&4)
- TESTS ON BWR MARK-I MELT SPREADING AND LINER FAILURE.

CURRENT STATUS

- CONTRACTOR ANALYSES ESSENTIALLY COMPLETE
- FINAL QA/QC OF CONTRACTOR EFFORTS IN PROGRESS
- STAFF EVALUATING RESULTS AND PREPARING NUREG-1150
- COMMITTED TO TIMELY COMPLETION!

CONSIDERATIONS

- SINGLE VALUE OF RISK RARELY SOLE BASIS FOR REGULATORY DECISION
- ACCIDENT MANAGEMENT PROCEDURES IN PLACE CAN SIGNIFICANTLY REDUCE RISK
- THERE ARE SUBSTANTIAL DIFFERENCES AMONG PLANTS. RESULTS ARE HIGHLY PLANT-SPECIFIC.
- THE NUREG-1150 ANALYSES DO NOT REFLECT MANAGEMENT INFLUENCES, EXCEPT THOSE REFLECTED IN PLANT-SPECIFIC FAILURE DATA.

NATURE OF RESULTS

- RESULTS FOR THIS BRIEFING LIMITED TO SEVERE CORE DAMAGE FREQUENCY FROM INTERNAL EVENTS.

- RESULTS PRESENTED AS STATISTICAL DISTRIBUTIONS - 5TH AND 95TH PERCENTILES, MEDIAN, AND MEAN IDENTIFIED

- RELATIVE CONTRIBUTIONS OF VARIOUS INITIATING EVENTS TO MEAN SEVERE CORE DAMAGE FREQUENCY PRESENTED.

INTERNAL CORE DAMAGE FREQUENCY DISTRIBUTIONS

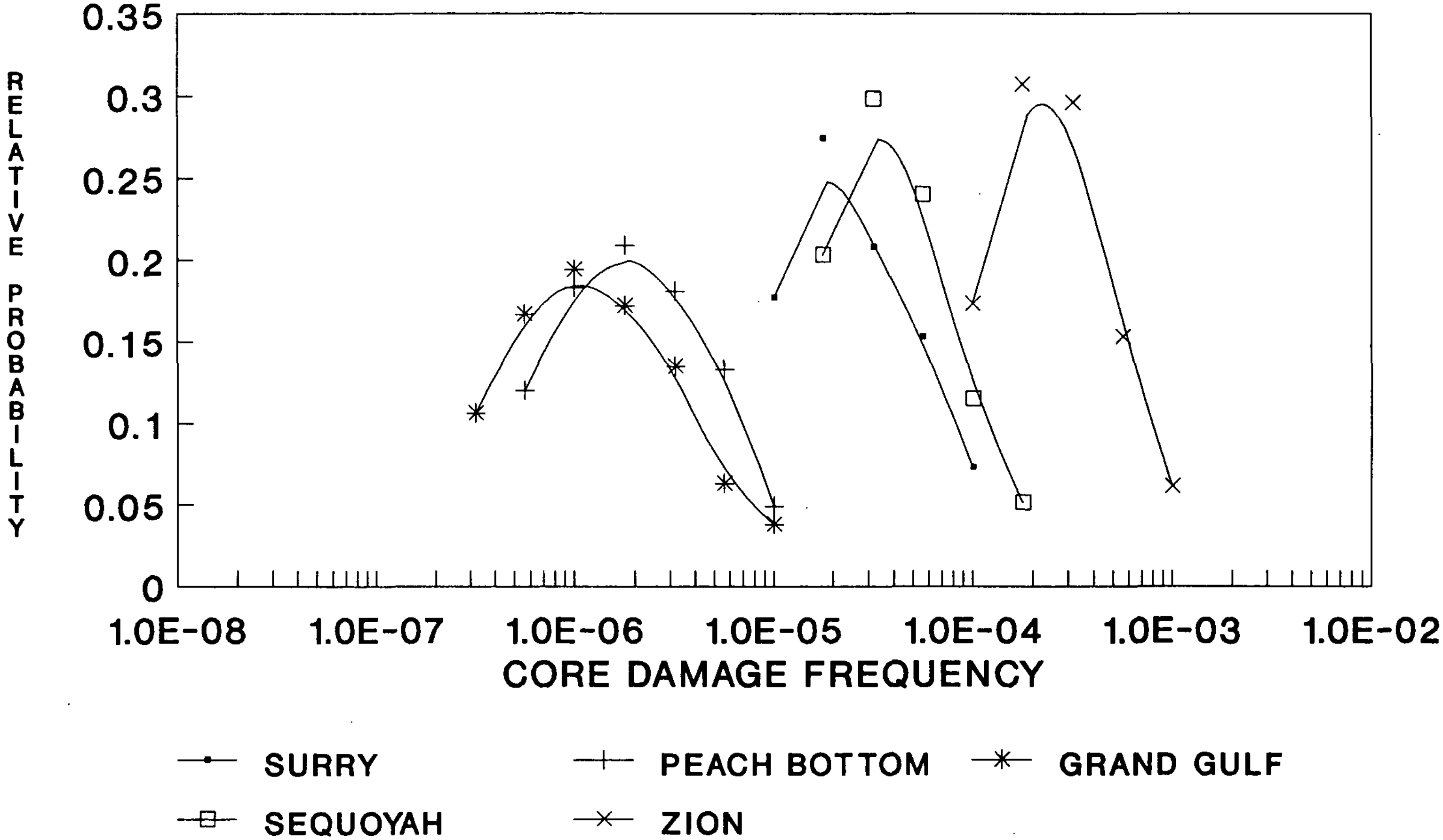

FIGURE 1

INTERNAL CORE DAMAGE FREQUENCY RANGES
5th and 95th Percentile

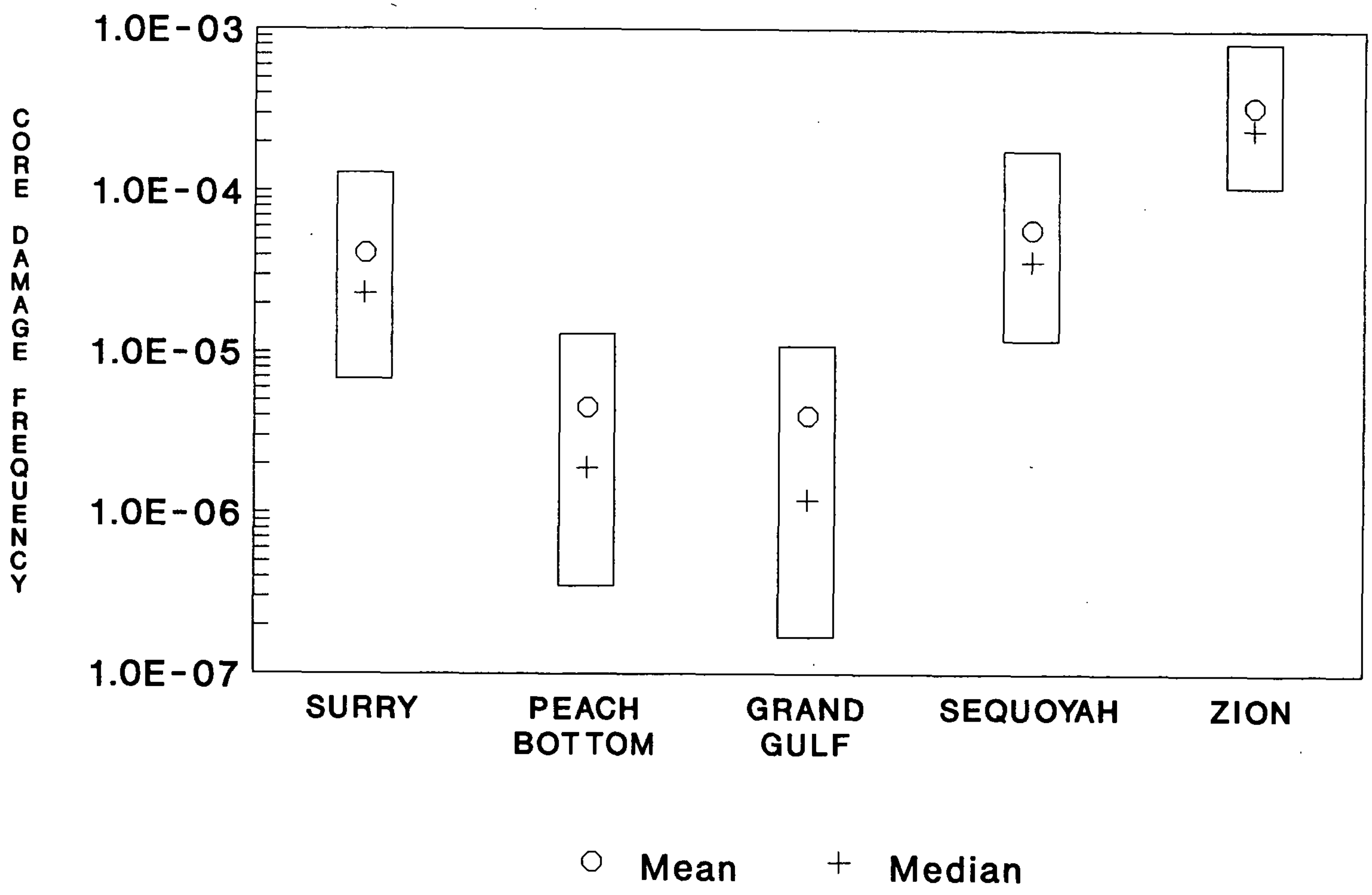

FIGURE 2

BWR PRINCIPAL CONTRIBUTORS TO INTERNAL CORE DAMAGE FREQUENCY

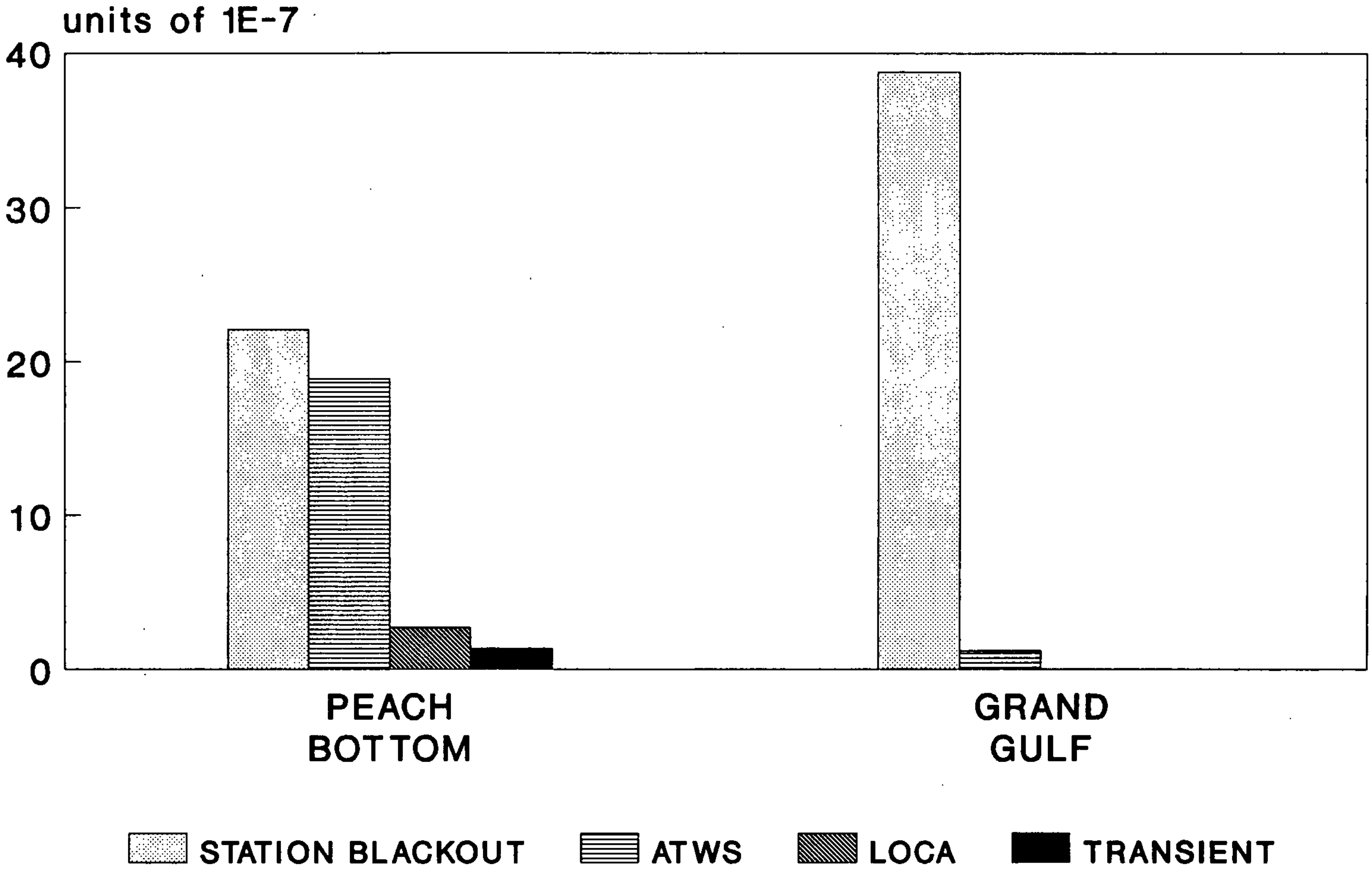

FIGURE 3

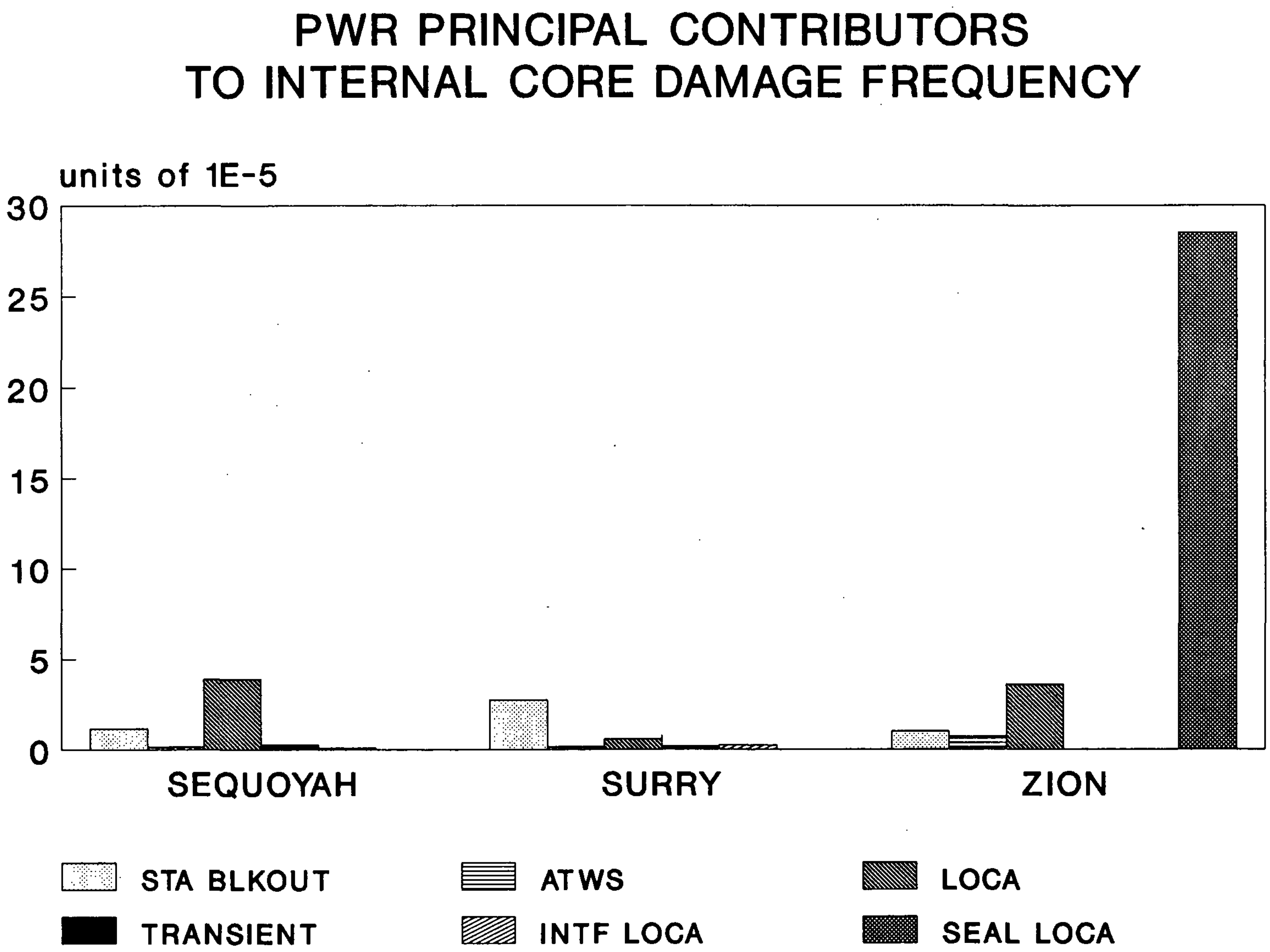
PWR PRINCIPAL CONTRIBUTORS
TO INTERNAL CORE DAMAGE FREQUENCY
units of 1E-5
30
25
20
15
10
5
0
SEQUOYAH
SURRY
ZION
STA BLKOUT
ATWS
LOCA
TRANSIENT
INTF LOCA
SEAL LOCA

FIGURE 4

INSIGHTS AND PERSPECTIVES

- NUREG-1150 ADDS TO THE COMPENDIUM OF PRA-BASED INFORMATION. INSIGHTS GAINED SHOULD BE TESTED AGAINST THAT DATA BASE OF RESULTS.

- BWRs STUDIED
  - THE TWO BWRs STUDIED ARE DOMINATED BY STATION BLACKOUT AND ATWS SEQUENCES, BUT THE ABSOLUTE VALUES ARE QUITE SMALL.

  - PEACH BOTTOM BENEFITTED FROM HIGHLY RELIABLE DIESEL GENERATORS.

INSIGHTS (CONT.)

- BWR INSIGHTS (CONT.)
  - THE INDEPENDENT HPCS DIESEL AT GRAND GULF PLAYS A KEY ROLE IN RESPONSE TO LOSS OF OFFSITE POWER.
  - BATTERY DEPLETION IS AN IMPORTANT CONSIDERATION IN A LONG-TERM LOSS OF ALL AC POWER.
  - FAILURES IN THE EMERGENCY SERVICE WATER SYSTEM CAN BE SIGNIFICANT AT PEACH BOTTOM. DEGRADATION OF THIS SYSTEM COULD MARKEDLY INCREASE CORE DAMAGE FREQUENCY.

INSIGHTS (CONT.)

- PWR INSIGHTS
  - THE ACCIDENT FREQUENCY AT ZION IS HIGHLY DEPENDENT ON THE AVAILABILITY OF THE COMPONENT COOLING WATER SYSTEM.
  - CORE DAMAGE FREQUENCY AT THE THREE PWRs TENDS TO BE DOMINATED BY STATION BLACKOUT AND LOCAs.
  - NORMALLY CLOSED BLOCK VALVES CAN REDUCE THE RELIABILITY OF ALTERNATE COOLING MODES.
  - STATION BLACKOUT IS SIGNIFICANT AT SURRY BECAUSE THERE ARE ONLY 3 DGs FOR 2 UNITS.

INSIGHTS (CONT.)

- PWR (CONT.)
  - THE EARLY INITIATION OF CONTAINMENT SPRAYS IN AN ICE CONDENSER CONTAINMENT CAN LEAD TO EARLY DEPLETION OF THE RWST, REQUIRING EARLY TRANSFER OF ECCS TO THE RECIRCULATION MODE. HUMAN ERRORS ASSOCIATED WITH THE TRANSFER CAN CAUSE LOCAs TO BE RELATIVELY MORE IMPORTANT THAN AT OTHER PWRs IF THE TRANSFER IS NOT AUTOMATED.

INSIGHTS (CONT.)

- GENERAL
  - OPERATOR RECOVERY ACTIONS DURING AN ACCIDENT HAD A SIGNIFICANT EFFECT ON CORE DAMAGE FREQUENCY (> FACTOR OF 10)
  - SYMPTOM-BASED PROCEDURES HAVE LED TO REDUCED ASSESSED HUMAN ERROR PROBABILITIES.
  - PROPERLY DESIGNED CROSS-TIES BETWEEN SYSTEMS OR BETWEEN UNITS CAN REDUCE THE CORE DAMAGE FREQUENCY SIGNIFICANTLY (ALTERNATELY, POORLY DESIGNED CROSS-TIES CAN HAVE A NEGATIVE EFFECT.)

INSIGHTS (CONT.)

- GENERAL (CONT.)
  - IF STEAM WERE RELEASED INTO THE REACTOR BUILDING DURING A SEVERE ACCIDENT, IT COULD AFFECT THE RELIABILITY OF EQUIPMENT LOCATED IN OPEN AREAS. SEALED PUMP AND ELECTRICAL ROOMS CAN OBVIATE THIS PROBLEM.

ZION PERSPECTIVES

- ■ IN THE DOMINANT SEQUENCE, LOSS OF COMPONENT COOLING WATER RESULTS IN FAILURE OF THE RCP SEAL (LEADING TO A SMALL LOCA), AND FAILURE OF THE HIGH HEAD SIS PUMPS (NO PUMP COOLING).
- □ CCW PIPE BREAK FREQUENCY IS HIGHLY UNCERTAIN.
- □ USE OF THE NEW WESTINGHOUSE RCP SEAL WOULD REDUCE THE SIGNIFICANCE OF THIS SEQUENCE.
- □ IF THE ZION HHSIS PUMPS DO NOT REQUIRE EXTERNAL COOLING, THE ACCIDENT FREQUENCY WOULD DROP SUBSTANTIALLY. (SEQUOYAH HAS DEMONSTRATED THAT SIMILAR PUMPS CAN OPERATE WITHOUT EXTERNAL COOLING FOR EXTENDED PERIODS OF TIME.)

February 17, 1989

POLICY ISSUE
(Information)

SECY-89-058

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: STATUS REPORT AND PRELIMINARY RESULTS OF NUREG-1150

Background: NUREG-1150 "Reactor Risk Reference Document," and its supporting contractor reports were published as drafts for comment in February 1987. Extensive public comments have been received. In addition, the draft documents were subjected to three independent peer reviews:

- A review of the uncertainty methodology by a committee chaired by Dr. H. Kouts of Brookhaven National Laboratory and sponsored by USNRC (NUREG/CR-5000),
- A comprehensive review of the entire report by a committee chaired by Prof. W. Kastenberg of UCLA and sponsored by USNRC (NUREG/CR-5113),
- A review of the report by a committee chaired by Dr. L. LeSage of Argonne National Laboratory and sponsored by the American Nuclear Society.

In addition, we have had informal interchanges with groups from the United Kingdom and the Federal Republic of Germany, and benefitted from an IAEA sponsored two-day workshop on NUREG-1150, held in Rome in April 1988.

We have now essentially completed the reanalysis to the extent where we can share a subset of the results with the Commission (that pertaining to accident frequencies from internal events), and discuss the possibilities for use of the information developed while NUREG-1150 is undergoing peer review. Final efforts are underway to complete and document the remaining portions of the analyses (those pertaining to external event accident frequencies and quantitative risk results) and to

Contact:
J. Murphy, RES
492-3510

present the perspectives gained from an examination of these results. We intend to transmit the remaining information to the Commission in mid-April 1989.

We have discussed our plans for further peer review of the completed work with the ACRS and the Commission (SECY-88-337) and are proceeding with such a review consistent with the Commission's February 9, 1989 memo to the EDO (Attachment 1).

Major Areas of Improvements: There have been several major changes to the report since the report was issued in draft form in 1987. The major areas are listed below:

- Expert elicitation is being used in NUREG-1150 to obtain subjective estimates of important physical parameters for which there are incomplete sources of information. A vastly improved elicitation process has been incorporated into the effort since the draft was published, using decision theoretic techniques, with the help of experienced decision analysts. The panels of experts were expanded considerably to reflect the judgement of experts not only from the national laboratories, but also from academia and industry, as suggested by the Kouts, Kastenberg, and LeSage Committees. Considerable analyses were performed using the best available codes and experimental information to support the expert elicitation process.

- The accident frequency analysis was modified to reflect the changes in plant design and operational practices which have occurred since the analysis was performed for the draft report. Sensitivity issues treated in the draft report have been incorporated into the uncertainty analysis. Further, where information on component performance under unusual conditions was made available by the licensee, due credit was given in the analysis.

- The analysis of containment loads, containment response, and source term estimations were completely redone. We have significantly increased the number of Source Term Code Package runs and, as previously noted, a significant number of analyses using codes such as MELCOR, MELPROG, and CONTAIN were performed by NRC contractors to support the elicitation process. Information obtained using the industry developed code MAPP was also made available by experts from industry.

- Up-to-date research results were utilized. In particular, technical reports and publications that appeared subsequent to performing analysis for the draft report were brought to the attention of all elicited experts. In some cases, results were recent enough that published reports were not available, and informal summaries were provided to the experts. Significant new results included in-vessel analyses with the newer more detailed codes (MELPROG/TRAC, SCDAP/RELAP5, and BWRSAR); the first severe damage test (DF-4) of a BWR fuel geometry (in the Sandia ACRR reactor); the first two full-length coolant boildown tests at sustained high temperatures (FLHT-4 and FLHT-5 in the Canadian NRU reactor); new evidence on hydrogen combustion, transition from deflagration to detonation, and detonability limits at high temperatures; the first direct containment heating tests (DCH-2 and DCH-3) in the new SURTSEY 1/10th-scale facility; small-scale tests using simulants to study debris retention in different cavities; initial testing (SURC-3 and SURC-4) on core-concrete interactions with sustained heating; and a variety of tests related to melt spreading in BWR Mark-I containments. With respect to these up-to-date research results, it must be kept in mind that to complete NUREG-1150 our knowledge base must be "frozen" at some point in time. We froze this base in March 1988.

- The consequence analyses has been redone using an improved code, MACCS 1.5, with sensitivity studies performed to examine different input assumptions, such as assumptions regarding the number of people evacuating, and protective action levels for crop interdiction and relocation of residents.

- The potential effects of external events, such as earthquakes and fires within the plant have been studied for Peach Bottom and Surry. Although the quality assurance checks on these analyses are not yet complete, it appears that these initiators may be major contributors to the core damage frequency at these plants. This may be due to the fact that these two plants have been extensively studied over the last 15 years by various PRAs and obvious contributors to the severe core damage frequency resulting from internal events may have already been rectified.

- A concerted effort is being made to improve the documentation of the work, making the analyses more scrutable and the results more traceable, in response to several comments received from the public, as well as from the three review committees. The distributions obtained

will be presented together with the means and medians of the distributions rather than presenting only the upper and lower bounds, as was done in the draft report.

Current Status: The technical analyses by our contractors are nearly complete (as of February 10, 1989) and are undergoing final review by the staff and by the analytical teams themselves. We are now evaluating the results and preparing the final documentation. We plan to have a near-camera-ready copy of the report available to the Commission prior to our next meeting in late April (to be scheduled).

Results to Date: NUREG-1150 estimates the risk associated with five nuclear power plants of differing designs: (1) Surry, a three-loop Westinghouse-designed PWR in a subatmospheric containment, (2) Sequoyah, a four-loop Westinghouse-designed PWR in an ice condenser containment, (3) Zion, a four-loop Westinghouse-designed PWR in a large, dry containment, (4) Peach Bottom, a General Electric-designed BWR in a Mark I containment, and (5) Grand Gulf, a General Electric-designed BWR in a Mark III containment. The analyses of the frequency of severe core damage accidents for four of the plants (Surry, Sequoyah, Peach Bottom, and Grand Gulf) were performed using the same methods and are essentially full Level 1 probabilistic risk assessments (PRAs). The severe core damage frequency for the fifth plant, Zion, was analyzed separately, in lesser depth, relying primarily on the utility-sponsored Zion Probabilistic Safety Study, published in 1981, the reviews of that document by the NRC staff and our contractors, with additional work done to reflect the current plant configuration. (The estimation of accident phenomenology, containment event trees, and risk analyses are consistent across all plants.)

In considering the information found in NUREG-1150, the following points must be kept in mind:

- Single value of risk results, in and of themselves, rarely would be sufficient to support a regulatory decision. There are a myriad of other factors that need to be considered, such as existing regulations, the nature and magnitude of the associated uncertainties, the needed margin of safety, and, if an acceptable level of safety has been attained, costs.

- Accident management techniques are important considerations. In general, they have been considered in NUREG-1150 when written procedures existed.

- There could be significant differences in risk among plants due to plant-specific design and procedure differences. Such factors must be carefully evaluated before using NUREG-1150 results for generic decisions or attempting to apply the results to other plants.

- NUREG-1150 does not reflect management influences, except as they affect component failure rate data. It does not present a complete and unchanging characterization of plant risk. Risk can be decreased in time by modifying systems and procedures and implementing accident management procedures. Similarly, degraded systems, poorly implemented procedures, inadequate maintenance, or ineffective management (to name a few) could drastically alter both the dominant contributors and the absolute value of risk at a plant.

Summary of Results

The severe core damage frequency as presented in NUREG-1150 is not a single value, but instead is expressed as a statistical distribution. Thus, it will be presented as a 5th percentile, a mean, median, and a 95th percentile value, statistics that describe the distribution. In addition, there is valuable information in the principal contributors which make up the estimated severe core damage frequency. The significance of these results in terms of perspectives on and insights into the frequency of core damage accidents will be discussed below.

The internal core damage frequency distributions are presented graphically in Figure 1. As noted above, the figure does not include the contributions of external events.

Figure 2 shows the ranges of the distributions in a different format. The lower and upper extremities of the bars in this figure represent the 5th and 95th percentiles of the distributions drawn in Figure 1. Thus, the bars include the central 90 percent of the distributions. (Note Figure 1 indicates that the distributions are not constant within these bars.) The mean and median of each distribution are also shown.

The contributors to the internal mean core damage frequencies are shown in Figures 3 and 4. The information presented on these figures is discussed in the Perspectives section presented below.

Perspectives: Internal Core Damage Frequency Distributions

Because of the large uncertainties associated with core damage frequencies, comparisons with limits or goals need not be simply a matter of comparing two numbers. One can also observe how much of the distribution lies below a given point, which translates into a measure of the probability that the point has

not been exceeded. For example, if the median were exactly equal to the point in question, half of the distribution would lie above and half below the point and there would be a 50% probability that the point had not been exceeded.

Similarly, when comparing severe core damage frequencies calculated for two or more plants, it is not sufficient to simply compare means. Instead, one must compare the entire distribution. If one plant's distribution were virtually entirely below that of another, then there would be a high probability that the first plant had a lower severe core damage frequency than the second. This is seldom the case, however. Usually, the distributions have considerable overlap and thus one can infer with some probability that one plant has a higher or lower severe core damage frequency than another.

In NUREG-1150, the degree of overlap of the core damage distributions is relatively high. As can readily be seen from Figure 1, there is a relatively high probability (but not certainty) that the severe core damage frequency for Grand Gulf is lower than that of Sequoyah or Surry. Conversely, it can readily be seen that the difference in severe core damage frequency between Surry and Sequoyah is not very significant.

The significance of a very low core damage frequency, such as that calculated for Grand Gulf, is also apparent when examining this figure. Although the peak of the distribution is in the 10^{-6} range, it is incomplete to simply state that the severe core damage frequency for this plant is that low, since the 95th percentile extends past 10^{-5}. Thus, although the central tendency of the calculations is very low, there is still a finite probability of a higher (or lower) severe core damage frequency.

Figure 2 is based upon the distributions of Figure 1, but gives a somewhat different perspective. Because these are 5-95 percentiles, there is a 90% probability that the severe core damage frequency for each plant is within the span indicated (although the density of the distribution is not constant within the bar). The median is defined as the midpoint of the distribution, i.e., half of the samples lie above this point, and half below. The mean is an arithmetic average calculated on a linear scale.

The asymmetry of the distribution causes the mean and the median to differ. Even though some of these distributions appear reasonably symmetric on a logarithmic scale, they are quite skewed on the linear scale upon which the mean is based. The effect becomes more pronounced for wider distributions since the mean, unlike the median, is quite sensitive to changes in the tails of the distributions.

Principal Contributors

Accident sequence perspectives are qualitative rather than quantitative. The various accident sequences which comprise the PRA can be grouped by common factors into categories. Older PRAs generally did this in terms of the initiating event, e.g., transient, small LOCA, large LOCA, etc. Current practice also uses categories which experience has shown to be of interest, such as "ATWS," "seal LOCA," "station blackout," etc. Generally, these categories are not equal contributors to the total core damage frequency. In practice, four or five sequence categories, sometimes fewer, usually contribute almost all of the core damage frequency. The group of sequence categories contributing most of the core damage frequency will for the purposes of NUREG-1150 be referred to collectively as the "principal contributors."

In some cases, one sequence category by itself will be found to contribute the majority (i.e. over 50%) of the core damage frequency. In such cases, this sequence category is said to be the "dominant" contributor and the core damage frequency is said to be "dominated" by this sequence category.

The significance of the principal contributor identification is that it provides physical insight into the core damage frequency. If the severe core damage frequency is to be changed, changing something common to the principal contributors will have the most effect. Thus, if a particular plant had a relatively high severe core damage frequency and a particular group of sequences were dominant, a valuable insight into that plant's safety profile would be obtained.

The existence of a highly dominant contributor does not of itself mean that a safety problem exists. For example, if a plant already had an extremely low estimated severe core damage frequency, the existence of a single dominant contributor would have little significance. Similarly, if a plant were modified such that the dominant contributor were eliminated entirely, the next highest sequence category may well by definition become the new dominant contributor.

Nevertheless, it is the study of the principal contributors that provides understanding of why the core damage frequency is high or low. This qualitative understanding of where the core damage frequency comes from and how it can be changed is necessary to make practical use of the core damage frequency distribution.

Given this background, the principal contributors for the five studies are illustrated in Figures 3 and 4. Several observations on these sequences and their effect on core damage frequency can be made.

BWR vs. PWR

It is evident from Figures 1 and 2 that the two BWRs in this study have core damage frequency distributions that are significantly lower than those of the three PWRs. An examination of the principal contributors gives part of the reason why. The LOCA sequences, often dominant in the PWR core damage frequencies, are barely visible in the case of the BWRs. This is not surprising in view of the fact that the BWRs have two low pressure ECC systems (low pressure core injection and low pressure core spray), each of which is multi-train, two high pressure injection systems (Reactor Core Isolation Cooling System and the High Pressure Coolant Injection/High Pressure Core Spray systems), still other high pressure injection capabilities (Control Rod Drive hydraulic system, service water, etc.) and a readily available method of rapid depressurization (Automatic Depressurization System). Moreover, LOCA events in a BWR can also be mitigated by the main feedwater system, which is both high pressure and high capacity.

In contrast, PWRs generally have only one high pressure and one low pressure ECC system (both multi-train), plus a set of accumulators. The PWR ECCS does have considerable redundancy, but not as much as that of a BWR.

Conversely, BWRs have historically been considered more subject to ATWS events. This is partly due to the fact that some ATWS events in a BWR involve a pressure surge which can cause a significant insertion of positive reactivity via the void coefficient, resulting in a rapid power surge. An ATWS event in a PWR is slower, giving more time for mitigative action.

In spite of this historical belief, it is evident from Figure 3 that the two BWRs are not dominated by ATWS events. Both of these plants have implemented ATWS modifications (pursuant to NRC rulemaking) and this has resulted in a lower contribution of ATWS sequences to the total core damage frequency.

For both BWRs and PWRs, the analysis indicates that the support systems are quite important. Because these systems are site specific, caution must be exercised when making statements about generic classes of plants such as PWR vs. BWR or extrapolating this discussion to all PWRs vs. all BWRs.

Station Blackout contributes a high percentage of the core damage frequency for the BWRs studied. However, when viewed on an absolute rather than relative scale station blackout is higher in absolute value at the PWRs than at the BWRs. This is because of the PWRs' different susceptibilities (particularly the seal LOCA) and the fact that unlike the BWRs which have at least one injection system which can work during a station blackout, PWRs cannot inject directly into the reactor coolant system during a blackout scenario.

BWR Observations

It is evident from Figure 3 that station blackout plays a major role in the core damage frequencies for the two BWRs. This is a case where the low total core damage frequencies for these two plants must be noted. Although station blackout meets the definition of dominant as used in this report, the reason for this is not that station blackout is a major problem for these two plants, but rather that other sequences are very low and station blackout is the contributor that is left.

Grand Gulf is equipped with an extra diesel generator dedicated to the high pressure core spray system, and is thus better equipped than many other plants to deal with losses of AC power. Peach Bottom is an older model BWR that does not have the extra diesel. However, the Peach Bottom diesels have an unusually good reliability record, historically. In addition, Peach Bottom is a two unit site with the appropriate cross-ties and four diesels, any one of which has sufficient capacity to power both units in the event of a loss of offsite power. Thus, Peach Bottom has still more redundancy than Grand Gulf in onsite AC power. However, the Peach Bottom diesels are susceptible to service water failures; in that if two particular diesels fail, the service water system will fail causing the other two diesels to fail. In addition, DC power is needed to start the diesels. (Some emergency diesel systems, such as those at Surry, have a separate dedicated DC power system just for starting purposes). This offsets some of the high redundancy and reliability at Peach Bottom.

Figure 3 indicates that, at Peach Bottom, station blackout with battery depletion accounts for about 40% of the core damage frequency. The likelihood of battery failure before 10 hours is judged to be approximately 0.5 (relatively good performance for station batteries). Thus, in combination with failure to recover AC power, battery life is a major factor for Peach Bottom.

Another support system of importance is Emergency Service Water. Emergency Service Water systems that are likely to contribute to the core damage frequency are those that lack sufficient redundancy or are susceptible to single passive component failures, and that affect multiple front line systems or other key support systems such as emergency power. Emergency Service Water at Peach Bottom appears in almost all of the accident sequences, where it either directly fails a system or indirectly causes failure through loss of diesel generator cooling.

Examination of Figure 3 also reveals that LOCA and transient sequences are significant (although minor) at Peach Bottom, but have essentially disappeared at Grand Gulf. This is primarily due to the fact that Grand Gulf is a BWR/6 design. Thus, Grand Gulf uses a motor driven High Pressure Core Spray System rather than a steam driven High Pressure Coolant Injection System for high pressure addition of coolant to the reactor vessel, which improves the reliability of this system for mitigation of transient and small LOCA events. In addition, Grand Gulf has a third train of low pressure coolant injection which injects directly into the reactor core volume, and thus has one more train than Peach Bottom for mitigation of large LOCAs.

Finally, the analysis has shown the importance of venting of the suppression pool or improving the reliability of the long-term decay heat removal function at a Mark I BWR. At Peach Bottom, the ability to vent steam from the suppression pool (by means of a 6 inch diameter pipe) during the course of an accident scenario prevents the reactor building from being flooded with steam from a rupture of the torus. Because of this, the various coolant injection systems are not rendered inoperable by the presence of a harsh steam environment, and also remain accessible to plant personnel. (At Grand Gulf, this does not apply, as its containment design is such that the suppression pool will not discharge into the building containing the safety systems.) Peach Bottom is equipped with a hard pipe for this venting. Other BWRs with a similar containment may use ducting for this purpose, which is far more likely to fail and flood the area with steam during venting.

PWR Observations

Among the PWRs, it is readily apparent from Figures 1 and 2 that Zion has a significantly higher core damage frequency distribution. Examination of Figure 4 reveals why Zion has an apparent susceptibility to loss of service water and loss of component cooling water that does not appear in the two other PWRs' principal contributors.

At Zion, it is assumed that component cooling water is needed for operation of the charging and high pressure safety injection pumps. (There is some controversy associated with this assumption. For Sequoyah, the licensee has provided evidence, based on an actual test, that the high head safety injection pumps can operate without external cooling for 24 hours.) Loss of component cooling water (or loss of service water, which will render component cooling water inoperative) will apparently result in loss of these high pressure systems. This translates into a loss of reactor coolant pump seal injection and a loss of the capability to inject high pressure makeup to the primary system. Simultaneously, loss of component cooling water will also mean loss of cooling to the reactor coolant pump seal thermal barrier heat exchangers. Thus, the reactor coolant pump shaft seals will lose both forms of cooling. Experience and expert judgement has indicated that there is a high probability of failure of the seals under these conditions with current seal materials. (New seals, currently being installed by utilities, will reduce the likelihood of significant leakage considerably, based on the opinion of the expert panel.) Thus, loss of component cooling water or service water can both cause a small LOCA and disable the systems needed to mitigate a small LOCA. The importance of this scenario is increased further by the fact that the component cooling water system at Zion, although it has appropriate redundancy of pumps and valves, nevertheless sends its flow through a common header.

With the exception of the reactor coolant pump seal LOCA contributions to the Zion core damage frequency, the makeup of the core damage frequencies at the three PWRs is reasonably similar. A comparison of the principal contributors to the mean severe core damage frequency for the three PWRs is presented on the bar graph in Figure 4. As can be readily observed, except for the large seal LOCA contribution at Zion, the core damage frequency at the three PWRs is governed by station blackout and the various LOCA sequences, with some contribution from ATWS. The danger of extrapolating conclusions to other plants (say, from Surry and Sequoyah to Zion) is also evident.

The importance of keeping PORVs and ADVs unblocked became evident during the analysis. Unblocking these valves is a manual operation, which must be done under stressful conditions and, in any event, is prevented by loss of power or hardware failures in certain scenarios. This reduces the availability of decay heat removal via feed and bleed cooling or secondary system blowdown. Also, in an accident scenario where pressure is rising, if PORVs and/or ADVs are blocked,

pressure will continue to rise until the safety valves lift. The higher pressure can make management of the accident more difficult; and in addition, there is a high probability of safety valve leakage after actuation.

Similarly, loss of some power (possibly only one bus) or other support systems can fail PORVs or ADVs or their block valves at some plants, precluding the use of feed and bleed or secondary system blowdown. At Surry, the ADVs are not Class 1E powered. Thus, given station blackout, the ADVs are inoperable, the secondary side code safety valves open, and the pressurizer PORVs are demanded and thus open, with a chance to stick open. A station blackout with a stuck open PORV can lead to core damage, if coupled with other failures.

In contrast to the two BWRs, which were governed almost exclusively by station blackout, the contribution of station blackout is very significant at Surry, moderately so at Sequoyah, and relatively minor at Zion. However, when viewed on an absolute scale, the mean core damage frequency from station blackout is in the 10^{-5} range at all three PWRs. (This illustrates once again how the principal contributors should not be examined without examining the mean core damage frequency as well.) The relatively high importance of station blackout scenarios at Surry is due to the fact that this is a two unit site with two dedicated diesel generators (one for each reactor) and one swing diesel generator shared between the two units. Since a loss of offsite power generally means that both units need their diesels, two of the three diesels must function. The lower redundancy of onsite AC power at this site increases the importance of station blackout.

For the LOCA scenarios (seal LOCA excluded), Sequoyah has the greater contribution. This is because of the ice condenser containment at Sequoyah. In this containment design, the containment sprays are automatically actuated at a lower pressure setpoint than at a corresponding large dry containment design, resulting in containment spray actuation over a larger break size spectrum. This causes faster depletion of the refueling water storage tank and an earlier switch to recirculation mode. The inclusion of this switchover, which at Sequoyah is a complex manual operation which must be done rather quickly and under stressful conditions, significantly increases the importance of the LOCA sequences. (Many later designs have incorporated circuitry for automatic switchover to the recirculation mode.)

General

It should be noted that changes in plant hardware or procedures can change the absolute contribution of an individual sequence category by a factor of 10 or more, greatly affecting the percentage contribution of that sequence category to the total core damage frequency. Yet, because the total severe core damage frequency is a sum of all the sequence categories, changing the contribution of any one sequence category will not necessarily greatly change the total severe core damage frequency unless that sequence category by itself is highly dominant. Moreover, total severe core damage frequency is usually observed on a logarithmic scale, which further reduces the apparent change. Thus, the makeup of the charts of Figures 3 and 4 can generally be expected to be more sensitive to plant changes than is the case for the total core damage frequency.

There are several general observations that can be made on the analysis:

Operator recovery during an accident sequence can have a significant effect on the potential core damage frequency. Each combination of faults identified that could lead to core damage is examined in a recovery analysis, since recovery actions are often very dependent upon the specific combination of failures involved. Recovery may be as easy as manual opening of a MOV when the motor fails or finding an alternate flow path. The key is having sufficient time and written procedures to support the recovery action, and there is significant variability in the quality of the procedures and training from plant to plant.

Symptom based procedures are superior to event oriented procedures in that prompt diagnosis of the specific accident sequence is not required and the possibility of incorrect action being taken is therefore reduced. Symptom based procedures will generally provide a faster response to the accident, in terms of correcting the actual functional problems.

Properly designed cross-ties between systems can significantly decrease the core damage frequency. Cross-ties allow for more options to work around failures, both for flow paths and for electrical power. When core cooling or heat removal is lost, alternate paths to combine those components that are functioning can be very beneficial. Similarly, bus failures or diesel generator failures can be circumvented by appropriate cross connecting. Since there is a potential for incorrect cross connecting, proper administrative control is very important.

Venting or steam release into the reactor building in a BWR may have serious effects on safety systems, and thus injection sources located outside containment have a significant advantage. Often pumps, motor control centers, and cables are located in open areas susceptible to harsh environments. Humidity and condensation are significant threats to electrical equipment that can be avoided by sealed pump rooms and electrical equipment rooms.

Injection pumps that can operate after containment failure will reduce the core damage frequency. If containment failure occurs before core damage, continued injection and long term heat removal could prevent core damage or lessen the severity. Given core damage, continued injection could mitigate the accident by reducing the fission product inventory that might be released and decreasing possible core concrete interactions.

Comparison with the Reactor Safety Study

Figures 5 and 6 show the internal core damage frequency distributions calculated in this present study for Surry and Peach Bottom along with distributions synthesized from the Reactor Safety Study. (The Reactor Safety Study presented results in terms of only medians, not means.) It can be seen that the median core damage frequencies are lower in the present work, although observation of the overlap of the ranges show that the change is more significant for Peach Bottom than for Surry.

There are two important reasons for the differences between the new figures and those of the Reactor Safety Study. The first is the fact that PRAs are snapshots in time. (In these cases, the snapshots are taken about 15 years apart.) Both plants have implemented hardware modifications and procedural improvements with the stated purpose of increasing safety, which drives core damage frequencies downward.

The second reason is that the state of the art in probabilistic analysis has advanced significantly since the Reactor Safety Study was done. Computational techniques are now more sophisticated, computing power has increased enormously, and consequently the level of detail in modelling has increased. In some cases, these new methods have reduced or eliminated previous conservatisms. However, new types of failures have also been discovered. For example, the years of experience with probabilistic analyses and plant operation have uncovered the reactor coolant pump seal failure scenario as well as intersystem dependencies, common mode failure mechanisms, and other items which were less well recognized at the time of the Reactor Safety Study. Of course, this same experience has also

uncovered new ways in which recovery can be achieved during the course of a possible core damage scenario. Thus, the net effect of the inclusion of these new techniques and experience is plant specific and can drive core damage frequencies in either direction.

In the case of the Surry analysis, the LOCA-induced core damage frequency for the present study is significantly reduced from that of the Reactor Safety Study, particularly for the small LOCA events. This occurred in spite of a tenfold increase in the small LOCA initiating event frequency. The reason for the reduction lies in plant modifications since the Reactor Safety Study was done. These modifications allow for the cross-tie of the high pressure safety injection systems, auxiliary feedwater systems, and refueling water storage tanks between units. These cross-ties provide a reliable means of recovering from system failures. Thus, a plant modification (the cross-ties) has driven core damage frequencies downward, but new PRA information (the higher small LOCA frequency) has driven them upward. In this case, the net effect was an overall reduction in core damage frequency. (It is worth noting that a poorly engineered cross-tie could defeat redundancy and have a correspondingly deleterious effect.)

In the case of Peach Bottom, the Reactor Safety Study found the core damage frequency to be comprised primarily of ATWS and of transients with failure of long term decay heat removal. The present study concludes that station blackout scenarios are dominant. The possibility of successful containment venting, and realistically allowing for successful core cooling after containment failure, have considerably reduced the significance of the loss of long term decay heat removal accidents. In addition, the plant has implemented some ATWS fixes, although ATWS events still appear as one of the principal contributors. Thus, for this plant's analysis, advances in PRA methodology and plant modifications have both contributed to a reduction in the estimated core damage frequency.

In summary, there have been reductions in the core damage frequencies for both plants since the Reactor Safety Study. The drop in the assessed severe core damage frequency for Peach Bottom is more significant when viewed in terms of the range. (Even here, there is still considerable overlap of ranges.) The conclusion to be drawn is that the hardware and procedural changes made since the Reactor Safety Study appear to have reduced the severe core damage frequency at these two plants, even when accounting for additional failure data and sequences.

Intended Uses: The intended use of NUREG-1150 was presented to the Commission in SECY-88-147, Integration Plan for Closure of Severe Accident Issues, and is summarized below:

- The probabilistic models of the spectrum of possible accident sequences, containment events, and offsite consequences have been and are being used to develop guidance for the review and conduct of the search for vulnerabilities through the individual plant examinations and the development of the framework for considering accident management strategies.

- The analytical information base obtained from NUREG-1150, including the analyses performed to assist the experts in the elicitation process will be a useful input to considerations of the need for improvement of containment performance under severe accident conditions. They have added significantly to the analytical data base.

- The NUREG-1150 analyses will add to the compendium of PRA information on the frequency of severe accidents and the dominant accident contributors which can be used to assist in identifying plant operational features or practices which have an adverse impact on plant safety.

- The models of NUREG-1150 will provide a testbed for the evaluation of alternative safety goal implementation strategies at five plants of differing designs and evaluation of the risk reduction benefit of various accident management options.

- In addition, for the plants analyzed, NUREG-1150 will identify the major contributing factors to core damage frequency, and the various measures of risk, and to the uncertainties associated with those estimates. These will form an important data base which can be used as one element in the evaluation of research priorities and the prioritization and resolution of generic issues.

The uses presented above do not rely heavily on the quantitative "bottom line" results of the risk calculations of NUREG-1150. Rather, they utilize the considerable information data base developed in the course of performing the study. Thus, we believe it would not be prudent for us to ignore the large numbers of analyses which supported the expert elicitation process or the understandings gained from evaluating the bases behind the expert judgement elicited on critical issues. Similarly, if items are found to be important

and potentially risk significant in the current version on NUREG-1150, we feel strongly that we should evaluate these from a regulatory standpoint. Consistent with the advice in the February 9, 1989 SRM, we will inform the peer review committee of our intended uses of NUREG-1150 work.

We note that the February 9, 1989 SRM also indicated that the Commission be informed on the membership, charter, and other aspects of the peer review committee. When available, this information will be provided to the Commission in a separate information paper.

Section 5 of this paper has presented some of the insights and perspectives gained in conducting the accident frequency portion of the analysis. They are not highly dependent on precise quantitative values, but rather present the broader knowledge gained from an integrated examination of the operation of the five reactors considering the complex interactions between systems under a variety of challenges.

This usage of NUREG-1150 is a matter of debate with the ACRS. We will meet with them and report back to the Commission in our next meeting on NUREG-1150.

Victor Stello, Jr.
Executive Director
for Operations

Enclosures:
Figures 1-6

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INTERNAL CORE DAMAGE FREQUENCY DISTRIBUTIONS

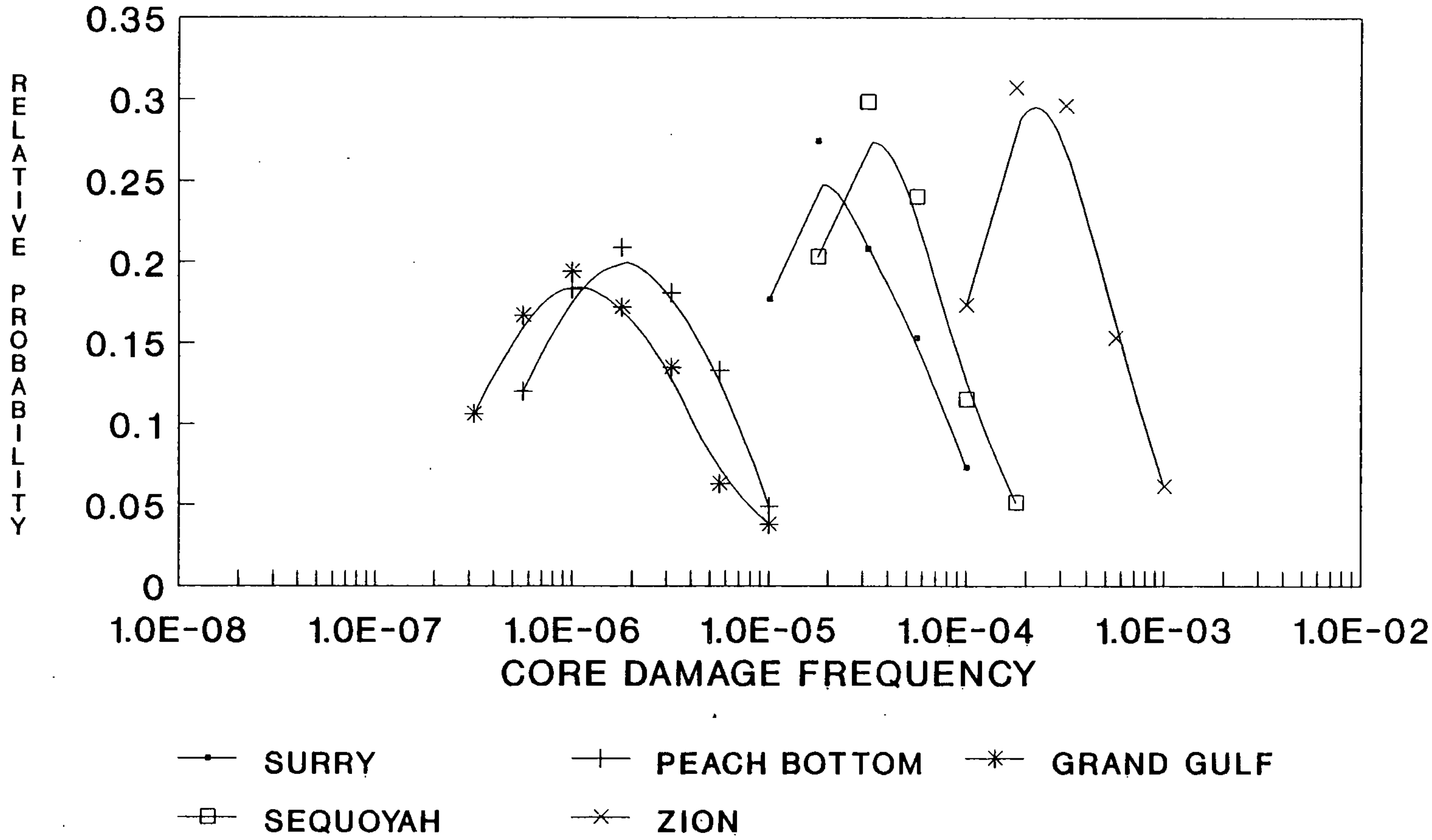

FIGURE 1

INTERNAL CORE DAMAGE FREQUENCY RANGES

5th and 95th Percentile

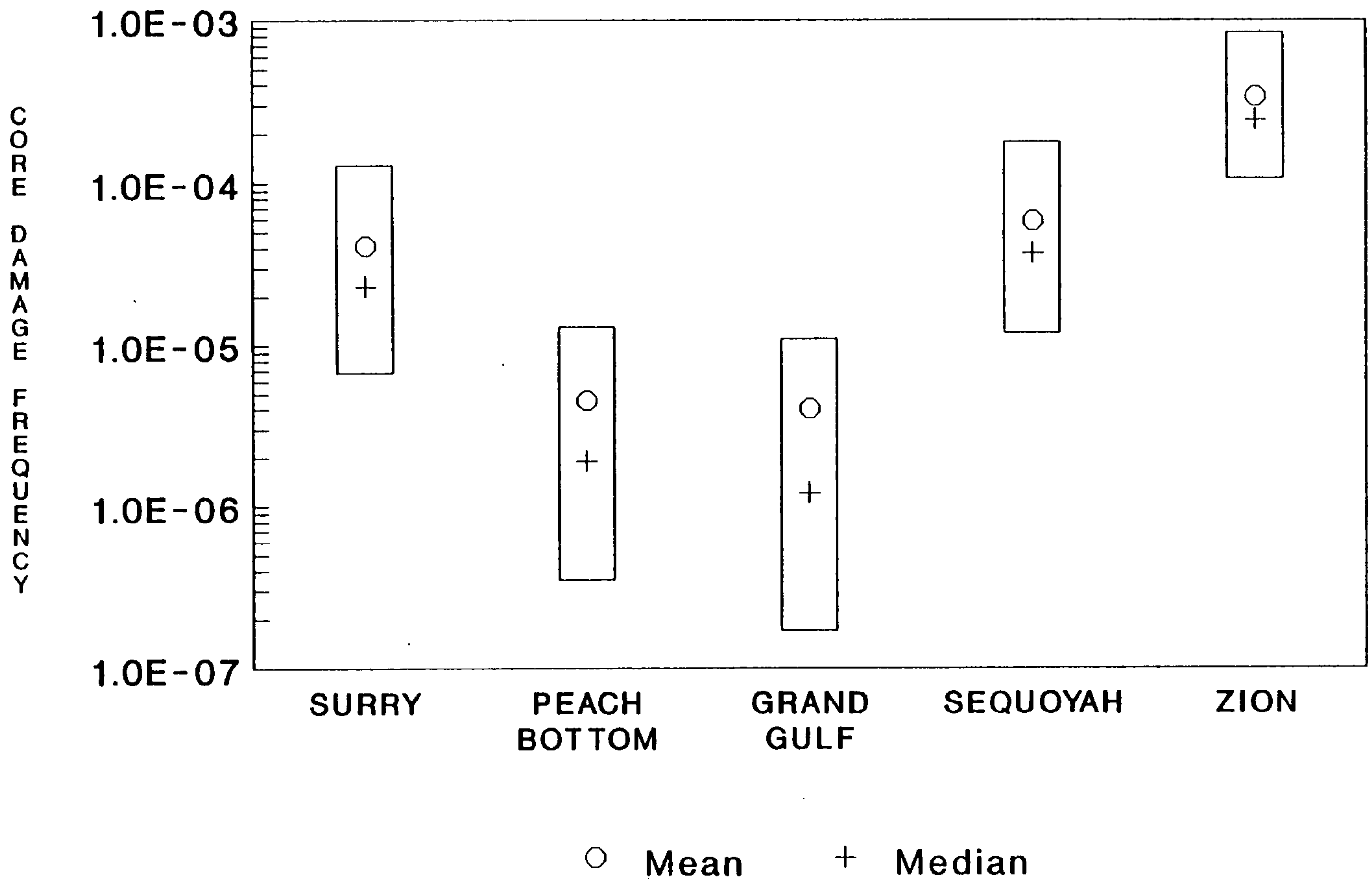

FIGURE 2

BWR PRINCIPAL CONTRIBUTORS TO INTERNAL CORE DAMAGE FREQUENCY

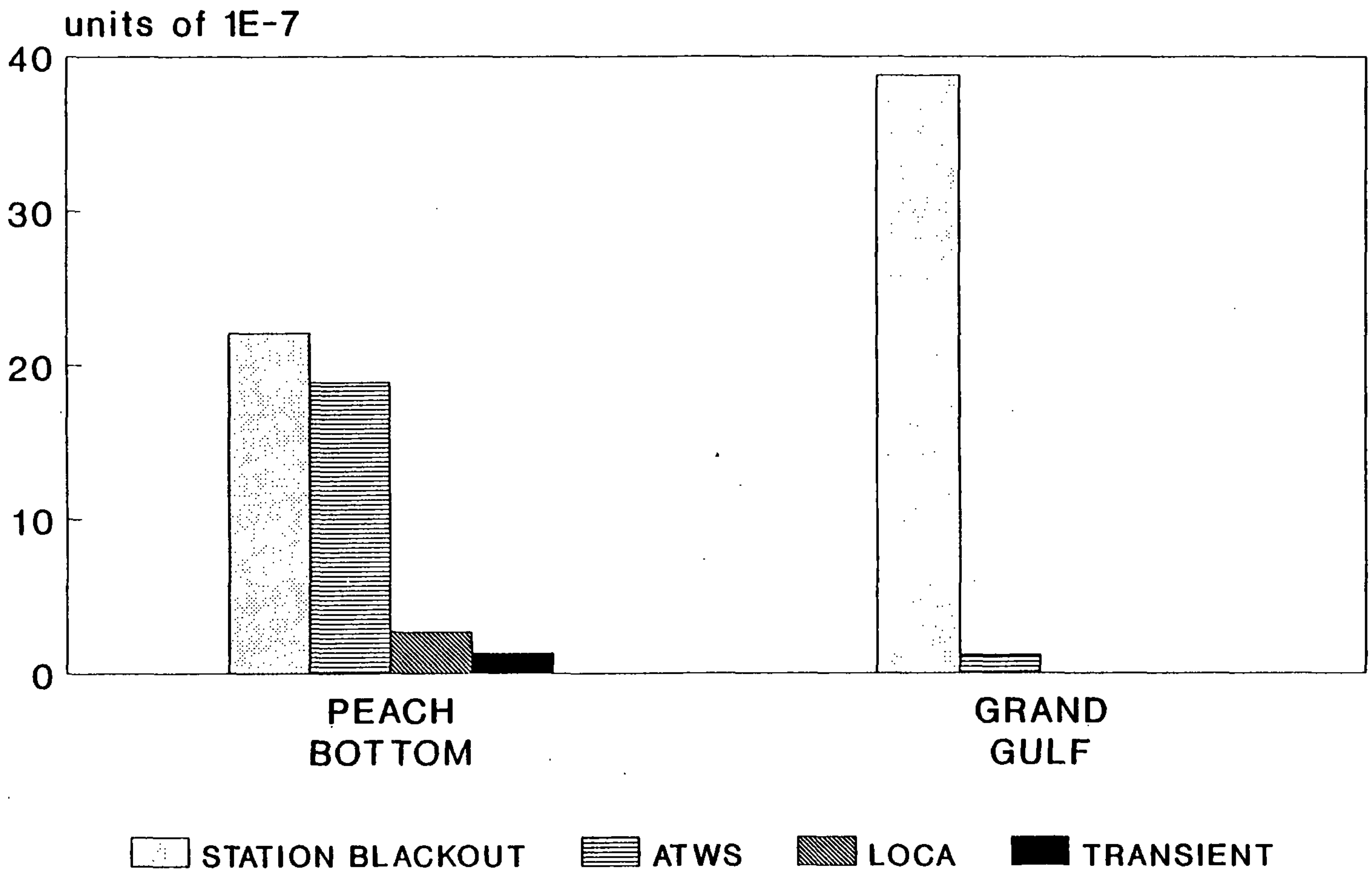

FIGURE 3

PWR PRINCIPAL CONTRIBUTORS TO INTERNAL CORE DAMAGE FREQUENCY

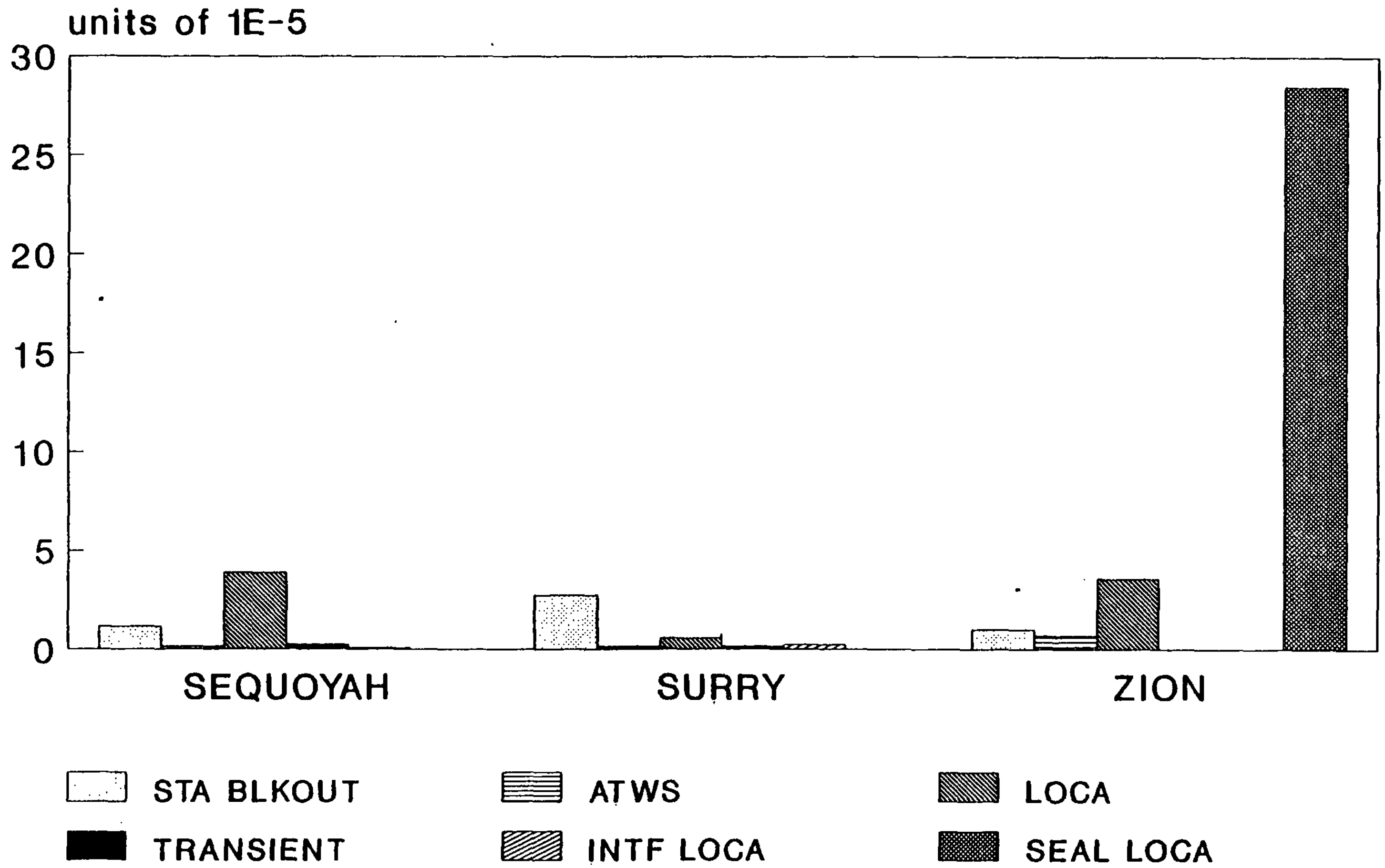

FIGURE 4

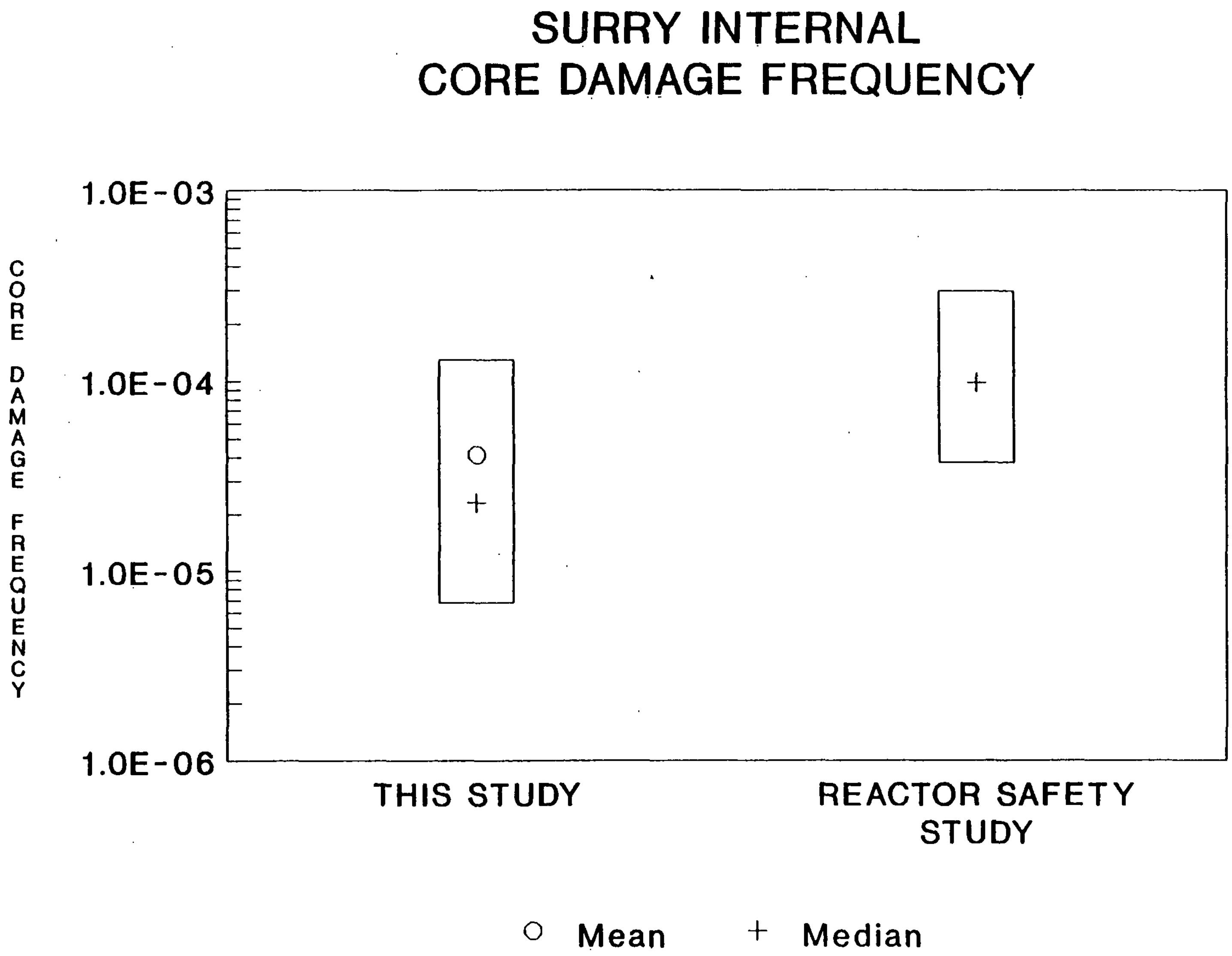
SURRY INTERNAL
CORE DAMAGE FREQUENCY
CORE DAMAGE FREQUENCY
1.0E-03
1.0E-04
1.0E-05
1.0E-06
THIS STUDY
REACTOR SAFETY STUDY
Mean
Median

FIGURE 5

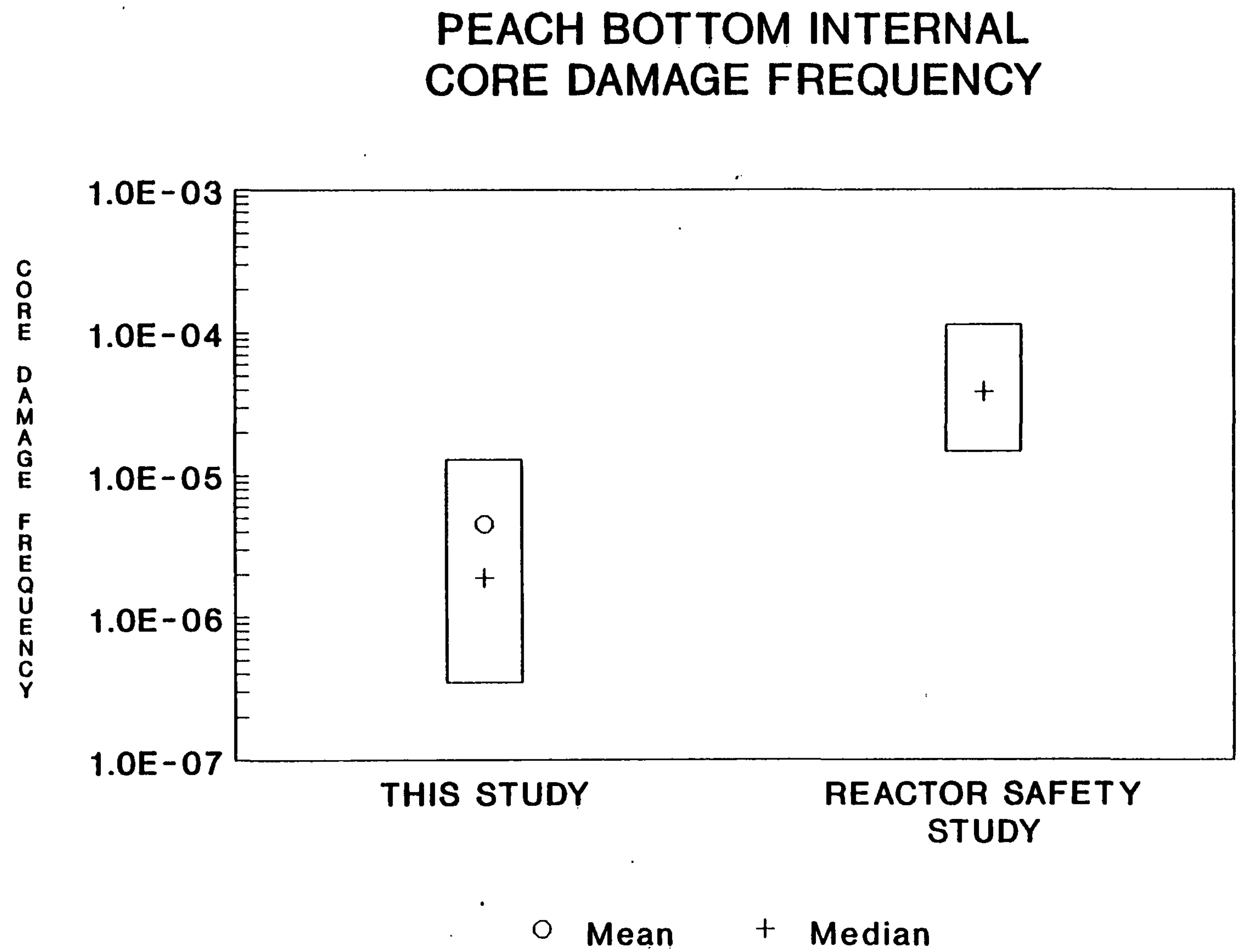
PEACH BOTTOM INTERNAL
CORE DAMAGE FREQUENCY
CORE DAMAGE FREQUENCY
1.0E-03
1.0E-04
1.0E-05
1.0E-06
1.0E-07
THIS STUDY
REACTOR SAFETY STUDY
Mean
Median

FIGURE 6

Attachment 1

ACTION - Beckjord, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE SECRETARY

Cys: Stello
Taylor
Thompson
Blaha
Murley
JMurphy, RES
Scroggins, CON
Bird, OP

February 9, 1989

MEMORANDUM FOR: Victor Stello, Jr.
Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: SECY-88-337 - PLANS FOR FUTURE REVIEW OF NUREG-1150

This is to advise you that the Commission (with all Commissioners agreeing) has approved the staff's recommendation to form a new Review Committee under FACA, subject to the comments noted below:

The Commission believes that the Committee should consist of nine members as follows:

* Chairman

* One or more members of the Kastenberg Committee. Kastenberg should be consulted and ideally would himself be a member.

* One or two members of the ANS Committee. LeSage should be consulted and ideally would himself be a member.

* One member from EPRI.

* Three members from other countries.

* One or more additional members selected to bring additional technical balance to the group (after all of the other members have been chosen and have expressed a willingness to serve.)

Guidelines for selecting individuals for committee membership should include provisions for avoiding real or apparent conflicts of interest, and an emphasis on individuals with broad as well as deep relevant professional expertise.

The staff should work with the FACA management officer and OGC to develop a charter for the Committee and to assure that the Committee is balanced and that no conflicts of interest exist. The Commission should be provided with the final list of Review Committee members.

Rec'd Off. EDO
Date 2-10-89
Time 9:00

The Review Committee should be given a schedule of dates by which the NRC would expect to have an interim report and final report.

After further deliberation, the Commission has decided that (subject to prior review by the Commission) NUREG-1150 should be issued as a second draft for reference by the NRC staff and NRC licensees in individual plant examinations. The final NUREG-1150 report should be issued after the Review Committee recommendations are resolved and after a final review by the Commission. Further, the Commission believes that the staff should not limit the scope of the review.

The review committee should be informed of the staff's plans for use of NUREG-1150 concurrent with its review and should be requested to identify areas of concern to the staff as quickly as reasonably practical. The staff should act quickly to resolve areas of concern identified by the review committee and through public comment, particularly where information, provided in NUREG-1150, pertinent to the individual plant examination may need to be revised. When issued, the forward to the draft of NUREG-1150 should inform the reader that the staff intends to solicit comments of a review committee and that a final report, responding to major comments, will be issued. An estimated date for completion should be provided.

Commissioner Rogers indicated that he believes that the questions the Review Committee is asked to address need more critical thought to improve their balance and scope. In particular, he finds that the question "Does NUREG-1150 represent a major advance in the state of the art of PRA?" an unacceptable tautology that can easily be replaced by several questions which together provide ample opportunities to develop the responses that the staff seeks.

At the next briefing on the status of NUREG-1150 (now scheduled for February 27, 1989), Commissioner Curtiss requests that the staff be prepared to address in more detail the intended uses of NUREG-1150 results.

cc: Chairman Zech
Commissioner Roberts
Commissioner Carr
Commissioner Rogers
Commissioner Curtiss
OGC
GPA