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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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BRIEFING ON MASTER PLAN FOR INTEGRATING ALL SEVERE ACCIDENT ISSUES

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PUBLIC MEETING

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Nuclear Regulatory Commission
One White Flint North
Rockville, Maryland

THURSDAY, JUNE 2, 1988

The Commission met in open session, pursuant to notice, at 10:00 a.m., the Honorable LANDO W. ZECH, Chairman of the Commission, presiding.

COMMISSIONERS PRESENT:

LANDO W. ZECH, Chairman of the Commission
THOMAS M. ROBERTS, Member of the Commission
KENNETH CARR, Member of the Commission
KENNETH ROGERS, Member of the Commission

STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

S. Chilk
V. Stello
T. Speis
W. Parler
T. Murley
A. Marchese

AUDIENCE SPEAKERS:

J. Murphy

P R O C E E D I N G S

[10:00 a.m.]

CHAIRMAN ZECH: Good morning, ladies and gentlemen. Today the Commission will be briefed by the Nuclear Regulatory Commission's Office of Research and the Office of Nuclear Regulatory Regulation on the integrated plan for closure of severe accident issues.

The integration plan is a description of all severe accident programs currently being undertaken by the Commission. It describes how the Agency will reach closure on these programs and describes the interrelationship among the various programs.

The main motivation behind the development of an integration plan for closure on severe accidents was to assure the consistency between programs and consistency with Commission policy and strategic goals.

It's my understanding that there are six main elements of the plan, which include individual plant examinations, containment performance improvements, improved plant operations, severe accident research program, external events, and accident management.

The Commission is interested in hearing how consistency between programs is maintained, the interrelationship between programs and policy implications of each program.

Additionally, we would be interested in hearing how the resolution of severe accident issues may impact the backfit rule.

It is my feeling that the backfit rule has been providing some order of discipline to the backfit process and is working quite well, but we would like to hear from the staff in that regard.

This is an information briefing today and no Commission vote is anticipated. Do any of my fellow Commissioners wish to make any opening comments?

[No response.]

CHAIRMAN ZECH: If not, Mr. Stello, will you proceed, please.

MR. STELLO: Yes, Mr. Chairman, and thank you. The meeting this morning I think is very, very important and the paper that is now before the Commission I think is also very important.

Over the years, a number of issues have been raised with respect to severe accidents. The idea behind the paper and the need for the paper became apparent after one started to ask the question of how do you get to the end of the process, when and what has to be accomplished to achieve closure of the severe accident issues, the satisfying of the Commission's policy statement on severe accidents.

The paper is intended to show you the relationship so

that as the various pieces are before the Commission, the Commission will better understand how each of those particular issues fit together and when is it that we will be able to finally conclude that we are done with the severe accident issues and find that issue closed.

You raised at the outset a question that I think is a very, very important question and that's the process of how we are going to make decisions regarding what additional modifications might be needed for the facilities.

The Commission has issued guidance in this matter in two forms at the moment. The first is in the Commission's backfit rule, which you have already indicated has given a structured philosophy and policy of how to proceed with backfit issues.

At the same time, the Commission has also given us some guidance with respect to safety goals. We intend to use both the backfit rule and the Commission's safety goal policy to the best of our ability to do so.

We have, however, recognized that in dealing with severe accident issues, it may be that the backfit rule or the policy statement does not adequately cover any particular issue that we're going to be dealing with.

And on Page 18 of the Commission paper, we particularly tried to make the point that if we ever came to a point where we thought something needed to be added to a

facility and it were an issue that we believed more work needed or modification, even if the backfit rule did not, if you will, pass that test or the Commission's safety goal policy did not provide enough guidance.

For those cases, we would come to the Commission on a case-by-case basis, present the issue to the Commission for the Commission's decision so that --

COMMISSIONER ROGERS: Aren't you talking about specifically the Mark 1 issue?

MR. STELLO: Well, that could be, but let me be more specific than that. Some of the plants that we are now analyzing, and I guess forecasting some results that have not yet even been published on NUREG 1250, some of the core melt frequencies are approaching ten to the minus six.

If you look at the Commission's safety goal policy, the most stringent requirement in that safety goal policy would suggest that a plant should have a frequency of less than ten to the minus six for a significant release.

Well, clearly if the core melt frequency is ten to the minus six, you're going to meet that one and you meet all other safety goal requirements.

If you then start looking at the backfit rule in terms of analyzing it for cost benefit analyses, at ten to the minus six for a core melt frequency you're obviously substantially less for any kind of a release, and when you work

out the equations, you're going to wind up with very few dollars which could justify doing anything.

Well, if there is a particular dominant sequence that is still remaining in Grand Gulf that judgment would suggest you might want to do more, then that is the kind of an issue we would clearly bring to the Commission.

With that, Mr. Chairman, and I think perhaps we ought to talk some more about what is the meaning of Page 18 so that we are clear on what our intent is and if there's more guidance by the Commission on that it would be useful for perhaps the Commission to make sure that we understand it.

I will then ask Dr. Speis to go through the presentation and this is the kind of presentation, I think it's probably a useful idea to stop whenever something is necessary for clarification and try to do it at that point so we're prepared to answer all of your questions at any point in the presentation. Themis?

CHAIRMAN ZECH: All right. You may proceed.

MR. SPEIS: Mr. Stello, Mr. Chairman, Mr. Commissioners, I will start with some background of this effort on Page 2 of the viewgraphs that you have in front of you.

If you recall, on August 8, 1985, the Commission issued the severe accident policy statement. In it, the Commission concluded that existing plants pose no undue risk to the public.

That conclusion was reached after extensive evaluations involving severe accidents and the research that was done after that time.

At the same time, however, based on NRC and industry experience with plant-specific probablistic risk assessments, the Commission recognized that systematic examinations will be beneficial in identifying any plant-specific vulnerabilities to severe accidents for which further safety improvements may be justified.

In February '86 we presented to the Commission an implementation plan for the severe accident policy statement which addressed the specific issues that were raised in 50 FR 32138.

At the same time, we provided some additional information to the Commission in two SECY's dealing with the treatment of external events in severe accidents as well as how does one evaluate a source term in a realistic sense.

In February '87, the Commissioners asked EDO to provide an integrated plan for some of the issues that they were discussing at that time, and at the beginning of '87, EDO provided to the Commission a preliminary plan for integration of some of the severe accident issues.

Finally, last year Mr. Murley from the Office of NRR and from the Office of Resource, briefed the Commission on the plan for closure of severe accident issues, including issues

relating to BWR Mark I containments.

If you recall, Mr. Chairman, at that meeting the staff indicated its intent to pursue an integrated approach to resolution of severe accident issues, which I'll say more about when I come to the issue of containment performance improvements.

In December '87, we presented to the Commission our plan to resolve the severe accident issues relating to the containment performance, starting with Mark I. That was described in SECY 87-297.

Finally, the EDO assembled together a number of senior NRC managers at the beginning of this year, from the Offices of RES, Nuclear Reactor Regulation, and the Office of AEOD, to discuss the whole spectrum of severe accident issues with the primary emphasis on how to proceed with the implementation of the Commission's policy statement.

Putting this plan together was a result of that Baltimore meeting. On the second page, on Page 3 of the handout, I want to kind of outline the objectives, even though, Mr. Chairman, you discussed them already.

Basically, the purpose of the integration plan is to present the staff's plan, how to integrate all the issues and how to proceed with the closure of severe accident issues.

The objectives are, a number of them, to provide an understanding of the staff activities, both the main ones and

the supporting elements that are underway to implement the Commission's severe accident policy and to assure that these activities are consistent with the Commission's policy and overall strategic goals.

It's important to assure that these activities are consistent among themselves, have a common goal which is to ultimately lead to improved plant safety and are properly coordinated among the responsible NRC organizations, and to assure that the Commission is aware of the key technical and policy issues, some of which will meet Commission guidelines for approval, and finally to describe the use of safety goals and backfit policy in the closure process.

I think Mr. Stello already mentioned, but I think it's important to stress that this paper presents the information, describes it, and tells you when the different packages with be reaching the Commission for decision-making.

There is no need to make any decisions on this paper itself today. The severe accident integration plan provides for a number of coordinated activities to ensure fulfillment of the severe accident policy.

These activities, which are addressed in the integration plan, are listed in the next viewgraph, on Page 4 of your handout.

Basically, the main ones are the IPE, the individual plant examinations, the containment performance improvements

program, which will start with the Mark I's and complete the cycle of the different types of containments by the end of next year, improved plant operations, the severe accident research program, external events, and accident management.

The supporting activities are such reports as NUREG 1150. Of course, the generic safety issues and the integrated safety assessment program, ISAP.

The severe accident policy for future plants is also addressed in this Commission paper and, of course, the process of closure and how we plan to utilize the safety goal in that process.

The next page, I have a schematic illustration of the severe accident integration plan and basically completion of the elements of this plan constitute the basis for assuring that the residual risk to the public from severe accidents of the nuclear power plants are minimized.

We already repeated the six main activities. You see that all of them fit directly into the severe accident closure.

The individual plant examination, which includes both internal and later on the external events, on the left, the middle left.

Above it, improved plant operations. They also fit into the severe accident closure via improved tech specs, via improvement in the procedures existing, emergency operating procedures, and reaching them with accident management

procedures which I'll discuss later on.

Also, at the lower part, you see the potential containment improvement, starting with Mark I and going into the other ones, and they also lead directly into the severe accident closure.

Now, to summarize at this point, severe accident closure is achieved once the IPE's, the individual plant examinations, have been completed, including external events, and any appropriate changes implemented.

A framework for an accident management program has been developed and implemented and any generic requirements resulting from the containment performance improvement programs have been implemented.

Later on, I'll discuss in some more detail the closure process. Now, besides the main elements listed here, I already mentioned the supporting elements, for example the 1150, which provides information and insights that help us in the decisions that have to be made as part of the main elements.

Also the backfit and the safety goal policy provide guidance and constraints in defining the implementation strategy for these main elements basically.

And the next viewgraphs, Mr. Chairman, I'd like to summarize some of the main activities of the integration plan --

CHAIRMAN ZECH: Before you on there, let me interrupt you just a moment.

MR. SPEIS: Yes.

CHAIRMAN ZECH: This integration effort, I think, is extremely important and very significant. I have mentioned the backfit issue in my opening remarks and Mr. Stello, and now you, Dr. Speis, also referred to the backfit issue as it applies to this integration process and this effort that we're undertaking.

Let me give you at least what I understand, there are provisions regarding the backfit rule, and ask for your comments to see if you would agree because the effort to integrate these different efforts towards improved public health and safety are extremely important.

And let me just ask you if you would agree, at least with my understanding, and for my colleagues too, with at least my brief summary of the backfit process as I see it applying to this integration program.

And although the backfit rule, as I understand it, does require an analysis of the safety benefits in relation to the cost, and it does put discipline into the system, the backfit rule does not preclude the Commission from taking a regulatory action to improve safety.

MR. SPEIS: Yes, sir.

CHAIRMAN ZECH: Is that correct?

MR. SPEIS: Yes.

CHAIRMAN ZECH: And therefore, as I understand it then, the staff should and would bring to the Commission's attention any proposed significant generic modification for a Commission decision, even though it may not meet the backfit rule. Is that correct, too?

MR. SPEIS: That is correct.

CHAIRMAN ZECH: And I think that's very important to recognize as we discuss integration. And I think, at this point, I should emphasize, too, the recent emphasis we've placed on the backfit rule and emphasize, too, that there are three conditions where the backfit rule would not apply.

The first, of course, is where modifications are needed to bring the facility into compliance with our regulations. The backfit rule, no cost consideration applies.

MR. SPEIS: Yes, sir.

CHAIRMAN ZECH: The second would be where the modifications are necessary to assure adequate protection of public health and safety. Backfit rule, no cost considerations in that regard.

And the third would be where modifications involve determining what the adequate level of protection, public health and safety protection, is -- in other words, if there's a concern about needing adequate protection, public health and safety, then the backfit rule would not apply in that regard

either.

Is that correct, too?

MR. SPEIS: Yes.

CHAIRMAN ZECH: All right. Well, those are important, I think, as we consider now your further presentation of the integration of these very important severe accident related issues.

So will you proceed?

MR. SPEIS: Yes. Starting on Page 6, Mr. Chairman, I would like to summarize some of the main activities of the integration plan as well as some of the supporting ones.

I will start with the individual plant examination which flows directly from the Commission's policy statement of '85 where a plant-specific examination will be undertaken.

And the Commission, at that time, told us to go ahead and work with the industry to formulate and integrate it, systematic approach to an examination of its nuclear power plant, now operating or under construction, for possible plant-specific vulnerabilities that might be missed without such a systematic search.

As I said earlier, previous PRA's always identify something, items that could be improved. The examination, also the Commission at that time said that the examination will pay specific attention to containment performance in attempting to strike a balance between accident prevention and consequent

mitigation.

In fact, it's in this area that possibly, as we go down the road to examine containment performance improvements, it's possible in some of those areas, in attempting to strike a balance, we might find out that in some cases the expenses to pursue such affixes might go beyond the backfit policy, in addition.

Those areas that will come to you on a case-by-case basis. You know, that's an example for them. The examination will be requested via a 10 CFR 5054F letter.

Again, as I said, the basis for the examination is the Commission's policy statement. We have interacted extensively with industry and developed appropriate documentation for this examination.

Later on, I'll discuss what type of methodology could be undertaken to do this examination, could be utilized rather. Now, the purpose of this examination is for the utilities to identify and understand the most likely severe accident sequences that could occur at their plant and the utilities should take the initiative to evaluate and implement any means for further improvements to the residual risk that we're talking about.

It's very important to develop an awareness of severe accident behavior. Now, it is expected that it would take about three years to complete and evaluate the examinations

conducted in accordance with the individual plant examinations.

I will continue on the IPE, on Page 7 of the handout. It is very important that the licensee staff should participate in all aspects of the IPE, the individual plant examination, so that knowledge gained becomes an integral part of operating, training, and procedural programs in their plants.

COMMISSIONER CARR: Are you going to require that?

MR. SPEIS: It is strongly urged.

COMMISSIONER CARR: I guess that means no.

MR. SPEIS: Well, it's part of the package that will be coming to you at the end of this month. We feel very strongly about that. I guess maybe I'm not an expert linguist --

COMMISSIONER CARR: No, no. I agree. I feel strongly about it, too.

MR. STELLO: I think we're about as close to saying that's the way it ought to be done.

COMMISSIONER CARR: Well, if we believe that's the way it ought to be done, why don't we just say that's the way to do it?

MR. STELLO: Maybe that's what we will wind up with in the package we send the Commission.

MR. SPEIS: Now, the licensees should conduct a systematic examination of plant design, operation, and maintenance and emergency operations to identify plant-specific

vulnerabilities, attempt to quantify the results to the extent possible for the sequences that contribute the most to the total core damage as well as to large release frequency, understand what could probably go wrong in the plant, what could possibly go wrong in a plant, and identify and evaluate means for improving plant containment performance.

And then the licensee should decide what improvements will be implemented and proceed to schedule such implementation.

Now, the IPE generic letter was reviewed by CRGR on 4-18-88 and by the full ACRS Committee on May 5, 1988, even though they had also reviewed the preliminary package of the IPE last year.

With that summary, I would like now to go to the containment performance improvements program on Page 8 of the handout, Mr. Chairman.

As a result of concerns related to the ability of containments to withstand some generic challenges associated with some very low likelihood severe accident scenarios, we have in parallel with IPE undertaken a program to determine what, if any, action should be taken to reduce the vulnerabilities of containments to severe accident challenges.

I should like to point out that this evaluation involves generic challenges and does not involve plant unique with the primary focus of the IPE's, but those two programs

compliment each other and there is no intent to undertake any containment performance improvements if any are found to be needed without a close integration and coordination with the IPE process itself.

Okay. That's very important and we articulate that as much as possible in the Commission paper itself. It's an issue that has been raised, whether we have two separate programs, and it's really one program.

The focus, again, is to -- I'm sorry. I would like to say that this program was described to you in a Commission paper about December of '87, SECY 87-297.

Again, it focuses on evaluating generic challenges and as part of that you have to understand the failure modes and we're also evaluating potential improvements.

We're staring this program with Mark I. Our approach that we are pursuing, the approach that we are pursuing involves both accident prevention and mitigation.

It's an integral one. We'll look very carefully at what additional things can be done to further reduce the core damage, core melt probability, what things can be fashioned in an accident strategy way to not only prevent an accident, but possibly if one is initiated to arrest it, or if you fail there, what things can be done to further reduce the threat to the containment or even reduce the source term that might be released given a severe accident.

So again, I want to stress that this approach is a very integral one and it involves both accident prevention, accident management, accident mitigation.

For example, we're exploring additional sources of water for both core cooling and possibly for containment and debris cooling and fission product scrubbing.

We're looking at things like utilizing existing systems, like fire water sprays, to connect them to the residual heat removal system or to the sprays themselves.

We're looking at improving the reliability of the automatic depressurization system. If you depressurize, if their liability is high, them you're able to introduce the water into the system because you have low pressure systems available in a power plant.

We're also looking at venting. All of us have heard quite a bit about venting. For Mark I's, we're looking at the suppression pool as the filtering medium.

We're looking very carefully at the pluses and minuses of venting and we're not focusing on venting itself. On all of these things, we're performing regulatory analyses and we will be giving you a preliminary report of this effort by the end of this month or some -- I guess at the middle of July, and our final recommendations from this will be to you at the end of this summer, Mr. Chairman.

MR. SPEIS: So that's a partial summary of the

program, and I'd like now to go to the improved plant operations program. On page nine. If you recall, the Commission's approach to safety was dramatically changed following the TMI-2 accident. Our emphasis shifted from providing safety by relying on the traditional design basis approach to a multi-faceted approach which emphasized improved operations, a human factor considerations, a realistic performance of systems, and performing probabilistic risk assessments to uncover weaknesses or strengths in the design which can be farther improved.

This program, it's very important, it's the bread and butter of really, of the Nuclear Regulatory Commission, and Mr. Murley here has spent I guess 110 percent of his time in this effort. And so do our regions. So this program to include plant operations considers the following efforts. And this is, of course, it's not complete. Continued improvement of the systematic assessment of licensee performance, the so-called SALP process. Our regular reviews by senior NRC managers of problem plants to identify and evaluate those plants that may not be meeting NRC and industry standards and operational excellence.

Team inspections to probe farther the performance of those plants identified involved that possibly do not meet the NRC and industry standards. Regulatory actions to improve operational performance where it has fallen below expected

standards, improved technical specifications, continued improvement of operating procedures, expanding the emergency operating procedures to include guidance on severe accident management strategies, I'll say more about that shortly. And research to establish the sensitivity of risk to human errors including the contribution of management at the level of human errors.

This is, of course, improved plant operations is, as I say, it's an ongoing program. It's the bread and butter of NRC. And we don't foresee any specific additional actions on your part as far as this program is concerned. I don't know. If Mr. Murley wants to add anything to this brief summary that --

MR. MURLEY: Well, we will have a paper to the Commission probably within a month or two, on accident management, which is an integral part of the improved plant operations. And, as you know, a lot of what we're doing, tech spec improvements, we're now looking at improving, actually reducing surveillance testing requirements during operation. Improved emergency operating procedures. A lot of the stuff that NRR does bears on this key part of closing severe accident issues.

MR. SPEIS: Now I'd like to summarize the severe accident research program. The severe accident research program was begun after the TMI-2 accident to provide the

Commission and NRC staff with the technical data and the analytical methodology needed to address the severe accident issues. Initially, the work was oriented to phenomena to gain more insight into the mechanisms of the different processes involved, given a severe accident, and consisted of a wide range of experiments and code development to gain more insights and understandings into those phenomena and processes.

Some examples of severe accident phenomena and technical areas that the severe accident research program has examined include fission product release, transport, deposition and revaporization, both in the primary system and in the containment, containment loadings, given severe accidents. We looked at the hydrogen phenomena involving both burning, conflagration, or detonations. We looked at, given a massive core melt that penetrates the vessel, and if it attacks the concrete basement, we looked at the mechanisms of core-concrete interactions, and the gases, and the forces that are generated as a result of those interactions.

We tested the containments themselves to see what is their capability. At this point, I would like to say that most of our, all of our containments of course have been designed for the so-called design basis accidents. The large break LOCA or steamline breaks. But, because of the conservatisms in the codes that have been utilized to design them, most or I would say all of our containments have ample margin to go beyond the

original design basis. In fact, they're able to withstand a large number of severe accident scenarios. Not of course, the complete spectrum. There are always some extreme low-likelihood scenarios that can challenge the containment. But I think it's important to stress at this point that our containments are able to accommodate forces, pressures, and temperatures that go beyond the original design basis. In fact, one of the objectives of the accident management program is to understand those margins and be able to utilize them in an accident management strategy. We also looked at the effect of natural circulation on the primary system. So these are some of the basic things that we did over the past few years.

The primary emphasis in the near future will be to resolve specific technical issues associated with the main elements of the plan to facilitate the formulation of staff positions, especially in the area of core containment performance improvement. And I have listed some of the specific issues that I'm talking about here. For example, meltspreading and potential containment or shell failure in Mark I's. The containment failure probability by the rad containment heating. Resource data and models to assess accident management strategies.

For the longer term, I will be pursuing confirmatory research on issues such as refining hydrogen behavior models, core melt progression, a look at the later stages, some

additional work on core-concrete interactions, especially with corium, that is reaching metallic components. And to pursue further model assessment and refinements. Again the main point in the next few years, the research program will emphasize resolving specific issues. Not dwell in the abstract. An example of that is given on page 11, Mr. Chairman.

CHAIRMAN ZECH: Before you go on. Let me just ask you in that regard. It would appear that there -- some of these confirmatory research projects are extremely important, and of course, have not been resolved as of now. That's why you need further research on them. But I guess my question would be, looking at the time schedule for completion of the IPE program, do you anticipate these research efforts will be resolved sufficiently to make you, to allow you to a confident recommendation regarding IPE program itself.

MR. SPEIS: Yes, we feel confident that we can integrate existing information, make decisions, and in some areas, the decision process will be such that you will need confirmatory. But I think we'll be able to make decisions based on information that exists. Many times you do make decisions that you don't have all of the information in front of you. It's decisionmaking that involves uncertainties. But those uncertainties can be bound and we're pursuing strategies that can at least for the time being put some of these uncertainties on the side, and still implement some of these

programs that will further reduce the residual risk that we're talking about.

CHAIRMAN ZECH: I'd encourage you to watch that very carefully. Because I think the Commission will be very interested in ensuring the research program and the associated data that's so necessary will be forthcoming in a timely manner so that you can make these IPE decisions with confidence. So, as we proceed along that line, I think it would be important to let us know what research has been completed and what confidence you have that it is progressing in an appropriate manner. All right? Thank you. Proceed.

MR. SPEIS: On page 11, I give an example of how we're attempting to focus the research to address some of the specific issues that you alluded to, Mr. Chairman. I have given an example here for a specific type of containment. I have described the associated issues and below I have addressed, the research to address. In the Commission paper itself, we have attempted to focus the research program on the issues associated with both containment types, all containment types, that are utilized in the United States.

Again, we have attempted to identify the issues. Some of these issues have been discussed with industry, with our research people at the laboratories, and again as I say, our focus in the next few years is to focus, to address those specific safety issues. On page --

COMMISSIONER ROGERS: Just before you leave that page, could you just clarify a little bit on this matter of cut-off pressure for HPE and the DCH issue. I'm not quite clear on what --

MR. SPEIS: Yes. What we're talking about here. There is a mechanism that has been postulated where the vessel will fail under high pressure conditions and the corium itself will blow down into the containment, and in addition to the thermal transfer of heat there will be chemical reactions that will augment the pressure. There we're talking about what is that pressure level where this phenomenon is not important. Okay? And that's --

COMMISSIONER ROGERS: A drop below this pressure.

MR. SPEIS: Yes. And that's what we're talking about.

COMMISSIONER ROGERS: Okay.

CHAIRMAN ZECH: Let me make a comment on that too, before you move along. In the discussion in your paper on containment performance, I would -- I didn't see as much discussion on prevention versus mitigation as I might expect. I know you've thought about that. But it seems to me as you go through the considerations of containment performance that you continue to balance the prevention versus mitigation considerations. It's extremely important. And if you're giving a certain emphasis to prevention, for example, rather

than mitigation, in accordance with perhaps the consideration of the safety goal or other considerations, those things should be pointed out to us. In other words, the rationale that you use in mitigation versus prevention-type thoughts, analysis, I think would be helpful for the Commission to get as you present these containment-performance considerations to us.

MR. STELLO: Mr. Chairman, you will be getting precisely that kind of thought process in the paper sometime in June on the Mark I's.

CHAIRMAN ZECH: Fine. That's what we'll be looking for. All right. Thank you.

MR. SPEIS: The next page, Mr. Chairman. I'd like to say something about the accident management program. Dr. Murley already mentioned the Commission paper that will be coming to you on it. We view the accident management as an important means of possibly achieving a substantial reduction in risk from severe accidents. I would like to start a presentation on this subject by saying that PRAs typically measure the loss of adequate core cooling and not core melt. Because since PRAs normally do not address in a substantive way the likelihood of recovery actions. So at present, there is no good way to quantify the fraction of severe core damage accidents that go to core melt.

And when I'm talking about severe core damage accidents, I'm talking about TMI. Where TMI was a core damage

accident, but it was arrested. And finally the corium was, stayed inside the vessel and the vessel was penetrated. So when you see a number that comes from a PRA, it normally measures the loss of core cooling and does not address the number that is associated with the vessel melting. Okay. So to us, accident management is very important because we're going to focus in that area to see even if an accident is initiated and the initiation of a core degradation process begins. You know, what things can be done to make sure that the accident is arrested. And accident management will play an important role in this area.

That's why I say it's a very important program. And we look at this as very important in reducing the risk of severe accidents.

Now accident management includes the measures taken by the plant operating and technical staff to prevent core damage and to us, preventing core damage is part of the severe accident program itself. Severe accident is not only when a severe accident takes place and proceeds to challenge the containment, and that comes to your prevention discussion that you had previously with Mr. Stello.

As part of the accident management, we look at the means to terminate the core damage if it occurs and retain the core within the reactor vessel, as I said earlier. And failing that, maintain containment integrity as long as possible. And

even failing that, to minimize the consequences of outside releases. So, it's a very integral process that we're looking at. Accident management encompasses hardware, human and organization factors, of course. It provides decision-makers at the plant a structured program for managing a severe accident. So you have to have thought about this thing ahead of time. You have to put the procedures in place. And last but not least, you have to train the operators and the technical staff in the management of the plant to be prepared to undertake such activities if the need ever arises.

Now in the proposed individual plant examination generic letter, the way we addressed accident management is as follows:

Utilities are expected to develop an accident management program for prevention or mitigation of risk important severe accident sequences. We described that in detail. And identify measures that plant personnel can and should take in case of severe accidents.

We're working with industry, with NUMARC to develop an accident management framework. We don't want to have a separate accident management program for every plant. I think there should be some coordination. But meanwhile, we're telling utilities that, as you go through your individual plant examination and you find some things that you think will improve your plant in this area, you go ahead and take the

initiative and do it and just use the criteria 10 CFR 50.59 as part of that process.

This 10 CFR 50.59, it's something that the utilities can take the initiative to do whatever they think is prudent as long as it does not change the balance of the previously analyzed safety of the plant.

So again, that is a brief summary of the accident management program. We'll be telling you more about it when we come to you with the individual plant examination letter.

Next, I would like to discuss the role of the NUREG-1150 as part of supporting our main line efforts in this area. In February of '87, NRC published this reactor risk reference document, NUREG-1150 in draft form for public comments. It is currently being revised to accommodate comments received as a result of a number of peer reviews that it has received. And our present plans are to complete it by the end of this year.

COMMISSIONER ROBERTS: It will be completed by when?

MR. SPEIS: By the end of this year, Mr. Roberts. December '88. I have to look. I saw Murphy in the back, who is the guru of this, to make sure I said the right thing. The objective of this document has been to provide a snapshot of the state-of-the-art PRA technology. It basically incorporates improvements in methods and data that have been accumulated since the WASH-1400 document that was put together in the mid '70s. It includes examination of severe accident frequencies

and risks and their associated uncertainties for five specific plants.

Its role in the regulatory process. It has many of them. It's to provide independent staff assessment of risk. It provides technical data on a number of issues and phenomena and integral studies to the individual plant examination. It provides insight to develop accident management strategies. It provides insights and calculations of containment performance in each of these that you have underway. Also it provides insights in the areas of resolving a number of generic issues.

Also, it's going to be utilized to prioritize, to focus our resource activities, especially for the next few years.

Now, Mr. Chairman, I would like to spend the remaining of my time to discuss, say some more about severe accident closure process itself and discuss some of its elements. This thing is described in some detail in the Commission paper starting on page 18. Basically, the severe accident closure defines for the industry those programs which are critical to resolving severe accident issues for their plants and identifies the specific steps that must be taken to achieve resolution.

Actions must be taken for generic issues and plant-specific issues. Closure for generic issues results when the Commission either takes action in the form of rulemakings or

explicitly states whatever its required approach is. And closure for plant-specific issues results when each utility has completed certain evaluations and implemented certain programs such that the events which comprise the dominant contributions to risk are identified and practical improvements are made to reduce the probability of risk contributors or their consequences to acceptable low levels. So, this is kind of an overview.

I would like to summarize now the use of the safety goal in the closure process. This is only an outline. And it kind of provides the, again the outline of something that will be coming to you, Mr. Chairman and Mr. Commissioners. In August, we will be providing you a very detailed paper that recommends to you how to go forward with implementing the safety goal. All elements of it. Our plan is that the safety goals and objectives will be used only for the resolution of generic issues. Resolution of plant-specific issues will be accomplished through the IPEs, and the plant-specific backfit criteria which are in existence.

Safety goal policy implementation plan will be published, as I say in August '88, and it will address the Commission concerns and the ACRS recommendations. We're interacting with the ACRS very extensively in this area as you recommended, Mr. Chairman. And those concerns expressed by the ACRS and the process of interaction will be described in this

Commission paper. Basically all PRA information will be used to make comparisons with applicable safety goal objectives.

We will determine why certain classes of plants don't meet safety goal objectives and assess reasons relative to current regulatory requirements. Again, this is going in the ACRS direction, when instead of comparing numbers with numbers, we'll examine the regulations and the specifics in plants to see in an engineering way why some plants don't meet safety goal objectives. Now such assessments will test the effectiveness of the present requirements and may result in changes to regulations on a generic basis and safety enhancements.

Such safety enhancements, of course, will be subject to regulatory analysis per the Commission's backfit rule, and proposed to the Commission for backfit in the form of rulemaking. We will go forward in some of these areas. The first application will be reflected in staff recommendations to Commission on BWR Mark I containments. Performance improvements which will be coming to you in the Fall of '88.

The last capsule on this viewgraph is the thing that Mr. Stello brought to your attention, Mr. Chairman, the appropriate -- if appropriate, any proposed plant operational improvements which cannot be justified by the backfit would be provided to the Commission on a case-by-case basis.

COMMISSIONER ROBERTS: I still have to ask, if

improvements don't result in substantial safety benefits and are not cost-beneficial, why would the staff propose them?

MR. SPEIS: Well, as I said earlier, Mr. Commissioner, in your policy statement, you talked about balance between prevention and mitigation, for example. Okay. And it's possible in attempting to get to that balance, we might find out that we're outside the balance of the backfit policy. Or -- and then for that case, if we find out that there's some good reason to do something, we'll come to you. Again, we take that guidance from that broad statement in the Commission policy statement.

Another example is in the implementation of the safety goal. The 10 minus 6 large release, the way that is defined, the way that you people will tell us what is the appropriate way to proceed might be outside the bounds of backfit policy. And in that situation, then we'll come to you on that. So these are two specific examples that it's possible we might find ourselves. But again, the bottom line would be safety and if we find out that we do things that don't improve safety, I don't think we as a staff will come to you.

MR. STELLO: I think if it isn't a significant safety improvement, we certainly aren't going to come. But whether or not when you strip into a cost benefit analysis you can reach to some of the other policy, to 10 to the minus 6 and the safety goal, or the balance in the philosophy and the policy of

the Commission for severe accidents. We needed to have, if you will, an escape valve, and since we're not sure we know how to draw every line for every issue and that the backfit rule in every case is as far as the Commission would want us to go. If there's a question, we're going to come to the Commission. As long as that philosophy, the backfit rule itself, allows us to continue, we're going to go. But whenever there's a problem, we're going to come back to the Commission on a case-by-case basis for you to give us further direction. That's simply what it means.

CHAIRMAN ZECH: And I think that's very important myself. Because I believe when there is any doubt at all, and recognizing that we're making judgments, engineering, scientific judgments, that aren't always quantifiable with a fine line, I think it's very important in those cases that staff does come to the Commission to either affirm or reject the staff's recommendation. I think that's very important and I think it's part of my understanding that the backfit rule does not prevent the Commission in any way from making a decision, if we, the Commission, do believe that safety can and should be improved. So I think those are the ones we are talking about. Those judgments should come to the Commission. The backfit rule should not preclude you from worrying greatly about how the balance flows between the cost benefit analysis. If you believe there's a doubt or judgments involved, I think

it's the Commission's responsibility to make those decisions.

MR. STELLO: We agree with that.

MR. SPEIS: On page 16, Mr. Chairman, I say something about the resolution of plant-unique issues. I alluded to it earlier. In performing the individual plant examinations, our utilities may find and it is expected that they will voluntarily remedy any uncovered outliers, and make any improvements they deem appropriate, conforming however, to 10 CFR 50.59. However, through the review of the individual plant examination submittals, if the staff finds on a plant individual basis areas that can -- that the staff thinks improvements should be pursued, then we have to exercise the backfit rule in that case.

MR. STELLO: Incidentally, with respect to the first issue, we have found that the utilities are in fact doing exactly what that says. As we've gone through with NUREG-1150, and in the process of doing an analysis, if an issue came up which suggests that there were things that could be done to solve a problem, the utility is moving forward expeditiously, making those changes themselves, putting in procedures. And the response of the utilities with respect to this kind of an effort seems to be very, very positive. So I put a lot of emphasis on that. I'm not so sure we're going to have very much of a problem. But wait and see. Based on what we see now, it looks good.

COMMISSIONER CARR: The second item says we're going to review all IPEs, and as I read it, they're going to be through in three years. How long is it going to take us to review them?

MR. SPIES: I have it someplace. No more than two years.

COMMISSIONER CARR: So it's five years total, probably.

MR. SPIES: Yes. On page 16, Mr. Chairman -- I'm sorry -- on page 17, I again summarize the closure process. I said at the beginning and I will say it again, in summary fashion, the steps which each utility is expected to take to achieve closure of the severe accident issues for its plants are completion of the individual plant examination, identification of potential improvements, evaluation and fix as appropriate or if necessary, develop and implement a framework for an accident management program that can accommodate new information as it develops, if we gain more understanding about how to improve the prevention or how to retain the core vessel or how to do other things that will improve safety, then this accident management program should be flexible enough to accommodate those strategies, and implement any generic requirements resulting from containment performance improvement, which, as I said earlier, it's very closely coupled to the IPE program itself.

Severe accident research will continue to confirm some of these judgment, and any new issues that arise in the future will be handled on a case-by-case basis.

On Table 1 on page 18 --

CHAIRMAN ZECH: Before you go to that, my review of your paper would lead me to believe that the Staff has, indeed, reasonably addressed the ACRS concerns and their suggestions that the IPE program may be broadened and combined with the ISAP program. I think you have addressed that properly.

But I guess my question would be, do you think there are some strong parts of the ISAP program which should be picked up in the IPE program, or are you still looking at that?

MR. STELLO: Well, the first part, I think, depends on the utility. I think the utility needs to make the commitment to want to go to the ISAP program themselves. I'm not so sure I see how you can stop short of doing the full process, which would be getting an integrated plan to make these improvements. So I think it really has to start with the utility. If they want to do it, and we can get the resources, I think it's a good idea.

If I were a utility executive, that's what I'd want to do.

CHAIRMAN ZECH: But I think the Staff should monitor that very carefully to make sure that we don't -- that we will agree that the decisions they are making are the appropriate

ones, and if there's any doubt, then you should get more heavily involved.

MR. STELLO: Absolutely.

CHAIRMAN ZECH: All right. Please proceed.

MR. SPIES: On the last page of the presentation, Mr. Chairman, I have a table in which I tried to summarize the individual pieces of information that will be reaching the Commission and on which of these items a Commission decision will be required. Starting with the individual plant examinations, the Commission paper will be reaching you in July of '88, and the Commission will be requested to approve the issuance of the generic letter initiating the IPEs.

Likewise on containment performance improvements, BWR MARK Is, we will be providing to you at the end of this month a status paper. We will be providing you our recommendations in the fall of '88, and for that issue, also we expect you to make a decision on our recommendations. Likewise for the other containment issues that will be coming to you by the end of next year.

On Item 3 and 4, improved plant operations and severe accident resource program, these are continuous efforts. We don't expect any actions on your part at this time.

On external events, we expect that you approve or you make a decision on our recommendations.

On accident management, there is -- will be a paper

for information now only. Tom, I missed if you said that there will be a Commission decision at that time on the accident management program. I guess we'll have to see as we -- we don't foresee it at this point in time, right?

NUREG-1150, no Commission decision is expected. There will be a paper for information only. Likewise for Issues 8 and 9, the generic safety issues and the integrated safety assessment program.

On advanced reactors, we will be coming to you. There are two types of advanced reactors, the LWRs and the non-LWRs, the DOE reactors, the HTGRs and the liquid metal reactors. On both of them, we will be coming to you with recommendations and expect you to make a decision on both of these -- in both of these areas.

And last, the safety goal policy implementation, a very important area, the Commission paper will be coming to you in August, and we would expect you to make a decision on our recommendations.

With that, Mr. Chairman, I think my presentation -- I've completed my presentation.

CHAIRMAN ZECH: Let me just make a comment or two on this last table. First of all, I expect that the Staff, as you continue to review containment performance and severe accident research and accident management and so forth and other generic issues, that if, during the course of your review, you

determine that something is more significant perhaps than you realize or something that should be done, you'll take action, or if you feel it should come to the Commission, I wouldn't want you to feel bound by the schedule.

In other words, if there is something that's safety significant, we want to know about it right away.

And the other comment I'd like to make is that I think that the schedule you've got, as near as we can tell, certainly on these very significant issues, looks to be, at least from my standpoint, reasonable, but in the column where you have expected Commission action as you go through and determine perhaps that you may want to change your mind about coming to the Commission, don't hesitate to come to the Commission at any time on any of those issues that you really think should come to the Commission as you proceed.

We want to be involved. If there's any doubt, come to the Commission.

All right. Is that --

MR. SPIES: That completes my presentation, Mr. Chairman.

CHAIRMAN ZECH: Mr. Stello, any other comments?

MR. STELLO: No.

CHAIRMAN ZECH: Then comments from my fellow Commissioners. Commissioner Roberts?

COMMISSIONER ROBERTS: Well, I only want to say I

agree with what you said in your opening remarks about you think the backfit rule is working and working well, and I agree with that, and I think it's good policy. And I think the backfit rule gives appropriate and adequate guidance to the Staff in making decisions on proposals for planned improvements. That's all I have to say.

CHAIRMAN ZECH: All right. Thank you. Commissioner Carr?

COMMISSIONER CARR: Yes, I have some comments on the accident management plan, and I'm a little curious as to what's going to be the definition of a satisfactory severe accident management framework that you're expecting the utilities to come in with. What are we talking about?

MR. SPIES: Mr. Commissioner, this is something we're discussing with the industry right now, and let me tell you what I meant by it. You know, we have the emergency operating procedures in place that you are very familiar with, so what we are talking about -- by the way, they go up to some point in these scenarios, like inadequate core cooling, even though for BWRs they go beyond into the venting area -- we're talking about, should we add, when we go beyond, to prevent core melt -- I mean, to contain core melt in the vessel or structure strategies to reduce its threat to the containment, are we going to put those procedures --

COMMISSIONER CARR: I understand that.

MR. SPIES: Together with the EOPs. Are we going to have them separately? Lots of things we are talking about.

COMMISSIONER CARR: I understand that part of it, but I'm a little curious about how they're going to close. You said one of the things they have to do for closure is to develop and implement a framework for an accident management program.

MR. SPIES: To commit. To commit to do an accident -- to have an accident management program. That goes beyond the existing --

COMMISSIONER CARR: It says develop and implement.

MR. STELLO: Let me try.

MR. SPIES: The framework is being developed now.

MR. STELLO: Let me try again. The procedures obviously have to be developed after you do the IPE, find out what the particular improvements are that can be made, what procedures ought to be implemented, and then go on and develop and implement them. I think that's what it says simply.

I'd use an easier example, I think, to understand. Direct heating at the containment, high-pressure injection of the corium. There are ways in which you can significantly change the course of that accident perhaps by opening the relief valves earlier in the transient, and that would be a severe accident procedure.

COMMISSIONER CARR: Which is going to depend on some

research.

MR. STELLO: Well, that work is ongoing right now, which we will have finished in time for them to make that judgment on the management.

COMMISSIONER CARR: I'm glad to hear you say that. I wish I had that same confidence.

MR. STELLO: I'm confident.

COMMISSIONER CARR: All right. Well, it seems to me that area is a little fuzzy, and maybe that's what you intended it to be, since we don't really know what we're going to require yet.

Okay. The other thing I'd like to recommend is, since you were mentioning that the generic letter on IPEs is going to formalize the Commission's approach to accident management, I'd like to see that accident management program plan paper before we approve the IPE letter, or I'd like to get in on that act somewhere, because it's going to a piece of something, and I don't know what we're saying.

So could you send those in at the same time? Can you separate the two?

MR. STELLO: I would prefer to have the Commission approve the accident management program, and we'll send that up for the Commission's approval to make sure it's okay. We're fairly close to having --

COMMISSIONER CARR: Well, one says July, and the

other one says fall. I don't know how close those things --

MR. STELLO: Well, the one that says July is --

COMMISSIONER CARR: One says July, and one says fall.

MR. STELLO: You got it.

MR. SPIES: There will be enough information in the IPE packet and accident management that hopefully it will satisfy you, Mr. Commissioner.

MR. STELLO: But if not, we would prefer to make the commitment to bring that up and have Commission approval at that point.

COMMISSIONER CARR: Well, you can see my problem.

MR. STELLO: Yes.

COMMISSIONER CARR: The other area, severe accident training, management training, I want to be sure we don't overblow that issue, you know. Most of the problems we've got come from training that leads to the severe accident or lack thereof. It's like teaching all drivers to be defensive drivers or handle quick reversals in their cars. We hope they never have to do that, but it's a handy thing to know once, but you don't have to train on it a lot and a long time.

I don't want to focus these people on managing severe accidents at the expense of running their plants day to day.

MR. STELLO: Absolutely.

COMMISSIONER CARR: They don't get enough practice in starting up and shutting down, in my opinion, because they are

designed to run X number of days without any problem. And so we don't want to overemphasize how much training we put on that area.

MR. SPIES: We agree fully with that. In fact, that's one of the things we are discussing with the industry. We're in total agreement with that.

CHAIRMAN ZECH: Anything else, Commissioner Carr?

COMMISSIONER CARR: No. I just -- do you think that schedule you've got there is reasonable, and you're going to make it?

MR. SPIES: Yes, yes. Most of this work --

COMMISSIONER CARR: I'll watch with interest. Thank you.

CHAIRMAN ZECH: Commissioner Rogers?

COMMISSIONER ROGERS: Well, I just want to say that I found this whole document a very interesting and comprehensive piece of work. There are a number of questions about details, how they will actually be carried out. But the attempt to bring this all together into what you would define to be closure, namely to close the loop, that you've gone through the whole cycle on this, I thought was an excellent approach, and I really want to commend the Staff for the clearly enormous effort of thought that has been required to bring it to this point. I think it really is very commendable.

I do have a number of little questions about things

along the way. Some of them have already been touched on. But if you would bear with me for a moment, I could perhaps try to go through my little list of questions quickly.

The Slide 4, Severe Accident Activities, it wasn't clear to me how you're going to effect closure between the efforts of the individual licensee in carrying out an IPE and the results that will be coming out of analysis or research activities that relate to containment performance and accident management and generic issues, how you get that back into the work of the licensee as those things proceed during the three-year period during which the licensee is developing the IPE.

So it wasn't clear to me just how that feedback mechanism or information flow or what your system is to get that out to individual licensees as they are proceeding with the development of their IPE. I wonder if you could just say a little bit about that?

Do you have a clear mechanism in place or in mind?

MR. MURLEY: Maybe I could take a cut at that. Closure as envisioned in this document and this chart is really a process. It's not going to be a package, let's say, that we come in with on each plant, so that --

COMMISSIONER ROGERS: No, I understand.

MR. MURLEY: Okay. What it envisions is that we will have addressed the major issues still open in severe accidents for each plant. This is meant to portray what those issues

are, and it's also contemplated that each plant will be different in terms of its vulnerability to severe accidents, so that, let's say, some MARK I plants may not have to do anything. Others may have to make some changes. Likewise with accident management.

Our intention would be that as we come to a resolution with the Commission of what's required to address the MARK I issue, for example, we would use the regulatory tools that we have, like generic letters or that sort of thing, to implement any additional requirements. So this program -- I'm not sure I'm getting at exactly your question, but this program envisions that over the next few years, there would be several regulatory requirements type documents that may go out. Each plant would be more or less susceptible to those, depending on how it relates to the issue that we're dealing with.

COMMISSIONER ROGERS: Well, I was just thinking of, you know, some way of providing ongoing communication with licensees as they are proceeding through the development of their IPEs as to the state of affairs of any issue that might be coming close to resolution that will affect their completion of their IPE. You know, it's kind of an intermediate feedback mechanism that is there.

MR. STELLO: NUMARC has organized itself to be able to interact in this area, and should that come to pass, that

will be the easiest and most direct way of having the results of research communicated to them and then in turn passed on.

COMMISSIONER ROGERS: I'm just concerned about someone essentially completing an IPE or being, you know, within six months of completing a three-year effort, and then all of a sudden there's a new finding that's coming out that now has to be accommodated in their plant, that if they were aware of, they could have rescheduled their efforts to accommodate more easily.

MR. STELLO: We hope that the industry will organize itself to do that, and if they do, that will be the easiest and most direct way. Otherwise, we will develop some mechanism of communicating the research results directly ourselves. I think it would be better for the industry if it did organize itself to be able to assimilate that kind of information and then reduce it to useful packages for the --

COMMISSIONER ROGERS: Well, presumably our efforts have to be easily accessible to that kind of --

MR. STELLO: Oh, we will definitely make that the case, make sure that happens.

COMMISSIONER ROGERS: In the full report, not your summary of it, Mr. Stello, but in the full report that we received, on page 15, in the reference to the ACRS' comments and your answers to those, there was a little point that bothered me a bit in an answer there. You say in reference to

the IPE generic letter, the letter states that if a utility concludes through the IPE and proposes to the Staff that -- one, two or three -- Item 2 is, there is not shown to be a cost-effective resolution to the issue, the Staff will consider the issue resolved upon review and acceptance of the IPE results.

That left some questions in my mind. This is page 15 of the full report, not Mr. Stello's comprehensive summary.

Just really what you had in mind there, because it looks as if that might just be -- that implied that if there isn't a cost-effective resolution to an issue, that it might somehow or other just get dropped.

I know that it doesn't necessarily say that, but are you following the --

MR. SPIES: Yes, I follow your --

COMMISSIONER ROGERS: And I'm just a little concerned about how you propose to deal with that, what that really means.

MR. SPIES: Well, here we're talking about a specific issue for a specific plant, and we're talking about Items 1, 2, and 3. They can go ahead and give us the information. They can go ahead and do the regulatory -- we have to pass judgment on that, okay? And that doesn't mean that, you know -- this implies that we will have reviewed that, and if we accept it, fine. If not, then we'll have to go through it farther. It

doesn't imply that we'll accept it at face value and write it off without reviewing their submittals. That's all it means.

MR. STELLO: I think in simple terms, the utility can, when they do their IPE program show that certain USIs and GSIs really don't apply to them, based on their unique situation, and find a way to close that issue on that plant at that time.

COMMISSIONER ROGERS: I'm worried about the "no cost-effective resolution," Item 2.

MR. STELLO: Yes.

COMMISSIONER ROGERS: Well, if they feel there's no cost-effective resolution of the issue, then how do we deal with that?

MR. SPIES: Well, we'll review it ourselves. We have a normal review process, and we'll either accept or reject. So that's the way it was meant anyhow.

MR. STELLO: The presumption is, we'll agree. If we don't agree, that may be one of the issues we'll bring to the Commission, if the utility proposed to do it that way, and we'll bring up to the Commission.

COMMISSIONER CARR: I have a little problem on that same page further down when you're talking about deterministic methods such as those being developed under the seismic design margin program, which may be just as effective in identifying vulnerabilities at a lower cost. I didn't understand that at

all.

MR. MURLEY: What is meant just to be sure that the lower cost is -- rather than doing a full PRA that includes all the latest seismic margins technology, there are some simpler methods that have been developed that we think can get -- can be just as effective.

COMMISSIONER CARR: It's only in the seismic area. Or there are a lot of areas like that?

MR. STELLO: No. No. We have several seismic programs that have been ongoing and have developed particular methods to assess the seismic capability of the facility. And it's in that context, saying -- those methods have --

COMMISSIONER CARR: You imply that those deterministic methods may extend across a further area than seismic.

MR. STELLO: No. It was limited with respect to the issue of the seismic program that's been underway.

COMMISSIONER CARR: All right.

COMMISSIONER ROGERS: I just come back to page 9 of your slides, the improved plant operations area. Just a comment, really, on your effort to improve technical specifications. I think that's a very important activity. It's one that I've been talking about the need to try to simplify, and weed out, and prune those things which really are not, which happen to be there historically, but are really not

necessary. Can you give me some idea of just what your level of effort will be in carrying out that part of the activity? How many people? How serious are you about that?

MR. MURLEY: Well, we're very serious about it. It's an ongoing program. I'd say we're probably two-thirds of the way through it now. We've sent out for owner's group comments, a document that proposes, which we've been working with -- this document proposes what can be left in the textbacks, what's important to safety, and what can be removed. And as a rough rule of thumb, it looks like, if these turn out to be accepted, and the staff goes along with it, it can remove, cut down on the number of LCL action statements by 40 percent, for example.

We have a branch that is, really one section of a branch, that's devoted full-time to carrying this out. My guess is it's probably 4 or 5 FTEs. So it is a substantial effort on the part of staff, and as I said, I think we're clearly, I can see the end is in sight.

COMMISSIONER CARR: Like when, do you think?

MR. MURLEY: I'd have to --

COMMISSIONER CARR: You've got farther vision than I have.

MR. MURLEY: It's a matter of like a year or two, rather than five, or six, as envisioned in this document here.

COMMISSIONER ROGERS: Also on that list, research to establish the sensitivity of risk to human errors, do you have

a common view within the staff and various areas, research, and NRR for example, of how to explicity take into account human factors probabilities into PRAs. Do you have a common view of that, or is it some -- this is a controversial --

MR. MURLEY: I can give you my --

COMMISSIONER ROGERS: This is a controversial area.

COMMISSIONER CARR: It was a little open in the human factors research program.

MR. MURLEY: I think that the models that we have today, in fact, are not adequate for us to determine the effect of recovery actions, just in one instance. Themmy mentioned that. And as a result, the PRAs don't give us the full benefit of improving human performance, because the models just don't allow it. Once the reactor reaches a certain state, in terms of emergency cooling pumps not working and the core starting to heat up, the models -- at least the older ones -- generally assume that goes directly to core melt. And we all know that there's many things that the operators can do at that stage. We don't have the models to actively quantify that.

A second area that I think we can work on is how to better estimate plant-specific human error rates. Generally, PRAs that are done today use a standard human error rate data. It's taken from a handbook that was developed some years ago, I believe for Air Force officers. It's meant to apply across the board, but in fact, every day we see wide variations from plant

to plant in human error rates. And that, I think, forms part of my reservations about using safety goals for plant-specific decisions. I think they're quite okay to use PRAs for a broad range of plants, to judge against safety goals. But when you get down to a specific plant, where the variation in human error can be substantially different from the average, we don't have the means for putting that into our models today.

COMMISSIONER ROGERS: Do you think there's some possibility of carrying that out?

MR. MURLEY: Well, I've asked that Research continue to develop these techniques and I believe they're doing it. So let me turn to Themmy on that.

Joe, do you want to say something?

CHAIRMAN ZECH: Yes, would you come to the microphone, please, and identify yourself for the reporter.

MR. MURPHY: Joe Murphy from the Office of Research.

I agree with what Dr. Murley has said. We have programs in place now that are going forward. The techniques in the more modern PRAs are the ones done in the last one or two years. We are taking into consideration some of the elements of difference in the nature of the procedures, the attitude of the operators; it's starting to be done, there's still a long way to go.

I think what we've seen is that there's promise that we can do this better. We have work ongoing in the field of

cognitive errors that the operator misinterprets the information coming in to him. Most of the earlier PRAs were on the order of procedural errors. The operator missed a step in this procedure or did something wrong in doing it, rather than that he misdiagnosed.

I think in the next year or two, we'll start along the line of being able to factor this in much better than we have in the past. And I think that with time, we'll have those in, but I agree with Dr. Murley, right now, we don't adequately include things like the influence of plant management on operator performance. The PRAs essentially assume that the operators are average, industry average operators.

MR. STELLO: I would want to emphasize while that those are important for the longer-term average, I think there are some short-term gains that we can make that will be important to safety. And that's in the area of looking at the LER data, the cost codes, what really, underlying reasons, whether it was poor training, poor procedures, or maintenance, or surveillance and testing, that perhaps should never even have been done during operation that caused the plant to be subjected to unnecessary trips and equipment failures. I think understanding the human element in that context, which I got the thrust was one of the major comments from the briefing from the Academy that the Commission had the other day. I think there we can probably do a great deal.

And there are significant differences, as you look country to country. For example, Japan and the United States. The number of challenges to the plant which is a direct coupling, if you will, to risk, I think can be significantly reduced just simply by being a little bit more careful about what people do.

COMMISSIONER ROGERS: Well, the point I was trying to get at was the quantification of these human factors in a PRA, whether, you know, there can be some way to -- with common agreement as to how to include that, because once you do that, then the PRA becomes a dynamic measure that can be looked at. It becomes an index in itself that changes. It's not just a number that happens to be calculated at a certain point in time for a plant and then goes up on a shelf and there it is forever. But is a dynamical measure of interest to us all. And I think that's something that's well to keep in mind. I recognize that it has certain tricky elements to it, but it is something that once one has a commonly-agreed upon methodology for including human factors in the PRA, then the PRA takes on a new significance --

MR. STELLO: I agree.

COMMISSIONER ROGERS: -- a new dynamical significance. I'll -- just one or two more things and I'll --

COMMISSIONER CARR: I'll piggyback on that a second. I think we'll get a lot of data out of that so-called research

and assimilator programs where we permit operators to take actions that either prevent or mitigate the accident, and give them credit for that. If they do it wrong, we don't really go that far and we don't let them take the action because it never works and because otherwise you don't have a drill. But I think we can get some data in those areas.

COMMISSIONER ROGERS: And coming to NUREG-1150, on page 13, you cite that. And just to bring us up to date on what your thinking is with respect to direct comparison between the technology or techniques of NUREG-1150 and the analysis of WASH-1400, with respect to those two specific plants, Peach Bottom and Surrey, it would be very nice to be able to have those direct comparisons of the new technology under NUREG-1150 applied to those same situations. I know that you mentioned that there have been improvements, but that was -- seemed to be a somewhat qualitative statement with respect to changes in those two plants based on NUREG-1150 rather than on the same kind of analysis that was carried out for the five plants to be carried out in comparison with the results of WASH-1400 for those two plants.

MR. STELLO: We will have those comparisons directly in NUREG-1150. Yes, they will be.

COMMISSIONER ROGERS: Under the full methodology. I see. All right. Because that was the question.

MR. SPEIS: At least the most important ones. The

ones that really make the difference, between then and now. That would involve --

COMMISSIONER ROGERS: Because I think that's very interesting.

MR. SPEIS: -- improvements in data, improvements in methodology, or whatever, you know, made the difference between then and now.

COMMISSIONER ROGERS: Do you expect to have any additional external review of the revised version of 1150?

MR. STELLO: My view at the moment is that we need to get the document out and be able to use it. And we've had extensive review of that document. Eric Beckjord has a meeting scheduled later this month, to revisit that issue with some people from the Academy and to discuss it further. We have not made a final decision, but I lean very heavy to let's finalize it, put it out. I think it's the kind of a document that you're going to update again in a few years. There's a lot of activity that's going on and I think we have just got to find a time to say, "This is enough. Let's get it out and use it for the purposes we can," and stop. That's my view at the moment. But we're still looking at it.

COMMISSIONER ROGERS: Well, with so much learned from the exposure of the document to comment around the world that in incorporating that all into a new document, there's still -- it will be a new document.

MR. STELLO: Agreed.

COMMISSIONER ROGERS: And --

MR. STELLO: But if we continue the cycle, we're never going to get anything on the street that we can really use, so I think --

COMMISSIONER ROGERS: It's a decision that has to be made.

MR. STELLO: I think my inclination would be to just get it on the street and of course, further comments, further review would be desired and be used to help us guide in the next time we visit this issue, which is clearly going to be in several more years. But we're already several years into trying to get this version out. We just have to stop at some point.

COMMISSIONER ROGERS: Just one final one. On your table 1at the end of, in the schedule, page 18, slide 18. The integrated safety assessment program, you didn't expect to bring that to the Commission. I think that's something that conceivably might have elements that the Commission would like to respond to. It has -- it's not quite as comprehensive as the IPEs, I guess, but it would seem to me that there might be some issues there that Commission action might be called for. That's item 9.

MR. STELLO: Okay. Well, it will be coming to the Commission, because the way we presented it in the paper that

you have before you, what we have said is that we are making the ISAP a program if utilities wish to come forward, our view is that to the extent we can have the resources, we'd like to do that. And we have not concluded or suggested that it ought to be made mandatory. In fact, we have recommended against making it a mandatory program.

COMMISSIONER ROGERS: Well, I just want to thank the staff for an excellent presentation. I found it very informative. Very helpful.

CHAIRMAN ZECH: But again, you are going to send that ISAP paper to me. I think you say that. And we will have it for action as we see fit.

MR. STELLO: Right.

CHAIRMAN ZECH: Well, let me just say -- Are there any other comments, my fellow Commissioners? Let me just say, I, too, compliment the staff on a very important and difficult task of bringing together all of these various programs and policies. What we're trying to do is build a structure, an integrated structure, out of the various pieces we have. And I think that's extremely important. Because we must put, in my judgment, consistency, predictability, common sense, and analytical credibility into the structure that we're building. The pieces should fit together. We recognize that they overlap in some areas. There are judgments to be made. But I think the effort to bring it together is going to be extremely

helpful to our entire regulatory process, and to public health and safety, and to our decisions in that regard. So I, too, compliment the staff on an extremely difficult task.

But, Mr. Stello, you remember we talked about this some time ago. I've felt, and I know my colleagues have agreed that the effort to bring these various programs together in some kind of a unified structure will enable us to make, the staff to make better recommendations, the Commission to make better decisions. So it's an extremely important endeavor. And none of us belittle the inability perhaps to get more exact judgments than you've come up with. I think you've done an outstanding job of trying to pull together these policy statements.

But the action itself, the effort itself, I think is extremely important. The results, of course, are terribly important, too. But the effort you've made to pull together, I'm sure, has made you question and look at some of the overlapping areas that will cause further analysis, perhaps. But in any case, I think you've done an excellent job, and I commend the staff for that.

My only last point would be to emphasize what I've said before and what others have said, too, and I think you sensed the Commission's desire to be very much involved in this process. And whenever there's any doubt in your mind -- these are important policy matters -- that they should be brought to

the Commission's attention. So I'd like to make that that last emphasis that we do feel that these matters we're discussing have tremendous impact on public health and safety, in the broad sense, and will be watched very closely, I'm sure, by other nations that we should make as good a decision on these matters as we possibly can. And so we do feel the Commission should be involved, and if there's any doubt in your mind at all about this, come to us for guidance, either informally or formally, and we want to keep involved in the whole process as it proceeds.

Again, let me compliment the staff on an excellent job, and let's continue working closely with the Commission in this regard.

No other comments? We stand adjourned. Thank you very much.

[Whereupon, at 11:35 a.m., the proceedings were adjourned.]

CERTIFICATE OF TRANSCRIBER

This is to certify that the attached events of a meeting of the U.S. Nuclear Regulatory Commission entitled:

TITLE OF MEETING: BRIEFING ON SEVERE ACCIDENT INTEGRATION PLAN

PLACE OF MEETING: Washington, D.C.

DATE OF MEETING: THURSDAY, JUNE 2, 1988

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.

Suzanne Young

Ann Riley & Associates, Ltd.

COMMISSION BRIEFING

ON

SEVERE ACCIDENT INTEGRATION PLAN

THEMIS P. SPEIS
(301) 492-3710
OFFICE OF RESEARCH
U. S. NUCLEAR REGULATORY COMMISSION
JUNE 2, 1988

BACKGROUND

- o AUGUST 8, 1985, "SEVERE ACCIDENT POLICY STATEMENT," 50 FR 32138
- o FEBRUARY 28, 1986, "IMPLEMENTATION PLAN FOR SEVERE ACCIDENT POLICY STATEMENT," SECY-86-76
- o DECEMBER 1, 1986, STAFF REQUIREMENTS MEMO, COMMISSION REQUESTED PAPER ON INTEGRATION OF SEVERE ACCIDENT ISSUES
- o FEBRUARY 17, 1987, MEMO FROM EDO TO COMMISSIONERS ON PRELIMINARY PLAN FOR INTEGRATION OF SEVERE ACCIDENT ISSUES
- o JULY 15, 1987, STAFF BRIEFED COMMISSION ON A PLAN FOR CLOSURE OF SEVERE ACCIDENT ISSUES
- o DECEMBER 8, 1987, "MARK I CONTAINMENT PERFORMANCE PROGRAM PLAN," SECY-87-297
- o FEBRUARY 9-11, 1988, BALTIMORE MANAGEMENT MEETING ON SEVERE ACCIDENT ISSUES

SEVERE ACCIDENT INTEGRATED PLAN

- PURPOSE: TO PRESENT STAFF'S PLAN FOR INTEGRATION AND CLOSURE OF SEVERE ACCIDENT ISSUES

- OBJECTIVES:

  - TO PROVIDE AN UNDERSTANDING OF THE STAFF ACTIVITIES THAT ARE UNDER WAY TO IMPLEMENT THE COMMISSION'S SEVERE ACCIDENT POLICY

  - TO ASSURE THAT THESE ACTIVITIES ARE CONSISTENT WITH THE COMMISSION'S POLICY AND STRATEGIC GOALS

  - TO ASSURE THAT THE STAFF ACTIVITIES ARE CONSISTENT AMONG THEMSELVES, HAVE A COMMON GOAL OF ULTIMATELY LEADING TO IMPROVED PLANT SAFETY, AND ARE PROPERLY COORDINATED AMONG THE RESPONSIBLE NRC ORGANIZATIONS

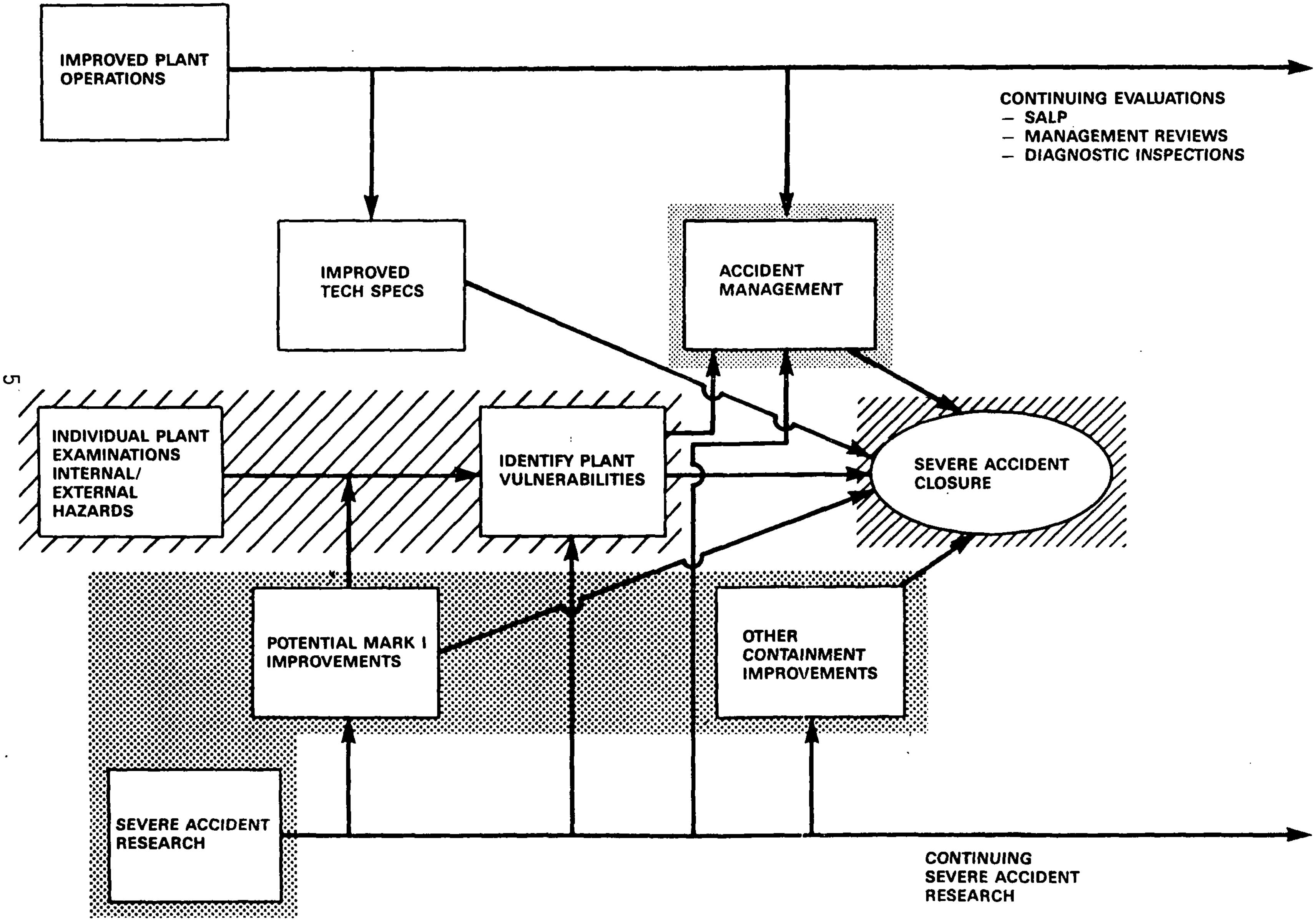
  - TO ASSURE THAT THE COMMISSION IS AWARE OF THE KEY TECHNICAL AND POLICY ISSUES, SOME OF WHICH WILL NEED COMMISSION GUIDANCE OR APPROVAL

  - TO DESCRIBE THE USE OF SAFETY GOALS AND BACKFIT POLICY IN THE CLOSURE PROCESS

SEVERE ACCIDENT ACTIVITIES

- INDIVIDUAL PLANT EXAMINATIONS (IPE)
- CONTAINMENT PERFORMANCE IMPROVEMENTS (CPI)
- IMPROVED PLANT OPERATIONS (IPO)
- SEVERE ACCIDENT RESEARCH PROGRAM (SARP)
- ACCIDENT MANAGEMENT (AM) PROGRAM
- NUREG-1150
- GENERIC SAFETY ISSUES
- EXTERNAL EVENTS
- INTEGRATED SAFETY ASSESSMENT PROGRAM (ISAP)
- SEVERE ACCIDENT POLICY FOR FUTURE PLANTS
- SEVERE ACCIDENT CLOSURE/USE OF SAFETY GOAL

FIGURE 1
SEVERE ACCIDENT PROGRAM - SCHEMATIC



IPE SUMMARY

- 10 CFR 50.54(F) LETTER REQUESTING IPEs

- BASIS: COMMISSION SEVERE ACCIDENT POLICY ISSUED ON AUGUST 8, 1985 (50 FR 32138)

  - PLANT-SPECIFIC PRAs EXPOSED RELATIVELY UNIQUE VULNERABILITIES TO SEVERE ACCIDENTS

  - UNDESIRABLE RISK CAN BE REDUCED BY LOW-COST CHANGES VIA PROCEDURES OR MINOR DESIGN MODIFICATION

  - ANALYSIS WILL BE MADE OF PLANTS THAT HAVE NOT UNDERGONE AN APPROPRIATE EXAMINATION

- STAFF HAS INTERACTED EXTENSIVELY WITH INDUSTRY AND DEVELOPED APPROPRIATE DOCUMENTATION FOR THE IPEs

- PURPOSE OF IPE IS FOR THE UTILITIES TO:

  - IDENTIFY/UNDERSTAND THE MOST LIKELY SEVERE ACCIDENT SEQUENCES THAT COULD OCCUR AT THEIR PLANTS

  - EVALUATE/IMPLEMENT MEANS FOR IMPROVEMENTS

  - DEVELOP AN AWARENESS FOR SEVERE ACCIDENT BEHAVIOR

IPE EXAMINATION PROCESS

- o LICENSEE'S STAFF SHOULD PARTICIPATE IN ALL ASPECTS OF THE IPE SO THAT KNOWLEDGE GAINED BECOMES AN INTEGRAL PART OF OPERATING, TRAINING AND PROCEDURE PROGRAM

- o LICENSEES SHOULD CONDUCT SYSTEMATIC EXAMINATION OF PLANT DESIGN, OPERATION, MAINTENANCE AND EMERGENCY OPERATION TO:

  - IDENTIFY PLANT-SPECIFIC VULNERABILITIES (DESIGN AND PROCEDURAL) TO SEVERE ACCIDENTS (FOR BOTH CORE DAMAGE AND CONTAINMENT PERFORMANCE)

  - QUANTIFY RESULTS OF EXAMINATION FOR THE SEQUENCES THAT CONTRIBUTE THE MOST TO THE TOTAL CORE DAMAGE OR LARGE RELEASE FREQUENCY

  - UNDERSTAND WHAT COULD PROBABLY GO WRONG IN A PLANT

  - IDENTIFY AND EVALUATE MEANS FOR IMPROVING PLANT/CONTAINMENT PERFORMANCE

  - DECIDE WHICH IMPROVEMENTS WILL BE IMPLEMENTED AND SCHEDULE FOR IMPLEMENTATION

CONTAINMENT PERFORMANCE IMPROVEMENTS

- o SOME CONTAINMENTS POTENTIALLY VULNERABLE TO EARLY FAILURE DURING SEVERE ACCIDENT (DRAFT NUREG-1150)

- o EVALUATING GENERIC CHALLENGES, FAILURE MODES & POTENTIAL IMPROVEMENTS

- o STATUS FOR MARK Is:

  - APPROACH BEING PURSUED INVOLVES BOTH ACCIDENT PREVENTION AND MITIGATION

  - ADDITIONAL SOURCES OF WATER BEING EXPLORED FOR CORE COOLING, CONTAINMENT AND DEBRIS COOLING, AND FISSION PRODUCT SCRUBBING

  - ADS RELIABILITY ENHANCEMENT

  - VENTING UTILIZING SUPPRESSION POOL FOR SCRUBBING USEFUL, BUT DOWNSIDES SHOULD BE MINIMIZED

  - REGULATORY ANALYSES OF ABOVE BEING PERFORMED

- o MARK I INTERIM AND FINAL RECOMMENDATIONS DUE TO COMMISSION BY JULY AND FALL OF '88, RESPECTIVELY

- o RECOMMENDATIONS FOR OTHER CONTAINMENT TYPES DUE TO COMMISSION BY FALL '89

IMPROVED PLANT OPERATIONS

- MOST PRAs HAVE SHOWN SENSITIVITY OF RISK TO HUMAN ERRORS
- STAFF ANALYSES OF OPERATING EXPERIENCE CONFIRMS THE IMPORTANCE OF REDUCING MAINTENANCE, SURVEILLANCE, TESTING & CONTROL ROOM OPERATOR ERRORS
- THEREFORE, STAFF'S PROGRAM TO IMPROVE PLANT OPERATIONS CONSIDERS THE FOLLOWING EFFORTS:
    - CONTINUED IMPROVEMENT OF THE SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE (SALP) PROCESS
    - REGULAR REVIEWS BY SENIOR NRC MANAGERS OF PROBLEM PLANTS
    - TEAM INSPECTIONS TO PROBE FURTHER THE PERFORMANCE OF THOSE PLANTS IDENTIFIED IN ITEM 2
    - REGULATORY ACTIONS TO IMPROVE OPERATIONAL PERFORMANCE WHERE IT HAS FALLEN BELOW EXPECTED STANDARDS
    - IMPROVED TECHNICAL SPECIFICATIONS
    - CONTINUED IMPROVEMENT OF OPERATING PROCEDURES
    - EXPANDING EOPs TO INCLUDE GUIDANCE ON SEVERE ACCIDENT MANAGEMENT STRATEGIES
    - RESEARCH TO ESTABLISH THE SENSITIVITY OF RISK TO HUMAN ERRORS

SEVERE ACCIDENT RESEARCH

- BEGINNING IN 1980, AFTER THE TMI-2 EVENT, RESEARCH HAS PROVIDED A DATA BASE AND MODELS FOR:

  - o FISSION PRODUCT RELEASE, TRANSPORT, DEPOSITION, & REVAPORIZATION
  - o CONTAINMENT LOADING BY HIGH PRESSURE MELT EJECTION (HPE)
  - o HYDROGEN DETONATION AND BURNING
  - o CORE/CONCRETE INTERACTIONS (CCI)
  - o CONTAINMENT PERFORMANCE TESTING
  - o EFFECTS OF NATURAL CIRCULATION ON THE PRIMARY SYSTEM
  - o CORE MELT PREGRESSION (EARLY STAGES)

- FUTURE RESEARCH EFFORTS WILL FOCUS ON SPECIFIC ISSUES SUCH AS:

  - o CONTAINMENT FAILURE PROBABILITY BY DIRECT CONTAINMENT HEATING (DCH) INCLUDING EFFECT OF NATURAL CIRCULATION
  - o MELT SPREADING AND POTENTIAL CONTAINMENT SHELL FAILURE IN MARK Is
  - o RESEARCH DATA AND MODELS TO ASSESS ACCIDENT MANAGEMENT STRATEGIES
  - o LONGER TERM CONFIRMATORY RESEARCH ON:

    - DCH CONSEQUENCES
    - REFINEMENT OF HYROGEN BEHAVIOR MODELS
    - CORE MELT PROGRESSION (LATE STAGES)
    - CORE/CONCRETE INTERACTIONS
    - FURTHER MODEL ASSESSMENT AND REFINEMENTS

AN EXAMPLE OF AN ISSUE AND ITS ASSOCIATED NEAR AND LONG-TERM RESEARCH

CONTAINMENT TYPE	LARGE DRY PWR
ASSOCIATED ISSUES	o POTENTIAL CONTAINMENT FAILURE MODES - DIRECT CONTAINMENT HEATING (DCH) - HYDROGEN BURN/DETONATIONS - LATE FAILURE BY CCI LOADS (OVER T&P) o CONTAINMENT PERFORMANCE o ACCIDENT MANAGEMENT STRATEGIES - DEPRESSURIZATION OF PRIMARY SYSTEM
RESEARCH TO ADDRESS ISSUE	o DCH - PROBABILITY OF HIGH PRESSURE MELT EJECTION (NATURAL CIRCULATION) - CUTOFF PRESSURE FOR HPE - MANAGEMENT THROUGH DEPRESSURIZATION - CONSEQUENCES - ANALYSES OF PERFORMANCE TESTS ON CONCRETE CONTAINMENTS (1/6 SCALE)

ACCIDENT MANAGEMENT

- ACCIDENT MANAGEMENT INCLUDES THE MEASURES TAKEN BY THE PLANT OPERATING AND TECHNICAL STAFF TO (1) PREVENT CORE DAMAGE, (2) TERMINATE CORE DAMAGE IF IT OCCURS AND RETAIN THE CORE WITHIN THE REACTOR VESSEL, (3) FAILING THAT, MAINTAIN CONTAINMENT INTEGRITY AS LONG AS POSSIBLE, AND FINALLY (4) TO MINIMIZE THE CONSEQUENCES OF OFFSITE RELEASES

- ACCIDENT MANAGEMENT ENCOMPASSES HARDWARE, HUMAN, AND ORGANIZATION FACTORS

- IT PROVIDES DECISION MAKERS AT THE PLANT A STRUCTURED PROGRAM FOR MANAGING A SEVERE ACCIDENT

- PROPOSED IPE GENERIC LETTER ADDRESSES ACCIDENT MANAGEMENT AS FOLLOWS:

  - UTILITIES ARE EXPECTED TO DEVELOP AN ACCIDENT MANAGEMENT PROGRAM FOR PREVENTION OR MITIGATION OF RISK IMPORTANT SEVERE ACCIDENTS

  - IDENTIFY MEASURES THAT PLANT PERSONNEL CAN AND SHOULD TAKE IN CASE OF SEVERE ACCIDENT. ASSESS AGAINST THE CRITERIA OF 10 CFR 50.59 AND, IF APPROPRIATE, SUBMIT FOR NRC REVIEW IN ACCORDANCE WITH 10 CFR 50.90

ROLE OF NUREG-1150

OBJECTIVE:

PROVIDES A SNAPSHOT OF THE STATE-OF-THE-ART PRA TECHNOLOGY, INCORPORATING IMPROVEMENTS IN METHODS AND DATA ACCUMULATED SINCE WASH-1400; INCLUDES EXAMINATION OF SEVERE ACCIDENT FREQUENCIES AND RISKS AND THEIR ASSOCIATED UNCERTAINTIES FOR FIVE PLANTS.

ROLES IN REGULATORY PROCESS:

- INDEPENDENT STAFF ASSESSMENT OF RISKS
- TECHNICAL DATA BASE AS INPUT TO:
  - IPE
  - ACCIDENT MANAGEMENT
  - CONTAINMENT PERFORMANCE INITIATIVES
  - SAFETY GOAL IMPLEMENTATION
  - GENERIC ISSUE RESOLUTION
- PRIORITIZATION/FOCUS OF RESEARCH

SEVERE ACCIDENT CLOSURE

- o DEFINES FOR INDUSTRY:

  - THOSE PROGRAMS WHICH ARE CRITICAL TO RESOLVING SEVERE ACCIDENT ISSUES FOR THEIR PLANTS

  - SPECIFIC STEPS THAT MUST BE TAKEN TO ACHIEVE RESOLUTION

- o ACTIONS MUST BE TAKEN FOR GENERIC ISSUES & PLANT-SPECIFIC ISSUES

- o CLOSURE FOR GENERIC ISSUES RESULTS WHEN THE COMMISSION EITHER TAKES ACTION IN THE FORM OF RULEMAKINGS, OR EXPLICITLY STATES WHATEVER ITS REQUIRED APPROACH IS; AND

- o CLOSURE FOR PLANT-SPECIFIC ISSUES RESULTS WHEN EACH UTILITY HAS COMPLETED CERTAIN EVALUATIONS & IMPLEMENTED CERTAIN PROGRAMS SUCH THAT EVENTS WHICH COMPRISE DOMINANT CONTRIBUTIONS TO RISK ARE IDENTIFIED AND PRACTICAL IMPROVEMENTS (DESIGN, PROCEDURES, OPERATION) ARE MADE TO REDUCE PROBABILITY OF THESE CONTRIBUTORS OR THEIR CONSEQUENCES TO ACCEPTABLY LOW VALUES

USE OF SAFETY GOAL IN CLOSURE PROCESS

- SAFETY GOALS AND OBJECTIVES WILL BE USED ONLY FOR RESOLUTION OF GENERIC ISSUES: RESOLUTION OF PLANT-SPECIFIC ISSUES WILL BE ACCOMPLISHED THROUGH IPEs AND PLANT-SPECIFIC BACKFIT CRITERIA

- SAFETY GOAL POLICY IMPLEMENTATION PLAN WILL BE PUBLISHED IN AUGUST 1988 AND WILL ADDRESS COMMISSION CONCERNS AND ACRS RECOMMENDATIONS

- ALL PRA INFORMATION WILL BE USED TO MAKE COMPARISONS WITH APPLICABLE SAFETY GOAL OBJECTIVES

- STAFF WILL DETERMINE WHY CERTAIN CLASSES OF PLANTS DON'T MEET S.G. OBJECTIVES AND ASSESS REASONS RELATIVE TO CURRENT REGULATORY REQUIREMENTS

- SUCH ASSESSMENT WILL TEST EFFECTIVENESS OF PRESENT REQUIREMENTS AND MAY RESULT IN CHANGES TO REGULATIONS AND SAFETY ENHANCEMENTS FOR SOME PLANTS

- SAFETY ENHANCEMENTS WILL BE SUBJECTED TO REGULATORY ANALYSIS PER COMMISSION'S BACKFIT RULE (10 CFR 50.109) AND PROPOSED TO COMMISSION FOR BACKIT IN THE FORM OF RULEMAKING

- FIRST APPLICATION WILL BE REFLECTED IN STAFF RECOMMENDATIONS TO COMMISSION ON BWR MARK I CONTAINMENT PERFORMANCE IMPROVEMENTS IN THE FALL OF 1988

- IF APPROPRIATE ANY PROPOSED PLANT AND OPERATIONAL IMPROVEMENTS WHICH CAN NOT BE JUSTIFIED BY THE BACKFIT RULE WILL BE PROVIDED TO THE COMMISSION WITH THE STAFF'S RECOMMENDED ACTION ON A CASE-BY-CASE BASIS

RESOLUTION OF PLANT UNIQUE ISSUES

- o IN PERFORMING IPEs, UTILITIES MAY FIND, AND IT IS EXPECTED THAT THEY WILL VOLUNTARILY REMEDY UNCOVERED OUTLIERS, AND MAKE ANY IMPROVEMENTS THEY DEEM APPROPRIATE, CONFORMING, HOWEVER, TO 10 CFR 50.59

- o HOWEVER, THROUGH THE REVIEW OF IPE SUBMITTALS, THE STAFF MAY FIND IT NECESSARY TO EMPLOY PLANT-SPECIFIC BACKFIT POLICY TO ASSURE THAT JUSTIFIABLE PLANT-SPECIFIC IMPROVEMENTS ARE MADE

CLOSURE PROCESS

- o STEPS WHICH EACH UTILITY IS EXPECTED TO TAKE TO ACHIEVE CLOSURE OF SEVERE ACCIDENT ISSUES FOR ITS PLANTS ARE:

  - COMPLETE THE IPEs, AND IDENTIFY POTENTIAL IMPROVEMENTS, EVALUATE AND FIX AS APPROPRIATE

  - DEVELOP AND IMPLEMENT A FRAMEWORK FOR AN ACCIDENT MANAGEMENT PROGRAM THAT CAN ACCOMMODATE NEW INFORMATION AS IT DEVELOPS

  - IMPLEMENT GENERIC REQUIREMENTS RESULTING FROM CONTAINMENT PERFORMANCE IMPROVEMENTS PROGRAM

- o SEVERE ACCIDENT RESEARCH WILL CONTINUE TO CONFIRM JUDGMENTS: NEW ISSUES WILL BE HANDLED ON A CASE-BY-CASE BASIS

TABLE 1

Schedule and Expected Commission Actions
for Severe Accident Issues

Item	Schedule	Expected Commission Action
1. Individual Plant Examinations	Commission Paper due July, 1988	Commission will be requested to approve issuance of generic letter initiating IPEs
2. Containment Performance Improvements		
a) BWR MARK Is	Interim recommendations due June, 1988 and Commission Paper with final recommendations due Fall, 1988	Approve staff recommendations
b) Other Containment Types	Commission Paper due August, 1989	Approve staff recommendations
3. Improved Plant Operations	Continuous	None at this time
4. Severe Accident Research Program	Continuous	None at this time
5. External Events	Commission Paper due end of FY 89	Approve staff recommendations for treatment of external events in severe accident policy implementation
6. Accident Management Program	Commission Paper due Fall, 1988	None--Paper for information only
7. NUREG-1150	Commission Paper scheduled for January, 1989	None--Paper for information only
8. Generic Safety Issues	Continuous	None at this time
9. Integrated Safety Assessment Program	Commission Paper due June, 1988	No action anticipated at this time
10. Advanced Reactors		
a) LWRs	Commission Paper due July, 1988	Approve staff recommendations
b) Non-LWRs	Commission Paper due June, 1988	Approve staff recommendations
11. Safety Goal Policy Implementation	Commission Paper due August, 1988	Approve staff recommendations

May 25, 1988

POLICY ISSUE
(Information)

SECY-88-147

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: INTEGRATION PLAN FOR CLOSURE OF SEVERE ACCIDENT ISSUES

Purpose: The purpose of this paper is to present the staff's plan for integration and closure of severe accident issues. This paper is for information only. There are no policy decisions that need to be made as part of this paper. Basically, this paper describes all of the information that the Commission will be receiving in the near future that will lead to closure of severe accident issues for operating reactors. More specifically, the main and supporting elements of the plan will be described, as well as how closure is to be achieved. Those elements of the plan which will require future Commission policy decisions to achieve closure are identified and discussed. In addition, this paper describes the interrelationship among these elements to assist the Commission in assessing the policy implications of each element. Finally, severe accident activities for advanced reactors are discussed.

The six main elements of the plan which will be discussed are: Individual Plant Examinations, Containment Performance Improvements, Improved Plant Operations, Severe Accident Research Program, External Events and Accident Management. The supporting elements which will be discussed include: Reactor Risk Reference Document (NUREG-1150), Safety Goals, Generic Safety Issues, and the Integrated Safety Assessment Program. With respect to future plants, the major activity which will be presented includes Severe Accident Policy for Advanced Reactors. Before a discussion of each of the main and supporting activities is presented, the main objectives, key background material, and an overall summary are provided.

Contact:
T. Speis, RES, 492-3710
B. Sheron, RES, 492-3500
A. Marchese, RES, 492-3554

Objectives: The principal objectives of this paper are:

1. To describe the main and supporting elements of the closure plan, including the use of safety goals and backfit policy in the closure process.

2. To provide an understanding of the staff activities that are under way to implement the Commission's severe accident policy.

3. To assure that these activities are consistent with the Commission's policy and strategic goals.

4. To assure that the staff activities are consistent among themselves, have a common goal of ultimately leading to improved plant safety, and are properly coordinated among the responsible NRC organizations.

5. To inform the Commission of the key technical activities and policy issues that need to be addressed in the near future.

Background: Severe accident evaluations and research progressed to the point that the Commission issued a Severe Accident Policy Statement (50 FR 32138) on August 8, 1985, which concluded that existing plants posed no undue risk to the public. However, based on NRC and industry experience with plant-specific Probabilistic Risk Assessments (PRAs), the Commission recognized that systematic examinations would be beneficial in identifying plant-specific vulnerabilities to severe accidents for which further safety improvements may be justified.

On February 28, 1986, the staff issued an implementation plan for the Severe Accident Policy Statement (SECY-86-76). Additional information was provided to the Commission relative to severe accident implementation in the following two areas:

1. Introduction of realistic source terms into licensing (SECY-86-228).

2. Treatment of external events in severe accident considerations (SECY-86-162).

On December 1, 1986, in a Staff Requirements Memorandum, the Commission requested the staff to develop a paper outlining how severe accident issues are being integrated into a consistent program. In a February 17, 1987 memorandum from the EDO to the Commissioners, the staff presented its preliminary plan for integration of some of the severe accident issues.

On July 15, 1987, the staff briefed the Commission on a plan for closure of severe accident issues, including matters relating to BWR MARK I containments. At the July 15th Commission briefing, the staff indicated its intent to pursue an integrated approach to the resolution of severe accident issues. Subsequently, in SECY-87-297, dated December 8, 1987, the staff presented its plans to resolve severe accident issues relating to containment performance, starting with MARK I plants.

The severe accident closure implementation program has been clarified and upgraded and now consists of the following six major elements:

1. Examination of existing plants for severe accident vulnerabilities.

2. Development of generic containment performance improvements with respect to severe accidents to be implemented if necessary for each of the six containment types.

3. Upgrading of staff and industry programs to improve plant operations.

4. A severe accident research program.

5. A program to define how and to what extent vulnerabilities to severe accidents from external events need to be included in the severe accident policy implementation.

6. A program to ensure that licensees develop and implement severe accident management programs at their plants.

This paper describes the staff plans for integration of these six main and three supporting elements that will lead to closure of severe accident issues for operating plants.

Summary:

Improved Plant Operations Since TMI-2

The Commission's approach to safety was dramatically changed following the TMI-2 accident. Emphasis shifted from providing safety by relying on the traditional design basis approach to a multifaceted approach which emphasized improved operations, human factor considerations, realistic performance of systems, and probabilistic risk assessments. By reviewing the operational history of the commercial nuclear plants since TMI-2, noticeable improvements in plant performance are readily evident from this regulatory shift in emphasis. We have seen an overall improvement in operational performance based on data obtained from Licensee Event Reports (LERs), and an overall

improvement in Systematic Assessment of Licensee Performance (SALP) ratings. Our willingness to shut down plants because of safety concerns emanating from poor management has resulted in a positive response by the industry to improve management performance at the plants. Hence, while many of the programs being conducted by the staff deal with understanding severe accident phenomena and examining the risk contribution from systems and components, improving plant operations is an area in which actions resulting in substantial risk reductions are possible and are being taken. It is also an area, however, in which risk can be increased when plant operations decline in quality. For this reason, emphasis on improving plant operations is a critical element of our overall closure plan for severe accidents as discussed later in this paper.

State of Technology

In 1975, the first comprehensive examination of the risk of light water reactors, the Reactor Safety Study (WASH-1400), was issued. This study, as well as many PRAs performed since in the U.S. and elsewhere in the world, concluded that core damage and core-melt accidents represent the major contribution to the residual risks from commercial nuclear power plants. However, those residual risks were found to be relatively small compared to most other risks encountered by the public in their day-to-day activities. The WASH-1400 results indicated that the conditions that could develop in such severe accidents as a result of the thermal-hydraulic-material interactions which take place could be more severe than those chosen for design basis accidents, and that failure of the containment, in some instances early into the accident phase, was a possible outcome of such accidents.

The NRC began to give added attention to severe accidents even before the accident at Three Mile Island (TMI). However, after the accident, the scope and diversity of the Commission's efforts in this area increased substantially. Some examples include allocating additional resources to R&D activities, including the construction of new experimental facilities, development of analytical tools, and in general, the development of the information needed to gain more knowledge and insights about severe accident behavior. Also, the Commission initiated and implemented new rules dealing with some aspects of severe accidents (i.e., accidents beyond the original design basis of the plant, but still assuming the retention of the degraded core in-vessel). For example, the NRC promulgated a rule (10 CFR 50.44) dealing with the generation of large quantities of hydrogen in the three types of BWRs and in Ice Condenser PWRs. Since then, the NRC, and to a lesser extent the industry as well, has developed a large body of information and data on severe accidents, including the

probability of severe accidents, core-melt phenomenology, associated accident sequences, and effects of severe accidents on plant systems, components and structures, especially those that provide barriers to fission product release to the atmosphere such as the containment.

This information has improved our understanding of what could go wrong in a nuclear power plant during severe accidents, as well as ways to reduce or eliminate their consequences. As is pointed out in the Enclosure, which describes in more detail the Severe Accident Research Program (SARP), there are still some severe accident phenomena for which additional research is needed to help reduce the wide range of uncertainty associated with their likelihood or their effect on containment integrity. However, issues associated with such identified phenomena are amenable to technical solutions using existing technology. In addition, the available information base is solid and broad enough so that a range of regulatory decisions regarding severe accidents can begin to be made. Therefore, the staff believes that implementation of the Commission's policy on severe accidents can be initiated. Accordingly, the staff's Severe Accident Integration Plan is described in the following sections.

Overview of Integration Plan

The Severe Accident Integration Plan provides for coordinated efforts to ensure fulfillment of the Commission's Severe Accident Policy Statement. An overall schematic illustration of the Severe Accident Integration Plan is presented in Figure 1. Completion of the elements of this plan constitute bases for assuring that the residual risks to the public from severe accidents at nuclear power plants are minimized in an effective manner. There are six main elements of the integration plan which lead to severe accident closure for operating plants. The six main elements are: 1) the individual plant examination (IPE) program, 2) a containment performance improvements (CPI) program for each of the six containment types, 3) a program to improve plant operations, 4) a severe accident research program (SARP), 5) an external events program, and 6) an accident management program. Severe accident closure is achieved once the IPEs have been completed (including external events) and any appropriate changes implemented, a framework for an accident management program has been developed and implemented, and generic requirements resulting from the CPI program have been implemented. A more detailed discussion of the severe accident closure process is presented later in this paper.

Besides the main elements listed above, there are other activities which are part of the staff's supporting programs on

severe accidents and are either input or output elements of the integration plan. For example, activities such as NUREG-1150 and the Systematic Assessment of Licensee Performance (SALP) provide information and insights to support one or more of the main program elements. Other input activities, such as safety goal and backfit policy provide guidance and constraints in defining an implementation strategy and making decisions. Each of these activities is discussed further in the next section of the paper.

It should be emphasized that this paper does not describe all of the severe accident decisions that need to be made over the next several years. However, key policy decisions and supporting documents that will be provided to the Commission in the near future are described below.

Discussion: This section will briefly describe the main and supporting elements of the severe accident integration plan and how closure is achieved. For each element, the current status is provided, along with a description of any anticipated Commission action. This information is also summarized in Table 1. In addition, Figure 2 provides a more detailed overall plan diagram that shows the flow of information, important interfaces, and schedular milestones.

The overall relationship of the elements is that each provides a specific action or condition that, when taken together and completed, results in closure. A more detailed discussion of each activity, including how they relate to each other and schematic diagrams that show the relationship with other activities and programs, is provided in the Enclosure.

MAIN ELEMENTS OF PLAN

1. Individual Plant Examinations (IPE)

The IPE involves the formulation of an integrated and systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant plant-specific risk contributors that might be missed without a systematic search. The examination will pay specific attention to containment performance in striking a balance between accident prevention and consequence mitigation. The licensee examinations may use 1) a PRA, 2) a method developed by the Industry Degraded Core Rulemaking (IDCOR) program and acceptable to the staff, or 3) other systematic examination methods. The staff expects each utility to promptly fix any significant vulnerabilities uncovered by the IPE process. As part of the NRC's review of the licensee's examinations, any significant individual plant vulnerability not addressed by the licensee but considered important by the

staff will be identified together with the most cost-effective options for reducing the vulnerability. The Commission's backfit policy will be used to decide which options need to be implemented. It is expected that it will take three years to complete and evaluate the examinations conducted in accordance with this program. The staff is almost ready to issue a generic letter to licensees containing guidance for implementing the individual plant examinations.

The specific objectives of the IPE are for each utility to:

A. Develop an overall appreciation of severe accident behavior.

B. Understand the most likely severe accident sequence(s) that could occur at its plant.

C. Gain a more quantitative understanding of the overall probability of core damage and fission product releases.

D. Reduce the overall probability of core damage and fission product releases, if necessary, by appropriate modifications to procedures and hardware that would help prevent or mitigate severe accidents. It is expected that the achievement of these goals will ensure that the severe core damage and large radioactive release probabilities for U.S. nuclear power plants are generally consistent with the Commission's safety goal policy.

In its letter dated May 10, 1988 from W. Kerr to Chairman Lando W. Zech, Jr., "Proposed Generic Letter on Individual Plant Examinations and the Proposed Integrated Safety Assessment Program II," the ACRS generally found the current draft of the IPE generic letter to be improved over the version they have commented on in a report dated June 9, 1987. The ACRS, however, recommended that the scope of the IPE be ". . . broader than the original intent of searching for outliers," and require each licensee to conduct a Level 2 PRA to subsume all outstanding safety issues, USIs, and GSIs. The ACRS also recommended that both internal and external initiators should be considered at this time, and conclusions about results and necessary changes should be determined by the licensees and reviewed by the staff through the ISAP process.

While we acknowledge and understand the ACRS recommendation and its basis, namely to provide a program that integrates a number of ongoing regulatory activities, we have concluded that it is not appropriate to completely implement the ACRS recommendation at this time. Our reasons for this conclusion are provided in Section 2.7 of the Enclosure.

Status and Expected Commission Action

The IPE generic letter was reviewed by CRGR on April 18, 1988, and by the Full ACRS Committee on May 5, 1988. At the Full ACRS Committee meeting, representatives from NUMARC stated that sufficient work on developing an examination methodology was done, and that the industry believed the IPE letter should be issued. We expect to forward to the Commission by July, 1988, a paper which transmits the proposed generic letter. The Commission will be requested to approve issuance of the generic letter to all licensees which initiates the conduct of the IPEs.

2. Containment Performance Improvements (CPI) Program

In SECY-87-297, dated December 8, 1987, the staff presented its program plan to evaluate generic severe accident containment vulnerabilities. This effort is predicated on the conclusion that there are generic severe accident challenges to each LWR containment type that should be assessed to determine whether additional regulatory guidance or requirements concerning needed containment features is warranted, and to confirm the adequacy of existing Commission policy. The bases for the conclusion that such assessments are needed include the relatively large uncertainty in the ability of LWR containments to successfully survive some severe accident challenges, as indicated by draft NUREG-1150. All LWR containment types are to be assessed starting with BWR MARK Is. Staff recommendations for regulatory closure are forecast for MARK I containments in late summer 1988, and for the other containment types in the fall of 1989. This effort is integrated closely with the IPE program, which will be examining plant-specific accident vulnerabilities that could threaten containment integrity. This program complements the IPE program and is intended to focus on resolving hardware and procedural issues related to generic containment challenges. Because of its generic origin, the perceptions about MARK I performance, Commission guidance, and the IPE schedule (approximately three years), this effort is being accelerated. New regulatory requirements from this program, if any, would be developed consistent with the safety goal and backfit rule and would constitute closure of generic containment performance issues. Additional discussion on the use of the safety goal in the closure process is provided in the closure section of this paper.

Status and Expected Commission Action

The study of potential improvements in containment performance from generic challenges has been initiated in three phases. The status of each phase is as follows:

A. The MARK I study is in its final stages. A status report to the Commission is in concurrence. A final report with recommendations is expected by the Fall of 1988. The report will discuss the preventive and mitigative effectiveness of proposed improvements, and will make recommendations.

B. Issues related to ice condenser, MARK II and MARK III plants are to be identified in an invitation to industry to participate in a workshop scheduled for February, 1989. A resolution of the issues on these containments is expected to be submitted in a report to the Commission, with staff recommendations, in August, 1989.

C. Issues related to dry containments are to be identified in an invitation to industry to participate in a workshop tentatively scheduled for April, 1989. A resolution of the issues on these containments is expected to be submitted in a report to the Commission, with staff recommendations, in August, 1989.

For each of the above phases, the Commission will be requested to approve implementation of the staff recommendations.

3. <u>Improved Plant Operations (IPO)</u>

Most PRAs have shown the sensitivity of risk to human errors. Staff analyses of operating experience confirms the importance of reducing maintenance, surveillance, testing and control room operator errors. Therefore, the staff's program to improve plant operations considers the following efforts:

A. Continued improvement of the Systematic Assessment of Licensee Performance (SALP) process.

B. Regular reviews by senior NRC managers to identify and evaluate those plants that may not be meeting NRC and industry standards of operational performance.

C. Diagnostic team inspections to probe further the performance of those plants identified in Item B.

D. Regulatory actions to improve operational performance where it has fallen below expected standards.

E. Improved technical specifications.

F. Continued improvement of operating procedures.

G. Expanding EOPs to include guidance on severe accident management strategies.

H. Industry programs to reduce transients and other challenges to engineered safety features.

I. Feedback from the IPE program of experience and improvements in operational areas, such as maintenance and training.

J. Continued research to evaluate the sensitivity of risk to human errors, the contribution of management to the level of human errors, and the effectiveness of operational reliability methods to help identify potential problems early and prevent their occurrence.

Status and Expected Commission Action

This is an ongoing program. No specific additional actions are expected or planned to be requested of the Commission in the near future.

4. Severe Accident Research Program (SARP)

The Severe Accident Research Program (SARP) was begun after the TMI-2 accident in March, 1979, to provide the Commission and NRC staff with the technical data and analytical methodology needed to address severe accident issues. The initial work was phenomena-oriented to gain insights into the mechanisms involved, and consisted of a wide range of experiments and code development activities. Today, however, with considerable knowledge amassed and the results of the Agency's new Reactor Risk Program (NUREG-1150), it is possible to apply the knowledge base and models to the resolution of specific issues, such as Containment Performance Improvements (CPI), Accident Management (AM), the Individual Plant Examination (IPE) program, and the implementation of the Commission's Safety Goal Policy Statement.

Some examples of severe accident phenomena and technical areas that SARP has examined include:

A. Natural circulation in the reactor coolant system.

B. Core melt progression and hydrogen generation.

C. Steam explosions as a potential failure mode for both reactor vessel and containment.

D. The potential for early failure of containment by high pressure melt ejection (Direct Containment Heating).

E. Core-concrete interactions, fission product behavior, and heat transfer.

F. Hydrogen ignition and burning in containment.

G. Fission product behavior and chemical form in the reactor coolant system.

H. Revaporization of previously deposited fission products.

The SARP effort has greatly enhanced our ability to analyze severe accident phenomena and has even identified certain phenomena and issues that were not completely considered during the Reactor Safety Study (WASH-1400), such as 1) the potential for early failure of PWR containments by Direct Containment Heating (DCH) induced by high pressure melt ejection and the potential mitigating effects of natural circulation in the reactor coolant system, and 2) the possible failure of BWR/MARK I containments by melt-through of the steel containment shell caused by interaction with the resulting molten corium-concrete mixture, expecially under dry cavity conditions. SARP has also provided the Agency with state-of-knowledge computer code analysis capabilities to realistically analyze the course of a severe accident and to develop accident management strategies. Over the next three to five years, SARP will support the other main elements of the plan and will continue to develop information in the areas of most significance to reactor risk and continue confirmatory work, including code assessment and validation. The primary emphasis will be on resolving specific technical issues associated with the main elements of the plan to facilitate the formulation of staff positions, especially in the area of containment performance improvements. A more detailed discussion of SARP is provided in Section 5 of the Enclosure.

Status and Expected Commission Action

This is an ongoing program. No specific additional actions are expected or planned to be requested of the Commission in the near future.

5. External Events

The Commission's Severe Accident Policy Statement does not differentiate between events initiated within the plant and externally initiated events. The staff believes it was the Commission's intent that both internally and externally initiated events be considered. To date, both the industry and the staff have concentrated on procedures for review of internally initiated events. Procedures for external events examination are under development, but will not be completed in a time frame consistent with internal event schedules. Therefore, the staff intends to proceed with implementation of the severe accident policy outlined in SECY-86-76 for

internally initiated events. The evaluation of external events will proceed separately and on a later schedule from that of internal events; otherwise, it would produce an unwarranted delay of severe accident closure (see Section 2 of Enclosure).

As stated in SECY-86-162, the staff is proceeding with the evaluation of severe accidents initiated by external hazards in two phases. The first phase, which is essentially completed, consists of a Lawrence Livermore National Laboratory (LLNL) study to 1) assess the margin that past design bases provide relative to external events that are beyond the design basis, and 2) identify areas where an examination for external vulnerabilities may be needed. The second phase will consist of developing specific guidance and criteria for each external hazard to be considered in the Individual Plant Examination for External Events (IPEEE).

Based on the results of the LLNL study and information developed during an external events workshop held August 4-5, 1987, some external events such as earthquakes and fires will need further consideration at some plants. Also, the design bases derived from some external events, such as winds and external floods, may be sufficiently conservative that they do not pose a significant risk at most plant sites. However, there may be some structures or facilities that were not designed to current criteria and may pose some risk at those plants. These studies reaffirm the need to include external events in accident reviews and provide insights to assist in developing review guidance for the IPEEEs. A new program has been initiated at LLNL to develop IPEEE guidance, procedures, and criteria. More specific guidance including evaluation and integration of ongoing NRC programs will be developed to help utilities to conduct their IPEEEs.

On December 21, 1987, NRC established an External Events Steering Group (EESG) to make recommendations to senior management concerning how best to proceed with implementing the severe accident policy with respect to externally-initiated events. This will include the development of guidance documents on which external initiators need to be examined, and acceptable examination methods. In addition, industry has also begun to work on this issue, and their efforts will be coordinated with the staff efforts.

In a recent letter to Chairman Zech dated May 10, 1988, the ACRS recommended that a Level 2 PRA including both external and internal events be conducted by each licensee to search for vulnerabilities. The staff does not agree with the ACRS recommendation, and our reasons are provided in Section 2.7 of the Enclosure.

Status and Expected Commission Action

A paper describing the staff recommendations for proceeding with implementation of the severe accident policy for externally initiated events will be forwarded to the Commission by the end of FY 89. The Commission will be requested to approve implementation of staff recommendations contained in that paper.

6. Accident Management (AM) Program

Certain preparatory and recovery measures can be taken by the plant operating and technical staff that could prevent or significantly mitigate the consequences of a severe accident (i.e., accident management). Broadly defined, accident management includes the measures taken by the plant staff to 1) prevent core damage, 2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, 3) failing that, maintain containment integrity as long as possible, and finally 4) minimize the consequences of offsite releases. In addition, accident management includes certain measures taken before an event (e.g., improved training for severe accidents, hardware or procedure modifications) to facilitate implementation of accident management strategies. Therefore, accident management is viewed as an important means of possibly achieving a substantial reduction in risk from severe accidents. We will examine the feasibility of establishing accident management goals in the forthcoming Safety Goal Implementation Plan.

Under the staff program, accident management programs will be developed and implemented by licensees. The NRC will focus on developing the regulatory framework under which the industry programs will be developed and implemented, as well as providing an independent assessment of accident management capabilities and strategies. It is expected that much of the information resulting from NRC research will be transferred to industry to support their accident management programs and improve overall plant safety. The other major elements of the integrated plan for closure of severe accident issues will help provide the technical basis for developing the elements of accident management programs.

The staff's accident management program will also be supported by the Human Factors Research Program. Human factors research will provide the technical basis for decisions about human-machine interfaces, procedures, qualifications and training, and organization and management.

Status and Expected Commission Action

In the proposed generic letter initiating the IPEs, the industry is informed that they will be expected to develop and implement severe accident management programs in conjunction with conducting their IPEs, but submittal of a program is not required with the IPE results submittal. The staff's request to the Commission for approval to issue the IPE letter will therefore result in formalizing the Commission's intentions on this matter.

Within the next several months, the staff will send to the Commission plans for proceeding with the Accident Management Program. This will include the Severe Accident Management Research Program Plan which will be carried out by RES and a description of the regulatory program to be carried out by NRR that will result in accident management programs being put in place by utilities. The Commission paper will address how research results will be used in the regulatory program and how industry efforts (e.g., NUMARC) will be coordinated. It is not clear yet if any specific Commission action will be requested.

SUPPORTING ELEMENTS OF PLAN

1. NUREG-1150

In February, 1987, the NRC published the Reactor Risk Reference Document, NUREG-1150, in draft form for public comment. Since that time, a substantial number of comments have been received. In addition, three major peer reviews have been conducted and have provided valuable suggestions regarding possible improvements to the report. The report is now being substantially revised in response to the comments received, and will be available in final form in January 1989. NUREG-1150 provides a snapshot of the state-of-the-art PRA technology, incorporating improvements since WASH-1400. The report will include examination of the severe accident frequencies and risks and their associated uncertainties for five licensed nuclear power plants, and will utilize the latest source term information available from both the NRC and its contractors and the nuclear industry. The information in the final report will provide valuable information and insights to the various other elements of the severe accident integrated plan. The information to be provided, and the uses it will have, include the following:

A. Probabilistic models of the spectrum of possible accident sequences, containment events, and offsite consequences of severe accidents, for use in:

- Development of guidance for the individual plant examinations of internally and externally-initiated accidents and accident management strategies;

- Analysis of the need and appropriate means for improving containment performance under severe accident conditions;

- Characterization of the importance of plant operational features and areas potentially requiring improvement; and

- Analysis of alternative safety goal implementation strategies (as provided and discussed in draft NUREG-1150).

B. Data on the major contributing factors to risk and the uncertainty in risk, for use in:

- Prioritization of research; and

- Prioritization of generic issues.

Status and Expected Commission Action

NUREG-1150 is currently being revised to accommodate comments received. It is planned to be completed by December, 1988. A paper transmitting it to the Commission for information is expected in January 1989.

2. Generic Safety Issues

Generic Safety Issues (GSIs) including Unresolved Safety Issues (USIs) are, for the most part, potential generic vulnerabilities to severe accidents. This is the case because such issues are identified primarily on the basis of their safety significance. Safety significance is, in turn, primarily a function of the potential for a severe core damage accident. All generic issues are listed and described in NUREG-0933. The imposition of any new requirements or guidance resulting from the resolution of generic issues must be in accordance with the backfit rule (10 CFR 50.109).

Generic safety issues are indirectly related to the individual plant examinations. As previously stated, generic safety issues are a safety concern that may affect the design, construction, or operation of all, several, or a class of nuclear power plants. Most of the generic safety issues being resolved today already have a generic resolution identified, and the principal work remaining is preparation of the regulatory analysis needed to determine whether the recommended

resolution is justified. In addition, most generic safety issues are on schedules which call for their resolution within approximately two years. Hence, with the exception of USI-A-45, it was not considered advantageous to try to resolve all or most of the outstanding generic safety issues in the IPE process.

Nevertheless, we recognize that in the course of conducting their IPE, some licensees may wish to propose resolutions to certain generic safety issues for their plants. In the proposed generic letter, the staff is encouraging licensees to submit generic safety issue resolutions, and upon review and approval by the staff, the issues would be considered resolved for those plants.

Status and Expected Commission Action

This is an ongoing program. No specific actions are requested or planned to be requested of the Commission in the near future.

3. Integrated Safety Assessment Program (ISAP)

This is a voluntary systematic program to address regulatory issues according to an integrated schedule. It includes an integrated assessment based on a probabilistic risk assessment (PRA). Issues would be ranked, and an integrated schedule would be developed from this ranking. The integrated schedule would allow both NRC-initiated regulatory issues and utility-initiated items to be included in the schedule. For a given plant, ISAP would require a minimum of a Level 1 PRA. A Level 1 PRA augmented with a containment vulnerability assessment is also an acceptable method for a utility to perform an IPE. Therefore, the analysis done to participate in ISAP is directly applicable to the work necessary to perform an IPE. The staff would implement the IPE and ISAP as separate but compatible programs. A utility could initiate ISAP either before or after completion of its IPE. The criteria for determining which of the identified risk reduction actions would be implemented will be established within the IPE program framework. Currently, there is a relatively low level of interest expressed by the licensees in ISAP; however, the staff will arrange to combine IPE and ISAP reviews when requested consistent with available resources.

Status and Expected Commission Action

Since the program is not being pursued at this time, no specific actions are requested or planned to be requested of the Commission in the near future.

ADVANCED REACTORS

The staff is currently reviewing, or planning to review, a number of advanced reactor designs. These designs include future LWRs and non-LWR reactors. In parallel with the closure of severe accident issues for operating reactors, the staff is proposing to develop procedural and possibly performance regulations for future LWRs, with supporting regulatory guides and other documents, in order to clarify the procedures and requirements for addressing severe accidents. The staff's plans for both LWR and non-LWR advanced reactors are described below.

The severe accident policy states that a new design can be shown to be acceptable for severe accident concerns if it meets certain criteria and procedural requirements. Therefore, the staff is considering initiating rulemaking to modify 10 CFR 50.34(f), since it currently applies only to cancelled plants, and the possible addition of performance requirement(s). The modification of 10 CFR 50.34(f) would adopt the procedural requirements from the policy statement. The performance requirement(s) would require a higher level of safety for future plants in accordance with the policy statement. The Regulatory Guide(s) would expand on the Rule(s) to provide a more detailed definition of the procedural steps, such as the scope and content of the required PRA documentation, and provide guidance on what severe accidents or phenomena should be considered. The staff believes that severe accident rulemaking for future LWRs will facilitate design certification rulemaking and that the requirements should be in place prior to initiation of design certification rulemaking for the LWRs currently under review.

The treatment of severe accidents for non-LWR reactors is being assessed as part of the reviews being conducted in accordance with the Advanced Reactor Policy Statement and NUREG-1226, "Development and Utilization of the NRC Policy Statements on the Regulation of Advanced Nuclear Power Plants." The guidance being developed by the staff for these designs is built upon and is to be consistent with existing Commission guidance as stated in the advanced reactor, severe accident, safety goal and standardization policies. In particular, the guidance contained in the safety goal and severe accident policies has been used to help define the range of events which need to be considered in these designs and the role of a PRA. Additionally, we are utilizing a licensing approach on advanced reactors which stresses deterministic engineering analysis and judgment, complemented by PRA. It is anticipated that much of the guidance to be developed in the rulemaking for LWRs will also be applicable to the non-LWR reactors.

Status and Expected Commission Action

A separate Commission paper on the staff's proposal for future LWR severe accident rulemaking is in preparation and will be provided to the Commission by July, 1988. With respect to the key licensing issues associated with future non-LWR designs (treatment of severe accidents, siting source term, containment, emergency planning), specific staff proposals are to be provided in a Commission paper by June, 1988. For both of these proposals, Commission approval of the recommended actions will be requested.

CLOSURE OF SEVERE ACCIDENT ISSUES FOR OPERATING REACTORS

The Commission has ongoing a number of programs related to severe accident behavior in operating light water reactors. Each program addresses a specific aspect of severe accident behavior and may in fact result in a proposed specific action on the part of the staff or Commission towards the regulated industry. However, neither the staff nor Commission has yet defined for the industry which programs are critical to resolving the severe accident issues for their plants and what specific steps must be taken by each licensee to achieve this resolution.

Completion of this resolution process is termed "closure" of severe accident issues. Actions resulting from two tracks; namely, generic issues and plant-specific issues, must be taken for severe accident closure. Closure for generic severe accident issues will be obtained when the Commission takes action in the form of rulemaking, or states whatever its required approach is. Closure for plant-specific severe accident issues will be obtained when each licensee has completed certain evaluations and implemented certain programs such that events which comprise the dominant contributions to risk for each plant are identified and that practical enhancements to the design, procedures, and operation are made such that further improvements can no longer be justified by backfit analysis pursuant to 10 CFR 50.109. However, specific plant and operational improvements may be identified which do not meet the backfit rule, but if implemented, would significantly alter the risk profile of the plant, improve the balance of reliance on both prevention and mitigation, or substantively reduce uncertainties in our understanding. Any such improvements identified will be brought forward to the Commission with recommended action on a case-by-case basis. Closure of a single issue or combination of issues is achieved when the above is satisfied for that issue or those issues addressed.

It should be noted that "closure" does not imply that all severe accident activities will cease. Certain activities, such as research in the areas of severe accident phenomena and human performance will continue beyond "closure." These activities are designed to provide confirmation of previous judgments. It is expected that as a result of continuing research, experience, and other activities, additional issues or questions regarding judgments related to severe accidents may arise. These will be considered and disposed of on a case-by-case basis, and are not expected to bring into question the previous conclusions regarding closure.

The following sections describe in detail the steps that each licensee is expected to complete in order to achieve severe accident closure for each of its operating reactors.

1. <u>Completing Individual Plant Examinations (IPEs)</u>

The IPE program is intended to be "an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributors (sometimes called "outliers") that might be plant specific and might be missed absent a systematic search."

Each licensee is expected to perform an IPE using a method acceptable to the staff. As will be described in the staff generic letter implementing the IPE, the staff expects that in many cases utilities, in the performance of their IPEs, may find and will voluntarily remedy uncovered vulnerabilities by making the necessary safety improvements (conforming to the requirements of 10 CFR 50.59). However, through the review of IPE submittals, the staff may find it necessary to employ established plant-specific backfit criteria to assure that justifiable corrections are made.

For the phase of the evaluation associated with identification of dominant core melt sequences (commonly referred to as the "front end" analysis of a PRA), there is little controversy regarding methods, and we expect the industry decision process with respect to potential modifications to be straightforward. For the phase of the evaluation associated with core melting, release of molten core to the containment, and containment performance, the staff recognizes that for a few of the phenomena, notably in areas which affect containment performance, there is a wide range of views about their relative probability as well as their consequences. For these issues additional research and evaluations will be needed to help reduce the wide range of uncertainties. Because of concern over the ability of containments to perform well during some severe accidents, the staff is conducting a Containment Performance Improvements Program (for more details see Item 3

below). This program complements the IPE program and is intended to focus on resolving generic containment challenges, including issues associated with the phenomena mentioned above.

The NRC and industry currently have ongoing research programs to address these few issues. However, until a sufficient understanding of these phenomena is developed, each licensee will be faced with the need to be able to understand the potential range of probabilities and consequences associated with these issues.

Accordingly, we would expect each licensee to implement a Severe Accident Management Program which provides training and guidance to their operational and technical staff on understanding and recognizing the potential consequences of these phenomena.

We do not plan to require a licensee to consider external events in its IPE at this time. The staff is currently studying methods it would find acceptable for examining plants for severe accident vulnerabilities from external events, and will be meeting with NUMARC regarding these methods as well as the scope of an external event examination. We expect completion of the methods development within 12 to 18 months. Closure with respect to external events will be achieved upon completion of an examination of each plant, as needed, for external event vulnerabilities consistent with the conclusions of the staff studies described above.

2. Accident Management

The staff has concluded that significant risk reductions can be achieved through effective severe accident management. We also believe that the IPE conclusions reached by licensees for their plants will explicitly rely on certain operator actions, or on operators not taking actions which could adversely affect both the probability and consequences of a severe accident.

Hence, a key element to severe accident closure for each plant will be the implementation of a Severe Accident Management Program. Since information on severe accident phenomena and effective accident management strategies will continue to be developed by both NRC and industry over the next several years, closure is not predicated on having a "complete" accident management program in place. Rather, closure is based on each licensee having an Accident Management Program framework in place, that can be expanded, modified, etc. to accommodate new information as it is developed.

3. Containment Performance Improvements

As a result of concerns related to the ability of containments to withstand some generic challenges associated with severe accidents, the staff has undertaken a program to determine what, if any, actions should be taken to reduce the vulnerability of containments to severe accident challenges, and to reduce the magnitude of releases that might result from such challenges.

Staff efforts have first focused on the BWR MARK I containment. The staff studies are primarily focused on the potential generic vulnerabilities of these containments, and not plant unique vulnerabilities, which is the primary focus of the IPEs. The staff schedule calls for an interim report on BWR MARK Is to be submitted to the Commission in June of this year, with final recommendations due in the fall of this year. The other types of containments are to be assessed by the fall of 1989.

The IPE generic letter is now expected to be issued by July of this year, and licensees will have approximately four months to respond identifying their plan for conducting the IPEs. Following the four-month period, it is expected they will commence with their IPEs. It is further expected that any modifications to Mark I containments that the staff may recommend will be available to the industry before they start their IPEs. For the other containment types, the fact that any staff recommendations will not be available until after they have commenced with their IPEs is a concern. However, the IPE generic letter will state that the staff does not expect the industry to make any major modifications to their containments until the information associated with the generic issues which affect containment performance has been developed by the staff. Hence, the industry will not be placed in a position of having to implement improvements before all containment performance decisions have been made.

4. Use of Safety Goal in the Closure Process

The staff expects to use safety goal policy and objectives, including the 10^{-6}/reactor-year "large release" guideline, to assist in the resolution and closure of severe accident issues. Resolution and closure of issues are expected to be of two different types, either plant unique or generic. Safety goals and objectives are to be used only for the resolution of generic issues, i.e., severe accident issues common to a defined generic class of plants. Resolution of plant unique issues is to be accomplished on a case by case basis, using the information developed by Individual Plant Examinations (IPE) as is described in Section 1.

The staff is preparing a Safety Goal Policy Implementation Plan (Revised) that incorporates the following, as directed by the Commission (Staff Requirements Memorandum dated November 6, 1987):

(1) Information on how the staff proposes to implement OGC guidance on the use of averted on-site costs in backfit analyses.

(2) Whether averted off-site property damage costs should be included in a more explicit manner in backfit analyses.

(3) Whether $1,000/person-rem remains an appropriate cost/benefit criterion.

(4) A discussion of options for defining a "large release."

(5) A discussion of options for specifying appropriate plant performance objectives.

(6) Responses to Commissioner Bernthal's questions regarding population density considerations, and whether it would be acceptable for a plant to have no containment if it met the large release criterion by prevention of core melt (core damage) alone.

This plan will also reflect the consideration given by the staff to ACRS recommendations and the results of several meetings with the ACRS on this subject.

Resolution of severe accident generic issues using safety goal objectives is expected to proceed as follows. PRA information from a variety of sources, including both staff generated PRAs, (e.g., NUREG-1150) and utility generated PRAs (IPE) will be used to make comparisons with applicable safety goal objectives in accordance with the implementation plan. The staff will identify the reasons why particular plants appear to meet or not meet these objectives and assess these reasons in relation to current regulatory requirements. This assessment will constitute a testing of the effectiveness of these requirements or their implementation and is expected to result in the identification of potential changes to regulatory requirements that, for some plants, would be expected to result in safety enhancements. These, in turn, will be subject to appropriate regulatory analysis as provided in the Commission's backfit rule 10 CFR 50.109. Those that can be shown to provide substantial safety benefit and are cost-effective will be proposed to the Commission for backfit, possibly in the form of rulemaking. The staff expects that this process would have no impact on classes of plants for which there is reasonable assurance that safety goal objectives are met. This

expectation is based upon the intent to identify those features of design and/or performance that are already in place at plants meeting safety goal objectives and to structure any new requirements such that they do not require changes or additions at these plants.

The staff's revised Safety Goal Implementation Plan is scheduled to reach the Commission in August, 1988. The first application is expected to be reflected in the staff's recommendations to the Commission in the Fall of 1988 on potential improvements to BWR MARK I severe accident containment performance.

5. <u>Summary of Closure Process</u>

In summary, the steps which each licensee is expected to take to achieve closure on severe accidents for its plants are as follows:

- Complete the IPEs; identify potential improvements, evaluate and fix as appropriate.
- Develop and implement a framework for an Accident Management Program that can accommodate new information as it is developed.
- Implement any Commission-approved generic requirements resulting from the staff Containment Performance Improvements Program; this should constitute closure of containment performance generic issues.

While programs for improved plant operations and research in the area of severe accidents will continue, completion of the above by a licensee is considered to constitute "closure" of the severe accident issue for the plant in question. Specific issues that may arise in the future as a result of ongoing research will be treated on a case-by-case basis and will not affect the closure process.

<u>Closure Schedule and Resources</u>

The staff estimates that it will take five to six years to complete the closure process. This estimate is based on the following:

- The IPE process for internal events will be completed in approximately three years.
- The staff guidance on conducting IPEs for external events will be available in approximately one to one and one-half years. The conduct of external event IPEs could take up to three years following issuance of staff guidance.

- The framework for Accident Management Programs would be developed and implemented within two years following completion of the internal events IPE.

For each closure activity, the following staff resources are estimated at this time:

° Internal Event IPE	8 FTE per year x 3 years = 24 FTE
° External Event IPE	5 FTE per year x 3 years = 15 FTE
° Accident Management Framework Reviews	7 FTE per year x 2 years = 14 FTE
° Generic Containment Performance Improvements	4 FTE per year x 2 years = 8 FTE
	TOTAL = Approximately 61 FTE

Resolution of any resource inconsistencies will be addressed in the current update of the five-year plan.

Implementation: The staff intends to proceed with implementation of the integration plan for severe accident closure as described above and in accordance with the schedule delineated in Table 1, unless the Commission has specific changes which it wishes to make.

Scheduling: The staff recommends this paper be considered at an open meeting.

Victor Stello, Jr.
Executive Director
for Operations

Enclosure:
Discussion of Severe Accident Integration Plan Activities

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TABLE 1

Schedule and Expected Commission Actions for Severe Accident Issues

Item	Schedule	Expected Commission Action
1. Individual Plant Examinations	Commission Paper due July, 1988	Commission will be requested to approve issuance of generic letter initiating IPEs
2. Containment Performance Improvements		
a) BWR MARK Is	Interim recommendations due June, 1988 and Commission Paper with final recommendations due Fall, 1988	Approve staff recommendations
b) Other Containment Types	Commission Paper due August, 1989	Approve staff recommendations
3. Improved Plant Operations	Continuous	None at this time
4. Severe Accident Research Program	Continuous	None at this time
5. External Events	Commission Paper due end of FY 89	Approve staff recommendations for treatment of external events in severe accident policy implementation
6. Accident Management Program	Commission Paper due Fall, 1988	None--Paper for information only
7. NUREG-1150	Commission Paper scheduled for January, 1989	None--Paper for information only
8. Generic Safety Issues	Continuous	None at this time
9. Integrated Safety Assessment Program	Commission Paper due June, 1988	No action anticipated at this time
10. Advanced Reactors		
a) LWRs	Commission Paper due July, 1988	Approve staff recommendations
b) Non-LWRs	Commission Paper due June, 1988	Approve staff recommendations
11. Safety Goal Policy Implementation	Commission Paper due August, 1988	Approve staff recommendations

FIGURE 1
SEVERE ACCIDENT PROGRAM - SCHEMATIC

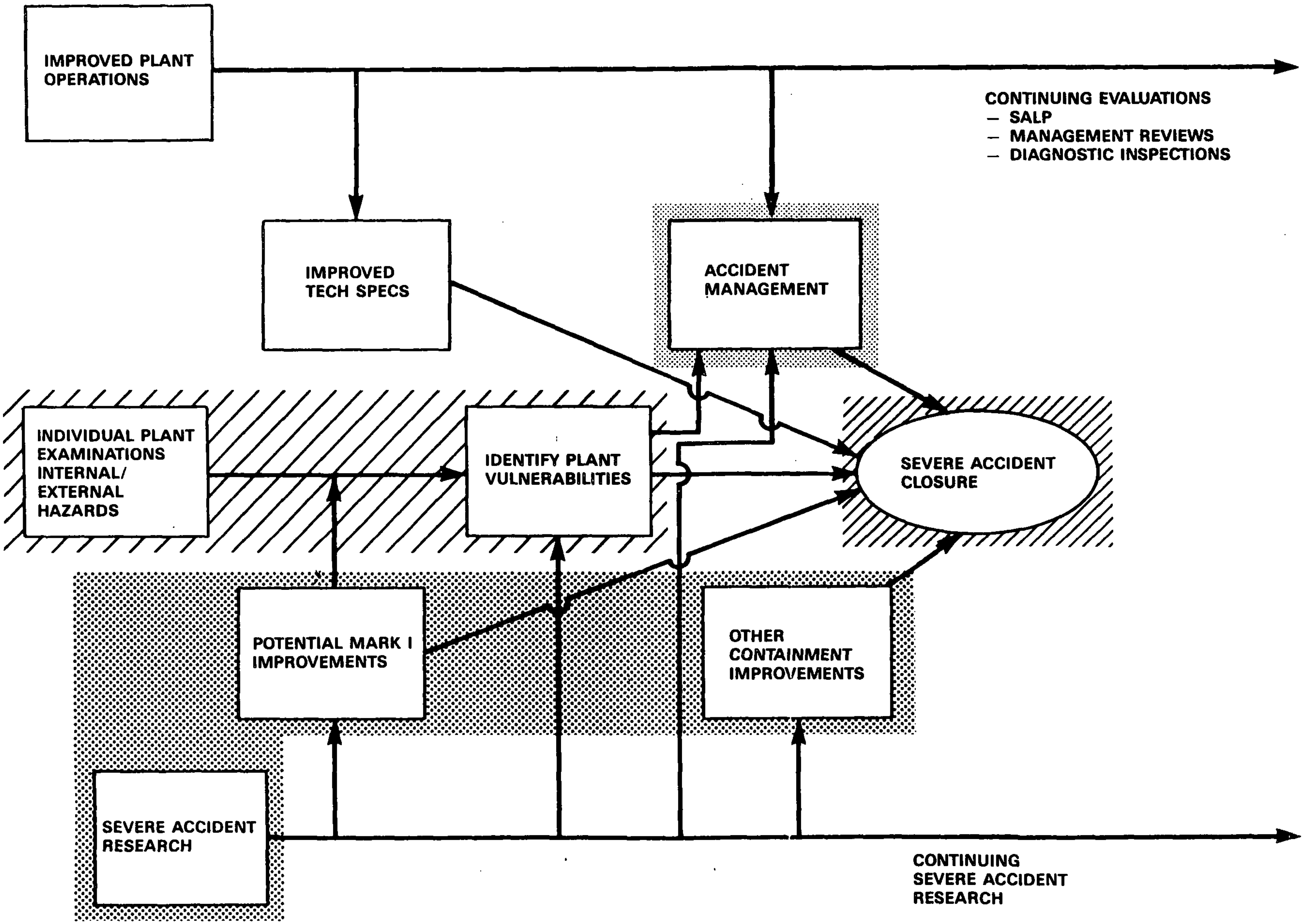
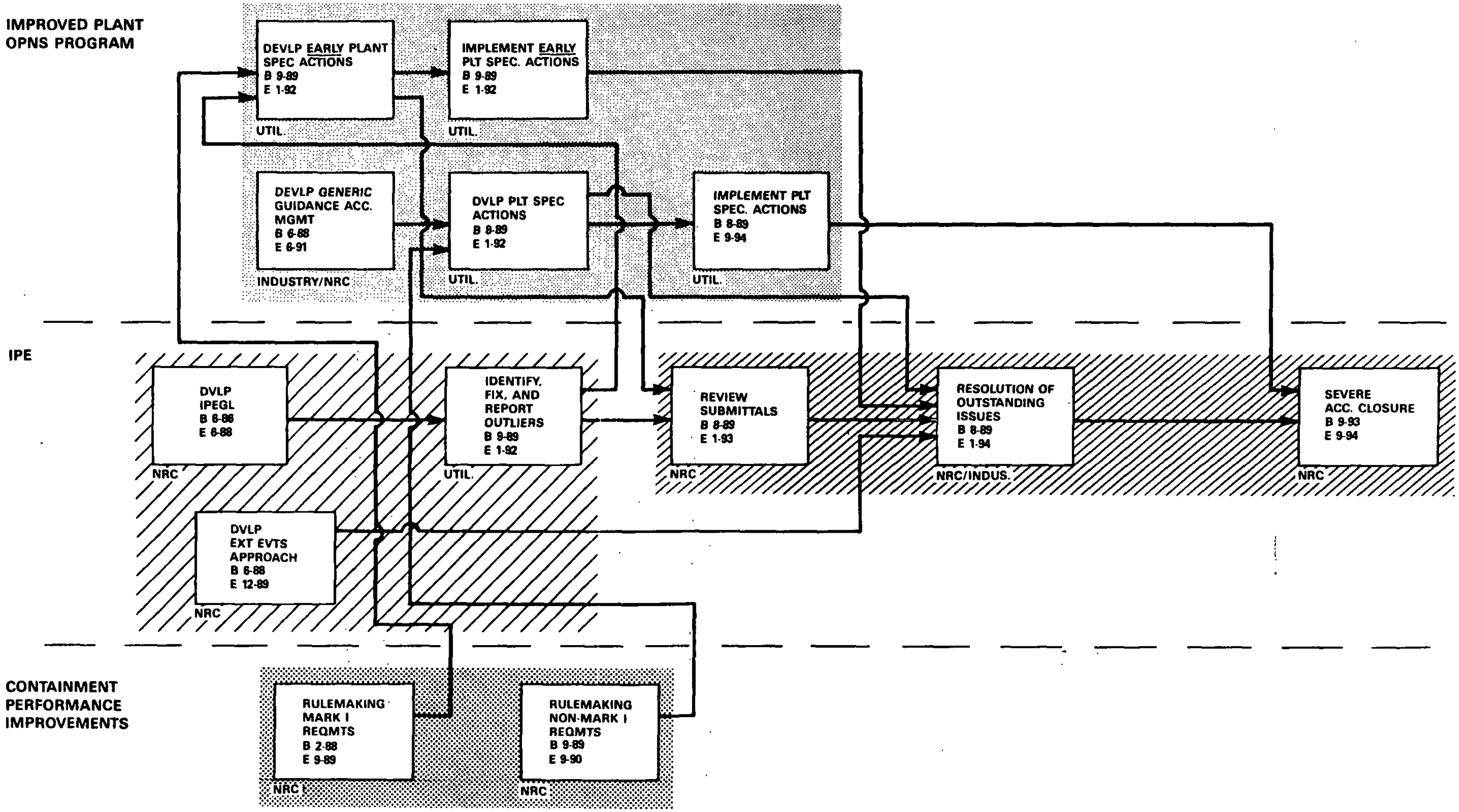


FIGURE 2

SEVERE ACCIDENT PROGRAM - SUMMARY PLAN



ENCLOSURE

Discussion of

Severe Accident Integration Plan Activities

Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission

May 1988

Table of Contents

1. INTRODUCTION AND STATE OF TECHNOLOGY

1.1 Introduction

This enclosure describes in more detail than the cover paper the main and supporting elements of the staff's Severe Accident Integration Plan, including the elements for advanced reactors. In addition, discussions are presented on how all severe accident activities relate to each other, as well as schematic diagrams that show the relationship with other activities and programs.

First, however, the state of technology on severe accidents will be described to provide perspective. Then, the main and supporting elements of the plan will be described, as well as severe accident policy for future LWRs and advanced reactors.

1.2 State of Technology

In 1975, the first comprehensive examination of the risk of light water reactors, the Reactor Safety Study (WASH-1400), was issued. This study, as well as many PRAs performed since in the U.S. and elsewhere in the world, concluded that core damage and core-melt accidents represent the major contribution to risk from commercial nuclear power plants. The WASH-1400 results indicated that the conditions that could develop in severe accidents, as a result of the thermal-hydraulic-material interactions which take place, could be more severe than those of design basis accidents, and that failure of the containment, in most instances early in the accident evolution, was the likely outcome of such accidents. A number of these studies have also identified areas in both the design and operation of a nuclear power plant where appropriate hardware additions or modifications, frequently coupled with procedural improvements, could help reduce these risks, in a cost-effective way.

The NRC began to give some attention to severe accidents even before the accident at Three Mile Island (TMI). However, after the accident, we increased the scope and diversity of our efforts in this area substantially. Some examples include allocating additional resources to R&D activities. This included construction of new experimental facilities, development of analytical tools, and in general, the development of information needed to gain more knowledge and insights about severe accident behavior. Also, the Commission initiated and implemented new rules dealing with some aspects of severe accidents (i.e., accidents beyond the original design basis of the plant, but still retaining the degraded core in-vessel). For example, the NRC promulgated a rule (10 CFR 50.44) dealing with the generation of large quantities of hydrogen in the three types of BWRs and in Ice Condenser PWRs. Since then, the NRC, and to a lesser extent the industry as well, has developed a large body of information and data on severe accidents, including the probability of severe accidents, the core-melt phenomenology, the associated accident sequences, and the effects of severe accidents on plant systems, components and structures, especially those that provide barriers to fission product release to the atmosphere such as the containment.

1.2.1 Containment Performance Capabilities and Failure Modes Beyond the Design Basis

The role of the containment structure as a vital barrier to the release of fission products to the environment has long been widely recognized. The good public safety record of nuclear power plants has been sustained by applying the "defense-in-depth" principle, which relies on a set of barriers to fission product release. The containment and its supporting systems are the last of these barriers. Current containment design criteria are based on a set of deterministically derived challenges. Pressure and temperature challenges are usually based on the design basis loss-of-coolant accident; radionuclide challenges are based on the source term of 10 CFR Part 100. Also, criteria based on external events, such as earthquakes, floods, and tornadoes are considered. The margins of safety provided by applying conservative loading

combinations have been the subject of considerable research and evaluation, and these studies have shown the ability of modern containment systems to survive realistic pressure loading challenges well beyond design levels. Because of these margins, the various containment types presently utilized in the United States have the capability to withstand, to varying degrees, many of the challenges presented by severe accidents even though not designed primarily for that purpose. For each type of containment, however, there remain failure mechanisms which could lead to either early or late containment failure, depending on both the accident scenarios involved and the containment types.

1.2.2 Current Understanding

In general, our knowledge and understanding of the phenomena and processes involved has increased substantially since WASH-1400. For example, challenges to containments from missiles generated from steam explosions are now considered very unlikely (NUREG-1116, NUREG-1150). Early challenges to large dry PWR containments from steam overpressure (i.e., from the quenching of the molten core as it is being released into the reactor cavity) are also considered unlikely, mostly due to the now known capability of these containments to withstand such loads. This scenario was considered as a possible failure mode in WASH-1400. Challenges to the class of smaller volume BWR and Ice Condenser containments from the generation and uncontrolled burning of large amounts of hydrogen have been reduced via specific containment-design actions already implemented (e.g., inerting of BWR MARK I and MARK IIs, and the installation of systems for controlled burning of hydrogen in Ice Condenser and MARK III plants, for a number of severe accident sequences).

However, even though containment designs can cope to varying degrees with many of the challenges presented by core melt accidents, there remain failure mechanisms for each type of containment. Some of these failure mechanisms could lead to early failure of containment. In some designs, these failure mechanisms are considered to be well understood while in others the failure

modes are not as well understood, and their likelihood is viewed as being high or low by various experts (NUREG-1150). There are several severe accident phenomenological issues where research has not produced definitive results on the challenges that certain phenomena could pose to containment integrity. These issues include direct containment heating, associated with molten core material released at high pressure; containment shell melt-through, especially as it could affect the integrity of a MARK I type of containment under dry conditions; and the issue of hydrogen which could principally have an effect on Ice Condenser plants, especially under station blackout conditions when the igniters have ceased functioning. These and some other issues also of importance to containment integrity are discussed in more detail below and in the enclosure in sections dealing with the Severe Accident Research Program (SARP). SARP is being directed to address these issues and to further reduce the uncertainties associated with them. Information developed from these programs will be utilized in the formulation of staff positions regarding proposed containment performance improvements and related accident management strategies. Additional research, well focused on specific issues or sub-issues, will continue on a confirmatory basis.

1.2.3 Summary and Conclusion

Considerable knowledge and insights on severe accidents have been developed over the past years. This information has improved our understanding of what could possibly go wrong in a nuclear power plant during severe accidents, as well as ways to reduce or eliminate their consequences. As was pointed out above, there are still some severe accident phenomena where additional research is needed which will help reduce the wide range of uncertainty associated with their likelihood or their effect on containment integrity. However, for either case, the available information base is solid and broad enough so that a range of regulatory decisions regarding severe accidents can now be made.

Accordingly, the staff is recommending that the implementation of the Commission's policy on severe accidents be initiated. A discussion of the severe accident integration plan activities is presented in the following sections.

2. INDIVIDUAL PLANT EXAMINATIONS (INTERNAL EVENTS)

2.1 Background

In the Commission policy statement on severe accidents in nuclear power plants issued on August 8, 1985 (50 FR 32138), the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on generic rulemaking or other regulatory requirements for these plants. However, the Commission recognized, based on NRC and industry experience with plant-specific Probabilistic Risk Assessments (PRAs), that systematic plant examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements.

The systematic examinations, defined as Individual Plant Examinations (IPEs), are intended to identify plant-specific vulnerabilities that contribute significantly to the overall risk from severe accidents. The maximum benefit from the IPE would be realized if the licensee's staff becomes involved in all aspects of the examination to the degree that the knowledge gained from the examination becomes an integral part of plant operations and training programs. Therefore, the staff is encouraging each licensee to use its staff to the maximum extent possible in conducting the IPE.

The IPE is to be directed at the two areas of severe accidents: (1) core damage prevention capability and (2) mitigation of the radionuclide releases associated with severe accidents. The methods used for the IPE should:

1. have the capability to find severe accident vulnerabilities and potential areas of improvement.

2. give proper consideration to visual examination of the plant.

3. give proper consideration to the insights from related analyses and examinations.

4. adequately document the methods used and display the results of the examination.

Typical examinations will be based on techniques developed for probablistic risk assessments to determine the system unavailabilities and accident sequence probabilities. This will involve definition of a complete set of initiating events and the development of event and fault trees that take into account single failures, common mode failures, and the failure propagation through interconnecting systems. Operator actions, human error, and plant design data, as well as actual as-built and as-modified data obtained by walk-through and talk-through are used in the examination process.

The dominant sequences are screened for both core damage frequency and containment failure rates. A set of screening criteria has been developed and is shown in Table 2.1. The screening criteria are not a measure of acceptability, but rather constitutes a set of thresholds for which sequences should be reported to the NRC in the IPE results report and may deserve further attention.

The results of the examination for core damage frequency are to include a listing of the driving features for the dominant sequences. These will include the systems, components and operator actions that influence significantly the core damage frequency.

Research and evaluation have shown the ability of many containment systems to survive pressure challenges of two to three times design levels and have the capability to withstand, to varying degrees, many of the challenges presented by severe accidents. For each type of containment, however, there remain failure mechanisms which could lead to either early or late containment

failure, depending on both the accident scenarios involved and the containment types.

Recent research results and studies have led to the identification of several key events and phenomena affecting containment performance under severe accident conditions. However, there are a few severe accident phenomenological issues, in particular direct containment heating and containment shell melt-through, where uncertainties are large and research has not produced definitive results on the challenges which these phenomena could pose to containment integrity. The staff does not plan to require each utility to perform analyses to assess the challenges such phenomena pose to their containment. However, the utilities should be aware of how these uncertainties can affect all possible outcomes of severe accidents so they will be prepared to handle them, should one occur. Therefore, the staff encourages utilities to give consideration to strategies to deal with those severe accident phenomena that account for these uncertainties. For example, although there appears to be no consensus on whether water availability will fully quench the debris and keep it coolable and hence prevent or substantially delay MARK I containment shell melt-through, there is a broad agreement that the presence of water will scrub the fission products and could substantially reduce the radionuclide release even if containment shell melt-through were to occur. Utilities are being made aware of these insights and experience (e.g., NUREG/CR-4920) so that when conducting the IPE they should develop appropriate strategies to minimize the effects of those phenomenological issues while awaiting for their generic resolution.

The examination is expected to identify those areas of potential improvement in the containment system that will minimize releases for the more likely severe accidents. The adequacy of the results from an IPE will be decided ultimately by regulatory judgement that will include, in addition to consistency with the Commission's Safety Goal Statement, an assessment of other factors such as the plant operational management.

2.2 Relationship Between IPE and Containment Performance Improvement

The IPE provides the mechanism for seeking out plant-specific design features that could affect the core damage frequency and influence the releases for the dominant severe accident sequences and would provide an understanding of the plant-specific factors important to containment performance.

For those severe accident phenomenological issues where research has not yet produced conclusive results, the NRC has established a research effort to deal with the prevailing questions. Since resolution is unlikely in the immediate future, utilities are not asked to make any design modifications concerning those issues to their plants at this time, but rather are asked to develop strategies to minimize the consequences of the challenges that such phenomena may pose to containment integrity. The Containment Performance Improvement Program is assessing the need for any further regulatory guidance for generic containment challenges.

2.3 Relationship Between IPE and Accident Management

Because the conclusions a utility will draw from the IPE for severe accident vulnerabilities (1) will depend on the credit taken for survivability of equipment in severe accident environment, and (2) will either depend on operators taking beneficial actions during or prior to the onset of severe core damage, or depend on the operators not taking specific actions which would have adverse effects, a logical outcome and important outgrowth of the IPE will be a severe accident management program. Utilities are being requested in the IPE generic letter to make all reasonable preparations for successfully executing measures they determine will help prevent core damage or prevent core debris from penetrating the vessel, if core damage could not be prevented.

2.4 Relationship Between IPE and USIs and GSIs

With regard to the ongoing work on USIs and GSIs, our experience shows that the resolution of many USIs and GSIs are not normally plant specific requiring

systematic examinations of plants and, therefore, can be reasonably excluded from the current IPE process. In addition, most USIs and GSIs have proposed resolutions identified and are scheduled for completion within approximately two years, in advance of the time when the majority of the IPEs will be received. However, recently we have concluded that for USI A-45, "Decay Heat Removal Requirements," there is no cost-effective generic solution to the issue, based on existing Commission backfit guidance, and cost-effective improvements can only be realized on a plant-specific basis. To accomplish this resolution would require each plant to perform a search to identify vulnerabilities associated with decay heat removal. Since such a search is already being performed by each plant as part of the severe accident policy implementation, the staff is proposing that USI A-45 be subsumed in the IPE. Another GSI that may be resolvable using the IPEs is GSI 121, "Hydrogen Issue for Large Dry PWRs." The main concerns are the possibility of containment failure resulting from global detonation in subatmospheric containments~~, or~~ of safety system failure resulting from local detonations or multiple burns in a hydrogen source compartment in any containment. Because these phenomena are a function of specific containment geometries and designs, the staff is examining whether plant-specific assessments of the potential adverse impacts of hydrogen burning or detonations should be conducted. The potential for conducting such assessments as part of the IPE containment performance assessments and hence subsuming GI 121 into the IPE process is also being considered.

2.5 Relationship Between IPE and NUREG-1150

The staff and its contractor have assimilated the insights gained from (a) previous industry sponsored PRAs, (b) NUREG-1150 analyses of five major containment configurations, and (c) IDCOR analyses of four containment configurations into five reports, NUREG/CR-4920, "Assessment of Severe Accident Prevention and Mitigation Features, BWR, MARK I Containment" with analogous reports for MARK II, and MARK III, Ice Condenser and large dry containments. It is not the intent of the information contained in these reports to specify a set of requirements to be applied uniformly to all plants.

Rather, they are intended to provide information and insights to the analysts performing the IPEs about plant features and operator actions that were found to reduce the overall risk for the relevant plants. It is our plan to issue these reports to the industry for their information but separate from the IPE letter. In addition, we will continue to pass on any new insights derived from NUREG-1150 to industry. In the IPE generic letter, the staff will encourage licensees whose plants have been extensively analyzed under the NUREG-1150 program to submit their IPEs on an expedited basis. This will enable the staff to excercise its review and decision process for determining acceptability of the IPE, the adequacy of the licensee identification of plant-specific vulnerablities, and the associated modifications by utilizing the extensive insights and experience from NUREG-1150.

2.6 Plan for Review of IPE Results

Following the issuance of the generic letter, workshop(s) with utility representatives will be scheduled to discuss the IPE objectives and to answer questions that utilities might have. A draft IPE Review Document that is being prepared by the staff to establish a framework for reviewing IPE submittals will also be made available to the industry and discussed in the workshop(s). The purpose of the IPE Review Document is to assure, as much as possible, the quality and uniformity of the reviews and to present a well-defined base from which to evaluate the adequacy of the IPEs. Following the completion of the workshop(s), the NRC, where appropriate, will revise the IPE Review Document and issue it in final form. In the generic letter, the staff is requesting the holders of operating licenses and construction permits of nuclear power plants to submit, within 60 days of issuance of the IPE Review Document, their proposed programs for completing the IPEs. This would include identifying the method, approach selected, milestones, and schedule for performing the IPE.

Current staff estimates are that although there are about 109 licensed plants in the U.S., replicate plant considerations would reduce the number of IPEs submitted to about 80. We also believe that there are perhaps as many as 25

plants with industry-sponsored PRAs already performed. The staff has reviewed approximately 15 industry sponsored PRAs over the past several years.

Based on the information presented in the licensee's response to the generic letter, the staff will select several lead plants to perform reviews that will be conducted by the staff and its contractors (see Section 2.5).

There are several purposes for the lead reviews. First, by reviewing several lead plants, the staff and contractor reviewers will gain valuable experience with regard to scope of the review, adequacy of the staff's review plan, and adequacy of the submittals. Other areas where valuable experience will be gained include methodology differences or assumptions that need careful attention to accurately interpret the results and the relationship of the individual results to those of other PRAs. This experience will be passed back to the licensees quickly so that later submittals meet the staff's needs.

The second purpose of the lead review concept is to assess the efficiency of the review team structure. Each plant review is to be the responsibility of one NRC team leader, several plant systems specialists and a PRA specialist. Two teams will do only large dry containments, one team will do MARK I and MARK II containments and one team will do MARK III and Ice Condenser containments.

Because the severe accident policy stressed that each utility should correct any severe accident vulnerabilities that are found, the submittal by each utility will also describe the disposition of each sequence it found to be a significant contributor to risk. It should also describe how amenable each significant sequence is to corrective action (i.e., procedure or hardware modification) and the basis for the utility's ultimate disposition of the item.

Currently, we estimate that a review of the IPE submittals will take an average approximately six person-months per plant. We also estimate that the

submittals will be received over a period of three years. RES will have the lead in evaluating IPE submittals and will form four teams, expending approximately 16 person-years effort per year (eight contractors, four from NRR and four from RES). Contractor costs for this review effort based on eight persons per year at $175K per person year are $1.4M per year.

Review results will be described by the review team in reports to the staff senior management. We fully anticipate that questions will arise in the course of the IPE reviews and plan to pursue these questions with the licensees. However, bearing in mind the voluntary nature of this information submittal, we do not intend to conduct our review with the formality of a design basis licensing review. Any critical issues regarding phenomenology, methodology, or data in which we reach an impasse with a utility will be brought to the attention of the Commission.

We expect each licensee to correct any significant severe accident vulnerabilities it finds. In the event that the staff disagrees with the conclusion of the licensee, particularly with respect to the need to fix a vulnerability, the staff will pursue the fix in accordance with the Commission's backfit rule.

Moreover, for those fixes that apply to a class of plants and go beyond the current regulations, the staff will have to pursue rulemaking in conjunction with the backfit analysis, since, with the exception of specific rules, the need for nuclear plants to accommodate severe accidents is beyond current regulatory requirements.

2.7 Comments on ACRS Letter Dated May 10, 1988

In its letter dated May 10, 1988 from W. Kerr to Chairman Lando W. Zech, Jr. (Proposed Generic Letter on Individual Plant Examinations and the Proposed Integrated Safety Assessment Program II), the ACRS generally found the current draft of the IPE generic letter to be improved over the version it commented on in a report dated June 9, 1987. The ACRS, however, recommended that the scope

of the IPE be ". . . broader than the original intent of searching for outliers," and require each licensee to conduct a full-scope PRA to subsume all outstanding safety issues, USIs, and GSIs. The ACRS also recommended that both internal and external initiators should be considered at this time. Conclusion about results and necessary changes should be determined by the licensees and reviewed by the staff through the ISAP process.

While we acknowledge and understand the ACRS recommendation and its basis, namely to provide a program that integrates a number of ongoing regulatory activities, we have concluded that it is not appropriate to implement the ACRS recommendation at this time. Our reasons for this conclusion are discussed below.

First, the Severe Accident Policy Statement was issued in 1985 to require licensees to perform a systematic examination of their plants to identify any plant-specific vulnerabilities to severe accidents, to understand the response of their plant to a severe accident, and report the results to the NRC. The staff developed the IPE generic letter package, which includes allowance for several optional IPE methods, including PRAs that could be used by utilities to participate in the ISAP II program at a future time. The IPE generic letter also allows, however, for use of the IDCOR IPEM method that was developed by industry in reponse to the policy statement. The staff has reviewed this methodology and concluded that, provided the staff recommendations are incorporated, the IPEM is an acceptable method of meeting the intent of the Commission's severe accident policy and, therefore, we have no basis for not allowing use of this method. We also feel that requiring all utilities to perform a PRA at this time will lead to unnecessary delay in implementing the Commission's severe accident policy. It should be noted, however, that the generic letter does not discourage, in fact encourages, utilities to perform PRAs and where appropriate, the staff will consider giving those utilities more time to finish their PRAs.

Second, the IPE generic letter does not discourage resolution of USIs and GSIs through the IPE program, rather, it encourages it. The letter states that if a utility concludes through the IPE and proposes to the staff that (1) an existing USI or GSI does not apply to their plant based on probabilistic considerations, (2) there is not shown to be a cost-effective resolution to the issue, or (3) a modification is proposed to resolve the issue, the staff will consider the issue resolved upon review and acceptance of the IPE results. The staff has decided to subsume A-45 into the IPE since in implementing the severe accident policy, utilities will perform an examination of the decay heat removal systems in addition to systems used for other safety functions. However, it is impractical to request utilities to resolve all USIs and GSIs since some of these safety issues do not require plant-specific examinations, and most GSIs and USIs have resolutions identified with resolution schedules within two years.

Third, the IPE generic letter does not request consideration of external events at this time, but advises utilities that they will be expected to examine and identify vulnerabilities to severe accidents due to externally initiated events separately. The staff agrees that a full scope PRA, if conducted in enough detail, is an acceptable approach to deal with external events. However, to be effective, ongoing staff and industry programs to develop guidance and procedures for external event reviews are needed as input to external event PRAs. In addition, PRAs are not the only acceptable methods of performing examinations for external event vulnerabilities. Deterministic methods, such as those being developed under the Seismic Design Margin Program, may be just as effective in identifying vulnerabilities at a lower cost. Whatever method that is used to conduct examinations for external event vulnerabilities, it must be integrated with a number of other ongoing regulatory activities involving external events, particularly in the seismic area. The External Events Steering Group, which has been chartered with developing guidance on external event IPEs and with integration of other activities related to external events, is expected to complete its work in FY 1989.

Based upon experience with simplified external event methods being developed for NUREG-1150 plants, it is believed that separation of internal and external event initiators will not result in loss of efficiencies or inconsistencies. This is as long as utilities recognize that an external events review will be required, and that they collect and retain information that will be helpful when later conducting the IPE for external events. In addition, the staff believes that it is unlikely that any plant modification which might be proposed on the basis of internal events would render the plant more vulnerable to external initiators. This is because, in general, external initiators do not introduce new sequences.

In summary, we believe that the IPE generic letter meets the intent of the Commission's Severe Accident Policy Statement and also addresses most of the concerns expressed by the ACRS. However, to expand the scope of the IPE to fully meet the ACRS recommendation cannot be justified and would result in a substantial delay in implementing the policy.

TABLE 2.1

Criteria for Reporting Important Severe Accident Sequences

The following screening criteria are proposed to be used to determine potentially important functional sequences and functional failures that might lead to core damage or unusually poor containment performance. All numerical values given in this Table are "expected" values.

1. Any functional sequence that contributes 1E-6 or more per reactor year to core damage.

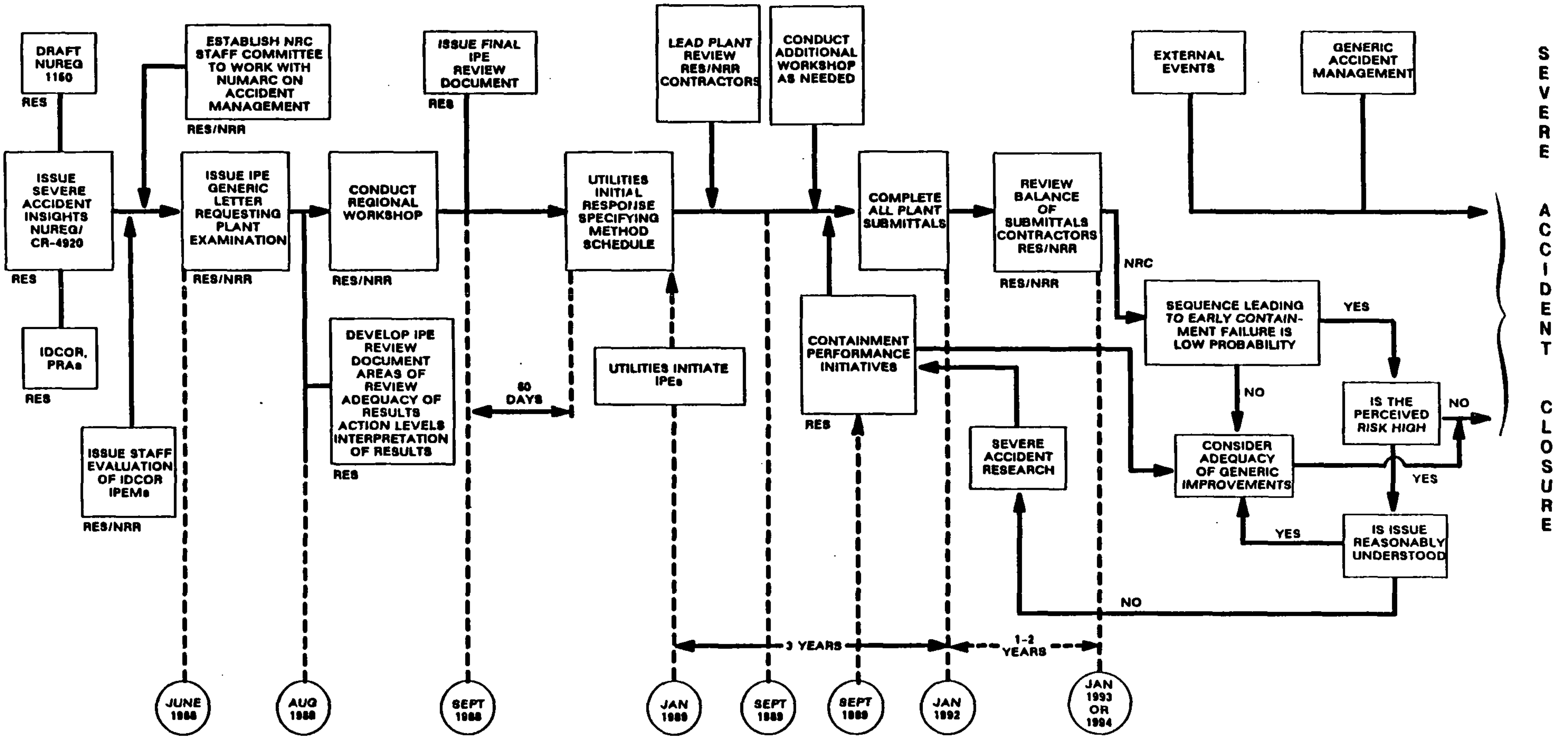
2. Any functional sequence that contributes 5% or more to the total core damage frequency.

3. Any functional sequence that has a core damage frequency greater than or equal to 1E-6 per reactor year and that leads to containment failure which results in a radioactive release of magnitude greater than or equal to BWR 3 or PWR 5 release categories of WASH-1400.

4. Functional sequences that contribute to a containment bypass frequency in excess of 1E-7 per reactor year, or

5. Any functional sequences that the IPE team determines from previous PRAs (e.g., as in NUREG/CR-4920) or by engineering judgment to be important contributors to core damage frequency or poor containment performance.

FIGURE 2.1 **PLANS FOR ADDRESSING INTERNAL EVENT VULNERABILITIES**



3. CONTAINMENT PERFORMANCE IMPROVEMENTS

A program to evaluate severe accident containment vulnerabilities was described for the Commission in SECY-87-297. The need for the program is predicated on the conclusion that there are generic severe accident challenges to each LWR containment type that should be assessed to determine whether additional regulatory guidance is warranted, or to confirm the adequacy of existing Commission policy. The bases for this conclusion include the relatively large uncertainty in the ability of LWR containments to successfully survive some severe accident challenges as indicated by draft NUREG-1150. All LWR containment types are to be assessed starting with BWR MARK Is. Staff recommendations for regulatory closure are forecast for MARK I containments in the Fall of 1988, and for the other containment types in August 1989.

The containment performance program consists of evaluations of generic severe accident containment challenges, failure modes and potential improvements. The challenges include early containment temperature and pressure excursions that would be expected before or immediately after a core melt which caused pressure vessel failure, late temperature and pressure excursions associated with the generation of noncondensible gases and steam spikes in the containment after vessel failure, and containment bypass (such as liner melt-through and basemat melt-through). Failure modes include overtemperature/overpressure-induced structural cracking, tearing, or penetration failures, and melt-through. The potential generic improvements would be the hardware and procedure modifications for both reducing the likelihood of containment failure and for limiting offsite accident consequences.

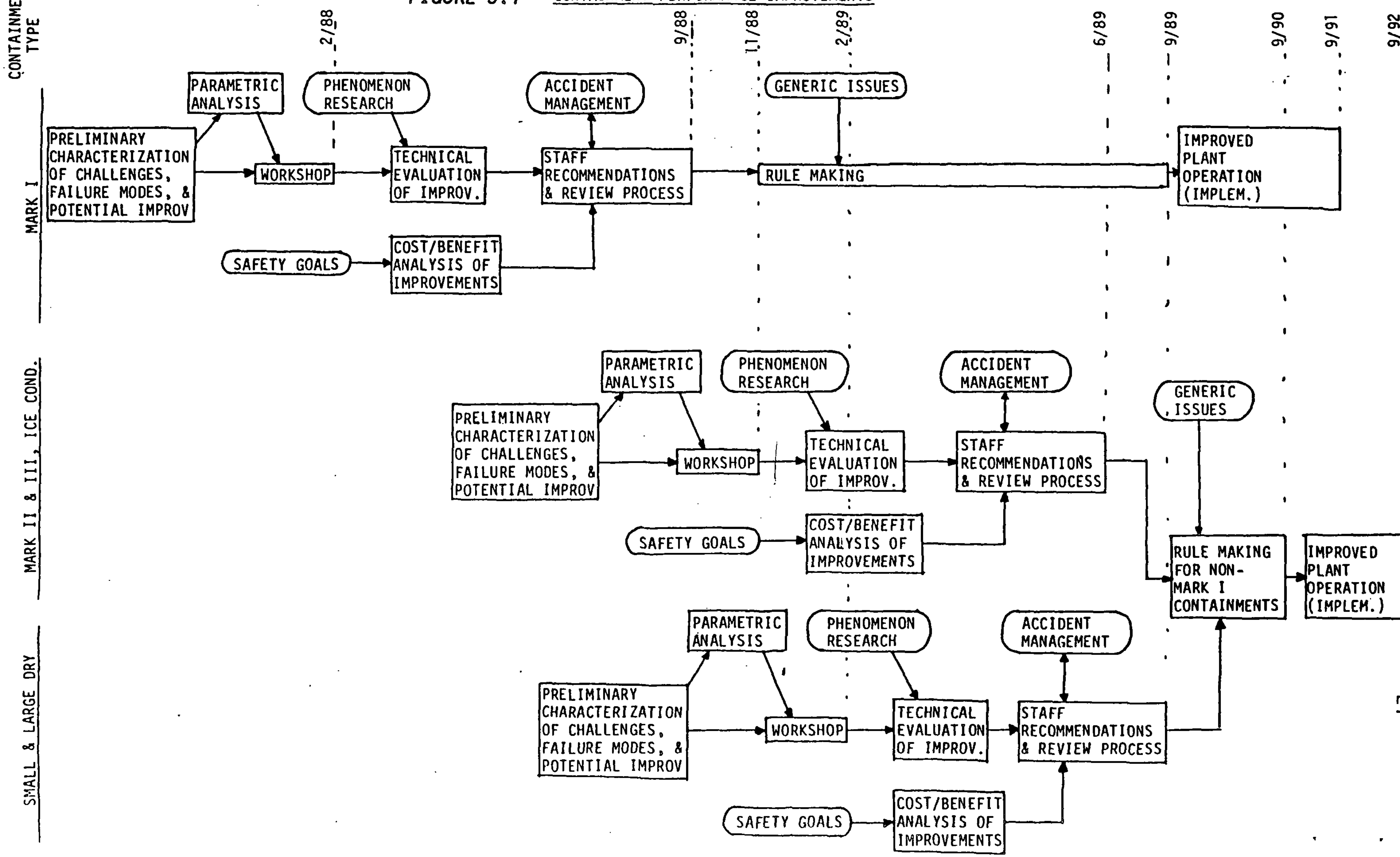
NUREG-1150 results are to be used to aid in identifying generic challenges and failure modes, and to provide a perspective on core melt and containment failure likelihoods and risks. Severe accident/source term research provides the basis for assessing core melt progression, in-vessel and ex-vessel combustible gas, fission product release and transport phenomena, and related uncertainties. This research is of two general types: that necessary to

provide sufficient information to effect regulatory decisionmaking on severe accident containment issues, and that necessary to confirm regulatory decisions. For the former, two areas have been identified as important: ex-vessel core debris control (shell melt-through) for MARK I containments, and direct containment heating for all containment types. Another potential area that may require additional research for regulatory closure is the manner in which a pressure vessel may fail and release debris to a containment.

The staff's criteria for judging the adequacy of industry decisions associated with IPE results are primarily those associated with the backfit rule (10 CFR 50.109) and the Commission's safety goals. Experience with application of the backfit rule indicates that any newly proposed regulatory guidance that results in substantial new requirements is not likely to be justified because of relatively low contemporary estimates of risks. However, inclusion of considerations of averted onsite and offsite property damage costs may alter such a view. Application of the Commission's safety goals would indicate that the early and latent fatality goals are not an issue with existing LWR containments. However, the proposed implementation criterion that a large release have a probability of 10^{-6}/reactor-year or less may be an issue requiring further policy considerations. Similarly, other issues such as venting may also indicate the need to consider the adequacy of existing Commission policy. Forthcoming Commission papers on containment performance initiatives and safety goal implementation will address these issues in more detail.

Because generic containment performance evaluations may be under way at the same time as plant-specific assessments, the question of potential duplication arises. This question has been resolved by informing the industry in the IPE generic letter that containment performance issues associated with generic severe accident phenomena with high uncertainties (i.e., liner melt-through and direct containment heating) need not be considered in the IPE.

FIGURE 3.1 CONTAINMENT PERFORMANCE IMPROVEMENTS



4. IMPROVED PLANT OPERATIONS

This program area is defined to include those regulatory activities which will yield improvement in the way plants are managed and operated by plant personnel. This approach is distinctly different from activities pointed toward improvements in the design or as-built configuration of plants' systems and components. Operating experience, precursor studies and probabilistic risk analyses all indicate that operator performance can affect the likelihood of substantial core damage in a significant manner. Current PRAs indicate that operator actions to recover from equipment failures can significantly reduce the likelihood of core damage. Conversely, errors in testing and maintenance activities, when coupled with equipment failures, and failure to follow emergency procedures have been found to contribute significantly to the assessed frequency of core damage.

The above considerations lead to a realization that a plant which appears to present a low risk, based on a PRA that represents the plant design and operational practices at some point in time can actually be operating at a significantly higher risk because of deterioration of the many human performance attributes which can influence the ability to cope and the attitude under which maintenance and testing are performed.

Known important and favorable human performance attributes include a well-trained staff that rigorously follows procedures, a diligent, probing plant operations review committee, and effective preventive maintenance. Some plant indicators that seem to go with good performance attributes are plant-specific training simulators, full staffing, very little overtime, a high standard nuclear work ethic, professional decorum in the control room, low incidence of scrams, a plant that usually shuts down to fix safety systems, a low maintenance backlog, immediate repair of malfunctioning equipment, a generally clean plant, and systems engineers available onsite.

The staff has concluded that improvements in plant operations will reduce challenges to plant safety and will enhance both human and equipment response to transients, thus avoiding or significantly reducing the probability of severe accidents. Several programs have been established to contribute to the enhancement of safe nuclear power plant operations. Taken as a whole these initiatives are seen as proactive in coping with severe accident scenarios. Some of these programs are:

4.1 Operator Licensing & Training

The staff typically administers reactor operator and senior reactor operator examinations to over 1000 qualified candidates annually. This activity termed "replacement examinations" has been on-going for more than twenty-five years. Much more recently INPO has accredited RO and SRO training programs and announced in the Spring of 1988 their intention to accredit continuing training as their eleventh program. The staff's efforts have taken a parallel path in the development of a new "requalification examination." The primary goal of this entire licensing and training activity is to ensure that power plant operators are current on all important operational aspects of their facility. This area, considered one of NRC's highest priority activities, encompasses training of power plant staff, utilization of simulators, scram reduction, adherence to procedures, and a strong nuclear work ethic. This activity includes the NRC efforts to ensure complete operator familiarization with EOPs.

4.2 Human Factors

Following the TMI accident the NRC promulgated numerous requirements to help solve the operational problems identified following the event. One of these is related to the staff's intent to assure that licensed operators are provided with accurate, reliable and useful information in the event of a plant transient or severe accident condition. An aggressive human factors program was initiated to enhance the information flow to the plant operations staff. The staff focused on upgrades to control boards, including upgrades in the information presented,

to assure that it was accurate. The safety parameter display system was designed to synthesize and provide on line analytical results to track and trend transient or accident conditions. Symptom oriented procedures were developed to assist the operators in responding to the information displayed on the control boards. These were considered essential programs to effectively improve the operator's ability to respond to an event. The three initiatives were an integrated attempt to logically translate plant conditions to correct operator actions. The staff, in bringing the EOP reviews to closure, has chosen to perform on-site walk-downs of the EOPs to be assured that they can be carried out, if necessary.

The EOP verification program assures operator knowledge and familiarization with systems and procedures which contribute to preventing severe accidents. The SPDS will be used to help prevent accidents by providing a continuous display of critical safety functions. The SPDS will also be used to monitor and manage the course of an accident from the EOP and Tech Support facilities. The staff has a multi-disciplinary task under way to develop general guidelines for use by Technical Support Center staff in the event of a severe accident. The effort embodies many of the good practices identified in the human performance and human factors areas. Other key areas addressed by the human factors discipline included establishment of an overtime policy, efforts to encourage a good nuclear work ethic, professional decorum, and the establishment of a rule in 1984 on minimum shift staffing.

4.3 Performance Evaluation

Performance evaluation methodology will improve the staff's ability to develop performance trends for individual plants. This will provide an information base to support anticipatory actions that might be taken to respond to degrading performance by a plant before significant safety concerns arise. The product of this activity will be the focusing of NRC attention on safe operation and all supporting functions. Key inputs to the NRC's overall evaluation of plant performance include: performance indicators, SALP results, Inspection Report

results, status of closure of generic issues, monthly operating reports, Licensee Event Reports and the results of special inspections.

Of note is NRC's development and implementation of quantitative indicators of plant safety performance. This set of performance indicators, together with other information such as inspection results and Systematic Assessment of Licensee Performance (SALP), provide input to NRC management decisions regarding the need to adjust plant-specific regulatory programs.

Staff is continuing to develop more objective and leading indicators, particularly for maintenance and availability of safety systems. These improved indicators will further enhance NRC's capability to recognize areas of poor or declining safety performance of operating plants.

4.4 Qual~~ity~~ Assurance

Quality assurance programs have been reoriented toward evaluating the implementation of utility quality verification programs in the identification and correction of safety significant technical problems. The performance oriented inspection activity concentrates on known problem areas such as software and configuration management. The product of this activity contributes directly to the improvement of plant operations by providing performance based feedback on the quality of operations and associated interfaces. As performance is enhanced the degree to which people and hardware can be relied upon are enhanced.

4.5 SALP/Self Assessment

The SALP methodology has recently been revised to place more emphasis on operations, maintenance and management. The staff is also developing revised policy guidance to seek improved licensee self assessment capabilities. These activities, in parallel with strong utility self assessment, lend themselves

strongly to ensuring enhanced plant performance. Through critical self assessment, licensees are expected to set and achieve higher standards in reducing challenges to the reactor protection system, in effective and efficiently managed activities for both on-site and off-site review activities, and in a strong on-site presence of dedicated systems engineers.
These initiatives provide insights to the tough questions which may be asked in the event of a severe accident. Critical self assessment lends itself to solving problems, identifying solutions, and developing general preplanned guidance to mitigate severe accidents.

4.6 Maintenance

The staff plans to issue guidance for region-based team inspections of licensee maintenance performance. The staff will also conduct maintenance inspections to determine what procedure changes or rulemaking initiatives are needed to improve plant maintenance operations. The product of this activity contributes to the overall improvement of plant operations through improved equipment availability.

4.7 Management

The staff routinely performs reviews of management and organizational effectiveness. Effective leadership is known to be a dominant contributor to safe plant operation. Through effective management and organization many of the above key attributes are addressed. Of significance is the role management must play should a utility be confronted with a severe accident. Management has the responsibility of either making or concurring in difficult decisions regarding all aspects of severe accident mitigation. Their decision making process is viewed, in an unchallenged state, to be an important predictor of performance in responding to critical plant challenges.

Also of significance is the licensee management's role in integrating routine plant activities in ways to help prevent accidents. For example, management

has the responsibility for establishing a closed-loop process for: monitoring plant performance; comparing performance with targets; identifying performance deviations and prioritizing them; and for important deviations (i.e., potential problems), identifying the causes, taking corrective action, and monitoring plant performance to verify that the corrective action worked to prevent important potential problems from occurring.

4.8 Technical Specifications

The NRC's Technical Specification Improvement Program has been developed with the principle goal of streamlining the TSs to remove line items, sections, or even chapters found to have little or no safety benefit. The resulting TSs are envisioned to be the minimal number necessary to assure regulatory requirements are fulfilled. TSs serve as a limiter in restricting plant operations to a specific envelope of operation. These restrictions provide assurance that power plants operate within all safety limits.

4.9 Inspections

NRC performs a number of different inspections focused on operations, maintenance, engineering, and outage management. In addition routine inspections encompass a full range of plant activities to include areas such as HP, chemistry and procurement. The staff performs risk based inspections including use of PRA in designing and conducting inspections, realizing that PRAs show risk of severe accidents (severe core damage). Dominant sequences are explored in-depth with key vulnerabilities being identified. These inspections provide valuable insight to the staff and utility regarding the actual plant condition versus a utility perceived condition. These inspections may find root causes of deficiencies that could lead to accidents beyond DBAs and may provide results on how such findings could be used to help move NRC toward resolution of severe accident concerns.

5. SEVERE ACCIDENT RESEARCH

5.1 Background

The Severe Accident Research Program (SARP) was begun after the TMI-2 accident in March, 1979, to provide the Commission and NRC staff with the technical data and analytical methodology needed to address severe accident issues. The initial work was quite general in nature due to the paucity of information available at that time, and consisted of experiments and code development of wide scope. Today the research results and models (including the insights derived from the five plant risk studies of NUREG-1150) are being applied to other Agency severe accident related regulatory activities such as Containment Performance Improvements (CPI), Accident Management (AM), and the Individual Plant Examination (IPE) project among others.

During the course of the SARP work, many research programs were conducted encompassing a wide scope of severe accident phenomenology. They consisted of (a) experiments and data analysis, (b) model development and validation, and (c) risk assessment and regulatory applications.

The experiments and data analyses either addressed or are addressing major severe accident issues and phenomena such as core melt progression; fission product release, transport, deposition, and revaporization; core/concrete thermal and chemical interactions; containment loading by high pressure melt ejection (direct heating), hydrogen detonations and burning, overpressure, and overtemperature; containment performance testing; and effects of natural circulation in the primary system.

Model development and validation consisted of the establishment of state-of-knowledge computer codes at two levels of complexity: namely, (a) highly-detailed, slow-running but accurate models and, (b) fast-running, simplified models. The fast-running models are being used for risk assessment and source term studies. The detailed models are being used to benchmark the

faster running models (i.e., determine their accuracy relative to detailed analyses), and for specific answers to risk-significant issues. All the data from the experimental program are being used to assess the validity of both types of models.

The risk assessment program assured that Probabilistic Risk Assessments (PRA's) used by the Commission include state-of-knowledge methodology and the procurement of up-to-date data on system reliability. The most recent accomplishment in this area utilized the foregoing data base and models to analyze and determine the risk of five U.S. LWR plants; i.e., plants chosen from each of the five major U.S. containment types. The resulting document (NUREG-1150) is currently being revised for final publication and its purpose and use is discussed in detail elsewhere in this document.

The SARP effort has greatly enhanced our ability to analyze severe accident phenomena and has even identified certain phenomena that were not completely considered during the Reactor Safety Study (WASH-1400), such as (a) the potential for early failure of PWR containments by Direct Containment Heating (DCH) induced by high pressure melt ejection and the potential mitigating effects of natural circulation in the Reactor Cooling System (RCS), and (b) the possible failure of BWR MARK-I containments by melt-through of the steel containment shell caused by interaction with the corium/concrete mass, especially under dry cavity conditions. For these phenomena, additional research is needed to help reduce the wide range of uncertainty associated with their likelihood or their effect on containment integrity. However, issues associated with such identified phenomena are amenable to a technical solution using existing technology.

In summary, the SARP has provided the Agency with new state-of-knowledge analysis capabilities. Among these are the simplified codes such as the Source Term Code Package (STCP), and its replacement, (MELCOR); interim specialized BWR models (BWR-SAR), and the highly-detailed mechanistic codes MELPROG/TRAC, SCDAP/RELAP5, CORCON/VANESA, and CONTAIN. These latter code

packages have been and will continue to be used for validation and benchmarking of simplified code results, especially for accident management studies, and potentially for following the course of any real accident.

Over the next 3-5 years research will (a) continue to develop information in the areas of most significance to risk, and (b) continue confirmatory work, including code assessment and validation. The remainder of this chapter will be devoted to showing the coupling of the severe accident safety issues with the research needed (both on-going and future) to address them by focussing on the safety issues associated with the most risk-significant challenges to the five major classes of containments. The chapter will also address (a) research that supports some of the other related severe accidents tasks involved in the implementation of the Commission's severe accident policy such as the IPE, Safety Goal, and Accident Management, (b) research in support of the resolution of specific generic issues, and (c) longer term research needed for decision confirmation, further model improvement, or the development of new analytical tools.

5.2 Relationship Between SARP and Containment Performance Improvements (CPI)

To a large extent the phenomenological research and associated code development has been directed to address some of the more significant issues associated with the integrity of the different containment types utilized in the United States (i.e., to understand and quantify the challenges to containments as a result of the in-vessel and ex-vessel processes and thermal/chemical interactions which can take place during the evolution of a severe accident). In this section, the risk-significant potential early containment failure modes for each of six containment types are discussed. Note carefully that the potential containment failure modes discussed in this section of the report on Severe Accident Research are conditional upon substantial core damage and core melting and in many instances melt-through the reactor pressure vessel itself and subsequent ex-vessel thermal-hydraulic-material interactions. This section does not address the absolute probabilities for these failure modes (i.e., core melt probability x containment

failure probability given core melt) which are addressed in such documents as NUREG-1150 as well as other PRAs. Also, the probabilities shown for the specific failure modes for each containment type are relative to each other and as indicated above are always conditional upon core melt. The SARP is being directed to address these issues and to further reduce the uncertainties associated with them. A schematic illustration of the flow of information for the containment performance research is shown in Figure 5.1. Information developed from these programs which addresses the issues and the phenomena associated with these potential containment failure modes will be utilized in the formulation of staff conclusions regarding possible containment performance improvements and related accident management strategies, with additional research, well focused on specific issues or sub-issues, continuing on a confirmatory basis.

5.2.1 MARK I & MARK II BWR Containments

The three risk significant potential early failure modes for these containments which require input from research are:

1. Failure by melt-through of the steel containment (MARK Is only) from interaction with the corium/concrete melt, especially under dry cavity conditions.

2. Failure by overpressurization due to non-condensible gas generation and steam.

3. Failure by overtemperature due to core/concrete interactions.

The research tools (i.e., models and data) required for decisionmaking on improvements for these containments were developed as part of the ongoing SARP and Source Term studies over the past several years. Research has been conducted both on containment loading phenomena and containment performance. The latter work is nearing completion to the extent that pressure ranges for failure and failure modes of these containments are fairly well-defined and

narrow enough to preclude the need for further studies. For the three failure modes listed above, only the first remains an outstanding issue. The last two are relevant only with respect to how late in time the failure will occur, and the source term associated with that failure.

It should be obvious from the description of the potential early failure modes listed above that the resolution of Item 1 requires accurate modeling of the core/concrete interactions and the initial conditions of the core melt (i.e., mass, temperature, and composition). Data from the core/concrete research programs at SNL has resulted in the development of the state-of-knowledge code module CORCON/VANESA which models the interactions with concrete and the release of fission products and hydrogen, steam, and carbon oxides. In March 1988, a new gas release model was developed which now accurately computes energy production in metal-rich core/concrete debris, and models the heating and spreading of debris much more realistically. Moreover, more accurate thermodynamic and material property data including the proper phase relationships for more complicated systems than the oversimplified urania-zirconia system used in former studies are being developed and will be available by June, 1988.

A fourth potential mode of failure involving the interaction of the molten core with water may be important for those unique MARK II and MARK III designs in which there exist direct pathways between the vessel and the water from the suppression pool. The introduction of molten core into the suppression pool may produce an ex-vessel steam explosion which could potentially fail the pedestal and lead to failure of the containment at a major penetration. A specific plant (Shoreham) is being analyzed to assess the likelihood of this phenomenon.

Longer-term research required for confirmation of decisions to be made in the near future is focussed on a realistic understanding of the in-vessel core melt progression process which determines the initial conditions of the exiting corium (mass, temperature, and composition). It should be realized, however, that obtaining accurate data at the stage of melt progression near vessel failure is very difficult and expensive, and fraught with many potential uncertainties.

Less important phenomena for this class of containments include (a) failure by steam explosions (very low probability), (b) failure to isolate (amenable to engineering, procedural and quality assurance fixes), (c) failure by hydrogen burns/detonations (very low probability), (d) early failure by steam spikes (low probability), and (e) failure by interfacing systems LOCA sequences (variable probability; amenable to engineering, procedural and quality assurance fixes).

5.2.2 BWR MARK III Containments

The two risk significant potential early failure modes requiring input from research are:

1. Failure by hydrogen burns and/or hydrogen detonations.

2. Failure by overpressurization due to non-condensible gas generation and steam.

The second mode above is the same as item 1 for MARK Is and MARK IIs and has been covered in the previous section. It should be noted that this mode is less of a threat in MARK IIIs because of their larger containment volume.

For failure by hydrogen burns or hydrogen detonations under severe accident conditions, the SARP has provided new data and models over the years via Commission supported programs as well as by intensive cooperative research with the industry. The results are generically applicable to all containment types. Large and small scale experiments have provided the data to develop and verify two analysis codes to study the effects of hydrogen in containments; namely, HECTR and HMS-BURN. The HECTR code is a lumped parameter code for fast analyses, whereas HMS-BURN is a slow, highly-detailed code, used for benchmarking HECTR results. Both codes are state-of-knowledge analysis packages and have been validated enough to use with confidence. The simplified HECTR models have been incorporated into the Source Term Code Package for

integrated analyses. Relative to hydrogen detonation issues, the CSQ code, which was developed in the weapon's program, is used to evaluate the energy and the structural response, given a detonation. The recently developed model ZND can be used for a prediction of the detonability of a given mixture of hydrogen, air, carbon oxides, and steam.

Further research, discussed below, is focused on continued assessment of the combustion codes' ability to predict the following phenomena:

a. Hydrogen transport and mixing.

   As part of a cooperative international program, NRC is providing support for large-scale mixing experiments (to ascertain the potential for detonable mixtures under severe accident conditions). The data will be used to validate and improve current mixing models in the NRC's HECTR and HMS-BURN codes. Updated code versions will be available for use by July 1989.

b. Behavior of diffusion flames.

   An assessment of recently completed large-scale experiments is being conducted to improve the flame propagation models in the NRC codes to better characterize the extent, consequences, and safety significance of a severe accident hydrogen burn. Updated code versions will be available by November 1988.

Less important phenomena for this class of containments include (a) failure by steam explosions (very low probability), (b) failure to isolate (amenable to engineering, procedural and quality assurance fixes), and (c) failure by an interfacing LOCA sequence (variable probability; amenable to engineering, procedural and quality assurance fixes).

5.2.3 Large Dry & Subatmospheric PWR Containments

The only risk significant phenomenon for these containments for which major research input is needed for decisionmaking is that of potential early failure by Direct Containment Heating.

Research in this area is four-pronged in nature. That is, (a) what is the probability of occurrence of a high pressure melt ejection (i.e., will failure be induced at other locations in the RCS?), (b) what is the role of the containment configuration in mitigating debris dispersal and pressurization, (c) how to manage such an event for example via depressurization using present hardware, hardware changes, and/or procedures and (d) what is the probability of early containment failure given (1) that high pressure ejection occurs, (2) the debris or the hydrogen generated through chemical reactions is effectively dispersed and (3) hydrogen combustion takes place.

For item (a) analysis indicates that the Reactor Coolant System (RCS) may fail at other locations in the RCS prior to lower head failure due to local heating effects caused by natural circulation of hot gases within the RCS. Such a failure, if large enough, will effectively depressurize the RCS and preclude the ejection of the core at high pressure, thus eliminating the DCH mode of early containment failure. (Evaluations are underway to understand why natural circulation induced heating seems not to have occurred at TMI-2.) Experimental work at INEL on the creep-rupture characteristics of RCS components is just now being completed along with natural circulation experiments by Westinghouse to assess and validate the codes' predictions on natural circulation effects. The natural circulation research should be completed in FY 1989. Depending on the outcome of this work this failure mode for PWRs may not be as important as perceived up to now.

For item (b) above, research into the effects of containment configuration will be completed in FY 1988. The work utilizes analyses and small-scale experiments to simulate various configurations to determine the overall effect

on pressurization. The research to date indicates that cavity geometry has little or no effect on debris retention (all being dispersive) for the higher pressure end of the pressure range. Therefore, the geometry of the lower subcompartments of the containment may be significant.

In the third area, research is being concentrated on determining the ejection pressure at which potential containment failure by DCH is no longer a threat. The cavity geometry may be important in this determination. Both SNL and BNL have scheduled their studies to fall within the decisionmaking schedule for this containment type. Preliminary results are due in April, 1988, with final applicable results by August, 1988. Accident management studies will also make a major contribution in this area by computing depressurization possibilities and their effectiveness.

To develop predictive capability for the decisionmaking process regarding consequences of DCH, the longer-term objectives of the research will attempt to reduce uncertainties in the following three areas:

1) Determination of Initial Conditions of the Melt Ejection -

   The mass, temperature and composition of the melt, and the RCS pressure, at the time of the RPV bottom head failure, as well as the size of the failure, all affect the severity of DCH. Research in this area involves the study of core melt progression through the bottom head failure. As was discussed in the case of BWR MARK I and MARK II containments, experiments for the back end of this phenomenon are difficult and expensive to conduct. Long-term work ongoing at SNL, at the NRU facility, and at the CORA facility in FRG should reduce uncertainties in this area over the next five years. Immediate results are not possible because of the complexity of the task and the need for further research.

2) Distribution and Interactions of the Core Debris in the Containment -

How well the ejected core melt mixes, and how quickly the debris particles interact, chemically and thermally, with the containment atmosphere will determine the magnitude of the peak containment pressure. Large-scale experiments in the Surtsey facility at SNL will provide data bases for development of analytical models regarding debris transport, ex-vessel metal-steam reactions, heat transfer, and containment heating and pressurization. Such models will be incorporated into the CONTAIN code. The possible mitigating effect of water will also be investigated in the Surtsey tests. In addition, small-scale tests using low temperature melt simulants are being conducted at BNL to determine the influence of containment structures on core debris dispersal, and consequently on the containment heating.

3) Aerosol Generation and Distribution -

Should the containment fail from DCH, sources of airborne radionuclides would be of the utmost concern. The Surtsey tests will provide data for development of predictive models to calculate aerosol generation during a pressurized ejection of the core melt from the RPV.

In addition to the above research on containment loading, SARP is continuing to provide valuable information on the performance characteristics of these containments by testing scaled-down models of steel and concrete containments to failure. Most testing has been completed and the results are available for decisionmaking. Future tests of importance include the British test of an unlined prestressed model in the Fall of 1988, and NRC tests on liner failure modes in the FY 89-90 time frame.

Less important phenomena for this class of containments include (a) failure by steam explosions (very low probability), (b) failure to isolate (amenable to engineering, procedural and quality assurance fixes), (c) failure by hydrogen

burns/detonations (low probability), (d) early failure by steam spikes (low probability), (e) late failure by overpressurization (high probability but low source term), (f) late failure by basemat melt-through (medium probability but low source term), and (g) failure by an interfacing LOCA sequence (variable probability; amenable to engineering, procedural and quality assurance fixes).

5.2.4 Ice Condenser PWR Containments

The two risk significant severe accident potential early failure modes for this class of containments requiring input from research are:

1. Failure by hydrogen burns and/or hydrogen detonations.

2. Failure by direct containment heating (DCH).

The uncertainties and related research for failure by DCH was covered under Section C. As in the case of BWR containments, evaluation of the conditions that could lead to failure from hydrogen burns and/or detonations can be addressed by using the NRC hydrogen behavior codes HECTR and HMS-BURN for hydrogen combustion and the CSQ and ZND models for hydrogen detonation. For ice condenser containments some improvement is needed in assessing the modeling of the transition from deflagration to detonation. Previous experiments on this phenomenon directed toward this containment type are being documented. These results will be used for immediate reassessment of the codes in this area so that decisionmaking on this failure mode can be made by September, 1988.

Less important phenomena for this class of containments include (a) failure by steam explosions (very low probability), (b) failure to isolate (amenable to engineering, procedural and quality assurance fixes), (c) early failure by steam spikes (low probability), (d) late failure by overpressurization (high probability but low source term), (e) late failure by basemat melt-through (medium probability but low source term), and (f) failure by an interfacing LOCA sequence (variable probability; amenable to engineering, procedural and quality assurance fixes).

5.3 Relationship Between SARP and Accident Management

Accident management is a set of practical actions that the plant's operating and technical team can take to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the damaged core within the reactor vessel, (3) failing that, maintain containment integrity as long as possible, and (4) ultimately to minimize the consequences of offsite releases. Given the initiation of a severe accident, the objective is to bring the plant to a safe, stable state. The preparations to achieve this objective made by the utilitiies in advance of a severe accident provide further assurance that the health and safety of the public will be protected. Accident management is discussed in Chapter 6.

Accident management preparations both (1) apply to the specific vulnerabilities at individual plants and (2) remain current by being based upon the current state of collegial knowledge about severe accidents. The collegial knowledge about severe accidents is currently rather extensive. However, there are some phenomenological questions that severe accident research has not yet conclusively answered and some initial answers where significant residual uncertainties still remain. It is these phenomenological questions and initial answers that are to be concluded by the ongoing severe accident research. The staff expects the ongoing research to expand the collegial knowledge about severe accident phenomena such that accident management actions will have a strong technical basis.

5.4 Relationship Between SARP and Individual Plant Examinations (IPEs)

Two major aspects of the IPE program are supported directly by SARP: (1) preparation of the Severe Accident Prevention and Mitigation Features report (under preparation), and (2) reviews of the IDCOR IPEMs.

The Severe Accident Prevention and Mitigation Features report identifies plant features and operator actions that have been found to be important to either

the prevention or the mitigation of severe accidents for a specific plant containment type. The report summarizes the insights gained from SARP, industry-sponsored PRAs, NUREG-1150, and IDCOR reference plant analyses, and indicates what may be important to risk. The report addresses the general issues of (1) survivability of equipment, (2) equipment capabilities, capacities, and duration of operability, (3) accessibility of equipment, (4) availability of support systems, (5) identification of necessary components, (6) identification of important operator actions, and (7) identification of parameters for initiation of mitigating systems and operator actions.

The IDCOR IPEMs review has recently been completed. The review was performed by NRC staff and was based on the same SARP-supported technology discussed above.

According to the current draft generic letter, IPEs should be completed by the licensees within three years. By the time NRC reviews are initiated, major results will be available from SARP on some of the phenomenological issues which relate to containment performance and where research is still going on: Among these issues are containment shell melt-through, high-pressure fuel debris dispersal, and hydrogen combustion. These SARP research results will be used by NRC to complete the staff review of licensee IPE submittals (either to confirm decisions already made or to take other actions as necessary).

5.5 Relationship Between SARP and USIs and GSIs

The following generic safety issues and licensing issues are related to severe core damage accidents for which severe accident research is applicable:

- A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment" and GSI-121 "Hydrogen Control for Large, Dry PWR Containments."

- For these issues, past experiments have provided data on the fundamentals of hydrogen combustion including the effect of aerosols and steam on

flammability limits and the possibility of deflagration. Future research of a confirmatory nature includes the investigation of the catalytic effect of steam at high temperatures on hydrogen combustion, the possibility of detonation in various LWR containments, and flame acceleration experiments to address remaining uncertainties on this subject.

- GSI-II.H.2, "Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure."

- An experimental program has been underway in cooperation with DOE to obtain data on all aspects of the TMI-2 accident. Technical data on the TMI-2 containment structure will continue to be assessed and put into the overall SARP data base.

5.6 Relationship Between SARP and NUREG-1150

In February, 1987, the Agency published the "Reactor Risk Reference Document," NUREG-1150, in draft form for public comment. This report summarized the results of, and provided perspectives on, the study of severe accident frequencies and risks for five licensed nuclear power plants.

The principal objective of NUREG-1150 was to provide a snapshot of the state-of-the-art of PRA technology, incorporating improvements since WASH-1400. The report utilizes the large amount of data and new phenomenological models developed from the SARP since WASH-1400.

In particular, the codes used for the phenomenological analyses were developed by SARP and the calculations for NUREG-1150 were actually performed under existing SARP programs.

In addition, newly addressed phenomena such as DCH and natural circulation effects were studied using the SARP detailed models and the data base accumulated to date. The SARP program results will continue to support such risk studies in the future by providing state-of-knowledge analysis codes and data on severe accident phenomenology.

FIGURE 5.1

CONTAINMENT PEFORMANCE FOCUSSED RESEARCH

PWR LARGE DRY		BWR MARK I & II		BWR MARK III		BWR ICE CONDENSER	
MAJOR ISSUES	RELATED RESEARCH	MAJOR ISSUES	RELATED RESEARCH	MAJOR ISSUES	RELATED RESEARCH	MAJOR ISSUES	RELATED RESEARCH
DIRECT CONTAINMENT HEATING (DCH) (EARLY)	D* - PROBABILITY (NATURAL CIRC.) D - CUTOFF PRESSURE D - MANAGEMENT (DEPRESSURIZATION) C** - CONSEQUENCES (SURTSEY TESTS) C - INITIAL CONDS. (MELT PROGRESSION)	SHELL MELT THROUGH MARK-I ONLY (EARLY)	D - MELT SPREADING TESTS D - HEAT TRANSFER TO LINER TESTS C - MELT SPREADING USING VARIOUS CORE DEBRIS C - MODEL COMPLETION C - INITIAL CONDS. (MELT PROGRESSION)	HYDROGEN BURNS & DETONATIONS (EARLY)	C - ASSESS COMBUSTION CODES WITH EXISTING DATA	HYDROGEN BURNS SAME AS MARK-III DCH - SAME AS PWR LARGE DRY (EARLY)	
		PEDESTAL FAILURE BY EX-VESSEL STEAM EXPL. (EARLY)	D - ANALYSIS OF SHOREHAM PLANT				
OVERPRESSURE OVERTEMP. (LATE FAILURE)	C** - LARGE-SCALE CCI TESTS*** C - FRG BETA TESTS ON CCI C - IMPROVE & ASSESS CCI CODES C - INITIAL CONDS.. (MELT PROGRESSION)	OVER P&T SAME AS PWR LARGE DRY (EARLY FAILURE)	D - MANAGEMENT (DEPRESSURIZE) (DRYWELL)	OVER P&T SAME AS PWR LARGE DRY (EARLY)	D - MANAGEMENT (DEPRESSURIZE)	OVER P&T SAME AS PWR LARGE DRY (LATE FAILURE)	

KEY
*D = DECISIONAL RESEARCH (COMPLETED BY 08/89)
**C = CONFIRMATORY (LONG-TERM) RESEARCH
***CCI = CORE/CONCRETE INTERACTIONS

6. ACCIDENT MANAGEMENT

6.1 Introduction

Through high standards imposed on the design and construction of nuclear power plants, as well as on the qualifications and training of the operating staff, severe accident probabilities have been made extremely low. However, they cannot be completely eliminated. Nevertheless, certain preparatory and recovery measures can be taken by the plant operating and technical staff that could prevent or significantly mitigate the consequences of a severe accident (i.e., accident management). Broadly defined, accident management includes the measures taken by the plant operating and technical staff to (1) prevent core damage, (2) terminate core damage if it occurs and retain the core within the reactor vessel, (3) failing that, maintain containment integrity as long as possible, and finally (4) to minimize the consequences of offsite releases. Thus, accident management is viewed as an important means of achieving a substantial reduction in risk from severe accidents.

The capability to effectively respond to reactor accidents is routinely assessed through various regulatory activities, including staff review of emergency procedure guidelines, operator qualification and training programs, and offsite emergency response plans. In general, however, the emphasis of these activities is either on assuring adequate core cooling, or on assuring effective offsite emergency response in the event that core damage were to occur. Very little regulatory emphasis has been given to managing accidents once substantial core damage has occurred. Recent risk studies, however, indicate that substantial reductions in risk might be achieved if plant personnel can intervene with effective actions that retain the damaged core within the vessel or maintain containment integrity if vessel failure occurs.

The staff (RES and NRR) are developing an Accident Management Program Plan (AMPP) to describe the principal accident management issues (within the

regulatory/research framework), define the research tasks and scope, and delineate the priorities and major milestones to assure appropriate consideration is given to the application of accident management insights at nuclear power plants. A summary of a proposed Accident Management Program Plan is presented in the following sections. A more comprehensive description and discussion of this program plan will be provided in a forthcoming Commission paper.

6.2 Program Summary

6.2.1 Objectives

The proposed Accident Management Program Plan has the following principal objectives:

- To provide a technical basis for the review of accident management programs to be developed by utilities.

- To assure that adequate guidance (developed from such sources as previous PRA studies, the NRC Severe Accident Research Program and the nuclear industry-sponsored IDCOR program) will be distilled, disseminated to the operating/technical support staff and integrated into utility training programs.

- To increase the emphasis on accident management as a part of annual emergency preparedness exercises.

- To develop strategies for dealing with prevention and mitigation of accident sequences based in part on input from CPI and IPE programs.

6.2.2 Approach for Enhancing Accident Management Capabilities

The approach that we currently envision for enhancing utility capability to cope with potential severe accidents is to emphasize the importance of

(1) developing procedural guidelines (beyond EOPs) written at an engineering or safety function level and defining the role of the operating and technical support center staffs in their use, (2) improving accident management skills by providing further clarification regarding what is expected from licensees in the area of accident management and performing more in-depth evaluations of accident management capabilities during the annual emergency preparedness exercises, and (3) assessing plant-specific applicability of the latest technical information on severe accident phenomena and strategies resulting from NRC and industry sponsored research to determine whether additional corrective measures, such as enhancements to procedural guidelines, or design or hardware changes, are warranted.

6.2.3 Proposed Research Program

A substantial amount of research has already been done on both probabilistic risk assessment (PRA) methods and on severe accident phenomenology to systematically discover severe accident vulnerabilities, both in the U.S. as well as abroad. Although considerable progress has been made, RES has initiated a research program to examine the efficacy of generic accident management strategies and related activities to meet the objectives of the Accident Management Program Plan, as stated above. A summary of this program plan follows.

6.2.4 Prevention of Core Damage

As a result of the TMI-2 accident, the current symptom or symptom/event-oriented emergency operating procedures (EOP) have been developed to provide guidance on accidents leading to inadequate core cooling. The relationship of the EOPs to severe accident management programs must be developed to ensure compatibility. Research tasks envisioned will (1) investigate whether cost-effective measures or improvements may be available to either prevent core

melt or substantially delay it, and (2) examine ways that severe accident management strategies that are implemented early in the event can be properly encompassed in existing procedural guidance without conflict.

6.2.5 In-Vessel Accident Management

In-vessel accident management addresses preservation of the reactor coolant system pressure boundary and the measures (i.e., strategies) that potentially could be taken to accomplish this. The obvious principal strategy is to restore water supplies to the reactor vessel to prevent molten core material from melting through the reactor vessel head. For PWRs, a related strategy is the management of steam generator secondary coolant inventory. There are concerns that must be addressed before the generic application of the strategy of water addition to the reactor coolant system (and to the secondary system for PWRs) can be supported technically. These include questions about possible conditions where water addition may increase the overall severity of the accident, the capabilities necessary for effective water addition, and the competing need to supply water to the containment sprays. Resolution of these concerns will also be strongly impacted by specific plant design features or specific accident sequences. Another proposed strategy is to depressurize the primary system at an appropriate time to allow water to be supplied to the reactor vessel from alternative low pressure systems (e.g., the fire system with previous provisions made to accomplish this) if principal safety systems (i.e., ECCS) are not available. An additional benefit from depressurization related to containment performance is discussed below. The proposed research program is being directed toward defining and investigating the issues related to establishing the efficacy of these strategies.

6.2.6 Ex-Vessel Accident Management

In the course of a severe accident, primary system failure can occur, releasing substantial amounts of energy to the containment, as well as the fission product inventory of the core. During this phase of a severe accident the

containment will be challenged and its integrity must remain intact to prevent radioactive releases to the environment. Also, at this stage of a severe accident, an operator may be faced with a number of decisions that can affect continued containment integrity. The proposed research will include consideration of options for mitigating severe accidents inside the containment or reactor building by determining advantages and disadvantages for various strategies, such as venting, hydrogen control, effective use of containment and reactor building sprays, and depressurization of the reactor coolant system to reduce the impact on containment from direct containment heating. The studies will also investigate what strategies may be required to accommodate the impact of recovering safety systems or other components during a severe accident progression. The proposed research program addressing this area is being directed toward systematically determining the feasibility and interrelationship of several diverse strategies and the technical issues that must be resolved prior to incorporating them in severe accident management guidelines.

6.2.7 Related Activities

In order to develop viable severe accident management strategies, the availability of information on the status of the core, containment, safety systems and other support systems under severe accident conditions needs to be established. Following the TMI-2 accident, the environmental qualifications of important monitoring and control instrumentation that could provide this information were upgraded. However, this issue will be examined to define the type of plant information systems (including control room displays, remote displays, and "information interpretation" systems such as safety parameter display systems and computerized decision aids) that may be needed to enhance the operating crew's capabilities during accidents, and whether the required instruments and displays will continue to function as needed under the conditions that result during severe accidents.

Any study of the technical issues associated with developing an effective accident management program must be accompanied by a consideration of the human

factors and organizational issues that play a large role in the effectiveness of such a program. Some areas being considered are (1) methods of assuring the adequacy of operational and technical staff performance, training, and staffing for severe accident situations, (2) identification of unique training requirements for the operating and technical staff in coping with severe accidents, and (3) development of criteria and guidelines for assessing the contribution of licensee management and organization as related to coping with severe accidents.

6.3 Relationship to Other Programs

6.3.1 Individual Plant Examination (IPE)

Although the Individual Plant Examinations (IPEs) being carried out as part of the Commission's severe accident policy implementation do not require accident management programs, it is expected that much of the information developed during the conduct of the IPEs will be used in the development of an accident management program. As noted elsewhere in this paper, development of an accident management program by each licensee is considered an essential ingredient of the "closure" process for severe accidents. As stated in the proposed generic letter implementing the IPEs, we expect licensees to conduct their IPEs with the understanding that much of the insights and knowledge they gain will ultimately be used in a severe accident management program. We are working with NUMARC to define the scope of an accident management program and the best way to implement one at each operating plant. The Accident Management Research Program is being designed to support the staff's capability to assess the licensees' development of severe accident management programs. This relationship has been discussed further in Section 2 of this Enclosure.

6.3.2 Containment Performance Improvement (CPI)

As described in Section 3 of this Enclosure, the staff has established the CPI research effort to deal with key issues affecting generic challenges to

containments. These are typically severe accident phenomenological issues where research has not yet produced conclusive results, or where uncertainties are (or may remain) large. Some of the uncertainties related to these issues may be reduced through future research. Therefore, the application of suitable accident management strategies may effectively preclude the occurrence of some phenomena or significantly reduce the associated challenges to containment.

6.3.3 NUREG-1150 and Severe Accident Research Program (SARP)

The results, to date, from the NUREG-1150 and SARP programs and other industry-sponsored programs (e.g., IDCOR and plant-specific PRAs) provide a broad understanding of severe accidents, the phenomenology of their initiation and progression, the important issues arising from the phenomena, and the influence of factors such as operator actions and emergency operating procedures. These studies show that the potential exists for certain preparatory and recovery actions to be taken by the plant operating staff that could substantially prevent or mitigate the consequences of a severe accident. For example, various measures that could be taken by the plant operating and technical staff to either prevent or mitigate an accident were considered in NUREG-1150.

The Accident Management Program will be coordinated with the remaining NUREG-1150 and SARP efforts to ensure that potentially effective severe accident strategies identified through these programs are systematically evaluated and forwarded for consideration by licensees of affected nuclear power plants.

7. NUREG-1150: REACTOR RISK REFERENCE DOCUMENT

7.1 Background

In February, 1987, the NRC published the "Reactor Risk Reference Document," NUREG-1150, in draft form for pubic comment. This report summarized the results of, and provided perspectives on, the study of severe accident frequencies and risks for five licensed nuclear power plants.

The principal objective of NUREG-1150 was to provide a snapshot of the state-of-the-art PRA technology, incorporating improvements in methodology and data accumulated since WASH-1400. Such data resulted from plant operating experience and from the extensive research programs sponsored by NRC, the U.S. nuclear industry, and foreign organizations. As a second objective, this risk information was used to provide technical data and perspectives on certain regulatory issues. In draft NUREG-1150 itself, for example, comparisons were made of the estimated risks to NRC safety goals and to possible supplemental goals (e.g., the probability of a "large" release, as defined in NUREG-1150). In addition, the technical data developed in the course of preparing the draft report was used by the NRC staff to support the development of guidance for licensees on the performance of individual plant examinations pursuant to the Commission's Severe Accident Policy Statement.

7.2 General Perspectives

Some general perspectives from draft NUREG-1150 include the following:

- Severe accident frequencies:

  Severe accident frequencies for the two plants originally studied in the Reactor Safety Study (Surry and Peach Bottom) were estimated to

have decreased since 1975, because of extensive design, operational, and procedural changes at these plants.

- Severe accident risks:

  Risk levels were estimated as generally lower for the two plants originally examined in the Reactor Safety Study, but uncertainties in these risk estimates were substantial.

- Containment performance:

  Considerable differences in containment performance under severe accident conditions were found among the five plant design types studied; these analyses also indicated that the specific phenomena potentially leading to early containment failure also varied considerably among the plants.

  Containment performance in severe accident conditions appeared better for the PWR studied in both the Reactor Safety Study and NUREG-1150, while such performance in the BWR plant examined in both studies did not significantly change.

- Comparisons with safety goals:

  Comparisons with NRC's safety goals indicated that all five plants studied in draft NUREG-1150 met the quantitative health objectives and all but one met the "large release" guideline (as defined in NUREG-1150), for the class of events examined for the draft report (internally-initiated events).

7.3 Areas of Improvement

The severe accident frequency and risk studies described in draft NUREG-1150 provide a "snap-shot" for each of the five plants as of 1986. Pursuant to both

public comment and peer review, major areas of improvement of the five plant analyses are well under way for the preparation of the final version of NUREG-1150, expected to be published by January 1989. These areas include:

- Updating to reflect design configuration and/or procedure modifications for each plant as of early 1988.

- Substantial improvement in the expert elicitation process using decision theoretic techniques and experienced decision analysts to obtain estimates of physical quantities and frequency distributions for which sources of information are at variance with one another or not available.

- Complete reanalysis of containment loads, containment response, source terms, and consequences.

- Estimates of external risks, particularly seismic and fire, will be included for two plants, Surry and Peach Bottom.

- Significant efforts are being made to make the NUREG-1150 report more scrutable and the results more easily traced.

7.4 Independent Staff Audits

NUREG-1150 will provide an independent staff audit of severe accident frequencies and risks in five licensed nuclear power plants. This audit process will continue after publication of final NUREG-1150, examining additional plants as "samples" of the overall set of U.S. plants.

7.5 Data Base Supporting Ongoing Staff Activities

NUREG-1150 provides a systematic and consistent PRA data base for five plants of different design, with quantitiatve evaluations of uncertaintiies, supporting ongoing staff activities in the following ways:

- Individual plant examinations:

    - Detailed technical data generated for NUREG-1150 on severe accident frequencies, risks, and important uncertainties were used in developing the analysis requirements described in the proposed IPE generic letter;

    - Via audit PRAs discussed above, future NUREG-1150-type analyses will provide a reference basis for the review of IPE submittals.

Containment performance initiatives:

- NUREG-1150 provides a systematic, comprehensive identification of vulnerabilities of five major containment design types.

Safety goal implementation:

- NUREG-1150 provides a probabilistic model for five plants, acting as "test-bed" for alternative goals and implementation strategies.

Generic issues:

- NUREG-1150 probabilistic models provide a capability for issue screening and prioritization in five plants of different design.

Accident management:

- NUREG-1150 probabilistic models of five plants provide a vehicle for the examination of benefits and downside risks of alternative accident management strategies.

Figure 7.1 presents a schematic diagram showing the schedule of effort and illustrating the flow of information from NUREG-1150 to the other activities of the Severe Accident Integration Plan.

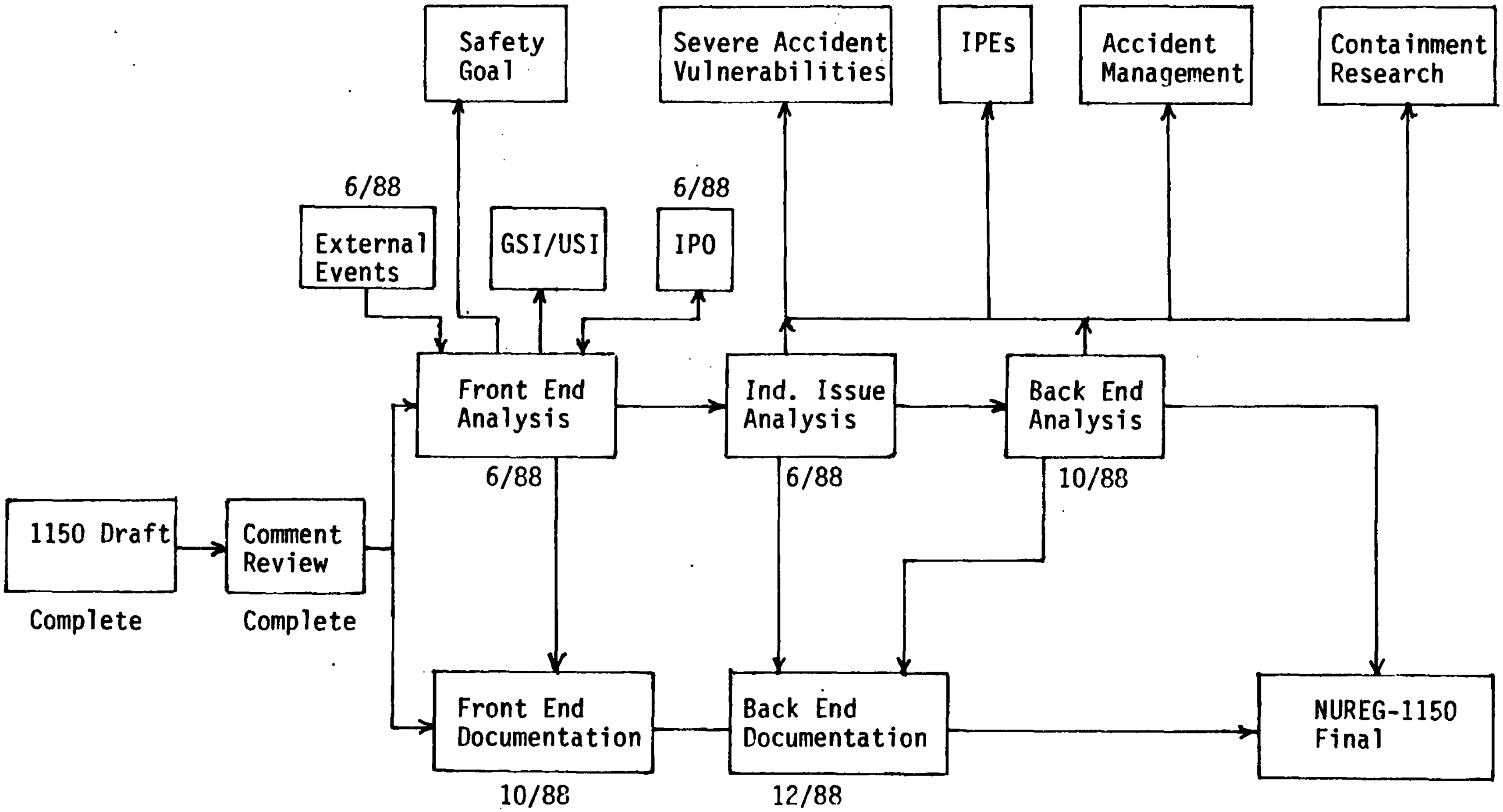


Figure 7.1 NUREG-1150 Flow Chart

8. GENERIC SAFETY ISSUES

Generic safety issues, which are possible deficiencies in the design, construction, or operation of several or a class of nuclear power plants such that the protection of the public from radiation may be inadequate or an enhancement of safety may be cost-effective, are, for the most part, generic vulnerabilities to severe accidents. This is so because generic safety issues are selected for resolution primarily on the basis of their safety significance. Safety significance is, in turn, primarily a function of the potential for a severe core damage accident. Most of the high priority issues were selected because the potential for a reduction in core damage frequency was estimated to be greater than 10^{-5} events per reactor year. The basis for this selection criterion is that by eliminating these sequences, the aggregate core damage frequency can be maintained at or near 10^{-4} per reactor year. As part of the resolution process for generic issues, sequences with frequencies that are an order of magnitude less (10^{-6}/Rx-yr) are also selected in recognition of the large uncertainty associated with the method of estimating core damage frequency as described in NUREG-0933.

In some cases, the resolution of generic safety issues may only require the enforcement of current requirements. However, the 64 issues currently being resolved by the staff are likely to result in changes in generic requirements (i.e., the rules) or generic guidance (e.g., Regulatory Guides, Standard Review Plan or Generic Letter). Since most of these current issues were identified from operating experience, most are directed toward preventing core damaging accidents. Only seven of the issues are directed toward mitigating the consequences of core damage accidents. Other containment performance improvements are being pursued, such as those described in SECY-87-297 for the MARK I containment, but are not identified or tracked as generic issues. None of these issues are directed toward mitigating the effects of potential releases of radioactivity from the containment. All generic issues are listed and described in NUREG-0933.

The imposition of any new requirements or guidance resulting from the resolution of generic issues is justified in accordance with the backfit rule (10 CFR 50.109). Resolutions without such justification can be imposed if the change is necessary to provide adequate protection of the public. However, the resolution of no generic issue identified to date is required in order to maintain the risk to the public less than the quantitative objectives for health effects adopted in the Policy Statement on Safety Goals (i.e., 0.1% of the risk from other accidents or cancer). Resolutions are usually imposed if the net cost of implementation is no more than approximately $1000 per person-rem of radiation dose likely to be averted, but other factors may be considered.

Generic safety issues are, in some cases, also identified in plant-specific PRAs as vulnerabilities that are significant contributors to the frequency of core damage or containment failure. In fact, several of the current issues were identified from the results of PRAs (e.g., GSI 23, Reactor Coolant Pump Seal Failures; GI 105, Interfacing Systems LOCA at BWRs; GI 130, Essential Service Water Pump Failures at Multiplant Sites). Thus, the systematic, probabilistic, individual plant examinations such as those proposed in the implementation of the severe accident policy, would be expected to identify some vulnerabilities that are the same as some of the current unresolved generic safety issues.

If changes to regulatory requirements (i.e., rules) or guidance (R.G., SRP) are needed, then the generic issue process must be followed. In any case, if a cost-effective generic resolution of an issue can be identified, this is usually the most efficient and effective method of imposing any changes. For example, the focusing of a relatively small amount of NRC and industry resources has resulted in a consensus for a generic method for implementing the resolution of USI A-44, Station Blackout. Other issues must be implemented as generic issues because the level of detail required to assess the issues is beyond that in the simplified method used in individual plant examinations. For most of the current issues, generic resolution would yield closure sooner

than the IPE process. Most generic issues are scheduled to be resolved within two years, and the results of individual plant examinations are not likely to be available before then.

Some vulnerabilities result from specific features of plant design or operation that are unique to a plant or a few plants and can be only identified by means of an individual plant examination. The results of numerous PRAs have shown that many vulnerabilities exist in support systems, such as the service water system, and are plant specific because of large variations among the designs of such support systems. Thus, some generic issues may not have a cost-effective generic resolution, but implementation could be cost-effective if plant specific. USI A-45, "Decay Heat Removal," is such an issue, since the proposed generic requirement (a dedicated decay heat removal system) was not cost-effective based on existing Commission backfit guidance. However, individual plant examinations should be able to identify plant-specific improvements of the decay heat removal function that would be cost-effective. Most such improvements would go beyond current requirements or guidance. While imposition by the NRC would have to be justified by plant-specific backfit analyses, it is expected that licensees will volunteer to implement cost-effective improvements that reduce risk based on their individual plant examination.

Thus, there is a close relationship between generic safety issues and individual plant examinations. Each generic issue has been reviewed to determine whether or not it can be most efficiently and effectively resolved and implemented as a generic issue or as part of the individual plant examinations. For most issues, a generic resolution was found to provide the fastest and most efficient and effective implementation.

9. EXTERNAL EVENTS

9.1 Background

On August 8, 1985, the Commission issued a policy statement (50 FR 32138) on severe accidents. The policy statement does not differentiate between events initiated within the plant and events caused by external initiators. Current risk assessments indicate that risk from external events could be a significant contributor in some instances, although a distinct possibility exists that risk from external events has been overestimated because of the conservatisms used in this evaluation.

The staff intends to proceed with the implementation of the severe accident policy outlined in SECY 86-76 using Individual Plant Examinations (IPE) for internally initiated events. The assessment methods and plan for Individual Plant Examinations for External Events (IPEEE) will require more development. Hence, the evaluation of external events will proceed on a different schedule from internal events. Delaying severe accident reviews for external events will not result in unacceptable public risk because design provisions for protecting nuclear power plants from external events are known to be conservative. The procedures and criteria outlined in industry codes and standards and specified in NRC Regulatory Guides and Standard Review Plans result in substantial inherent margin in the design. Criteria for protection from external floods, high winds/tornadoes, transportation accidents and earthquakes have traditionally been conservative, and more severe than the maximum expected event. The capability of plants to withstand external events is therefore, in most cases, substantially beyond the design basis. As an example, seismic experience data collected as part of USI A-46, "Seismic Qualification of Equipment in Operation Plants," indicates that, in general, equipment in nuclear plants has seismic capability far in excess of its design level provided that it is properly anchored. However, the staff believes that plant- and site-specific conditions and design and construction errors can decrease the inherent margin from external initiators. In addition, use of

maximum expected events has not always been on probabilistic bases. Therefore, it is appropriate that an IPEEE be conducted to identify and correct plant-specific vulnerabilities which could reduce the inherent margin.

The staff, as stated in SECY-86-162, is proceeding with the evaluation of severe accidents initiated by external hazards in two phases. The first phase, which is essentially completed, consisted of a Lawrence Livermore National Laboratory (LLNL) study (a) to assess the margins that past design bases provide relative to external events that are beyond the design bases, and (b) to identify areas where an examination for external vulnerabilities may be needed. During this period, a trial application of a simplified seismic review procedure, developed by NRC's Seismic Design Margin Program was tried on a PWR plant and found to be very useful and could be used for IPEEE with minor modifications. The second phase of the external events evaluation program will consist of developing specific guidance and criteria for each external hazard to be considered in the IPEEEs.

Based on the results of the LLNL study and information revealed during an external events workshop held August 4-5, 1987, some external events, such as earthquakes and fires, may be significant risk contributors at some plants. Also, the design bases for some external events, such as winds and external floods, may be sufficiently conservative that they do not pose a significant risk at most plant sites. However, there may be some structures or facilities at some sites which were not designed to current criteria and may pose some risk at those plants. These studies reaffirm the need to include some external events in accident reviews, and provide insights to assist in developing review guidance for the IPEEEs. A new program has been initiated at LLNL to develop IPEEE guidance, procedures, and criteria. More specific guidance will be developed from these and other ongoing studies to help utilities to conduct their IPEEEs. This program includes evaluation of industry programs as well as ongoing NRC programs.

There are many on-going NRC programs addressing the potential threat of external events to the safety of nuclear power plants. Just to name a few: the Seismic Hazard Characterization Project to determine seismic hazard at eastern U.S. sites; the Seismic Design Margins Program developed a method which can be used to address the seismic concerns of eastern U.S. sites; A-46 to address the seismic qualification of equipment at older plants; the Fire Risk Scoping Study to address the adequacy of fire protection; and the NAS study to address the extreme flood frequencies. It became evident that an integrated approach, to avoid any unnecessary duplication of effort, is absolutely essential for NRC to address these external events. On December 21, 1987, NRC established an External Events Steering Group (EESG) to make recommendations to senior staff management regarding the degree to which external events need to be considered in the context of the severe accident policy implementation, and the scope and methods for such an evaluation.

9.2 External Event Phase 1 Program

Under the auspices of the NRC, Lawrence Livermore National Laboratory (LLNL) made a study on the risk of core damage and large release at nuclear power plants due to externally-initiated events. These events included internal fire, high winds/tornadoes, external floods, transportation and seismic events.

Two figures-of-merit were used as evaluation criteria to discriminate between the significant and the less significant levels of risk. These two figures-of-merit are defined as 10^{-5} per reactor year for a core damage accident and 10^{-6} per reactor year for a large release of radioactive materials to the environment. These figures are compatible with the Commission's safety goal policy.

The results of LLNL's study showed that those external initiators (seismic, fires, floods and high winds) are important (exceeded the figure-of-merit) with respect to core damage frequency. It also showed that the design bases for some external events, such as winds and external floods, may be sufficiently

conservative that they do not pose a significant risk at most plant sites. However, there may be some structures or facilities at some sites which were not designed to current criteria and may pose some risk at those plants. The seismic initiator was found to have frequencies of large releases exceeding the figure-of-merit while fires were found to have frequencies of large releases less than the figure-of-merit. Fires and seismic events are of importance to all plants, while the other external events are site specific. A full-scope PRA, a screening type PRA, or an abbreviated method can be used for external vulnerability analysis.

These results reaffirm that external events should be included in the vulnerability analysis of nuclear power plants. However, the scope of the examination may differ depending on the site locations. A close examination of the vulnerable areas identified in the LLNL's study and those considered in the actual plant design should be made to ensure adequate protection against severe accidents.

9.3 On-Going NRC Programs and Related USIs/GIs

There are a number of ongoing NRC programs that either plan to examine, have examined, or have some potential to assess the risk from external events. They are as follows:

- USI A-46, "Seismic Qualification of Equipment in Operating Plants," is developing an alternative method and acceptance criteria to verify the seismic adequacy of operating plants equipment with construction permits before about 1972 against the current safe shutdown earthquake (SSE).

- USI A-45, "Shutdown Decay Heat Removal Requirement," has examined the ability of the decay heat removal systems of six operating plants (Quad Cities, Cooper, Point Beach, St. Lucie, ANO-1 and Turkey Point) to perform its function during and after seismic events up to and beyond the SSE, with internal flooding, internal fires and high winds.

- USI A-17, "Systems Interactions in Nuclear Power Plants," addresses ACRS concerns regarding the interaction of various systems with regard to whether actions or consequences could adversely affect the redundancy and independence of safety systems.

- USI A-40, "Seismic Design Criteria, A Short-Term Program," investigated selected areas of the seismic design process, and is proposing alternate approaches to part of the design sequences, as well as modifying the NRC criteria in the Standard Review Plan to reflect the current state of technology and industry practice.

- "Seismic Hazard Characterization of the Eastern United States Project" has developed a method to assess the seismic hazard for the region east of the Rocky Mountains.

- "Seismic Design Margin Program" has developed a simplified method to estimate available seismic margins in operating plants at a predetermined seismic level above the current SSE.

- NUREG-1150, "Reactor Risk Reference Document," is performing external event analyses on Surry and Peach Bottom. Earthquake, fires and other plant specific external events are to be included in the analyses.

- "Fire Risk Scoping Study" was performed to: 1) review and requantify certain past fire risk scenarios in light of updated data bases and updated computer fire modeling capabilities, 2) identify potentially significant fire risk issues which have not previously been addressed and to quantify the potential impact of those identified fire risk issues, and 3) review current fire regulations and plant implementation practices for relevance to the identified unaddressed fire risk issues.

- "Robust Techniques for Estimating the Probabilities of Extreme Floods" identified and reviewed various approaches to estimate extreme flood probabilities.

Similar to NRC's ongoing programs, the nuclear industry has a number of programs that have, or will have examined risk associated with external events. EPRI is sponsoring a seismic margins program and an eastern U.S. seismic hazard characterization program complementary to the staff programs.

9.4 External Event Phase II Program

Based on the results of the External Event Phase 1 Program, it appears that some selected external events need to be considered in the IPEEE. The scope and the approach of the examination for each individual event is expected to differ.

More detailed guidance and criteria needs to be developed during Phase II for use in the IPEEE. This guidance should enable the licensees and/or the NRC as appropriate to:

1. Determine which external hazards need to be included in their IPEEE programs based on their specific site and plant conditions.

2. Assess systematically plant-specific vulnerabilities to severe accidents initiated by those external hazards.

3. Integrate NRC's ongoing safety programs which are related to external events.

4. Establish criteria to resolve external event issues.

5. Identify areas for industry/NRC cooperation.

A new program is being conducted to develop IPEEE guidance, procedures and acceptance criteria for operating plants. Since the industry has sponsored numerous safety programs, we will continue to invite industry participation in the phase 2 program to identify needs, scope and methods for the IPEEE.

10. INTEGRATED SAFETY ASSESSMENT PROGRAM

The original Integrated Safety Assessment Program (ISAP) was a pilot program initiated in May 1985 for Northeast Utilities' Millstone Unit 1 and Haddam Neck facilities. It was designed to provide a comprehensive review program for operating reactors that would address all safety issues and allow for an integrated, cost-effective implementation plan using deterministic and probabilistic techniques.

ISAP II is a systematic program to address regulatory issues according to an integrated schedule. The program is defined in detail in a SECY paper expected to reach the Commission in June 1988. ISAP II includes an integrated assessment based on a probabilistic risk assessment (PRA) and, as necessary, an operating experience review. Issues would be ranked, and an integrated schedule would be developed from this ranking. The integrated schedule would allow both regulatory issues and utility-initiated items to be included in the schedule. As in the pilot program, identified benefits of ISAP II include (1) finding common elements in separate review areas and proposing a single integrated action to resolve the concerns; (2) addressing regulatory and utility items on a plant-specific basis; and (3) dropping issues from further consideration because of low safety significance. Unlike the pilot program, however, ISAP II participants would not be expected to address unresolved generic issues before the staff reaches a generic resolution. ISAP II would include only current and future items on a plant's regulatory agenda, i.e., those issues that would be required of nonparticipating utilities.

Current issues include those required by regulation, order, license condition, bulletin, or generic letter; items committed to by the utility; and the utility's own initiatives. If an issue resolution was required by NRC, ISAP II could serve as a basis for rescheudling or dropping that issue but ISAP II would not obviate the need for a required exemption or other regulatory filing.

ISAP II for a given plant requires a minimum of a Level 1 PRA. A Level 1 PRA augmented with a containment vulnerability assessment is also an acceptable method for a utility to perform the IPE. Thus the analysis done to participate in ISAP II is directly applicable to the work necessary to perform an IPE.

The staff proposes to implement the IPE and ISAP II as separate but compatible programs. A utility could initiate ISAP II either before or after completion of its IPE. Criteria will be established within the IPE program framework for determining which of the identified risk reduction actions will be implemented.

If a utility elects to use a PRA for an ISAP II review after the same PRA is used as the basis for an IPE review, the staff's review of the ISAP II submittal would not be expected to identify any significant risk reduction actions that were not considered by the NRC in the IPE review and approval process. If, for any reason, additional potentially significant risk reduction modifications are identified in the ISAP II review, those proposed modifications will be considered with respect to the same criteria used for the IPE determinations, i.e., the need to provide reasonable assurance of safety as determined at plant licensing, and the backfit rule 10 CFR 50.109. If the ISAP II review precedes the IPE, proposed risk reduction actions will be identified, but a final determination on the absolute risk importance of the proposed actions will be made at the conclusion of the IPE program for that plant. All approved actions resulting from the IPE then will be evaluated, prioritized, and scheduled within the ISAP II process.

If a utility elects to participate in the ISAP II program after having submitted an IDCOR IPEM evaluation for the IPE program, the subsequent ISAP II results will not be expected to reveal additional potentially significant risk reduction actions, but in that event, the additional actions will be evaluated to determine whether backfit action is warranted.

Currently, there is a relatively low level of interest expressed by the licensees in ISAP II; however, the staff will arrange to combine IPE and ISAP reviews when requested consistent with available resources.

11. ADVANCED REACTORS

The staff is currently reviewing, or planning to review, a number of advanced reactor designs. These designs include the evolutionary LWRs (W SP/90, GE ABWR, and CE System 80+), the passive advanced LWRs (SBWR, and AP-600), and the three DOE-sponsored, non-LWR advanced reactor conceptual designs (MHTGR, PRISM and SAFR). The staff is also reviewing the EPRI ALWR utility requirements document. In parallel with the closure of severe accident issues for operating reactors, the staff is proposing to develop procedural and possibly performance regulations for future reactors, with supporting regulatory guides and other documents to clarify the procedures and requirements for addressing severe accidents. This activity is considered to be consistent with the intent of the Severe Accident Policy Statement and is intended to support the design certification rulemaking (10 CFR 52). The staff's plans for the treatment of severe accidents for both LWR and non-LWR advanced reactors are described below.

11.1 Advanced Light Water Reactors (LWRs)

The staff has increased consideration of severe accidents at LWRs since the TMI accident. A severe accident policy was issued and new requirements (10 CFR 50.34(f)) were established. The severe accident policy stated that a new design can be shown to be acceptable for severe accident concerns if it meets the following criteria and procedural requirements:

a. Demonstration of compliance with the procedural requirements and criteria of the current Commission regulations, including the TMI-related requirements;

b. Demonstration of technical resolution of all applicable USIs and GSIs, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems;

c. Completion of a PRA and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety; and

d. Completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by PRA.

The staff is considering initiation of rulemaking to modify 10 CFR 50.34(f), since it currently applies only to cancelled plants. The modification of 10 CFR 50.34(f) would adopt the above stated procedural requirements and make them applicable to advanced reactors. Requirements that specify an acceptable level of protection against severe accidents are also being considered. These requirements, if adopted, would require a higher level of safety for new plants thereby implementing the following from the Commission's Severe Accident Policy Statement:

> "The Commission fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs."

Supporting Regulatory Guide(s) would also be developed to expand on the rule(s) to provide a more detailed definition of the procedural steps, such as the scope and content of the required PRA documentation, and provide guidance on what severe accidents or phenomena should be considered. The staff believes that severe accident rulemaking for advanced reactors will facilitate design certification rulemaking, will ensure a uniform treatment of severe accident issues, and provide useful guidance to future reactor designers. Accordingly, it is our intent to put the advanced reactor severe accident requirements in place prior to initiation of design certification rulemaking for the LWRs currently under review.

During the course of the development of the proposed rule(s) and Regulatory Guide(s), the staff will be monitoring the closure of severe accident issues for operating reactors and will be holding periodic public meetings in order to collect information for the rulemaking proceeding. These meetings will reduce the potential for impacting the LWR designs currently under review by keeping the public and industry informed on the status of the severe accident rulemaking. The first public meeting is scheduled for June 9, 1988, in Rockville, Maryland.

11.2 Advanced Non-Light Water Reactors

The treatment of severe accidents for advanced non-LWR reactors is being assessed as part of the reviews being conducted in accordance with the Advanced Reactor Policy Statement and NUREG-1226, "Development and Utilization of the NRC Policy Statements on the Regulation of Advanced Nuclear Power Plants." Currently, these activities involve the review of three DOE-sponsored advanced reactor conceptual designs and are intended to provide preliminary licensing guidance to the designers of these concepts in pre-application SERs. It is anticipated that in the review of an actual application for one of these designs, the staff would utilize and build upon the review done at the conceptual design stage.

In the current review of the DOE reactors, the guidance being developed by the staff for these designs is built upon and is to be consistent with existing Commission guidance as stated in the advanced reactor, severe accident, safety goal and standardization policies. In particular, the guidance contained in the safety goal and severe accident policies has been used to help define the range of events which need to be considered in these designs and the role of a PRA. Additionally, we are utilizing a licensing approach on advanced reactors which stresses deterministic engineering analysis and judgment, complemented by PRA. Specific staff proposals with respect to the key licensing issues associated with these advanced designs (treatment of severe accidents, siting source term, containment, emergency planning) are to be provided to the Commission in a SECY paper. It is anticipated that much of the guidance to be developed in the rulemaking for advanced LWRs will also be applicable to the DOE reactors.