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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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BRIEFING ON SEVERE ACCIDENT RESEARCH PLAN

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PUBLIC MEETING

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Nuclear Regulatory Commission
One White Flint North
Rockville, Maryland

Tuesday, May 2, 1989

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

COMMISSIONERS PRESENT:

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

STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

SAMUEL J. CHILK, Secretary

STUART TREBY, General Counsel's Office

VICTOR STELLO, JR., Executive Director for Operations

ERIC BECKJORD, Director, Office of Research

BRIAN SHERON, RES

FRANK COSTANZI, RES

DENNY ROSS, RES

P R O C E E D I N G S

(10:04 a.m.)

CHAIRMAN ZECH: Good morning, ladies and gentlemen.

Commissioner Carr will not be with us today.

This is an information briefing in which the Office of Research will describe its revised severe accident research program plan. Details of the revised plan are described in a recent NRC staff paper, SECY 89-123.

The severe accident research program provides confirmatory information and technical support in implementing the NRC integration plan for closure of severe accident issues. That plan was described in an earlier staff paper, SECY 88-147, and has been the subject of several Commission meetings.

Copies of the slide presentation and the staff's paper, SECY 89-123, should be available at the entrance to the meeting room.

Do any of my fellow Commissioners have any opening comments before we begin?

(No response.)

If not, Mr. Stello, you may proceed.

MR. STELLO: Thank you, Mr. Chairman. I wanted to take a moment to review some of the background of the

research activity. As you are aware, a great deal of the work that comes out of the agency starts with research. Some it spanned many years and involved the expenditures of very large sums of money, such as emergency core cooling that spanned probably the better part of 20 years. We finally resolved that issue and at very large expenditure of funds. Other issues which have gone on for quite some time and are very, very significant in terms of dealing with safety issues such as PTS.

When we provided you with our integration plan for closure of severe accidents, it was apparent then that this was pretty much dictating, in a general sense, the strategy of where the severe accident research ought to go and, at that time, the Commission, at that meeting and in documents since that time, has indicated the need to have a status and an update and a briefing of the Commission of the general thrust of where we ought to go.

Eric and his staff have done what I think is a very commendable job in trying to lay out the framework for that program; have gone out and sought comment and review by various organizations, trying to make sure that what we are now doing is, in fact, viewed by those who normally review it, such as the ACRS and our peer review group for research, indicates to us at least our belief that we are, in fact, on the right track.

And what we want to do today is to brief the Commission on where we are with respect to the program, and we'll brief you in terms of where we intend to go both in the long-term as well as the short-term, filling the more immediate needs that we have for the IPE, CPI, the containment performance and the plant performance, as well as our accident management safety goal. And with that brief introduction, let me ask Eric to have some opening remarks, and we'll get into the meat of the briefing.

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

MR. BECKJORD: Thank you, Mr. Chairman.

On April 13th, I reported to you on carrying out the integration plan for the closure of severe accident issues document that you've already referred to.

The revised severe accident research program plan is an important element of the closure plan, and today we are presenting it to you in detail. Specifically, the severe accident research program provides a technical base for support of the other elements of the closure plan.

The severe accident research program has four goals. First, to provide new knowledge for assessing containment performance over the range of risk-significant core melt events; secondly, to evaluate the effectiveness

of generic containment performance improvements; third, to support the development of generic accident management procedures; and, fourth, to assess fission product behavior and release in severe accidents in the event of containment failure.

It focuses on the resolution of severe accident issues having high risk significance in the near-term, especially the containment performance questions.

The long-term part of the program aims at reducing the uncertainties in the knowledge of severe accident phenomena. This will help to reduce the uncertainties in future probablistic risk assessments.

The revised severe accident research program was developed with input from experts in the university, the national laboratory, and the industrial community. The performers of the research, university people and the NRC program offices, have reviewed the plan. The Advisory Committee on Reactor Safeguards has strongly supported the plan. Also, our Nuclear Safety Research Review Group has reviewed the plan, under Professor Todreas. I strongly recommend to you this plan.

CHAIRMAN ZECH: Thank you very much.

DR. SHERON: Could I have the first slide, please.

CHAIRMAN ZECH: You may proceed. Thank you.

DR. SHERON: (Slide) Just to repeat, the objectives of the research program are -- just a different form than what Eric just said -- is to develop a technical base of severe accident information, which could be used by the Commission and the staff, in any and all actions that are taken related to severe accidents; also, to support by developing a technical base, the various actions already underway -- the IPE, the containment performance program, or accident management.

And another one, the last one here, is to basically help develop an understanding of plant response to severe accidents, to make sure there are no unexpected phenomena that would change our perception of risk.

Could I have the next slide. (Slide) In setting out to develop this revised plan, keep in mind that severe accident research has been underway in this agency about ten years.

Our underlying motive was to make sure that the research was consistent with the integration plan, 88-147. We also wanted to make sure this was not just something that was dreamed up by the staff, but had the support of the technical community. And what we did is, we tried to draw upon a broad spectrum of expertise across the nation.

We called on experts from the national labs, industry, universities, and the experts within the staff,

to help identify how the current research activities in severe accidents were related to SECY 88-147.

We broke these experts into three groups, basically functional, and asked them to look at the ongoing research and how it related, and to provide the results to a senior oversight group. This was made up of staff management -- Dr. Costanzi headed it -- as well as senior managers from the laboratories and university community.

This oversight group helped assimilate all this information from these three groups, and helped focus it in the proper direction. The staff then took the results from this oversight group and went to work in developing a plan. So, this was the basic approach to developing this.

Could I have the next slide, please. (Slide) We also used NUREG 1150 as a guidance document, in defining and ranking severe accident issues. As you know, 1150 pulled together a lot of experts in terms of looking at what the uncertainties were and what the importance for various phenomena in determining risk, and we made sure that we used that information to the fullest.

Once we had developed a plan, we circulated it widely for comment, to the national labs, again; the universities; EPRI; Dr. Todreas' committee, the Research Review Committee; and the ACRS.

We were encouraged because almost all the comments we got back were virtually all favorable on the plan, that it was in the right direction, had the right elements to it, and the ACRS was particularly encouraging.

COMMISSIONER ROBERTS: That's a major accomplishment.

DR. SHERON: Could I have the next slide, please. (Slide) This you've seen before, is the schematic of the severe accident program. It's just here to remind you where the research program fits in in the overall closure process. You'll see it down in the lower, right corner. You'll see that it's principally designed to contribute to the containment performance improvements, as Eric said. It does feed into both the IPE process as well as accident management, and you'll notice that the arrow continues, which means there will be a longer-term confirmatory research program even beyond closure.

CHAIRMAN ZECH: Before you go off that one, it seems to me that's very important. Could you be a little bit more specific on how the research that -- described in that schematic, could impact on the IPE program -- in other words, in identifying vulnerabilities, or any accident management, or containment improvements that you might be considering? How does that really affect the IPE program?

DR. SHERON: It affects it in a number of ways. I guess the most striking example I could give you is that the Electric Power Research Institute approached me about four or five months ago, and asked if they might conduct an independent review of a severe accident computer code, BWRSAR, that was developed by Oak Ridge.

We gave them our blessing to go forth and do that, and my understanding is that they were trying to put together analysis capability for the utilities so that they could do their IPE analysis.

We've also had requests for other of our analysis codes which, again, are incorporating the latest technology from research, so that they could use these for conducting the IPE.

So, the principal impact that we are having on the IPE is that the technology we are developing which, in turn, gets embodied in our analysis computer codes, is being picked up and used directly by the utilities in conducting the IPE, the back-end analysis.

CHAIRMAN ZECH: All right. Thank you very much.

DR. SHERON: The next slide, please. (Slide) The program is divided into a near-term and a longer-term effort, and I'll get into that in a little bit here.

Could I have the next slide. (Slide) The major areas that we're emphasizing in the near-term are those

that are related to the early containment failure. Basically, these are the scenarios that have the greatest risk significance. Since, obviously, anything that fails the containment early, you have substantial release before any settling or decay can occur, of the radioactive fission products.

The issues that we're focusing on, I'm sure you're aware of, are the direct containment heating issue for the PWRs, and the MARK I liner melt-through issue for the MARK I BWRs.

We're also looking at an area that has been sort of smoldering, I would say, in the back of people's minds, and that is that if a severe accident were occurring at a plant and if, for example, it was caused by a lack of ability to add water to the primary system, and you restored that capability, say, by restoring electric power, an operator would naturally opt to put as much water as they could on the core, as fast as they could.

Depending upon the location of the core and, during the course of the accident, whether it's in the lower cavity or still in the vessel, their putting water on a very, very hot molten substance can produce such things as steam explosions -- very, very large steam generation -- and there's always been a question as to what are the effects of this, and one of the things we

want to do is to help quantify those effects, not so much from the standpoint of saying, gee, we would never tell an operator to do that, but if an operator were to put water on a molten core, the operator should be prepared to understand what the next sequence of events would be in that plant.

If they saw a lot of steam generation, a lot of hydrogen generation, they should understand why that is occurring -- okay -- and if there is anything, any precautionary measures they should take in advance. So, this is something we're going to be studying. It's more or less in the area of accident management, as well.

There are two other areas that we're going to be focusing on, and these are related to how we carry out the research program. This is the issue of scaling, or similitude, and the issue here, which Dr. Costanzi will talk about a little bit more, basically, is that, as you know from when we studied ECCS, one of the questions we always had was scale, and we started out with small scale facilities and worked our way up to a large scale, like the LOFT facility and then almost a full scale with the UPTF facility in Germany.

In severe accidents, which is a much more complex type of an accident to understand, most of our experiments, with the exception of very few, are small

scale and, therefore, in order to extrapolate up to the large plant, we have to be very certain that we understand the scaling of that. And, so, we're going to be taking a very systematic and deliberate approach in looking at that, to make sure that we are able to scale or, more importantly, understand when we are not able to scale.

The last item we're going to be looking at also is accuracy criteria for our computer codes. When we start developing these codes, it's almost from scratch. We are trying to understand the basic phenomena, and I think we've been very successful in that.

The computer codes, right now, are starting to mature. They're certainly not at a stage where we would want to say we're -- say where we are now with the ECCS codes, but I think we've reached a point where we want to take -- when we now develop our codes further, we want to understand a little better what the uncertainties are, what the benefits are of further improvements and, really, a question of how accurate do these codes have to be in order to reach the conclusions that the Commission needs to reach. So, we're going to be taking a hard look at the further development of codes, with regard to how much more accurate that they need to be.

Next slide, please. (Slide) This is just a little more detail of these major areas. As I said, in

addition to restructuring the programs into a near- and a longer-term, we're revising the approach, and we've talked to you about the scaling issue.

The third bullet there is the -- talks about the increased emphasis on understanding late-phase core melt behavior. The objectives here are to understand much better, the expected amount, composition and temperature of the corium, the molten core, that would be available for release from the lower head of a light water reactor,

For both the direct containment heating and the MARK I issue, the loads on the containment are very much impacted by how the core is released from the vessel -- if it's released very fast, if it's released in a slow stream, what the driving pressure is, what the super heat of that material is, and what it's composition is. Right now that's a very big uncertainty in the analysis, and we are looking to develop experiments that will help better quantify what that is, so we can reduce those on the uncertainties.

COMMISSIONER ROBERTS: Have you seen anything from the Three Mile Island incident that would help in this?

DR. SHERON: We have a program underway right now, as you're aware, to take samples of the lower head, which we feel is a major element of this part of the

program, which is the next bullet, which is understanding the modes of lower head failure.

Right now, the analysis that we would do would indicate that if that amount of molten material were on the lower head, it would have failed, yet TMI did not fail. And, so, one of the things we want to try and understand from this examination of the lower head, is why it didn't fail.

As you know, there are competing effects. There's heat transfer from the molten material to the lower head, as well up, through, across, and into an overlying pool of water, and relative heat transfer rates and the like determine the temperature of the lower head and how it would structurally fail.

CHAIRMAN ZECH: But the investigation of Three Mile Island, it seems to me, really, you should certainly incorporate that in your research work, and try very hard to learn what we can from that --

DR. SHERON: Yes.

CHAIRMAN ZECH: -- and integrate it into the research that you're doing, as far as the lower head is concerned. I presume you intend to do that?

MR. BECKJORD: Yes.

DR. SHERON: Yes, that is a major element of the lower head examination program.

CHAIRMAN ZECH: Okay. Thank you.

DR. SHERON: And the last bullet is -- (slide) -- our anticipation is that if we are successful in carrying out the above research, that we will be able to reduce the current phenomenological uncertainties that are associated with early containment failure.

Next slide, please. (Slide) I talked a little bit -- I mentioned the approach we were going to take on further development of the computer codes that predict severe accident response.

What we're going to be doing is putting these on a more structured schedule for development. This would be consistent with the current approach we've taken for the thermal hydraulic codes. And what this is is that we want to make sure that the codes are all documented thoroughly, at the time that they are released for general use.

Right now, typically, the -- in many cases, the development precedes the documentation by several months to maybe a year or two, and this can be difficult when one is trying to understand exactly what is in the code-- what the equations are, what is being solved, what needs improvement, what doesn't.

Likewise, the need for the improvements in the code will proceed on a more structured basis, namely, that before improvements are just made, we will be looking for

justification from the code developers as to why these are needed. In other words, what is their importance? What is this improvement going to buy us in terms of increased accuracy, better understanding of the performance, and the importance to risk.

The fourth bullet, in the development of the research program to-date, there's been a development of four core melt progression codes -- the MELPROG-TRAC code, the RELAP-SCDAP code, the BWRSAR code, and the MELCOR code. We think it's appropriate at this time now that we assess the need to continue development of four codes, that there may be some duplication of effort.

So, we'll be doing an assessment, probably starting in FY90, to determine which codes we should further develop and which ones we can terminate development on.

And as I said before, the last bullet here is that we're going to attempt to identify the code accuracy criteria for that development. The objective here would be to improve our ability to clearly determine when a code is sufficient, and that development can be stopped.

On the next slide -- (slide) -- I'm going to ask Frank Costanzi now, to talk to you in a little more detail about the near-term and longer-term goals.

CHAIRMAN ZECH: Thank you very much. You may

proceed.

DR. COSTANZI: Mr. Chairman, Commissioners.

As we mentioned -- Dr. Sheron mentioned earlier, the revised severe accident research program has both a near-term portion and a long-term. I'm going to speak first to the goals which we are going to attempt to achieve in the near-term, near-term being the next three years.

Specifically, the near-term goals are two: One, to provide technological base for assessing containment performance over the range of risk-significant core melt events; and, secondly, develop the capability to evaluate the efficacy of generic containment performance improvements.

In reaching these goals, we're going to be focused on five issues: First, being an issue of scaling, or similitude; direct containment heating; MARK I liner melt through; the effect of adding water to a degraded core; and the accuracy criteria for codes and code development, as Dr. Sheron mentioned earlier.

Next slide, please. (Slide)

CHAIRMAN ZECH: The next slide.

DR. COSTANZI: Yes, next one after that.

Next slide, please. (Slide) The first issue is the question of similitude and scaling. What we're trying

to address here is the question of how to extrapolate from small scale experiments to full scale nuclear power plant accidents.

As Dr. Sheron mentioned, the research that we're doing, the bulk of the experiments are lab scale experiments. We're also using simulant materials, since trying to conduct an experiment with actual core materials is very difficult as well as very expensive.

So, the questions we're trying to investigate here are with regard to this, to make sure that we can extrapolate with confidence from these laboratory scale experiments to the reactor accidents. So, we ask the questions, are the correct phenomena being investigated? Are distortions of scale important to the process under investigation? For example, on distortion of scale, are ship models of the towing tank, the scale model of the ship is constructed and dragged through the towing tank, to look at wave propagation and the like, but things like viscosity of the water and the wave velocity in the water are not things which are directly scalable to a full-size ship, so there's a distortion there that needs to be accounted for.

And that leads to the third question -- are distortions characterized, understood and accounted for when using these large codes, which are ultimately the

embodiment of our understanding of severe accidents.

To address this, we've designed a program to develop a generalized approach to scaling severe accident phenomena.

Next slide, please. (Slide) The second issue we're addressing is depressurization and direct containment heating. We're tracking this problem from a two-prong approach. First, we're looking at the question of depressurization as a mechanism to avoid DCH. The theory is that if you depressurize the primary system early enough in the accident when the vessel fails, there will not be any energy to forcibly eject the debris from the vessel, and you will not fail the containment.

The second part of our research here focuses on whether or not there may be down-sides of depressurization, and to weigh those against what the threat of direct containment heating might be.

Next slide, please. (Slide) In looking at depressurization as a means to avoid direct containment heating, questions that we're asking is, what is the likelihood that the reactor coolant system will fail by natural circulation prior to significant core damage? You know, high pressure sequence as the effluent boils off the core, the hot gases could sort of circulate through the primary system, could fail either a surge line or a steam

tube, which would lead to depressurization.

Next question we're asking, is there a low-pressure cut-off below which there is no DCH threat? That is to say if, by whatever mechanism, the reactor coolant system depressurized to a point where the amount of material ejected into the containment upon vessel failure would not be sufficient to fail the containment.

And the third related question is, if, indeed, a low-pressure cut-off exists, is this pressure going to be reached either by natural circulation-induced failure of the RCS, or through operator action, or both?

Next slide, please. (Slide) In looking at the effects of depressurization, we're asking the question, if operator action is needed to depressurize to avoid DCH, is the operator going to have sufficient time to take that action? And what are the hardware procedural requirements needed to make intentional depressurization successful?

Secondly, are there adverse consequences to depressurization? Is something untoward going to happen if the operator proceeds to do this? Or, if natural circulation induces failure of the RCS, what is the risk significance of that?

A third question is, what is the nature of the DCH threat to begin with, and what mechanisms may exist ex-vessel to mitigate the threat of DCH?

Next slide, please. (Slide) The third issue which we're going to address is BWR MARK I containment shell melt-through. Specific questions which we're going to address in our research is, what are the relationship of the BWR bottom head failure mode to variations in the quantity, composition, temperature, and timing of arrival of the melt on the bottom head?

Perhaps one of the major uncertainties we have in understanding severe accidents are the details of the core melt, and that seems, to our present understanding, to be very significant to understanding what happens both in terms of the way the vessel fails and, given the vessel failure, of the way the melt comes down from the vessel and what happens with the melt interacting with the containment.

What we're asking here is, well, how sensitive is the bottom head failure to the details of core melt progression? And this may be particularly important in BWRs since there is a tremendous amount of thermal mechanical inertia in the bottom head of a BWR, which may make it less sensitive to the details of core melt progression. That's one of the questions which we're going to investigate.

What is the effect of water on the drywell floor when the melt pours out from the pressure vessel?

Obviously, water is a coolant, and it's going to have some effect on, and perhaps even cooling, the debris as it comes out. Certainly, water will tend to retain any fission products that are being involved from the interaction of the debris with the concrete. The question is, how much can we quantify that?

Third, how does the answer to the above question depend on the initial conditions? Again, this is the melt ejection rate, the composition, the temperature of the melt coming down.

And the final question under BWR MARK I is, under what conditions would the crust that forms the initial contact between the melt and the shell freeze? How long would it be stable? What conditions would make it stable, hence, going from an early containment failure to perhaps a recovered accident or late containment failure, which would certainly have far fewer consequences.

Next slide, please. (Slide) The fourth issue which we're addressing is adding water to a degraded core. I want to say beginning here that we're assuming in this research, that if the operator has the opportunity in the course of an accident, to add water, the operator is going to add water. That's our initial assumption.

What we're not trying to do in this part of the

program, is address the question of when should the operator add water, or how much water, at what rate. What we're trying to do here is simply understand that if the operator gets the opportunity to add water and does so, what is the operator likely to see? How is the plant likely to respond? Basic questions there -- how much energy is going to be transformed into hydrogen, or how much into steam?

Also, there is the question with BWR in-vessel, if the operator adds water, the potential for criticality, and what are the consequences of it?

The last issue -- next slide, please -- (slide) -- goes back to the question of codes. As Dr. Sheron mentioned earlier, we're focusing very heavily on our code program right now because that is the embodiment of our knowledge. When you understand something about severe accidents, we put it in the code, so we can manipulate, we can do calculations, and look at consequences.

The questions we're asking here is, how well do the mechanistic models reflect the phenomena believed to be important in severe accidents? This is kind of a question which you always ask yourself, am I modeling the right phenomena? Am I doing the right physics? How well does the interactive program of code advancement and experimentation achieve the objective applied in (1)

above? We mean here the general way we proceed in developing a code is, you take what physics you think is important in the accident, you put it into a code. You use the code to help you design an experiment to test that. You run the experiment. You go back and it helps you improve the code, and it's iterative process continues.

Now, the second question, you have to make sure that you always are going back to the accident, that you're not just going off on a tangent, that the phenomena and the experiments that you do all relate to the accident you're worried about. That's very closely coupled with the question of similitude, the first issue that we're examining.

Third, is the level of detail in the codes appropriate to their use? We want to make sure that we're not going off and developing very highly detailed codes to answer questions which we don't really -- where the significance or the sensitivity to impacts is not as great as the detail. We want to make sure that we're not putting more money into trying to understand phenomena in minute detail, which we really don't need to understand because it doesn't affect risk that much.

Which stages of an accident need to be modeled by detailed mechanistic codes, and which need coupling to

adjacent stages? This is a two-part question. One, as I mentioned earlier, we have great uncertainty about the details of core melt progression. That's still our major uncertainty.

The question is, do we need to resolve that uncertainty in the kind of detail that we seem to think right now. We're going to examine that so that we don't spend a lot of resources in understanding the fine details of core melt progression, but find out that the consequence of the accident is really not as sensitive to the details as we might have originally thought.

And the question of coupling, is there feedback in the course of an accident, between one stage to another, that requires that you couple modules together so that when you're doing a calculation, that you have this feedback process going on so you don't over-estimate or under-estimate by a significant amount because you forgot this coupling when you do calculations.

Next slide, please. (Slide) Is the level of precision needed for regulatory use being considered? This relates to the question of uncertainties, but also relates to the question here of, what are we going to do with this code?

Right now, we have a program of what we call a two-tier code program where we have very detailed

mechanistic codes which help us do the understanding of experiments and understanding of the details of sequences in an accident, but when we do analyze plants or, presumably, we will begin analyzing accident management strategies, we're not going to use that level of detail.

We're going to use codes which are much simpler. They give broader-brush sorts of answers, still valid, but not with the same degree of precision. You don't need them. We want to make sure that the codes that we develop for those kinds of purposes, again, have a level of precision which matches what the regulatory need is.

And that relates to the last question which is, is the level of precision needed from a given code consistent with the expected overall level of accuracy required in the integrated analysis package for applications. Accident management is probably the primary one.

Okay. The next slide, please. (Slide) The long-term goals of this program -- I might mention at this point, before I forget, so that it is very clear -- we are not going to proceed with the program over the next three years, which is the near-term program -- finish that, and then pick up a long-term program.

The near-term program is being drawn out of the continuing research which we are doing in severe accident

area. It's a focusing. It's not switching from one program to another. So, when I talk about long-term goals, these are still things which we are studying right now. They are not things which we're going to wait three years before we start looking at them.

Our goals for the long-term is to provide better understanding of the range of phenomena in severe accidents. This is essentially so that we can understand the utility and benefits of accident management strategies. I think that's our primary use of this, although there are others.

And, secondly, relates to the source term, develop improved methods for assessing fission product behavior and the availability of fission products for release in a severe accident.

Next slide, please. (Slide) In achieving these long-term goals, we're going to have our research directed at, one, reducing the uncertainties in the estimate of risk, but also making sure that we have a broadly based severe accident research program, so that we can respond to changing technical priorities as they may arise over the next three years, and we've tried to lay out a program in which we think we're going to hit on what's important in the near-term and address those, but, you know, we're not prissy and things may change, and we want to make sure

that our program is sufficient broad and robust that we can respond to changing technical priorities.

What we are going to be addressing are seven areas of modeling the severe accident phenomena, is the first -- we just want to make sure we're pursuing the most risk-significant phenomena first.

In-vessel core melt progression and hydrogen generation. This is, again, the question of details of core melt progression and what happens there.

Hydrogen transport and combustion. Here, we're going to be focusing on, primarily, on high-temperature hydrogen, high-steam environments within the containment.

Fuel coolant interaction and molten core concrete interactions is primarily for accident management purposes.

Fission product chemistry and transport source term question came up again.

And the last thing is the fundamental data needs. Here, we're looking at both materials properties since, ultimately, it's how the materials respond that determine what happens in the course of an accident, and also the experimental data base, what data is available or needed for validation of these severe accident codes which, as I say, is our embodiment of our understanding.

With that, I will turn the presentation back to

Dr. Sheron.

CHAIRMAN ZECH: Thank you very much.

DR. SHERON: Next slide, please. (Slide) I guess this is the important stuff, is the money. Right now, what we've done is, we've done our resource projections for carrying out this program from FY90 through FY92.

What we are proposing is that we will carry this out, this revision, consistent with the current five-year plan approved by the Commission. What that means is, we're not going to ask for anymore money.

What we will do to implement this is to fund it by either reducing, or deferring, or perhaps even canceling research that we think would be of lower consequence in terms of its results.

So, basically, it's -- we're just going to re-prioritize where we spend our money, where we think we'll get our best results.

What you see there in this abbreviated table is the FTE that we will be expending on it -- that's the branch FTE in Dr. Costanzi's branch -- and the contractor assistance, which is the research money that the laboratories would -- and that the contractors, universities and the like, for carrying these out, and these are the numbers that are in the five-year plan.

COMMISSIONER ROBERTS: But the bulk of that is to the national labs?

DR. SHERON: Yes. I don't -- do you know what the split is right now?

DR. COSTANZI: I don't know, but I think the bulk is certainly --

DR. SHERON: The bulk is primarily because of the large experimental costs.

Next slide, please. (Slide) This just shows a breakdown of the -- what we would be spending from FY90 through FY92, to carry out the near-term issues 1 through 5. These are the scaling, the depressurization and DCH research, the MARK I, and the add water to degraded cores, and also the computer codes, and that's about $14.6 million.

On the next --

COMMISSIONER ROGERS: Excuse me.

DR. SHERON: Yes, sir?

COMMISSIONER ROGERS: Is that a three-year or a four-year period?

DR. COSTANZI: It's three years, '90 through '92.

DR. SHERON: It's three years, '90 through '92.

COMMISSIONER ROGERS: It says '93.

DR. SHERON: I'm sorry, there's a typo there on

the first bullet. I thought that had been corrected on your copies. That should be a "2".

COMMISSIONER ROGERS: Okay. So, this is a three-year program then?

DR. SHERON: Three-year, yes.

DR. COSTANZI: The same thing on the next page.

DR. SHERON: Okay. Also, on the first bullet on the next page, page 23, that also should be '92.

COMMISSIONER ROGERS: Okay.

DR. SHERON: Okay. These are the resources for the longer-term issues, again, for the three-year period, and these numbers, again, if you add the two, they are consistent with the five-year plan.

And beyond 1992, the -- any further research needs, the budget would be -- come, you know, from the normal Commission approval process for the budget.

COMMISSIONER ROGERS: Do you have handy there, the breakdown, year-by-year, of near-term and long-term expenditures? In other words, on page 21, slide 21, what the breakdown is for the $18 million and each succeeding year, in near-term and long-term?

DR. SHERON: Is that anywhere --

DR. COSTANZI: I think it was in a Commission paper. It's roughly level. It's roughly level.

COMMISSIONER ROGERS: Roughly level for each of

them? Okay.

DR. SHERON: And the last slide, 24 -- (slide) -- just to conclude, that is that we are starting to implement this program now. There are some programs which we are starting to revise right now, in FY89, to implement this program. We think, though, that the full implementation will begin in FY90.

One of the things we're doing right now is starting to develop our statements of work to be sent out to the contractors, from which we would expect proposals to come back in. These statements of work will reflect the emphasis on the short-term and consistent with this overall revised approach.

As you know, we met with the ACRS on a number of occasions, in developing this plan. You've received a letter from them. We are going to continue meeting with both the Subcommittee on Severe Accidents as well as the full committee, as we go forth and put this in place.

We are -- hopefully, we will get strong support from across-the-board, to assure that we can put this program in place. Obviously, it's a major change in some corners, and so, obviously, there may be people that are hesitant to either embrace it or the like. So, we're hoping that we will get strong support across-the-board on this, and we would propose to come back and brief you

periodically, on our progress in putting this plan in place. And that concludes my presentation.

CHAIRMAN ZECH: All right. Thank you very much.

MR. STELLO: That's all we have, Mr. Chairman.

CHAIRMAN ZECH: All right. Thank you.

Questions from my fellow Commissioners? Commissioner Roberts?

COMMISSIONER ROBERTS: I have no questions, just a comment. This is a terribly important program and, as you well know, the agency certainly had its critics of its research program. I'm delighted you got the ACRS onboard, and I hope that continues. That's all I have.

CHAIRMAN ZECH: Thank you. Commissioner Rogers?

COMMISSIONER ROGERS: Yes, I've got a couple of questions. Just coming back to this budget question, to what extent have you prioritized your budget in such a way as to be able to use the research results for -- in 1992, when the IPEs and PRAs come in, or have to be dealt with, from an NRC decisionmaking point of view?

Will the work be done during the first three years in such a way that you have the most useful results in hand for those purposes, and have your budget priorities been set to do that?

DR. SHERON: Yes. By its very nature, the short-term issues are high priority issues. If you

remember, the IPE generic letter essentially has told the industry that in conducting their IPEs, that they did not, at that time, have to address these specific issues that we're looking at here, the DCH or the melt through, the liner melt through.

The approach right now is to try and maximize this information that will be available for the industry at the time when they will be making their submittals and the like.

In terms of the actual budget money, what we're looking at is, what research do we need to do? Another question is, how much does that cost? It may not be a matter of if I put more money I'll get more information. We're putting -- sort of optimizing the amount of money that will get us the maximum benefit for -- within the next three years.

COMMISSIONER ROGERS: Well, have you had to define or put any research in the long-term category, just simply because you're trying to tuck it into the five-year plan expenditures? Has that driven a definition of long-term versus near-term to any extent?

DR. COSTANZI: No, I don't think so. What determined what was near-term and what was long-term was primarily 88-147, the closure plan -- severe accident closure plan, and what we thought we could achieve in

three years.

We took a hard look at where we were in understanding severe accidents, and what we thought was achievable in three years or, as I said, a good shot at it, and what we just recognize is going to have to be either a long-term goal or part of a -- you know, consistent part of an ongoing program. Materials property is an example of one of -- it's just something you continue to do, but that was basically how it was broken down, and we added up the money accordingly.

DR. SHERON: Yes. If you recall, I think back on slide 5, as I said, one of the things we really used in helping us do this was 1150. We wanted to see what was that telling us about where we should put our resources.

COMMISSIONER ROGERS: Well, I guess the thing that puzzles me a little bit, just in that you're spending -- you're projecting to spend about three times as much each year for long-term projects, as for near-term projects, during the first three-year period, and I don't know whether you expect to finish the near-term projects within the three-year period, or whether they extend out beyond that -- certainly, the long-term ones will -- and how that relates, again, to the usefulness of these research results for 1992, when you're going to start to need them.

DR. SHERON: The best way I can answer that is that this is our best guess at this time. Obviously, as we get further into defining this program, there may need to be some adjustments made, but, you know, obviously, if somebody walks in and says, "I can give you an experiment for X-amount of dollars that will definitively solve direct containment heating", and maybe it's going to cost more than what we've put here, obviously, we would consider that, but the objective here is to try and to reach some sort of a conclusion on these major issues within about three years, rather than leaving them hanging.

COMMISSIONER ROGERS: But do you really expect the near-term issues to be resolved by 1992? Is that --

DR. SHERON: That's our objective, yes.

COMMISSIONER ROGERS: That is.

DR. COSTANZI: Yes.

COMMISSIONER ROGERS: You don't explicitly say that. You say that you have these near-term objectives and you have a budget, a near-term budget, but they are not necessarily -- they don't necessarily terminate at the same point in time.

DR. COSTANZI: On the near-term portion of the program, we have very specific questions which we're going to try and answer, and the figures, the budget figures for

the near-term, are related to the programs to answer those specific questions.

Now, those questions aren't going to be answered in a vacuum. The ongoing research which we call the long-term, is going to help support the answers to those questions, but the specific programs which are aimed at developing the answers to those specific questions, are what appears in the budget figures for the near-term.

COMMISSIONER ROGERS: Well, just the point that you expect that these five topics to be more or less dealt with at the end of 1992.

DR. COSTANZI: Yes.

COMMISSIONER ROGERS: All right.

MR. ROSS: Commissioner Rogers?

COMMISSIONER ROGERS: Yes?

MR. ROSS: There's a variation on this that I think the Commission should know about. The utilities sponsored research -- for example, Commonwealth Edison is doing its own plant-specific direct containment heating research. They're doing cavity designs of their own reactors, and doing ejection of simulants and measuring dispersal, in furtherance of IPE, and the BWR owners groups are doing their own liner melt-through research.

So, it well may be -- we'll have to wait and see what the results are, but the definitive solution for IPE,

to the extent research is needed, may be that that's tendered by the industry.

COMMISSIONER CURTISS: I guess the more -- on the schedule topic, the more immediate schedule or question that I have is related to the CPI program where we've got the MARK I fixes now before us, and the remaining fixes are due in about eight months from now.

I gather that a large part of the focus of this program, particularly in the direct containment heating and the hydrogen area, will go to containment performance and containment integrity. With the near-term program running three years -- I gather we're kind of into it right now, but will be geared up in FY90 through '92, I guess the question that I have is, will this research be available prior to or incorporated in the CPI fixes, or is it envisioned that the research here in the short-term area for CPI, is going to be confirmatory? Can you talk about that?

DR. SHERON: For the MARK I, I believe it would be confirmatory. The MARK I improvements --

COMMISSIONER CURTISS: I understand that.

DR. SHERON: -- were basically balanced between a prevention accident management and a mitigative approach.

COMMISSIONER CURTISS: I guess I'm focusing on

the other fixes that come in in January of '90 -- the ice condensers and the large drys and the sub-atmospherics, where it looks, if that schedule holds, like we'll be getting recommendations on the CPI fixes more than two years before the conclusion of the near-term research program. Can you talk about that a little bit?

DR. SHERON: Yes. Right now, I guess, in talking with the people who are working on CPI, it's not clear yet how they intend to proceed. In other words, I think if they feel that this program will produce information that would provide a more definitive conclusion on the vulnerability, say, of a large dry to DCH -- for example, they may defer addressing specifically the DCH issue until that information becomes available, and maybe they would just focus in on perhaps the hydrogen aspect of it, of the large drys.

I don't think it's clear yet because it's not really clear what the fix is, I guess, for the large drys either, for DCH.

COMMISSIONER CURTISS: I guess my point is, you don't know what -- it won't be clear what the fixes are until you get the research done, will it, or is that not correct?

DR. SHERON: There may be improvements that can be made, that would not hinge on the research results.

COMMISSIONER CURTISS: I'm sorry. Go ahead.

COMMISSIONER ROGERS: No, that's fine. Just how are you incorporating plant operating and systems engineering expertise in your program? How do you actually incorporate kind of hands-on experience rather than the research laboratory experience, into the research program?

DR. SHERON: This would primarily come through the accident management -- in other words, the interface of the operator with the system. That is primarily being accomplished through the accident management program, and you had a briefing on that, I guess it was in January.

One of the things we are doing in that program right now is looking to contract with either organizations or individuals as consultants, that have actual hands-on operating experience. That was one of the comments we had received from Dr. Todreas', you know, group. So, that's where that impact would come from.

We also have a lot of human factor work going on in conjunction with the accident management research, on the human interface in accident management, but I'm not aware of any right now that we have in conducting this actual phenomenological type of research.

COMMISSIONER ROGERS: Well, I think that's a very important aspect of this, and I saw on your slide 14

that one of the questions on depressurization is -- you know, is there time available for operators to take action. Well, how do you decide that? To what extent are operators involved in arriving at that kind -- in answer to that kind of a question?

MR. STELLO: Remember, Commissioner Rogers, we've asked industry, and I understand they will put together a group to help us do just this, through bringing in people with that kind of expertise and dealing with the severe accident management issues, to make sure they --

COMMISSIONER ROGERS: But what would the mechanism for that be between those two plans?

MR. STELLO: Well, they're going to get a group in to make sure that the coupling, on how you handle the severe accident procedures, has that kind of thinking, from the hands-on plant operator point of view, and making sure that those are, in fact, reasonable procedures and processes that we propose, and they have committed to put together that group and help deal with that issue.

COMMISSIONER ROGERS: Okay. On your experimental programs, those laboratory experiments of various kinds, I had a little difficulty identifying how much experimental work there is and how much is computer code development and theoretical work in the overall program and, in connection with that, I wonder what your

thinking is of the necessity for duplicative experiments to verify some of these results. It's a tricky question because, if you're talking about large scale experiments, how many can you mount, but how are you coming to-- arriving at that kind of a decision, when you've got a large scale experiment that is unique? What verification experiments can you conduct, or are you thinking of being able to -- what kind of data can you bring onto this to verify that it's been correctly assessed?

DR. COSTANZI: Well, you know, that's a difficult question which we're constantly dealing with in this program, but we do attack it from a number of positions. First, we do certainly duplicate on a selected basis, the small scale experiments.

We tend to rely even more heavily on the questions of scaling, and this is scaling methodology which we're attempting to develop, to then extrapolate to larger scales, and we'll do some larger scale experiments and then, of course, ultimate extrapolation to the accident.

We're also trying to coordinate our research with EPRI-sponsored research, as well as research that is taking place in other countries, both for large scale and small scale experiments, so that there is some overlap, that there are certain benchmarks provided in their

experimental program and our experimental programs, by which we can compare and then, essentially, leverage the utility of several research programs, to try and come to grips with the severe accident question.

So, it's a combination of doing some very-- some duplicative experiments at the lab scale, occasionally, on a large scale, comparing and then benchmarking our experiments with experiments by industry as well as experiments and experimental programs in other countries.

COMMISSIONER ROGERS: Well, it would be interesting to know, when all is said and done, when you've more or less finished, what pieces of important data are essentially unique, that they're one of a kind, that you haven't been able to do that, for just practical reasons.

DR. COSTANZI: One of the programs which we're looking at right now is to try and assemble a catalog of severe accident research experimental data, to try and put it all together and find out just those kinds of questions -- what data is available, what experiments have been done, how good is it, what it can be used for, has it been verified or matched by other experiments, or is it unique -- and we have a contract under consideration right now, to do just that.

DR. SHERON: The scaling work, which is being headed up by Dr. Zuber, is also designed to do just that. It's to look at these experiments, and he's -- right now, he's focusing in, for example, on the SERTSI facility, which is the one-tenth scale direct containment heating facility out in Sandia, and that's to primarily look at whether there's any unique aspects of that facility that are either distorting it in terms of its scalability, and -- but that's really one of the reasons we focused in on that program, is to address the very question you're asking.

COMMISSIONER ROGERS: Does your scaling program -- I assume it does -- deal with not only size scale, but dimensional scale -- 2D, 3D questions?

DR. SHERON: Yes.

COMMISSIONER ROGERS: You do that.

DR. SHERON: Yes.

COMMISSIONER ROGERS: All right. Good. Thank you very much. It looks like a good program.

CHAIRMAN ZECH: Commissioner Curtiss?

COMMISSIONER CURTISS: I just have maybe an observation and perhaps a couple of requests. One, it seems to me that the program's well thought out. I, too, am pleased that we've gotten a consensus with the ACRS and the review committee and EPRI, and it looks to me like

it's well thought out and a forward-looking program.

Just a couple of requests. When you reach the point when you identify those research programs that will need to be postponed or cancelled, if it would be possible to let us know, maybe just in an information paper, what those are, to get a sense of what the choices are here, first.

And then, secondly, maybe a question that goes more to the folks working on CPI. I still, I guess, have questions about the schedule or relationship between the research results of this program and the CPI program, and perhaps if the folks who are working on the CPI can give us some sense of what impact, if any, the research program that you've laid out will have on their schedule for the remaining CPI fixes, I'd be interested in that.

DR. SHERON: Okay.

COMMISSIONER CURTISS: That's all I have. Thank you.

CHAIRMAN ZECH: Thank you.

Well, let me just make a comment. First of all, I think this has been an excellent presentation, not only with the information you've presented us, but I think Research Office, Mr. Beckjord, you and your people are to be commended on the coordination that you're doing with other activities.

I think it's important that you do this. The fact that you've coordinated well with the ACRS, with, I notice, too, in your paper, with AEOD and the NRR, in our own organization, as well as with EPRI and your Nuclear Safety Research Review Committee, other industry groups, universities, too. This is the kind of thing I think the Commission has been wanting to see for a long time.

We recognize that there's a lot of talent in our Research people here, but I think that your efforts to coordinate with others, and to listen to what they have to say is extremely important and it certainly broadens our base and gives us a greater confidence that our research program is doing what we want it to do. I commend you for that, and I would ask you to continue in that direction.

So, I think that, in itself, as much as the details you've presented, is extremely important, and I commend you and your people in Research, who are doing that. I think you recognize that the Commission has a great deal of respect for what research can offer to our regulatory decisions, but they must be integrated, and they must be part of our program.

They can be. We know they are, but your efforts to go outside your own organization to make sure that your research is consistent with and supportive of our

regulatory decisions is extremely important, and I commend you for that, again.

I think that the -- also, I'd like to just mention that the ACRS -- I read with great interest the letter that they wrote to the Commission, and also the comment they had that said that regarding the proposed severe accident research program plan, the ACRS commented that they felt it represents a substantial change from previous severe accident research programs, and is a very positive step. I commend you for that, too, because the ACRS do have a lot to offer us in an advisory capacity, and you briefed them, I notice, on several occasions, and not only the subcommittee but the full committee. I think that is important, too, because we do listen to their comments, and your efforts to be close to them is important to the Commission.

And the ACRS comment that you have a -- you made a specific effort to ensure that your contractors show that the proposed continuing work that they're doing addresses the analysis and the important phenomena that are aimed at predicting risk, and were clearly defined objectives. And, so, the focus you've made on that, as pointed out by the ACRS, I think, is important also.

So, I think the plan certainly shows that you've attempted to go beyond your own office and to interface

with others who can contribute to our effort in severe accident research.

I also note that you've emphasized efforts to eliminate duplication of effort, and to prioritize your research program, and I'd just simply note that that's good management.

We don't have unrestrained resources. Research is so important to our programs. We simply must put our dollars where we can get the most impact on our regulatory safety decisions, and I clearly commend you for that.

The Commission has stressed on many occasions, the need to better integrate our research program with our regulatory programs, and it would look to me, Mr. Stello, Mr. Beckjord, that you're clearly doing that. That's what we want to see, and I think we're seeing it. And, so, that's why it's particularly encouraging to me to hear those views at this briefing.

I think, again, that the Commission recognizes that we'll be able to make better regulatory decisions when we have solid research that's well integrated, not at the last minute, but continually as you're moving through these difficult analyses and tests and studies that research contributes to so heavily. And, again, it will help us carry out our statutory responsibilities for public health and safety, if we recognize that the

research program is making such a strong contribution, and I only look ahead a little bit to the waste issues we have in front of us, as well as the severe accident issues, and look to the plant life extension program where we are going to count on strong research efforts.

So, in those areas, too, as well as severe accident, I would encourage you to continue the efforts you've obviously made in this area, to go outside our agency and to seek the advice and counsel of those in universities and -- as well as your review committee, and the ACRS, EPRI, and others, who can assist.

So, I'm very encouraged by what I've heard today, and I think that you're showing the proper direction in your severe accident research plan, and in particular your strong efforts, obviously, to coordinate and cooperate with others.

Do any of my fellow Commissioners have any other -- yes?

COMMISSIONER ROGERS: Just one more brief thing, and that is that I was very favorably impressed with everything I heard today, and I certainly learned a number of things.

It seems to me that it would be helpful, perhaps, for you to emphasize in your whatever summaries of this program that you write up, that what the rationale

was for your division into near-term and long-term work, not what the division was but the rationale for that division, so it's clear what your philosophy is and your point of view in making those decisions and assignments because I think it will help you to give a greater sense of integrity to the entire program.

CHAIRMAN ZECH: Are there any other comments?

(No response.)

If not, thank you for an excellent presentation.

We stand adjourned.

(Whereupon, at 11:14 a.m., the meeting was adjourned.)

CERTIFICATE OF TRANSCRIBER

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

TITLE OF MEETING: BRIEFING ON SEVERE ACCIDENT RESEARCH PLAN

PLACE OF MEETING: ROCKVILLE, MARYLAND

DATE OF MEETING: MAY 2, 1989

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

Phyllis Young

Reporter's name: PHYLLIS YOUNG

COMMISSION BRIEFING

ON

REVISED SEVERE ACCIDENT RESEARCH PROGRAM PLAN

BRIAN SHERON, DIRECTOR
DIVISION OF SYSTEMS RESEARCH
OFFICE OF NUCLEAR REGULATORY RESEARCH

FRANK COSTANZI, CHIEF
ACCIDENT EVALUATION BRANCH
DIVISION OF SYSTEMS RESEARCH
OFFICE OF NUCLEAR REGULATORY RESEARCH

MAY 2, 1989

OBJECTIVES

- DEVELOP A TECHNICAL BASE OF SEVERE ACCIDENT INFORMATION WHICH CAN BE USED IN ALL COMMISSION ACTIONS RELATED TO SEVERE ACCIDENTS

- DEVELOP A TECHNICAL BASE OF SEVERE ACCIDENT INFORMATION TO CONFIRM DECISIONS RELATED TO SEVERE ACCIDENTS (I.E., CPI, IPE, ACCIDENT MANAGEMENT)

- DEVELOP UNDERSTANDING OF MOST LIKELY PLANT RESPONSE TO SEVERE ACCIDENTS AND HELP ENSURE THERE ARE NO UNEXPECTED PHENOMENA THAT WILL CHANGE OUR PERCEPTION OF RISK

DEVELOPMENT OF THE REVISED SARP

- o REVISION NEEDED TO ENSURE RESEARCH WAS CONSISTENT WITH SEVERE ACCIDENT INTEGRATION PLAN (SECY 88-147)

- o DEVELOPMENT OF REVISED SARP DREW UPON BROAD SPECTRUM OF NATIONAL EXPERTISE

  - EXPERTS FROM NATIONAL LABS, INDUSTRY, UNIVERSITIES, AND STAFF WERE CALLED TO HELP IDENTIFY HOW CURRENT SARP ACTIVITIES RELATED TO SECY 88-147

  - EXPERTS WORKED IN 3 GROUPS. RESULTS WERE PROVIDED TO SENIOR OVERSIGHT GROUP (STAFF MANAGEMENT AND CONTRACTOR MANAGEMENT)

  - OVERSIGHT GROUP ASSIMILATED AND FOCUSED INFORMATION FROM 3 GROUPS.

  - STAFF THEN USED RESULTS FROM OVERSIGHT GROUP TO DEVELOP PLAN.

DEVELOPMENT OF THE REVISED SARP (CONT.)

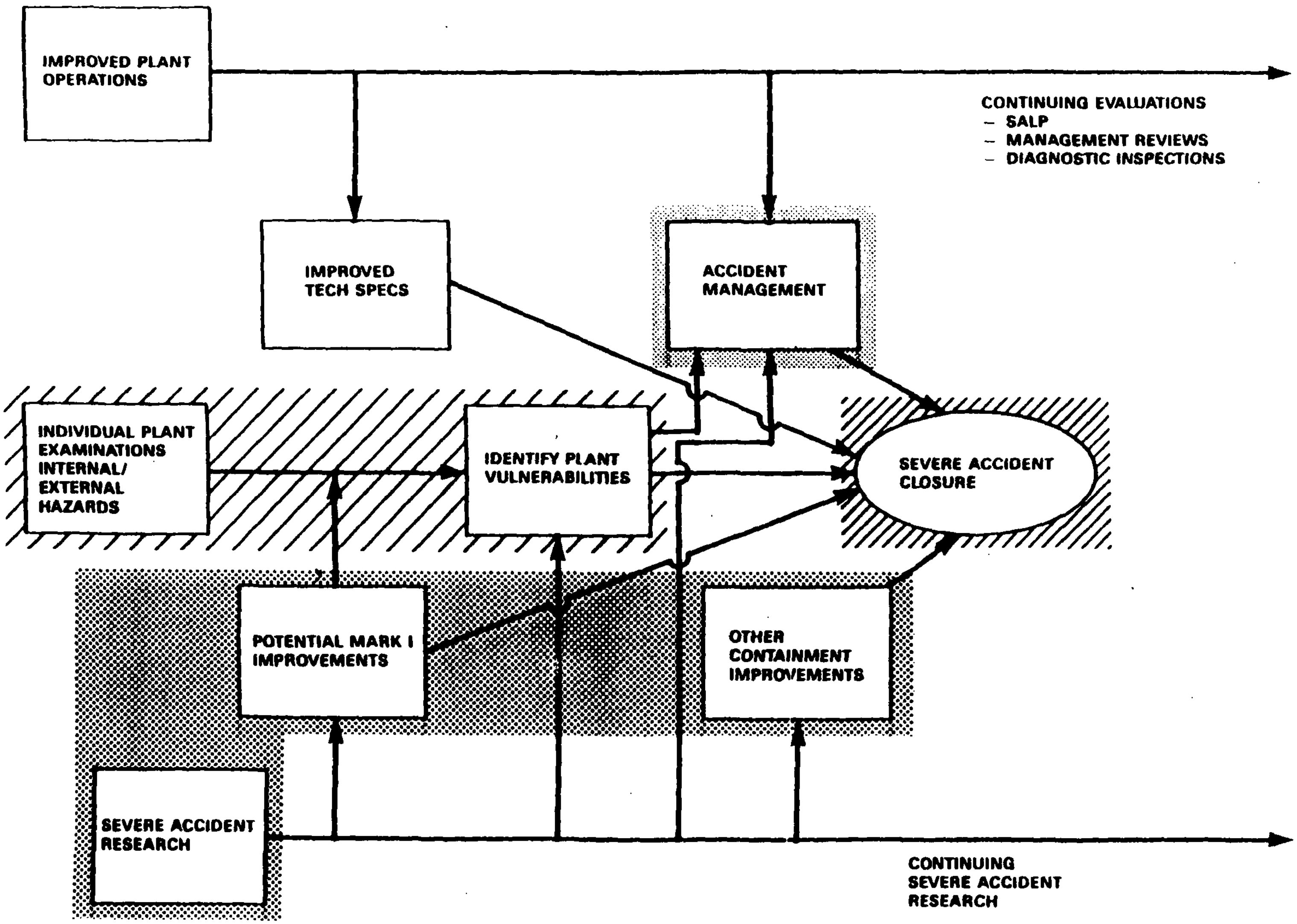
- DEVELOPMENT OF REVISED SARP USED NUREG 1150 AS GUIDANCE FOR DEFINING AND RANKING SEVERE ACCIDENT ISSUES TO BE ADDRESSED

- DRAFT PLAN WAS WIDELY CIRCULATED FOR COMMENT TO NATIONAL LABS, UNIVERSITIES, EPRI, RESEARCH REVIEW COMMITTEE (NSRRC) AND ACRS.

- COMMENTS RECEIVED VIRTURALLY WERE ALL FAVORABLE

- ACRS WAS PARTICULARLY ENCOURAGING

FIGURE 1
SEVERE ACCIDENT PROGRAM - SCHEMATIC



REVISED SARP IS DIVIDED INTO NEAR-TERM AND LONGER-TERM WORK

MAJOR AREAS OF EMPHASIS IN NEAR TERM THOSE RELATED TO EARLY CONTAINMENT FAILURE

(RISK SIGNIFICANT IN NUREG 1150)

- DIRECT CONTAINMENT HEATING
- MARK I LINER MELT THROUGH
- EFFECT OF WATER ON A MOLTEN CORE

AND TWO OTHERS THAT ARE RELATED TO HOW THE RESEARCH IS CARRIED OUT

- SCALING (SIMILITUDE)
- "ACCURACY" CRITERIA FOR CODES/CODE DEVELOPMENT

MAJOR AREA OF EMPHASIS IN NEAR TERM (CONTINUED)

- IN ADDITION TO RESTRUCTURING PROGRAMS INTO NEAR AND LONGER TERM ELEMENTS, PROGRAMS ALSO REVISED APPROACH TO RESEARCH METHODOLOGY

- SPECIAL, INCREASED EMPHASIS ON SIMILITUDE (OR SCALING) RATIONALE OF EXPERIMENTS

- INCREASED EMPHASIS IN UNDERSTANDING LATE PHASE CORE MELT BEHAVIOR. OBJECTIVE TO BETTER UNDERSTAND EXPECTED AMOUNT, COMPOSITION, AND TEMPERATURE OF CORIUM THAT IS AVAILABLE FOR RELEASE FOR THE LOWER HEAD OF A LWR

- INCREASED EMPHASIS ON UNDERSTANDING AND PREDICTING EXPECTED LOWER HEAD FAILURE MODE

- SUCCESS IN CARRYING OUT ABOVE APPROACH HOPEFULLY WILL REDUCE CURRENT PHENOMENA UNCERTAINTIES ASSOCIATED WITH EARLY CONTAINMENT FAILURE.

CODE DEVELOPMENT REVISIONS

- CODE DEVELOPMENT WILL BE PUT ON A MORE STRUCTURED SCHEDULE (CONSISTENT WITH CURRENT APPROACH FOR T/H CODES)

- DOCUMENTATION OF CODES WILL BE UPGRADED AND IMPROVED

- NEED FOR CODE IMPROVEMENTS WILL PROCEED ON A MORE STRUCTURED, DEPENDABLE BASIS. NEEDED IMPROVEMENTS WILL BE BETTER JUSTIFIED ON THE BASIS OF IMPORTANCE TO UNDERSTANDING BEHAVIOR (AND IMPORTANCE TO RISK)

- NRC CURRENTLY SUPPORTS 4 CORE MELT PROGRESSION CODES (MELPROG-TRAC, RELAP-SCDAP, BWRSAR, MELCOR). NEED FOR THIS MANY CORE-MELT PROGRESSION CODES WILL BE ASSESSED, AND PROGRAM REVISIONS WILL BE MADE ACCORDINGLY.

- AN ATTEMPT WILL BE MADE TO IDENTIFY CODE ACCURACY CRITERIA. OBJECTIVE IS TO IMPROVE OUR ABILITY TO CLEARLY DETERMINE WHEN A CODE IS SUFFICIENT FOR INTENDED PURPOSE AND DEVELOPMENT CAN BE COMPLETED

GOALS OF THE REVISED SARP

- NEAR TERM GOALS

  PROVIDE TECHNOLOGICAL BASE FOR ASSESSING CONTAINMENT PERFORMANCE OVER THE RANGE OF RISK-SIGNIFICANT CORE MELT EVENTS.

  DEVELOP THE CAPABILITY TO EVALUATE THE EFFICACY OF GENERIC CONTAINMENT PERFORMANCE IMPROVEMENTS.

GOALS OF THE REVISED SARP

- o NEAR TERM GOALS
  - SCALING (SIMILITUDE)
  - DIRECT CONTAINMENT HEATING
  - MARK I LINER MELT THROUGH
  - EFFECT OF WATER ON A MOLTEN CORE
  - "ACCURACY" CRITERIA FOR CODES/CODE DEVELOPMENT

ISSUE 1 SIMILITUDE/SCALING

- o HOW TO EXTRAPOLATE FROM SMALL SCALE EXPERIMENTS TO FULL SCALE NUCLEAR POWER PLANTS ACCIDENTS
  - ARE THE CORRECT PHENOMENA BEING INVESTIGATED?
  - ARE DISTORTIONS OF SCALE IMPORTANT TO THE PROCESS UNDER INVESTIGATION?
  - ARE DISTORTIONS CHARACTERIZED, UNDERSTOOD AND ACCOUNTED FOR WHEN USING CODES?
- o PROGRAM DESIGNED TO DEVELOP A GENERALIZED APPROACH TO SCALING FOR SEVERE ACCIDENTS PHENOMENA

ISSUE 2 - DEPRESSURIZATION AND DCH

A. DEPRESSURIZATION AS A MEANS TO AVOID DCH

B. OTHER EFFECTS OF DEPRESSURIZATION VS. THREAT OF DCH

ISSUE 2 - DEPRESSURIZATION AND DCH (CON'T)

SPECIFIC QUESTIONS TO BE ADDRESSED

A. DEPRESSURIZATION AS A MEANS TO AVOID DCH

1. WHAT IS THE LIKELIHOOD THAT THE RCS WILL FAIL BY NATURAL CIRCULATION PRIOR TO SIGNIFICANT CORE DAMAGE?

2. IS THERE A LOW-PRESSURE CUTOFF BELOW WHICH THERE IS NO DCH THREAT?

3. IF SO, WILL THIS PRESSURE BE REACHED BY NATURAL CIRCULATION INDUCED FAILURE OF THE RCS, OR THROUGH OPERATOR ACTION TO DEPRESSURIZE, OR BOTH?

ISSUE 2 - DEPRESSURIZATION AND DCH (CON'T)

SPECIFIC QUESTIONS TO BE ADDRESSED

B. OTHER EFFECTS OF DEPRESSURIZATION VS. THREAT OF DCH

1. IF OPERATOR ACTION IS NEEDED TO DEPRESSURIZE TO AVOID DCH, IS THERE TIME AVAILABLE FOR THE OPERATOR TO TAKE ACTION? WHAT ARE THE HARDWARE AND PROCEDURES NEEDED?

2. ARE THERE ADVERSE CONSEQUENCES TO DEPRESSURIZATION? WHAT ARE THE POTENTIAL CONSEQUENCES? WHAT IS THE RISK SIGNIFICANCE OF NATURAL CIRCULATION INDUCED RCS FAILURE?

3. WHAT IS THE NATURE OF THE DCH THREAT, AND WHAT MECHANISMS (E.G., SPRAYS) AND CONFIGURATIONS EXIST EX-VESSEL THAT CAN MITIGATE OR ELIMINATE THE THREAT OF DCH?

ISSUE 3 - BWR MARK I CONTAINMENT

SHELL MELTTHROUGH

SPECIFIC QUESTIONS TO BE ADDRESSED ARE THE FOLLOWING:

1. WHAT IS THE RELATIONSHIP OF THE BWR BOTTOM HEAD FAILURE MODE TO VARIATIONS IN QUANTITY, COMPOSITION, TEMPERATURE, AND TIMING OF ARRIVAL OF THE MELT ON THE BOTTOM HEAD?

2. WHAT IS THE EFFECT OF WATER ON THE DRYWELL FLOOR WHEN THE MELT POURS OUT FROM THE PRESSURE VESSEL?

3. HOW DOES THE ANSWER TO THE ABOVE QUESTION DEPEND ON THE INITIAL CONDITIONS (MELT EJECTION RATE, MELT SUPERHEAT, WATER SUBCOOLING, MELT COMPOSITION) AND WATER ADDITION RATE?

4. UNDER WHAT CONDITIONS WOULD THE CRUST THAT FORMS AT INITIAL CONTACT BETWEEN THE MELT AND THE SHELL BE STABLE AND FOR HOW LONG? WHAT IS THE EXPECTED RATE OF HEAT TRANSFER BETWEEN THE CORE MELT MATERIALS AND THE SHELL FOR VARIOUS MELT CONDITIONS?

ISSUE 4 - ADDING WATER TO A DEGRADED CORE

THE QUESTIONS TO BE ADDRESSED ARE THE FOLLOWING:

ACCIDENT MANAGEMENT STRATEGIES INCLUDE RESTORING A SOURCE OF COOLANT TO THE PRIMARY SYSTEM. RESEARCH ASSUMES OPERATOR WILL ALWAYS ADD WATER IF AVAILABLE.

1. WHAT ARE THE AMOUNTS AND RATES OF HYDROGEN AND STEAM GENERATION DURING REFLOODING OF DEGRADED CORE AND DURING MELT RELOCATION ONTO THE BOTTOM HEAD?

2. WHAT IS THE POTENTIAL FOR RECRITICALITY IN BWR SEVERE CORE DAMAGE ACCIDENTS? WHAT WILL BE THE CONSEQUENCES?

ISSUE 5 - THE USE AND STATUS OF SEVERE ACCIDENT MODELS

AS PART OF THE CODE DEVELOPMENT PROGRAM, A METHOD MUST BE DEVELOPED THAT CONSTANTLY ASSESSES THE STATE OF CODE DEVELOPMENT FROM A NUMBER OF STANDPOINTS. THESE ARE:

1. HOW WELL DO THE MECHANISTIC MODELS REFLECT THE PHENOMENA BELIEVED TO BE IMPORTANT TO SEVERE ACCIDENTS?

2. HOW WELL DOES THE INTERACTIVE PROGRAM OF CODE ADVANCEMENT/ EXPERIMENTATION ACHIEVE THE OBJECTIVE IMPLIED IN (1.) ABOVE

3. IS THE LEVEL OF DETAIL IN THE CODES APPROPRIATE TO THEIR USE? ARE SOME CODES MORE DETAILED THAN NEEDED, OTHERS NOT DETAILED ENOUGH?

4. WHICH STAGES OF AN ACCIDENT NEED TO BE MODELED BY DETAILED MECHANISTIC CODES, AND WHICH NEED COUPLING TO ADJACENT STAGES?

ISSUE 5 - THE USE AND STATUS OF SEVERE ACCIDENT MODELS (CONT.)

5. IS THE LEVEL OF PRECISION NEEDED FOR REGULATORY USE BEING CONSIDERED IN THE CODE DEVELOPMENT PROGRAMS?

6. IS THE LEVEL OF PRECISION NEEDED FROM A GIVEN CODE CONSISTENT WITH THE EXPECTED OVERALL LEVEL OF ACCURACY REQUIRED OF AN INTEGRATED ANALYSIS PACKAGE FOR APPLICATIONS AS DISCUSSED ABOVE (E.G., ACCIDENT MANAGEMENT)?

GOALS OF THE REVISED SARP (CONT.)

- o LONG TERM GOALS

  PROVIDE AN UNDERSTANDING OF THE RANGE OF PHENOMENA IN SEVERE ACCIDENTS

  DEVELOP IMPROVED METHODS FOR ASSESSING FISSION PRODUCT BEHAVIOR AND AVAILABILITY FOR RELEASE

LONG TERM GOALS

- o DIRECTED AT
  - REDUCING UNCERTAINTIES IN ESTIMATE OF RISK
  - BROADLY BASED SEVERE ACCIDENT RESEARCH PROGRAM

  1. MODELING SEVERE ACCIDENT PHENOMENA
  2. IN-VESSEL CORE MELT PROGRESSION AND HYDROGEN GENERATION
  3. HYDROGEN TRANSPORT AND COMBUSTION
  4. FUEL COOLANT INTERACTION
  5. MOLTEN CORE-CONCRETE INTERACTIONS
  6. FISSION PRODUCT CHEMISTRY AND TRANSPORT
  7. FUNDAMENTAL DATA NEEDS

RESOURCES PROJECTIONS FOR FY 1990 THROUGH FY 1992

- o RESOURCES TO CARRY OUT THE REVISED SARP ARE CONSISTENT WITH THE CURRENT FIVE YEAR PLAN APPROVED BY THE COMMISSION,

- o NEW PROGRAMS TO IMPLEMENT THE REVISED SARP WILL BE FUNDED BY REDUCTION, DEFERRAL, OR CANCELLATION OF LOWER CONSEQUENCE SEVERE ACCIDENT RESEARCH.

	FY 1990	FY 1991	FY 1992
FTE	10	10	10
CONTRACTOR ASSISTANCE	$18M	$23.4M	$23.8M

RESOURCES PROJECTIONS FOR FY 1990 THROUGH FY 1992 (CONTINUED)

- NEAR TERM ISSUES RESOURCE PROJECTION FOR FY 1990 THROUGH FY 1992 ARE AS FOLLOWS:

ISSUE #1	SCALING ANALYSIS	$1.8M
ISSUE #2	DEPRESSURIZATION AND DCH	$7.4M
ISSUE #3	BWR MARK I CONTAINMENT SHELL MELTTHROUGH	$2.5M
ISSUE #4	ADDING WATER TO DEGRADED CORES	$2.0M
ISSUE #5	USE AND STATUS OF SEVERE ACCIDENT CODES	$ .9M
SUBTOTAL		$14.6M

RESOURCES PROJECTIONS FOR FY 1990 THROUGH FY 1992 (CONT.)

- LONG TERM ISSUES RESOURCE PROJECTION FOR FY 1990 THROUGH FY 1992 ARE AS FOLLOWS:

L1	MODELING SEVERE ACCIDENTS	\$16.0M
L2	IN-VESSEL CORE MELT PROGRESSION AND H_2 GENERATION	\$17.0M
L3	FUEL-COOLANT INTERACTIONS	\$3.0M
L4	HYDROGEN TRANSPORT AND COMBUSTION	\$3.0M
L5	MOLTEN CORE-CONCRETE INTERACTION	\$6.7M
L6	FISSION PRODUCT BEHAVIOR AND TRANSPORT	\$4.5M
L7	FUNDAMENTAL DATA NEEDS	\$0.5M
SUBTOTAL		\$50.7M

- RESOURCES NEED BEYOND FY 1992 FOR THE LONG-TERM PROGRAM WILL BE SUBMITTED FOR COMMISSION APPROVAL AS PART OF THE NORMAL BUDGETARY PROCESS

CONCLUDING REMARKS

- o STAFF IS STARTING TO IMPLEMENT PROGRAM NOW. FULL IMPLEMENTATION WILL BEGIN IN FY90.
- o STAFF WILL CONTINUE TO MEET WITH ACRS ON PROGRAM AS IT DEVELOPS.
- o STRONG SUPPORT FOR PROGRAM IS NEEDED TO ASSURE ITS IMPLEMENTATION
- o COMMISSION WILL BE BRIEFED PERIODICALLY ON PROGRESS OF PROGRAM.

April 20, 1989

POLICY ISSUE
(Information)

SECY-89-123

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: REVISED SEVERE ACCIDENT RESEARCH PROGRAM PLAN

Purpose: The purpose of this paper is to inform the Commissioners of the staff's revised Severe Accident Research Program (SARP) which supports the tasks and objectives discussed in the staff's "Integration Plan for Closure of Severe Accident Issues," SECY-88-147. The principal objectives of this paper are:

1. To describe the major objectives and elements of the revised SARP.

2. To describe how the SARP activities relate to the Commission's policy, strategic goals, and other activities associated with closure of severe accident issues.

3. To describe how the SARP activities relate to those actions resulting from the Commission's review of the FY 1989-1993 Draft Five Year Plan, namely:

   a. the staff's program for resolution of uncertainties in the source term, particularly as they impact closure of severe accident issues and

   b. the need for additional research on hydrogen transport and combustion beyond FY 1989 (provided as Appendix 1 to this paper).

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<u>Background:</u>

For the past 10 years, since the Three-Mile Island accident, NRC has sponsored an active research program on light water reactor severe accidents as part of a multifaceted approach to reactor safety. In August 1985, the Commission issued a Severe Accident Policy Statement (50 FR 32138) in which the Commission concluded that, based on available information, existing plants posed no undue risk to the public health and safety and that there is no present basis for immediate action for any regulatory requirements for these plants. However, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), the Commission was convinced of the need for both continuing research on severe accidents and a systematic examination of each existing plant to identify any plant-specific vulnerabilities to severe accidents. These systematic examinations are now being accomplished under the Individual Plant Examination (IPE) program as described in the staff's Integration Plan for Closure of Severe Accident Issues (SECY-88-147) which was presented to the Commission in May 1988. That plan consists of six major elements:

1. Examination of existing plants for severe accident vulnerabilities (individual plant examinations).

2. Development of generic containment performance improvements with respect to severe accidents to be implemented if necessary for each of the six containment types used in the USA, namely, BWR Mark I, Mark II, and Mark III; and PWR large dry, subatmospheric, and ice condenser.

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.

Discussion: The subject of this Commission paper is to describe the revised Severe Accident Research Program (SARP), Item 4 above, and how the revisions are designed to provide confirmatory information and technical support to the NRC staff in implementing the staff's Integration Plan for Closure of Severe Accident Issues as described in SECY-88-147. (The revised SARP is provided as Attachment 1 to this paper.) The revised SARP addresses both the near-term research directed at providing a technical basis upon which decisions on important containment performance issues can be made, and the long-term research needed to confirm and refine our understanding of severe accidents. In developing this plan, the staff recognized that the overall goal is to reduce the uncertainties in the source term sufficiently to enable the staff to make regulatory decisions on severe accident issues. However, the staff also recognized that for some issues it may not be practical to attempt to reduce uncertainties further, and some regulatory decisions or conclusions will have to be made with full awareness of existing uncertainties.

The near-term goals of the revised SARP, as stated in Attachment 1, are:

1. Provide the technological base for assessing containment performance over the range of risk-significant core melt events.

2. Develop the capability to evaluate the efficacy of generic containment performance improvements.

and the long-term goals are:

3. Provide an improved understanding of the range of phenomena exhibited by severe accidents that includes the impacts of generic accident management schemes.

4. Develop improved methods for assessing fission product behavior and availability for release in the event of containment failure at various phases of severe accident sequences.

These goals and their related issues set forth in the revised SARP plan are consistent with the Integration Plan for Closure of Severe Accident Issues (SECY-88-147). For example, in the near-term, the revised SARP addresses issues and phenomenological uncertainties associated with accident sequences that lead to early containment failure: direct containment heating, BWR Mark I containment shell meltthrough, ex-vessel molten fuel-coolant interaction in BWR Mark II and III containments, and hydrogen combustion in BWR Mark III and PWR ice condenser containments. For each of these issues, a program has been developed that attempts to obtain key data and information needed to make an informed decision on each issue within a fixed timeframe.

In Appendix 2 to this paper a summary is provided of how the results of the research to be conducted under the revised SARP will provide the required information needed to resolve near-term issues. The staff's current estimate is that the near-term programs will yield information that will provide tangible results within 3 years useable in the decisionmaking process in considering (a) the likelihood of the phenomena in question, (b) various preventive or mitigative actions that can either reduce or eliminate the probability of recurrence of the phenomena, or reduce or eliminate any adverse consequences associated with the phenomena, and (c) the likelihood that further research can or will substantially reduce uncertainties and the associated costs of such research.

DEVELOPMENT OF THE REVISED SARP

During the development of the revised SARP, experts actively involved in severe accident research and severe accident phenomenology were drawn from laboratories, universities, industry, and NRC to consider the detailed technical issues and their status and to help the NRC staff define the needs for further research within the framework of SECY-88-147.

The staff subsequently developed a draft revised SARP in February 1989 that was sent to the DOE laboratories, industry groups, individuals at several universities, and the experts mentioned earlier, asking for their comments. ACRS, AEOD, and NRR comments were also sought, as were those of the recently created "Nuclear Safety Research Review Committee" as to whether the specific research tasks discussed in the revised SARP plan are reasonable and appropriate with respect to achieving the goals set forth in the plan. The comments received on the revised SARP were virtually all favorable, with most comments focused on selected specific details in the plan.

The staff met with and briefed both the ACRS Severe Accident Subcommittee and the full committee on the revised SARP on March 7 and 9, 1989, respectively. In its letter dated March 15, 1989, from Forrest J. Remick to Chairman Lando W. Zech, Jr., "Proposed Severe Accident Research Program Plan," the ACRS stated that the revised SARP represents a substantial change from the previous severe accident research program and is a very positive step. The ACRS endorses the requirement of the revised SARP that contractors show that their proposed continuing work address analyses and phenomena important in the prediction of risk, and have clearly defined objectives.

The ACRS also recommended, as an alternative to studies of various phenomena associated with direct containment heating (DCH) which amounts to a major fraction of the near-term program's resources, that greater priority should be given to studies that might very well demonstrate that the risk from DCH is negligibly low or could be made low by readily achievable plant modifications or procedural changes, thus making much of the proposed DCH-related research unnecessary. In general, the staff agrees with this comment, and the revised SARP in Attachment 1 to this paper contains a balanced approach, consistent with their comment, to assess whether DCH presents a real challenge to containment integrity. In addition, under the revised SARP, whether such a threat is likely, and what is the incremental increase in the risk from DCH will be assessed.

Finally, the staff also has made changes to the revised SARP, per an ACRS comment, to emphasize how results of previous work or expected results from existing research programs sponsored by the U.S. industry or foreign organizations are to be factored into the NRC program.

SUMMARY OF THE REVISED SARP

The plan provides a description of the general approach to the needed research. Naturally, many of the details associated with specific experiments or analyses have not yet been developed and can only be developed after the specific research is identified. Therefore, for each near term issue, a detailed research plan will be developed, describing each experimental and analytical program and how these programs fit together and lead to a resolution of the issue.

In the remainder of this paper, the salient features of the revised SARP are brought to the Commission's attention.

1. Similitude (Scaling)

Because of the excessive costs associated with large-scale experiments involving molten materials that cover the range of potential severe reactor accident conditions of interest, most severe accident research is performed in small-scale facilities, many conducting only separate effects tests. Although conducting research in such a manner is not uncommon, questions about whether and to what extent the small-scale tests reflect the fundamental realities of a reactor accident must be addressed. Hence, a program to systematically investigate similitude (commonly called scaling) has been started. The objective of this program is to develop methods and criteria that will allow information from small-scale experiments to be confidently scaled up to full-size reactors. More importantly, it will also tell us where information from small-scale facilities cannot be confidently scaled up. These methods and criteria would be applied

before proposed new experiments were funded. This program initiative is not intended in any manner to constrain closure of those other near-term issues defined herein (and in SECY 88-147). Rather, it is a more demanding rigour being applied to similitude of the experimental work of smaller-scale which has prospective benefits to both the near- and long-term issues.

2. <u>Late-Phase Core Melt Behavior</u>

The progression and consequences of severe accidents is highly dependent upon how and when the vessel lower head fails, and the amount, composition, and temperature of the molten core in the lower head at the time of failure. New research programs will be started that will be designed to allow us to get a much better understanding of late-phase melt progression, including lower head failure mode, within the next several years. The result of this research will help confirm resolution of the direct containment heating and BWR Mark I containment shell meltthrough issues.

3. <u>Code Development</u>

The current severe accident code development program involves the development of over a dozen codes to describe the various phenomena involved in, and phases of, a severe accident. For the most part, criteria have not been established regarding when development is finished. Under the revised SARP, care will be taken to ensure that criteria will be developed to assist the staff in deciding when a code is good enough. Further, all codes are to be properly documented and the level of quality assurance under which they were developed established. Future code development will be based on a structured approach such as the one currently being used for the thermal-hydraulic code development. This approach requires the developers to assess the current version of the code, identify potential deficiencies and proposed improvements, and justify why it is

necessary to develop a new version of the code to incorporate the proposed improvements (e.g., how is our perception of safety or risk expected to change, what is the likelihood the proposed improvements will in fact correct the deficiency). Employing this approach to code development along with criteria to help determine when the codes are "good enough," is expected to ensure a balanced approach to uncertainty reduction that will prevent the development of any one code to the point where its uncertainty has been reduced far below the uncertainties associated with the phenomena dominating the risk.

Finally, there are currently four in-vessel core melt progression codes under development. These are RELAP-SCDAP (INEL), TRAC-MELPROG (SANDIA), BWRSAR (ORNL), and MELCOR (SANDIA). The need to continue development of all of these codes is not clear. Each code will be critically examined, its capabilities and other attributes determined, and a decision will be made as to which codes should continue to be supported.

4. Long Term Research

Simultaneous with the near-term effort, the staff will continue to pursue a longer-term, broad-based research program aimed at reducing uncertainties in estimates of severe accident risk. Although resolution of issues associated with this broader-based research are not considered of immediate consequence as are the issues associated with the near-term goals of the SARP, nevertheless they are not insignificant contributors to risk uncertainty.

In pursuing this long-term research, the staff will apply many of the same elements of the near-term program. For example, code development will follow the approach previously outlined, and scaling methods and criteria will be applied to ensure fidelity of similitude in the experimental undertakings. In addition, the long-term programs will be periodically reviewed to ensure that their continuation remains essential to understanding severe accidents and reducing severe accident risk uncertainties. The operative criterion will be

whether there is a high likelihood that their continuation will result in a significant reduction in risk uncertainty.

Accordingly, for the long-term program some new programs are likely to be initiated and some existing ones will likely be either terminated or revised.

Resource Commitment:

The NRC resources associated with carrying out the revised SARP over the next three years including regulatory closure of those severe accident issues (of immediate priority that need to be addressed in order to achieve the near-term goals for regulatory closure as set forth in SECY-88-147) as summarized in this paper are consistent with the budget set forth in the current Five Year Plan approved by the Commission. This is approximately 10 FTE of staff effort and about $18M of contractor assistance in FY 1990, 10 FTE of staff effort and about $23.4M of contractor assistance for FY 1991 and 10 FTE of staff effort and about $23.8 of contractor assistance for FY 1992. Resource needs beyond FY 1992 for the long-term program will be submitted for Commission approval as part of the normal budgetary process. It should also be noted that some of the tasks identified in the attached revised SARP are being carried out under the accident management research program element and are presented in the revised SARP to provide a comprehensive description of the integrated program. There are no duplicative efforts in any of these elements.

Increases in funding levels to carry out the revised program are not proposed in the foreseeable future. In revising the program, in addition to identifying new or reoriented programs, the staff plans to either terminate or reduce funding of existing programs consistent with the revised program emphasis.

With regard to advanced LWRs, the staff is not aware of any features of the evolutionary LWRs under development that are unique with respect to severe accident analysis and require special research programs. As such, no funding has been allocated for work in this area. However, the staff will keep abreast with the development of the evolutionary LWRs and notify the Commission if the current perception

changes. The staff intends to advise the Commission periodically on the SARP progress toward achieving closure of the near-term severe accident issues and as appropriate, the status of the long-term program.

Scheduling: This paper is scheduled to be considered at an open meeting on May 2, 1989.

Victor Stello, Jr.
Executive Director
for Operations

Enclosures:
Appendices 1 and 2

Attachment 1 provides
Revised Severe Accident
Research Program

DISTRIBUTION:
Commissioners
OGC
OIA
GPA
REGIONAL OFFICES
EDO
ACRS
ACNW
ASLBP
ASLAP
SECY

Appendix 1

Hydrogen Transport and Combustion

In a memorandum dated September 13, 1988, from Victor Stello, Jr., to Eric S. Beckjord, "Actions Resulting from Review of the FY 1989-1993 Draft Five Year Plan," the staff was requested to provide EDO with a Commission Paper that justifies work on hydrogen transport and combustion beyond FY 1989. The memorandum also requested the staff to identify the cost of the proposed work. In a memorandum dated November 8, 1988, from Eric S. Beckjord to Victor Stello, Jr., "Commission Papers on Source Term Uncertainty Resolutions and Hydrogen Transport and Combustion Work," the staff indicated that a comprehensive SARP will address both the source term uncertainty resolution and hydrogen transport and combustion research. Attachment 1 to the Commission Paper transmits the revised SARP. The purpose of this appendix is to summarize the hydrogen transport and combustion issues.

Hydrogen Behavior for Degraded Core Accidents

The major concerns regarding hydrogen in LWRs are that the static or dynamic pressure loads from hydrogen combustion and detonation may breach containment integrity, or that safety-related equipment may be damaged as a result of either pressure loads or high temperatures. To assess the possible threat to containment and safety-related equipment, it is necessary to understand how hydrogen is transported and mixed within containment and to determine the likelihood of various modes of combustion. The hydrogen behavior issues have been extensively investigated since 1979, but several important areas of uncertainty remain:

(1) high-temperature/high steam concentration combustion and
(2) deflagration-to-detonation transition

Research programs on these issues of combustion have been sponsored in the U.S. by the NRC and the nuclear industry as well as in the international community. However, the combustion processes are sufficiently complex such that many aspects are still not well understood. The resulting uncertainties in the threat to containment integrity are unlikely to be reduced significantly by the existing research programs. The intent of the revised SARP (attachment 1) is to reduce these uncertainties. It should be noted that in some cases, reduced uncertainties are not required to make a near-term regulatory decision. Each category of uncertainty is discussed below.

High-Temperature Combustion

The Zeldorich-von Neumann-Doering (ZND) chemical kinetics theoretical model developed under NRC sponsorship predicts that

increasing temperature has a strong effect on the combustion and the detonation of off-stoichiometric hydrogen-air mixtures and on all hydrogen-air-steam mixtures. It is believed that the steam inerting effect is reduced greatly at elevated temperatures. There is a limited supporting data base for this phenomenon. Experiments are necessary to resolve the uncertainties associated with high temperatures and steam concentrations typical of those likely to be encountered in severe accident scenarios. It is also necessary to extend or develop more mechanistic models to predict, by either extrapolation or interpolation, the temperature sensitivity of hydrogen-air-steam mixtures that have not been tested.

The small-break LOCA and TMLB scenarios are two examples of high-temperature hydrogen-air-steam mixtures that may exist below the auto-ignition temperature (550 C minimum value for stoichiometric, hydrogen-air mixtures). There are two aspects of the high-temperature/high steam concentration combustion problem. The first is the injection of high-temperature hydrogen and steam mixtures that auto-ignite upon contact with pre-existing and premixed hydrogen-air-steam mixtures. The second aspect is the injection of hydrogen and steam mixtures at elevated temperatures that do not auto-ignite upon contact with pre-existing hydrogen-air-steam mixtures. This allows the possibility of a premixed condition to form and a subsequent deflagration or detonation. In both cases, the competition between chemical reaction rates and physical mixing rates will ultimately determine the ensuing combustion mode. Data are needed to determine the hydrogen-air-steam flammability limits and steam inerting criterion at elevated temperatures if reliable predictions are to be made as to the likelihood and the potential threat resulting from various combustion mode(s) for a wide range of accident conditions.

Therefore, the research approach is to determine hydrogen-air-steam flammability limits, volumetric oxidation (chemical reaction) rates, physical mixing rates, and the competition of these rates during the injection of high-temperature hydrogen and steam mixtures into cooler pre-existing and premixed hydrogen-air-steam mixtures.

A draft proposal is currently under consideration by the NRC that addresses feasibility; construction design, cost, and schedule; and an experimental test plan. It is our current estimate that if a facility is needed, it can be constructed by early FY 1990 and that experimental results can be generated by late FY 1991 or early FY 1992. Final data reduction and formal documentation should be available in late FY 1992.

Containment Loads for Detonations

Deflagration-to-Detonation Transition (DDT)

Direct initiation of a detonation would require a concentrated high-energy source for insensitive, steam-diluted mixtures. This is not considered a credible mechanism of initiation by almost all researchers. However, it is possible to initiate a flame with a low-energy source such as a spark or a glowplug, and the subsequent propagation through orifices and around obstacles such as pipes can result in flame acceleration that culminates in a transition to detonation.

The possibility of DDT in realistic and prototypic containment geometries and conditions needs to be resolved. The uncertainty in this area has increased because of recent experimental and theoretical results that indicate an increased likelihood of detonations at high temperatures and large steam fractions as discussed earlier. The current data base on flame acceleration and DDT suggests that the mixture composition, obstacles, and venting are all important factors. These uncertainties result from a lack of experimental data on flame acceleration and DDT for conditions that include the effects of steam dilution, elevated temperature, large-scale and prototypical obstacle types, and spacing.

At present, no reliable model exists to predict or extrapolate DDT results from small-scale experiments to containment scale. Reactor safety studies and fundamental combustion research is being carried out in the Federal Republic of Germany and Canada. These data along with data generated for space shuttle application will be applied to reducing the uncertainty associated with DDT as well as in assessing the potential threat of DDT to containment integrity. Specifically, this data base will then serve as the basis to improve or develop correlations and models to allow extrapolation of experimental results to reactor scale and accident conditions.

Appendix 2

Elements of the Near-Term SARP Related To Containment Challenges

As stated in the Commission paper, a program has been developed that attempts to obtain key data and information needed to make an informed decision on each issue. A more detailed description of this program is presented in the revised SARP (Attachment 1) and is summarized below.

(1) Depressurization and Direct Containment Heating

The conduct of the research enumerated under Issue 2, "Depressurization and Direct Containment Heating," would provide the technological base for assessing the primary containment integrity to the threats posed by DCH and in assessing mitigating strategies. In particular, executing the DCH research will enable the staff to gather the information needed to make an informed decision regarding the likelihood of a DCH threat to containment integrity. If such a threat exists, the staff will also be able to make recommendations on the feasibility and the effectiveness, including any adverse consequences, of depressurization on the reduction of the risk associated with DCH. The assessment would always include the current state of operator training and the dominant accident sequences anticipated. In addition, the proposed research will examine the effectiveness of water on quenching the melt and subcompartment effect on reducing the consequence of DCH.

(2) BWR Mark I Containment Shell Meltthrough

It is widely agreed that adding water under severe accident conditions is a desirable strategy to attempt to quench the debris and keep it coolable and hence may prevent Mark I containment shell meltthrough. Moreover, water can act as a scrubbing mechanism for fission products and could substantially reduce the radionuclides released even if containment shell meltthrough were to occur. Executing the research activities described in the revised SARP will enable the staff to determine the mode and timing of the RPV lower head failure and hence identify the various melt conditions, i.e., quantity, composition, and temperature. The staff will also be able to better quantify the effect of water on the drywell floor on melt spreading and on the heat transfer between the melt and the containment shell.

(3) BWR Mark III and PWR Ice Condenser Containment Hydrogen Combustion

The risk-significant potential of early containment failure by hydrogen burns and/or hydrogen detonations will be assessed using data and models provided by the SARP, namely, the HECTR and HMS-BURN, codes which have been validated. Relative to hydrogen detonation issues, the recently developed model ZND will be used for a prediction of the detonability of a given mixture of hydrogen, air, carbon oxides, and steam.

(4) BWR Mark II and Mark III Fuel-Coolant Interactions

A potential mode of containment 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 exists direct pathways between the area underneath the vessel and the water from the suppression pool.

The research to be conducted will enable the staff to address some well-defined questions dealing with key parts of ex-vessel molten core-water interactions and the potential that such interactions could lead to failure of the containment at a major penetration.

REVISED SEVERE ACCIDENT RESEARCH PROGRAM PLAN

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

April 1989

SEVERE ACCIDENT RESEARCH PROGRAM PLAN

Revision 1

APRIL 1989

Severe Accident Research Program Plan

I. INTRODUCTION

In a memorandum dated July 8, 1988, the Director of the Office of Nuclear Regulatory Research charged the Division of Reactor and Plant Systems (now the Division of Systems Research) to develop "a detailed plan identifying individual goals, discrete products, and anticipated schedules for the Severe Accident Research Program (SARP) that clearly demonstrates the implementation of the SARP portion of the staff's Integration Plan for Closure of Severe Accident Issues (Integration Plan) (SECY-88-147). [Preparing such a plan will require] a detailed review of the current activity of the SARP against the framework of the Integration Plan and the detailed activities and schedules of the other five elements of the Integration Plan, [together with] an assessment of what redirecting/reprioritizing/rescheduling within the SARP may be needed to fulfill its portion of the Integration Plan and how and when such can be accomplished." This revised Severe Accident Research Program plan reflects the evolution of the severe accident research program and was developed in response to that July 8 memorandum. It will be used to guide the formulation and conduct of severe accident research, addressing in the near-term those issues pertinent to the implementation of the Integration Plan: viz., Individual Plant Examination (IPE) and Containment Performance Improvement (CPI) activities. The plan also will be

used to guide the formulation and conduct of the program of long-term research needed to support the NRC's accident management activities and to confirm the Commission regulatory decision on severe accident issues. The plan provides a description of the general approach to the needed research. Naturally, many of the details associated with specific experiments or analyses have not yet been developed and can only be developed after the specific research is identified. Therefore, for each near-term issue, a detailed research plan will be developed, describing each experimental and analytical program and how these programs fit together and lead to a resolution of the issue. The plan may also give an impression that we are just now entering the arena of severe accident research; the facts are quite the contrary. A tremendous amount of severe accident research now exists and this plan is intended as a needed step to the critical and focused re-examination of our further needs and the regulatory questions involved.

This plan is organized as follows. Section II presents the goals of the SARP and a sketch of its structure. In Section III, the work directed at achievement of the near-term goals articulated in Section II is described. Research dealing with the longer-term goals is covered in Section IV. Appendix A to this plan provides background information on the existing SARP and the relationship of SARP to other elements of SECY-88-147, and Appendix B is an

exposition of the role of scaling in the SARP. Estimated costs and significant milestones for implementing the revised SARP are presented in Table 1.

II. GOALS

In developing the revised Severe Accident Research Program (SARP) plan, the staff recognized that the overall goal is to reduce the uncertainties in source term sufficiently to enable the staff to make regulatory decisions on severe accident issues. However, the staff also recognized that for some issues it may not be practical to attempt to reduce uncertainties further, and regulatory decisions or conclusions will have to be made with full awareness of existing uncertainties.

As the Severe Accident Research Program represented in this plan is a goals-oriented program, it is critical that such goals be clearly stated at the outset. The staff expects to use safety goal policy and objectives, as appropriate, in determining what potential improvements in the technological base are needed for closure of severe accidents and in order to ensure that the safety of existing plants are reasonably consistent with the safety goals . The goals of the revised SARP are as follows:

1. Provide the technological base for assessing containment performance over the range of risk-significant core melt events.

2. Develop the capability to evaluate the efficacy of generic containment performance improvements.

3. Provide an understanding of the range of phenomena exhibited by severe accidents that includes the impacts of generic accident management schemes.

4. Develop improved methods for assessing fission product behavior and availability for release in the event of containment failure at various phases of severe accident sequences.

Goals "1" and "2" above are primarily near-term, while goals "3" and "4" are longer-term objectives that aim for additional depth of understanding of both accident evolution and final consequences. These goals are consistent with NRC's Integration Plan for Closure of Severe Accident Issues (SECY- 88-147), and the guidance provided in Generic Letter 88-20 for Individual Plant Examinations. It should be noted that there are not two SARPs, a near-term and a long-term. Rather the revised SARP draws from and focuses specific tasks of the continuing program of severe accident research experiments and analyses to address the near-term implementation of SECY-88-147.

A. Near-Term Work

Subsequent to the Three Mile Island Unit 2 accident in 1979, the U.S. Nuclear Regulatory Commission undertook a broadly based research program to develop an understanding of severe accident behavior. Over the past decade, major experimental and model (code) development programs have been performed that provide a greatly improved understanding of severe accident phenomena, reflected in an ability to model those phenomena. Now, as the Commission is preparing to close on severe accident regulatory questions, the immediate priority of the NRC's SARP is to support the closure process. That process[1] is displayed schematically in Figure 1. Hence over the next 3 years a significant portion of the severe accident research effort will be directed to issues that relate to the dominant topic area of the Integration Plan, viz., the accident sequences that lead to early containment failure: direct containment heating (DCH), BWR Mark I containment shell meltthrough, molten fuel-coolant interactions in BWR Mark II and Mark III containments, and hydrogen detonation in BWR Mark III and

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[1] In SECY-88-147, the staff identified the steps that each licensee is expected to take to achieve closure on severe accident for its plant, namely:

1 - Completion of the individual plant examination (IPE) and identification of potential improvements
2 - Development and implementation of a framework for an accident management program that can accommodate new information as it is developed, and
3 - Implementation of any Generic Containment Performance Improvements with respect to severe accidents.

Mark III containments, and hydrogen detonation in BWR Mark III and PWR ice condenser containments. Note that the hydrogen behavior issues for accidents in which the core is degraded but is not into the severe accident domain have been extensively investigated which resulted in the resolution of USI A-48 "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment". The additional hydrogen research discussed in this revised SARP is intended to address the hydrogen threat from severely damaged cores and will serve to reduce uncertainties in the threat to containment integrity resulting from various hydrogen combustion mode(s) for a wide range of accident conditions. In the near-term, severe accident research will also support the resolution of broader, more general questions regarding the response of various containment designs to severe accidents. Task descriptions have been developed for the near-term work on these issues and are presented in Section III of this plan. The majority of the near-term severe accident issues are considered relevant to evolutionary LWRs currently under review by the NRC. Therefore, these near-term efforts are equally important to evolutionary LWRs. We are currently not aware of any features of the evolutionary LWRs under development that are unique with respect to severe accidents and require special research programs. As such, no funding has been allocated for work in this area. However, we will keep abreast with the development of these evolutionary LWRs and notify the Commission if our current perception changes.

B. Long-Term Work

Achievement of the longer-term goals demands a broadly based severe accident research program. To be responsive to regulatory needs, particularly confirmation of closure of severe accident issues, that program must both explore severe accident phenomena and develop the methodologies appropriate to quantitative assessment of severe accident risks. The plan to guide such a program must reflect a balance among the risk importance of phenomena being investigated, the desire for technical completeness, the expected likelihood that the research will result in a significant reduction in the uncertainty of risk, and the need to promote the maintenance of technical expertise and analytical capabilities. The features of the SARP that address the long-term goals are presented in Section IV of this plan. Figure 2 depicts the elements of this revised SARP and the relationships among them.

C. Results of Previous Work

Review of information available from previous and ongoing research including that sponsored by U.S. industry or foreign organizations is being undertaken by the staff and expected to be completed by end of FY1989. The purpose of this review is to identify and redirect the research program as appropriate to ensure that the needed information to reach closure or resolution of severe accident issues important for regulatory decisions is developed.

This review will also ensure that the various research projects are consistent and well integrated among themselves, have a common goal of ultimately leading to closure of severe accident issues, and that the long-term confirmatory research is properly focused.

FIGURE 1
SEVERE ACCIDENT PROGRAM - SCHEMATIC

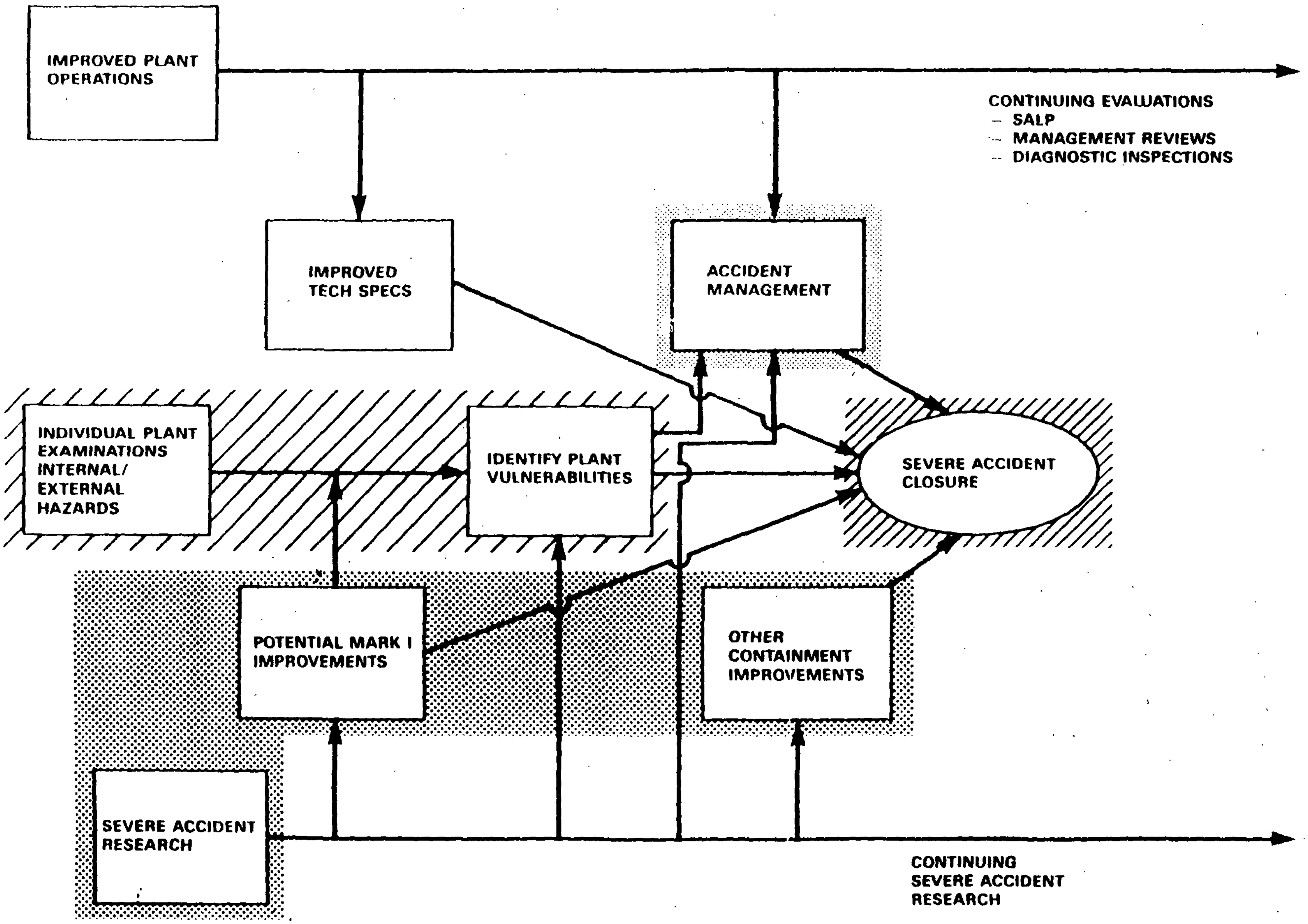
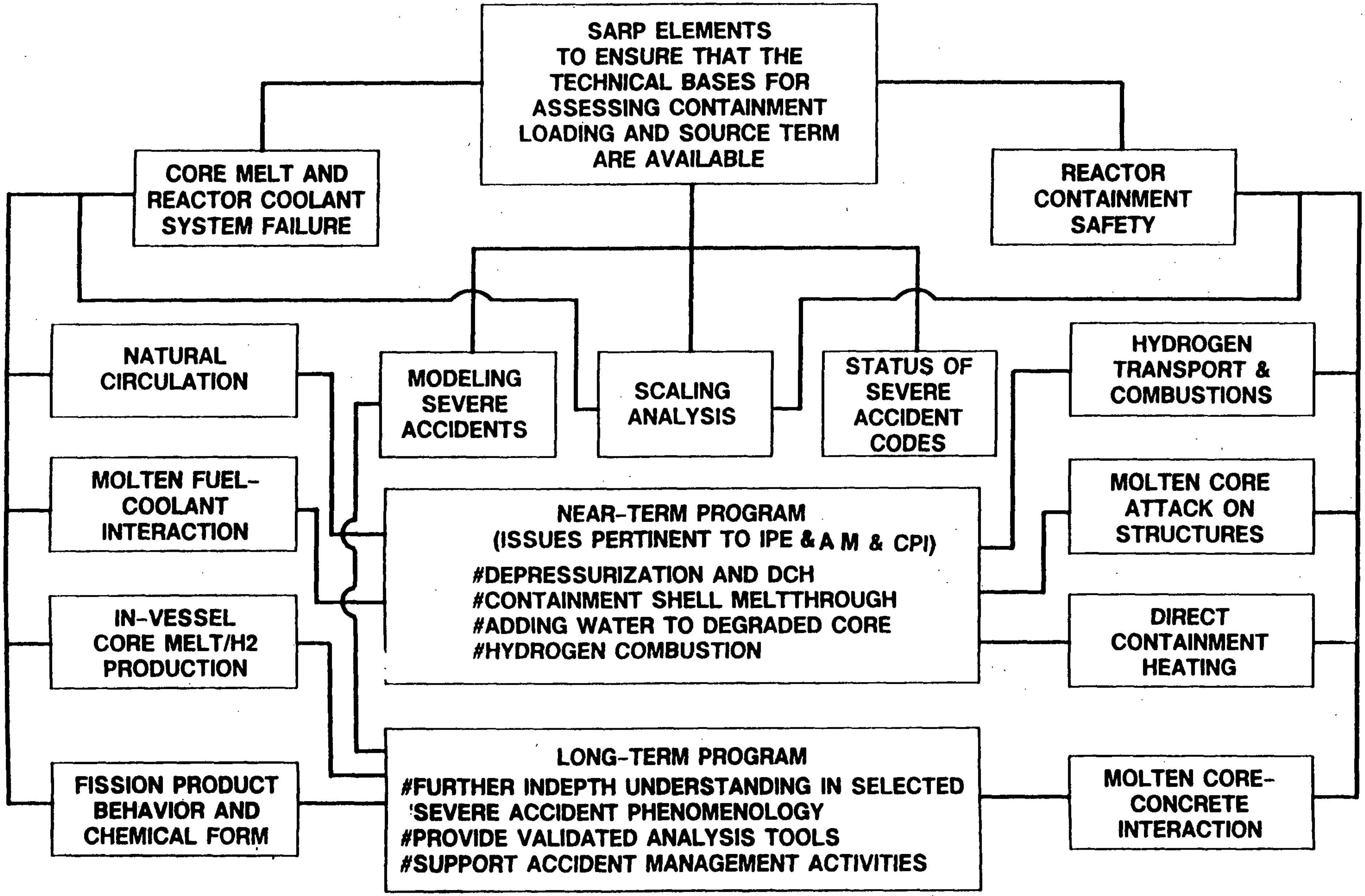


FIGURE 2
REVISED SARP
NEAR- AND LONG-TERM PROGRAMS



III. MEETING NEAR-TERM GOALS

This section of the SARP plan presents those issues of immediate priority that need to be and can be addressed in the next few years in order to achieve the near-term goals set forth in Section II. These are the issues that are pertinent to the NRC's programs of individual plant examinations and containment performance improvements (direct containment heating (DCH) and hydrogen transport and combustion, issue 2, and BWR Mark I containment shell meltthrough, issue 3). The research to be performed to address those issues is also presented. This research is primarily directed to assessing containment performance under severe accident conditions and is constructed to enable as definitive a judgment as is possible to be made about the potential threats and likelihood of early containment failure in the event of a severe accident given the present paucity of (or sparse) data describing core melt progression. Also included in this section is a discussion of the issue of scaling of severe accident experiments and analyses (issue 1). Obviously, the question of verisimilitude in severe accident research is critical, and scaling is one element of the same. Whether the correct physical and chemical processes are being investigated is another and is appropriately dealt with on a case-by-case basis. But scaling as an issue is common to all areas of severe accident research since full-scale experiments simply are not feasible. For this reason, a systematic examination of scaling of severe accident experiments and analyses is included among the key elements of the near-term research. This element is

not intended to constrain closure expected of the other near-term issues herein but is expected to be of benefit to decisions made on recommending further experiments over the near- and long-term periods addressed in this plan.

In addition, this section contains a discussion of the issue of molten fuel-coolant interactions (issue 4) and a discussion of the issue of the state and use of severe accident codes (issue 5). The molten fuel-coolant interaction issue is included here because it is directly relevant to considerations of intentional reactor vessel depressurization to cope with DCH, as well as low reactor coolant system pressure accident sequences in general. Specifically, the question of what happens when water is added to molten fuel (in-vessel or ex-vessel) appears to be significant to understanding accident management and some severe accident sequences. How to answer that question is not immediately obvious, and some initial work needs to be done to allow development of a rational approach to the research. As for the issue of codes, it is recognized that code development is integral to the achievement of long-term goals, since the understanding of severe accident phenomenology ultimately is reflected in those calculational tools. However, achieving the near-term goals will necessitate reasoned judgments based essentially upon our present understanding of severe accidents, and an assessment of just how well present codes describes the key behavior of severe accidents. The confidence that can be placed in the severe accident codes also directly

translates into confidence that can be vested in the near-term judgments about containment performance. For this reason, the code assessment issue is included in this section.

Severe Accident Phenomena - Core Melt Accidents

Before the issues and research tasks to address them are discussed, a general description of the important phenomena associated with core melt accidents to be dealt with in the near-term is presented. This is done to provide a perspective and context for the issues and tasks.

In light water reactors, core melt accidents can occur either at low reactor coolant system pressure or at high reactor coolant system pressure. At low pressures, the core heats up more or less independently from the rest of the primary system, and the process is accelerated by metal-water reactions. At high pressures, energy redistribution by steam from natural circulation may be significant, transferring core heat to the remainder of the system, including upper vessel internals and, for PWRs, steam generators. This transfer of energy affects the progress of the accident in several ways. It delays somewhat the onset of gross core degradation. It affects the steam availability in the metal-water reaction process, tending to limit hydrogen generation. It affects radionuclide release and transport within the primary system. Perhaps most importantly, it also raises the possibility of

competing high temperature and stress related of failures of the primary system pressure boundary (e.g., RCS piping, steam generator tubes) in PWRs with attendant depressurization, hence avoiding the possible ejection of molten core material from the reactor vessel while under high pressure. Current understanding suggests that for BWRs natural circulation is restricted by the fuel assembly wall channel boxes. Although natural circulation may affect the core melt progression, failure of the reactor coolant system due to excessive heating from high-temperature naturally circulating gases would not be likely. However, because of the enhanced reliability of the Automatic Depressurization System (ADS) suggested from the containment performance improvement program, high-pressure scenarios are not likely to be dominant contributors to early containment failure in risk assessment of BWRs. Hence, high-pressure scenarios in BWRs are not being addressed at this time.

Although the failure of the reactor coolant system (RCS) by natural circulation appears to be possible, the ability to predict its occurrence has not been established to the point that confident conclusions regarding the likelihood of RCS failure can be made. Moreover, while analyses to determine the extent to which natural circulation occurred in the TMI-2 accident are under way, it must be noted, however, that no evidence of high temperatures in the range of interest to cause failures in the primary system components has yet been seen. Thus, competing RCS pressure boundary

failures from high temperature cannot be presumed with a high degree of confidence, and scenarios involving gross core degradation or core melt at high or intermediate RCS pressure must be considered. Since such high pressure PWR scenarios have the potential to produce ejections of molten core material into the lower containment following lower vessel head failure, a variety of concerns regarding the possibility of containment overpressurization have been raised, in particular, direct containment heating. In fact, the high-pressure scenario seems to influence risk in terms of the variety of mechanisms with potential to overpressurize the containment and thus importantly contributes to the uncertainties in predicted risk. In addition, core melt accidents at low RCS pressures that lead to early containment failure have been identified as potentially important to risk, in particular, BWR Mark I containment shell meltthrough.

The following observations are intended to supplement the above general perspectives:

1. At this time, competing primary boundary failure locations, size, and timing are highly uncertain, particularly considering the role of relocated fission products, and the possibility of loss of integrity of steam generator tubes.

2. Core slump at high pressures may yield significant fuel-coolant mixing. Triggering of massive steam explosions at such high pressures is generally considered highly unlikely by the experts, but the potential driving forces associated with such core slump interactions may not be altogether ignorable in the process of understanding fission product relocation and the potential for altering the accident progression.

3. Natural circulation may for example, cause a weakening of the upper internals and upper vessel head (e.g., seals), with concomitant result that competing relief paths may be formed that influence both the fission product relocation and the potential for direct containment heating.

4. Lower vessel head specific failure modes and the timing remain highly uncertain. Local failures of instrument guide tubes could occur. However, melt attack of the vessel lower head forging (in the absence of local penetrations) could lead either to a local failure of the forging or to creep rupture of a large portion of the lower vessel head when heated to 600 to 700C. Large failures at high reactor coolant system pressure can potentially create thrust loads and threaten failure for example, through overstress or tearing at piping junctures.

5. At high pressure, a large coherent failure of the lower head would likely be accompanied by prompt expulsion of melt, steam, and hydrogen. High-pressure expulsion has been postulated to lead to direct heating of the containment (DCH) atmosphere and an increased potential for hydrogen combustion and perhaps even detonation.

6. The ex-vessel course of accidents in which reactor vessel breach occurs with low pressure in the reactor coolant system is highly dependent on the mass, thermal and chemical properties of the melt upon exit from the reactor pressure vessel. Direct attack of the Mark I containment shell; molten fuel-coolant interactions in the BWR Mark II and Mark III containments and hydrogen combustion in BWR Mark III containments are examples of where low system pressure scenarios are likely to dominate the probability of a severe accident but also give rise to early containment failure threats.

Issue 1 - Scaling

Full-size severe accident experiments are impractical or simply not feasible, for many reasons, particularly costs. Hence severe accident research must rely on small-scale experiments that are carefully crafted so that the conclusions drawn from those experiments about the phenomenology of severe accidents can be appropriately applied to predict severe accident progression in typical nuclear power reactors. The confidence in this application

is greatly enhanced through scaling analyses. A key element of the SARP, therefore, is a scaling methodology for severe accident experiments. In those cases where an acceptable methodology can be developed, it will be applied to relevant experimental programs. As necessary, recommendations regarding revisions to experimental programs will be made and needs for new facilities will be addressed in terms of the regulatory value in pursuing this research. A useful scaling methodology should be capable of providing the following:

1. A scaling rationale and similarity criteria.

2. A procedure for conducting comprehensive review of facility design, test specifications, and results.

3. A measure or index to indicate the applicability of correlations or models based on test data from sub-scale facilities to full-scale nuclear power plant conditions.

4. Quantification of the effects of scale distortion.

5. Quantification of the effects associated with extrapolating correlations or models beyond their data base. This information is needed for quantifying code uncertainty.

Although scaling analyses have been performed for some of the severe accident research experiments, there as yet is no systematic application of a scaling methodology that would allow confident extrapolation of observations made in intentionally scaled experiments to severe reactor accidents. The work to be done is to first determine whether a scaling rationale and similarity criteria for severe accidents research can be developed to provide assurance that the correlations or models developed can be properly scaled to reactor condition. Second, if such a methodology is developed, a systematic application of the same will be made to appropriate experiments and computer codes carried out under the SARP.

Task 1 Develop and Apply a General Severe Accident Scaling Methodology

Research Approach:

A review of scaling methods will be made. This review will be of scaling methods in general and methods used in the experiments of the SARP to date. From examination of the nature of the phenomena being studied, the experiments being conducted to study those phenomena, and the review above, scaling methodology will be developed for the SARP. The efficacy of the methodology will be tested initially by applying it to the direct containment heating experiments that have been conducted in the SARP to date.

This task will not address directly whether the correct phenomena are being investigated in a particular experiment - that is done for each task throughout the SARP - but rather whether the experiment will indeed yield information about the phenomena being investigated relevant to behavior at the scale of an actual reactor.

Useables and Use:

To the degree that this work is successful, the SARP will have a scaling methodology that can be applied to the experiments of the SARP to provide a more rigorous and systematic link between experiment and accident phenomena than is present today. The application of the methodology will help in the planning and conduct of the experiments directed at the longer-term goals. In the near-term, application of the methodology should help establish the level of confidence that can be placed in the present understanding of containment performance from scaled experiments and code analyses performed to date.

Issue 2 - Depressurization and DCH

Present understanding of the in-vessel progression of a PWR core degradation accident is limited but suggests the formation of melting-solidifying fronts as the molten core materials "candle down" into the lower, colder portions of the fuel bundles and solidify. This "frozen crust" of previously molten core material may form in a layer that inhibits steam flow from the lower plenum. Unsupported fuel pellet stacks may collapse onto the crust and melt. As this process proceeds, a crucible-shaped crust supporting a molten mass of core material may form in the central region of the core. Failure of this crust at a radial location ("sidewall") is expected from considering thermal convection of the melt within the crucible. In fact, this is what was concluded to have occurred at TMI-2 upon examination of the reactor core. Upon crust failure, molten core material would flow downward, possibly ablating through the core shroud and flowing onto the flow distribution plate, eventually reaching the lower head. However, very little

experimental information on this process exists, and computer code predictions are at best uncertain. Hence, prior to dismissing crust failure on the bottom portion of the crucible (not a plausible failure mode), this issue will be assessed. The main difference between these two failure modes seems to be the amount of molten material that pours into the lower head and is ultimately available to be released to the containment. Our current understanding indicates anywhere from around 25 to 60 percent of the core could relocate to the bottom head upon crucible failure. Because of the difficulty of conducting appropriate tests to better characterize these phenomena, questions associated with core relocation are likely to remain highly uncertain in the near-term, although modeling of these processes will continue under the SARP. Thus, the question of lower head failure mode can be approached by considering a sufficiently broad range of debris quantities, composition, temperatures, and relocation rates. The TMI lower head inspection plays an important role in understanding the relocation process and the threat to the lower head integrity. On the other hand, depressurization of the RCS, either by structural failure of the pressure boundary (e.g., piping, steam generator tubes) resulting from natural circulation, or deliberate depressurization by operator action, may alleviate concerns over DCH. In this regard, any analysis with respect to depressurization would be assessed in terms of the current state of operator training including information available to the operator (instrumentation) and it would include the dominant accident sequences anticipated for a given plant.

As part of the development of the Reactor Risk Reference Document (NUREG-1150), the DCH issue was presented to an expert panel that included experienced severe accident analysts. Monte Carlo analysis was performed that integrated all of the pertinent phenomena. The resultant likelihood of containment failure was found to be small and the chance that containment would survive the loads associated with DCH is much greater than had been originally estimated and reported in the first draft of NUREG-1150. The experts based their judgments, including the likelihood of RCS failure by natural circulation leading to RCS depressurization, on the current body of research evidence.

Consistent with the above, in the near-term the SARP will address the DCH issue by a two-pronged approach:

1. Before we can rigorously conclude on the benefit to intentionally depressurize the primary systems of PWRs, an improved understanding of the challenge to containment integrity must be obtained. The existing research efforts will be critically examined with respect to the following:
    a. What is the nature and character of the information most likely to be provided on high-pressure core melt accidents in the next several years? Are continued commitment of time and resources likely to result in a commensurate advancement of understanding of DCH phenomena?

b. Are the existing research programs addressing DCH optimized with respect to providing the best information for closure of the DCH question in the near-term?

2. The efficacy of PWR depressurization will be examined. In particular, both beneficial and detrimental aspects will be explored. Review of relevant emergency procedures, dominant accident scenarios, and timing of required operator actions to depressurize also would be included in this examination.

Using the results from 1 and 2 above, an updated estimate of the risk associated with high-pressure core melt accidents will be made, along with an estimate of the risk reduction that would be obtained by intentional depressurization, taking into account any increase in risk associated with depressurization. Based on a careful weighing of the net benefits of depressurization, a recommended course of action will be proposed.

Some of the key questions that the SARP will attempt to answer in the near-term in order to make the above determinations are:

1. What is the likelihood that the RCS will fail by natural circulation prior to lower head failure? If so, are these failures of much lesser concern to overall risk? (Task 2.2)

2. Is there a low-pressure cutoff below which there is no DCH threat? (Task 2.3)

3. If so, will this pressure be reached by natural circulation-induced failure of the RCS, or through operator action to depressurize, or both? (Tasks 2.1 and 2.5)

4. If operator action is necessary, is there time available for this action? Given that operator action is to be taken what are the hardware and procedure specifications that would enable successful depressurization actions. (Tasks 2.3 and 2.4)

5. Are there adverse consequences to early depressurization? If so, what are these and their potential significance in terms of causing either earlier core damage or earlier containment failure? (Tasks 2.4 and 2.9)

6. What is the nature of the DCH threat, and what mechanisms (e.g., sprays) and configurations exist ex-vessel that will mitigate or eliminate it? (Tasks 2.4-2.8)

Logically an answer to this last question should precede the rest. However, even with DCH apparently presenting much less of a risk than previously believed, there is not yet sufficient confidence in this belief that it warrants delay in seeking answers from the other relevant tasks, especially in light of the 3-year horizon of the Integration Plan.

Task 2.1 <u>Evaluate current research program addressing high pressure melt ejection challenges to containments.</u>

<u>Research Approach:</u>

Current research programs, both experimental and model development programs related to DCH and hydrogen production during high-pressure melt ejection (HPME), will be reviewed and assessed to quantify the current level of uncertainties to determine if any of the questions raised in the following tasks have already been answered. In addition, the assessment will determine the nature and character of information likely to result from these programs over the next 3 years. Progress made to date, current research topics, major uncertainties, and methods being used, both experimental and analytical, will be examined. This task will be carried out prior to the planning and conduct of any new experiments or analyses.

<u>Useables and Use:</u>

Based upon the results of the above assessment, a judgment will be made as to whether a significant increase in understanding the threats to containment from high RCS pressure accidents will be achieved in the next 3 years. In particular, the question of whether a significant reduction in uncertainties will be achieved only with a significant additional expenditure of time and resources will be addressed.

Task 2.2 <u>Assess the likelihood of RCS structural failure by natural circulation</u>.

<u>Research Approach:</u>

Calculations of RCS failure by natural circulation will be scrutinized and alternative models and assumptions examined. Comparisons of calculations with available experiments (Westinghouse) will be made to identify potential weaknesses in both experiments and models, particularly with regard to scaling. An assessment will be made on why there were only minor natural circulation effects in the upper vessel structure and none in the hot leg piping at TMI.

High RCS pressure PWR sequences will be selected by initiating event, characteristics of the early stages of the sequence, and potential for evolving to intermediate or low RCS pressure sequences (e.g., potential failure of a safety valve in a station blackout during core heat-up). This selection should include loop seal clearing and pump seal failures. Codes predicting energy redistribution due to natural circulation will be applied to these sequences and will be verified, validated, and examined for underlying assumptions that could affect the results. Analyses and comparisons of the selected sequences as to likelihood, location, and time of failure of the RCS boundary will be made. The effects of fission product deposition and revaporization as an additional source of thermal energy during natural circulation will be considered.

<u>Useables and Use:</u>

The result of this task will be an estimate of the likelihood of RCS failure from natural circulation in high RCS pressure scenarios for typical primary system geometries. It is possible that this item will alleviate concern with the threat of direct containment heating.

Task 2.3: <u>Investigate the influence of cavity and containment compartment structures on DCH, low-pressure cutoff for DCH, and hydrogen production from HPME.</u>

<u>Research Approach:</u>

The scoping calculations performed for NUREG-1150 to estimate the DCH load from the dispersal of melt into the containment correlated DCH load with melt quantity, ejection pressure, and ejection rate. The results of this analysis identified the key unknowns in the prediction of DCH containment loads from melt dispersal

that need to be determined by experiment, particularly, the effect of lower containment compartment geometries; the effect of water (co-dispersal with the corium or added by spray) on reducing the magnitude of DCH pressure; and the production of hydrogen by HPME.

Experiments at different scales (using existing facilities insofar as possible) with high temperature, thermitically generated melt and steam as a pressurizing fluid will be performed. Measured quantities will be identified based on the scaling analysis of Task 1. Variables believed at this time to affect the magnitude of the challenge to the containment are melt mass expelled from the cavity and its distribution within the containment, containment atmosphere pressure and temperature, melt metal content of the dispersed melt heat transfer rates to structures, and hydrogen production in the dispersing melt-steam mixture, and its transport and combustion in the containment.

Useables and Use:

Direct comparisons of containment pressurization observed in tests at different scales (existing facilities) provides insight on the dominant factors affecting scale up of the results. The data may be used to estimate melt entrainment as a function of quantity of melt and ejection rate. The melt entrainment will then be used to calculate at what pressure DCH from core ejection does not threaten the containment integrity (low-pressure cutoff).

Task 2.4: Explore the feasibility of intentional RCS depressurization over the relevant spectra of PWR severe accident scenarios.

Research Approach:

The overall framework for analyses should be based on a comprehensive treatment of representative scenarios, including possible perturbations by operator actions (based on the current state of operator training, Emergency Procedures or Functional Restoration Guidelines, and timing for when the operator is required to depressurize) and equipment failures that may have a bearing on the issues addressed here. The research should consider, but not be limited to, the normally available power-operated relief valves (PORVs) as a depressurization device. One such strategy is to decrease the system pressure sufficiently to allow the accumulators to dump prior to significant cladding oxidation, so that gross fuel degradation is approached at a pressure that will not threaten the containment

integrity should the vessel lower head fail. The approach is essentially analytic, using thermal-hydraulic codes such as RELAP5 or TRAC for the most part, although selective use of core degradation codes may also prove useful. The use of models in these codes that have not been tested in circumstances peculiar to this application, e.g., reflooding of a highly overheated core would be scrutinized to facilitate judgment as to the meaningfulness of the results. The research will examine both beneficial and detrimental effects of early intentional depressurization.

Useables and Use:

The result of this work will be a defined relationship between depressurization strategies and final system pressure at the time of gross fuel degradation, or the time of lower head failure, and an identification of any detrimental effects of intentional early RCS depressurization.

Task 2.5: Determine the mode of bottom head failure of the reactor pressure vessel in a high RCS pressure sequence.

Research Approach:

Assess and determine the likely modes of PWR pressure vessel bottom head failure considering the reasonable range (Task 2.6) of quantities, composition, and timings of core melt arriving on the bottom head. From the above assessment, identify those experiments that may need to be conducted to determine the values of parameters that are keys to the assessments and/or confirm those assessments. Application of the scaling methodology (Task 1) will be made in devising and interpreting the results of the experiments.

The TMI lower head inspection plays an important role in understanding why vessel failure did not occur for this accident. It is not sufficient to look for damage to the lower head. It is essential that the debris on the lower head be carefully characterized in order to interpret the observed condition of the lower head.

Useables and Use:

An understanding of the sensitivity of reactor pressure vessel (RPV) lower head failure mode to the quantity, composition, and timing of arrival of molten corium on the lower head. This understanding may be quantified in

the form of system level codes that predict the mode and timing of reactor vessel failure in a manner consistent with the predicted melt progression and thermal-mechanical loading of the RPV. The results of this task will contribute to the work of Task 2.3.

Task 2.6: <u>Determine the likely range of quantity, composition, and timing of molten core material arriving on the bottom head of a PWR during a core melt accident at high and intermediate RCS pressures.</u>

<u>Research Approach:</u>

Review and assess existing experiments and associated analyses to ensure that the analysis tools represent the important phenomena. A series of core degradation and melt relocation calculations will then be made using boundary conditions and initial conditions that span the range of core melt sequences. For each set of conditions considered, the calculations should yield the quantity, composition (proportion of metals and oxides), temperatures, and timing (rate of arrival) of melt reaching the bottom head. Based on these calculations, best estimates will be made on the characteristics of melt reaching the bottom head of a PWR in high and medium RCS pressure core melt accidents for assessing DCH. Low RCS pressure conditions will be used in evaluating the potential effects of depressurization, including the effect of molten fuel-coolant interaction on core melt progression.

<u>Useables and Use:</u>

The result of this work will be the most likely characteristics of melt to be considered in addressing PWR bottom head failure (Task 2.5).

Task 2.7: <u>Use the results obtained in Tasks 2.3 to upgrade DCH models.</u>

<u>Research Approach:</u>

This task should proceed only if the results from the scaling program (Task 1) and Task 2.3 indicate that further analytical developments are necessary for calculation of DCH loading and closure of this issue. Analysis may be used for extracting detailed information from experiments that can then be used to develop models that sufficiently represent plant-specific geometries. The suitability of the models will be affirmed by

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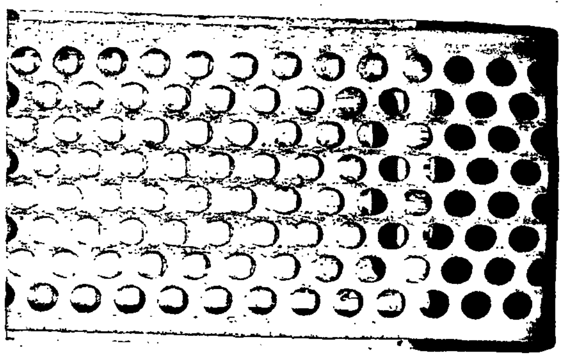

t/benefit analysis.

2.1 through 2.7, and the results of
3 dealing with fuel-coolant
system pressures, will be considered
ses of intentional PWR system
a mechanism to avoid DCH. The
cused on assessing the efficacy of
bring about a net benefit to safety
nelt scenarios (e.g., station blackout
onditions associated with loss of
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in risk of early containment failure
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ther factors to be considered include
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dicate that the RCS will fail and
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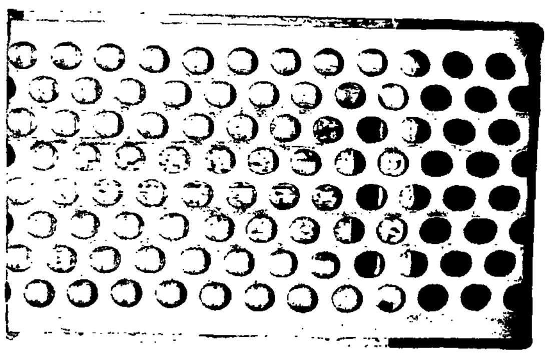


task will provide additional bases
mend for or against requiring
PWRs during core melt accidents to

Issue 3 - <u>BWR Mark I Containment Shell Meltthrough</u>

An accident sequence leading to early containment failure has been postulated for BWR Mark I containments. This sequence involves the deposition of molten core onto the drywell concrete floor at the time of vessel breach, the subsequent spread of the melt to, and its contact with, the drywell steel wall that is the containment boundary, ultimately causing failure of the wall. Some experimental work has suggested that the addition of water in the drywell could provide a mitigative effect by causing fragmentation and quenching of the melt before it reaches the wall or by providing significant cooling of the melt layer. The heat transfer processes from the melt to the wall, the melt-concrete interactions, and the heat removal processes from the melt and from the wall are not completely understood, although some work has been done at Sandia National Laboratories on heat transfer from steel melts to steel structures and on melt-concrete interactions. There have also been experiments on bubbling heat transfer and on melt spreading at Brookhaven National Laboratory, using simulants, and experiments related to melt spreading and quenching have been conducted by Fauske and Associated, Inc. In addressing this issue in the near-term, the literature will be carefully reviewed to clearly identify the critical questions to be answered with respect to Mark I shell failure and the capability of the existing data to answer those questions before any new research efforts are begun.

One such question that has been identified is the sensitivity of the time and mode of reactor vessel lower head breach to the quantity, composition, and timing of arrival of molten core on the bottom head of a BWR. The core melt of a BWR may proceed much differently from that of a PWR. For example, one possibility that has been suggested is that the channel box and control blade geometry of a BWR suggests that stable crusts supporting the melt may not form (as in TMI) and that molten fuel will more or less continuously flow onto the lower head. Further, the lower head volume of a BWR is larger than that of a PWR and is densely occupied by the control rod drive structures, and this may result in a different failure process (e.g., less coherent failure) of the bottom head from that of a PWR.

It has been postulated that a BWR core melt progression proceeds by gradually accumulating fully quenched debris in the spaces between the control rod guides, in the lower head. Another postulated progression is similar to the TMI-2 accident with melt relocating into the water-filled lower head. In either case, following dryout, debris heatup will be accompanied by lower head heatup, and eventual failure of the bottom head. At this time, the composition and characteristics of the debris arriving on the drywell floor beneath the reactor vessel is uncertain. The debris could range from mostly solid oxidic with low superheat to mostly molten metallic with high superheat. The nature of the debris exiting the failed bottom head of the BWR can determine the subsequent course

of the accident. However, the bottom head itself may play a significant role in determining the characteristics of that debris. In light of the above, the SARP will address the questions of how sensitive (or insensitive) to the quantity, composition, temperature, and timing of melt arrival is the failure mode of the bottom head of a BWR and to what extent are the hydrodynamic and thermal properties of melt arriving on the drywell floor determined by the mode of bottom head failure.

The BWR Mark I containment shell meltthrough issue, like the DCH issue, was presented to a panel of experienced severe accident analysts who based their judgment on the current body of evidence. The panel was equally divided as to whether failure would occur or would not. When the drywell floor is water covered, the experts were more confident that failure would be avoided. Similarly, when the floor is dry and superheat is high, the experts had less confidence that containment shell would remain intact. The panel concluded that the degree of belief regarding containment shell meltthrough is 0.33 or 0.87; the lowest failure probability corresponds to cases in which water is assumed to cover the drywell floor; and the highest corresponds to cases in which the drywell floor is dry and debris flow rate, debris superheat, and debris unoxidized metal content are all high. Nevertheless, it was generally agreed that the composition and temperature of the debris exiting the reactor pressure vessel and the presence of water on the drywell floor are important variables in estimates of

containment shell meltthrough probability. This issue is of course, of interest where venting might be accomplished after a severe core damage accident but it does not bear on the recognized efficacy of venting prior to such an accident in order to prevent the same.

Specific questions that the SARP will attempt to answer in the near-term are the following:

1. What is the relationship of the BWR bottom head failure mode to variations in quantity, composition, temperature, and timing of arrival of the melt on the bottom head? (Task 3.1)

2. What is the effect of water on the drywell floor when the melt pours out from the pressure vessel? (Task 3.2)

3. How does the answer to the above question depend on the initial conditions (melt ejection rate, melt superheat, melt composition, initial presence or absence of water) and water addition rate? (Task 3.2)

4. Under what conditions would the crust that forms at initial contact between the melt and the shell be stable and for how long? What is the expected rate of heat transfer between the core melt materials and the shell for various melt conditions? (Task 3.3)

In addition to addressing the questions related to attack of the Mark I containment shell, the results of the following tasks will provide initial conditions for a variety of ex-vessel phenomena that depend on the quantities and dynamics of spreading of the released corium.

Task 3.1: Investigate debris relocation phenomena into the lower plenum of a BWR, including the failure mode of the core plate.

Research Approach:

Review and assess existing BWR core relocation experiments and associated analyses. Analyze core plate and RPV failure mode and timing assuming a range of quantities, composition, and timing of melt. Based on these analyses, identify the key unknowns that can only be determined by further experiments, and assess the feasibility of those experiments. If such experiments are determined to be both necessary and feasible, design and conduct properly scaled out-of-pile tests in which simulated molten core debris is poured onto a scaled core plate or bottom head to verify the failure mode as a function of the nature of the debris (quantity, composition, and timing). The results of this task, including that for any additional experiments carried out under this task will be used to develop a model to predict lower head failure. All the analyses and experiments should assume expected degraded BWR core conditions (e.g., water in the lower plenum at the time of melt arrival on the core plate and ADS operation.)

Useables and Use:

This work will address the expected mode, timing, and temperature of core plate and bottom head failure and passage of molten core debris through core plate and bottom head for use in severe accident analyses. The results of this task will provide crucial initial conditions for a variety of ex-vessel phenomena that depend on the quantities and spreading characteristics of released corium.

Task 3.2 <u>Determine the effect of water on melt spreading, melt cooling, and melt-concrete interactions.</u>

<u>Research Approach:</u>

The results of melt spreading, cooling, and core-concrete interaction experiments will be reviewed to determine the extent to which a consistent picture of critical variables and conditions can be drawn. Among the questions to be considered are the role of superheat, the effect of unoxidized Zr in the melt, crust formation and stability, the generation of aerosols and noncondensible gases from core-concrete interactions, and the effect on the behavior of the melt of the presence of water--both water beneath the RPV upon bottom head failure and water added atop the melt following the arrival onto a previously dry floor. Consistent with the results from the scaling program (Task 1), calculations will be conducted either to confirm the conclusions of the review (or clarify issues that were concluded to be ambiguous) and define those near-term experiments that can be conducted to resolve ambiguities of data.

<u>Useables and Use:</u>

This work will produce refined estimates of melt spreading, hydrogen and other noncondensible gas generation from melt-concrete interactions, and the effect of water beneath the RPV. Of equal importance, however, this task will bring to focus those questions that may only be addressable by experiment over the long-term since this task goes to some of the basic chemical and process effects of high temperature interactions of the core-concrete materials.

Task 3.3 <u>Determine the conditions under which the interactions between the spreading melt and the Mark I drywell shell will lead to containment shell failure.</u>

<u>Research Approach</u>:

Experimental results of heat transfer from steel melts to steel structures will be examined, to determine their applicability to reactor accident conditions. In particular, questions such as the effect of the inclination of the shell to the drywell floor and the effect of concrete decomposition gas agitation of the melt will be addressed. Experiments will be devised and conducted to determine in the near-term whether a stable crust can form and persist at the drywell shell. Unless

otherwise indicated from the results of Task 3.2, this task will assume the presence of water in the drywell. The staff recognizes that it may well be that this task might not be completed in the near-term, however, available insights and data will be used in the closure process.

The presence of Zr metal in the melt arriving at the shell depends on the extent of its oxidation both in-vessel and while the melt is spreading over the drywell floor. If it is determined that significant amounts of Zr remain unoxidized before contact with the shell, then the impact of the presence of Zr will be determined.

Useables and Use:

The results from this task will enable the staff to determine the conditions under which the shell of a Mark I containment may fail, and, in particular, determine the proximate melt amount, composition, and superheat necessary for failure.

Issue 4 - Adding Water to a Degraded Core

There is little doubt that in a severe accident situation the primary efforts of the operators will be directed toward making water available to the reactor vessel. An important question that needs to be considered in view of such likely efforts is what are the likely consequences of those efforts. Given the uncertainties in core melt phenomena, the uncertainty associated with the operator's knowledge of the condition and location of the core during an actual severe accident, and given the intuitive drive to put water onto the core in the event of an accident, it is not likely that an operator ever would be told not to put water into the reactor vessel should water become available during the course of an accident. However, along with the potential benefit of

achieving a stable, coolable configuration, restoring water to a core that has been severely damaged, can have effects of which the operator should be aware. The operator also should maintain cognizance of possible symptoms and response of the plant to adding water in such circumstances (e.g., molten core-coolant interaction, increased hydrogen generation, increased containment pressure).

A related question that also needs exploration is what are the circumstances in which a grossly degraded core can be prevented from melting its way through the vessel lower head. Although important parameters can be identified as water availability, system pressure, and debris configuration, it is not likely that complete resolution to this issue will be achieved in the near-term. Nonetheless, there are questions related to the addition of water to a degraded core that can be addressed in the near-term that will provide some insights into the longer-term issues of accident management as well as improve understanding of low RCS pressure core melt accident phenomena. The questions to be addressed are the following:

1. What are the amounts and rates of hydrogen and steam generation during the reflooding of a degraded core and during relocation of the melt onto the bottom head? (Task 4.1)

2. What is the potential that reflooding a severely damaged BWR core will result in a recriticality? What will be the effects as opposed to not adding water? (Task 4.3)

Task 4.1 <u>Determine the effect of water injection on the generation of steam and hydrogen during reflooding of a degraded core.</u>

<u>Research Approach:</u>

Identify basic variables governing heat transfer and hydrodynamics of melt-water interaction, including the effect of water injection on debris reconfiguration. The work would be oriented toward scoping the range of hydrogen generation that can be produced during quenching, and assess the impact on containment performance. In consideration of the results of the scaling analysis (Task 1), establish whatever additional experiments may be needed or conduct analysis needed to determine the effect of adding water to a degraded core. We have already performed a lot of TMI-2 analyses and will assess these results.

Additionally, a benchmarking exercise, using available models (to be selected), against the TMI-2 accident beginning at 174 minutes (when core was reflooded by the start of a reactor coolant pump and high-pressure injection) will be undertaken as part of this task.

<u>Useables and Use:</u>

This task will produce estimates on the amounts and rates of steam and hydrogen generation as a function of water addition to degraded core geometries. The effect of adding water to degraded cores has important application to accident management as well as improving the quantification of the resultant containment loads.

Task 4.2: <u>Stability of Melt-Supporting Crusts.</u>

<u>Research Approach:</u>

The formation of a crust (crucible) supporting a melt, the predicted failure of the crust, and the eventual relocation of the melt into the bottom of the reactor vessel depend strongly on the heat transfer coefficients at boundaries of the core melt crucible. The anticipated approach is to review and assess existing analysis for

both the growth and demise of a melt-supporting crust and subsequently perform a set of calculations of heat transfer along boundaries of the crust crucible, analyzing thermal and mechanical crust stability as a function of melt accretion.

Useables and Use:

The work will affirm whether the heat transfer coefficient and the code treatments of core behavior, including crucible formation and collapse and debris relocation into the lower plenum, are suitable.

Task 4.3: Investigate the possibility and consequences of recriticality in degraded BWR cores.

Research Approach:

There are two configurations to be considered with regard to the recriticality problem. One is the possibility that standing fuel pellet stacks will become critical upon reflooding (with control rod materials previously melted out). The other is the possibility that core material will become critical upon relocation to the bottom head.

The approach is to first review and assess existing analyses of both the likelihood and consequences of recriticality of a damaged BWR core. Where deficient, additional calculations will be performed assuming the existence of critical masses for expected degraded BWR core geometries reflooded with unborated water.

Useables and Use:

The products of this task would be an estimate of the likelihood and effects of recriticality on accident progression and the alternative design mitigative measures (e.g., minimum boron concentrations in the reflood water) that might eliminate the problem if one is found to exist.

Issue 5: Use and Status of Severe Accident Models (Codes)

As noted earlier, the confidence in the analyses and judgments that will be made in implementing the staff's Integration Plan will depend on how well the tools used in performing those analyses and making those judgments conform to the realities of a severe accident. The tools that have been developed under the SARP, the physical and phenomenological models embodied in computer programs (codes), have had and continue to have a key role in furthering the objectives of severe accident research, viz., support of the U.S. Nuclear Regulatory Commission's policy towards severe accidents. First, and perhaps most critical, is the use of the codes in identifying the important sources of risk. In this capacity the impact of the codes is important since the results of code calculations support the evaluation of risk. The evaluation of risk, in turn, significantly determines both what is researched and the relative priority of what is researched. Second, through an iterative process, exercising the codes helps define the experiments needed to improve the representation of physical phenomena by the codes. Third, the complex mechanistic codes provide a benchmark for less detailed but faster-running codes that are used in applications that require extensive calculations, such as parametric studies for evaluating accident management strategies. This "two-tier" code strategy has been pursued because the use of the detailed mechanistic codes to evaluate entire accident sequences is costly, impractical and unnecessary.

Any large code development program is always faced with the difficulty of determining when the codes are "good enough." That is, when have they achieved an accuracy sufficient for the application for which they were intended? At this time, an acceptable level of accuracy has not been defined. As a result, the current code development program is continually iterative. Codes are developed and assessed against selected experiments, and needed model improvements identified. Overlaid on this process is the identification of new phenomena considered important to typical plants that "must be" incorporated into the code.

In addition to the above, a problem that is not unique to the severe accident program relates to the number of codes that must be developed to have a complete analysis capability. If one phase of the accident is difficult to model, or has inherently large uncertainties or variabilities, or perhaps is best understood as stochastic processes with a distribution of potential outcomes, it is not appropriate to model other phases of the sequence in greater detail, since both the overall variability and uncertainty will be driven by the least precise and least certain processes, respectively. Currently, this appears to be the situation with regard to late-phase core melt progression. The degree to which the early phase is modeled, and those models tested and validated by experiment, must be tempered by the degree to which the late phase through melt relocation and vessel failure can be understood and modeled, and that understanding tested by experiment.

Therefore, as part of the code development program, a method must be developed that constantly assesses the state of code development from a number of standpoints in addition to how well a particular sequence or facet of an accident can be modeled. The following questions will be considered:

1. How well do the mechanistic models reflect the phenomena believed to be important to severe accidents? (Are the correct phenomena being modeled?)

2. How well does the interactive program of code advancement and experimentation achieve the objective implied in 1 above?

3. Is the level of detail in the codes appropriate to their use? (Are some codes more detailed than needed, others not detailed enough?)

4. Which stages of an accident need to be modeled by detailed mechanistic codes, coupled to adjacent stages? (While clearly there is an interdependency among stages of a severe core melt accident, a detailed mechanistic code coupling all stages of accident phenomena from the onset of core uncovery through containment failure is not needed to either understand or make regulatory judgments about severe accidents. Further, a detailed mechanistic model of all aspects of a severe accident may not be possible, let alone necessary.)

5. Is the level of precision needed for regulatory use being considered in the code development program?

6. Is the level of precision from a given code more than needed and is it suitable to the expected overall levels of uncertainty from an integrated analysis package for applications as discussed above (e.g., accident management)?

Answering the above questions is a major undertaking and is not likely to be achieved in the near-term. Nonetheless, some measure of confidence in the tools that will be used in achieving the near-term goals must be made. Therefore, the following task will be done. Although not a complete code assessment, this task is believed to be a workable compromise in the near-term and should provide sufficient information to enable the staff to gauge the appropriate level of confidence to be placed in calculations using the codes.

Task 5 <u>Code Documentation and Review.</u>

<u>Research Approach:</u>

For each code developed under the SARP, the developer will be asked to supply the following information:

a. Stages in a severe accident to which the code is intended to apply and the accident sequences in which those stages are found.

b. Current capability of code. (Of the intended accident stages and phenomena, which does the code now model, which does it not).

c. Limitations of the code. (Portions of accident stages and/or special assumptions under which the code operates).

d. Statement of why the code needs to be or does not need to be coupled sequentially to adjacent codes to yield sufficiently precise calculations.

e. The degree to which the relevance of the physics, as embodied in the code, to severe accidents has been tested by experiment. (This point questions not simply the conformance of code to experiment, but experiment to severe accident.)

f. What features (physics and chemistry) not presently in the code/model need to be incorporated to enable the code to be a reasonable representation of accident phenomena.

g. What quality assurance program was the code subjected to.

<u>Useables and Use:</u>

The above information will be reviewed and scrutinized by the staff, and an assessment will be made of each code as to its present scale up capability and utility to support achievement of the near-term goals of the revised SARP. Based on that assessment, recommendations as to use of the present form of the code, further development, or abandonment will be made.

Criteria for judging the suitability of each code are being developed. These criteria will take into consideration the intended end use of the code and the expected user of the code. Deciding when code development is completed would require a subjective judgment and a sense of proportion between the level of precision needed for application and the levels of uncertainty and variability in models of severe accident phenomena.

This latter task is applicable to both the near and long-term research as is Task 1 on similitude. The conduct of Task 1 and 5 is believed logical and necessary to the critical reexaminations of SARP needs. However, neither is intended to constrain closure of the near-term issues described above.

IV. MEETING Long-Term GOALS

The research described in this section of the SARP plan consists of work primarily directed at reducing uncertainties in the estimation of the risk presented by severe accidents. This orientation is consistent with the direction provided in "Review of Research on Uncertainties in Estimates of Source Terms from Severe Accidents in Nuclear Power Plants," NUREG/CR-4883, known as the "Kouts Report." That direction suggested a broad base approach, investigating manifold phenomena with the common goal of reducing uncertainties. Both the approach and the goal remain valid, as does the more specific direction in NUREG/CR-4883 that identifies individual technical questions as contributors to uncertainty in risk estimates. However, there are sources of new information that bear on how to order research and distribute resources among the topics of the SARP. These sources are the revised "Reactor Risk Reference Document," NUREG-1150, and the guidance provided in SECY-88-147 that identifies accident management as a strategy that the NRC will pursue in dealing with severe accidents. In addition, the successes of the SARP itself over the years are a source of information providing a clearer picture of which phenomena are critical to understanding severe accidents, insight into what avenues of investigation are likely to be most fruitful, and which topics are likely never to be fully resolved in a reasonable time frame within realistic resources.

Taking the above together with the needed balance discussed in Section II, specific topics have been identified that characterize the SARP efforts directed toward achieving the long-term goals cited in Section II - providing an understanding of the range of phenomena exhibited by severe accidents, including impacts of generic accident management schemes, and developing improved methods for assessing the "severe accident source term." These topics are the following:

1. Review of the SARP approach to modeling severe accident phenomena

2. In-vessel core melt progression and hydrogen generation

3. Hydrogen Transport and Combustion

4. Fuel-coolant interaction

5. Molten core-concrete interactions

6. Fission product behavior and transport

7. Fundamental data needs

It is not at all clear that detailed mechanistic models can be, or need be, developed for items 2 through 6. Some phenomena may only be tractable when treated statistically as a population of related events with differing outcomes. In recognition of this possibility, the first item listed is a systematic review of the approach taken in the SARP to modeling severe accident phenomena.

Issue L1 - Modeling Severe Accidents.

Because of the difficulty in performing prototypic experiments and the variety of scenarios possible, substantial reliance must be placed on the development and validation of complex computer codes for analyzing severe accidents. A number of "mechanistic" codes (e.g., SCDAP/RELAP, MELPROG/TRAC) have been developed for various stages in severe accidents, both in-vessel and ex-vessel, for both PWRs and BWRs. However, it appears that there are practical limits to the feasibility of deterministically modeling all aspects of severe accident behavior. Moreover, it is not clear that all aspects of severe accidents need to be mechanistically reflected in deterministic codes. The resolution of some severe accident regulatory issues may be achieved with bounding analyses alone. Further, some phenomena may be better addressed employing stochastic rather than deterministic techniques. Priorities for additional development can be determined through use of probabilistic risk assessment techniques. When risk is unsensitive to the uncertainty in the phenomena or variabilities in parameter values, there should be no need to attempt to reduce the uncertainty or variability further.

Using the "documentation" provided by Task 5 of Section III of this plan as a point of departure, the approach of developing mechanistic, deterministic codes for each stage of severe accidents will be reviewed. For each code development program/phenomenon being addressed, a review of progress to date, magnitude of uncertainties remaining, and degree of validation will be made. Based upon this information, a determination will be made as to whether an alternative approach might be appropriate for dealing with some phenomena. The efficacy of the SARP's "two-tier" code strategy (fast-running lumped parameter system codes benchmarked against a combination of individual stand-alone and coupled mechanistic, deterministic codes) also will be examined. This review will begin in the beginning of FY 1990 and recommendations will be made at the end of FY 1990.

The NRC staff assessment of the modeling program will be assisted by various individuals outside the NRC and will be drawn from universities, national laboratories, industry personnel who are expert in accident phenomenology (have appreciation for the purpose of the analysis and the uncertainties in the all parts of the analysis), regulatory concerns, and programming and numerical techniques. This effort would identify the strength or weakness of different modeling approaches, quantify the uncertainties in code models, and identify additional research, if any, where uncertainties and risk importance are significant.

Issue L2 - <u>In-vessel core melt progression and hydrogen generation.</u>

In-vessel core melt progression concerns the state of the reactor core from the start of core uncovery to reactor vessel failure. Included phenomena are thermal attack by the core debris upon the reactor structure and the reactor vessel, in-vessel hydrogen generation, in-vessel natural convection and heat transfer, and in-vessel steam generation. The details of core melt also determine rates and amounts of in-vessel fission product release and aerosol generation and much of the fission product and aerosol transport (and retention) in the reactor coolant system. (Some aspects of this issue relating to fuel-coolant interaction were discussed in Section III.)

In considering melt progression, BWRs and PWRs need to be treated separately, mainly because of their different fuel assembly, control element, and lower plenum structures. For example, the effect of water injection on (1) the integrity of upper in-vessel structures, (2) the mode of core relocation, (3) hydrogen generation upon core relocation, and (4) the mode of bottom head failure and the details of the release of core and structural debris into the drywell following failure are questions common to both PWRs and BWRs. However, the answers are quite distinct arising from the physical and geometric differences between BWRs and PWRs. Bounding calculations and few small-scale experiments may be continued to identify which phenomena are critical for each

reactor type and whether further reduction in uncertainties and calculational variabilities related to melt progression and hydrogen generation are needed and achievable.

Areas in which work will continue are the examination of TMI-2 data for whatever insights might be provided, the NRU fuel melt experiments, and participation in the existing cooperative program with the FRG at the CORA facility. New experiments that evolve in the near-term research may continue as well.

Issue L3 - Hydrogen Transport and Combustion

The major concerns regarding hydrogen in LWRs are that the static or dynamic pressure loads from hydrogen combustion and detonation may breach containment integrity, or that safety-related equipment may be damaged as a result of either pressure loads or high temperatures. To assess the possible threat to containment and safety-related equipment, it is necessary to understand how hydrogen is transported and mixed within containment and to determine the likelihood of various modes of combustion. The hydrogen behavior issues have been extensively investigated since 1979, but several important areas of uncertainty remain:

(1) high-temperature/high steam concentration combustion and
(2) deflagration-to-detonation transition

Research programs on these issues of combustion have been sponsored in the U.S. by the NRC and the nuclear industry as well as in the international community. However, the combustion processes are sufficiently complex such that many aspects are still not well understood. The resulting uncertainties in the threat to containment integrity are unlikely to be reduced significantly by the existing research programs. The intent of the this research is to reduce these uncertainties, however in some cases reduced uncertainties are not required to make a near-term regulatory decision. Each category of uncertainty is discussed below.

High-Temperature Combustion

The Zeldorich-von Neumann-Doering (ZND) chemical kinetics theoretical model developed under NRC sponsorship predicts that increasing temperature has a strong effect on the combustion and the detonation of off-stoichiometric hydrogen-air mixtures and on all hydrogen-air-steam mixtures. It is believed that the steam inerting effect is reduced greatly at elevated temperatures. There is a limited supporting data base for this phenomenon. Experiments are necessary to resolve the uncertainties associated with high temperatures and steam concentrations typical of those likely to be encountered in severe accident scenarios. It is also necessary to extend or develop more mechanistic models to predict, by either extrapolation or interpolation, the temperature sensitivity of hydrogen-air-steam mixtures that have not been tested.

The small-break LOCA and TMLB scenarios are two examples of high-temperature hydrogen-air-steam mixtures that may exist below the auto-ignition temperature (550 C minimum value for stoichiometric, hydrogen-air mixtures). There are two aspects of the high-temperature/high steam concentration combustion problem. The first is the injection of high-temperature hydrogen and steam mixtures that auto-ignite upon contact with pre-existing and premixed hydrogen-air-steam mixtures. The second aspect is the injection of hydrogen and steam mixtures at elevated temperatures that do not auto-ignite upon contact with pre-existing hydrogen-air-steam mixtures. This allows the possibility of a premixed condition to form and a subsequent deflagration or detonation. In both cases, the competition between chemical reaction rates and physical mixing rates will ultimately determine the ensuing combustion mode. Data are needed to determine the hydrogen-air-steam flammability limits and steam inerting criterion at elevated temperatures if reliable predictions are to be made as to the likelihood and the potential threat resulting from various combustion mode(s) for a wide range of accident conditions.

Therefore, the research approach is to determine hydrogen-air-steam flammability limits, volumetric oxidation (chemical reaction) rates, physical mixing rates, and the competition of these rates during the injection of high-temperature hydrogen and steam mixtures into cooler pre-existing and premixed hydrogen-air-steam mixtures.

A draft proposal is currently under consideration by the NRC that addresses feasibility; construction design, cost, and schedule; and an experimental test plan. It is our current estimate that if a facility is needed, it can be constructed by early FY 1990 and that experimental results can be generated by late FY 1991 or early FY 1992. Final data reduction and formal documentation should be available in late FY 1992.

Containment Loads for Detonations

Deflagration-to-Detonation Transition (DDT)

Direct initiation of a detonation would require a concentrated high-energy source for insensitive, steam-diluted mixtures. This is not considered a credible mechanism of initiation by almost all researchers. However, it is possible to initiate a flame with a low-energy source such as a spark or a glowplug, and the subsequent propagation through orifices and around obstacles such as pipes can result in flame acceleration that culminates in a transition to detonation.

The possibility of DDT in realistic and prototypic containment geometries and conditions needs to be resolved. The uncertainty in this area has increased because of recent experimental and theoretical results that indicate an increased likelihood of

detonations at high temperatures and large steam fractions as discussed earlier. The current data base on flame acceleration and DDT suggests that the mixture composition, obstacles, and venting are all important factors. These uncertainties result from a lack of experimental data on flame acceleration and DDT for conditions that include the effects of steam dilution, elevated temperature, large-scale and prototypical obstacle types, and spacing.

At present, no reliable model exists to predict or extrapolate DDT results from small-scale experiments to containment scale. Reactor safety studies and fundamental combustion research is being carried out in the Federal Republic of Germany and Canada. These data along with data generated for space shuttle application will be applied to reducing the uncertainty associated with DDT as well as in assessing the potential threat of DDT to containment integrity. Specifically, this data base will then serve as the basis to improve or develop correlations and models to allow extrapolation of experimental results to reactor scale and accident conditions.

Issue L4 - <u>Fuel-Coolant Interactions.</u>

Molten fuel contacting water can give rise to a range of phenomena. Very energetic steam explosions could result in early containment failure (alpha-mode failure). Less energetic interactions do not threaten containment directly but could change the course of the accident and the magnitude of the source term. Among these are

sudden coherent failure of the vessel lower head, ex-vessel debris dispersal and steam generation pressure pulses. (Core debris dispersal, and steam and hydrogen generation in some BWR Mark II and Mark III containments also warrant further evaluation as to whether fuel-coolant mechanism can pose a threat to containment integrity.)

The research to be conducted will be selective, confirmatory in nature, and focused to address some well-defined questions dealing with key parts of steam explosion phenomenology, such as degree of premixing, triggering, and fragmentation within the detonation wave of a steam explosion for various premixture conditions. It is clear that the program will necessitate limited scale experiments to test the predictive capability of calculations of premixing and the fragmentation rate of melt drops in the explosion zone of a propagating explosion. Further, the utility of the calculational tools to assess the effect of (1) reflood of a damaged core or debris bed in-vessel, and (2) fuel-coolant interaction in a suppression pool will be examined.

In addition to providing final confirmation of the staff's position on the issue of the alpha mode of failure, this work also will provide analytical tools, additional data, and insights for use in evaluating the dynamics of molten fuel-coolant interactions in in-vessel and in various containment configurations.

Issue L5 - Molten Core-Concrete Interaction (MCCI).

This is a subject fundamental to severe accidents. It has received considerable research attention, both experimentally and analytically, mainly addressed at understanding the quasi steady-state interaction that would develop when the core melt materials form a pool above the concrete. However, some significant uncertainties remain. An important question is the long-term coolability of initially molten corium pools interacting with concrete and flooded with water. Other uncertainties involve the transient (early) stage of core-concrete-water interactions, which include the spreading and relocation of the melt over concrete and the associated thermal-hydraulic characteristics of the attack. Also uncertain is the effect of pour rate on melt spreading.

Specific topics that are addressed by the research include the key characteristics of molten-corium pools interacting with the concrete basemat, the rate of melt cooling in the early stages of MCCI, the energy balance of a corium-concrete interaction in the high-temperature regime, the rate of fuel cooldown as it spreads over the concrete, the effect of water on the spreading behavior and associated heat losses of the corium melts, the erosion and ablation of concrete structures (in particular, the reactor vessel pedestal), and the volatilization of fission products and production of aerosols during MCCI and estimation of the effects of

these phenomena on the source term. Several of these phenomena investigations are being carried out under the existing cooperative agreement with the EPRI ACE program and FRG BETA program. The design of any additional experimental and analytical programs will follow from the work discussed in Section III, Tasks 3.2 and 3.4.

Issue L6 - Fission Product Behavior and Transport.

The NRC has had a substantial program to investigate fission product release from fuel in-vessel. Today, PRA studies no longer identify this as an area of major uncertainty in risk assessment. The recent NUREG-1150 elicitations on a number of source term issues do indicate, however, that late iodine release, revolatization,and fission product release from core-concrete interactions can have an effect on the overall risk uncertainty. Hence, research in these areas is being continued. Nonetheless, because severe accident issues associated with phenomena that have the potential to result in containment failure have higher priority than modeling of fission product release and transport, source term issues will be addressed in the long-term program with lower priority.

Issue L7 - <u>Fundamental Data Needs</u>.

For some ranges of phenomena, fundamental data (physical and chemical properties and constants) do not exist or are so poorly known that their use provides no confidence in the fidelity of experimental or analytic results. Data needs have been briefly touched upon in Issue L2 above in connection with core melt progression, but similar needs exist with respect to MCCI. Among these needs are thermal properties (including melting points, latent heats, and thermal conductivities) and phase relationships among the various constituents of the debris. A continuing program to identify and measure (or calculate) the required basic data and to incorporate the results in the codes will be instituted. However, in supporting this basic data effort, we will attempt to strike a balance between the accuracy of the data and its significance relative to regulatory applications.

TABLE 1

SARP MILESTONES AND ESTIMATED COSTS

Task:	Title	Description	1st Deliverables	Expected Completion	Expected Cost (FY90-92)
Issue 1:	<u>Scaling</u>				
1	Scaling Analysis	Develop and apply severe accident scaling methodology	3/89	6/91	1800K
TOTAL	ISSUE 1				1800K
Issue 2:	<u>Depressurization and DCH</u>				
2.1	**Containment Challenges Research**	**Assess both experimental and model development programs to determine the appropriateness of additional expenditure**	**6/89**	**12/89**	**300K**
2.2	**Natural Circulation**	**Confirm natural circulation calculations**	**9/89**	**9/90**	**1800K**
2.3	**DCH Scaled Experiments**	**Experiments to study effect of structures and influence of water in the reactor cavity and containment atmosphere and to verify analyses of Task 1**	**11/89**	**9/91**	**2500K**

TABLE 1 (CONTINUED)

Task:	Title	Description	1st Deliverables	Expected Completion	Expected Cost (FY90-FY92)
2.4	Depressurization (intentional)	Calculations exploring efficacy and consequences of possible methods of intentional depressurization	9/89	9/90	600K*
2.5	RPV Bottom Head Failure	Analyses over a range of melt configurations. Confirmatory experiments.	9/89	6/90	800K
2.6	Material at Lower Head at Time of Breach	Determine quantity, composition and timing of molten core arrival at bottom head	8/89	8/90	600K
2.7	Upgrade DCH models	Incorporate test results into DCH analysis model	9/89	4/90	600K
2.8	Depressurization Cost/Benefit Analysis	Provide bases upon which to recommend for or against depressurization	6/89	12/89	220K
TOTAL ISSUE 2					7420K

* This Task will be conducted under the Accident Management research. The allocated $ is for code validation/development

TABLE 1 (CONTINUED)

Task:	Title	Description	1st Deliverables	Expected Completion	Expected Cost (FY90-FY92)
Issue 3:	BWR Mark I Containment Shell Meltthrough				
3.1	Lower Plenum Study	Analytical/experimental studies to determine sensitivity of BWR RPV failure to characteristics of melt in lower plenum	12/89	9/90	1000K
3.2	Effect of Water on Melt Spreading	Analytical and experimental work scoping melt-water interactions	12/89	6/90	900K
3.3	Mark I Containment Shell Failure	Examination of past experiments and conduct of new experiments to verify expected heat transfer to drywell shell in Mark I under expected ex-vessel melts	12/89	6/90	600K
TOTAL ISSUE 3					2500K

TABLE 1 (CONTINUED)

Task:	Title	Description	1st Deliverables	Expected Completion	Expected Cost (FY90-FY92)
Issue 4:	Adding Water to Degraded Cores				
4.1	H_2 Generation Upon Reflood	Analysis of melt-water interaction, design of needed experiments	9/89	6/90	1000K
4.2	Stability of Crust Crucible	Analysis to examine formation and stability (mechanical and thermal) of crust crucible	9/89	1/90	800K
4.3	Recriticality Investigation	Analysis to determine the extent of recriticality concern	9/89	9/89	200K*
TOTAL	ISSUE 4				2000K
Issue 5:	Use and Status of Severe Accident Codes				
Task 5	Role and Future Development of Severe Accident Codes	An in-depth assessment of the entire area of NRC-sponsored severe accident codes development	9/89	3/90	855K
TOTAL	ISSUE 5				855K

* This Task is being done under the Accident Management research. The allocated $ is for defining different critical masses expected for degraded BWR core.

TABLE 1 (CONTINUED)

Long-Term Goals (Confirmatory Program 3 years projection)

Task:	Title	Description	Expected Cost (FY90-FY92)
	Role of Uncertainties in Regulatory Decision and Future Research	Seek further depth of understanding of accident phenomenology, validate analysis tools. The issues are described below.	
Long Term Research			
Issue L1	Modeling Severe Accidents		16000K
Issue L2	In-Vessel Core Melt Progression and Hydrogen Generation		17000K
Issue L3	Hydrogen Transport and Combustion		3000K
Issue L4	Fuel-Coolant Interactions		3000K
Issue L5	Molten Core-Concrete Interaction		6668K
Issue L6	Fission Product Behavior and Transport		4538K
Issue L7	Fundamental Data Needs		500K
TOTAL			50,706K

APPENDIX A

Background on Severe Accident Research Program and Relationship to Other Elements of SECY-88-147

For the past 10 years or so, since the Three-Mile Island accident, NRC has sponsored a research program on severe nuclear power plant accidents as part of a multifaceted approach to safety. Other elements of this approach included improved plant operations, human factor considerations, and probabilistic risk assessments. In August 1985, the Commission issued a Severe Accident Policy Statement (50 FR 32138), which concluded that existing plants posed no undue risk to public health and safety. However, the Commission recognized that systematic examinations of existing plants could identify plant-specific vulnerabilities to severe accidents for which further safety improvements could be justified.

In May 1988, the staff presented to the Commission an Integration Plan for Closure of Severe Accident Issues (SECY-88-147). The Integration Plan consists of six major elements:

1. Examination of existing plants for severe accident vulnerabilities (individual plan examinations)

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.

During the past few years, the SARP has generated a large amount of insight into the progression of severe accidents. An extensive experimental program has led to a vastly improved understanding of in-vessel core melt progression and associated phenomena of hydrogen generation and fission product release. Models of core melt progression are being coupled to thermal-hydraulic in-vessel and primary cooling system models to allow more rigorous treatment of the in-vessel stages of a severe accident. The ex-vessel phenomena of melt ejection and direct containment heating in high-pressure accident scenarios and molten core-concrete interactions in lower-pressure scenarios, together with their associated threats to containment integrity, also are becoming better understood as a result of the ongoing experimental and

analytical efforts under the SARP. Understanding containment challenges due to steam and noncondensible gases is much improved as well. The "two-tier" code strategy of developing detailed "mechanistic" codes for understanding and faster-running "integral" codes for application has furthered the NRC's ability to calculate the impact of accident sequences and their associated risks to public health and safety. Both levels of codes have been used in the risk assessment document, NUREG-1150.

As can be inferred from the above, the thrust of the SARP up to now has been to establish and refine the technical and scientific base of knowledge in the area of severe accident phenomenology, to apply it at the scale of reactor accidents, and to reduce the uncertainties in this knowledge base and the risk assessments that depend on it. The program to date generally has been appropriate to the above motivations. However, because the needs for the SARP are changing as a result of actions being taken by the NRC to bring severe accident regulatory issues to closure, it is natural to question whether the directions of the program should change accordingly. Obviously there will remain practical and resource related questions as there as to how many fronts can be pursued in understanding and characterizing accident phenomenology and which of these fronts will prove the most fruitful ones to follow to achieve closure of regulatory issues.

I. DEVELOPMENT OF THE REVISED SARP

To assist the NRC RES staff in this reappraisal and revision of the SARP, a set of four expert groups was established to help the RES staff identify and define the status of present SARP activities that relate to the Integration Plan, identify and focus research necessary for sound regulatory decisions to be made within the framework of the Integration Plan, and identify and rank the base of confirmatory research activities directed at achieving the long-term goals. Each group was composed of contractor personnel from the DOE laboratories, consultants from universities and industry, and NRC staff. Each member was active in severe accident research. The groups consisted of three "working groups" that addressed the technical points of the research and an "integration group" that dealt with programmatic considerations.

The working groups considered detailed technical issues, defining their status and needs for further research within the frameworks of the Integration Plan and confirmatory research. The integration group took the reports of the working groups and evaluated and synthesized them, along with other considerations, into an earlier draft of a recommended Severe Accident Research Program plan. The NRC staff then used this as a basis for preparing the Revised SARP plan discussed in this report.

II. RELATIONSHIP OF THE REVISED SARP TO OTHER ELEMENTS OF THE INTEGRATION PLAN

In order to place the revised SARP plan in perspective, it is useful to first discuss in general terms the relationship of SARP to the other elements of the Integration Plan. From the discussion it will be apparent that SARP provides, or will provide, important and often essential data to most of the other elements of the Integration Plan.

1. Relationship of SARP to Individual Plant Examinations

The Integration Plan provides the specific objectives for the IPEs that each utility is expected to meet.

a. Reduction of the overall probability of core damage and fission product releases by appropriate hardware and procedure modifications;

b. Development of an overall appreciation of severe accident behavior; and

c. Development of an understanding of the most likely severe accident sequences that could occur at its plant.

It is envisioned that a principal tool for an NRC audit review of IPE submittals will be a relatively fast-running, integrated severe accident analysis code such as MELCOR. (This code in turn is benchmarked and validated against detailed and more mechanistic codes.) All of these codes draw or have drawn heavily for their phenomenological modeling on the SARP experimental and analytical efforts (including the foreign and industry contributions under cooperative arrangements). Up to now, these codes have been under continual development and validation. However, it is presumed that when the staff begins its review of the IPEs, some versions of these codes will be "frozen" for use by the NRC staff.

It should be noted that the new SARP addresses all the phenomenological uncertainties discussed in Appendix 1 of the IPE Generic Letter No. 88-20.

2. <u>Relationship of SARP to Containment Performance Improvement Program</u>

The Integration Plan states that the Containment Performance Improvement (CPI) program complements and is closely integrated with the IPE program and is intended to focus on resolving hardware and procedural issues related to generic containment challenges.

A detailed summary of the relationship of SARP and CPI is given in Section 5.2 of the enclosure to the Integration Plan (SECY-88-147).

3. Relationship of SARP to Improved Plant Operations

The Improved Plant Operations (IPO) program includes elements relating to continued improvements of the Technical Specifications, Emergency Operating Procedures (EOPs), expanding EOPs to include guidance for severe accident management strategies, and industry programs to reduce transients and other challenges to the engineered safety features (ESFs). These elements are obviously closely related to accident management strategies, discussed below.

The relationship of the SARP to these elements of the IPO program involves providing analytical descriptions of severe accident sequences, including the effects of EOPs and other mitigation strategies. These analyses would predict the effects of accident mitigation schemes, including both a determination as to whether they will be effective and identification of possible undesirable consequences. This is particularly true with respect to the later phases of in-vessel (core degradation) progression of a severe accident and the ex-vessel and containment phenomena. Accident sequence analyses, based on SARP results, could be used to judge the appropriateness and effectiveness of EOPs under severe accident conditions.

4. Relationship of SARP to External Events

According to the Integration Plan, the evaluation of external events will proceed separately with a different schedule from that of internal events. At the moment, there is no part of the SARP that is explicitly addressed to external events. However, to the extent that external events such as earthquakes or fires act as accident initiators or constrain the availability of ESFs, safety-related equipment, and other mitigation strategies, the accident progression can be analyzed with the severe accident codes developed under the SARP plan.

5. Relationship of SARP to the Accident Management Program

The Integration Plan defined accident management to include measures taken to prevent core damage; to terminate the progress of core damage if it begins; and, failing that, to maintain containment integrity as long as possible to minimize offsite releases. The NRC accident management program concentrates on near-term improvements based on well-understood accident management procedures and strategies. The SARP will provide the data base to allow the staff to examine procedures and strategies whose benefits and adverse effects are not well understood.

As in other areas, the role of SARP in accident management is to provide the tools (i.e., codes) to analyze the progression of

severe accidents and to evaluate the effects of given mitigation strategies (both beneficial and detrimental). The staff is currently in the process of defining the analytical tools that are available or should be developed and required for accident management purposes. We expect to complete this effort by the end of FY 1990. It is important that SARP should continue to address certain important issues--in particular, the consequences of adding water, both in-vessel (reflood) and ex-vessel, in attempting to cool a severely damaged core or quench core debris--and that further attention should be given to evaluation of the uncertainties associated with the use of the codes.

APPENDIX B

A Severe Accident Scaling Methodology (SASM)

1. Objective and Outline

The objective of this appendix is to outline a Severe Accident Scaling Methodology (SASM) that could be used to address in a systematic and practical manner questions concerning such scaling topics of interest to severe accidents as:

a. The adequacy of the design and operation of a reduced-scale test facility to provide experimental data that can be used in safety analyses of full-scale nuclear power plants.

b. The appropriate initial and boundary conditions for experiments of interest.

c. The use of a set of experimental data or of a correlation based on the data in nuclear power plant safety analyses.

d. The effects of test facility scale distortions (if present) on physical and chemical processes of interest to nuclear power plant safety analyses.

e. The capability of a computer code to scale up physical and chemical processes observed in reduced-scale test facilities to full-scale nuclear power plant conditions, etc.

The role that scaling plays in nuclear power plant safety analyses is discussed in Section 2 of this appendix. The need for establishing a severe accident scaling methodology (SASM) is discussed in Section 3, together with the requirements that it would have to meet. The elements of such a methodology are outlined in Section 4, together with their rationale. The last section describes a program directed at demonstrating the methodology by applying it to the DCH problem.

A more detailed discussion of SASM together with worked-out examples are presented in Reference 1.

2. Elements of Safety Analyses and Premises

As was the case in studies of LOCAs, analyses concerned with severe accidents will have to be supported by a suitable experimental data base, on appropriate models and computer code calculations, and, when appropriate, on bounding calculations. Implicit to this approach are two premises: one pertaining to experiments and the other to codes; both are concerned with scaling.

For reasons discussed below, both premises must be addressed and evaluated before a high degree of confidence in the analysis will exist.

Premise Related to Experiments

In nuclear reactor safety research, experimental data are used (a) to develop correlations and/or models for a particular process or (b) to assess code capability to calculate such a process.

Full-scale test facilities capable of generating the experimental data of interest are prohibitively expensive to construct and to operate. In severe accident research, difficulties and costs are compounded because some phases of an accident scenario involve failures of the reactor vessel and of containment structures. Thus, large (or even smaller) scale integral facilities needed to provide experimental data on processes leading to or attending such failure events become prohibitively expensive.

Consequently in severe accident research, the majority of experiments will have to be performed in reduced or small-scale separate effects test facilities. The premise made in following this approach is that experimental data from such facilities are applicable and relevant to nuclear power plant conditions. This implies that test facilities as well as the initial and boundary conditions of experiments are properly scaled so that distortions (if and when present) will not affect the evolution of physical and chemical processes of interest.

Whether a facility and the initial and boundary conditions of an experiment are well scaled will have to be evaluated for each facility and set of experiments because such an evaluation will determine whether the data can be used in nuclear power plant safety analyses of a postulated severe accident.

Premise Related to Codes

The reliance on computer codes to simulate the behavior of a nuclear power plant during a postulated accident scenario is predicted on three factors. First, it is too costly, and for severe accidents not even feasible, to subject a nuclear power plant to such an event. Second, very often one cannot directly apply results from test facilities to nuclear power plants; this is particularly true for separate effects test data. Third, the study of various plant recovery techniques can only be performed by computer codes.

Implicit to applications of computer codes to analyses of postulated accident events in nuclear plants is the premise that these codes have the capability to scale up phenomena and processes from test facilities to full-scale plant conditions. For reasons discussed below, this premise must be evaluated on a case-by-case basis, that is, for each postulated accident scenario or for a set of scenarios.

It is often stated that advanced codes are "mechanistic" and are based on "first principles." Therefore, ipso facto, they have the scale-up capability. However, as a matter of fact, for the following three reasons this is not the case.

First, because the conservation equations used in computer codes are space averaged and because of their dependency on numerous empirical correlations, computer codes are not based on "first principles." As the capability of a code to model a particular process and/or phenomenon is provided by particular closure relations, the scale-up capability of a code will depend on whether or not the empirically determined closure relations have this capability. If the test facilities that generated the data are well scaled and the experiments are performed with appropriately scaled initial and boundary conditions, the empirical closure relations, and therefore a code, can be used in analyses of full-scale nuclear power plants. Otherwise limitations due to scale distortions must be assessed before meaningful safety analyses can be performed.

Second, because of discretization schemes used to nodalize a nuclear power plant and perform calculations, the computed values are not local but are averages over very large volumes. Consequently, these averages are functions of node size and may affect the evolution or the timing of a physical or chemical process calculated by the code. Furthermore, as nodalization used

to model a nuclear plant and small-scale test facilities differ (the latter have most often a much finer nodalization) the events calculated to occur in a full-scale plant may differ from those observed in test facilities. This problem becomes even more serious if a test facility has some scale distortions that could affect a particular process of interest.

Finally, because of "compensating errors" that may be present in a code, there is no assurance that a code has the capability to scale up processes observed in small-scale test facilities to full-scale nuclear plants. "Compensating errors" are generated most often during the code validation process. Since advanced codes have numerous parameters, coefficients, that is, "dials," improved agreement with experimental data is very often achieved by adjusting some of these "dials". This "tuning" process of a code to a set of experimental data can introduce "compensating errors" in the code. The effect of such errors on the scale-up capability of a code becomes even more difficult to assess if scale distortions are present in the facility or if the initial and boundary conditions of an experiment are not properly scaled.

It can be concluded from this brief discussion that if meaningful safety analyses concerned with postulated severe accidents in nuclear power plants are to be performed, then questions related to scale-up capabilities of experimental data and of computer codes will have to be addressed.

3. Needs and Requirements

The important role that scaling has in experimental and analytical investigations related to severe accidents was noted in the preceding section. This role establishes a need for a scaling methodology that would be applicable to both experiments and computer codes.

In order to meet the needs of research and development activities conducted for a regulatory agency, the methodology should:

1. Be systematic and practical, auditable and traceable.

When applied to a specific severe accident scenario, the methodology should:

2. Provide the scaling rationale and similarity criteria.

3. Provide a procedure for conducting comprehensive reviews of facility designs, test specifications, and results.

4. Provide a measure or index to indicate the applicability of correlations or models based on test data from sub-scale facilities to full-scale nuclear plant conditions.

5. Quantify the effects of scale distortion.

6. Quantify the effects associated with extrapolating correlation and/or models beyond their data base.

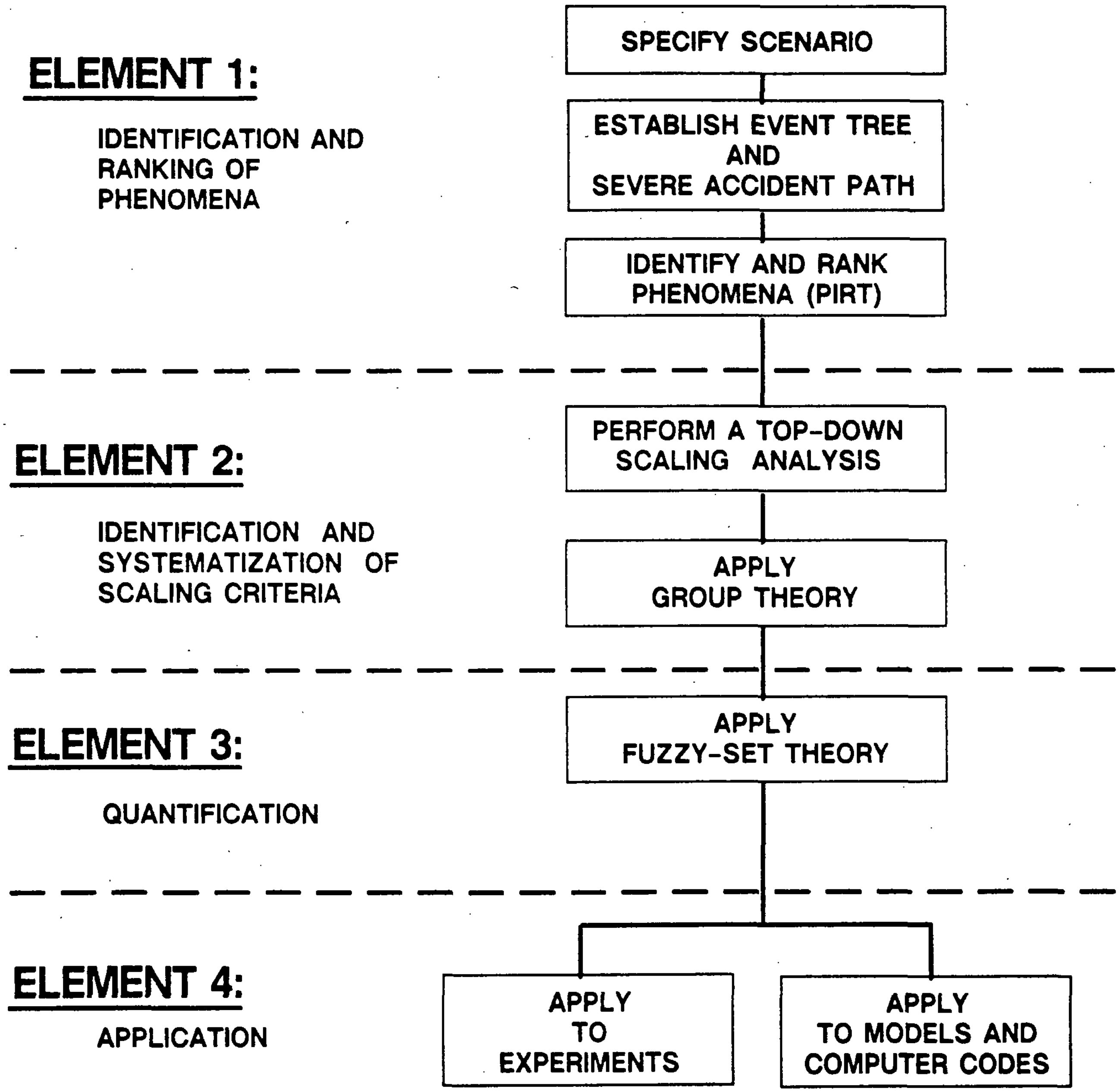
7. Relate in a systematic and unifying manner
   - test facility design and operation
   - test data accuracy and applicability
   - correlations (or model) accuracy and applicability to:
   - computer code scale-up capability and applicability to calculate the postulated scenario for a full-scale nuclear power plant.

8. Provide a quantifiable and traceable procedure for specifying and prioritizing future experiments if needed.

It should be noted that requirements 4 through 7 provide the information needed to quantify the code uncertainty to calculate the postulated severe accident scenarios.

A scaling methodology that meets these requirements is outlined in what follows.

FIGURE 2.1
SEVERE ACCIDENT SCALING METHODOLOGY (SASM)



4. Elements of SASM and Their Rationale

The proposed Severe Accident Scaling Methodology (SASM) consists of four primary elements as shown in Figure 2.1.

The first element: Identification and Ranking of Phenomena contains Steps 1-3. In this element, scenario modeling (and therefore scaling) requirements are identified by (a) specifying the scenario (Step 1), (b) establishing the event tree and selecting the accident path (Step 2), and (c) identifying phenomena/processes along this path and ranking their importance (Step 3). Activities carried out in this element are designed to meet the first requirement set forth in the preceding section. These activities are similar to those discussed and developed in References 2 and 3, that is, they:

1. Provide a comprehensive, physically based framework for analyzing an accident scenario. This is essential to a scaling methodology that is systematic, auditable and traceable.

2. Decompose the scenario in elementary components and provide a casual relationship approach. This is essential for identifying modeling (and therefore scaling) requirements and understanding the role of each component (and therefore model) in the selected accident path.

3. Identify and rank processes and phenomena along the accident path. This is important to a scaling methodology that is not only systematic but also practical. The need for this screening process arises from the fact that it is not feasible either to design an experiment or to develop a code that will scale properly all processes occurring during an accident. What is needed, however, is to ensure that physical and chemical processes important to the evolution of an accident are properly scaled.

4. Identify and prioritize research activities directed at resolving potential safety concerns related to an accident scenario. This is essential to an expeditious closure of a safety issue.

The second element: Identification and Systematization of Scaling Criteria contains steps 4 and 5. In this element, scaling criteria are (a) identified by performing a top-down scaling analysis (Step 4) and (b) systematized (Step 5) by applying group theory methods discussed in References 4, 5, and 6. Activities carried in this element are designed to satisfy the second and third requirement listed in the preceding section, that is, they:

1. Structure the scaling process to follow the physically based, hierarchical approach detailed in Element 1. Such a structure is essential as it provides not only the rationale for establishing scaling and similarity criteria but it ensures also that processes important to the scenario are taken into account in the similitude analysis.

2. Provide a structure and procedure that starts from a global, top-down view point and introduces complexity and detail at each lower level. This is important for conducting comprehensive reviews of facility design, test specifications, and results.

3. Provide a systematic and traceable approach to derive and select similarity parameters. This minimizes the arbitrariness, that is, the ad hoc approach used so often in facility design and test specifications.

4. Provide a method that which can yield similarity criteria for processes that can limit the operational range of a system.

The third element: Quantification contains Step 6, with activities designed to meet requirements 4 through 6, that is, to quantify the effects associated with scale distortion and/or with extrapolating correlations beyond their data base. Several methods can be used to achieve these objectives. One among them based on the fuzzy sets theory of Zadeh (Refs. 7 and 8) appears very promising in view of its successful application by Kubic and Stein (Refs. 9 and 10) to be a problem concerned with quantifying model uncertainties and system similarity.

The fourth element: Application which consists of Step 7, is designed to satisfy requirements 7 and 8, that is, to provide the same methodology that can be used to:

1. Design and operate test facilities.

2. Evaluate test data accuracy and/or applicability.

3. Evaluate correlations and/or model accuracy and applicability.

4. Evaluate computer code scale-up capability and applicability.

It should be noted that treating experiments and code model/correlation development in a systematic and unifying manner (by applying the same scaling methodology to both activities) is a prerequisite for quantifying efficiently and more accurately code uncertainties to calculate the postulated severe accident scenarios in a full-scale nuclear power plant.

5. Application and Demonstration

A program has been initiated at the Brookhaven National Laboratory with the objectives to:

1. Develop a scaling methodology (SASM) that meets the eight requirements listed above, and

2. Demonstrate the methodology by applying it to the DCH problem.

Furthermore, a Technical Program Group (TPG) has been formed to assist the staff and provide guidance to this work. The members were selected on the basis of their:

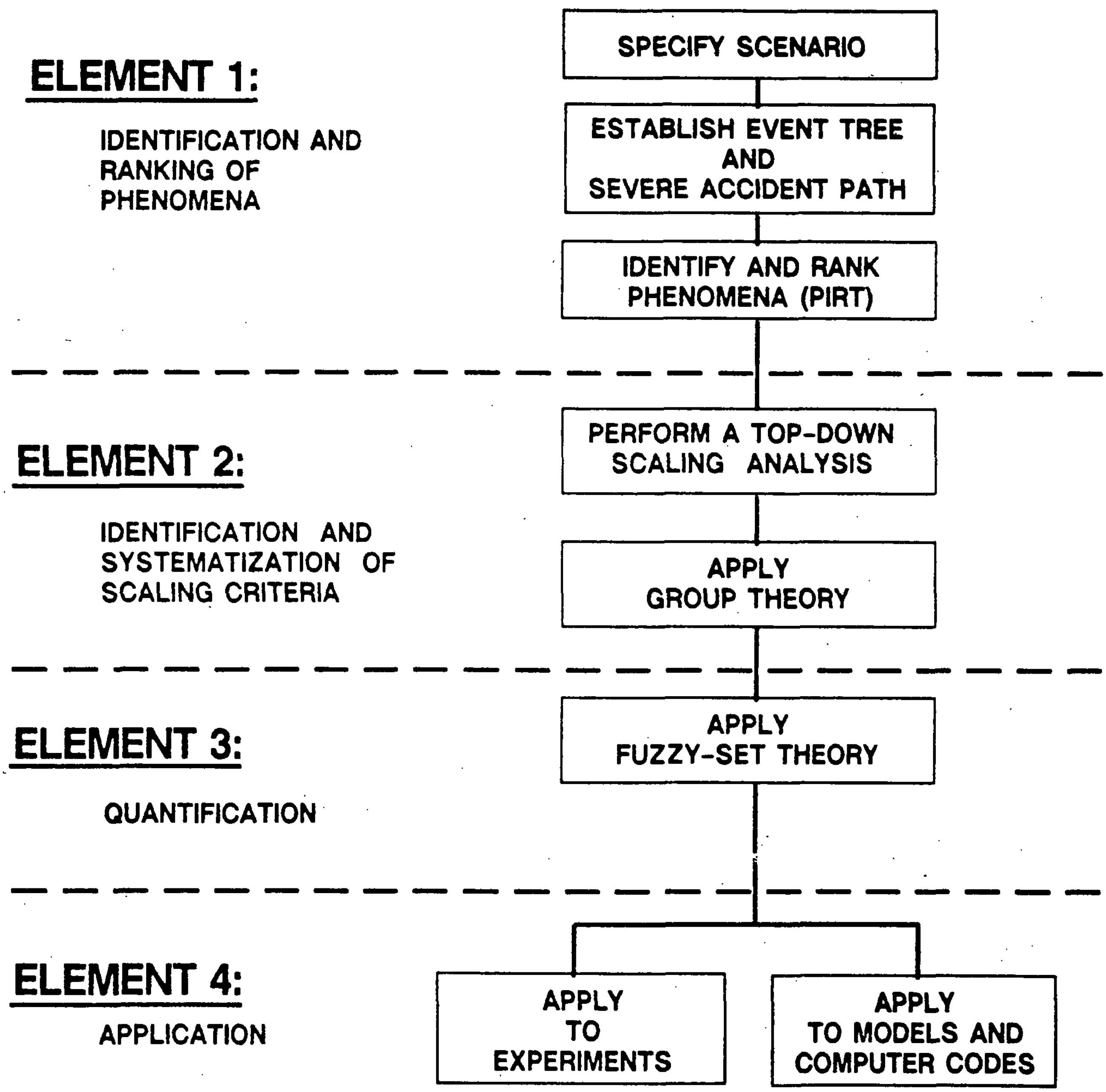
1. Knowledge of severe accident phenomena and issues, and/or

2. Internationally recognized expertness in modeling and scaling of complex systems and phenomena.

In order to provide input from a broad spectrum of technical sources, the composition of the TPG was specifically designed to include technical talent from universities, national laboratories, and industry.

This work that is, the development and demonstration of SASM, is expected to be completed by the end of December 1989.

FIGURE 2.1
SEVERE ACCIDENT SCALING METHODOLOGY (SASM)



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