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NUCLEAR REGULATORY COMMISSION

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BRIEFING ON SEVERE ACCIDENT RESEARCH PROGRAM

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PUBLIC MEETING

Nuclear Regulatory Commission
One White Flint North
Rockville, Maryland

Tuesday, October 26, 1993

The Commission met in open session, pursuant to notice, at 10:00 a.m., Ivan Selin, Chairman, presiding.

COMMISSIONERS PRESENT:

IVAN SELIN, Chairman of the Commission
KENNETH C. ROGERS, Commissioner
FORREST J. REMICK, Commissioner
E. GAIL de PLANQUE, Commissioner

STAFF SEATED AT THE COMMISSION TABLE:

SAMUEL J. CHILK, Secretary

KAREN CYR, Office of the General Counsel

JAMES TAYLOR, Executive Director for Operations

ERIC BECKJORD, Director, Office of Research

THOMAS MURLEY, Director, NRR

BRIAN SHERON, Director, Division of Systems Research, RES

ASHOK THADANI, Director, Division of System Safety and Analysis, NRR

FAROUK ELTAWILA, Chief, Accident Evaluation Branch, RES

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

10:00 a.m.

CHAIRMAN SELIN: Good morning.

We're pleased to welcome members of the staff to brief the Commission on the Severe Accident Research Program. The goal of this research is to achieve an adequate level of understanding of severe accident phenomena in order to allow regulatory decisions to be made on these severe accident issues.

Understanding severe accident phenomena is necessary for evaluating the extent to which a plant has design features which will prevent these severe accidents or, if necessary, to mitigate their consequences.

As most of you know, the NRC has sponsored an active research program on severe accidents in light water reactors for quite a few years. Substantial benefits have also been derived from our cooperative agreements on severe accident research with other countries. The focus of the research has been on specific severe accident phenomena that could result in early containment failure and also on code development.

Today's briefing will focus on staff accomplishments since the issuance of its last program

plan update in September 1992. The briefing will cover the current status of the severe accident issues, as well as closure plans for issues that have not yet been closed. The research results are also being used in support of the certification reviews for the evolutionary and for the advanced light water reactors. So, the staff will discuss not only the research program itself and its results, but also the use of those results in our regulatory programs.

Copies of the viewgraphs are said to be available at the entrances of the room.

Commissioners?

Mr. Taylor, would you proceed?

MR. TAYLOR: Good morning. With me at the table from the Office of Research, Eric Beckjord, Brian Sheron, Farouk Eltawila. From NRR, Doctor Tom Murley and Ashok Thadani.

Mr. Chairman, I won't repeat. You've outlined the basis upon which we do most of our work in the severe accident arena and its contributions to the review of the evolutionary and passive reactors. But as you'll hear from the staff this morning, this program has undergone a significant revision in its scope and priorities since being set up in 1987. A number of key issues associated with our understanding

of severe accidents have been resolved over the last few years and current plans are to resolve the remaining issues by 1996.

I'm pleased with the scope of the program that is currently ongoing and also the progress.

With that, I'll turn the meeting over to Eric, who will continue.

DOCTOR BECKJORD: Thank you.

(Slide) Mr. Chairman, Commissioners, slide 2 shows the presentation outline. I'm going to give a background in brief. Doctor Sheron will review the revised Severe Accident Research Program and the update since September 1992. Doctor Eltawila will give a presentation on fuel coolant interactions and debris coolability. Doctor Sheron will then review the utilization of the research results, plans for future work in the international programs, and if we have some time Doctor Eltawila will come back to the additional information.

(Slide) If I could have the third slide, please.

This is the background. Severe accidents are important safety considerations for several reasons. First, we're able to calculate in many cases severe accident loads from first engineering

principles, and they can exceed design limits in some cases. Also, we are able to estimate the frequency of severe accidents by means of probabilistic analysis and equipment failure rates that are derived from operating experience. The combination of these two flows of information by means of what we call the PRA shows that severe accidents are major contributions to risk. We know this from WASH-1400, from the update a few years ago of NUREG-1150. We know it from PRAs that have been done on individual plants over a period of 15 years and we also know it from the experience of Three Mile Island Unit 2.

The margins, as to margins in existing plants, we have studied the failure of primary systems and subsequent possible containment failures and we are able to evaluate the margins in plant designs to accommodate severe accident loads, the frequencies of the challenge and the probability given the challenge that the containment would fail.

The containment, the importance of containment has been recognized for a long time. I would say it goes back to certainly the Brookhaven Report, WASH-740 which came out many, many years ago. I think it was 1957 and this was clearly demonstrated at Three Mile Island.

The source term, which is listed here, has been investigated very thoroughly. After Three Mile Island, questions about the size of the source term came up and spurred an extensive review which was carried out by an office, the source term office established by the Commission. The conclusion was that the source term magnitude and the composition of radionuclides depends on accident sequences and on severe accident phenomena. It is not a single value for a reactor type and it had to be determined sequence by sequence.

As a consequence on that conclusion, the Severe Accident Research Program was undertaken in the early 1980s and its purpose was to develop an understanding of severe accident phenomenon and to provide a technical basis for regulatory decisions.

In 1985, the Commission approved a severe accident policy statement. That statement, in brief, said that based on then current knowledge of operating plants that there was no undue risk posed by the operation of those plants. However, the Commission called in that statement for a systematic evaluation of all operating plants to identify the vulnerabilities to severe accidents that could be corrected, all vulnerabilities and with focus on those

that could be corrected in a cost effective way.

The next item is NUREG-1150. That was an assessment of five nuclear power plants which, in effect, updated the technology of doing risk assessment from the WASH-1400, making use of the available results from the Severe Accident Research Program, which were beginning to come forth in the 1986 to '88 period. In 1988, there was issued a SECY paper, 88-147, on the subject of the individual plant examination, the Containment Performance Improvement Program, accident management, the Severe Accident Research Program and the use of operating experience. In effect, this document described how the Severe Accident Research Program was being focused on the vulnerabilities of severe accidents that had important risk significance. In particular, the core damage frequencies and the containment failure mechanisms and risks.

There have been several revisions of the Severe Accident Program since that time. The latest is NUREG-1365 and that is the December 1992 update and Doctor Sheron will take over to describe that to you.

DOCTOR SHERON: Thank you.

(Slide) The next slide, please.

As Jim and Eric said, in 1988 we undertook

a complete reexamination of the Severe Accident Research Program. Prior to that it had focused principally on just generating general information about severe accidents. What we wanted to do with this reexamination was to refocus it on those areas where there were larger remaining uncertainties and on the phenomena which seemed to be posing the highest risk in terms of severe accidents.

We did this reexamination and went through a rather extensive review process with the ACRS and the NSRRC. It was documented in NUREG-1365 and that was issued in August of '89. So, you can see this took us about a year to really refocus the program.

(Slide) Next slide, please.

The refocused program tried to focus in on four major areas. One was we tried to emphasize resolving uncertainties associated with scenarios that produced early containment failure because these are the scenarios which lead to the highest risk. The two major issues there were the Mark I -- the BWRs with the Mark I containment, the liner failure issue. What this was was basically that in a core melt accident you would postulate the melt, after it is released from the vessel would spread through the pedestal region into the dry well and contact the liner of the

containment. It was postulated that this would then burn through or fail the liner, which then gave you a direct pathway to the environment. The question was would this liner fail under these conditions.

The other main issue we were focusing on is the direct containment heating. This basically pertains to PWRs and a few postulated core melt accident with the primary system remaining at a relatively high pressure. When the vessel fails the high pressure steam that is in the vessel would drive the core material out into the key way, up into intermediate and perhaps even into the main compartments of the containment. This high temperature melt would interact very quickly with the containment atmosphere, causing an over pressurization and a failure.

Other areas were the source term research we were using to update the source term, core concrete interactions which generate the non-condensible gases basically that would cause a slow over pressure failure of the containment, hydrogen combustion which also contributes to the containment loads.

The other thing we focused on was our computer codes. At that time when we took a look at our suite of codes there were numerous codes which

seemed to duplicate one another. They were actually calculating the same thing or nearly the same thing, they were just being done at different laboratories or different groups. Our objective was to eliminate any duplication and just have one code to do a single job.

The other thing we also discovered was that many of our severe accident experiments really had not put together a very rational scaling analysis. By this I mean that we wanted to be able to take the results from smaller scale experiments and extrapolate them up to a large plant. The detailed scaling analysis, the dimensionless groups, whatever, that would provide that rationale had really not been developed to the extent that we would like. So, we focused in on developing a scaling rationale.

Then also, some of our other research was refocused. One area is the melt progression, the core melt progression, which we'll talk more about, fuel coolant interactions, which can provide a source of energy to possibly fail certain parts of the primary system and so forth, debris coolability which is an issue not only from the standpoint of accident management, also for the advanced passive reactors, and then hydrogen combustion.

(Slide) The next slide.

This just is a little cartoon which tries to illustrate where the loads on a containment come from during a severe accident. As you can see, pressure is the driving force. What we're worried about is really over pressurizing containment beyond its ultimate strength and failing it. The things that can contribute to increasing the pressure obviously is if you put a break in the primary system you'll release steam and possibly hydrogen from the primary system. Melt, as it relocates to the lower part of the vessel and perhaps out in -- if it fails the vessel into the containment, will interact with the concrete base mat. It will generate non-condensible gases, hydrogen. Also, the melt itself can interact with the containment atmosphere producing heating and a pressurization. All of these contribute to pressurizing the containment and ultimately to its possible failure.

(Slide) Next slide, please.

In the intervening period from 1988 to '92, after we had revised the program and refocused it, we in fact completed research on a number of these issues that I just mentioned. The first one being the Mark I liner failure issue. The results of that research demonstrated that the Mark I containment

integrity would be maintained provided water was added to cover the core debris. This was documented in NUREG/CR-5423 and it underwent a peer review, a very extensive peer review. As a result of that peer review, we identified four areas that required some confirmatory research to confirm the conclusions of the report. We set forth and did that research, carried it out, completed it. The results of that research were incorporated back into the report. It was then peer reviewed a second time to make sure that we had, in fact, addressed the issues of the first peer review. It was and the final report, NUREG/CR-6025, was issued in October of '93.

And we have taken the results of that research and we are using it as we review the IPEs right now. So, as we review IPEs, in particular the plants with the Mark I containments, we are looking at how they were dealing with the question of having reliable water supplies to provide water to the dry well.

(Slide) On the severe accident scaling methodology, we had a task group that was put together consisting of a number of experts in the country which met over a period of -- I guess it was well over a year, to put together this methodology. They applied

it as an example to the direct containment heating issue and it was done successfully and we now have a NUREG report, NUREG/CR-5809, which is sort of a guidance document for all of our contractors to follow. For any new experiments that are being considered, they have to do a thorough scaling analysis to show why the results of that experiment can in fact be extrapolated up.

On source term issues, our research program is essentially complete with regard to the quantity of the source term, the timing of it, its release and the chemistry or its composition. One thing we are doing right now is participating in the Phebus project in France which you may know something about. That's a large scale, an integral experiment designed to look at the source term behavior not only in the primary system but as its transported basically from the core to the containment.

In terms of our source term research, we are planning on updating the source term. There's a SECY-92-127, which i believe has already been sent down to the Commission last year, and we have a final report, NUREG-1465, which we plan to publish at the end of '93.

COMMISSIONER REMICK: Brian, before we go

on, I have a couple questions.

DOCTOR SHERON: Yes, sir.

COMMISSIONER REMICK: Do we know if other groups are using our scaling methodology that we've developed here?

DOCTOR SHERON: Other groups, you mean --

COMMISSIONER REMICK: International groups.

DOCTOR SHERON: International groups. Let me ask Farouk.

Are you aware of any?

DOCTOR ELTAWILA: I'm not aware of any, but we received a request for the report from different people in the -- member of the CSR, but we have not got any clear indication if they have used it or not.

COMMISSIONER REMICK: You indicate that the research program is complete in source term. What's the final resolution? I think there was a difference between EPRI and the staff on the low volatility radionuclides. Has there been anything more recent?

DOCTOR ELTAWILA: I think that difference still exists between us and the reason for the database that we have is not that complete to cover

all the expected range of severe accident. So, the staff still believe that the source term that we propose in that NUREG document is still adequate. So, I don't think that difference has been resolved yet.

COMMISSIONER REMICK: Nothing on that?

At the meeting in Seoul on advanced nuclear power system, Pierre Bashear from, I think, EDF, and some others made some strong statements about the need to get various international regulatory groups to come to closure on source term and there were statements that there are differences of several orders of magnitude between source terms in different countries. Would that be related to results from the Phebus project or would you have any idea what would be the basis of the concerns like that that different regulatory bodies were using source terms that differed by several orders of magnitude?

DOCTOR ELTAWILA: I think the main reason I think some of the other countries that they have not developed their database as much as we developed ours in the United States here and they are relying in a much more conservative estimate in their evaluation until they have their own information. I don't believe that the Phebus project is going to change our opinion about the source term. I believe it's going

to confirm that we are choose the source term. The source term that we are developed in this country is based on a lot of experimental data and the Phebus as an integral experiment is not going to assess each individual issue, but rather is going to give us the overall picture and our ability in calculating the source term can't confirm that rather than to produce any new information in that area.

COMMISSIONER ROGERS: Could I just --

COMMISSIONER REMICK: Excuse me. Go ahead, please.

COMMISSIONER ROGERS: -- piggyback on that a little bit?

Earlier you mentioned that the source term, in fact, depends very much on the particular reactor in the scenario that you're considering. The question is the variability in those results for U.S. reactors in different scenarios, is that outside of this gap between that Bashear is referring to of orders of magnitude difference between U.S. results and estimates done elsewhere.

COMMISSIONER REMICK: Yes. I have to say he did not say U.S. and elsewhere, he just said between different regulatory bodies, yes.

MR. TAYLOR: Maybe when we finish NUREG-

1465 we ought to be sure it gets distributed widely and then perhaps have a meeting of people on the subject in an international forum since this represents the best of the efforts that we have had.

COMMISSIONER ROGERS: That would sound like a very useful --

MR. TAYLOR: And I think that would be very useful among the people who practice severe accident --

COMMISSIONER REMICK: Yes. I did not read his comments with any difference of U.S. necessarily. It was kind of an appeal internationally to the need for regulatory bodies to come to closure and be consistent because otherwise it will lead to lack of public confidence.

MR. TAYLOR: Maybe if we stimulate that description.

COMMISSIONER REMICK: Yes.

MR. TAYLOR: That sounds to me if we're about to wrap up our review, it would be appropriate to do that.

Eric, how do you feel about that? I think that we could make those types of arrangements.

DOCTOR BECKJORD: Yes, I think so. Did he give a paper? Is it available? Could we --

COMMISSIONER REMICK: These were ad hoc comments --

DOCTOR BECKJORD: I see.

COMMISSIONER REMICK: -- of a discussion. I'm trying to think if Pierre gave a paper. I'm not sure. I'll check on that. But the comments came as part of a discussion of an international regulatory body, a harmonization basically.

DOCTOR BECKJORD: The director and some of his staff from the CEA's Institute of Safety are here this week and we can talk to them and see if they have some insight on this.

COMMISSIONER REMICK: Yes.

DOCTOR MURLEY: Could I add a point here? I don't think the situation is as divergent as its been portrayed. We've, in the context of the EPRI requirements document, of which EDF is a member of this, we've resolved the source term basically for evolutionary plants and also we've issued our FSER for the passive plant. So, we've at least got a regulatory resolution of how to proceed.

There will always be some loose ends, of course, but I don't see it as being a major issue that's going to affect the designs for the passive or the evolutionary.

COMMISSIONER REMICK: No, that certainly was not the point. In no way do I want to infer that there was a difference with U.S. source term. It was an appeal to international bodies to come together and on what should be the source terms used to have some kind of consistency. He claimed there were several orders of magnitude in different parts of the world. He said nothing specifically about U.S. and indicating that this can result in lack of public confidence if one nation are using this source term and another nation are using another and so forth.

DOCTOR MURLEY: I do agree that we need to understand any differences. It may be the difference between a source term for design basis accidents and a source term for severe accidents. It may be used differently.

COMMISSIONER REMICK: Yes.

DOCTOR THADANI: In fact, I think -- as Tom was saying, I think that's where the difference is. What source term do you apply for design basis accident? Do you include in-vessel as well as ex-vessel part? As Commissioner Rogers said, that's definitely driven by sequences and the source term will depend on specific sequences. The concept, however, in terms of design basis accident would be a

recoverable accident. So, perhaps one would only consider the in-vessel aspects. But for severe accidents, where you're looking at containment performance, you do have to look at the sequences and the challenges to the containment and actually what you would expect in terms of ex-vessel interactions as well. That's the approach we've been using in evolutionary and passive designs.

DOCTOR ELTAWILA: Can I add one more thing, just not to belabor the points here? There is an inconsistency in the definition of the source term. In the United States we define the source term as the amount of fission product in the containment itself. Some other country defines the source term about what's released outside the containment. So, if you take the effect of contamination due to the containment, that can be the reason for the differences. So, what we might need is a consistent definition of the source term, rather than the quantity of it.

DOCTOR SHERON: (Slide) Slide 9, please.

In the area of core concrete interactions, we've completed all of our dry core concrete interaction experimental research. By dry I mean that there is no overlying pool of water. We have

validated models that are currently available for assessing a plant's response and we are continuing to do additional validation on experiments that are being conducted worldwide, both in the U.S. as well as in the Russian Republic.

In hydrogen transport and combustion, a major effort has been completed on a deflagration to detonation transition criteria under a variety of conditions. This is one that transitions from burning to a detonation.

We have research going on right now on some residual issues related to high temperature hydrogen combustion research. This is going on at Brookhaven National Laboratory and it's a joint program between the NRC as well as NUPEC in Japan.

(Slide) Next slide, please.

We have just recently completed our TMI-2 vessel investigation project. This was -- I guess, as you know, we had a meeting in Boston just a week ago which was the final meeting in which we presented the results. What this program did is it provided a substantial amount of information to help us benchmark some aspects of our core melt progression models. We gained a lot of information about reactor vessel integrity. It gave us significant insights on

accident management and, most importantly, it allowed us to evaluate the margin of failure of the TMI vessel.

Some of the highlights we had uncovered from this program is that we saw a hot spot on the lower head of the vessel, about a median diameter. It reached a temperature of 1100 degrees C for about 30 minutes. But away from the hot spot the vessel temperature was below the 727 degree C transition temperature for the steel and, as a consequence, we were able to conclude that from that the vessel was not that close to failing under the conditions that were seen at TMI.

There was a number of other highlights there, I think, from the program. Instrument tube failures, for example, for the TMI type scenario are not considered to be a dominant failure mode. Creep rupture is the more controlling failure mode. The other thing we inferred from the results is that there is greater heat transfer between the core debris and the overlying pool of water. There may have been water ingress into the melt and that removed much more heat from the melt than what would be expected if we did our calculations with our current codes.

All of this indicated that there was

substantially more margin in that vessel than we would have predicted had we just done an outright calculation with our analysis codes.

COMMISSIONER REMICK: Have we attempted to calculate the progression of the melt with our codes, what we believed to be the case in TMI and if we did so how accurately were we able to do that? Have we attempted to use MELCOR or something?

DOCTOR BECKJORD: Yes, we did calculations.

Farouk?

DOCTOR ELTAWILA: There was an effort under the sponsorship of OECD, a comparison exercise using the different code and we used the SCDAP/RELAP code and the MELCOR code. But that was done about five years ago, an older version of this code. We are planning now after we add all the updates to our code to go over the scenario again and see how well we predicted.

COMMISSIONER ROGERS: Just before you leave that topic, as you know I was at that presentation last week and I just want to say that I think that whole project was an enormous success. I think it was very well managed and I think the NRC did a superb job in bringing the international effort

together, getting the support from overseas that was vital to be able to do the job as well as it was done, and the quality of the work. I couldn't stay for the whole meeting, but certainly what I heard was very impressive indeed. I think that it really was a superb piece of work. I think that a great deal of credit should go to NRC's management of that effort in bringing it all together. Fine effort.

DOCTOR BECKJORD: Thank you very much.

(Slide) Slide 11, please.

Last year, which was around four years since we started to look at our revision to the Severe Accident Research Program -- as you remember that was done in 1988 with the 1365 being issued in 1989 -- we decided that we ought to take another look and now update and see where we are again. What we did is we updated it through revision 1 of NUREG-1365 and what this revision done is it identifies the issues that have now been completed or are very near completion. It describes the progress and our understanding of important severe accident phenomena. It defines the research that would lead to closure of core melt progression, direct containment heating, which as I said before was an early containment failure mode, and fuel coolant interactions and it includes research

that would be applicable to the advanced passive light water reactors, especially in the area of containment performance criteria during severe accidents.

(Slide) Next slide, please.

In the revised Severe Accident Research Plan we're focusing our research on three phenomenological issues and on code development and assessment.

One thing I want to point out is, and I think I said this a couple weeks ago at our -- I may have mentioned this at the thermal hydraulic briefing, and important aspect of our code development program now is peer review. Each one of our major codes has or is undergoing a peer review. These are with outside experts, not within the agency, but people that are knowledgeable in the area with a wide variety of backgrounds, both in large codes as well as in phenomena and so forth. And the objective is to identify deficiencies in the code and to identify any necessary further work in order to say these codes have reached a good state of maturity and which we can have confidence in their application to severe accidents.

COMMISSIONER ROGERS: How was that being conducted? How was that peer review actually carried

out? Some of these codes are enormous things that are developed over many, many years. How are they actually doing a peer review of the codes?

DOCTOR BECKJORD: What we do is --

COMMISSIONER ROGERS: Are they piece by piece?

DOCTOR BECKJORD: No. No. It's for the whole code. Let me give you an example for MELCOR. What we did is we asked Los Alamos National Laboratory to head up the peer review effort. They had one individual who was acting as sort of the project manager or the coordinator. We wanted to do this just so that the NRC was still at arm's length from the peer review process. The individual then selected peer reviewers and I can't remember how many.

Do you remember how many there were for MELCOR?

DOCTOR ELTAWILA: Maybe eight or so.

DOCTOR BECKJORD: It was around eight or so. They had a wide variety of backgrounds. Some knew large codes. Some knew certain phenomena better than they knew others. So, we had a real spectrum. They would then go off and they would conduct a number of -- they would do their review. They would come back, they would periodically meet. During those

meetings the contractors, the developers of the code would give them presentations on the models, verification, whatever information that they would need. Sometimes the peer reviewers would need some additional calculations done in order that they could look at, say, sensitivities. So, our contractor would go back and do those calculations. Then this was all pulled together again by Los Alamos in the form of a final peer review report, which was then issued.

We, the NRC, then took the peer review report. We looked at the recommendations and the conclusions. We had to decide what things were important for the NRC, for regulatory. In other words, you would get a lot of recommendations, for example if some model may not have enough validation. But the question is was that important for the NRC's work? If it wasn't, then we may not want to go off and spend a lot of time or money working on that one item.

We then would take the peer review report and we would put together an implementation plan on how we would address the peer review comments. Then we would go forth and do that.

Farouk, do you want to add anything to that?

COMMISSIONER ROGERS: What were the most significant modifications or deficiencies that you felt should be addressed?

DOCTOR ELTAWILA: The most deficient things in our code right now is the late phase melt progression because the database is inadequate. About two years ago when we started that program the only database that we had is the TMI accident itself and we did not have any information about that. Some of the other important problem is the numerics with the code, lack of diversions and it's difficult to run. Documentation was an important issue and assessment of the codes. So, all of these we had an expedite, very aggressive program to address all of them. We still have additional database on late phase melt progression, but that has not been incorporated in the codes yet.

COMMISSIONER ROGERS: And the documentation, where does that stand>

DOCTOR ELTAWILA: The documentation right now, I think we improved the documentation tremendously and most of our codes right now, the documentation is very adequate. I think we have learned one lesson from the peer review process, that we should not let -- in the future, anybody develop a

code should not let the code developer start developing a code and to go along developing very fast without having a peer review process along the way.

So, what we have instituted in the division and in the branch in particular, that any time we have a modification to the code, before it is incorporated in the code we form smaller group of expert to review the model, to see if the database support the implementation of that model. Once we get the approval of the peer reviewer we give the code developer the go-ahead to add this model into the code. So, this way we'll not fall into trap that we have to review a very complex code like we did in the case of the MELCOR code.

COMMISSIONER ROGERS: Thank you.

COMMISSIONER REMICK: Has RELAP/SCDAP also had a peer review and have any issues there been resolved?

DOCTOR ELTAWILA: The SCDAP/RELAP has gone peer reviewed. The main issue that were resolved is the user friendliness, the documentation and the ability to run the code for full plant analysis. We have resolved all this problem. The peer review was just finished about six or eight months ago. So, we have a plan in fiscal year '94 to implement the high

priority issue of the peer reviewer comments.

DOCTOR BECKJORD: Okay. On direct containment heating closure, we've done a lot of work in this area. We've finished our scaling analysis, as I mentioned. We started running experiments again at the SURTSEY facility at Sandia. We also then found a way to utilize the containment technology test facility which is a one-sixth scale concrete containment model at Sandia. So, we had now experiments at Argonne which we had a small one-fortieth scale. We had the SURTSEY facility at Sandia, which was one-tenth scale. We then went to one-sixth scale. So, I think we have a real good base to extrapolate to the prototype plant on a DCH.

We have developed a resolution for DCH for the Zion plant. That report has been issued and is undergoing peer review. We also have a resolution plan now that has been issued for the Surry plant. These were the two designs that we looked at experimentally because of their difference in their lower compartment geometries.

The report on Surry came out about a month or month and a half after the Zion report. It also has a generic -- it's called the integrated report. It also looks at the general methodology of

extrapolating to other designs. Both these reports are undergoing peer review and will follow basically the same process we did on a Mark I issue.

COMMISSIONER REMICK: In the SECY document there was no mention of B&W plants. What is being done for the B&W plants.

DOCTOR BECKJORD: On direct containment heating?

COMMISSIONER REMICK: Yes.

DOCTOR BECKJORD: Direct containment heating, B&W plants for direct containment heating look very similar to a Westinghouse design. The plant that does not in this case is the combustion. The reason is because they do not have lower penetrations on their vessel head and they do not have that keyway to bring out the instrument tubes.

COMMISSIONER REMICK: I see.

DOCTOR BECKJORD: Right now we're looking at whether we have to run one or two more experiments to possibly get some information on the CE design.

(Slide) On core melt progression, early phase melt progression we think is fairly well understood. We've run a number of experiments, both in the U.S. as well as abroad to understand the early phases of core melt, the candling down of the cladding

and the early relocation. One remaining uncertainty is the question of blockage versus drainage, which applies primarily to BWRs. Because you have an ADS system you blow most of the coolant out of the core, out of the primary system during the blowdown. So, when the core melts, it is not melting where the lower part of the core is underwater still. So, there's a question as to whether when the melt comes down it will freeze and the cooler lower portions are whether it will run right down and relocate immediately to the lower head.

On fuel coolant interactions and debris coolability, we've completed most of our research on the fundamentals of fuel coolant interaction. We have work going on right now cooperatively at the FARO facility in Ispra, Italy involving a large scale integral test. But our ex-vessel debris coolability research results are still inconclusive.

These --

COMMISSIONER REMICK: How do you see that closure on that?

DOCTOR BECKJORD: On the fuel coolant interaction?

COMMISSIONER REMICK: Yes.

DOCTOR BECKJORD: Doctor Eltawila is going

to discuss that in his presentation.

COMMISSIONER REMICK: Okay.

DOCTOR BECKJORD: (Slide) On our computer codes we just talked about those. I'll just touch on these very quickly.

Our major severe accident codes are the MELCOR code, which is an integral risk assessment code. When I say risk assessment it's used primarily for risk assessment. This has been peer reviewed. The comments of the peer reviewers, as I said before, have been taken into account and we have work going on to incorporate these. I think most of it's already been done.

On the SCDAP/RELAP code, this is our detailed core melt progression code. The peer review is complete and we have a program underway right now to address the peer review comments.

The CONTAIN code, which is our containment analysis code, a detailed peer review was started in April of '93 and we expect it to be completed in about four months or five months, which would be in March '94.

We do have other specialized codes which we use for specific type phenomena analyses. These are HMS, which looks at hydrogen transport, COMMIX,

which we are looking at. That's a three dimensional computer code which looks at the containment behavior for the AP-600 where you're in a condensing environment.

(Slide) Next slide, please.

COMMISSIONER REMICK: For MELCOR, SCDAP/RELAP and CONTAIN, I'm really pleased with the fact that you are having the peer review and resolving issues. How about long-term maintenance? Do you have plans to make sure that once you've brought them up to date that we maintain the codes and the expertise knowledgeable in the codes over a longer period of time?

DOCTOR BECKJORD: I address that in my last slide.

COMMISSIONER REMICK: Okay.

DOCTOR BECKJORD: I just want to touch very quickly on some of the research facilities that the NRC sponsors. We have funded significant experimental research in facilities both in the U.S. as well as around the world. We've done in-vessel crust behavior studies in the Annular Core Research Reactor at Sandia. We have ex-reactor experiments to study this question of blockage versus drainage for the BWRs, also at Sandia. When we put these slides

together we also had a full-length core melt experiment scheduled at the Canadian NRU reactor. Unfortunately, after several delays on the part of the Canadians, we went to test our test bundle and found we had a leak in it. So, the test has been canceled.

As I said before, we have fuel coolant interaction experiments going on at the FARO facility in Ispra, Italy and I think you're probably aware we are now looking at a lower head coolability program using a RASPLAV facility in Moscow.

COMMISSIONER ROGERS: Could you say just a quick word or two about that RASPLAV program?

DOCTOR BECKJORD: Basically what it is is we're looking at a scale model of a cylindrical vessel lower head and what we want to understand are two things. One is the behavior of the melt in the lower head itself, the circulatory patterns and therefore the heat transfer. Then hopefully later on the question of having -- if there is water on the outside of the vessel, what kind of cooling that would provide.

The Russians right now are still designing the test facility itself. There's questions about whether we go to a complete hemispheric lower head or whether we use a slab geometry because the heating

obviously is by induction heating and the Lorenz force becomes very significant when you have a large hemispherical head and it may distort the phenomena that you're interested in.

DOCTOR BECKJORD: That will become another OECD Nuclear Energy Agency program. It's been approved and I think CS&I will put its blessing on it at the December meeting. As I think you've seen, there's a very strong logical follow-on to the TMI vessel program.

COMMISSIONER de PLANQUE: Brian, just a general question and a qualitative answer will do. If you look at all the experimental research being done internationally, what proportion of that research is being done in U.S. facilities and do you see a trend in the change of that proportion?

DOCTOR SHERON: In terms of number of experiments or in dollars per experiments?

COMMISSIONER de PLANQUE: Well, I don't know what unit I want it in. What will best describe the situation?

DOCTOR SHERON: Well, I mean, I would, and this is just my judgment, maybe Farouk -- I would say maybe about, what, two-thirds in the U.S., one-third -- maybe one-half, one-half. Okay?

In terms of dollars, there's a couple of major experiments. For example, the FARO facility is fairly high cost. We have very good leverage on that because we're only paying, what, 15 percent, I believe, of the cost?

DOCTOR ELTAWILA: We have the fixed cost of close to $1.75 million that we're sharing with EPRI, but the FARO or CEC and the European Community is paying the salaries and the costs of the facility itself, so we're going to run five experiments over there at that fixed cost regardless of the outcome of how long it takes or anything like that.

COMMISSIONER de PLANQUE: Do you see any indication that more and more of this work as we go on or get into the advanced reactors will have to be done at facilities outside the U.S.?

DOCTOR ELTAWILA: In the area of severe accident, we have not identified any new area that can affect the advanced light water reactor. The reason that some of this work now is starting, for example in Europe, and we don't have any of that work, they start very late compared to us.

We have our experimental program going on since the '80s, so we developed a wealth of database and now they are trying to develop their own database

and we are taking advantage of that by becoming a member of this international program to get the information at very cheap cost and we keep our expertise maintained and involved in this program at the same time. So although they have started, most of the work in severe accident will be in European countries. I think we have our own database that they are trying to catch up with.

COMMISSIONER de PLANQUE: Okay.

DOCTOR BECKJORD: I think I'd like to add something to that. I think it's very clear that there is a benefit to the international cooperation and I think that's recognized throughout the world and that has meant that there are now more cooperative programs outside of the U.S. whereas before ten years ago it was primarily inside the United States.

I think the direction here is going to depend very much on the direction of the nuclear industry in this country. If there's a strong nuclear industry, I think you will find that there will be a lot of projects that are carried out with international cooperation here.

COMMISSIONER de PLANQUE: Okay. Thank you.

DOCTOR SHERON: For the sake of time,

slide 16 really doesn't say anything other than list some facilities we have.

I'd like to just -- if you'd look at the monitors, we've got some much better pictures of some of these facilities.

COMMISSIONER REMICK: Brian, before going to that --

DOCTOR SHERON: I'm sorry.

COMMISSIONER REMICK: -- I've recently heard that the Japanese have a program of severe accidents called Alpha. It's not listed here. Do we have any involvement with an Alpha program on severe accidents?

DOCTOR ELTAWILA: Yes. As part of our severe accident research program, the Japanese are entitled to our data and we are entitled to their data. I apologize for not including it here in the viewgraphs.

COMMISSIONER REMICK: So we will get the data from their programs?

DOCTOR ELTAWILA: We will get the data.

DOCTOR SHERON: (Slide) Okay. If you could put the pictures on the monitor, the first one is a picture of the SURTSEY vessel. As you can see, this is one-tenth linear scale. This is at the Sandia

Laboratories. At the bottom, it's not very clear, but right inside the vessel in the lower part is where we have the melt generator which drives the melt into the containment and we measure the parameters like the hydrogen and the pressure and so forth.

COMMISSIONER REMICK: What is the melt material that they're using?

DOCTOR SHERON: This is usually Thermite, which is an iron-alumina compound.

(Slide) The next picture is the CTT facility, the Containment Technology Test facility. This is a one-sixth linear scale model of the Westinghouse containment, I believe, and this was actually tested previously where it was pressurized to see its failure mode, and so it was pressurized up to its ultimate strength and then it showed signs of failure. I think what happened was that the liner cracked and the pump that was pressurizing it just could not keep up with the leakage.

DOCTOR BECKJORD: Reinforced concrete.

DOCTOR SHERON: Yes.

DOCTOR BECKJORD: It got up to a little bit over 130 pounds per square inch.

DOCTOR SHERON: So, we decided that we could use that for our DCH testing as well.

(Slide) The next picture, please.

This is a picture of our high-temperature high-speed combustion facility. I think this picture was taken at the construction site where it was built. It has since been taken apart and relocated to Brookhaven and, if anyone is traveling to Brookhaven, I think in about another couple weeks you would be able to see this. It's put in a long tunnel down there and we will be looking at high-speed high-temperature combustion phenomena on hydrogen.

(Slide) Next slide, please.

This is just a schematic of the Phebus facility which will show you the mock-up of the primary system and the containment and the -- the steam generator is in there too and the pressurizer and, again, it's a matter of looking how fission products distribute in the core, the primary system and in the containment.

With this, I'm going to let Doctor Eltawila talk to you about the next two items on the agenda.

DOCTOR MURLEY: Brian, could I just add a point here?

DOCTOR SHERON: Yes.

DOCTOR MURLEY: From our perspective, this

is very valuable research and we do use it and I'll give one example.

Brian was talking about the TMI-2 vessel results and I had -- as we were wrapping up our reviews of the evolutionary plants and writing our FSERs, I wanted to know whether there had been any kind of integral insights, licensing insights, safety insights that have changed as a result of this research, and so I asked Ashok's staff what would have been -- would the containment have failed if the vessel had failed at TMI-2? Now, there was a preliminary study done by the Rogovin group right after the accident, but, in light of all this new information, I asked them to look into that and maybe he can say a few words.

The bottom line is that probably it would not have failed using the codes and the best methods that Research has developed as long as the containment sprays were available. If they were available, it would not have failed at all.

DOCTOR THADANI: Yes, that's basically what we concluded and in fact some of the calculations have helped us think about some helpful accident management strategies that one can develop to protect containment for long-term.

DOCTOR MURLEY: I think our FSER will include a detailed discussion of severe accident behavior of these evolutionary plants and it will of course use -- depend greatly, almost totally on the methods and research that has been developed in the last decade or so.

COMMISSIONER REMICK: That's a good introduction to a question I was going to ask later on, a question I was recently asked. How would you characterize how we are doing severe accident analysis in the reviews? It's not really a design basis approach, but how would you express how we are going about looking at severe accidents in the designs?

DOCTOR MURLEY: I can express it broadly and then perhaps Ashok can add. What we've said from the beginning is that a core melt accident, severe accident, is not a design basis accident in the classical sense because there what one does is insist on diversity and redundancy and basically gold-plating all the equipment that needs to deal with it and we said that's not necessary. But one must consider core melt accidents in the design and do what you can to mitigate their consequences and where it makes sense, where it seems to be cost-effective, put features in the design to cope with those things, so we're quite

confident that we have reduced the conditional containment failure probability given a core melt accident, but we've stopped short of all the paraphernalia that goes with making it a design basis accident.

COMMISSIONER REMICK: I think I didn't do too bad in providing an answer. One thing I did add is rather than conservative calculations too we used best estimate and realistic--

DOCTOR MURLEY: Yes.

COMMISSIONER REMICK: That's correct?

DOCTOR THADANI: That's right. And where there are questions -- I'll use an example. Initially, if you recall, when we came up with SECY-90-016 we said quench the core debris was our first thinking and then we realized the experimental basis couldn't really support that. Well, what was our goal? Our ultimate goal was to make sure that the containment integrity would be maintained for those accidents for some time period. You recall we talked about 24 hours, and that then required that we do some analysis and use some best estimate analysis to see what coolability considerations would be, what heat transfer mechanisms would be involved and if in fact one would impact some of the support structures and so

on. And it was these codes that we used to analyze to get an understanding of what could happen and how much time there would be before the containment would be pressurized beyond the ultimate point, so I would say that this has been very helpful and timely for us.

COMMISSIONER REMICK: Are the new designs using concrete aggregate that produces less of the noncondensibles or not?

DOCTOR THADANI: No. They are using aggregate that uses less gas production, but then there are some other downsides to that also. So that's the way they're going right now and that's what's in our, in fact, ITAACS.

COMMISSIONER REMICK: I see.

DOCTOR ELTAWILA: I would like to talk about two phenomenological areas of research that we are doing right now in the Accident Evaluation Branch. The first one is the fuel-coolant interaction and the second one related to the debris coolability, and they are related somehow. That's adding water or molten material and getting in contact with water.

But before I start the detailed discussion, I would like to make a point that whenever we talk about fuel-coolant interaction people always think of the alpha mode steam explosion that was

defined in the WASH-1400. There are other aspects of fuel-coolant interaction that we are interested in. I will repeat myself later, but we have concluded the alpha mode containment failure is a low probability event and we are not looking at it at all.

For example, during the TMI event when the material relocated into the lower head there was a mild fuel-coolant interaction which had some beneficial effect which resulted in the quenching of the material in the lower head and that's why we did not see the challenge in the lower head that we used to assume in our analyses before. So, because fuel-coolant interaction can occur in the vessel itself as a result of accident management through flooding the core, it can happen in the lower plenum as a result of molten material falling into the water coolant and it can happen ex-vessel as a result of reactor pressure vessel failure and pouring of the material into flooded cavity.

We have interest in pursuing the issue of fuel-coolant interaction and our main focus is try to identify what are the relocation scenario likely to lead into a coolable configuration, what are the effects of fuel coolant interaction on lower head integrity and on the cavity and pedestal integrity,

and what is the likely energy release and resulting load from containment? Again, we're not talking about the alpha mode failure. Here we're talking about the potential of producing large quantities of steam and hydrogen and see how can that charge the containment hydrostatically or quasi-static pressurization.

COMMISSIONER ROGERS: Excuse me. Just before you move off that, this question of the relocation scenarios, do you have in mind that it's the possibility of influencing the development of scenarios? In other words, accident management through this work or is that just beyond what you're looking at?

DOCTOR ELTAWILA: Absolutely not. I think we -- our position that any time you have a water -- you add the water as much as you can. It's not the idea of to influence the scenario itself. It's try to understand the consequences of adding the water, what a person observing severe accidents should one occur expect to happen as a result of adding the water at the different stages of the accident. So, he will have indication about -- should have an idea about the amount of steam that would produce, the amount of hydrogen that can be produced, where that hydrogen will go, how the water addition itself can influence

the source term, can influence the result the rest of the core melt progression.

So, it's not to alter anything, but to understand the consequences of the action that we'll be taking.

COMMISSIONER ROGERS: Well, that troubles me a little bit, frankly. This is not some abstract kind of scientific study. You don't want to just know what happens. If there's some possibility of a management decision being made to influence the evolution of the scenario, you want to be able to be able to pass that kind of information on. So, it sounded to me as if either there isn't any work going on in that or that that's -- I'm just uncomfortable about your answer on that one because -- maybe I misunderstood it, but it didn't seem to me that you can pull out of this what would be ultimately the ultimate most useful result, namely not what has happened, but how do you see that the worst thing doesn't happen when you're adding water. The rate at which you add it and when you add it.

DOCTOR THADANI: If I may comment on that. I think this is very valuable information as part of accident management considerations. In fact, this is the kind of information that the industry would be

looking at and in fact they are looking at today as part of their accident management program to see what kind of strategies they need to develop and what kind of information would be needed in the development of that strategy and the information that one needs is what comes out of an activity like this. There is a program in place that the industry has. They expect to implement the accident management program. They expect to implement by the middle of 1997. I mean there's a lot more to it, including the completion of IPEs and so on, but this is very useful information that goes into that program and assessment of what actions should be taken given one understands the state of conditions.

COMMISSIONER ROGERS: Yes, but I would hope that it wouldn't be just an open look feedback system, that somehow you'd close that loop so that whatever studies are being done also consider what the management actions might possibly be and then the consequences of those.

DOCTOR THADANI: I was being very simplistic, I admit.

COMMISSIONER ROGERS: Okay. All right.

DOCTOR THADANI: But yes, indeed. In fact, including what kind of instrumentation and

capability you need to be able to feed back and assess.

COMMISSIONER ROGERS: All right. Okay. That is part of your objective here.

DOCTOR THADANI: Yes, and also the decision making process. This is --

COMMISSIONER ROGERS: Now I hear NRR talking, but my problem was with an RES response. So, I would hope that you would be closing the loop internally here.

DOCTOR BECKJORD: The conclusion of my paper dealt with this at the Boston meeting last week and I think there is a very clear lesson from TMI-2 as to how you handle the scenario and the pressurized water reactor that's following that way.

There are still some questions that have to be answered and Doctor Eltawila is talking now about the fuel coolant interaction and that's one of the questions that needs to be dealt with. In particular, it's clear that you want to put water in the vessel. It's almost as important for the vessel as it is for cooling the core. If the vessel was overheated, and I think we can conclude now that the vessel will be overheated if the core melts and falls to the bottom, you also want to limit the pressure and

bring the pressure down. The question is how far to bring the pressure down and that's still an outstanding question that he is dealing with.

COMMISSIONER ROGERS: Okay.

DOCTOR BECKJORD: And then when you know that the core has been quenched, then you're free and clear. You just keep the water in there, keep the cooling going.

COMMISSIONER ROGERS: Sure. Okay.

DOCTOR ELTAWILA: (Slide) If I may jump to a viewgraph on page 23 just to try to describe the phenomena of fuel coolant interaction and I just would like to emphasize that this schematic here is a very old and does not represent our understanding right now, but I put it in totality here to see how far we came in understanding the issue of fuel coolant interaction as a result of the research.

When the molten core material is accumulated on the lower core plate, it was assumed in WASH-1400 that would be a catastrophic failure of the lower core plate and the material will enter into the water pool in the lower plenum. It will fragment and interact with water produced large quantities of steam. If an explosion occur as a result, for example, of the material impacting the lower plenum or

any of the plates located in the lower plenum and that shock wave propagates through the system, additional fragmentation will happen which provides more surface area which will interact with the steam and accelerate a slug of liquid material upward and impacting on the upper head of the vessel, dislodging it and impacting the containment and penetrating the containment. That was the WASH-1400 assessment.

Since then there was some other research here that indicated if we have enough energy to impact the upper lead and dislodging, that same energy would be able to fill the lower head and that will reduce the amount of energy available to impact on the upper head and that was the reason for concluding that the alpha mode failure is a low probability.

Again, right now, our research result, which I will discuss it in detail, indicate now there is not enough energy in the system to fill either the lower head or the upper head. So, instead of relying on a failure in the system to eliminate another failure, we believe that the failure mode of the vessel has been under control right now.

(Slide) Just to put our approach -- I'll go back to page 22 -- our approach in perspective, in 1985 the NRC formed the steam explosion review group

and that group concluded that the occurrence of a steam explosion was sufficient energy to fail containment has a very low probability. At that time the steam explosion review group used engineering judgment, used probabilistic approach and used subjective assessment about uncertain phenomena. They did not have enough database to help them put this issue to rest and they recommended additional research to try to explore the certain key energetic generic phenomenological aspect of fuel coolant interaction to address the complex variation in core melt progression, whether we have a large coherent pour, as they assume in the WASH-1400, or we can have a slender slow pour as happened at TMI and in accident management whether you add water early, in the early stage of core melt progression or at advanced stage of core melt progression.

So, we devised our research program based on the recommendation of the steam explosion review group and we performed a small scale experiment to look at the fundamental of the phenomena itself, which mainly the premixing and this is a hypothesis that when the material located into the lower head, the largest steam that occurred as a result of that relocation would pull the water out of the mixing zone

and will result in depletion of water in that region. So, you have a physical limit on the amount of material that can participate in fuel coolant interaction.

That was a hypothesis in 1989. We ran experimental program and we proved that the issue of water depletion is a real issue and now we have confidence in our assessment of these loads.

The second issue about additional fragmentation that can happen as a result of being in an explosion zone and we have completed our research in that area too. We have used the research result and developed analytical tools which have been validated using this experimental result and other results from other countries too.

(Slide) May I have page 24, please?

As Doctor Sheron indicated, that we are performing a large scale experiment with prototypic material, UO_2, ZrO_2, and Zirconium. And under prototypic conditions representative of accident in the in-vessel at the FARO facility, we hopefully will run a test later on this year to investigate the melt jet fragmentation, the quenching, the vapor generation, the hydrogen generation, any thermal load on the lower head and any erosion in the lower head.

By using our analytical tool that we developed based on separate effect tests to prepredict that experiment and then try to compare the result between prepredictionand after the experimental result, we'll be able to have additional confidence in our analytical tool.

Our approach to resolve fuel coolant interaction, like any other issue in severe accident, is using a probabilistic framework. Since there are too many variations of the accident scenario, we have to rely on developing a probabilistic framework like the one we did for DCH or the Mark I liner issue and we used the same approach, fuel coolant interaction issue.

The result from our fuel coolant interaction research so far indicated that the assumption, for example, that was made in WASH-1400 about a massive catastrophic failure of the coil support plate is unrealistic and that came from our core melt progression research. We've seen in the TMI that you get very minor break and relocation process. The fact of the water depletion phenomena again reduced the amount of material that can participate in fuel coolant interaction and reduce the energy that will be dissipated in the system either on the lower

head and the upper head. And as a result of that work, we reassess the alpha mode containment failure probability and will confirm the conclusion that we reached in 1985. That's indeed a low probability, but in addition to that we have concluded too that lower head failure is not expected to occur as a result of fuel coolant interaction.

We feel that our analytical code are adequate right now for doing fuel coolant interaction loads for in-vessel as well as ex-vessel. The issue of premixing and the fragmentation phenomena, which are the key for fuel coolant interaction phenomena, are really well understood right now. For ex-vessel phenomena we might need -- we're waiting for the FARO test program to see if any variation in the hole size or in the fall high, the water depth and so on might affect our perception about ex-vessel. But right now we have confidence that it might not change that much.

Our plan right now is to complete the ongoing research program which is mainly at FARO and at the University of Wisconsin. At the conclusion of that work we will convene a group of experts to identify the need for and the value of any additional research in that area.

COMMISSIONER REMICK: What do you foresee

as a schedule for convening that group?

DOCTOR ELTAWILA: Of course we're not going to wait until the end of the FARO program because that will take maybe about two or three years. What we expect to see after running one FARO experiment this year and finish the University of Wisconsin work, sometimes maybe by March, April of next year we can convene the group immediately after that.

COMMISSIONER REMICK: '94?

DOCTOR ELTAWILA: '94, yes.

(Slide) The second issue that I would like to discuss here is the issue of the debris coolability and what our interest is try to understand what is the effect on containment performance from debris-structure-water interaction. If we have core concrete interaction and add water as a result of accident management or as a result of design specification, like an advanced light water reactor, we want to see the effect on how much can the debris be cooled enough to arrest the core concrete interaction and further erosion and reduce the amount of non-condensible that enter in the containment. And at the same time we'll try to understand the mechanism that can lead into sustained coolable configuration.

We have run several experiments in the past at Sandia National Laboratory and we are a member of the MACE project which is run by EPRI and include, in addition to NRC, DOE, EPRI and other foreign countries.

The results from the past testing performed at the small scale showed that there are some potential for coolability. Immediately after adding water there are significant amount of energy released or removed from the melt. However, because of the configuration that we test -- run this experiment which are very small experiment, usually about 20 by 20 centimeter or 50 by 50 centimeter, the maximum that we ran, sustained debris coolability appears to be encumbered by the insulating crust formed on the debris surface that attached to the wall.

(Slide) If I may go to the figure on page 28 and try to explain that slide in details here.

In the left-hand side you see the crucible which you have the concrete at the bottom and on top of it we have a charge of uranium oxide, zirconium oxide and some concrete material. It's heated electrically and when the melt is complete the charge is melted and the core concrete interactions start.

Water is added on top of it to cool the debris. What happened in the experiment, if you look at the right-hand side, that you see that the dark line on the top, a crust formed and as the core concrete interaction continuing, that the surface of the melt keep receding down, which caused creating a void between the actual melt and the water on top of it. So, we were not able to remove heat from that.

COMMISSIONER REMICK: Just intuitively, it would seem like when you had the melt concrete interaction you'd be producing a lot of gases which would agitate the melt and break up the crust, but that does not happen?

DOCTOR ELTAWILA: Absolutely. That's exactly what should have happened. But unfortunately the crust is formed and because of the size of the crucible is so small it attached itself to that wall and will not be able to penetrate it.

COMMISSIONER REMICK: So, in the actual case, one might expect that the gases coming off from the concrete interaction would agitate and allow water to get into the melt, I would think.

DOCTOR ELTAWILA: In fact, in some of these tests we see some burst of material coming out, gases pumping out of that void, breaking the crust and

disputing material on top of the water and causing it to quench.

(Slide) Now I'll go back to page 27 and again that there are some believe among the researchers, as Commissioner Remick indicated, that the crust may be fractured by thermal mechanical loads and might be fractured by just the aspersion of the gases from the melt.

So, what the MACE project is planning for the next integral test, M3, is to run it as large a scale as possible and we are planning right now to run it at 1.2 meters by 1.2 meters. It's going to have a charge of 2000 kilogram of reactor melt material and 20 centimeter of debris bed.

We're currently assessing the usefulness of any additional testing versus accepting the current uncertainty if we can live with the fact that containment will not be pressurized beyond its capability within certain time as a result of core concrete interaction. That might be an acceptable answer to us.

(Slide) I'd like now to return back to Doctor Sheron.

DOCTOR SHERON: Okay. I would just point out that Ashok had told you -- you asked a question

about what kind of concrete they were using in advanced designs and you could argue that in one scenario you want to have a low gas content concrete that doesn't put a lot of non-condensible gas into the containment. But from the standpoint of breaking up across and allowing water ingress, you would like a lot of gas. So, it's not clear what's the best way.

I'd like to finish up here by talking about utilization of our research results and then talk a little bit about our future plans.

With regard to the results from all this research we've been doing, when we talked about the source term research we intend to modify 10 CFR Part 100 based on the new information. We've updated the source term for the advanced light water reactor which was documented in NUREG-1465. It's improved all of this research we've done and the area of severe accidents has, I think, significantly improved our understanding of containment performance. This has been reflected in our evaluation of the IPEs, the individual plant examinations, containment performance improvements as well as our reviews of the accident management programs being developed by the industry.

We've also used these results to close, as I said, on some of the severe accident issues that

were dominating the risk uncertainty, namely those that resulted in early containment failure. As I said before, we've already closed on the Mark I liner issue and put that one to bed and hopefully within a year we will have direct containment heating resolved for all of the PWRs.

COMMISSIONER REMICK: What's the anticipated schedule for Part 100? Not surprisingly, when I was in Taiwan and Korea recently there was a lot of interest in knowing the status of that. What is the schedule for sending that back up to the Commission?

MR. TAYLOR: I'm supposed to see something, an options paper, by November 19th, I believe, and that will then be coming up to the Commission before we receive -- this will provide the options. I think that's right.

DOCTOR SHERON: (Slide) Slide 30, please.

The other areas we utilized research results. One is obviously, I think as Tom said before, we're providing NRR with a database that they can use for evaluating the advanced light water reactor's performance against severe accidents. The areas that are being evaluated were documented in SECY-90-016 and also, as you know, we had an advanced

notice of proposed rulemaking on Part 50 to deal with severe accidents for future reactors.

Both of these specifically require technical basis for assessing the capability of the containment to withstand the severe accident loads and these loads, as I said before, they're just listed right here, the DCH, the FCI, direct melt attack on the concrete, the hydrogen and then the slow pressurization due to steam and non-condensibles.

And then lastly, understanding severe accident phenomena and their range of behavior is important for taking advantage of existing reactor and containment capabilities. This is primarily in exploring accident management strategies for the different containment types and developing plant performance criteria for future light water reactors against severe accident challenges.

So, I think in summary all of this research is being utilized rather extensively in various facets of the whole regulatory process.

(Slide) This will get to Commissioner Remick's concern from before with our future plans.

First is we would like to -- if all goes well, we'll complete our research on these major issues that we're still working on by 1996. That's

sort of a goal that we've set for ourselves. However, 1996 is not when we turn out the lights, close the door and walk away. We intend to maintain a level, albeit reduced, capability for the agency, for the staff to be able to evaluate and predict severe accident behavior. In order to do this we're going to need to identify the necessary disciplines for which expertise should be maintained. We need to identify the minimum suite of analysis codes that we will need to maintain. Once we've done that, then we will identify future efforts needed to maintain this expertise.

I think I said at the thermal hydraulic briefing you can't maintain the expertise by just keeping your researchers sitting around on their hands waiting to run a code or something. The way we would plan on doing this would be to continue our involvement in international programs, participate in international standard problems, participate in cooperative research with international partners and so forth. We would maintain a code development and assessment effort. I think this will provide the impetus for the researchers to want to improve the codes the same way as I said in the thermal hydraulic area. There may be a need to continue to run small

scale experiments, although we don't really under -- we don't see a need right now for any large major experimental facilities for severe accidents.

COMMISSIONER ROGERS: Before you leave that, just on this whole code question, I know Commissioner Remick is very interested in it. I'm interested in it too. But I'll tell you, I would be a little uncomfortable about developing codes to pursue code development unless there is a really important reason to do that other than to maintain the expertise of the people. I think that's a separate question of how to maintain expertise and whether codes need further development or not.

I think -- I wouldn't mix those two together automatically just to pursue code development, for fine tuning it a little bit more, I know, to keep people alive, their interest alive. I'd be very concerned about that because it seems to me at some point these things ought to stabilize and not change. Unless there is a good solid technical reason to pursue additional code development, I wouldn't -- I mean there are some systems that once you get -- once they're complete, they have to be complete. If you keep tweaking them, you're going to drive everybody crazy.

So, it's just an issue there, a bit of a concern that I have with respect to continued code development. I'm not opposed to it if there's a really important technical reason why codes need to be further developed, but just linking that with the necessity of maintaining expertise is a connection that to me is a doubtful one.

DOCTOR SHERON: Yes, I apologize. I didn't mean to leave the impression that we were just going to do this just to keep expertise.

COMMISSIONER ROGERS: It's sort of a gratuitous comment, but --

DOCTOR SHERON: Yes, I think on the areas where we feel very comfortable we're certainly not going to pursue any major code development just for the sake of it. But I think that in the area of severe accidents there's always going to be some areas where we're just not really totally comfortable and we could always feel we could do a little better. I think I said in the thermal hydraulic briefing that the area I see is going to be in the numerics. With the computer environment changing the way it is with these new machines coming on, with much more capabilities, as you saw the other day we can now run these large codes right here from NRC, whereas five

years ago that was a dream.

COMMISSIONER REMICK: I think that will help maintain the codes too since we have now capability in Research and in NRR to run some of these on work stations. Our own people will realize where there is a need for improving.

CHAIRMAN SELIN: Underground code testing.

DOCTOR MURLEY: Mr. Chairman, could I be excused?

CHAIRMAN SELIN: Yes, of course. Thank you, Doctor Murley.

Before you go on, Commissioner Remick, did you get your questions answered?

COMMISSIONER REMICK: Yes. I'm very pleased with that answer.

DOCTOR SHERON: And our long-term planning is needed right now to determine the requisite funding level to maintain this capability and that's something we're going to take an IOU because that's something we have to do.

(Slide) Just a last slide. I just wanted to touch on very quickly some of the international programs that we are involved in.

We have bilateral agreements with, as you see, that list the countries who have joined our

Cooperative Severe Accident Research Program. What joining our Cooperative Research Program means is that the participants will receive the NRC severe accident research results, our computer codes that we've developed and any other information that comes out of our program. In return, what we usually ask is that we receive some cash and, depending upon the size of the country, perhaps some in-kind contribution from that country. If they're doing research, then we would like to see their research results.

Just as our thermal hydraulic codes, our severe accident codes, I think, set the standard for the world. Most of the Europeans and even in Asia are relying extensively on our severe accident codes to analyze their designs.

That concludes our presentation. I see it's 11:30, which was the scheduled completion time, so I don't -- unless you would like, we won't propose to discuss anymore issues.

CHAIRMAN SELIN: I do have a couple of questions.

I was quite impressed with the presentation and in particular how you were able to tie the research work to needs in NRR and in the regulatory program. That's very positive and deserves

commendation.

I want to make sure I understand a remark that Doctor Eltawila made, your discussion of our databases being more effected by experimental work and therefore not having to have such large margins of computational -- well, just large computational margins. Is the implication that we use experimental results more than some of our more -- the other advanced colleagues overseas in doing the codes and in doing the regulatory work?

DOCTOR ELTAWILA: We're using the experimental result in assessing our code and try to expand the database to a large full-scale facility and we are using it more than our colleague overseas because we had that database available to us as a result of that research program that we have over the past ten years.

Our colleague overseas, they just start the process itself. They are developing their own code. Even the most advanced one, you know, they are in the beginning of developing their own code or performing their own experiments, so that's mainly the reason for the big difference.

CHAIRMAN SELIN: I must be missing something. I understand there's quite a bit of

communication about the experimental results and that our programs are available. Are people slow to pick them up or do they disagree with the way we're using them?

DOCTOR ELTAWILA: I don't see disagreement about the way we use them, but I think it's a matter of timing that they start developing their own capability over there. They have been participating in this program. They see the data. They have been using our codes, but they mainly in the past, especially in smaller countries, relied on the NRC for using the code, developing the database and try to just develop the code to a state of maturity that they can use them themselves. Nowadays they are using the code themselves. In addition to that, they are trying to assess it against either their own developed data or data that develop internationally. I don't know if I answered your question.

CHAIRMAN SELIN: You answered the question. I have to tell you that your answer is not consistent with my general view of the sophistication of some of these regulatory areas. I do believe that we invest more and run a larger regulatory research program than our colleagues, but I don't have the sense that the Germans or the French or even the

Swedes are slow to use the results and to integrate them.

COMMISSIONER de PLANQUE: Maybe what we're looking here for as an answer, are they repeating some of these experiments, duplicating them, or are they doing additional or variants on the experimental work that's been done here?

DOCTOR ELTAWILA: For example, the alpha program that Commissioner Remick mentioned is duplicating everything we have done in the past. They have done only one -- they have only one unique program about reactivity insertion accident. That's the only difference between their program and our previous program.

The German, we relied a lot on the German data, and there was, for example -- their research program is not as comprehensive as ours. They focus on the particular issue and they address it.

For example, if you look at the French, they have a major program ongoing right now. Most of that program, we have information about this result and it's sort of duplication of ours.

CHAIRMAN SELIN: We can leave this for future discussion.

I'm concerned about the image of a lot of

chauvinistic duplication of results rather than just accepting this as an international database and each one building on the other's results, but that's a big topic and I'd be interested in your insights at some future point as to what you think is happening in this international program.

I've had the image of a rather seamless international research program where we do some things and other people do some things but the answers are used sort of regardless of source and the image that you present is a little more nationalistic and chauvinistic than that.

DOCTOR BECKJORD: My sense, Mr. Chairman, is that they take some time to incorporate results, but I don't think they're duplicating in every case. Maybe the Japanese --

CHAIRMAN SELIN: This is not a good topic to enter in at 11:35 without some concrete example, but perhaps in the future we could have a little discussion about that.

MR. TAYLOR: Let's take a look at it.

CHAIRMAN SELIN: Commissioner Rogers?

COMMISSIONER ROGERS: Yes. We've had quite an extensive presentation and I've enjoyed it very much. I think that it was very helpful and there

are a lot of things we could talk about and go on all day, so I'll just touch on two things that I thought I might like to hear just a little bit about.

The scaling methodology, to what extent have those results been published or are going to be published in peer review journals that would be of interest to a broader community? I'm kind of interested perhaps along the way that Commissioner Remick started to touch on. It seems to me these are very fundamental kinds of problems, scaling of these phenomena experiments up to larger applications. And I don't know where they might be useful aside from the nuclear industry, but they might very well be in certain other applications and it would seem to me that it would be a good idea to try to see that that methodology gets published in a peer review journal that is widely read by interested people of different technologies, not just nuclear.

Excuse me?

DOCTOR SHERON: The scaling report, the NUREG report that was done, had not been published in any peer review journals. I think one of the reasons is that we kind of considered the whole development of it was a peer review process because we had 17 experts and --

COMMISSIONER ROGERS: I'm not thinking of it from the point of view of questioning the quality of it. I'm just saying that I think the dissemination of it will only be meaningful in peer review journals, not in internal documents, and for that reason I'm talking about peer review journals because those are the ones that everybody reads that looks at it from outside of this industry and that's the reason I'm stressing that.

DOCTOR SHERON: Well, let us take a look and see if there's a way we can get this published.

COMMISSIONER ROGERS: Because I think these scaling questions are very sticky questions and when people are grappling with them I'm sure that they would love to be able to use something that has the attention of a large collection of experts that our work has had.

DOCTOR SHERON: The information is available. It's published in NUREG reports, NUREG CR reports, but I don't think it's been really -- yes, I'm saying it's not available in peer and we're going to take a look and see if we can --

COMMISSIONER ROGERS: I'm saying, if somebody has got a problem and they're doing a literature search, they're not going to be looking in

NUREG reports if they're outside the nuclear industry.

DOCTOR SHERON: We will see if we can find a way to get these published in peer review journals.

COMMISSIONER ROGERS: I'd like to see that. I think that's very important.

The other thing is just on the codes, the international aspects of codes. Are we getting codes? I mean, our codes are being used and we're providing them to others internationally. What is the reciprocity on that? Are we getting codes from other cooperating bodies?

DOCTOR ELTAWILA: We have recently requested our colleague from CE to give us their ECAR II code and we have received it and given it to our contractor at Idaho National Laboratory to try to run it and benchmark it against our code.

In addition to that, we've received an older version of the Russian code, RASPLAV, but we have not received any update since about two years ago and we're looking for other updates in that area.

The other codes that are available, most of the other countries are using our codes so we rely on them, if they have for example an international center problem and things like that, to participate and we'll get feedback about user input, the effect of

user on the analyses and so on, but we have been getting these codes.

Some of the Japanese codes, we have not received them yet because they have not released them out to the public because they are not complete.

COMMISSIONER ROGERS: Thank you.

COMMISSIONER REMICK: I had not intended to raise it, but, since you mentioned thermal hydraulic codes and he talked about international sharing, anything new on CATHARE?

DOCTOR SHERON: I made a note here to find out if we've sent the request over.

COMMISSIONER REMICK: Okay.

DOCTOR SHERON: Let me take a commitment to get back to you on that.

COMMISSIONER REMICK: Fine. I just want to say that I'm very pleased with the presentation. I congratulate the staff on what they've done in the severe accident program area.

I think the refocusing that went on in 1988, and I think, Brian, you played a major role in that, was a significant development in bringing a chance of closure on many of these issues. It's obvious we've come a long way and I agree that internationally people look to these because it's

setting the pace throughout the world on severe accident programs and it seems to be now that, although even five years ago some countries didn't want to talk about the possibility of severe accidents, there seems to be a general consensus that one must look at severe accidents.

The work that has been done here I think has been quite significant. In preparing for your presentation, I went back and read an ACRS letter of about a year ago and I found it quite interesting because they were quite complimentary and ACRS doesn't tend to always complimentary. They say, "In general, the presentations were well organized and well presented and our questions were dealt with patiently and with good humor," and I thought, gee, that's a good idea to provide it with ice cream and make them happy.

CHAIRMAN SELIN: Commissioner de Planque?

COMMISSIONER de PLANQUE: I have no further questions, just to commend you on excellent work and an excellent presentation.

DOCTOR SHERON: Thank you.

CHAIRMAN SELIN: Thank you very much.

(Whereupon, at 11:43 a.m., the above-entitled matter 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 PROGRAM

PLACE OF MEETING: ROCKVILLE, MARYLAND

DATE OF MEETING: OCTOBER 26, 1993

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.

Carol Lynch

Reporter's name: Peter Lynch

SEVERE ACCIDENT RESEARCH STATUS

BRIEFING TO THE COMMISSION

Presented by

Eric S. Beckjord
Brian W. Sheron
Farouk Eltawila
Office of Nuclear Regulatory Research

October 26, 1993

Contact: F. Eltawila
492-3525

SEVERE ACCIDENT RESEARCH PROGRAM

OUTLINE

- Background
- Revised Severe Accident Research Program
  - Issues and Results
- Severe Accident Research Program Update since September 1992
  - Issues and Results
- Detailed Presentations on:
  - Fuel coolant interactions
  - Debris coolability
- Utilization of Research Results
- Future Plan
- International Programs
- Additional Information
  - Core melt progression
  - Hydrogen Combustion

SEVERE ACCIDENT RESEARCH PROGRAM
BACKGROUND

- Severe Accident Considerations: WASH-1400, other PRAs, TMI-2 and Chernobyl accidents, all tell us that severe accidents represent major contribution to risk from commercial nuclear power plants
- Margins (for severe accident challenges) in existing plants
- Even before the Chernobyl accident, importance of containment was recognized
- Source terms (accident sequences/phenomena)
- Severe Accident Research Program (SARP)
- Severe Accident Policy Statement
- NUREG - 1150
- SECY 88-147 (IPE, CPI, AM,SARP)
- NUREG-1365

REVISED SEVERE ACCIDENT RESEARCH PROGRAM

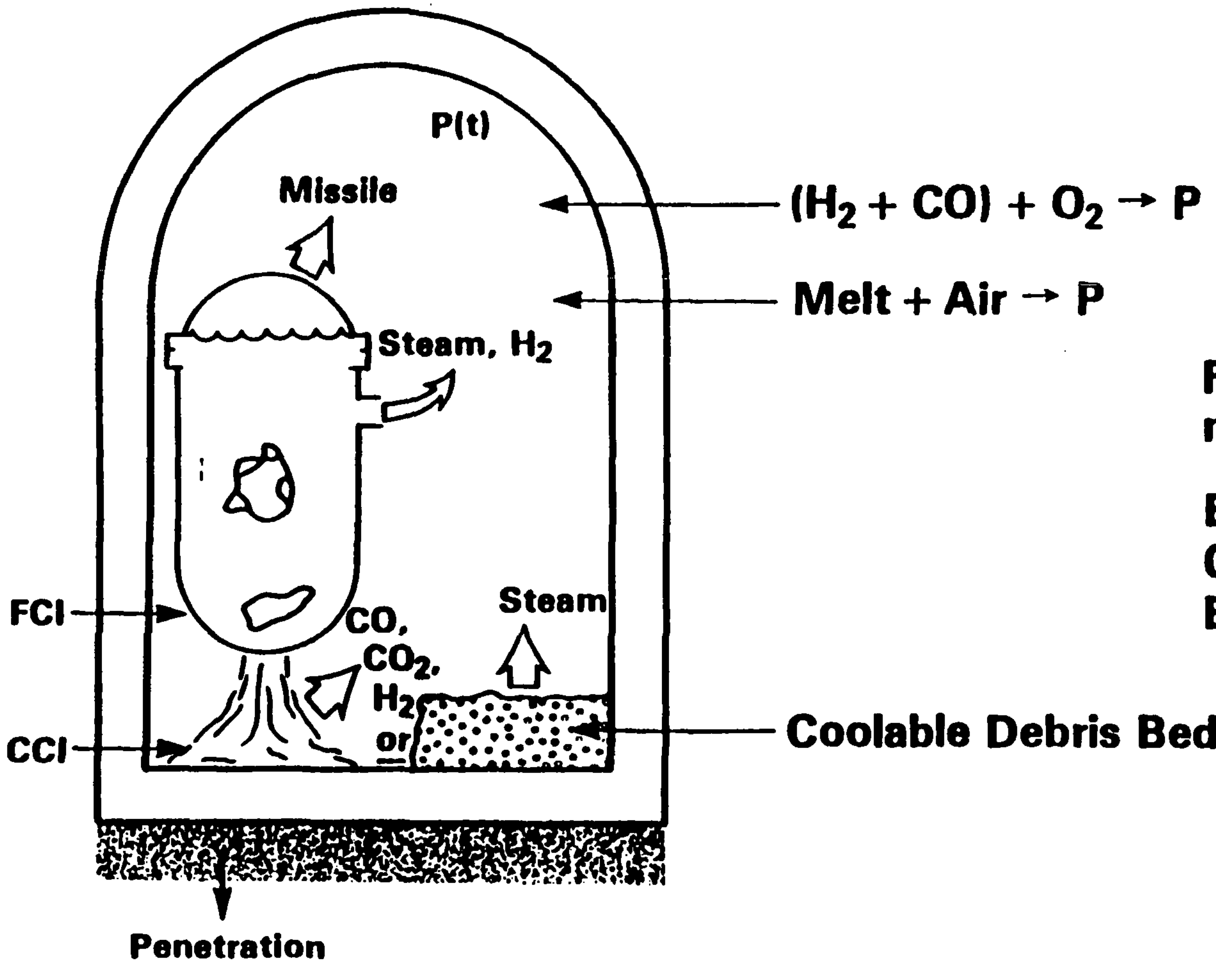
- In 1988, the staff undertook a complete re-examination of the severe accident research program. The objective of this re-examination was to refocus the research on those areas where remaining uncertainties still had a potentially large influence on risk, and where important residual phenomenological uncertainties still remained high.

- This re-examination of the severe accident program was documented as NUREG-1365, and was issued in August 1989. The program was extensively reviewed both internally as well as externally. Supportive letters were received from the ACRS and NSRRC.

REVISED SEVERE ACCIDENT RESEARCH PROGRAM (CONTINUED)

- The major elements of NUREG-1365 were:
  - Emphasis on resolving uncertainties associated with scenarios that produced early containment failure (Specific issues: Mark I containment liner failure, direct containment heating, source term update, core concrete interaction, H_2 combustion)
  - Eliminating duplication of analytical codes
  - Developing a scaling rationale for severe accident experiments
  - Refocusing research in selected areas (i.e., core melt progression, fuel coolant interactions and debris coolability, hydrogen combustion)

CONTAINMENT LOADS



Fuel Path is not Unique

Energetic vs. Quasi-Static Events

REVISED SEVERE ACCIDENT RESEARCH PROGRAM RESULTS

- In the intervening period from 1988 to 1992, the staff completed research on a number of these issues:

  - BWR Mark I liner failure

    - Demonstrated integrity of Mark I containments under core melt conditions provided water can be added to cover the core debris (NUREG/CR-5423) -- follow up work is complete. NUREG/CR-6025 was peer reviewed and issued in October 1993.

REVISED SEVERE ACCIDENT RESEARCH PROGRAM RESULTS (CONTINUED)

- Severe Accident Scaling Methodology
  - Effort was completed (NUREG/CR-5809) -- application to DCH by SNL and ANL
  - Developed method for applying scaled experimental results to full size plants

- Source Term Issues
  - Research program completed (source term, quantity, timing, chemistry)
  - We are participating in the Phebus project
  - Source term update is imminent [SECY-92-127], NUREG-1465 Final report to be published end of 1993

REVISED SEVERE ACCIDENT RESEARCH PROGRAM RESULTS (CONTINUED)

- Core Concrete Interaction
  - Completed all dry core concrete interaction experimental research
  - Validated models are available for assessing plants' response
  - Additional validation (against experiments conducted worldwide) is ongoing in USA and Russia

- Hydrogen Transport and Combustion
  - Major effort completed (deflagration-detonation criteria under variety of conditions).
  - Research on residual issues related to high-temperature hydrogen- combustion -- Joint NRC-NUPEC effort continues

REVISED SEVERE ACCIDENT RESEARCH PROGRAM RESULTS (CONTINUED)

- TMI-2 Vessel Investigation Project (OECD)
  - Program completed in October 1993
  - Provided information to benchmark some aspects of core melt progression and RV integrity
  - Evaluated margin to failure of reactor vessel
  - Insights on accident management

SARP UPDATE

- A SARP update (SECY-92-329; NUREG-1365 Rev. 1) was issued in September 1992:
  - Identifies issues that have been completed or are near completion
  - Describes the progress in our understanding of important severe accident phenomena
  - Defines the research that would lead to closure of core melt progression, direct containment heating (potential early containment failure mode) and fuel-coolant interactions
  - Includes research on issues applicable to ALWRs, especially the containment performance criteria during severe accidents

SARP MAJOR ACTIVITIES AND RECENT RESULTS

- Focus research on three phenomenological issues and code development and assessment

- An important aspect of the Code Development Program is the Peer Review. Each of our major codes has undergone a peer review to identify deficiencies and any necessary further work.

- Direct containment heating closure
    - New insights and improved data base allowed quantification of DCH loads for Zion and Surry.
    - Developed generic criteria for other types of containments and assess the need for any additional work
    - Peer review of the closure reports ongoing

SARP MAJOR ACTIVITIES AND RECENT RESULTS (CONTINUED)

- Core Melt Progression
  - Early-phase core melt progression is well understood
  - Remaining uncertainty related to
    - Blockage vs. drainage
    - Melt pool formation, growth and failure

- Fuel-Coolant Interactions and Debris Coolability
  - Completed most of the research on the fundamentals of FCI
  - Work at FARO, (Italy) involving large scale integral test
  - Ex-vessel debris coolability research results remain inconclusive

SARP MAJOR ACTIVITIES AND RECENT RESULTS (CONTINUED)

- Severe Accident Codes
  - MELCOR - risk assessment code -- peer review complete; program to address high priority peer reviewers comments is complete
  - SCDAP/RELAP - detailed core melt progression code -- peer review complete; program to address peer reviewers comments is underway
    - CONTAIN - detailed containment analysis code, peer review started April 1993 and will be completed March 1994
  - Other special codes for specific functions include HMS (H_2 transport) and COMMIX (AP600)

SEVERE ACCIDENT RESEARCH

- Facilities

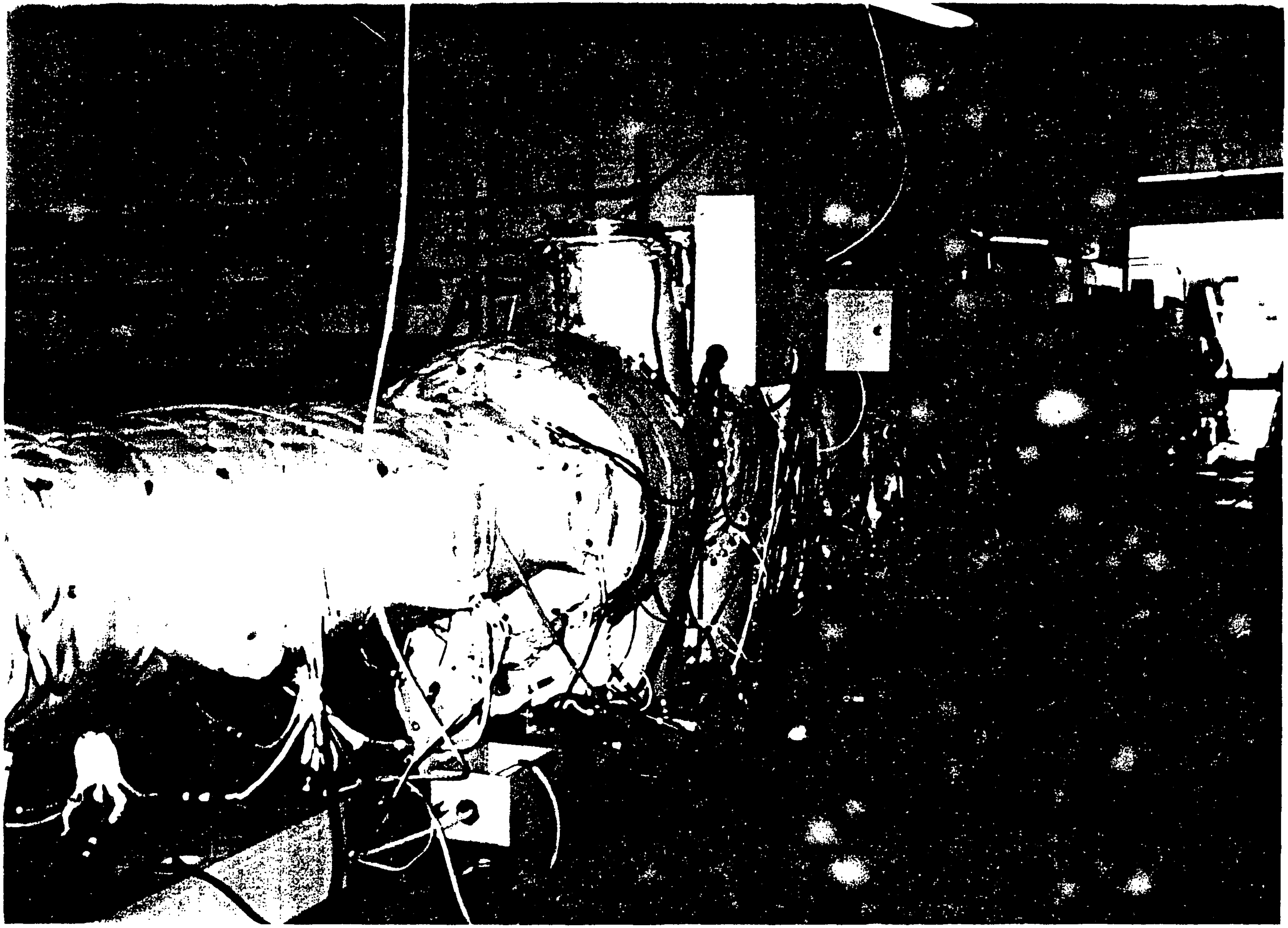
  - The staff has funded significant experimental research in facilities both in the U.S. as well as around the world.

    - In-vessel crust behavior studies in Annular Core Research Reactor (SNL)
    - Ex-reactor experiments to study blockage vs drainage issue (SNL)
    - Full length core melt experiment in the Canadian NRU reactor
    - Fuel Coolant interaction experiments in the FARO facility (Ispra, Italy)
    - Lower vessel head coolability in the RASPLAV facility (Moscow)

SEVERE ACCIDENT RESEARCH (CONTINUED)

- DCH testing in SURTSEY facility, CTT facility (SNL), and the COREXIT facility (ANL)
- Debris coolability experiments (MACE test at ANL)
- Hydrogen detonation testing at BNL (cooperative with Japanese); and Russia (Kurchatov)
- Fission Product transport in the core region, RCS, and containment (Phebus reactor, France)

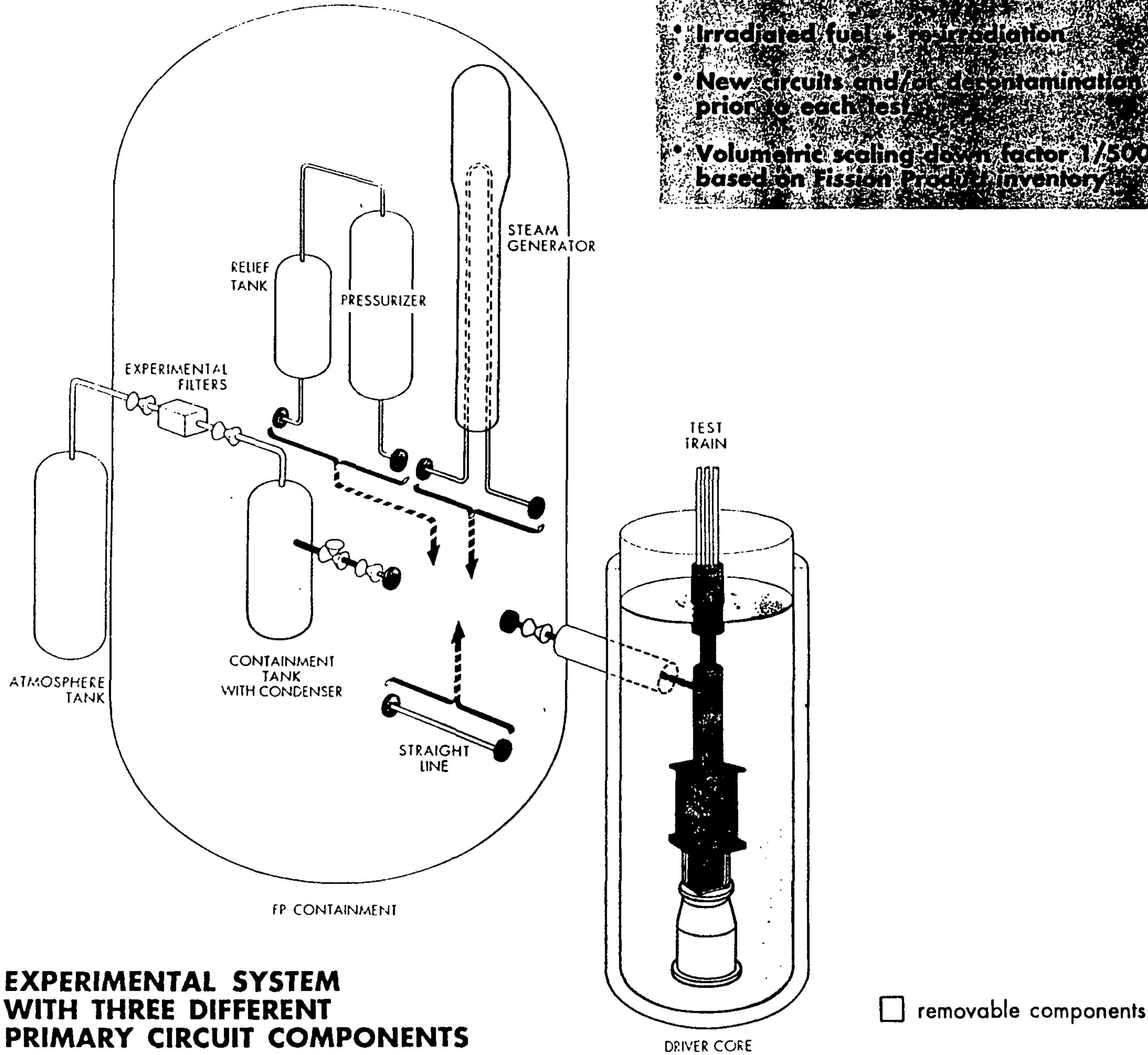

SCALE
TEST



GENERAL CHARACTERISTICS OF THE FACILITY

In pile test

- **Main plant components simulated**
- **Irradiated fuel + reirradiation**
- **New circuits and/or decontamination prior to each test**
- **Volumetric scaling down factor 1/5000, based on Fission Products inventory**



EXPERIMENTAL SYSTEM WITH THREE DIFFERENT PRIMARY CIRCUIT COMPONENTS

FUEL COOLANT INTERACTIONS

Fuel coolant interactions do not imply only steam explosions (α-mode containment failure); TMI-2 event exhibits another, in this case, beneficial, example of effects of FCI on melt quenching

Issues:

- What are the relocation scenarios likely to lead to coolable configuration?
- What are the effects of FCI on lower head integrity and cavity/pedestal integrity?
- What is the likely energy release and resulting load on containment?

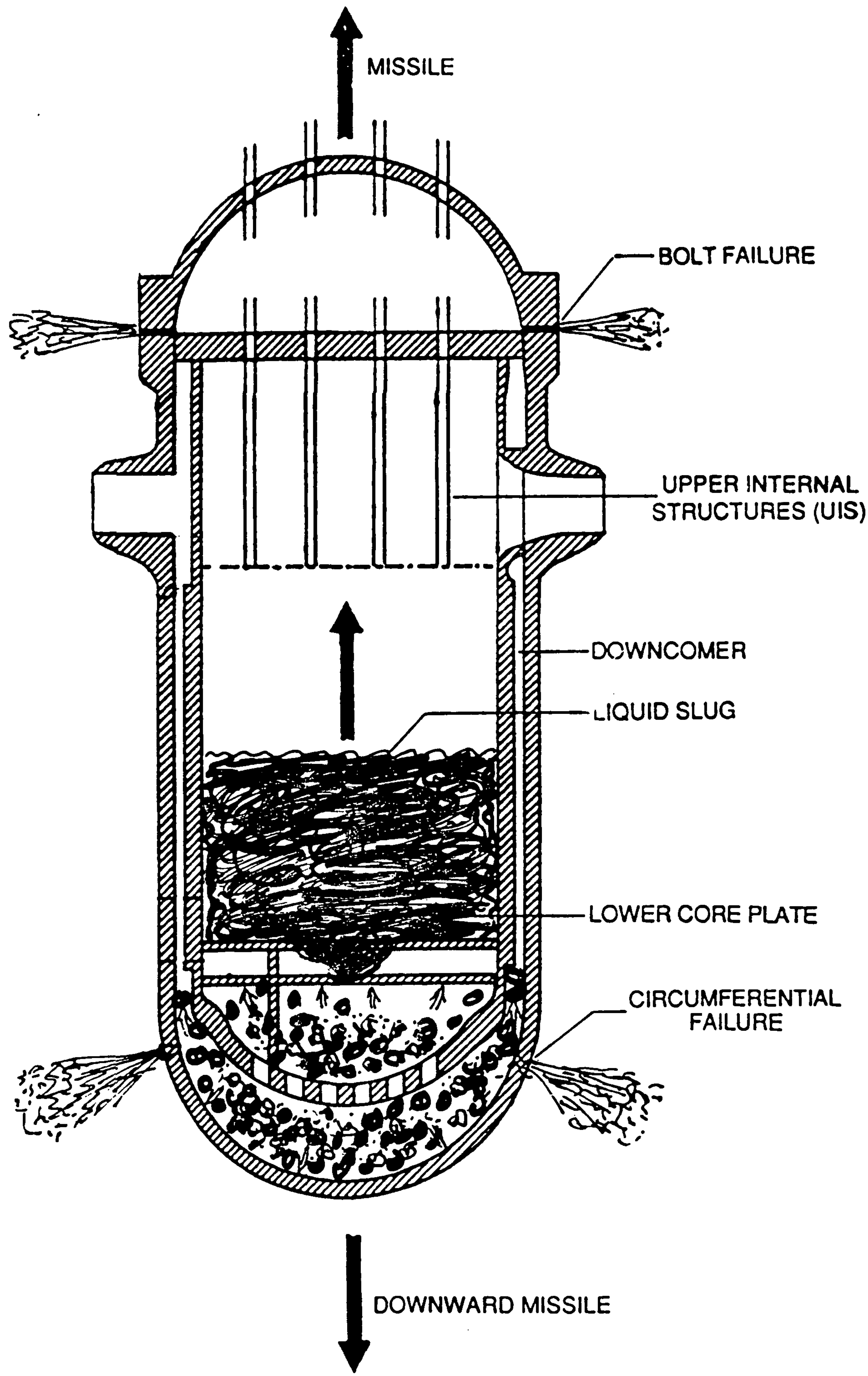
FUEL COOLANT INTERACTIONS
FCI (CONTINUED)

Approach:

- Explore certain key, generic phenomenological aspects of FCI to address the complex variations found in core melt progression and accident management.

- Small-scale experiments to explore the fundamentals of FCI (premixing, propagation/fragmentation, yield)

- Develop/assess analytical models


MISSILE
BOLT FAILURE
UPPER INTERNAL
STRUCTURES (UIS)
DOWNCOMER
LIQUID SLUG
LOWER CORE PLATE
CIRCUMFERENTIAL
FAILURE
DOWNWARD MISSILE

FUEL-COOLANT INTERACTIONS
FCI (CONTINUED)

- Large-scale experiments with prototypic melt and prototypic conditions to investigate melt jet fragmentation, quenching, vapor generation, thermal load and possible erosion on the lower head -- FARO (Ispra)

- Use probabilistic framework to couple the different stages of FCI -- used to quantify the α-mode failure

FUEL-COOLANT INTERACTIONS
FCI (CONTINUED)

Results:

- α-mode of containment failure is low probability
- Initial assessment of loads due to FCI is possible
- Premixing and fragmentation phenomena are adequately understood
- More research may be needed on ex-vessel fuel coolant interactions and to establish accident management options

Plan:

- Complete the ongoing programs
- Convene a group of experts to identify the need for and value of any additional research

DEBRIS COOLABILITY PROGRAM

Issue:

- What is the effect on containment performance from debris-structure-water interactions?
- What are the mechanisms, if any, by which molten corium, released from the RPV, may be maintained as a coolable debris bed?

Approach:

- Limited number of ex-vessel coolability tests conducted at small scales

Results:

- Following water addition, significant amount of energy was extracted from the melt
- No firm demonstration of yet long-term coolability, and results still inconclusive

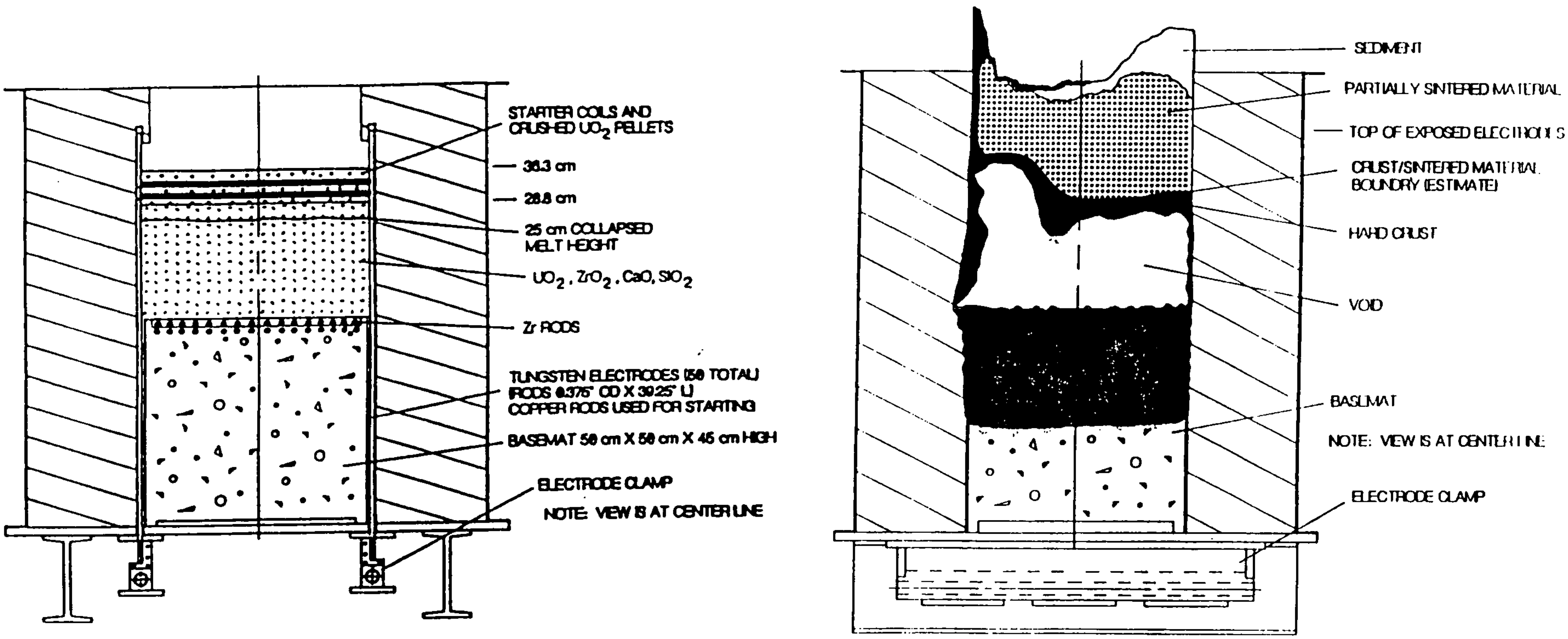
DEBRIS COOLABILITY PROGRAM
(CONTINUED)

- Demonstration of debris coolability may be hampered by physical scale of test apparatus; at larger scale any top crust more likely to be fractured by thermal/mechanical stresses

Plan:

- MACE project planning for next integral test (M3) (February 1994)
  - Large scale 1.2m x 1.2m (2000 Kg reactor melt, 20 cm deep debris bed)

- Currently assessing usefulness of additional testing versus accepting current uncertainty



Test Section Before and After the Test

UTILIZATION OF THE RESEARCH RESULTS

- Source Term Research

  - Modification to 10CFR Part 100
  - Updated source term for ALWR (NUREG-1465)

- Improved understanding of containment performance

  - Individual Plant Examinations, Containment Performance Improvements, and Accident Management

- Closure of severe accident issues that could potentially lead to early containment failure (Mark I liner failure and direct containment heating)

UTILIZATION OF RESEARCH RESULTS (CONTINUED)

- Provide data for severe accident plant performance criteria for ALWRs
  - SECY 90-016
  - Advance Notice of Proposed Rulemaking (Part 50)
- Specifically SECY 90-016 and ANPR require technical basis for assessing capability of the containment to withstand the severe accident loads:
  - Direct containment heating
  - Energetic fuel coolant interactions
  - Direct melt attack
  - Hydrogen combustion and detonation
  - Slow pressurization (steam, non-condensibles)

UTILIZATION OF RESEARCH RESULTS (CONTINUED)

- Understanding the severe accident phenomena and their range of behavior is important for taking advantage of existing reactor and containment capabilities; exploring accident management strategies for different containment types; developing plant performance criteria for future LWRs against severe accident challenges

FUTURE PLANS

- We plan to complete research needed to close on major remaining issues by 1996

- However, staff intends to maintain reduced level of capability to evaluate and predict severe accident behavior beyond 1996

  - staff will identify necessary disciplines for which expertise should be maintained
  - staff will identify minimum suite of analysis codes to be maintained
  - future efforts will involve moderate level of continuing research to maintain expertise (involvement in international programs, maintain code development and assessment efforts, possible small scale experiments)

FUTURE PLANS (CONTINUED)

- Staff will continue to address issues as they arise
- Plan to maintain cooperation with foreign participants

- Long term planning needed to determine requisite funding level to maintain capability - in progress

INTERNATIONAL PROGRAMS

- NRC has bilateral agreements with the following countries who have joined the Cooperative Severe Accident Research Program:

Bulgaria	Czech Republic	Finland
France	Germany	Hungary
Italy	Japan	JRC
Korea	Lithuania	Netherlands
NUPEC	Russia	Slovak Republic
Spain	Sweden	Switzerland
Taiwan	UK	

- Participants receive NRC Severe Accident Research Results, codes, etc.
- NRC receives some cash and in-kind contribution
- NRC codes are used extensively internationally

ADDITIONAL INFORMATION

CORE MELT PROGRESSION

Issue:

- What are the characteristics of the melt released from the core and the vessel?
  - Need melt mass, temperature composition (Metallic or ceramic), and rate and time of release
  - Used as input for assessment of vessel integrity, containment loads, hydrogen production, fission products.
- What are threshold and location of ceramic pool meltthrough from a blocked core?
- Does metallic melt drainage form a core blockage in BWR dry core accidents?

Approach:

- Ex-reactor experiments (blockage vs. drainage)
- In-reactor experiments (ceramic pool meltthrough)
- Analysis

CORE MELT PROGRESSION (CONTINUED)

Results:

- Blocked core gives large ceramic melt release like TMI-2; very low metal content ($\sim$ 1% at TMI-2)
- Ceramic melt release occurs on failure of the growing ceramic crust that contains the melt pool.
- If metallic melt drains from the reactor core, the later release from the lower head (following dry out) is lower temperature metallic melt

CORE MELT PROGRESSION (CONTINUED)

Plan:

- Metallic melt drainage or core blockage
  - Plan two simplified XR1 tests and four XR2 experiments within the next year
- Meltthrough from a blocked core
  - Performed MP-1 and MP-2 experiments in ACRR
- Analysis of the data from experiments is ongoing
- In late 1993, review group will consider current results and the adequacy of understanding, and the need for and likely benefits of further research

HYDROGEN COMBUSTION

Issue:

- At high temperature the limiting concentration for detonability may approach the deflagration limit (for downward propagation)
- High-temperature high-speed hydrogen combustion phenomena:
  - Effect of elevated temperatures on inherent detonability and on deflagration to detonation transition

Results:

- Higher temperatures increase the mixture sensitivity
- Steam addition to hydrogen-air mixture decreases the mixture sensitivity
- Detonations observed to propagate in hydrogen-air mixtures with 10% hydrogen at 650K (no steam)

HIGH-SPEED HIGH-SPEED
HYDROGEN COMBUSTION PROGRAM

Plans

- Complete construction of the high-temperature combustion facility (HTCF)

- Use the HTCF to perform:
  - Inherent detonability experiments,
  - Deflagration to detonation transition experiments with and without venting,

- Experiments will start in January 1994

- Program will be completed in April 1996