UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Title: BRIEFING ON NRC THERMAL-HYDRAULIC RESEARCH PROGRAM

Location: ROCKVILLE, MARYLAND

Date: AUGUST 3, 1989

Pages: 68 PAGES

NEAL R. GROSS AND CO., INC.

COURT REPORTERS AND TRANSCRIBERS
1323 Rhode Island Avenue, Northwest
Washington, D.C. 20005
(202) 234-4433

DISCLAIMER

This is an unofficial transcript of a meeting of the United States Nuclear Regulatory Commission held on August 3, 1989, in the Commission's office at One White Flint North, Rockville, Maryland. The meeting was open to public attendance and observation. This transcript has not been reviewed, corrected or edited, and it may contain inaccuracies.

The transcript is intended solely for general informational purposes. As provided by 10 CFR 9.103, it is not part of the formal or informal record of decision of the matters discussed. Expressions of opinion in this transcript do not necessarily reflect final determination or beliefs. No pleading or other paper may be filed with the Commission in any proceeding as the result of, or addressed to, any statement or argument contained herein, except as the Commission may authorize.

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

- - - -

BRIEFING ON NRC THERMAL-HYDRAULIC RESEARCH PROGRAM

- - - -

PUBLIC MEETING

Nuclear Regulatory Commission
One White Flint North
Rockville, Maryland

Thursday, August 3, 1989

The Commission met in open session, pursuant to notice, at 2:30 p.m., Kenneth M. Carr, Chairman, presiding.

COMMISSIONERS PRESENT:

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

STAFF AND PRESENTERS SEATED AT THE COMMISSION TABLE:

SAMUEL J. CHILK, Secretary

WILLIAM C. PARLER, General Counsel

JAMES TAYLOR, Deputy Executive Director for Operations

ERIC S. BECKJORD, Director, Office of Research

BRIAN W. SHERON, Director, Division of Systems Research, NRR

LOUIS M. SHOTKIN, Chief, Reactor and Plant Systems Branch, Division of Systems Research, NRR

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

2:30 p.m.

CHAIRMAN CARR: Good afternoon, ladies and gentlemen.

The purpose of today's meeting is for the NRC staff to brief the Commission on the thermal-hydraulic research program. The Advisory Committee on Reactor Safeguards commented on the program by a letter dated June 15th, 1989 and the Commission would like the staff to specifically address those comments.

Copies of the slide presentation are available at the entrance to the meeting room.

Do my fellow Commissioners have any opening comments?

If not, Mr. Taylor, you may proceed.

MR. TAYLOR: Good afternoon, sir. To my left is Eric Beckjord, the Director of Research and to my right, Brian Sheron and Lou Shotkin, also from the Office of Research.

This has been a very important program within the Office of Research and has given us many good results which we have used in regulatory application.

I'll now ask Eric to start the formal briefing.

MR. BECKJORD: Thank you.

Mr. Chairman, nuclear engineering is the beneficiary of technologies in heat transfer and fluid flow, that have their roots long before nuclear reactors. Nuclear engineering itself has invested great resources first in the problems of critical heat flux at reactor fuel elements and determination of steady state conditions of reactors in normal operation. Second in the problems of transient performance of reactors and third in their performance during loss-of-coolant accidents and the functioning of emergency core cooling systems.

NRC research and the AEC before it initiated the loss-of-coolant and the emergency core cooling system work and that was more than 20 years ago. That effort in terms of experimental confirmation of emergency core cooling system performance has been very successful in terms of the regulatory needs which have been defined in the research philosophy.

Today, we're, in a sense, declaring victory in these efforts, specifically the major experiments are done, proposals been developed and it's now time to propose what is needed in thermal-hydraulic research to maintain the technical base, for surely that base will be needed as long as reactors are operating.

In fact, we've already moved to make major changes, scaling annual expenditures down from a few years ago at a level of about $20 million to something of the order of $5 million annually in another year.

There are other higher risk matters and, hence, higher priority matters to turn our attention to. I mention a few of these. They include the revised severe accident research program, accident management developments, human factors, the expansion of reactor aging research, radiation protection and health effects and I don't mention the expense to complete the risk studies in 1150 which have exceeded their early cost projections.

In a time of declining research budgets overall, it is the reduction of expenditures in thermal-hydraulics that has made it possible to fund these other activities, in some cases in total, in some cases in part. And this is properly done, in my opinion.

Today, you're going to hear in some detail of the research plans for maintaining thermal-hydraulic technology at an appropriate level so that NRC will have the skills available when needed in terms of people and operating codes for analysis, codes such as TRAC, RELAP, and RAMONA and there are many others.

The thermal-hydraulic skills today are being used in these new efforts as well to support the severe accident research program and the accident management program. And I believe this will continue to be the case.

This is not to say that nothing remains to be done. The advanced reactors, as they are developed, will pose new problems and issues in the operation of natural circulation systems and in what I will call self-energized safety systems because I don't like the adjective "passive" in this context. NRC should watch these developments carefully to determine to what extent and depth additional research may be needed.

I believe the program that we are proposing to you based on the $108 million fiscal year 1990 research budget and including a few small-scale testing loops at universities and also code maintenance, will maintain the capability to make an appropriate judgment and response on these new issues as they arise, in particular as the details of the advanced reactor designs unfold.

With that, I'd like to turn it over to

Doctor Sheron.

DOCTOR SHERON: Thank you.

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

Just quickly on this slide, I'll go through a brief history of where we've been.

As Eric said, we've been developing these codes. I have 15 plus years. Twenty is probably even a better estimate. We've gone from -- the code development and verification effort has gone from the large break loss-of-coolant accident from the early '70s to right before TMI when we started to focus in on a small break loss-of-coolant accidents. TMI sort of fixed our attention on that and we spent considerable effort making sure we understood small break behavior in light water reactors.

In the early '80s, there was a number of transients that occurred at plants. I believe Saint Lucie had one, Davis-Besse, several others that focused our attention now on transients and our ability to understand and be able to calculate transient performance of these plants.

And then, most recently, in particular with regard to accident management, we've been focusing in on making sure that these codes can and are capable of predicting the front end of the risk dominance severe accident sequences.

Based on all of this, we've concluded that the codes have, in fact, reached an acceptable level of maturity for calculating performance of the current generation of light water reactors. Basically, this conclusion is reached because we don't believe that any more major changes in these codes are going to substantially change our understanding of performance or the consequences of accidents.

We've also, in reaching this conclusion, went forward and revised the ECCS rule, as you know. We have quantified the uncertainty in the large break LOCA analysis using a method that has been extensively peer reviewed. We are now in the process of confirming the applicability of that method by applying it to a small break. That should be finished within about a year, I believe.

(Slide) Could I have the next slide, please?

Based on this history, what we've done is we've sat down and we've asked ourselves what are our planning and direction objectives over the next foreseeable future, considering that most of our major experimental programs are finishing up. LOFT, as you know, is finished up. SEMISCALE has been closed down. The international facilities we're involved in, in particular the 2D/3D program is finishing up in a couple months. So, we had to ask ourselves the question, where are we heading with the program now?

We identified some major overall objectives. The first is that the Agency needs to maintain a thermal-hydraulic analysis capability at a minimum level. The question, obviously, is what is a minimum level? We'll talk about that in a little bit.

The other is to take these computer codes, which we have developed over the years, and to apply them to reactor issues in a proactive fashion. Past issues we've looked at, you'll see listed here, are obviously the loss-of-coolant accident. We've used the codes to help us in coming up with a pressurized thermal shock rule. We looked at the consequences of the TMI accident, the major USI-845 decay heat removal, the codes were used. It has helped us with the ATWS rule. Most recently we are looking at the BWR stability for NRR as a result of the La Salle event.

(Slide) Next slide, please.

CHAIRMAN CARR: The consequences of the Three Mile accident, you say you used them. What do you mean the consequence?

DOCTOR SHERON: I'm sorry. Not in terms of going into the severe accident part, but the -- in the beginning of the accident there was the question, as you know, about the coolant pump operation and understanding the consequences of tripping the reactor coolant pumps part way through an accident. And the codes were an instrumental part of the staff's resolution of that issue.

CHAIRMAN CARR: Okay.

DOCTOR SHERON: (Slide) On the next slide, what we've done is we've gone further from our overall objectives to specific objectives. And the first one here is to maintain a capability within the Agency for thermal-hydraulic analysis of light water reactors. These needs were embodied in a letter from NRR to the Office of Research in 1984. The major areas we requested that the codes be able to address and maintain capability for is in the area of operating reactor events. In particular being able -- when an event does occur at a reactor, we are normally asked many times to do "what if" calculations, what if this happened, what if that happened, and the like, and to demonstrate that we understand it. So, we have developed these codes with the understanding that they should be able to calculate these type of events.

Licensing issues, many times a licensee will come in with an amendment to their reactor in which we may question the proposed response that they've calculated. We may with to independently confirm the licensee's analysis. So we would use the codes to calculate these changes in the plant and independently reach the same conclusion the licensee has.

And then lastly, we want to maintain a capability for these codes to help us in more generic areas, as I said before, the front end of severe accident events.

The second objective is we need to maintain a cadre of experts, both in-house and at our contractors, to achieve this capability in item 1 above. We also need to maintain a code development and research activity at a minimum level. And this is necessary first to ensure that the codes are acceptable for predicting some of the advanced LWR and CANDU designs that we understand are in the wings now. Also to review new information which is still being generated, primarily overseas now, to ensure that nothing that is learned over there would invalidate our current understanding of the codes or their ability to calculate plant behavior.

And then lastly, we need to maintain this activity to maintain a cadre of experts. Very plainly put, if our contractors at the laboratories that are responsible for developing and maintaining these codes do not have active and interesting work, they will mostly likely leave the program, in which case then we don't have any experts. There's a real benefit to that because NRR typically when an issue comes up, they will turn to our contractors, the expertise that we've had developed at the laboratory and are maintaining, and use them to work on their specific problems. So, if we lose this expertise, then NRR also loses their experts as well.

(Slide) Next slide, please.

CHAIRMAN CARR: Well, are we the source sole -- sole source of funding for those experts?

DOCTOR SHERON: In the thermal-hydraulic area? Yes. I don't believe the Department of Energy has very much funding in that area.

CHAIRMAN CARR: What do they do for a living, I guess, is what I'm asking.

DOCTOR SHERON: Who's that, the --

CHAIRMAN CARR: The experts, I mean. You mean if we kill the contracts, those guys are out of work?

DOCTOR SHERON: They'll probably move to other programs.

MR. TAYLOR: They go to other work within the lab system, I'm sure.

CHAIRMAN CARR: All right.

MR. BECKJORD: There's a lot of competition in places like Sandia and Los Alamos for those people with those talents.

CHAIRMAN CARR: But they'll still exist and be working on the same kind of problems, I would assume.

DOCTOR SHERON: Many move over to like the Star Wars program, for example.

CHAIRMAN CARR: Okay.

DOCTOR SHERON: On the next slide, number 5.

One of the things we want to do is a new initiative here to help maintain capability within the Agency, is to establish and maintain low-cost experimental capability at universities through the construction and operation of scaled loops representing major U.S. reactor types. We've done this in the past at the University of Maryland. There's a scale loop of a Babcock and Wilcock lower loop design, which we've used as part of this overall MIST program which is basically to understand the B&W

plant behavior under small break conditions. It helped us very much in terms of scaling and understanding the scale factors involved from going to a basically a full-height, full-pressure, but very thin type of loop, skinny tubes and everything and the like.

We had substantial success with that. We've also ascertained that the results from a small-scale loop can, in fact, be used very effectively in code validation and verification.

We've proposed to maintain an experimental capability with the Agency to construct additional loops at universities over the next several years. We're now examining and discussing with NRR what type of reactors these ought to represent. We've have some suggestions that maybe they should really look at the more advanced designs that are being proposed. These loops can be built relatively cheaply, usually for within $1 million and they can be run even cheaper, usually about $150,000.00 a year, primarily because you're paying just for the professor and the graduate students.

A side benefit too is that we now have a source of graduate students that will be versed in thermal-hydraulics and hopefully will perhaps even consider working, you know, for the government or for our contractors. So, we maintain the expertise in that way.

Number 5 is we want to retain involvement in international thermal-hydraulic activities. However, I would put the caveat on that it's only provided that our resource commitment is minimized. By that I mean we don't intend to really get into any large financial commitments, long-term commitments with large facilities. And two is that we expect to see a substantial benefit to the Commission by being involved in these loops or these programs.

And then lastly, we've developed these codes. We want to apply them rather than just have them sit around, now that we've developed them, and systematically -- use them to systematically assess reactor behavior, looking at operating reactors, the advanced LWRs, the CANDU, the PIUS. Hopefully work in a proactive mode, so perhaps we can flush our problems before they actually -- before they actually happen.

(Slide) On the next slide, and we've put, I think, one that's a little probably clearer up on the screen. This is primarily just to show you, as Eric said, how we have projected the funding. You'll see it is reaching an asymptotic level as we go in the out years beyond '90. Right around '80, I think that was -- LOFT had just run its first test, I think a couple years ago. And -- you know, since then, what you're seeing is the phased shutdown of many of the experimental facilities.

(Slide) If you turn the page to slide 7, what you'll see is what we project as the costs necessary to maintain this thermal-hydraulic capability within the agency at a minimum level in the out years. This is, say, beyond FY '90.

You'll note that areas we have. There's code maintenance. We have about three or four separate thermal-hydraulic codes which have to be maintained. We have the TRAC code, the RELAP code, as Eric said the RAMONA code. We also have a plant analyzer at Brookhaven.

The scaled facility development, these are the university loops we were talking about, these would be built over a period of several years. So we're not building them all at once, at one time.

On the applications area --

CHAIRMAN CARR: Well, you're building one a year plus, I guess, or at least building one and running one, huh?

DOCTOR SHERON: Well, I don't --

CHAIRMAN CARR: You said it's about $1 million to build one and you're doing a million and a half per year.

DOCTOR SHERON: Yes, roughly once they're-- yes, once we've started the construction on them.

CHAIRMAN CARR: The question that comes to my mind is how do you decide where to build those? Is it a competitive contract?

DOCTOR SHERON: It would be a competitive contract. We would go out for a request for proposals.

On reactor applications, the advanced -- I'm sorry, the operating reactor issues, you'll note we've continued funding there and we've allocated funding for performing analyses on the advanced reactors that are expected to be coming in as well as the CANDU design.

So you can see we've -- you know, this totals up, we're expecting, right around $6.75 million a year and depending upon what's needed, I think, out in the future years, in particular for the advanced reactors, that number could either go down or up, depending.

CHAIRMAN CARR: Let me make sure I understood what you told me on the competitive. The universities bid against each other for these facilities?

DOCTOR SHERON: Well, we would go out with a request for proposals. We would identify what the research is, in other words the -- you know, we would ask someone to come in with a proposal to build a facility and operate it for a number of years. We would hopefully receive --

CHAIRMAN CARR: So it might not go to a university?

DOCTOR SHERON: I guess not. If someone else came in and could competitively bid, we would have to consider it.

MR. BECKJORD: I do think that the costs though favor a university in this kind of an activity.

CHAIRMAN CARR: I guess what worries me is do they favor a particular university?

DOCTOR SHERON: We have no -- we have none in mind, if that's what you mean.

CHAIRMAN CARR: Well, if these teams are out there we want to keep together, why aren't they already in business, I guess is my problem. You tell me these guys are out there and we've got to keep funding them, but it's going to be a competitive bid and we don't know who's going to get it. So --

DOCTOR SHERON: I think -- I think you're misunderstanding a little bit what we had in mind.

CHAIRMAN CARR: I must have misunderstood what you told me.

DOCTOR SHERON: The basic work on analyzing -- for example analyzing operating events, analyzing these new reactors, maintaining the codes and doing minor development work would be most likely maintained at the laboratories, okay, which would probably be Idaho and Brookhaven. The experimental work would be done at universities. Now, normally what we would do --

CHAIRMAN CARR: Well, I wouldn't think that one of these cracker jack guys we got is going to be very interested in just maintaining codes.

DOCTOR SHERON: No, no, no. They would-- the data that is obtained from the university loops or these loops that we want to build, okay, would be fed into the code maintainers and developers. They, in turn, would analyze the data, determine if the codes needed any more improvement, propose making whatever improvements were necessary. We would also use them in, I think, interesting areas such as the La Salle event in which we are right now actively engaged with NRR in doing the evaluation of BWR stability.

CHAIRMAN CARR: Okay.

DOCTOR SHERON: With that, I'm going to let Doctor Shotkin walk through the details of our programs.

DOCTOR SHOTKIN: (Slide) If I could have the next slide, slide number 8.

As we sat down to figure our plans for the future, one of the major factors was that most of our major thermal-hydraulic research programs are going to be completed by fiscal '92. We've already put that out in a NUREG report. For example, the computer codes, TRAC and RELAP, have been frozen, that is, no new models have been put in since December of 1984. They've gone through a period of five years of assessment through the international code assessment code, primarily. They've been improved with models and we're going to come out with the final versions of these codes in December of this year. This will complete all of our work in planned code development for our major systems codes. And this was something that was started over 15 years ago.

These codes will have flexible capability. That's one of their hallmarks. You can change the noting, we can simulate all the current vendor geometries, we can include the balance of plant. We can analyze a variety of scenarios, those that we know about, LOCA, those that we -- that might come up that have a lot of operator in -- actions or inactions, multiple equipment failure that would lead to risk dominance scenarios.

And we believe that our codes that we'll come out with at the end of this year, plus those that have already been completed, give us sufficient capability and that no new systems code is needed at this time as ACRS has recommended. And this is where we -- one area where we disagree with the ACRS.

One area that we're starting now with these codes is to start evaluating the newer reactor designs such as the 600 megawatt advanced reactors and possibly starting to look at CANDU and maybe PIUS.

(Slide) Another -- on the next slide --

CHAIRMAN CARR: While you're on that previous sheet, who exercises configuration control over them, do we or --

DOCTOR SHOTKIN: Each lab has a QA control that they have established and they exercise the configuration control.

CHAIRMAN CARR: And how many people around the world would you say are using the current versions?

DOCTOR SHOTKIN: Every major nuclear country.

CHAIRMAN CARR: So, when you come out in December '89, is somebody going to ensure that everybody gets the updated version?

DOCTOR SHOTKIN: Oh, yes. We have the international code assessment program, which I'll cover later, that we have about 14 countries that are members. Each one of them will get these codes and they're going to do the assessment for us over the next two years. And if you're talking of interesting work, this is something that we'd keep people for at least another two years while these codes are being assessed.

CHAIRMAN CARR: But we do have some method of ensuring that everybody is using the same version?

DOCTOR SHOTKIN: Absolutely, yes.

CHAIRMAN CARR: Okay.

DOCTOR SHOTKIN: We're reasonably careful with that. However, I should point out that if a country has experts, and many of their experts have been trained over here, and they want to go in and make their own changes, we can tell them, "Don't call it that, call it something else," but we can't stop them from making changes. We just don't let them call it the same code.

CHAIRMAN CARR: Okay.

DOCTOR SHOTKIN: On slide number 9, another -- I've gone through on the previous slide all of the code programs that will be completed by '92. Most of our testing will be finished by '92. A lot of it has already been finished. The 2D/3D project, which is the -- this large-scale German, Japan, U.S. program, will end in 1990. This will give us a complete database for use with the ECCS rule revision and we do expect industry to start coming in with submittals on this and they will be using this data to support their submittals.

The ROSA-IV is a large-scale, small break LOCA test facility, the world's largest small break LOCA facility. It's in Japan. We have a cooperative agreement with Jerry where they do testing that we recommend, we and the Japanese recommend. This testing program that they have will end in 1992 and we do not plan any renewal of this agreement after that date.

B&W testing, that is our last major domestic testing program. We have just finished testing in the MIST facility this year, in 1989. We're coming out with a final report. We are in the process of negotiating with the B&W Owners Group for further testing in a once through steam generator geometry. We've spent the year coming up with recommendations and now they -- we're in the process of finalizing whether or not we go ahead with that program.

After that is done, there are no further testing needs identified, either for B&W or, for that matter, any other of the existing geometries.

CHAIRMAN CARR: When we're through with these things, do we turn them down or do we mothball them or --

DOCTOR SHOTKIN: I'll just -- I can give examples. Of the LOFT facility, when we finished testing, we turned it over to the OECD and they ran that for about five years as an international program. There was a facility on the West Coast. We called it FIST, which was a mock-up of a GE reactor. After we got out of it, GE and DOE came in and then used that to test out the natural circulation capability of the new 600 megawatt design. They renamed the facility GIST. And after they are finished with it, it's just sitting there. SEMISCALE has been sitting there. We've been using the power supplies to do some other testing, but otherwise it's just sitting there decaying.

CHAIRMAN CARR: Who owns them?

DOCTOR SHOTKIN: We do. We -- I mean the government. I believe it's DOE that owns them.

CHAIRMAN CARR: DOE has the title? Okay.

DOCTOR SHOTKIN: (Slide) On the next slide, slide number 10, the second factor that we used in making our plans for the future was to see what are the requirements of the user offices. Last year in 1988, we had a series of meetings at both the branch level and the division director level with staff from Research, from NRR and AEOD. As a result of these meetings, of the task group meetings, senior program review group meeting, we came up with a joint recommendation of the entire NRC staff that would use this capability. And the conclusion was that the staff will continue to need independent expertise to address transients in the operating reactors that exist today, both the PWRs and the BWRs.

The second major conclusion of this review by the user offices was that the principal codes that have been developed, and these are the two TRAC versions, one for PWRs, one for BWRs, the RELAP5 codes, RAMONA as well as the plant analyzer, these should all be maintained for active use.

And the third major conclusion was that we should prepare these codes for quick response capability for transients that might occur in operating reactors. One of these was to expand the number of input decks that are needed to generate transients in these codes, to extend that up to about a base of about 25 representative input decks. Right now we have about ten. The number 25 comes from a careful survey of all of the vendor geometries that exist in the U.S. today and grouping them together in plants of similar kinds. We could probably do a good job on any transient that occurs with about 25 input decks.

CHAIRMAN CARR: What is an input deck? Is it really a deck of cards or is a computer tape or --

DOCTOR SHOTKIN: Well, when I was younger, it used to be a deck of cards. Right now it's on a tape.

CHAIRMAN CARR: But we still call it an input deck?

DOCTOR SHOTKIN: Yes, we still do.

CHAIRMAN CARR: Okay.

DOCTOR SHOTKIN: And it's what is needed to generate a calculation on a computer code. These input decks are also being developed not only for these representative plants but we're also doing work for AEOD so that they can benchmark the simulators that they have at the NRC technical training center. We're also developing input decks for each one of those simulators.

CHAIRMAN CARR: Now, the input deck itself programs the computer so that it looks like a reactor plan?

DOCTOR SHOTKIN: That's correct.

CHAIRMAN CARR: Okay.

DOCTOR SHOTKIN: (Slide) The next slide, number 11.

The third factor that we considered when we made our long-range plans was this need to maintain expertise to meet future Agency needs. As Doctor Sheron has said, just telling a guy that he's there to be an expert in the future isn't going to keep him very long.

What we've done is in 1987 we formed a thermal-hydraulic technical support center at Idaho where we made a commitment to provide support, baseline support, for people at this center with the -- and this came out as NUREG-1244, with the understanding that when a priority issue came up they would put aside this deferable research and work on the priority issue. And I can tell you over the last two years, just about everything they've been doing has been new priority issues. For example, working on the BWR stability issue of La Salle. We had to develop an accident management research plan and we told them to stop work and help us with that.

CHAIRMAN CARR: So, for a fixed amount of money, we have priority use on a bunch of technicians and equipment. And if we're not using them, they can contract out to anybody?

DOCTOR SHOTKIN: No, we give them baseline activities to work on these -- developing the input decks.

CHAIRMAN CARR: So, GE can't hire them or--

DOCTOR SHOTKIN: They're dedicated. That doesn't mean that they don't actually switch people around and do other projects.

COMMISSIONER ROGERS: How large is that group that you're talking about?

DOCTOR SHOTKIN: For the current funding right now, we have about eight people.

COMMISSIONER ROGERS: Are they all professionals? I mean that's eight professionals?

DOCTOR SHOTKIN: Yes, eight professionals, eight engineers.

CHAIRMAN CARR: What I guess what we asked for is we asked for eight dedicated bodies that we can use and you say we're managing to keep them busy.

DOCTOR SHOTKIN: Yes, for right now.

CHAIRMAN CARR: But if we didn't keep them busy, somebody else could use them, I assume.

DOCTOR SHOTKIN: Who? For another project? Well --

CHAIRMAN CARR: I mean a guy designing a reactor plant could hire them.

DOCTOR SHOTKIN: DOE -- let's say at Idaho, they've been trying to get some production reactors there. So, these people would help out on that.

CHAIRMAN CARR: Sure. Okay. There would be enough to keep them interested, I would think.

DOCTOR SHOTKIN: But then, when we have a priority issue, what we say -- we can't go in there and say, "Give us these three guys." We can ask for three bodies, but we wouldn't know which ones we'd get.

CHAIRMAN CARR: Yes, I understand. So, what we're discussing is how many guys or is it required to fund that cadre, is it three or eight or 15 or --

DOCTOR SHOTKIN: Right now we have about eight at that lab. We think a total cadre of about 15, including some work that we have going on at other labs that will eventually transfer to Idaho, would be appropriate.

CHAIRMAN CARR: Okay.

DOCTOR SHOTKIN: Some of this baseline research that we have going on at this technical support center includes the three elements recommended by the ACRS in their letter. These three elements are, analysis of plant transients that occur in operating reactors. Second is the preparation of these input decks that I've mentioned, and the third is analysis of special transients, those that have been and continue to be of particular interest.

Some examples here that we are working on or will be working on in the next few years are feed and bleed scenarios This has direct application to accident management. It's one of the most important accident management strategies. Another accident management one is secondary depressurization scenarios. And finally, natural circulation is again what could happen during a severe accident where you lose your pumps. The question is how well do we know that natural circulation will cool the reactor core? We feel that we've already generated enough data on the thermal-hydraulics program to come up with some sort of judgment on that and we'll be putting that together.

As I mentioned, when a priority issue arise, and I'll give some examples on viewgraph 13, we do delay this baseline research.

So, in closing on this slide, I want to point out, since you asked about ACRS, we do agree with their recommendation of the three baseline elements on this viewgraph.

CHAIRMAN CARR: Now, let me go back to the 15 guys. What do they charge us, roughly, $100,000 a year per --

DOCTOR SHOTKIN: It's about $150,000 per year.

CHAIRMAN CARR: $150,000.

DOCTOR SHOTKIN: That includes computer time.

CHAIRMAN CARR: So, we're talking $2.25 million per year to keep the team together.

DOCTOR SHOTKIN: Something like that, yes.

CHAIRMAN CARR: Okay.

DOCTOR SHOTKIN: (Slide) On the next viewgraph, 12, again the previous one was just maintain expertise. This one now talks about maintaining testing expertise, which is one of the factors that we considered in making these plans.

First of all, we would continue to seek and support cooperative programs with both international organizations and domestic industry. As you know, we have several of these cooperative international programs already, like 2D/3D and ROSA-IV. The MIST program was a cooperative program with industry. The one that's coming up that's a proposal by Germany is to continue testing in their facility under the 2D/3D program but do some accident management testing after the 2D/3D program ends. And they have made a proposal to us which we are considering.

Another way that we want to maintain testing expertise that's already been mentioned is these low pressure scaled loops at universities. This is based on the favorable experience that we've had with the B&W loop at the University of Maryland. We're going to plan to go out with a RFP to do some of this work at universities and if they come back with a proposal that they don't want to do a loop, that they want to do some separate effects testing, we'll consider that also.

CHAIRMAN CARR: What drives which loop you pick?

DOCTOR SHERON: I think our users, discussions with our users. As I said, we had a meeting two days ago, or was it yesterday, I can't remember which, with the NRR and AEOD. And we discussed this a little bit. There seemed to be an indication that perhaps rather than build, for example, a loop that represents a Westinghouse or a CE design, we should look towards a loop that perhaps looks like a 600 megawatt, the small advanced design coming in.

What we need to do --

CHAIRMAN CARR: Do we know what that looks like well enough to model it yet?

DOCTOR SHERON: Probably not. Probably not.

CHAIRMAN CARR: Yes.

DOCTOR SHERON: But that's, I think, a consideration we have to make.

CHAIRMAN CARR: I'm trying to tie it to if we know we've got a problem in somebody's loop, I assume that's what we had built first to try to figure out what the problem was. That may be a dumb assumption on my part, but I'm trying to figure out-- when you build that loop you have something in mind, I would assume, that you're trying to solve.

DOCTOR SHOTKIN: Yes, The loop at Maryland was built because we were putting a lot of effort on B&W geometry with the MIST facility and we needed some scaling information and we built a similar loop at Maryland to get the scaling information. That loop we plan to maintain. So we'll have something sitting there that can B&W geometry. We've actually never done any testing in exactly Combustion Engineering geometry. So that's one candidate, is to have a loop -- to build a loop for the CE geometry, test it and see if the behavior is as we expect, similar to the Westinghouse geometry.

What is probably of more interest is to build a loop for one of these advanced designs that are coming on. As I said, GE has already tested it out in this GIST facility out on the West Coast. We would like to do some independent testing of our own on that geometry.

CHAIRMAN CARR: Well, it seems to me that we've got enough problems out there with things in plants that vibrate and break and wear out that we don't really understand why some do and some don't. Doesn't that generate a desire to build something to figure out what's happening in there?

DOCTOR SHOTKIN: Yes. Yes, that does, but we would have to have a fairly strong user request in order to do something like that.

DOCTOR SHERON: Are you referring to like flow induced vibration or --

CHAIRMAN CARR: Yes. I don't know what I'm referring -- I don't know what causes them to break. But, for instance, the -- well, I guess it's the sensors that go in and out of the core on their little -- and we've got these tubes that sit -- the thimble tubes. Like I think it's South Texas that's got a problem with vibration that happened in Europe or something. I forgot which the plant is. But it looks like that's a problem that nobody knows how to solve. In fact, some plants have it, some don't, so we don't even know what the problem is.

I don't understand why that's not a valid research item, but maybe it's not ours.

DOCTOR SHOTKIN: As we go down -- that's one of the things that ACRS has mentioned that we get involved in and we do intend to put out a broad Agency announcement for some of these what I would call smaller research items, or something that perhaps a small company or university might help us solve. I don't know whether we need a big facility to look at that.

CHAIRMAN CARR: Yes. I guess I'm trying to figure out what the criteria are for figuring out which one of these loops to build.

DOCTOR SHERON: Basically it's going to be the -- it's going -- it's not -- it's going to be a matter of which type of geometries do we have the least confidence in our codes to calculate.

CHAIRMAN CARR: So the loops are really going to be for code verification rather than problem solving?

DOCTOR SHERON: Well, the codes are used for the problem solving. You cannot take a small-scale loop and go directly and apply it to a large reactor.

CHAIRMAN CARR: I guess you're -- okay.

MR. BECKJORD: I don't think we want to use our resources to get into every design problem that develops. I have to say that I think these thimble tube problems are design problems that I think the vendors ought to address.

CHAIRMAN CARR: Well, but you're using the codes to check the designs, aren't you?

MR. BECKJORD: Yes.

CHAIRMAN CARR: You would think the code would turn up those problems.

DOCTOR SHERON: These codes will not calculate something as specific as say a flow induced vibration or water seam or the like which would vibrate. They're just too -- too crude.

CHAIRMAN CARR: So, are they flow and heat transfer mostly?

DOCTOR SHERON: Yes.

CHAIRMAN CARR: All right.

DOCTOR SHOTKIN: (Slide) If I can go to the next slide, number 13, the fourth factor that we used in considering our -- developing our long-range plans is that issues that are related to thermal-hydraulics will continue to arise with regularity. They have been arising at the rate of once every year or once every two years that there's some major issue that comes that thermal-hydraulic expertise has to become involved in.

Current issues that we're working on is the BWR stability. That's related to the La Salle event. We have a two day meeting that's just going on upstairs right now. Again, as I mentioned before, we're checking the fidelity of the NRC simulators at the technical training center.

CHAIRMAN CARR: Against what?

DOCTOR SHOTKIN: There's some new software going in to upgrade their capability and we're using our already developed thermal-hydraulic codes to benchmark that new software that's going in.

DOCTOR SHERON: TRAC and a RELAP code

primarily.

CHAIRMAN CARR: So, against some -- well, those simulators are designed to reflect only one plant at a time, aren't they?

DOCTOR SHOTKIN: Yes.

DOCTOR SHERON: Yes.

CHAIRMAN CARR: So, you're going to check them against that particular configuration and plan?

DOCTOR SHOTKIN: Yes.

DOCTOR SHERON: That's correct.

DOCTOR SHOTKIN: That particular plant design.

CHAIRMAN CARR: Okay.

DOCTOR SHOTKIN: Another problem that's of interest to NRR recently has been interfacing systems LOCA and some of our codes are going to be used to help out with that.

And finally, as I mentioned, we are discussing with the industry some possible testing in the once through steam generator to look at the behavior under secondary depressurization.

(Slide) Next viewgraph, number 14.

The fifth item that we considered was that new initiatives will continue to be explored. So, what are our new initiatives? Well, first, the advanced light water reactors are coming down the pike. The 600 megawatt designs are quite different from the designs we have today. There may be new safety issues. We expect at DOE decision on which types they're going to develop through this year.

Some of the questions that we're asking are, how effective are the proposed new passive or, to use Doctor Beckjord's term, different term, safety systems? What they've done is instead of having pump -- ECC injection, emergency injection, they use natural circulation to get the water to the core.

And then, the next question is "Are the current computer codes applicable to these new geometries?" In fact, GE is using the TRAC code to analyze their new design. It's a version that they have -- they developed jointly with us. They've had it by themselves for a couple of years and they have made some changes to it.

We would expect to have an international code --

COMMISSIONER ROGERS: What do they call it?

DOCTOR SHOTKIN: TRAC GE.

CHAIRMAN CARR: How do you compare it? Is there a code for the new geometries that you compare it to?

DOCTOR SHOTKIN: No. What they --

CHAIRMAN CARR: Well, does this require some guy to sit down and redesign the code all over again?

DOCTOR SHOTKIN: Well, here's what they did. They marked up their new geometry in experimental test facility and they took data on the natural circulation patterns that developed. They then analyzed that data with their version of TRAC, TRAC GE, and it didn't fit. And they had some intelligent people sit down and say, "Well, it didn't fit because of these two models," and they change or improve those two models based on other considerations, reran their calculation and it did fit the data. And they then said, "It's good enough, we can now predict this behavior."

CHAIRMAN CARR: So it wasn't applicable, but they modified it.

DOCTOR SHOTKIN: At the beginning it wasn't, yes. They had to modify it.

CHAIRMAN CARR: Okay.

DOCTOR SHOTKIN: We have an international code assessment program with these 14 or so countries that will end in 1991. Most of these countries have expressed interest in continuing this program beyond 1991. And what we've said is, "Yes, we'd be willing to continue it if you would provide support for maintaining the codes with us." In this way, this is another way to maintain our expertise, is to have our people involved not only worrying about our problems, but also problems of the code that come up from other countries.

CHAIRMAN CARR: What does that run us per year after 14 --

DOCTOR SHOTKIN: Right now, the ICAP program itself is running us about $750 K per year.

DOCTOR SHERON: Keep in mind that the-- this proposed code maintenance consortia is designed such that instead of getting in return from these countries code assessment, in other words calculations, we basically said we would like money to help offset our costs. So, what we're trying to do is to actually bring in money --

CHAIRMAN CARR: Well, is $750 K the total cost right now or is that just our part of it?

DOCTOR SHOTKIN: Oh, that's just our part, yes. They are putting in much more money than that.

CHAIRMAN CARR: What's the total price of that code maintenance consortia?

DOCTOR SHOTKIN: Oh, of the code maintenance would be about $500 K per code per year. We would expect to have two or three codes maintained, depending on the international interest. There's interest in at least two of these codes. There might be interest in a third one.

CHAIRMAN CARR: Okay.

DOCTOR SHOTKIN: Some of the countries that are most interested are the UK and Japan.

Another new initiative is we are considering the ACRS suggestion to broaden research into something they call thermal sciences. We are asking them to help us define what they mean by thermal science. We are coming with some suggestions and we will discuss this with them.

And finally, as -- if and when CANDU and PIUS come on to the American market, and we would consider looking at the safety issues connected with those reactors.

CHAIRMAN CARR: Where do we maintain that? Where is that international code maintenance done?

DOCTOR SHOTKIN: That is done at the labs that have developed and maintained the codes now and that's primarily Idaho and Los Alamos.

CHAIRMAN CARR: And is that the same guys-- those are our dedicated guys or is that separate from them?

DOCTOR SHOTKIN: No, those are different

people who are working on the 14 --

CHAIRMAN CARR: Same kind of thing, but different guy. Okay. So they're going to stay busy if we keep this one going anyway.

DOCTOR SHOTKIN: (Slide) And finally, on slide 15, the sixth factor, just some other planning assumptions that we've used in developing our long-range plans.

First, we've looked at what we need for accident management software capability. And right now we believe that the existing codes that we have will be sufficient and no new software development is needed.

Second, we will continue to support fundamental research where needed. This is one of the ACRS suggestions. We intend to do this both through broad Agency announcements on certain topics as well as through the continuing grants program that we have.

Some of the ACRS suggestions that we will consider as part of this fundamental research program will be flow induced vibrations. That came up in one of the meetings. There's a question of heat transfer under oscillating flow. Sometimes valve behavior depends not only on the mechanics of the valve but on the flow conditions that are near the valve. And finally, the problem of erosion and corrosion does have a flow effect connected with it and we would propose to ask for some research proposals on that effect.

And finally, ACRS has suggested that we look at the way we are preserving our research results for future use and determine the best method so that future generations of people who come along will be able to have access to these results.

Do you want to do the last one?

DOCTOR SHERON: You can do it. You're doing so well.

DOCTOR SHOTKIN: (Slide) The final viewgraph, number 16, I've put on one viewgraph a summary of our response to the ACRS letter of June 15th. We agree with all the ACRS suggestions, except one, the one being that we disagree with their suggestion to initiate development of a new systems code.

What we agree with is that the codes are now good enough after the final versions are finished in calendar year 1989 and that no further development need be planned.

On experimentation, we would support, if funding is available, the university loops as well as the fundamental research that I mentioned.

Their suggestion on data analysis, we do intend to come up with proposals to them of how we would propose to preserve research results for future use. I should say that we already have several programs. We have an experimental data bank and we have the code maintenance program that we feel go a long way towards preserving these results, but we would discuss with ACRS what specific improvements we could make.

On the applications research, we do support their suggestions primarily for maintaining the expertise at the thermal-hydraulic technical support center, which would involve analysis of transients that occur in operating reactors, the development of the input decks and then to pick out some of the specific transients that I mentioned that would be of particular interest.

Finally, we agree that we would be interested in broadening our perspective of thermal sciences once we can mutually agree on what thermal sciences are of interest to the NRC.

Now, the reason we disagree with the ACRS suggestion to initiate development of new systems code is that we feel our current capability is quite sufficient to meet our needs in the future. Let me tell you what some of that capability is.

As I say, the new issues only come up about once a year, once every two years. And there isn't a heck of a lot of analysis that goes on when we analyze those issues. So, even though these major codes take a long time to run, most of the cost that goes into the analysis is people cost rather than computer cost. So, we don't feel that we need to put in a large expense in order to develop a faster running code.

For those people within the agencies such as AEOD and NRR who do like to have a fast running code on a PC, we do have a version of the code that AEOD is putting into the simulators that they do have working on a PC and they can run sensitivity studies and get the kind of answers they want. Then they can always benchmark those answers against the major thermal-hydraulic codes.

That's all that we have.

CHAIRMAN CARR: Does that complete the presentation?

DOCTOR SHOTKIN: Yes, sir.

CHAIRMAN CARR: Commissioner Rogers? Commissioner Curtiss?

COMMISSIONER ROGERS: I've got some.

CHAIRMAN CARR: Excuse me, Commissioner Rogers.

COMMISSIONER ROGERS: Yes.

CHAIRMAN CARR: Didn't mean to pass you by there.

COMMISSIONER ROGERS: Okay. Yes, a couple of things. Just on this systems question. I wonder if ACRS and you are talking exactly about the same thing though when they talk about system codes. In looking at their letter, it sounded to me at first as if they were talking about something more general that might include more than just thermal-hydraulics interactions. On the other hand, the examples they gave are things that you say you can handle right now. To do the La Salle station event, you say you can do that right now with what you have, or you will be able to, I guess, with just a little bit more work. And they draw an analogy to the Brookhaven BWR code as the type that we're suggesting.

So, I don't know if I'm reading their letter correctly or not, but it certainly seems to me that this question of systems interaction is something I'd like to hear more about because the Chairman's question, for example, was one that relates a little bit to systems interaction. So, what I'd like to know is what kind of system codes do you have in research that aren't simply thermal-hydraulics codes? Do you have systems interaction codes of other kinds?

DOCTOR SHERON: I don't know of any that --

COMMISSIONER ROGERS: The only system codes that you have are those all just fluid and thermal effects?

DOCTOR SHERON: If you're saying are there any in which you interact, say, structurally and the like --

COMMISSIONER ROGERS: Well, or electromechanical.

DOCTOR SHERON: I think the severe accident codes probably come the closest to combining thermal-hydraulics, say, with structural interaction core, the physical melting of structures and relocation of fuel.

COMMISSIONER ROGERS: Well, I'm not thinking about those. I'm thinking about before you get to that point.

DOCTOR SHERON: Maybe it would be helpful if I -- let me give you my impression of the ACRS concern and why they recommended that. The question that was asked is these codes, RELAP and TRAC, all evolve from the large break LOCA codes which were developed. In order -- during the particular blow down phase of a

LOCA, where things are happening very, very quickly, you have to solve mass, momentum and energy simultaneously. It's an iterative process. It's very expensive. You have to use very small time steps.

These codes were developed specifically to calculate that kind of an event. Over the years, as we went from a large break into the small break and then the transient and then severe accident, I guess the best way to describe it is that the time scale of the transient was extended. They become slower. Okay. Things don't happen as fast. When things don't happen very fast, you don't need to solve, for example, the momentum equation. And you can speed up the running time. The concern was that these codes, if you apply them to a long, slow, evolving transient, can be expensive and time consuming to run because of the way they're structured and the way they've been built. And in fact, if you were to go back today and say, "I want a code that can calculate a slow, long drawn out transient where things don't change very fast," I would do it entirely differently and I could probably build a code that could run faster.

That was their concern. We did build a code at Brookhaven called the BWR analyzer, which runs on an AD-100 computer. It runs, depending upon how it's nodalized, faster than real time. It's useful. We've been using it extensively with the La Salle analyses.

Now, the question is, "Gee, should we build one for a PWR?" The cost of building a code like that is probably in the range of $2 to $3 million and then you have to go through the entire assessment process and then you have the maintenance of it as well. Once you have a code, you have to maintain it. You have to maintain people there. So, there is a lot of long-term cost associated with a new code.

Our feeling was -- is that we have, for example, the TRAC -- it's the RELAP code, RELAP5. INEL, we had developed what they call their plant analyzer. What this is it's basically a code that tacks onto the front end of RELAP so that a user can interact on a terminal with the code and use it as an analyzer. If it's nodalized properly, okay, it can run almost in real time, sometimes even faster.

CHAIRMAN CARR: Any plant?

DOCTOR SHERON: Yes.

CHAIRMAN CARR: PWRs, BWRs

DOCTOR SHERON: Yes.

CHAIRMAN CARR: Any configuration?

DOCTOR SHERON: Wait a minute. It doesn't do the Bs, does it or not?

DOCTOR SHOTKIN: The Idaho one? Oh, it does the Bs.

DOCTOR SHERON: It does the Bs. They do both, yes.

CHAIRMAN CARR: And it's no matter how many loops or what?

DOCTOR SHERON: Right.

CHAIRMAN CARR: Okay.

DOCTOR SHERON: And basically the speed is dependent upon how finely you nodalize it, how many loops you choose.

Our feeling was that for the number of analyses that we are called upon to do every year, combined with what Doctor Shotkin said, and that is the principal cost of doing an analysis is not computer time, it's people time. People have to set up the decks, people have to analyze the results, they have to document the results, they have to QA the results. You're going to do that with any code. And our feeling was that it was more cost beneficial to maintain what we have and use it, even though it may be a little more expensive to run a calculation with the RELAP analyzer than it would be with a new code, than to put all those front end costs right in there.

And as an aside too, AEOD has also developed their own PWR analyzer recently. And that's still under development. They're using it for their own purposes, but certainly I think that if either Research or NRR needed something of that capability, they could turn and use that one. So that was the basis for saying we really didn't feel it was beneficial to develop a whole new code at this time.

I don't know if that answers your question.

COMMISSIONER ROGERS: Well, to some extent it does. I think maybe we could get into too long an extended conversation on it here that would be inappropriate for this meeting. But I'd like to hear a little bit more about what the -- that whole situation because I wonder whether we are addressing the systems interactions as -- of different parts of the plant in an adequate way and whether that was a question that they were raising, ACRS, in their letter. It isn't so clear to me, from what you've said, that that's what their thinking was, that they may have been thinking of something else.

So, I think I need to learn a little bit more about their thinking and yours on the whole matter. So I don't think I want to pursue it too much more today, but I do want to hear more about it from you if I can.

The question of the considerations of where you'd place a scaled facility, I just ask you, if you would in looking at that, consider the whole impact of that facility on the general question that we're facing in the country of the quality of nuclear engineering programs. Nuclear engineering programs are gradually going down the tube and facilities at universities to maintain high quality -- the interest of high quality students in going into nuclear engineering programs is a real problem.

I would wonder whether it's appropriate in thinking about where to place something like this to look at a little broader picture than just the proposal for the hardware and maintenance of that hardware and running it to look at the impact of that on the whole question of a nuclear engineering program, the quality of the people that might come out of it, and the maintenance of the faculty skills that are very important in maintaining nuclear engineering.

So, that I would hope that we could think about that question more broadly when we write a request for proposal so that it isn't just that we look at the most narrow view of what is the best proposal.

DOCTOR SHOTKIN: I think the nuclear

engineering departments are not the only places that could respond, but they certainly have an advantage because of the interest in codes and analysis of the systems. So, I would certainly expect that most of these would finally be associated with places that have a nuclear engineering department.

COMMISSIONER ROGERS: All I'm saying is I think that the broader question of not only the immediate data that comes out of such a facility, but the quality of the students and the quality of the faculty that are involved with it in terms of what they'll help to produce in the way of student output is very important.

MR. BECKJORD: That certainly is an important consideration.

COMMISSIONER ROGERS: Yes, right. So, it's not just who comes in with the cheapest proposal to build and operate something for us, that there are subsidiary questions that I would think would be important in making that placement.

MR. BECKJORD: Yes, I agree.

COMMISSIONER ROGERS: The -- this data analysis question and the preservation and organization of data from the test programs, I hope that will really get some hard thinking. It's a tough problem of how to preserve for future use not only the data but the smarts that were involved in generating that data as well. And once you have a table of numbers, that's not the whole story, as we know. How -- what some of the background of that is and what's required to understand and interpret it and so on.

So, it seems to me that that's a common problem with any sort of a situation where you build a big facility, you can't use it for some special purposes, people have to get trained up and skilled to build it, design it, operate it, get the numbers and then you tear it down. And there's a great loss, a human loss in various ways in that kind of situation. And you don't, you know, -- just that final output of a few numbers, even though it may be a long table, is not the full story. So, I think that somehow-- otherwise, ultimately somebody goes back and redoes the whole thing because they have the numbers, but there was something else that was missing, so they have to go back and do it.

So, I think that's something that requires a good deal of creative thinking, this question of how to preserve not just the data itself but some of the background thinking and skills that went into the acquisition of that data for the future, for future generations.

DOCTOR SHERON: One thing we are doing, which I don't think we've mentioned here, and that is that this technical support center at INEL -- I guess Chairman Carr was concerned about what these people do when they're not working on priority issues. One of the things we have them doing is a thing called the synthesis reports. What we've asked them to do is to look at special issues. And I guess natural circulation was one, feed and bleed cooling, I guess, was one. We've run a lot of experiments at various experimental facilities, LOFT, SEMISCALE, the FIST facility, MIST facility and the like.

The question is, for the users in particular, NRR, it's not sufficient for us to dump a bunch of contractor reports in their lap and say, "Here it is." That's not going to help them. What we're trying to do is synthesize it and that's what these reports do. They try and take all this data and collect that wisdom that was learned from all of this and put it in one document in a succinct way. And we've been trying to do that and this is the kind of work that, as I said, when something comes along like a priority type of work, we would ask them, "Okay, defer doing this to work on the priority work, but when that's done go back and do this." So, it's an ongoing program.

COMMISSIONER ROGERS: Well, I think that's an excellent and very important activity. But I wonder if it wouldn't be a good idea to try to think of connecting universities into that kind of thing because that's the sort of thing that universities are supposed to do very well, to be able to bring a general view towards the acquisition of knowledge and put it in a form that is useful for future generations.

And as you consolidate your thermal-hydraulic research in one center, I would think it would be very worthwhile considering encouraging universities to interact with that center and somehow participate in activities of precisely that type, because that's exactly the sort of thing that they can do well and there's no great rush on it. Universities don't respond very quickly, in general, to things, in solving problems. But they do respond in depth. And I would think this is the kind of thing that a university participation would be very useful to have and it could be part of your grants program.

Okay. Thanks very much.

CHAIRMAN CARR: Commissioner Curtiss?

COMMISSIONER CURTISS: Just a quick question. There was a report in the trade press last month about a fellow out at Idaho, Makowitz, I think his name was, who had run the codes and, I gather, tinkered with the time steps and came up with some results that surprised him at least and I gather suggested in turn that the differences in the results may call into question whether the people and values that were using the codes yield the most accurate results. Is that a concern that we have or have we appropriately responded to that?

DOCTOR SHOTKIN: We've received a letter from Idaho, EG&G, telling us that they are -- this is nothing new. There is not a concern. We do intend to submit this letter as well as some other background information to the Commission. There is no real concern.

COMMISSIONER CURTISS: Are the differences that he came up with predictable or expected differences when you run the codes by tinkering with the default values?

DOCTOR SHOTKIN: They're expected differences. The words that he used to describe them were -- I would call sensational. We went back and looked at the actual clocks that he had looked at and there was nothing really exciting about it.

DOCTOR SHERON: The divergence that he referred to was like differences in like ten percent, I think in the answer.

DOCTOR SHOTKIN: Yes.

DOCTOR SHERON: We were well aware that if you don't judiciously choose your time step, you'll get differences. And we make sure that when we put out our code manuals, that there's proper guidance for the users on how to chose the time steps.

DOCTOR SHOTKIN: Actually, he defeated something that's in the code. The code automatically chooses the time step. It does not let the user use it. But he knew how to go in there and defeat it. So I don't -- we really don't think it's a problem, but we'll communicate that to the Commission.

COMMISSIONER CURTISS: Okay, thank you.

That's all I have.

CHAIRMAN CARR: What are the benefits and objectives of us participating with the CSNI?

DOCTOR SHERON: In general --

CHAIRMAN CARR: Committee on Safety of Nuclear Installations. You mentioned in here that we were going to continue to --

DOCTOR SHERON: In general or in the area of thermal-hydraulics?

CHAIRMAN CARR: Any way you want to take it.

DOCTOR SHERON: In the area of thermal-hydraulics --

CHAIRMAN CARR: What are we getting out of it is my question.

DOCTOR SHERON: In the area of thermal-hydraulics, the principal benefit used to be the participation in international standard problems where one country would put up a test and all of the countries would participate in predicting it and we would then cross check our results to see if we were all on the same wavelength basically.

We've stopped participating in those type of international standard problems about two years ago. The reason is is because we felt that, again, our codes were pretty much at a mature level. We were not getting any real benefits out of it and we didn't really have the resources to spend on that.

The other area that we are participating in international standard problems right now in CSNI, analytically, is in the severe accident area. I can speak for the group that I represent. I represent the Commission on principal working group 2. We sponsored just recently the first severe accident international standard problem, which was the CIRC 4 test run by Sandia. I think there was substantial benefit, primarily because nobody predicted it very well. What we learned from it was that you have to include the condense phase chemistry in the core concrete reaction. Otherwise, you will not get the temperature of the melt right and that principally would effect things like the gas evolution rate off of the melt, the core concrete interaction and to some extent the oblation rate of the concrete.

So, there was, I think, a real benefit to participating in that test. We are now looking towards establishing a standard problem on early phase melt progression. We're going to get some proposals, as a matter of fact, in September of the next meeting.

To answer your question in general, I think there are benefits to participating in international standard problems because it helps us understand on an international basis how well we do compared to the rest of the world. I think it also helps us maintain a preeminence in this area because basically the rest of the world are using our codes.

CHAIRMAN CARR: Well, is it important that we maintain the preeminence? I guess is it worth paying for? Everybody else is building and we're not.

It looks to me like we might as well let them foot the preeminence bill awhile and take advantage of some of their efforts.

DOCTOR SHERON: Well, what the problem would be is that if we did not maintain the preeminence, they would most likely go off and build, for example, their own codes. The problem we would have is, number one, we would sort of -- if we stopped doing the work, then we would be dependent upon them for our analytical models. And also, as usual, if there are two codes to do the same thing, ten to one you'll get different answers. And then we will most likely wind up spending a lot of our time answering questions of critics as to why our codes don't give the same answers as theirs. And I guess my feeling is it's better if they're all using ours.

MR. BECKJORD: I'd like to make a couple of points on that, to take up CSNI in general. I think that the participation in that committee is useful in terms of keeping up to date on the developments because about twice a year you have a chance to sit-- on a regular basis you have a chance to sit down with counterparts in the other countries and you find out right away what's going on.

I think with regard to the working groups, those are important because these questions, the current issues are taken up in more detail. They can be pursued in those working groups, in both in breadth and in depth. I think that we would like to see some changes in the emphasis.

As I see it in CSNI, there is great emphasis today still on the kind of thermal-hydraulics that we're saying we know enough about now and I think not enough emphasis on risk. Perhaps one of the least active groups there is the group on PRA. I would like to see that more active and I would also like to see more attention given to the severe accident questions and the accident management. And we are trying to encourage the other parties to move in that direction, that is away from the thermal-hydraulics problems which we regard as largely solved and toward some of these other very important questions. That's what we're trying to convince people to do.

CHAIRMAN CARR: Okay. Are the costs which you project in your research paper here, are those reflected in the five year plan?

MR. BECKJORD: Yes.

CHAIRMAN CARR: And that's what you're currently planning in your five year plan?

MR. BECKJORD: Yes.

CHAIRMAN CARR: The --

MR. BECKJORD: They are not reflected-- however, they would not be reflected in what I see of reduced budgets in the out years, but what we're currently working with.

MR. TAYLOR: We don't know the impact of those projected cuts.

CHAIRMAN CARR: But the plan that's the latest submitted plan, they're in there.

MR. BECKJORD: Yes.

MR. TAYLOR: Yes.

CHAIRMAN CARR: Will the nuclear plant analyzer, or does it, have application in the development of accident management strategies?

DOCTOR SHOTKIN: Yes. We are right now using -- can I broaden your question to say the computer codes, not just the plant analyzer? We are using that to look at several of the more difficult strategies right now.

CHAIRMAN CARR: Okay. Let me ask one final thing here. In your recommendation it says that the Commission take note of the approach proposed herein for the future goals and directions of thermal-hydraulic research. If we take note of the approach, what does that mean? Have I committed myself to something? It doesn't sound like it.

MR. BECKJORD: No.

MR. TAYLOR: Does to me.

CHAIRMAN CARR: What are you asking me to do?

MR. BECKJORD: He's rapidly searching for that.

CHAIRMAN CARR: That happens to be the last page. It says recommendation.

MR. TAYLOR: The staff -- the staff wanted you to understand where we were and what we were doing really.

CHAIRMAN CARR: You left me a policy issue and all you asked me to do was take note of it.

MR. TAYLOR: The staff intends to proceed in this direction unless there's some --

CHAIRMAN CARR: That's why I'm speaking out. If I don't do anything, I've acquiesced by -- yes-- and as long as we have the money.

MR. BECKJORD: If I could characterize it, what we would like is approval of the plan that is laid out here.

MR. TAYLOR: You didn't ask for that specifically.

CHAIRMAN CARR: What I'm trying to find out is did you hope it would go away by default or did you really want approval for it?

MR. TAYLOR: We didn't make this an approval paper. I think the intention was that the Commission should be informed this is such a major program. We've had a lot of staff work on where we are heading. We've had ACRS input. So -- but we wanted the Commission to understand what we were doing. I believe unless the Commission indicates otherwise, we'd leave here and if the budget money is available, we would intend to follow the course that we've outlined here.

CHAIRMAN CARR: So, the onus is on us to do something or hold our peace until the budget scrub, I guess, huh?

MR. TAYLOR: That's right, sir. We may yet have parts of this program that we're not able to do, depending on the budget. But right now, this is-- this is where we think -- and we've had input from the user, everybody -- that this is the right course in this general, very broad, but important area.

COMMISSIONER ROGERS: I like it myself. I still have this question of systems interaction code. I'd like to hear a little bit more about that.

MR. TAYLOR: Managerially, we, I think,

across all the senior management, have talked about the thermal-hydraulic program knowing we were coming to this time where these kind of base decisions would have to be made. And as Eric mentioned, there are other very important demands on research money. But we wanted to tell you what we were doing. We weren't going to close shop on thermal-hydraulics, but we were definitely going into a maintenance mode because we have more important demands on our research dollars. That's really the message. That's the message.

CHAIRMAN CARR: I guess we can let the record show we've taken note and you're going to hear from us.

MR. TAYLOR: Maybe we'll change words in our next paper.

CHAIRMAN CARR: I'd like to thank the staff for the meeting. It appears that the staff has done a good job in developing this program by coordinating with the Advisory Committee on Reactor Safeguards and other NRC offices that use the information developed through the research program.

I believe it's appropriate and necessary, especially in the time of budget constraint, for the NRC to pursue cooperative areas and areas where research is deemed necessary. I would caution you to ensure that clear objectives which are of benefit to the NRC regulatory program are laid out and followed through in any cooperative effort.

Do any of my fellow Commissioners have any additional comments?

COMMISSIONER ROGERS: Yes, I just have one brief one.

I've made some noises in the past about some of the presentations to us by staff with an excess of acronyms and a lack of identification of who's going to make the presentation. I thought this one from you folks was an excellent job that I can't criticize in that way at all. I thought you did identify who the people are that are making the presentation and I didn't see a single acronym in it.

MR. BECKJORD: Thank you.

CHAIRMAN CARR: We stand adjourned.

(Whereupon, at 3:59 p.m., the above-entitled matter was concluded.)

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 NRC THERMAL-HYDRAULIC RESEARCH PROGRAM

PLACE OF MEETING: ROCKVILLE, MARYLAND

DATE OF MEETING: AUGUST 3, 1989

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

Carol Lynch

Reporter's name: Peter Lynch

FUTURE PLANS FOR NRC THERMAL-HYDRAULIC RESEARCH

PRESENTATION TO THE COMMISSION
AUGUST 3, 1989

BRIAN W. SHERON, DIRECTOR
DIVISION OF SYSTEMS RESEARCH
OFFICE OF NUCLEAR REGULATORY RESEARCH
AND
LOUIS M. SHOTKIN, CHIEF
REACTOR AND PLANT SYSTEMS BRANCH
DIVISION OF SYSTEMS RESEARCH
OFFICE OF NUCLEAR REGULATORY RESEARCH

BRIEF HISTORY

- NRC HAS BEEN DEVELOPING THERMAL-HYDRAULIC SYSTEMS CODES FOR 15+ YEARS.
- CODE DEVELOPMENT AND VERIFICATION WENT FROM LARGE BREAK TO SMALL BREAK LOSS OF COOLANT ACCIDENTS (LOCA) TO TRANSIENTS TO FRONT END OF SEVERE ACCIDENTS.
- CODES HAVE NOW REACHED AN ACCEPTABLE LEVEL OF ACCURACY AND MATURITY FOR CURRENT GENERATION LWRs.
  - FURTHER DEVELOPMENT NOT LIKELY TO PRODUCE MAJOR CHANGES IN OUR UNDERSTANDING OF PERFORMANCE OR CONSEQUENCES;
  - NRC HAS QUANTIFIED UNCERTAINTY FOR THE LARGE BREAK LOCA. SMALL BREAK LOCA UNCERTAINTY ASSESSMENT PLANNED.

PLANNING AND DIRECTION OBJECTIVES
OVERALL OBJECTIVES

- BRING MAJOR THERMAL-HYDRAULIC CODE DEVELOPMENT PROGRAM TO SUCCESSFUL CONCLUSION.

- MAINTAIN CAPABILITY FOR THERMAL-HYDRAULIC ANALYSIS AT MINIMUM LEVEL.

- APPLY DEVELOPED CODES TO REACTOR ISSUES.
  PAST ISSUES INCLUDE:
  - LOSS OF COOLANT ACCIDENT
  - PRESSURIZED THERMAL SHOCK
  - CONSEQUENCES OF THREE MILE ISLAND ACCIDENT
  - DECAY HEAT REMOVAL
  - ANTICIPATED TRANSIENTS WITHOUT SCRAM
  - ETC.

PLANNING AND DIRECTION OBJECTIVES
SPECIFIC OBJECTIVES

- 1. MAINTAIN CAPABILITY WITHIN AGENCY FOR THERMAL-HYDRAULIC ANALYSIS OF LWRs
  - a) OPERATING REACTOR EVENTS (e.g., DAVIS-BESSE, LASALLE)
  - b) LICENSING ISSUES (e.g., LICENSE AMENDMENTS)
  - c) GENERIC RESEARCH (e.g., FRONT-END of SEVERE ACCIDENT SEQUENCES)

- 2. MAINTAIN CADRE OF EXPERTS (CONTRACTOR AND IN-HOUSE) TO ACHIEVE (1) ABOVE

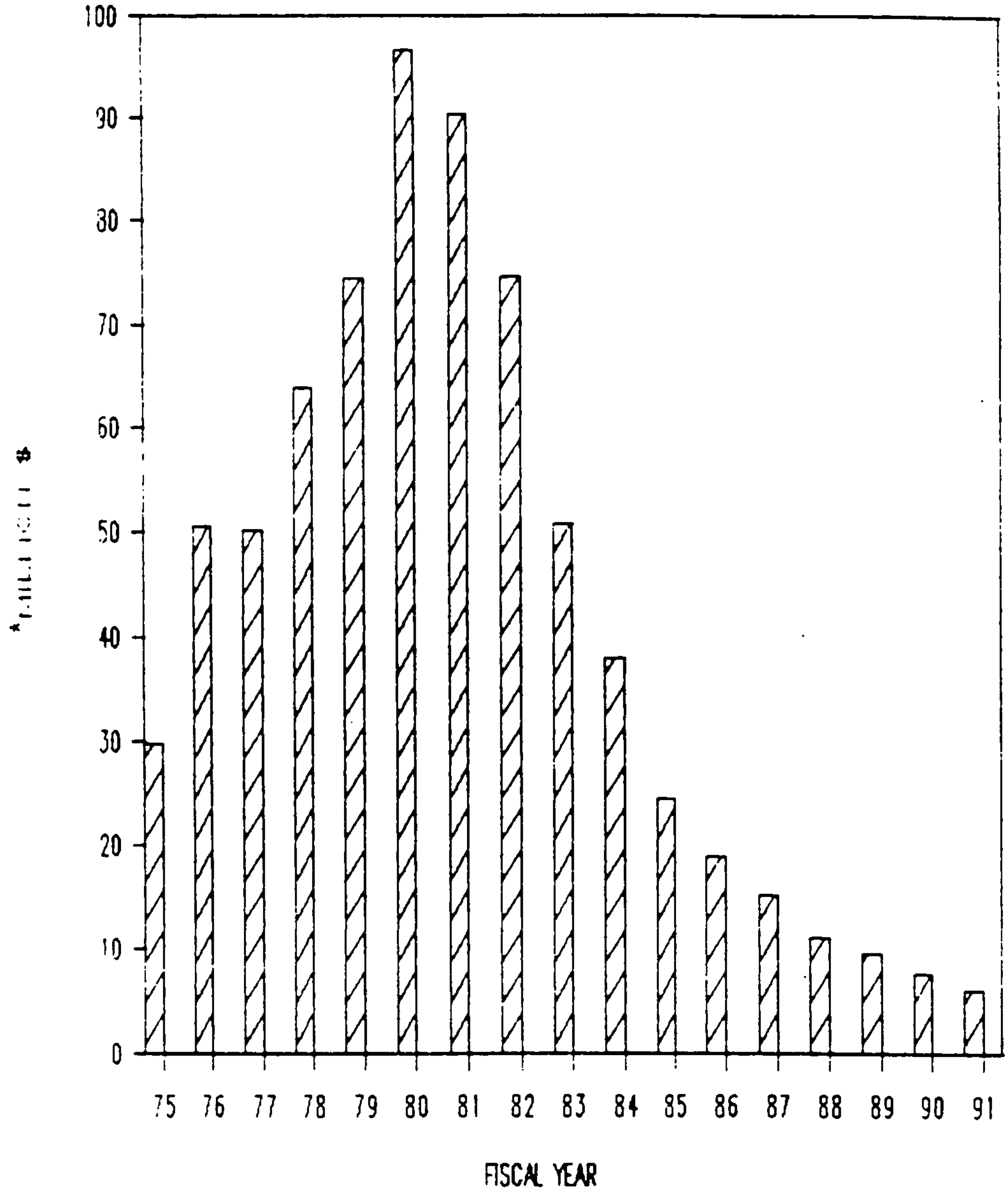
- 3. MAINTAIN CODE DEVELOPMENT/RESEARCH AT MINIMUM LEVEL NECESSARY TO:
  - a) ENSURE CODES ARE ACCEPTABLE FOR ADVANCED LWR AND CANDU ANALYSES;
  - b) REVIEW NEW INFORMATION TO SEE IF IT INVALIDATES CURRENT UNDERSTANDING OF CODE ACCURACY; AND
  - c) ACHIEVE (2) ABOVE.

- 4. ESTABLISH AND MAINTAIN LOW-COST, EXPERIMENTAL CAPABILITY AT UNIVERSITIES THROUGH CONSTRUCTION AND OPERATION OF SCALED LOOPS REPRESENTING MAJOR U.S. REACTOR TYPES.

- 5. RETAIN INVOLVEMENT IN INTERNATIONAL THERMAL-HYDRAULIC ACTIVITIES PROVIDED RESOURCE COMMITMENT IS MINIMIZED AND THERE IS SUBSTANTIAL BENEFIT TO NRC.

- 6. EXPAND APPLICATIONS RESEARCH USING CODES TO SYSTEMATICALLY ASSESS REACTOR BEHAVIOR.
  a) OPERATING LWRs
  b) ADVANCED LWRs
  c) CANDU
  d) PIUS

FUNDING FOR T/H RESEARCH



* Dollar Amounts not Adjusted for Inflation

FUTURE COSTS TO MAINTAIN THERMAL-HYDRAULIC CAPABILITY

- THERMAL-HYDRAULIC RESEARCH

CODE MAINTENANCE	$1.5M/YR	
SCALED FACILITY DEVELOPMENT	$1.5M/YR	
		$3.0M/YR

- REACTOR APPLICATIONS

OPERATING REACTOR ISSUES	$2.0M/YR	
ADVANCED REACTORS AND CANDU	$1.75M/YR	
		$3.75M/YR
TOTAL	$6.75M/YR	

NOT SHOWN: MAJOR PROGRAMS COMPLETED BY FY92

1. ALL MAJOR THERMAL-HYDRAULIC RESEARCH PROGRAMS WILL BE COMPLETED BY FY92 (NUREG-1252)

- COMPUTER CODES (TRAC AND RELAP) WERE FROZEN IN DECEMBER 1984.
- FIVE YEARS OF ASSESSMENT AND IMPROVEMENT PROVIDE FINAL VERSIONS IN DECEMBER 1989.
- FLEXIBLE CAPABILITY TO SIMULATE ALL CURRENT VENDOR GEOMETRIES, INCLUDING BALANCE OF PLANT.
- CAN ANALYZE LOCA, RISK DOMINANT SCENARIOS, AND OPERATIONAL TRANSIENTS WITH MULTIPLE FAILURES AND OPERATIONAL ERRORS.
- STAFF DOES NOT FEEL A NEW SYSTEMS CODE IS NEEDED AT THIS TIME, AS ACRS HAS RECOMMENDED.
- EVALUATION STARTING FOR 600 MW ADVANCED LWRs, CANDU, AND PIUS.

1. THERMAL-HYDRAULIC TESTING COMPLETED BY FY92

- 2D/3D PROJECT ENDS IN 1990.
  THIS COMPLETES DATA BASE FOR USE WITH ECCS RULE REVISION.
- ROSA-IV AGREEMENT FOR TESTING OF W/CE SMALL BREAK LOCA AND OPERATIONAL TRANSIENTS ENDS IN 1992.
  NO RENEWAL IS PLANNED.
- B&W TESTING (NUREG-1236)
  MIST FACILITY TESTING COMPLETED 1989
  OTSG* TESTING 1989-1992
  NO FURTHER TESTING NEEDS IDENTIFIED.

*OTSG= ONCE-THROUGH STEAM GENERATOR

2. FUTURE REQUIREMENTS OF USER OFFICES.

- INTERNAL REVIEW OF FUTURE NEEDS DURING 1988 BY STAFF FROM RES, NRR AND AEOD.
- NRC STAFF WILL CONTINUE TO NEED INDEPENDENT EXPERTISE FOR ADDRESSING TRANSIENTS IN OPERATING PWRs AND BWRs.
- PRINCIPAL NRC CODES (TRAC-PWR, TRAC-BWR, RELAP5 AND RAMONA) SHOULD BE MAINTAINED FOR ACTIVE USE.
- DEVELOP PLANT INPUT DECKS FOR SELECTED CODES TO HAVE A SET OF ABOUT 25 REPRESENTATIVE PLANT INPUT DECKS. (PRESENTLY HAVE ABOUT 10 INPUT DECKS)
- INPUT DECKS ALSO BEING DEVELOPED TO BENCHMARK SIMULATORS AT THE NRC TECHNICAL TRAINING CENTER.

3. NEED TO MAINTAIN EXPERTISE TO MEET FUTURE AGENCY NEEDS

- THERMAL-HYDRAULIC TECHNICAL SUPPORT CENTER FORMED AT INEL IN 1987 (NUREG-1244)

- BASELINE APPLICATIONS RESEARCH INCLUDES THE THREE ELEMENTS RECOMMENDED BY THE ACRS:
  - ANALYSIS OF PLANT TRANSIENTS IN OPERATING REACTORS
  - PREPARATION OF CODE INPUT DECKS FOR SEVERAL CLASSES OF PLANTS
  - ANALYSIS OF TRANSIENTS OF PARTICULAR INTEREST
    - FEED AND BLEED SCENARIOS
    - SECONDARY DEPRESSURIZATION SCENARIOS
    - NATURAL CIRCULATION
  - DELAY BASELINE RESEARCH WHEN PRIORITY ISSUES ARISE (EXAMPLES ON VUGRAPH 13)

3. MAINTAIN TESTING EXPERTISE

- COOPERATIVE PROGRAMS WITH INTERNATIONAL ORGANIZATIONS AND DOMESTIC INDUSTRY WILL CONTINUE TO BE EXPLORED.
  - ACCIDENT MANAGEMENT TESTING IN UPTF (PROPOSED BY FRG)

- BASED ON FAVORABLE EXPERIENCE WITH THE LOW-PRESSURE, SCALED B&W LOOP AT UNIVERSITY OF MARYLAND, OTHER SMALL-SCALE LOOPS (FOR OTHER VENDOR DESIGNS) ARE BEING PLANNED AT UNIVERSITIES
  - SEPARATE EFFECTS FACILITIES WILL BE CONSIDERED, AS REQUIRED.

4. THERMAL-HYDRAULIC RELATED ISSUES WILL CONTINUE TO ARISE WITH REGULARITY

- CURRENT ISSUES:
  - BWR STABILITY, RELATED TO THE LASALLE EVENT
  - NRC SIMULATOR FIDELITY
  - INTERFACING SYSTEMS LOCA
  - ONCE-THROUGH STEAM GENERATOR BEHAVIOR UNDER SECONDARY DEPRESSURIZATION

5. NEW INITIATIVES WILL CONTINUE TO BE EXPLORED

- ADVANCED LWR (600 MW) SAFETY ISSUES
  - DOE DECISION ON TYPE(S) TO DEVELOP DUE THIS YEAR.
  - HOW EFFECTIVE ARE PROPOSED NEW PASSIVE SAFETY SYSTEMS?
  - ARE CURRENT COMPUTER CODES APPLICABLE TO THE NEW GEOMETRIES?
- INTERNATIONAL CODE MAINTENANCE CONSORTIA, AFTER THE INTERNATIONAL CODE ASSESSMENT PROGRAM ENDS IN 1991.
- CONSIDERING ACRS SUGGESTION TO BROADEN RESEARCH INTO "THERMAL SCIENCES"
- CANDU AND PIUS SAFETY ISSUES

6. OTHER PLANNING ASSUMPTIONS

- NO NEW SOFTWARE DEVELOPMENT FOR ACCIDENT MANAGEMENT
  - USE EXISTING CODES (RELAP5/SCDAP, MELCOR)

- WILL CONTINUE TO SUPPORT FUNDAMENTAL RESEARCH, WHERE NEEDED, THROUGH BROAD AGENCY ANNOUNCEMENTS, AS WELL AS THROUGH CONTINUING GRANTS PROGRAM. WILL CONSIDER ACRS SUGGESTIONS:
  - FLOW-INDUCED VIBRATIONS
  - HEAT TRANSFER UNDER OSCILLATING FLOW
  - VALVE BEHAVIOR WITH CHANGING FLOWS.
  - FLOW EFFECTS ON EROSION/CORROSION

- DETERMINE BEST METHOD FOR PRESERVING RESEARCH RESULTS FOR FUTURE USE (ACRS SUGGESTION)

SUMMARY OF RESPONSE TO ACRS LETTER OF JUNE 15, 1989

- AGREE WITH ACRS SUGGESTIONS FOR:
  - CODE DEVELOPMENT: CODES NOW GOOD ENOUGH, AFTER FINAL VERSIONS FINISHED IN CY 1989.
  - EXPERIMENTATION: UNIVERSITY LOOPS AND FUNDAMENTAL RESEARCH.
  - DATA ANALYSIS: PRESERVE RESEARCH RESULTS FOR FUTURE USE.
  - APPLICATIONS RESEARCH: ANALYSIS OF TRANSIENTS IN OPERATING REACTORS, DEVELOPMENT OF INPUT DECKS, AND ANALYSIS OF TRANSIENTS OF PARTICULAR INTEREST.
  - BROADEN PERSPECTIVE TO "THERMAL SCIENCES."
- DISAGREE WITH ACRS SUGGESTION TO INITIATE DEVELOPMENT OF NEW SYSTEMS CODE.

POLICY ISSUE
(Commission Meeting)

July 24, 1989

SECY-89-219

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: STATUS AND PLANS FOR THERMAL HYDRAULIC RESEARCH CONDUCTED BY THE OFFICE OF NUCLEAR REGULATORY RESEARCH

Purpose: The purposes of this paper are to:

1. Brief the Commission on the current status of this research.

2. Inform the Commission of future goals and directions of this research.

Background: Thermal hydraulic research conducted since the inception of the NRC in 1975 and its predecessor agency, the AEC, provided the basis for the recent revision of 10 CFR 50.46 and Appendix K, issued in October 1988. This rule revision reflects the results of the large amount of work performed during the late 1970's and early 1980's, summarized in Reference 1. The revised rule permits realistic analysis of loss-of-coolant accidents (LOCA), while retaining the option to use the former prescriptive, artificially conservative approach. Rescinding the requirement to use artificial, overly conservative analysis methods will allow licensees more operational flexibility, such as extending useful life by lowering neutron flux exposure of reactor vessels and concomitant rate of vessel steel embrittlement, and increasing plant capacities. Eventual revision of the LOCA Rule was a goal set forth by the Atomic

Contacts:
D. Bessette, RES, 49-23572
L. Shotkin, RES 49-23530

Energy Commission when it adopted the rule in 1973 (Reference 2), and this has now been achieved. In addition, research has studied a spectrum of small break LOCAs and transients, and computer codes have been developed and assessed for such events. As a result of these research accomplishments, our understanding of and confidence in predicting LWR thermal hydraulic performance during transient and accident conditions has been greatly improved. Commensurate with this improved level of understanding, our research efforts in this area have decreased significantly over the last several years and are scheduled to decrease further. At this juncture, therefore, it is appropriate to inform the Commission of the current research status and the future goals and directions for thermal hydraulic research (Reference 3). Appendix 1 reviews the history of this research.

<u>Discussion</u>: Thermal hydraulic research is intended to support the Staff in the following areas:

- Understanding reactor transient events and their broad implications for operational safety;
- Detection of previously unrecognized issues important to safety;
- Investigating and resolving specific issues, e.g., the effectiveness of decay heat removal via feed and bleed;
- Evaluating the effect of design and operations-related changes, including operating procedures and changes to technical specifications and setpoints;
- Analyzing the early phases of risk dominant accident sequences and other postulated severe accident scenarios;
- Evaluating strategies and procedures for accident management;
- Evaluating the reactor and plant systems designs of new standardized LWRs; and

- Confirming safety margins in licensee analyses by performing audit analyses.

Thermal-hydraulic research is conducted under the plant performance program element, and includes three activities. The status and schedule of each activity is summarized briefly in the following.

1. Babcock and Wilcox Testing

The basic objective of this work, endorsed by the ACRS, is to provide a data base for B&W designs that is comparable to that which exists for the other NSSS vendors. Test facilities such as LOFT, SEMISCALE and FIST provided integral facility data for Westinghouse, Combustion Engineering and General Electric designs. RES developed and is carrying out a plan to provide comparable data for B&W designs (Reference 4).

Following the TMI-2 accident, the best-estimate codes TRAC and RELAP were used to analyze Babcock and Wilcox (B&W) designs for small break LOCAs. Similar analysis was performed by B&W. Significant discrepancies in calculated plant response were noted which could not be resolved due to insufficient experimental data. The lack of data to validate the calculated results led to the establishment of the Integral System Test (IST) program in 1983. This program included the construction of the Multi-loop Integral System Test (MIST) facility and the performance of a small break LOCA test series under a cooperative, cost-sharing arrangement among the B&W Owners Group, B&W, EPRI, and NRC. The successful completion of this program has provided a small break LOCA data base, and the codes are currently being validated against these data.

Several transients that occurred in B&W plants (e.g., 1985 Davis Besse and Rancho Seco events) since the establishment of the IST program indicated that the unique thermal hydraulic behavior of B&W plants resulting from steam generator design was not limited to small break LOCAs, but included non-LOCA transients as well. This resulted in a follow-on test series in MIST to investigate transient behavior, as well as initiating discussion on the need for experiments on once-through steam generator (OTSG) performance. Recently, agreement was obtained with the B&W Owners Group to conduct a cooperative experimental program on OTSG performance. This experimental program, expected to be completed in FY 1992, will make the experimental thermal hydraulic data base for B&W plants comparable to that which exists for the 3 other vendors.

2. Experiments and Analysis

It became apparent around 1975 that the cost for NRC to unilaterally obtain large scale experimental data necessary to resolve the LOCA/ECCS issue was prohibitive, so RES began discussions with Japan and the Federal Republic of Germany on the conduct of a joint program. Several years of planning and negotiation led to the formation of the 2D/3D program, which began in 1980 and will be completed in 1990. Three large facilities were constructed, two in Japan (Cylindrical Core Test Facility, Slab Core Test Facility) and one in Federal Republic of Germany (Upper Plenum Test Facility). The RES contribution included advanced instrumentation for these facilities, and the development of advanced analytical tools needed to model the complex phenomena being studied. Experimentation will be completed in the current fiscal year, analysis in FY 1990, and final reporting in FY 1991.

Following the TMI-2 accident, Japan decided to build a large scale (1:50) integral test facility to investigate small break LOCAs in PWRs. A number of ancillary facilities also make up the program known as ROSA-IV. RES interacted with Japan from the start of the program and provided advanced instrumentation to the facility. A bilateral agreement was signed in 1984, experimentation began in 1985, and the cooperation currently extends to 1992.

Several past experimental programs such as LOFT, SEMISCALE, and FIST successfully completed their mission to provide thermal hydraulic data for Westinghouse, Combustion Engineering and General Electric designs. These facilities were decommissioned since the cost to maintain them in standby mode could not be justified in the absence of an immediate demonstrated need. The closure of these facilities left the United States with no domestic experimental facilities to provide information on small breaks or transients in non-B&W designs. ROSA-IV helps fulfill this role, and RES periodically makes requests to the Japanese to perform experiments relating to different issues.

3. Modeling

This activity includes code development, code assessment, and code applications.

The International Code Assessment Program (ICAP), a cooperative effort among 14 countries, is the principal source of independent assessment of the RELAP and TRAC codes. Additional input is received from the experimental programs described above. The assessment results and discussions held under the ICAP program provided a common understanding of the performance of the RELAP and TRAC codes for modeling PWRs. RES developed a code improvement plan to guide the code development during the FY 1988-1989 time period. This will culminate in the last planned versions of these codes, which will be released by the end of the current year. Following their release, the codes will be assessed under the ICAP program through 1991. No new versions of these codes are planned unless assessment results show significant deficiencies in the ability to predict nuclear plant performance.

In this code development activity starting with the effort to revise the ECCS rule, RES has emphasized code documentation and software quality assurance. Documents were issued to describe all models and correlations contained in the codes, their source, data base, and range of applicability. User guidelines for nodalization and applying the code were improved and documented. An independent review was conducted of software quality assurance procedures in use at INEL and LANL to assure conformity with ANSI standards and prevailing industry practices.

Future Research Plans

The governing objective for future thermal hydraulic research was developed as a joint, staff position. RES and the principal user office, NRR, interact on a routine basis, both formally and informally (Reference 5). AEOD also participates in this process. The formal mechanisms are: (1) the Reactor Systems Safety Senior Research Program Steering Group, headed by the Director, Division of Systems Research, RES and constituted at the Division Director level; and (2) a Regulatory Research Review Group constituted at the Branch Chief level.

During 1988, these groups reviewed future thermal hydraulic needs. This review concluded that:

- The NRC staff will continue to need independent expertise and analytical capability for addressing transients in PWRs and BWRs; and

- The principal RES analysis codes (TRAC-PWR, TRAC-BWR, RELAP5, and RAMONA) should be maintained for active use.

The recent revision to 10 CFR 50.46 and Appendix K does not terminate the need to maintain expertise in the thermal hydraulic area. Nuclear safety encompasses a number of engineering disciplines; one of the more important being the field of thermo-fluid mechanics. Issues requiring thermal hydraulic analysis arise periodically and the Staff is called upon to address them. Such issues typically require code calculations. The codes are large, complex and require experienced users.

Code development has proceeded in an interrelated fashion with experimental programs. Most of this work is complete; however, certain projects remain. The 2D/3D program will end in 1990. The experimental programs associated with B&W designs will finish in 1992. The Japanese ROSA-IV program of integral and separate effects experiments on small breaks and transients will continue until 1992. The ICAP program continues until 1991. RES will complete these efforts, to the extent appropriate, and incorporate important information into the codes.

Several policy issues are present with respect to the near term and longer term future. These are described in the subsequent bullets, followed by the Staff's plans.

o Thermal hydraulic research has been oriented towards resolution of specific issues such as ECCS performance and pressurized thermal shock. These and other issues have been resolved, and the current program is aimed at resolution of other remaining issues. Absent specific new issues, what level of thermal hydraulic research should be pursued and to what end?

Future projects must be technically challenging to attract and retain researchers and must be consistent with the NRC's mission. We plan to:

1. Complete the current projects within the three activities described previously, which should occur by 1992.

2. Continue code maintenance for the RES PWR and BWR systems codes, and keep the software in pace with advances in computers.

3. Perform research as appropriate, directed at improving the accuracy of the codes in areas determined to warrant improvement.

4. Interact with domestic and foreign research programs in the subject area.

5. Continue and improve our utilization of university expertise.

We have made a determination of the minimum research support level needed for the long term to assure that adequate expertise is available to the NRC. In doing so, our contractors have been consulted. This determination considers the specialties and numbers required for group dynamics and research peer review. The Staff believes that, in the long term, the budget for contract research in this program element should be maintained at about $3M. The current fiscal year budget for plant performance research is $8.1M and declines to $5.8M in FY-1990. For reference, Figure 1 shows the history of RES thermal hydraulic funding. Planned RES professional staffing for the plant performance program element is three FTEs.

It is expected that a significant fraction (i.e., approximately 25%) of the work will be university projects, or joint national laboratory/university projects. University involvement will be sought to utilize academic expertise. With the success of the B&W test loop at the University of Maryland, plans are being formulated to initiate other test loops at universities to provide experimental data and models in the future.

o In the past when a large effort was underway, research was conducted at most of the national laboratories, as well as with other contractors. Now, the level of effort is greatly diminished and is limited primarily to three national laboratories. To what extent should the research be further consolidated?

Several years ago as part of its plan for technical integration of thermal hydraulics (Reference 6), the NRC established a policy of concentrating its remaining thermal hydraulic research at INEL. This step was taken because research reductions had raised a concern that an effort dispersed among several national laboratories would result in groups too small to have effective group dynamics and internal peer review. INEL was chosen since it had the largest remaining thermal hydraulics group and has performed well.

Certain work continues to be performed at LANL and BNL. At LANL, the research centers around the TRAC-PWR code, which was developed at that laboratory. The BNL effort is associated with the RAMONA code, used for analysis of reactivity transients in BWRs. To date and for the near future, this arrangement is the most efficient. We would of course continue support for university and international test programs, as required.

The possible negative aspects of consolidation were considered, three of which can be identified. First is the effect of competition, or lack thereof, of different laboratories submitting proposals on a given issue. Second, is the question of assuring sufficient external input and review of the research. Here, interactions with universities and domestic and international organizations can be utilized to overcome this difficulty. In terms of project management, a benefit is obtained from consolidation in that the tasks of coordination and integration become less important compared to planning and review of the research. The third possible detriment of consolidation is the question of whether unique expertise becomes lost if laboratory programs are terminated arbitrarily. These factors mitigate against immediate, complete consolidation. We are proceeding with these factors in mind.

o Industrial and international cooperation have played an important role in the research effort. The intent was to pursue cooperative programs with industry and international organizations whenever subjects were identified which would be mutually beneficial to resolve via cooperative research. To what extent should industrial and international cooperation be continued?

Our current domestic industrial thermal hydraulic cooperation consists of the B&W Testing Program, while our international cooperation includes: four agreement based programs (2D/3D, ROSA-IV, BETHSY, and ICAP); participation in CSNI; and temporary assignments of technical personnel. The NRC has, in the past, played a leadership role in fostering international cooperation. Also, because of the extensive past research, the RES codes RELAP and TRAC are currently the world standard for safety analysis of reactor transients. This technological leadership may be difficult to maintain in the future due to strong research programs underway in certain countries, notably France, Japan, and Federal Republic of Germany.

We propose to: (1) continue a multilateral program based on RELAP and TRAC following expiration of the current ICAP program in 1991; (2) continue to explore bilateral research agreements of benefit to the NRC; (3) continue to explore cooperative programs with domestic industry (including EPRI); (4) continue to participate in CSNI activities; (5) continue to accept temporary assignments of foreign technical personnel (although few of these are anticipated in the future); and (6) consider temporary assignments of U.S. technical personnel to leading foreign research programs. The multilateral program (Item 1) will be developed with the following objectives: (1) share the effort and resources required for code maintenance; (2) retain access to foreign experimental facilities and expertise; and (3) continue international cooperation concerning improved analysis of reactor transients.

- Considering that a large amount of information and considerable expertise has been developed from past research, what future applications of such knowledge to assist the Staff are envisaged?

A new program element, Reactor Applications, was initiated last year to apply the research results from the Plant Performance element. Reactor Applications research consists of performing LWR systems studies for both operating reactors and those of advanced design. Thus, Reactor Applications is concerned with resolving issues by application of information and computer codes developed through Plant Performance research. This element includes three activities: (1) Analysis of Operating Reactor Events; (2) Light Water Reactor (LWR) Systems Studies; and (3) Thermal Hydraulic Technical Support Center.

The first activity relates to issues highlighted by operating experience. For example, a subject of recent interest, as a result of the LaSalle event of March 1988, is BWR stability. The purpose of the research in progress is to determine the extent of safety concerns associated with BWR oscillations and whether any unacceptable conditions exist that might warrant regulatory action. To accomplish this, we are: (1) performing code validation for such applications; (2) performing additional ATWS analysis to determine whether oscillations would be expected and their consequences; and (3) determining the key parameters affecting onset and amplitude of oscillations and their mode (uniform or nonuniform).

The instability research is the most recent example of research needs identified as a result of operating experience, either directly or indirectly. Issues continue to arise requiring analytical assessment in the area of plant response to off-normal transients.

The second activity (LWR Systems Studies) is associated with advanced LWR designs. The first task is to determine whether, and in what ways, the transient response of advanced LWRs differs from current designs. From this, a determination will be made of whether the thermal hydraulic codes are validated for such applications. The second task includes code validation and model development, as required, to establish or improve the applicability of current analytical tools to transient analysis of advanced LWRs. The goal of this effort is to establish the accuracy and reliability of the current RES LWR systems codes for the new plant designs. Since these new designs have not been fixed, the long-term funding level for this activity may require further definition.

The third activity cited (Technical Support Center) includes three tasks. First is to help resolve regulatory issues. Two examples of the types of issues that are addressed are long-term cooling following a LOCA, and the consequences of an interfacing systems (high-low pressure) LOCA. The second task is the preparation of plant input decks. Plant designs are generally unique, and the particular design is important to the outcome of a given transient scenario. This task is concerned with extending the library of plant models available for analytical studies. The third task within this activity is concerned with the synthesis and integration of information on given subjects. Information normally exists in the form of a multitude of reports on experiments and code calculations. The task of synthesis and integration is intended to distill this information into a form that can be more easily utilized.

The Reactor Applications program element is funded at $5.3M for FY 1990 and planned at about $3 to $4 million in later years. This assumes that there will be no major thermal hydraulic test facility for advanced LWRs. The RES staffing level is three FTEs.

Coordination: The ACRS has reviewed this approach for future thermal hydraulic research and they agree with the general objective of the research program to maintain, within the NRC and its contractors, a capability for thermal hydraulic analysis sufficient to deal with safety and regulatory concerns that might arise in the future. Also, they agree with the general level of funding projected for the next several years. In addition, the ACRS provided several relatively specific recommendations which were, for the most part, consistent with our plans. We intend to continue to interact routinely with the ACRS to ensure they are kept informed and to obtain their recommendations. See Attachment 3 for the ACRS letter and Staff response.

Recommendation: That the Commission take note of the approach proposed herein for the future goals and directions of thermal hydraulic research.

Scheduling: This paper is scheduled to be considered at an open meeting on August 3, 1989.

Victor Stello, Jr.
Executive Director
for Operations

Attachments:
1. Appendix 1: Historical Perspective of Thermal Hydraulic Research
2. Figure 1: Funding for Thermal Hydraulic Research
3. ACRS letter of June 15, 1989 and Staff response

This paper is tentatively scheduled for discussion at an Open Meeting during the Week of July 31, 1989. Please refer to the appropriate Weekly Commission Schedule, when published, for a specific date and time.

DISTRIBUTION:
Commissioners
OGC
OIG
LSS
GPA
REGIONAL OFFICES
EDO
ACRS
ASLBP
ASLAP
SECY

References

1. Compendium of ECCS Research for Realistic LOCA Analysis, NUREG-1230, December, 1988.

2. Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled-Nuclear Power Plants, United States Atomic Energy Commission, Docket No. RM-50-1, December, 1973.

3. Nuclear Power Plant Thermal-Hydraulic Performance Research Program Plan, NUREG-1252, July, 1988

4. Thermal-Hydraulic Research Plan for Babcock and Wilcox Plants, NUREG-1236, January, 1988.

5. Plan for Integrating Technical Activities Within the U.S. NRC and Its Contractors in the Area of Thermal Hydraulics, NUREG-1244, March, 1987.

6. Reviews of Modern Physics, Volume 47, Supplement 1, 1975.

APPENDIX 1

Historical Perspective of Thermal Hydraulic Research

For many years following its emergence in the mid-1960s, the issue of Emergency Core Cooling System (ECCS) performance was dominant in reactor safety. The need to standardize ECCS analyses for licensing proceedings eventually led to the adoption of 10 CFR 50.46 and Appendix K to 10 CFR 50. Upon promulgation of the original ECCS rule in 1973 the Commission mandated that a research program be carried out to develop a better understanding of LOCA related phenomena. A subsequent review of the research program by the American Physical Society highlighted phenomenological and modeling issues which needed to be addressed to develop a realistic treatment of LOCA/ECCS (Reference 6). This review noted the importance of scaling and systems modeling.

The NRC, as well as other organizations, planned and carried out a large program of thermal-hydraulic experimentation and model development. Fuel behavior experiments were also performed to study LOCA-related fuel issues. This work concentrated initially on large break LOCA since this was believed to be the most limiting event from the standpoint of ECCS effectiveness. The advent of WASH-1400 began to focus attention additionally on small break LOCAs and transients, a process which was greatly accelerated and enhanced by the occurrence of the TMI-2 accident. Emphasis on thermal hydraulic research likewise shifted to small break LOCAS and transients.

The NRC successfully accomplished the original mission of the LOCA/ECCS research. Questions and concerns associated with ECCS performance are no longer important issues. When questions arise regarding LOCAs or transients the experimental data base, analytical tools, and expertise are available to be applied to resolve the issue. This being so, 10 CFR 50.46 and Appendix K were revised in 1988 to permit the use of best-estimate analyses of LOCA for licensing purposes, with appropriate accounting for uncertainties.

The principal products of thermal hydraulic research were computer codes that can be applied to understand and predict plant response to deviations from normal operating conditions. The codes model the plant behavior by describing the processes of heat transfer and fluid flow. Code development and experimental programs proceeded according to a feedback process. As different scenarios were encountered or postulated or potential modeling deficiencies identified, particular experiments were run to obtain data necessary to establish the code accuracy or to improve the code. The interlinkage of code development and experimental programs is such that one cannot exist without the other. Events involving new phenomena were encountered periodically in operating reactors for which code applicability had not been verified. This necessitated a dual analytical and experimental approach to help resolve such issues. The attached table provides examples of issues resolved using this approach.

Since the ability to determine the uncertainty of a large complex systems code was subject to some debate, RES undertook to develop a suitable methodology. The approach is known as Code Scalability, Applicability and Uncertainty (CSAU) methodology. The method is general in concept and was applied to the TRAC-PWR code modeling a large break LOCA in a Westinghouse 4-loop plant. Briefly, CSAU determines whether the code, which is developed and assessed against scaled experimental facilities, is appropriate for full scale applications. It also includes a determination of whether the code has appropriate models for the important phenomena. Finally, CSAU incorporates the ranging of important parameters over their uncertainty ranges, as determined from separate effects experiments. RES believes that the CSAU application produced final closure on the issue of large break LOCA. It provided a best-estimate core peak temperature history for the event, along with a statement of uncertainty. This was done in a scrutable, traceable, and auditable manner.

The process by which the CSAU methodology was developed was as essential as the product. It was a cooperative effort among personnel from three national laboratories, Idaho National Engineering Laboratory (INEL), Brookhaven National Laboratory (BNL), and Los Alamos National Laboratory (LANL). It also utilized input from university professors and consultants. The personnel had extensive experience and expertise, developed largely through their participation in past NRC thermal hydraulic research. The participants proceeded under a technical program group structure headed and closely coordinated by a RES Project Manager. This successful method is now being applied by RES to address scaling issues associated with severe accident research.

While not relevant to the mission of the NRC, it is noteworthy that past RES thermal hydraulic research has produced other unintended benefits. Currently, RES codes are being utilized for safety analysis of production and research reactors. Research into two-phase flow has also been applied to petroleum and chemical industries and to space applications.

Role of International Cooperation

International cooperation has played an important role in thermal-hydraulic research. The goals RES pursued were to: share safety technology in the interest of international nuclear safety; obtain access to foreign experimental results and expertise; and save resources by carrying out large programs with other countries on a shared basis. The principal means of international cooperation RES has undertaken are described in the following.

It became apparent around 1975 that the cost to obtain large scale experimental data necessary to resolve the LOCA/ECCS issue unilaterally was prohibitive, so RES began discussions with Japan and the Federal Republic of Germany on the conduct of a joint program. Several years of planning and negotiation led to the formation of the 2D/3D program, which began in 1980 and will completed in 1990. Three large facilities were constructed, two in Japan (Cylindrical Core Test Facility, Slab Core Test Facility) and one in Federal Republic of Germany (Upper Plenum Test Facility). The RES contribution included advanced instrumentation for these facilities and the development of advanced analytical tools need to model the complex phenomena being studied in these facilities.

Following the TMI-2 accident, Japan decided to build a large scale (1:50) integral test facility to investigate small break LOCA's in PWRs. A number of ancillary facilities also make up the program known as ROSA-IV. RES interacted with Japan from the start of the program and provided advanced instrumentation to the facility. A bilateral agreement was signed in 1984, experimentation began in 1985, and the cooperation currently extends to 1992.

Until 1984, RES made its thermal hydraulic codes openly available through the National Energy Software Center. Different countries obtained the codes and modified versions proliferated. RES did not attempt to obtain feedback from foreign users nor were foreign code assessment results particularly useful, being performed as they were with various unique code versions. Until that time RES had sponsored its own independent code assessment program. Budget constraints made this no longer possible, therefore, RES organized the International Code Assessment and Applications Program (ICAP) through a series of bilateral agreements with foreign safety authorities. The ICAP duration is from 1986-91. The intent was to:

- o Develop a common understanding of the ability of the code to appropriately represent important physical phenomena;

- o Share user experience on code assessment and present a well documented assessment data base;

- o Share experience on code errors and inadequacies and to cooperate in removing the deficiencies to maintain a single, internationally recognized code version; and

- o Establish and improve user guidelines for applying the code.

The ICAP program is successfully accomplishing these goals. Participating organizations have benefited from the establishment of a code users group. ICAP provides the NRC access to foreign facilities, experimental results, and expertise.

An additional role was played by the Committee on the Safety of Nuclear Installations (CSNI) thermal-hydraulics working group. This provided a forum for multilateral discussion and information exchange. It also provided a structure for the conduct of international standard problem exercises to evaluate how well codes and analysts could calculate thermal hydraulic experiments.

Mention should be made of the role of temporary assignments of foreign technical personnel to national laboratories, principally INEL, where about 70 such assignments have occurred since 1974. The purposes of these assignments included contributing to collaborative programs, training, and liaison. On the U.S. side, a limited number of assignments to collaborative programs with Japan and Federal Republic of Germany have taken place, the last of which is ending this fiscal year.

Cooperation with Domestic Industry

Where appropriate, RES has cooperated with U.S. industry in carrying out research programs. The establishment of joint programs is normally sought as a means of sharing the costs involved and in broadening the technical input to given projects. The industry groups most often involved are the Electric Power Research Institute (EPRI), the NSSS vendors, and utility owners groups. Past examples include cooperation with: General Electric and EPRI on the development of the TRAC-BWR code and the conduct of the FIST and BWR-FLECHT experimental programs; and Westinghouse and EPRI on the conduct of the FLECHT, FLECHT-SEASET and MB-2 experimental programs. Currently, the only cooperative program with industry is the Integral System Test program with the B&W Owners Group, B&W, and EPRI.

TABLE 1

EXAMPLES OF DUAL EXPERIMENTAL/ ANALYTICAL APPROACH TO RESOLVE ISSUES

Issue	Experiments	Analyses
Margin of Conservatism in Appendix K; Revision to Appendix K (LOCA)	LOFT, Semiscale, UPTF, CCTF SCTF, TLTA, SSTF, FIST, LOBI, Marviken, PKL, Creare, BCL	Analyses of Test Facility Data and Full-Scale LWRs
Pressurized Thermal Shock	Creare, Purdue, Semiscale, UPTF, HDR, Finland, and H. B. Robinson	TRAC-PWR and RELAP5 Analysis of Oconee and Calvert Cliffs
Small Break LOCA in W PWRs	Semiscale, LOFT, ROSA-IV, LOBI	TRAC-PWR and RELAP5
Small-Break LOCA and Natural Circulation in B&W Reactors	MIST, OTIS	TRAC-PWR and RELAP5 Analysis of Data
Small-Break LOCA in BWRs	FIST, TLTA	TRAC-BWR Analysis of Data
Feed-and-Bleed Procedures for Decay Heat Removal in PWRs	Semiscale S-SR-1, 2, and S-PL-3. LOFT LP-FW-1.	SASA Analyses
Performance of Upper Head and Upper Plenum Injection in W Reactors	Semiscale S-UT series, 2D/3D, CCTF, SCTF	COBRA/TRAC, TRAC-PWR
Small-Break LOCA with loss of High-Pressure Injection	ROSA-IV, LOFT, LP-SB-3, Semiscale S-NH series	RELAP5, TRAC-PWR, SASA Analyses
Liquid Holdup in Steam Generators During Small-Break LOCAs	Semiscale S-UT-6, S-UT-8, S-LH-1, S-LH-2, ROSA-IV	RELAP5, TRAC-PWR, NOTRUMP
Steam Generator Tube Rupture (SGTR)	ROSA-IV, Semiscale	TRAC-PWR, RELAP5
Anticipated Transients Without Scram (ATWS) in PWRs and BWRs	LOFT L9-3, L9-4, Semiscale S-PL-7, FIST	RELAP5, RAMONA-3B, TRAC-BWR
Iodine Behavior Following SGTR	MB-2, ORNL, Northwestern University	CITADEL, TRAC-PWR, RELAP5
Natural Circulation	ROSA-IV, BETHSY, MIST, SEMISCALE	TRAC-PWR and RELAP5

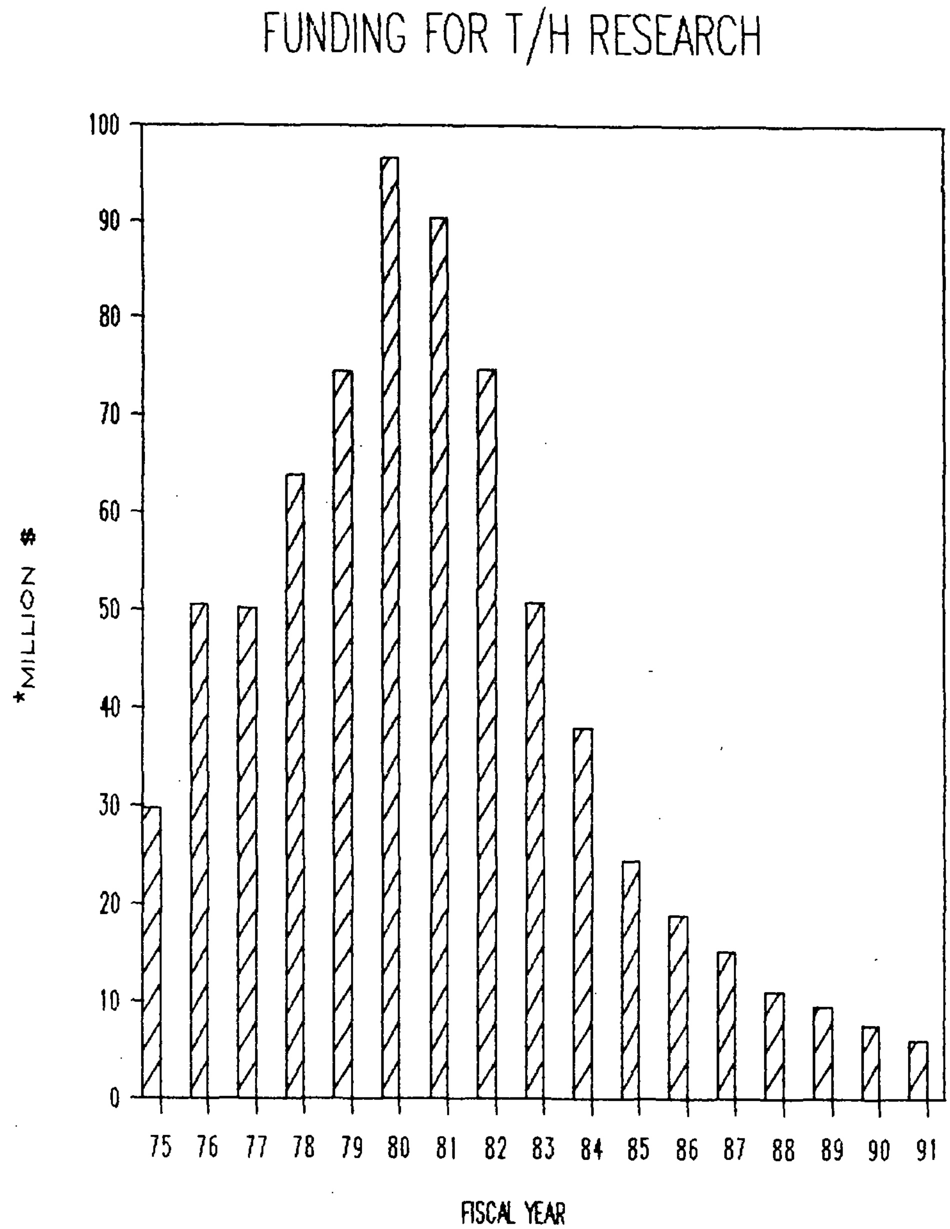
TABLE 1 (CONTINUED)

EXAMPLES OF DUAL EXPERIMENTAL/ANALYTICAL APPROACH TO RESOLVE ISSUES

Issues	Experiments	Analyses
Fluid Structure Interaction on Reactor Core Barrel and Vessel Internals after LOCA	HDR, SAI	K-FIX (FLX)
BWR Containment Pressure Suppression Pool Loads	MIT, GE, Livermore Marviken	PELE, SOLA
Stability Margins for BWRs	DRESDEN, FRIGG	NUFREQ
TMI-2 Accident	Semiscale, S-TMI series, MIST, OTIS	TRAC-PWR, RELAP5
Plant Transients		
ANO-II	LOFT L6-7	TRAC-PWR, RELAP5
Crystal River	Semiscale S-PL-3	
Ginna	LOFT L6-8C	
St. Lucie	Semiscale S-FS series	
Davis-Besse	MIST, OTIS	
Effects of Reactor Coolant Pump Operation During PWR Small-Break LOCA	LOFT L3-5, L3-6, LP-SB-1, LP-SB-2 Semiscale S-SB-P1, -P2, -P7	TRAC-PWR, RELAP5

ATTACHMENT 2

Figure 1



* Dollar Amounts not Adjusted for Inflation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 15, 1989

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: NRC THERMAL-HYDRAULIC RESEARCH PROGRAM

During the 350th meeting of the Advisory Committee on Reactor Safeguards, June 8-10, 1989, we reviewed the NRC's plan for continuing thermal-hydraulic research as related to the design and operation of nuclear power plants. This matter was also considered by our Subcommittee on Thermal Hydraulic Phenomena at a meeting on May 23, 1989. During these meetings, we had the benefit of presentations by representatives of the Office of Nuclear Regulatory Research (RES). We also had the benefit of the documents referenced. The Committee last commented to you on this subject in our report of June 7, 1988.

Thermal-hydraulic research has always been a central and major part of the NRC's research program. Much of the work was inspired by the perceived need to better understand hypothetical large-break loss-of-coolant-accidents (LB-LOCAs) and the performance of emergency core cooling systems (ECCS). Experiments and analytical models, such as the RELAP and TRAC codes, have confirmed compliance with the ECCS rule. Continuing research on LB-LOCAs culminated with a 1988 revision to the ECCS rule which permits licensees to use more accurate means of analysis and makes possible certain safety and operational improvements in existing plants. NRC contractors have demonstrated a methodology that can be used to estimate the magnitude of uncertainty associated with code predictions.

In addition, the experimental information base and the codes have been found useful in assessing and predicting the consequences of transients and small-break loss-of-coolant-accidents (SB-LOCAs) which are now recognized to be much more risk significant than the LB-LOCAs. The codes are also being used to analyze the early stages of severe accident scenarios.

Proposed Research Program

We understand the continuing NRC program in thermal-hydraulic research to have two principal purposes:

- Bring development of the major computer codes to a successful completion.

- Maintain, within the NRC and its contractors, a capability for thermal-hydraulic analysis sufficient to deal with safety and regulatory concerns that might arise in the future. This includes the continuing availability of a cadre of experts.

RES representatives indicated these general purposes would be realized through achievement of several specific objectives:

- The major codes will be maintained indefinitely and some further development will be carried out. The scope and depth of further development seems not to have been decided. Apparently, it will include appropriate reactions to new data from foreign experimental programs and assessments which are expected to continue for some time. It may also include a review and redevelopment of the important constitutive equations in the codes.

- The current experimental programs related to specifics of the Babcock and Wilcox (B&W) nuclear steam supply (NSS) system will be completed. Beyond this, any further experimental programs will be carried out at universities, rather than by the creation or operation of any major facilities at national laboratories. Relatively inexpensive "integral" facilities, of scope similar to the facility now operating at the University of Maryland, are being considered as contrasted with what have been called "separate effects" facilities. These would be mockups of specific NSS systems and of an advanced LWR (600 MWe size) design.

- An expanded program of applications research is planned. Apparently, much of this activity is expected to be in response to issues that arise from experiences with operating plants. But, it will include preparation of input data for several more plant types than are now available to the NRC. This will permit more rapid analysis than would otherwise be possible in response to future safety or regulatory issues. This program may also include exploratory, in-depth studies of a range of possible transients for a variety of plants.

In addition, two other specific program elements were mentioned:

- A further demonstration of the "Code Scaling, Applicability, and Uncertainty" methodology will be carried out for an SB-LOCA with RELAP5/MOD2, similar to that recently completed for an LB-LOCA.

- Improvements will be made to the NSS system process models now incorporated in training simulators at the NRC Technical Training Center. This will permit more accurate simulation of off-normal scenarios for the study of emergency and accident management procedures.

Before commenting on these research proposals, it is pertinent to consider two statements made by the NRC staff at the May 23, 1989 Thermal Hydraulic Phenomena Subcommittee meeting, because the ideas expressed have an influence on our recommendations:

> A representative of the Office of Nuclear Reactor Regulation said, "NRR is not relying extensively on the codes to address current licensing issues."
>
> A representative of RES said, "Codes have now reached an acceptable level of accuracy and maturity... further development is not likely to produce major changes in our understanding of [plant] performance or [accident] consequences."

ACRS Recommendations

We agree with the general objective of the research program to maintain, within the NRC and its contractors, a capability for thermal-hydraulic analysis sufficient to deal with safety and regulatory concerns that might arise in the future. Also, we agree with the general level of funding projected for the next several years. However, we believe there is too much emphasis on further development of the existing codes in the planned program. Maintenance of the needed NRC capability is more a matter of ensuring the availability of a cadre of experienced and expert analysts and access to the general body of experimental data, than it is of improving or even ensuring the availability of large systems codes. The Committee reiterates its comments in the report of June 7, 1988, that "marginal improvements that could be made [in the codes] over the next few years by extrapolating the recent levels of development work will not be sufficient to attain a significantly higher plateau of code accuracy and validation."

To accomplish this general purpose, we recommend a program of four primary elements:

(1) Code Development

Maintain the present large system codes, TRAC-PF1/MOD1, RELAP5/MOD2, TRAC-BWR, and RAMONA-3B, for an indefinite period. Limit improvements only to those required by: (a) the discovery of important errors or (b) crucial new information from the foreign experimental and assessment programs or the B&W testing program. Do not undertake major new restructuring or "zero-based" improvements to the constitutive equations or numerical algorithms in these codes. We are not convinced by the arguments given for the need to develop TRAC-PF1/MOD2 and RELAP5/MOD3. It is our view that the proposed modifications will not substantially improve the codes.

Instead, consideration should be given to the development of a new type of systems code that will be more useful for analysis of extended plant transients involving interactions of plant systems. The Committee also made this recommendation in its June 7, 1988 report. TRAC and RELAP were originally designed to analyze the LB-LOCA, a rapid and severe reactor transient, in great detail. There is a need for a more empirical and efficient analytical tool. We envision a code that would be able, for example, to make a rapid and sufficiently accurate analysis of the power oscillations observed last year at the LaSalle County Station, Unit 2 plant. Such a code would be more akin to advanced simulator codes than to TRAC and RELAP. The BWR code (HIPA) now in use at Brookhaven National Laboratory is an example of the type of code we are suggesting.

(2) Experimentation

The staff proposal to develop relatively inexpensive "integral" test facilities at universities is sound. We see this as consistent with our recommendation for a new type of systems code. We agree that it would be inappropriate to build several such facilities at one time. A gradual approach is warranted. The first such new facility might be one that would incorporate features of the advanced LWR designs. Also, it will be better to completely assess the benefit that has been obtained from tests with the University of Maryland facility mentioned above.

In addition, a small program to deal with more fundamental research should be maintained. These are experiments of the sort that have been previously called "separate effects" tests. An effort should be made to develop a consensus among experts as to which particular phenomenon should be investigated. At this time, we suggest consideration be given to the investigation of:

- fluid-elastic instability related to vibration of tubes in U-tube steam generators,
- departure from nucleate boiling with oscillating flow and power in BWRs,
- dynamic instabilities and loads on valves.

(3) Data Analysis

A major effort is needed to organize data from test programs into a useful form other than the large systems codes. In particular, with the 2D/3D, ROSA-IV, and the B&W test programs all coming to closure, measures are needed to ensure that these expensive and valuable bodies of data are preserved and used. In addition, older data from, for example, the FIST and FLECHT programs can be of greater value if they are effectively organized into more useful forms.

(4) Applications Research

A program in this area should include three elements:

- Analysis of transients indicated to be of interest as a result of plant operating experience.
- Preparation of input data decks for several classes of plants so that turnaround time for analyses in response to experience is shortened.
- Analysis of transients that are indicated by PRA or other sources of information to be of particular interest, but which are not presently well understood. We suggest the following for consideration:
  - feed and bleed scenarios
  - secondary depressurization scenarios

Finally, we suggest that RES broaden its perspective as to what other research in the thermal sciences should be included in its program, rather than being limited to the traditional scope of concerns in thermal-hydraulic areas. We suggest that it include studies of a broad range of thermal and fluid transport issues related to reactor safety.

ACRS Members William Kerr and Forrest Remick did not participate in the review of this matter.

Sincerely,

David A. Ward
Acting Chairman

References:
1. U.S. Nuclear Regulatory Commission, draft SECY Paper: "Status and Plans for Thermal Hydraulic Research Conducted by the Office of Nuclear Regulatory Research," provided to the ACRS in May 1989
2. U.S. Nuclear Regulatory Commission, NUREG-1252: "Nuclear Power Plant Thermal-Hydraulic Performance Research Program Plan," Office of Nuclear Regulatory Research, July 1988

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

JUL 24 1989

Mr. David A. Ward, Acting Chairman
Advisory Committee On Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Ward:

SUBJECT: NRC THERMAL-HYDRAULIC RESEARCH PROGRAM

The Advisory Committee on Reactor Safeguards (ACRS) letter of June 15, 1989, provided comments on the subject program. The ACRS agreed with the general objective to maintain expertise in thermal-hydraulics to meet future agency needs in this field. The Committee also agreed with the general level of funding projected for the next several years. Parallel to the staff's plans, a program comprised of four elements was recommended. staff review provides the following responses:

(1) Code Development

The ACRS may have misinterpreted a staff statement to mean that older versions of computer codes were already mature. This would have led the ACRS to recommend that no further work is needed on the final code versions, TRAC-PF1/MOD2 and RELAP5/MOD3.

The staff believes that the final versions of TRAC-PWR (TRAC-PF1/MOD2) and RELAP (RELAP5/MOD3), as completed in 1989, have an acceptable level of accuracy and that further development is not likely to produce major changes in our understanding of plant performance and accident consequences. Previous versions of these two codes (TRAC-PF1/MOD1 and RELAP5/MOD2) were released in December 1984. Since then, peer review and code applications studies identified a number of modeling deficiencies which we determined must be resolved to provide a reasonable, sufficiently accurate representation of physical phenomena. The knowledge and experience gained over the last five years is reflected in the final code versions. The codes are being provided to the International Code Assessment Program (ICAP) members for assessment to be completed in December 1991, when the ICAP program ends.

We believe that these final code versions, plus the other NRC codes already completed (RAMONA-3B, HIPA, TRAC-BWR, and COBRA/TRAC) provide NRC with sufficient analytical capability to meet its future needs, including any needed analyses of extended plant transients involving interaction of plant systems. This has been confirmed at several meetings among the RES staff and the user offices of NRR and AEOD that were convened for this purpose. The staff thus concludes that no new systems code development is required at this time.

(2) Experimentation

We note that the ACRS endorses our plan to develop university experimental facilities. We intend to clearly define what we expect from any new test facilities at universities before any proposals to perform this research are issued. Concerning the suggestion to maintain a small program to deal with more fundamental research, we intend to include this in a Broad Agency Announcement for proposals similar to the ones you have suggested. In addition, we will continue our grants program which has supported such research efforts for several years.

(3) Data Analysis

You have raised a concern about how best to preserve our research results for use by "future generations" of reactor safety experts. You are aware that we have an experimental data bank, code maintenance programs, and publish research synthesis reports on special topics. However, these existing programs may not fully address your concern. The staff will review our current capabilities and identify any ways we can improve on putting completed research into the most useable form. We would appreciate further interactions with the Committee regarding where they think our current programs are deficient or could be improved.

(4) Applications Research

We agree with the three elements you have suggested, and would add a fourth, code applicability studies. Such studies better define the classes of reactor geometries and accident scenarios for which specific computer codes are presently applicable. This would help to assure that code performance is well understood.

With regard to the suggestion of including other research in the thermal sciences, we propose to discuss this further at a subcommittee meeting in order to understand what the ACRS has in mind.

Sincerely,

Victor Stello, Jr.
Executive Director
for Operations

cc: Chairman Carr
Commissioner Roberts
Commissioner Rogers
Commissioner Curtiss
OGC
SECY

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

OFFICE OF THE
COMMISSIONER

June 26, 1989

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

FROM: Kenneth C. Rogers

SUBJECT: ACRS LETTER OF JUNE 15, 1989 ON NRC THERMAL-HYDRAULIC RESEARCH PROGRAM

I consider the subject ACRS letter excellent and lucid advice on the Office of Nuclear Regulatory Research (RES) thermal-hydraulics research program. I support the ACRS recommendations in the letter and the four primary elements of this program. Please inform me of any RES comments on the ACRS recommendations or their plans for implementation.

Kenneth C. Rogers

Kenneth C. Rogers
Commissioner

cc: Chairman Zech
Commissioner Roberts
Commissioner Carr
Commissioner Curtiss
OGC
SECY

Rec'd Off. EDO
Date 6-27-89
Time 9 A

EDO---004581