JWBeach
10/23/80
NRC-ACRS
Idaho
Falls
CS Sb
ARC:970

J 195

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

* * *

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON EMERGENCY CORE COOLING SYSTEMS

* * *

Westbank Motel Coffee Shop,
475 River Park,
Idaho Falls, Idaho,

Thursday, 23 October 1980.

The subcommittee was reconvened, pursuant to recess, at 8:30 a.m., with Dr. Milton Plesset, Chairman of the Subcommittee, presiding.

PRESENT FOR THE ACRS:

DR. MILTON PLESSET, Chairman
HAROLD ETHERINGTON, Member
WILLIAM MATHIS, Member
JEREMIAH RAY, Member
DR. ZUDANS, Consultant
DR. WU, Consultant
DR. ACOSTA, Consultant
DR. CATTON, Consultant
DR. THEOFANOUS, Consultant
DR. BATES, Federal Employee

PRESENT FOR THE NRC:

Messrs. Sheron, Sullivan, and Lyon

* * *

8010310002

CONTENTS

* * *

PROCEEDINGS

(8:30 a.m.)

DR. PLESSET: Good morning. This is the second day of our subcommittee meeting, and our agenda today is to go into the LOFT program and some information about the LOFT tests.

Let me first see if there are any comments that our committee members would like to make before we begin.

(No response.)

DR. PLESSET: Consultants, any comments you would like to make before we begin?

(No response.)

DR. PLESSET: They are going to save them for later.

I think we will turn to Mr. Kaufman, if he is prepared to give us an overview of the LOFT program.

MR. KAUFMAN: Good morning, and thank you.

I consider it really an honor and a privilege to give you an overview of the LOFT program today. I think it is very important, because LOFT as a program has been evolving and changing very rapidly, and I think sessions such as today give us a chance to highlight what that program is like by giving you a few presentations that are designed to illustrate the various parts of our program.

The point that I hope when we're done today I can illustrate is that LOFT was originally of course conceived to be a program or a project designed to study the large-break loss-of-cooling accident, and particularly to take thermal-hydraulic data related to that.

The program has since evolved very rapidly into a program no longer confined to large-break LOCA, or LOCA for that matter. Indeed, we are trying to look at a full spectrum of accidents of the sort that are postulated in the FSAR. Our data taking is no longer confined to thermal-hydraulic data or measurements. Indeed, we are trying to measure what happens in the pipe, inside the plant, what went on, but we're also trying to study how the sensors that are typical in a PWR measure that phenomena and how the operator interprets what those sensors see.

Additionally, we have moved into the area that becomes very esoteric, where in fact we have questionnaires and psychologists, and human factors people working with our operators, reviewing our methodology of display to try to assess such immeasurable things as frustration and goodness, areas where we really are without good tools and techniques for measurement.

(Slide.)

Now the LOFT program as we see it today has evolved to a program wherein we take a nuclear reactor, we intentionally place it into conditions that are characteristic of an accident in order that we can measure what happens, test techniques for recoveries, assess methods for accident recognition. We hope that these data can be used to improve predictive or anticipative tools. We hope these then could be used to gain some perspective of the balance between operator actions and the actions of automatic safety systems such as the ECC system.

(Slide.)

We have come a rather long way in a short period of time. It is hard to remember that LOFT became a nuclear testing facility only in December of 1978, less than two years ago. Since then, we have performed two large-break loss-of-cooling experiments simulating a double-ended break to the main cooling line. We have run four small-break experiments, two each, looking at simulations of four-inch and one-inch equivalent line breaks. We have run four operational transients.

So we have run experiments that span a spectrum from the most probable to the design-basis sort of event. I call your attention particularly to the transition that occurred in the program associated with the accident at TMI and the issues that were raised by TMI, and also the early successes that we found in the program wherein we saw margins as a result of hydraulic behavior that were significant relative to the predictions used in the licensing process.

At that point in time, at that juncture in May of 1979, we accelerated the program by over a year-and-a-half to begin small-break testing and begin operational transient testing.

Since then, we have a record wherein we've been able to modify the program in real time on fairly short notice to add experiments, as we did in the case of test L3-7, and to delete tests when we thought they had limited value as we did in the case of test L3-4.

Today, we view our principal program orientation to be essentially two-fold: One is the more traditional development of an experimental data base that can be used to address issues that confront us in the licensing process; but a very heavy emphasis in our program currently is to use and to evaluate the methods by which accident conditions and accident phenomena can be recognized, can be controlled, and the plant stabilized and recovered.

Now the other speakers today will illustrate some of these aspects of the program. They will talk particularly about our augmented operator capability program which is a program designed to develop methods that can be useful to the operator in recognizing and responding to accidents.

We will talk to you about some of the perspectives and observations that we have obtained in our small-breaks program. And, most importantly I think, too, we want to introduce you to some of the issues and things we have seen in our anticipated or operational transient testing program, a program where in fact we are looking at a different class of insights and gaining some new perspectives.

(Slide.)

To illustrate the value and the need, I think, for this sort of orientation, I have selected examples taken from the NRC Action Plan. I am not sure that it was entirely recognized at the time, but the NRC Action Plan created a rather extensive need for data, and for information, and for experience.

In other words, the items that required response presumed the existence of a basis of knowledge and information which would allow us to improve the operation of the plants.

What we have done is gone through the Action Plan to try to identify those issues -- and I've cross-referenced them -- in which we feel that we can make a contribution by providing a data base and an experience base.

Now the data that we're obtaining -- the data base that we're obtaining -- among other things, is important to training of the shift technical advisors. Indeed, what do we train the advisors and the operators to? What perspectives and what insights do we give him or teach him? What requirements do we demand of him?

Of course the next issue, we've talked here about small-break LOCA, the issue of inadequate core cooling, and the need for experiments and analytical techniques that allow us to address that issue.

Another that again we'll talk about this morning, Dr. Lienbarger will, is an understanding of coolant inventories, mass inventories as a function of time, what plant aspects significantly affect those things, and the stability and the effectiveness of natural circulation -- all information required to respond to these referenced items in the Action Plan.

(Slide.)

The plan goes on and requires the development of emergency procedures and their upgrade. It then requires the NSSS vendors to review those, and for the NRC to review those.

All of that presumes that we know how to upgrade them. All of that presumes that the vendors have an understanding of their plant, and that NRC has a body of information that allows them to do that upgrade and review.

There is a requirement for training for the mitigation of core damage. In fact, the mainstream effort of LOFT currently is to look at techniques by which the operator can intervene into the longer term, longer duration kinds of accidents and effectively take the plant to a cold shutdown state.

The Action Plan further requires the development of instruments for monitoring of accidents and accident conditions, and for the determination of inadequate core cooling. That requirement leads right to a mainstream effort of LOFT wherein in fact we have been measuring accident conditions, and learning about the effectiveness of various devices for that purpose, and in fact the efforts currently are heavily focused on instrumentation for the assessment of liquid level, one component in adequate core cooling.

In fact, the test that we ran just last month, we allowed the liquid levels in the plant to decrease to just slightly above the top of the core. We anticipate very soon to perform an experiment where we allow the liquid levels to reach and in fact penetrate into the top of the core -- experiments which place a very high reliance on our ability to determine liquid level in real time.

(Slide.)

The Action Plan goes on. It requires consideration of the installation of coolant system vents -- vents to release noncondensibles. It presumes that we have the data base to know where those noncondensibles will accumulate and enough understanding of the conditions to know that we can appropriately relieve them.

It requires the development and the location of post-accident sampling systems. Again, a mainstream effort in our planning for tests wherein we expect to release fission products.

We are in the process of developing instrumentation, building on the experiences of PBF for real time sampling of the fission product inventory in the plant in various locations for both during-accident and post-accident sampling.

We have been looking extensively at the interplay between the secondary cooling system, and hence the feedwater in the coolant system and the control valves, and the plant, to gain a perspective as to the kinds of accident conditions under which the feedwater system, the feedwater initiation systems and the scram systems are important. And indeed, we know there's going to be degraded core rulemaking. Where LOFT can contribute there is the development of the thermal-hydraulic conditions associated with the entry into severe damage. These then can be built upon by programs at PBF to look at severe core damage. So again I think we can contribute to that aspect of the Action Plan.

(Slide.)

Now the items in the Action Plan, although extensive and I think related very heavily to the kinds of work we're doing, I think there are some other important utilization of the data base that we're generating.

Specifically, we can use our plant to resolve specific licensing concerns, specific issues that come up in the regulatory process. And I think the pumps-on/pumps-off issue is an example of that kind of participation.

We believe that we can use the data that we're doing to improve the methods by which we characterize plant accident response. I think that that is a point that in one sense seems obvious, and yet in another sense is perhaps subtle.

We have found that as you move into the regime of trying to assist the operator, as you move into the area of trying to improve control systems and training, it becomes necessary to more and more characterize what actually happens in the plant, rather than to bound the conditions that will arise.

The safety analysis process very heavily is focused on assuring that we've bounded or enveloped what happens in the plant under transient conditions. Training requires you to know what actually happens. And as you move into that philosophical area, or into that issue, you find that the codes are still wanting, our analytical techniques are still wanting, and indeed our understanding is still somewhat wanting.

So although in the one sense certainly we have shown that the analytical tools used in the safety analysis process are likely to produce significant conservatisms, we are finding that when challenged at the level of actual prediction, there is still some room to go.

Finally, as you are well aware, throughout the world there are separate effects components tests going on. Our facility can be used to establish boundary conditions to provide some perspective as to what's important to study and what not. It is through this area that we are very closely tied to the Semiscale program that you heard about yesterday. Indeed, Semiscale has the flexibility to study parametrically many sensitivities and many issues. We can use Semiscale in conjunction with LOFT to try to focus in on the issues of concern to try to complement the importance of the nuclear side of LOFT, and the greater flexibility inherent in Semiscale.

(Slide.)

Now in addition to efforts associated with data base, I wanted to highlight some of the activities associated with the development of operational methods, operational techniques, and indeed to focus on the operator operating a nuclear plant under an accident condition.

Again, the Action Plan has a lot to say about it. The Action Plan requires us to assess control room staffing requirements which presumes that we know what instrumentation is important as a function of time, what controls are sufficient as a function of time, and how operators inter-react with the process and react to what they see on the consoles, and how they can put the puzzle together.

Again, we're required to develop emergency procedures in an operational sense, and to train to them, and to review to them. Again it presumes a body of information -- a body that I think LOFT is uniquely suited to provide.

It also requires upgrade plans for control rooms and NRC audit of those plans. Again, that presumes the existence of a body of information and a body of experience on what is good and what is not good. Certainly there have been many studies of control rooms since TMI, but I think in fact the practical body of experience of what is important and what is not important in coping with an accident condition is sorely lacking.

And in that area in order to try to assure that our program is integrated with the activities in the rest of the industry, we have participated in the industry groups, and in fact with the aerospace industry as well as the human factor societies, and on and on. I am sure Mr. Meyer will talk more about that.

We have moved, in the course of one year, from almost a standing start in the area of advanced diagnostics and advanced control rooms, to literally a position where we are recognized internationally as being a center for those kinds of studies.

(Slide.)

The NRC Action Plan goes on to talk about training of operators to cope with core damage and to mitigage the consequences.

It again requires the development of instruments so that the condition of the plant can be monitored.

It requires developing and upgrading of emergency support facilities. Again, another area that LOFT has moved into, we recognized early on that we had an opportunity to look at things like technical analysis centers, technical support centers. So very quickly we set up an operational technical support center, and had it manned and operating as we conduct these accident tests. We are learning a number of things that I think have value. For example, how you qualify personnel to man such a center; what requirements you place on those personnel for knowledge.

The difficulty of developing a meaningful technical support center which uses advanced computer technology, and yet confronting the requirements for safety-relatedness and for coming up-to-speed in the technical emergency in a very rapid fashion. Our center today doesn't meet all of the regulations and requirements on safety-relatedness, but indeed it has forced us to confront a great many of the problems, and I think our experience is of high value.

Incidentally, when we hosted the Utility Technology Transfer Meeting last week, that was an area where we received a great deal of feedback and a great deal of interest. The problems that we are confronting in terms of selecting information to be displayed in the technical support center I think are also of high value. They're the same problems that the industry will have to face -- they get hidden under various names: safety-state vectors, and so on -- but fundamentally it is the selection of parameters that permit us to follow in a technical sense, and to cope with abnormalities on an accident condition. These are the same problems, of course, that are going to be associated with development of emergency response centers.

(Slide.)

But in addition to these items from the Action Plan, I think our efforts in operational methods have additional value. We are in fact looking at ways in which you can collect information to recognize that an accident is in progress, and to help give the operator some feedback as to whether the situation is getting better or not.

We are trying to look at methods by which information can be validated in real time. The worst thing we can do is to focus the operator on a display of information about the state of the plant, and then put wrong information into that display.

We are trying to take the techniques that have been developed here at INEL for the validation of data --

300 7TH STREET, S.W., REPORTERS BUILDING, WASHINGTON, D.C. 20024 (202) 554-2345

techniques developed in Semiscale and LOFT and PBF -- and apply those in real time to qualify data provided in those displays. In other words, this signal is believable because I can verify it in certain ways; this signal is of questionable believability; this signal, by a certain series of screenings, is likely to be failed -- to provide, in real time, that kind of advice. And incidentally, we do have some prototypical regimes running now in real time in computer hardware.

Finally, we are looking to say: How can we take what we know, both in terms of operator actions, in terms of planning, and in terms of plant responses, and develop new ways by which the operator can respond to accident conditions? Is there a way that we can simplify the controls? Are there preferred responses such that we can simplify the training and the technical knowledge required of the operator? Because indeed the operation of these plants is becoming extremely complex; the amount of information we're requiring him to know as a result are becoming higher and higher; and I believe that there is a strong need to look at new ways and preferred ways of responding to accident conditions to simplify the process.

(Slide.)

Now the fact that we are a nuclear facility, the fact that we're looking at some of these things, has forced us into developing what amount to small programs within the larger LOFT program. I wanted to give you a brief introduction or overview to some of those.

Of course for years we have been heavily involved in the development of instrumentation and commercialization -- that is, teaching commercial industry how to build these kinds of instruments so that they would be available. I think you are quite familiar with those kinds of activities.

But in fact, we are developing equipment to cope with and clean up from severe fuel damage. LOFT, in its overall plan, has tests wherein we expect to damage fuel. That means for us that that is not a contingency situation; it is a situation for which we have to plan, and plan to effectively conduct. Therefore, we're developing techniques by which the plant can rather quickly be re-entered, cleaned up, and turned around so that we can run these tests again.

As a result, we have developed equipment covering a range. For example, we have a unique system on waste gas cleanup, a system that both filters gaseous fission products, as well as cryotraps. We've had to worry about all of the problems with safety analysis, and so on, that goes with such a system.

We have had to worry about isolability of our control room -- the ability to continue to man the control room under design-basis accident conditions. That data, that information I know has high value to the utility industry dealing with the same problems. They have expressed that as recently as last week's meeting.

Because LOFT is required to meet the same kinds of codes and standards that are typical of the industry, because we then design our plant to withstand seismic events and to withstand the severe design loads associated with accident conditions, we have a tremendous number of snubbers on our plant, and pipe restraints, and all of that. In fact, that has led to a mini-program where in fact we are building snubber test equipment and testing an extreme range of kinds of snubbers, both by manufacturer and by size.

So in fact we are developing information that we are finding is of high value to the structural people in NRC. We are also developing facilities, building on the LOFT test support facility, and in fact have begun testing of relief valves at 1000 psi -- characteristic of our secondary side -- and have been able to conduct full-discharge kinds of tests.

So we again are moving into a new regime of information -- information that right now is very relevant and of high value.

Another one I think is important is we are performing routine field utilization of ultrasonic -- automatic ultrasonic methods. Automatic ultrasonic inspection has been developed in the labs for some time -- developing in the labs for some time. It is very important, if we're going to keep the radiation exposure to workers to a minimum. The difficulty of course is that it is a very sophisticated computer-based system. The movement in the field in the conditions typical of an industrial process is very difficult. We now have two years in that transition, and I think we have learned quite a lot about it.

So those are some of the mini-programs that developed part and parcel with the conduct of the larger program.

(Slide.)

I think in summary for my prepared remarks, I guess I would like to make the following points:

LOFT has been repeatedly placed into accident conditions, into conditions that characterize those accidents postulated for PWRs, a fairly wide spectrum. We are in fact planning, as you will hear later today, to extend the kinds of events that we look at into the multiple-failure type of issue.

In all of those cases, the plant has been successfully stabilized and recovered. The operators, the equipment, and the emergency systems have performed exceedingly well. That is a strong statement about the methodology by which the operators are selected, trained, qualified, and certified. The processes by which we teach them to anticipate failures occurring and arising while the experiment is in conduct; the performance of the plant equipment, I think is an important statement about the viability of the design codes and the margins implicit in those codes.

The emergency systems have in every case worked as they were expected to work -- and I think, again, there is a powerful statement of that experience.

We are looking at new instrumentation. We are looking at new operational methods. And we are trying to refine our analytical techniques in order to keep up with the changing kinds of experiments and the progressive change in severity of the events that we're studying.

And finally, I think we have shown that indeed there are some significant conservatisms in the assumptions and calculations used in the licensing process that the safety conclusions are underpinned by. At the same time, we are finding new perspectives every time we run a test. We have gained new perspectives about what is important and what is not important, and are finding that we still have a ways to go if we are going to characterize what actually happens if we're to use that characterization to build simulators and to train our operators.

Well, I think that is a conclusion of what I would like to say. The other speakers today will highlight some of the test series, and some of these other operator-assisting techniques.

DR. PLESSET: Well, thank you. Would you be willing to have us ask you some questions?

MR. KAUFMAN: Oh, certainly.

DR. PLESSET: Yes, Theo?

DR. THEOFANOUS: Just very briefly, what were the significant conservatisms that the data have shown?

MR. KAUFMAN: Well, I think the largest one was the one we reported to you at some length at the last meeting. Specifically, that there are hydraulic processes that we observed in LOFT that resulted in significant core flow in following a large-break loss-of-cooling event, which led in LOFT to some quenching of the fuel long before the ECC systems operated.

Our calculations which characterized the behavior in LOFT when applied to a PWR -- and we look at only one PWR -- show that that hydraulic phenomena should also occur in a large PWR. Therefore, I think that that is significant and important in saying that the techniques used in the licensing process for that class of events are conservative.

I think what we're also seeing is that, in the small break particularly, that the secondary cooling system was extremely effective over a fairly wide range of break sizes in controlling the pressure in the primary system; and that an operator can in effect take hold, or control the primary system pressure from the secondary when the plant is in single-phase as well as natural phase circulation conditions that we have studied so far. I think that that is important that we have assurance and confidence that we can handle that.

DR. PLESSET: Are there any comments -- Yes?

DR. ZUDANS: In your previous nuclear tests, I understand you already had your technical support center manned.

MR. KAUFMAN: Yes.

DR. ZUDANS: Have you come up with something that would be considered a good list of parameters to be displayed in that particular environment?

MR. KAUFMAN: Yes, we have come up with a list of parameters that we've found useful for small-break and for operational transients. I would not presume at this point to say that it is a sufficient set. We have identified some necessary parameters for those classes of accidents, and in fact we have found some things that are to be voided, particularly in terms of displaying correctly to the operator what is going on.

For example, as I think I mentioned to some of you the other day, the interpretation of limit switches is very important. What do I say about a parameter once it has reached its limit, once the limit switch is actuated? It raises a whole series of questions about the believability of the signal in the first place, and then what can I say about the process once that switch has been reached?

We are finding certain areas of frustration because we can't follow the event beyond the normal range of conditions.

DR. PLESSET: Go ahead.

DR. CATTON: Have you made any contribution to Reg Guide 197, or its ANS counterpart?

MR. KAUFMAN: We have Reg Guide 197 in review. We are reviewing that and will provide our comments, as others will. Whether that will make a contribution, I don't know; I am hopeful that it would, because in fact the movement into a sophisticated "following," if you will, of these kinds of events generally steers you in the direction of a computer. The introduction of computer technology into the safety process is a very difficult bridge to cross -- not only in terms of how you buffer and how you address the question of redundancy and diversity, but the question of how you control the configuration of the software, and in fact the configuration of the hardware.

Now we have done some limited research that way. We have had contracts with Georgia Tech to look at the application of some military concepts in terms of hardware hardening. We are trying to draw on their data base of software configuration control that we use to set up our data acquisition computers, and we will respond to 197 based on that experience.

DR. CATTON: This is going to be reviewed November 5th.

DR. ZUDANS: Yes. It is interesting that you really have an appropriate environment, because you create the accident and you expect to proceed in certain ways. And then certainly you follow with the information that you get in the control room and otherwise that accident.

Now have you come to some specific opinions with respect to 197, and particularly with respect to following the course-of-accident instrumentation?

MR. KAUFMAN: I would not like to comment on that, because I don't think that my comments at this point would be well enough thought out.

DR. ZUDANS: But you think you will be able to contribute to that?

MR. KAUFMAN: I think we can contribute not only in the initial review, but as we learn still more. I will give you an opinion, and the opinion is this: That 197 raises a class of issues and problems for which we really don't have a lot of the kind of experience base we really wish we did. I wish that we were another year ahead in our program of where we are today because I think we could then say something a lot more powerful than we can do today. I am not pleased with the degree that we'll be able to comment by the end of this month.

DR. ZUDANS: Let's forget 197. Let's just think one of your accidents. I am sure that you have a very strong opinion about what you really have to know.

MR. KAUFMAN: Yes. And it turns out that the parameters that we found are of value for following the accidents. The best source of those has been the operators. So we have been through the business of looking at the biases of engineers and their conceptualization and understanding of accidents, and find that -- and we think once you say it, it is obvious -- operators have a different conceptualization and characterization of an accident. And their process by which they put together various pieces of information to draw conclusions are different than an engineer or a safety analyst might come up with.

Our operators have been most precise in terms of -- have been the best able to anticipate what the conditions were, and have found a fair amount of frustration with the kinds of things engineers cooked up. We don't have time to go into it in great length, but we do have examples of displays that we have come up with our colorgraphics that were developed by engineers, that when the operators looked at it they frankly thought it was totally garbage, and it was absolutely unfit for them to follow an accident condition.

DR. ZUDANS: True.

300 7TH STREET, S.W., REPORTERS BUILDING, WASHINGTON, D.C. 20024 (202) 554-2345

MR. KAUFMAN: And they have modified it and come up with alternatives.

DR. ZUDANS: So what you are saying is that you don't design such things unless your operator is involved in it?

MR. KAUFMAN: You bet.

DR. ZUDANS: Good. Are you going to give us more about your color selections?

MR. KAUFMAN: Yes, Mr. Meyer will talk for about an hour sometime today.

DR. PLESSET: Gerry, you --

MR. RAY: I would just like to comment that your overview this morning and your visit the other day to your facility was very enlightening to me, to the effect that your facility is not just one that is designed to answer specific questions of accident scenarios, but is one of potential -- significant potential as a resource to development, and for development, to benefit the industry as a whole, and not just the regulatory process.

Now you mentioned your meeting last week. Was this a first with industry in general?

MR. KAUFMAN: The generalized meeting last week was the first of that size. Prior to that, we had contacted for example, NPOV(?) and asked them to send us some operationally oriented people to witness the test, and we had about a half-a-dozen.

On other times with specific utilities we've had a few contacts. Frankly, last week was an effort to try to get past some of the problems we've had before in trying to reach the utilities. We've had difficulty with it.

MR. RAY: Well, you're not restricted are you, in any way?

MR. KAUFMAN: No, we've not been restricted. The difficulty is crossing the barrier of interest to help the utilities recognize that in fact we have something that may be of value to them, and that in fact it isn't downside; that LOFT isn't a threat to an extend that we'll find something that will be bad news to them, that will lead to some new requirement or regulation or whatnot. We have to break down that feeling.

MR. RAY: I understand that situation. Has NSAC and NPO shown any interest to continue this narrative?

MR. KAUFMAN: NPO, I think our response there was very good. We've had limited contacts with NSAC. Our contacts with EPRI have been through LOFT review group meetings and participation in our meetings generally as observers. But in fact, to date it has been confined to limited task orders.

It is an area, again, where we are actively trying to establish a dialogue. Our program has evolved very rapidly. Our techniques to disseminate information, frankly, we're still developing. How do you get this kind of information about operator behavior and operator performance, how do you reach people with that? The traditional forms of technical societies, research information letters, quick-look reports, and so on, don't suffice because we're not reaching the right audience, and we're trying to learn.

MR. MATHIS: Nick, have you worked at all with the people that are trying to develop the nuclear data link and that system?

MR. KAUFMAN: We've had limited contact with those people. We've I think opened the door with the people at Sandia. On the other hand, I think we have a good deal to contribute. This is an issue we have raised in NRC Research, and have indicated very strongly that we felt we had something that we could contribute.

DR. PLESSET: Well, I also want to express my appreciation for your presentation, but you won't mind if I'm a little bit negative just for playing the devil's advocate.

I would like you to come here sometime and start out a meeting discussing LOFT in which the atypicalities and flaws in LOFT were listed first, before you talk about what a useful facility it is.

I think that to have an explicit presentation and discussion of the atypicalities and the flaws in the facility are very refreshing and helpful, because there are flaws. It is not a PWR, and it has a lot of important differences from the PWR that we're all aware of -- but you're more aware than we are, and it is up to you to tell us about them, and what you think you might be able to do about them. That is one comment that I would like to have you reflect on. You don't need to tell me your answer today.

MR. KAUFMAN: Just briefly, we have obviously thought quite a lot about our atypicalities and our problems. One of the things that we're seeing as a result of the program is that there is no unique list of atypicalities for a particular phenomena that we would like to make comment on, or a particular kind of transient condition and for a particular kind of conclusion, and there are a different set of atypicalities.

Indeed, I think certainly I have encouraged, and I think all of our managers, that in fact the evaluation of that is an essential part of making any conclusion about our test. One of the difficulties, however, to have a simple single list is -- for even a brief presentation is that indeed the atypicalities are virtually a function of the conclusions we're trying to draw and the kind of tests we're running.

DR. PLESSET: Well, that may very well be. I haven't seen them in your reports. I suspect that they may be there, but not as evident as some of the optimistic results we hear about.

I don't mean to imply that your results are useless, but they do have limits and bounds which it seems to me you have to emphasize if you're going to have an important influence in the technical community in this business. I'm sure you agree with that.

I take it that your answer is that these are presented and discussed?

MR. KAUFMAN: I believe they are. In fact, with our utility meeting, lest our utility folks feel that -- not be sucked into something that they don't fully appreciate, our first presentation at our utility fair was a review of the atypicalities of LOFT, and a comparison with PWRs, to highlight the degrees to which we were not a PWR -- lest the conclusions that we presented be inappropriately interpreted.

DR. PLESSET: All right, let's accept that then, and we'll move on.

MR. RAY: May I ask a question?

DR. PLESSET: No, I'm not quite finished yet. I was listening to your presentation, and I think that it has a lot of interesting value in your discussion about the Action Plan and what's required of the industry. But it seems to me that almost all of this is unrelated to the fact that you have a kind of limited nuclear facility, for I would say a large part of the program you discussed -- you know, control room design, and operator training. One could do a great deal of this and not have a nuclear facility. And if that is the case, you might say: Well, how can one justify the added complication of having a nuclear facility? It might be one that's electrically heated, or it might have a very elaborate simulator and computer, or even have little boys running around behind a panel with a lot of lights and cathode ray tubes and the like.

(Laughter.)

DR. PLESSET: Now that might be unfair, but there is an element in that that I would like you to reflect on.

Also, it seems to me that the Action Plan was directed not so much at research; it was directed into two different directions. One is the NRC itself, and the other was the industry.

Now if you try to play a role in this dual thrust of the Action Plan, I think you have to do it with considerable care and, again, ask yourself the question: Can my nuclear facility contribute to either one of these important aspects of the Action Plan?

I think the Action Plan is a great thing. It's going to cost the industry billions of dollars. If you can save them just one of those billion, you're home free it seems to me. Or if they're not going to listen to you, then forget it.

Now that might be a harsh way to describe the situation, but let me go on and express some more negative ideas.

If you have a role to play -- maybe it's with the Action Plan, with the industry, or with the NRC -- again it seems to me you have to specify very clearly what is the necessity for having this particular machine to do that? Can I do most of it with another device? Can I do most of it with an electrically heated system? Or do I need any kind of facility at all?

Development of control rooms does not require a reactor; it seems to me you can do that without a reactor. Operator training doesn't require a reactor for almost all of it. The question is: What is the responsibility of the NRC in subsidizing control room development and operator training? After all, these are functions of the industry.

If you are doing this, the industry should support it. There is nothing wrong with that. Actually it would be a very good idea if the industry would support your program, because it might be difficult to get this support from the NRC alone.

Now this is just a thought I had. Maybe some of our consultants or members here have different ideas, but I think that it is worth discussing these and not just keeping them buried in the back of your mind and not facing up to them, because these are realities.

MR. KAUFMAN: Can I address some of those?

DR. PLESSET: Yes.

MR. KAUFMAN: I think the first point that you raised is the value of the "nuclear-ness," if you will, of the facility. In my mind, the issue of nuclear-ness is much larger than just simply having fuel pins that have decay heat. It's the rigor, or the discipline that is forced upon us because we have an operable nuclear reactor, because we are required to select and train people to do analyses against, to have technical specifications against the rules and regulations that a nuclear facility must go through.

Now one can make the argument that: Well, I could have a non-nuclear facility, and I could require all of the same things of the organization or the operation associated with that non-nuclear facility. I could put them through the same drills in terms of limiting their operation to only those kinds of operations that they would do if the- were a nuclear plant.

But in fact, I think if you do you wind up with the same kind of cost, and the same kind of constraints that we fact at LOFT; and as a result, in the practical world we are all threatened with cost pressure and whatnot, and I think if we had a non-nuclear facility we would start off with the objective of trying to pretend that it was a nuclear facility and subject to the same constraints, and very quickly under the realities of budget pressure we would abandon that.

So when we run an accident condition, and when we recover, we operate in a set of constraints or trajectories that are characteristic of the kinds of things that a PWR has to confront. Our operators have -- to use a word I hate, but it's conventional -- a "mindset"; that they and our engineers have the same mindsets as people in PWRs. Because in fact it is a nuclear reactor.

So there is a value to that kind of activity.

Secondly --

DR. PLESSET: So that point is regarding operator training?

MR. KAUFMAN: Yes.

DR. PLESSET: Have you made an impact on the need of the industry for operators?

MR. KAUFMAN: I would like to make an impact on the need in the industry for operators, and in fact my concern -- and it relates to one of these points here -- is that we have not had a good data base by which the industry can go to NRC and say: I don't need those operators that I'm required to have, and I don't need them because in an accident situation these are the conditions that arise on a control console, these are the controls that need to be used, and therefore I don't need them.

I have a data base, then, or an experience base of responding. Now what we see in LOFT, for example, is that when we go into a small break experiment, normal redundancies in instrumentation no longer exist. You have instruments that typically read the same thing; once you're in an accident condition, they're reading much different things.

You get into the question of what is believable and what is useful. We are not seeing those kinds of disparities in our smaller facilities, because indeed you have to anticipate their existence to some extent in order to assure that they're present. In fact, I think you need an operating facility, an integral facility, you need at least one in the world someplace that shows what presumptions, what redundancies no longer exist.

And in fact, we have seen several. I think another kind of thing that we will hear when we look at the operational transients -- and in fact we have seen when we looked at the small breaks -- rather small, routine plant issues like the fact that a steam valve doesn't purely close, big valves don't absolutely close. Our models tended to assume that they did. Our computer models tend to exist in a very absolute world where things work or don't work, or they work to some precision. What we see in a plant is that, yes, we always have some leakage in the plant -- our plant, as well as commercial plants. We have steam valves that open and close with varying degrees of perfection.

One could say: Well, let's build a facility as large, and make it non-nuclear; but again I would submit the costs are about the same, if you're going to maintain it, if you're going to build against the same kinds of codes and standards used in the nuclear industry.

So we are finding what I think are significant perspectives about the value of presuming that our analytical techniques correctly characterize even the most benign transients that we typically train our operators to.

So again, I think we are getting perspectives that are unique. I think you will see that when you look at some of our operational transients data.

DR. PLESSET: Yes, Ivan.

DR. CATTON: I would just like to comment on the training aspect. I think you have an excellent facility for that. But I don't know how you're going to get the information into the hands of the utilities, or I'm not even sure the utilities care.

I think for the most part don't they believe that what they're doing themselves is the best way the business should be handled?

MR. KAUFMAN: Well, I can talk to discussions I've had with them. We have, for example, developed what we call "alternate action procedures and methods" that are patterned on the aerospace techniques. They are the techniques and methods that allow us to, with confidence, enter an accident situation. Obviously we're extremely vulnerable --

DR. CATTON: How is the fellow at San Onofre benefiting by that?

MR. KAUFMAN: Well, all right. What we're trying to do is to tell the fellow at San Onofre, there's a better way to write your procedures.

DR. CATTON: I guess I would like to hear how you are going to do that.

MR. KAUFMAN: Well, I will tell you three approaches that we've got.

One was to try to get the fellow from San Onofre here last week, and to give him a presentation on alternate action procedures.

Another is to go to the AIF forums and the ANS forums that are operationally oriented and give papers about what we do. We are trying to travel as much as our budgets will permit to these plants. In fact, we call the plants and offer to come to their plants and give them presentations on what we know. In fact, we have done that.

We have been, most recently, in fact two weeks ago we were in the plant in Sacramento. We have been to San Onofre in the last month.

MR. POINTER: They're there now.

MR. KAUFMAN: Now? Okay.

DR. CATTON: Well, but they --

MR. KAUFMAN: So we're trying to travel to the utility and then find -- there's a third point. That is, we're feeding everything we know, our insights, into the NRC. I hope that is another vehicle by which they are reached, or at least their existence is appreciated.

DR. CATTON: Well, there is a lot of paper between your feeding them information and it being fed to the operator level.

MR. KAUFMAN: Yes, and that is why we are going directly now to the operators.

MR. ETHERINGTON: What is the level of education and training of your operators?

MR. KAUFMAN: Our operators are all ex-Navy, ex-Nuclear Navy. They came to LOFT generally with experience on two plants with the Navy. They are generally -- well, all of our operators, I think, are non-degreed at this point.

MR. POINTER: No, we have a couple that are not technical degreed.

MR. KAUFMAN: We have two without technical degrees. Our shift supervisors, of those we have one non-degreed and three degreed -- three non-degreed and one degreed. We don't have a simulator, and so we had to look at techniques to get the people ready to do these unique things without simulators. So we had to develop techniques for dry runs in preparation.

MR. ETHERINGTON: Do these people have on-the-job training? Or do you have a training course for them?

MR. KAUFMAN: Yes. We put them through an 18-month certification and training program that is on the job. It is a combination of practical experience and theoretical knowledge.

DR. PLESSET: Well, I was just trying to think, since we are going to make out a research budget review in a couple of months, that we go and discuss with the lightwater safety research people, that: Well, you thought you had a facility that was studying thermal-hydraulics out there at LOFT, but it isn't that way. What they're doing is doing operator training.

And they will say back: Well, that's none of your business. Safety research is not involved in operator training. I heard a lot this morning about operator training. Not that that isn't a necessary thing, but the question that I still have hanging around is: How does that fit with the mission of safety research people, which has been spelled out in connection with studying thermal-hydraulics?

MR. KAUFMAN: We train operators because we have to operate the nuclear plant. But I think the techniques that we are learning, because those operators also operate a plant under accident conditions and do it very successfully, that that has value in the safety process. I think TMI said that safety has to be viewed in the whole concept of operations as well as hardware.

DR. PLESSET: But if the utilities don't see this, and value it, and support it, I think you're operating in an isolated environment.

MR. KAUFMAN: I think the utilities are responding to many items that came from Three Mile Island, and in fact those that we've talked to are deeply involved in doing that. I think we're talking about questions of the quality of those responses, and our ability to improve the safety of plants by better equipping the operators and the plant equipment -- I don't want to stress too much the operator business, because there's also the issue of how the sensors monitor what is going on in the plant, and how the operator responds to those. That is a quasi-hardware-operator issue.

So it isn't sufficient to know what's happening in the plant; it is how what is happening in the plant is being interpreted and characterized by the sensors. And then how the operator, through controls, again interfaces back into the plant. That is an integral loop and I think it is right at the heart of the safety issue, how well that can be done.

DR. PLESSET: Yes, Theo?

DR. THEOFANOUS: Yes. I would like to -- to start with, I would like to support the need that the Chairman of

Subcommittee expresses for discussing some of those matters in greater depth.

Along those lines, I would like to express my concern that we often come here -- sometimes once, sometimes twice a year. And every time, we get a long list of things to be done, and plans for the future, and then we get some discussion of some results, and some comparisons that are similar to the kind of thing we got yesterday, but we never seem to have come to grips with concrete, identifiable, very specific contributions that the facility has made up to that time.

Now the facility has been operating for some time now. It seems to me that it would be very essential, in view of the costs that are involved, to both look ahead in time and make comprehensive plans and try to minimize all this shifting around of targets and try to make the targets well defined. And then try to come back here one or two years later and say: Those were my targets at that time; that's what I promised you I was going to do; and that's what I have done, very concretely.

I go along with you, Nick, that -- and in some of your responses I believe, in fact I agree with you that indeed it is very difficult, impossible I think, to simulate the nuclear environment. I think that you have a completely different feedback from the people, the operations, everything if you try to make that artificial kind of situation. So I agree with you there, that I think the nuclear-ness of the LOFT is a very, very important and essential component.

I think it is one of the -- well, in fact the only one around, a facility that one can actually gain information probably right next to the real thing.

On the other hand, just because of those two reasons I don't think that one can conclude directly that we should be running it for the next 20 years. I think that we need to look very, very carefully every year, or every two or three years, at what are the potential contributions LOFT can make, and in what specific way. And then, to come back after two or three years later and show how the contribution or how the problem was met.

MR. KAUFMAN: I understand your point. I think there is a point to be made that the understanding of the industry of itself, and particularly the settling out and the identification of the issues that are left after the turmoil of TMI.

LOFT has been in operation for a year-and-a-half. During that period, TMI happened four months after we went into operation. The industry has been in a great state of turmoil, and we have tried to move the program very rapidly to try to stay with the change in the industry.

Prior to the disruption of Three Mile Island, the facility was quite well focused in terms of an issue it was going to address. It did address that issue, and it addressed it, I think, on schedule and on time. And we talked the last time you were here at some length about just that, and those things that we had learned, and those questions that exist.

But since then we have been in the process of changing radically, because I think in fact the NRC itself, I think the industry in its recognition of itself has changed rapidly, too. Nothing would please me more than to reach a point of some better stability. I've changed my budget schedules five times this year.

DR. PLESSET: Yes?

DR. CATTON: I think the uses of LOFT as a training center is encouraging, but it seems to me that you ought to seek industrial support in its most recognized form, which is "money."

What is your reaction to other operators being in your control room operating your controls under the guidance of your trained people?

MR. KAUFMAN: I think you have really raised two issues. One is the likelihood that the industry will provide money --

DR. CATTON: Because that is an indication of whether they have interest.

MR. KAUFMAN: Well, I don't think it is.

300 7TH STREET, S.W., REPORTERS BUILDING, WASHINGTON, D.C. 20024 (202) 554-2345

DR. CATTON: They want something free?

MR. KAUFMAN: That's one piece of it. But I think their association and intermixing in a program that is partly funded by NRC and partly not is perhaps another issue.

I think whether the focus of the plant should be in safety issues or to economic issues might be another factor. I don't think that it's likely that industry will support LOFT because I think, on balance, it's a downside risk. It would be a downside risk to them, and one that they wouldn't fund.

DR. CATTON: It's the only place I know of where you can put an operator where he can experience a small break, or a large break. From that point of view, I think it is an excellent facility.

MR. KAUFMAN: That's absolutely true.

DR. CATTON: But if industry won't support it --

MR. KAUFMAN: I don't think that's necessarily so. I think that's the reason that we have research sponsored by NRC and DOE, because there is a certain class of research, there's a certain degree of "turning over the rocks," if you will, that I don't think it's reasonable to expect the utilities to do.

Now let me answer your second question. The second question is about utility people using our controls. What we -- We require 18 months of training to certify an operator to operate on our console. We require that, even though he has an extensive nuclear background, because we are extremely vulnerable to errors and malfunctions when we've got a plant placed in a severe accident condition. So we require a great deal of preparation and training.

I think that we would welcome, and in fact I have made the offer to the industry, that: If you can send your people for a long enough time that I can train them against our standards and criteria, I would do that.

On the other hand, bringing in a crew for two or three months just for a test and expect them to conduct it safely, I don't think we can do that. But in thinking about the issue and our possible value -- because I think that's an area where we've got some value -- we have proposed to build the control room of the '90s; take all of the best that people talk about, and build a control room of that sort. We have proposed this to DOE.

We would then equip that control room not only to run the reactor, but with sufficient computer capability to play back through all of the sensors the totality of the accident. Then we would run it, and we can play it back, and we can, with the computers, I think provide enough flexibility to allow them to make a few modifications to errors. Then they could do it.

Our concept of it would have monitoring not only of exactly what they do, but the way in which they did it, and then compare it against the norm of our people, with a given degree of training, doing it in a certain way.

Now that we have looked at, and I think that's the viable program. I think it is a program that potentially could be in operation in three years, if the funding is there.

DR. PLESSET: Well, I think that I am a little bit confused in this sense: LOFT represents about a fourth of the total research budget, and a lot of what I've heard seems to be that it's a mistake that it's part of the research budget. I think that Vic Stello, and Inspection and Enforcement should take it over, and he might be able to afford it. But that's just a bigger thought that I'll put out, because from the point of view of research, which is where their LOFT is now located, these things are laudable, worthwhile.

I was making a little bit of an estimate of what it would cost you to train operators for the industry. It might be on the order of a million dollars per operator, but maybe it is worth it.

Anyway, it seemed to me that this is getting a little remote from lightwater reactor research, which is in this branch, or even NRR or its branch, and getting away from that. And if the utilities won't take it over, and EPRI won't take it over, and Vic Stello won't take it over, we are in a difficult situation -- not "we," but "you."

So I think we can continue this indefinitely, and maybe not very profitably, and I think we've got to watch our schedule. I hope you don't mind that there is a bit of a devil's advocate running around. I think it has been well run, well managed, and the question is: What is the mission? And how are we going to get it continued?

That's why we brought up all of these critical comments which don't necessarily mean that we're unfriendly.

MR. KAUFMAN: I understand that, and in fact it is in the interplay of different ideas that I think we will be stronger.

I guess what I would like to do is now acquaint you with some other aspects of our program, and perhaps in more detail, and perhaps you will change your mind.

DR. PLESSET: Maybe. That's good.

MR. KAUFMAN: Dr. Charles Solbrig.

(Pause.)

DR. PLESSET: Dr. Solbrig is also an old friend of this subcommittee.

DR. SOLBRIG: Thank you.

(Slide.)

My first slide tells you who I am, again. The subject matter which I will talk about today is the results of our anticipated transients. I think a lot of the questions which you have asked in Nick's talk will be touched upon briefly in this talk. I would like to offer that any of these topics or any of these questions that you have, we would be more than happy to make presentations on. Last week we did touch on a lot of these items in our presentation to the LOFT Utility Technology Transfer Conference, but due to the limited time of this, and I guess the limited subject matter of this ACRS Subcommittee meeting, we really were not able to include that information, but we would really be happy to work out a schedule with you -- and even if you would like some more of this information presented today, we would be prepared to do that.

(Slide.)

We have performed four anticipated transient experiments thus far: Loss of load; loss of flow; excessive load increase; and the loss of feedwater.

These experiments can be performed quickly in the facility and in fact we performed three of these experiments in one week. They are not severe transients and do not require a tremendous amount of operation time to perform. As a matter of fact, the end-state of each of these transients is a hot standby condition.

So instead of going to -- attempting to go to a cold shutdown condition, you can approach instead a hot standby condition and you're in good shape to go back to power.

Each of these experiments was successful.

(Slide.)

The topics which I will discuss will include why I think it is important that LOFT perform anticipated transients; and what the results are from these experiments.

(Slide.)

The need for anticipated transients -- "experiments" in LOFT -- that should be "experiments"; we don't need any anticipated "transients" -- is to provide a basis for our anticipated transient with multiple-failure series.

These tests that we have performed are in fact non-trivial. We have seen several places in which the predictions could have been better with the codes, and we used the RETRAN computer code which was developed for utility use particularly for describing anticipated transients and operational transients.

The adequacy of most safety analyses models that are included in, for example, Chapter 15.1 or .2, have not been verified out. Often these models are of a simple nature and do not take into account complexities which can occur in such a transient.

I think another very important use of anticipated transients is in providing information for simulators -- and I will talk about this a little bit more. Simulators are in fact also a computer code, a simplified computer code, and they do not handle anything but the set, particular transients which have been previously included in the simulators.

Anticipated transients are also very interesting because of the fact that they are high-probability events and are expected to occur once per year.

(Slide.)

TMI requirements for simulators has really increased the need for the capability of the simulator. They're going to have to, I think, drive these simulators in the future with digital computer codes on the order of RETRAN or RELAP complexity.

The RELAP or RETRAN computer codes, however, really do not represent all of the aspects of the nuclear plant as yet, and so the anticipated transients will help us to determine which aspects they do not represent.

Some of the things that they don't represent are secondary side models, pressurizer heaters and sprays, adequately. These sort of things can be improved.

I believe that these codes must be able to represent in the future anticipated transients, anticipated transients with multiple failures, large breaks, small breaks, and all transients in between.

(Slide.)

Now a lot of these issues are in fact heavily related to the regulatory process, and we think we are contributing to the regulatory process. Certainly answers about anticipated transients and the course in which they will proceed are important regulatory processes.

There are a lot of questions that are asked in operations that are not answerable with the current simulators. For example, talk about justifying tech spec changes. Yankee Atomic and VEPCO have been using heavily the RETRAN computer code for justifying changes in their technical specifications. Also, they have been using it heavily for justifying changes in their operation procedures.

Therefore, the verification of the codes are very important.

An example of a procedure which one might want to verify is on proceeding from hot standby to cold shutdown in the plant. At some point in the operation and the procedure you must valve out the accumulators to prohibit them from injecting into the system. The normal procedure that was done in the plant that I'm familiar with was to valve them out at 1000 psi. Now the question is: Is this really a reasonable pressure to valve them out at? Is there any problem in doing that? The 1000 psi was arrived at just by good judgment; there really is no analytical data for why it should be done in that particular pressure.

The training programs -- the question that Dr. Catton asked. The TMI requirements for the increased operator training and technical advisor training, I'm aware that at the plants where they require this training, they really don't know what to include in any training. They know that they have to train for certain procedures for certain actions that they have to accomplish, but they really don't know how the plant is going to respond.

So when I was at Commonwealth Edison, we were looking at trying to back up this training and provide a tool to answer these questions. They have to go through a several-month training process. Technical advisors have to go through a several-month training process with the material that they have in those courses that they are uncertain about.

I think that this is heavily a licensing issue.

DR. ZUDANS: Could I ask just one question?

DR. SOLBRIG: Yes.

DR. ZUDANS: You say a "verification of current procedures."

DR. SOLBRIG: Yes.

DR. ZUDANS: I knew of the atypicalities that Chairman Plesset raised a question about. Do you feel confident that you can render a useful service to PWR?

DR. SOLBRIG: Well, when you take a look at the procedures that are used today and how they are verified, they are verified -- and I'm sure you're familiar with the types of models that are included in, for example, Chapter 15.1 of the SAR. These analyses are very simplistic. They are specifically designed for that particular transient in the way in which they proceed.

Now in our overall program, the method that we use for providing the information to plants is through verified codes. Now of course we have many atypicalities, and the question is: Are we in fact looking at all of the important phenomenon in LOFT to make certain that we have verified the aspects of the code which will be used in developing the procedures.

There are some areas in which we cannot aid. For example, the containment -- when you talk about containment coolers, containment fans, how the operator will be interacting during a transient with these pieces of equipment, because we have the pressure suppression pool -- we can't answer those questions. But there are many areas that we can.

It is my feeling that something on the order of 90 percent of the questions on procedures can be verified with the LOFT facility.

DR. ZUDANS: But then I gather that the root of your answer is via application of a code that's validated in LOFT, in essence?

DR. SOLBRIG: That's true; that's correct.

DR. ZUDANS: And of course within the limitations of the LOFT capability to validate that code.

DR. SOLBRIG: That's correct. That's correct.

DR. CATTON: Chuck?

DR. SOLBRIG: Yes, sir.

DR. CATTON: You mentioned RETRAN several times in simulators --

DR. SOLBRIG: Yes.

DR. CATTON: -- and as far as I can tell, simulators are used mostly for operational type training and this view of the transient. I know that RETRAN presently is being validated against several different reactors through the low-power testing program. What is LOFT going to contribute that's more, that's needed? It's a small percentage of need that is left.

DR. SOLBRIG: Well, the operational transients which are included in the plant startup tests --

DR. CATTON: No, I'm referring to the extended tests.

DR. SOLBRIG: -- the extended tests, are in fact not very severe, and they do not excercise many of the options of the code. For example, the anticipated transients with multiple failures has been looked at once in one of the reactors on an unanticipated basis, and in general they are not scheduled to look at anything except straight operational issues in those transients.

So I don't think it's a small delta, but I think it's a large delta. They can only provide a very limited amount of testing; whereas, LOFT can encounter almost any condition that we want to and answer any -- look into any area of operation, and we really do not have particular problems in doing that with our reactor. So I think that we can look at a lot more areas to verify the computer code.

DR. CATTON: Is EPRI going to give a preprediction of your L2-3 test, or L6, whatever it is, the next test coming up?

DR. SOLBRIG: The L3-5 and --

DR. CATTON: The next test that's coming up.

DR. SOLBRIG: The L3-5?

DR. CATTON: The L3-6. Is EPRI going to make a preprediction, or get involved in that game?

DR. SOLBRIG: No.

DR. CATTON: They're the closest ones to the utilities as far as favorable codes.

DR. SOLBRIG: You mean in terms of RETRAN?

DR. CATTON: In terms of RETRAN, yes. RETRAN is used all over the place.

DR. SOLBRIG: It is. It's used quite heavily. I would suggest that anticipated transients be included in the standard problem program. We use RETRAN just for the very particular reason that it is so heavily used in the industry. We are intending to use both RELAP and RETRAN in our future work. We have both operating, and we have capabilities with both computer codes. They are in fact very similar in their objectives and their end point that they're aiming at I think is somewhat similar. But during the course of development with these codes, I think we need both on board and are using both.

DR. CATTON: What I'm asking is: Is EPRI on board with you guys?

DR. SOLBRIG: I don't really know how to answer that.

DR. CATTON: I mean, they have distribution systems with their code, and anything they do with it is immediately throughout the industry. And if they're not involved --

DR. SOLBRIG: Okay. Excuse me. They know what we're doing. They have allowed us to use an unreleased version of their code -- for example, RETRAN2 -- and we're up to speed on that. Energy, Incorporated, has been working for us in these predictions. So in that sense --

DR. CATTON: Okay.

DR. ZUDANS: I would just like to add a point to this comment. Now if the actual power plant uses some sequence of an accident to validate the problem, are they adequately instrumented really to account for all the things that happen, as compared to LOFT which is highly instrumented?

DR. CATTON: Probably not.

DR. ZUDANS: So that means that, from that point of view where you can't really replace the need for LOFT.

DR. CATTON: Well, in part that is right. But from what I understand, EPRI actually developed an instrumentation package that they carried into the plants. And the real problem was not insufficient instrumentation, but the problem was the method of reporting and the inaccuracies, and they've corrected that by carrying their own system in.

DR. SOLBRIG: There is a problem with the amount of data that's taken, the amount of instruments that are used. For example, they don't put gamma densitometers on the system, and if you're looking at a cold-water accident where you can get boiling around the system, I think gamma densitometers will be very important.

So I appreciate the point on the instrumentation. With regard to the recording, they do have available advanced recording systems which are very equivalent to LOFT; but on most plants, this is not a standard equipment that's kept in the plant. They are used primarily for startup testing and then the computer goes back to the vendor --

DR. CATTON: That's right.

DR. SOLBRIG: -- to GE, or somebody like that, who would own the computer.

(Slide.)

To give you another example of the diesel generator loading test in, for example, PWRs such as Zion, on a periodic basis they are required to do an operational check on the diesel generator to make sure that they can pick up the load on all of the ECC systems.

Now you don't want to be injecting borated water into the system during this test, because obviously you would change the chemistry.

They would like to do this test during a hot standby condition, when they are in the process of shutting down for refueling or something like that; that the best time for them to perform this experiment would be under a hot standby condition.

The high pressure injection system has to be valved out for a certain period of time while they stroke the valves in the system. It has to be blocked so it doesn't inject into the system. So you have the system on your high pressure conditions, hot standby conditions, and you valve out the HPI.

The main purpose for the HPI is to predict against small breaks. So the question is: How long a time period would you have to have in order to re-enable the HPI if you did have a small break during that fact. And this was a question that I looked at in the past, and this can be answered by computer codes such as RETRAN or RELAP. It's been verified. But it can't be answered by a simulator.

(Slide.)

LOFT I think is uniquely suited to perform these types of experiments because it does have many of the systems that a regular plant has. It has multiple ECC trains; it has secondary side components; it represents such things as auxiliary feedwater. For example, in Semiscale you might be able to put in the right flow rate representing the change from main feedwater to auxiliary feedwater, but you wouldn't have the right temperature, because the main feedwater goes through the preheaters, of course, and comes in at a much higher temperature than the auxiliary feedwater.

LOFT is also designed for single-failure components so we have double parallel lines representative of many of the areas of the large PWR. Small systems have large heat losses, which we talked about yesterday.

The point that we just mentioned before about not reporting enough information. We think that LOFT can provide realistic experiments. We are really the one that's looking at not doing conservative experiments, the interaction of the fuel with the primary coolant as a realistic interaction.

(Slide.)

The objectives that we have in performing our anticipated transients were to increase our understanding of the phenomena which could occur in such transients; threshold phenomenon; to look at the augmented operator program; the tech support center program; to look at the engineered safety features, the plant control systems; and to provide data for a code assessment.

(Slide.)

The characteristics of these transients were that the coolant inventory was initially increasing or constant.

A main characteristic of the performance of a transient is that the primary coolant system energy balance is the most important aspect, and it controls the primary system pressure as well as the pressurizer level.

(Slide.)

The secondary and primary initiating events were investigated within these transients. We performed these transients in such a way that they provided a minimum impact on our overall loss schedule.

(Slide.)

The scaling or atypicality for these four experiments were not planned for in the experiments for an anticipated transient. We are really equivalent to a four-loop plant, whereas in a LOCA we could be comparing it with three loops of a large plant. So the typicality of the scaling is not nearly as good in the anticipated transients as it would be in the LOCA experiments that we have done.

As I mentioned before, each experiment was predicted with the RETRAN code.

Now the first experiment that we performed was the loss-of-steam load. This next slide shows the pressure comparison on the primary system between the calculations and the experiments.

(Slide.)

I am showing you results for 200 seconds. That is as long as we made the calculations. The experiment, however, did continue on to record data until we reached hot standby condition. We see in this experiment that the experimental conditions were different than the predictions when the spray was turned on in the pressurizer. In the experiment, we got a decrease in the pressure; whereas in the prediction it continued on up.

This means that the heat transfer characteristics of the spray, the condensation characteristics of the spray in the pressurizer are not adequately modeled.

We also noticed that when the spray was turned off, the code shows something, whereas the experiment doesn't show any effect of the spray being turned off. At this point (indicating), the heaters were turned on. The heaters in the experiment were able to keep the pressure up at this point in time. In the calculations, the heaters were turned on at this point (indicating), and they were not able to keep up with the pressure. So again you have a difference in the characteristics of the real heaters as opposed to the RETRAN heaters.

The experiment showed that we had more heat deposited in the primary because of a later scram time in the experiment than in the calculations, and you had more heat retained in the primary, and the main steam control valve had to open up in the experiment; whereas, this was not looked at or observed in the calculation.

(Slide.)

The next slide just summarizes those three inadequacies that we noticed in this experiment.

(Slide.)

The next experiment we performed was basically the stopping of a primary coolant system. We were looking again at movement toward single-phase natural circulation in the system. Scram occurred in this experiment at 2 seconds. This is followed almost immediately by an automatic trip of feedwater at about 3 seconds. The steam valve was completely closed at about 14 seconds, and the flow in the primary system, due to the pump or pumps are basically stopped by about 17 seconds into the transient.

Now this difference during the first 80 seconds is due to initial conditions. When we reran the code with the exact initial conditions of the experiment, we had very close agreement out to about 80 seconds. However, from this point on, we did get disagreement. We feel that this is due to the nonequilibrium model that is in the RETRAN code. Basically it assumes that the pressurizer is constant -- the vapor space is adiabatic and a constant mass system. So you get an adiabatic compression and apparently we should account for some heat transfer to the walls between the steam and the liquid. So basically this experiment showed --

(Slide.)

-- that nonequilibrium models in the pressurizer needs to be improved.

DR. THEOFANOUS: How many nodes in the pressurizer did you have in the calculation?

DR. SOLBRIG: RETRAN uses the non-equilibrium pressurizer model, which means that you can only use one node. They have a special model to treat the pressurizer.

(Slide.)

Now the next experiment was an excessive load increase. Here you see quite a large divergence here between the experiment and the predictions. This is primarily due to the fact that the code did not pick up the scram signal. In the experiment, the pressure decreased much faster in the experiment than in the calculation. The heaters were turned on at this point (indicating) in the transient, as well as in the calculation; however, the pressure didn't proceed on down much faster, and we encountered a scram.

Basically the experiment was over at this point in time. Now the reasons for this disagreement -- oh, I wanted to make one other point on this slide.

Basically the heaters were turned on at this point (indicating), and the pressure leveled out. The high pressure injection system was not turned on until this point (indicating). So this constant level of the pressure here (indicating) was turned around just by the heaters. Then the rapid increase in the pressure was due to the turning on of the high pressure injection system.

Now although the operators were not planned to intercede in this experiment during the first 200 seconds, they did at this point in time. They turned the high pressure injection system off, because you can see the rapid increase in the pressure. So they interceded in this case, rather than having a case like Crystal River where we would continue putting in high pressure injection and opening the PORV.

So basically we really ended the experiment at this point because of the large disagreement and the fact that we had a scram in the experiment.

DR. ZUDANS: Is this RETRAN calculation pre-test mode? Or post-test mode?

DR. SOLBRIG: These are all in the pre-test mode.

DR. ZUDANS: If you would repeat it without changing the code in a post-test mode with better knowledge of your initial conditions, would that be any better?

DR. SOLBRIG: No. I'll show you why it won't. One point to know here is that we have both low-pressure and high-pressure scrams. The code kept you right in between the scram set points, and it's very difficult to get into -- excuse me, it's got a high collar and a low pressure scram, and it's very difficult, I guess, to keep the reactor operating under those conditions because it's very easy to scram.

It is a very delicate situation. You know, here again training, if you're trying to train the operator or the technical advisor and tell him what's going to happen in such a transient, are you going to prepare him for a scram, or a much different set of events?

(Slide.)

The next slide shows the feedwater. Now the feedwater input -- the feedwater on the plant is controlled either by a single input or three inputs. The operators can select as to how they automatically control -- the feedwater will control.

Now in LOFT we were controlling on both the water level in the downcomer region in the steam generator, as well as the steam flow rate. Obviously, the coefficients that we put into the code to represent what was going to happen were not very good. The experiment shows an increase in the feedwater through the control system, whereas the calculation at the initiation of the experiment shows a decrease. And finally at this point when the scram occurs, the feedwater is shut off automatically and we see the complete opposite nature of the transient.

Now we redid the calculation with this feedwater flow rate and it improved things somewhat, but it didn't completely.

(Slide.)

The next slide shows that the steam flow rate that we calculated was lower than in the actual experiment. We're going back, having done this calculation, and putting in the specified steaming rate out of the steam generator.

I would like to point out that these three experiments were just done two weeks ago, and the state of our analysis is still progressing. So I can't answer all your questions.

DR. ZUDANS: It is interesting that clearly at this time you maximize the differences. You show how different the code would be from the test. So that in a way indicates that there is a lot more to do with the code.

What does it really mean in terms of everybody using that code, if it is such a poor predictor?

DR. CATTON: It's a worrisome thing.

(Laughter.)

DR. SOLBRIG: Okay. I don't want to say that the code is not good --

DR. CATTON: Well, it's obvious.

DR. SOLBRIG: -- for predicting LOCAs. In these type of transients, we're talking about system changes that don't matter at all in LOCAs. If you've ever been in a simulator and observed a LOCA going on in the simulator, the operators just stand there and watch it. It's over in a minute.

But these types of transients, there's a lot of opportunity for operator interaction. So we're talking about -- just like Nick mentioned, the steam control valve. We have a control valve on the secondary side of our steam generator which you could say is equivalent to, or corresponds to turbine bypass valves and the atmospheric dump valves in a power plant.

Now the question could be asked to the utilities: How much do those valves leak, these turbine bypass valves? They can't answer that question because they don't have any instruments to measure that. However, they will tell you that you can hear the flow through those valves when the turbine stop valves stop the steam to the turbine.

It's an unquantifiable thing. It's somewhat of an erratic thing. When these valves close, you know, you're going to get a different amount of leakage each time The

LOFT system behaves in the same way. We have taken great care to try and reduce that leakage to zero, but we haven't been able to. I think we've done as good a job as you can on that type of a valve, and I'm not sure that we want to reduce it to zero because all we do then is make it equivalent to the code, as opposed to representative of an actual operating situation.

DR. CATTON: Has EPRI seen these results which you're showing us?

DR. SOLBRIG: Well, these experiments were done three weeks ago and were presented at the Utility meeting last week. The quick-look report is just going out today. So the answer is "no."

DR. CATTON: It was my understanding that one of the uses of RETRAN was for study of transients.

DR. SOLBRIG: That's correct.

DR. CATTON: And, gee, if it's this bad and the utilities are all -- or a lot of the utilities have this code available to them through EPRI --

DR. SOLBRIG: Now in trying to predict what is going to happen into the experiment, you not only have the computer code to deal with, you have input to deal with. I think that this is one particular area where we could stand some more documentation, is in the inputs to computer codes. These things are related to the inputs. This particular item for example on the feedwater control, if we had a better representation of the coefficients which went in to represent the feedwater control valves, it's really not a code problem. Some of these are code problems, such as the non-equilibrium pressurizer model, but the input -- especially for these operational anticipated transients -- are very important.

DR. THEOFANOUS: I think it is true to say that RETRAN was expressly put together for analyzing transients, because I know what the people in fact, even the consulting firms, in fact are using it for this purpose.

DR. SOLBRIG: Yes.

DR. THEOFANOUS: And I also think it is fair to say that RETRAN is really nothing else but a LOCA code put together rather hastily with rather limited data or information as far as the applicability of the different models to transients. I think that is what we are seeing here. You just commented a minute ago that the pressurizer doesn't work and it's not too surprising if you're going to have one node and you're going to start running the heaters at one point, and you won't be able to predict depressurization.

Now how important, however, is it? How accurately do you need to predict it? That's a separate question.

So I don't think we should be surprised. What I'm saying is, we shouldn't be surprised by the differences; in fact, you have anticipated them. Again, it bothers me to see that we are kind of looking at those results and saying something else again to explain away. I don't feel that way, because that should be expected in the first place.

DR. SOLBRIG: I don't think it's a serious problem, but it is a problem that has to be recognized when these computer codes are used for operator training, and for establishing procedures.

DR. THEOFANOUS: But it could have been recognized before that; and you could have presented this information with a little introduction saying that I expected a lot of differences, and here they are, and I can use those to tune things all the better, then I think I would agree all the more with that perspective.

DR. SOLBRIG: Yes, I expected a lot of difference.

(Laughter.)

(Slide.)

DR. WU: Pardon me. Did I understand correctly that those are pre-test calculations using RETRAN?

DR. SOLBRIG: Yes, sir.

DR. WU: So apparently the inputs can differ from that. Suppose you improve those inputs for the reuse of RETRAN for the post-test? How much of a difference would that be?

DR. SOLBRIG: For example, on this last experiment on the L6-3 experiment, we should be able to improve that quite a bit, because we think it is only related to the feedwater flow and the steam flow rate, and so we should be able to improve those models.

However, for the non-equilibrium pressure model, that is not an input and it is something that would have to be improved in the code itself. So there are both contributions and experiments from both aspects.

DR. WU: Chuck, a follow up. Suppose the aim objective is to use some of the verified codes, or well-assessed codes for operator training? What is the margin, then, plus and minus of what would be left for the operator to react within that few minutes of time?

DR. SOLBRIG: Okay. In this particular transient such as L6-3, I think we would first of all have to find out how sensitive this feedwater control is to the model for the feedwater control. We didn't do that before the transient. It's my own feeling that probably the feedwater control valve, that this is the type of accuracy that we have in trying to represent it. You know, it's not an exact control.

So I think that you have to look at sensitivity and provide the operator with alternative courses of action. You know, if it in fact does scram instead not not scramming, this is what you will do. So you could in fact have three course of action mapped out here as a result of a sensitivity study.

DR. ZUDANS: You didn't analyze any of these transients with a code such as RELAP?

DR. SOLBRIG: RETRAN is very, very similar to RELAP4/MOD7. We didn't do that because RELAP4/MOD7 is not set up for transients. There is additional capability that's in RETRAN, although the basic structure is the same. For example, control systems are included in the RETRAN program in a very general sense. You can turn valves on and off, whereas in RELAP you are very limited in the number of times you can do this.

DR. ZUDANS: And now the actual question, the important one: What is the next step? Now you have found that the actual data in the transient departs so greatly, what is the next step in your thinking in the process? Where are we going from this point?

DR. SOLBRIG: Our next step is to go back and determine in fact that these particular issues aren't the cause for differences. In the L6-3 experiment we will go back and make certain that we can post-calculate what in fact happened.

We will, on the non-equilibrium pressurizer model, transmit that information to EPRI. They already know it, because Energy, Incorporated, was involved in this work, and we'll ask them for an improvement in that model.

Then we are intending to use this code for a prediction of our L6-7 experiment, which is basically a turbine-trip experiment with multiple failures. And we will use that to predict that experiment, as well as our L9 series, which is a series devoted entirely to anticipated transients with multiple failures.

DR. ZUDANS: So what you are really doing, you are helping industry to fix up its codes, and you are not being paid for it.

DR. SOLBRIG: Well, you can interpret it that way; however, Licensing is intimately involved in the training process, and they have to approve the training process. How do they know that the results of the training sessions are in fact valid?

DR. ZUDANS: So if Licensing is involved in this process, then it creates a conflict, kind of. Because if the industry goes back and uses the information that you've generated with Licensing agency's funding, then they will already have a strong argument in saying: Hey, we did what you asked us to do. Here it is. We accept it.

DR. PLESSET: Well, I think we are eating into Chuck's presentation time. We are running out of time for you, Chuck. You're aware, so these are general questions.

(Slide.)

DR. SOLBRIG: Yes. The L6-5 experiment was developed two months ago. It's a loss-of-feedwater experiment. The major thing to note here is the steam valve on the secondary side was probably important and contributed to the difference between the prediction and the experiment.

In the calculation, the main steam valve did open to relieve pressure on the secondary side; whereas in the actual experiment, we think there was sufficient leakage so that the pressure didn't build up on the secondary side.

Also in the calculations we had too high a decay heat--

(Slide.)

-- and we used the input of the radiation time rather than the actual 20 hours in the experiment, so that was also responsible for some of the differences. So those two aspects are responsible for the differences in that experiment.

(Slide.)

The long-term behavior of the experiment is shown here. As I said, the calculation time proceeded down to 200 seconds.

(Slide.)

The objective of the experiment was to bring the facility to a hot standby condition. In anticipated transients, you want to bring the facility to a hot standby condition instead of a cold shutdown condition, as you would want to do in a LOCA. You would want to bring it to a cold shutdown condition in a small -- any time when you're losing system mass.

This illustrates that in fact the operators were able to control the plant and bring it to a safe, stable state. Here the pressurizer sprays were used. The ring steam control valve opened and closed automatically. The pressurizers were used; the charging pumps were used; and in fact we brought the system to a safe shutdown condition.

(Slide.)

The conclusions of the work so far as that we feel like we were able to perform these experiments -- although they are not severe on the system, we were able to conduct these in a reasonable amount of time, a short time.

We think that current models in SARs should be looked at to make certain that they are in fact realistic, and compared with an advanced computer code such as the RETRAN code after it's been calibrated.

RETRAN was able to predict the trends and events in the transients in general, although we did run into a problem with the scram -- a very critical issue there, and a very sensitive calculation that was needed in that area.

Several areas have been determined for improvement. The engineering safety features in the plant protection system and the operator action were effective in bringing the plant to a hot standby condition.

In the future, we will be performing more anticipated transients with multiple failures. The Arkansas Nuclear transient will be simulated in our L6-7 experiment. This experiment was basically a turbine trip with two failures. The spray in the pressurizer stayed on, and the atmospheric dump valve remained open, which led into a coldwater transient or a power excursion due to that. The operators in that specific case were able to control. In the LOFT experiment, we will go in from that point into a more severe cooldown experiment.

DR. PLESSET: Well, thank you, Chuck. That is very interesting, and I think it should be useful and very helpful. I appreciate it, and I am sure all of the --

MR. ETHERINGTON: Could I ask one quickie?

DR. PLESSET: Yes, sir. Go ahead.

MR. ETHERINGTON: There should be some qualitative explanation of why RETRAN called for shutting off the feedwater on a large increase of load. Is there any?

DR. SOLBRIG: Well, we haven't looked into that sufficiently, yet, but obviously the coefficients -- there are three inputs to controlling the feedwater and it's a playoff between the three inputs, and obviously we didn't have the coefficients correct. It was related to the water level in the downcomer of the steam generator, and the steam flow rate, and we obviously did not have that modeled correctly. But we will look into that.

DR. CATTON: Harold, it's the "fiddler concept."

MR. ETHERINGTON: What?

DR. CATTON: It's the "fiddler concept." No matter how good the fiddle, if it's a poor player you don't get good music.

(Laughter.)

DR. WU: May I ask a question?

MR. ETHERINGTON: Yes, I won't waste time on that.

DR. WU: Have you finished?

MR. ETHERINGTON: Yes.

DR. WU: I appreciate your presentation and seeing some of the differences between the data and the prediction by RETRAN. Do you think your conclusion number three, that RETRAN can predict the trend into the events a little too generous and lenient, especially like feedwater? They go just opposite in trend.

DR. SOLBRIG: Well, I think there is much cause for optimism. I think we can clearly see what the problems are. Perhaps I should say that at some small amount of time in the future we will be able to reach that objective.

DR. PLESSET: There is a short comment from the back, I take it. Is it short?

MR. RICHERT: Kent Richert from Energy, Incorporated. The success that utilities have found with RETRAN is the fact that there is lots of plant data which can be used iteratively to refine the input. I would propose that the results of the last couple of weeks have shown that there has not been sufficient time for the people using the code who are still learning how to use the code; that this is a big difference between RETRAN and RELAP. RETRAN covers a broad spectrum of accidents where there is a lot of data which can be used to refine the input.

And as the utilities have found, particularly with TMI-2, once you have refined a code and have confidence in it, then it can be immediately applied to calculating, for example, how to operate the plant when it came time to shut off the pumps -- this was after the accident, during the recovery process.

I think that the experience in LOFT has yet to reach this state of refinement which the utilities have already reached with the data they have.

DR. PLESSET: Okay. Thank you.

DR. CATTON: Could I ask the gentleman from EI a question?

DR. PLESSET: Yes.

DR. CATTON: It is my understanding that RETRAN is a copyrighted code. Also, it's very expensive if you want to use it. How does this all fit together, that the LOFT program is helping you at EI and EPRI develop a copyrighted code? That's government money.

MR. RICHERT: Well --

DR. CATTON: Or is it going to become publicly available?

MR. RICHERT: -- EPRI has exclusive domestic rights to license RETRAN.

DR. CATTON: And you have foreign rights.

MR. RICHERT: Many government installations have RETRAN licenses. Los Alamos has been licensed to use RETRAN, EG&G has been licensed --

DR. CATTON: But it's not available to the public.

MR. RICHERT: -- also Brookhaven. So the government obviously benefits from all of this development, since they have access to the code, as well. I guess I don't understand the problem.

DR. SOLBRIG: Ivan, I would like to point out that the objective of RELAP5 is to include these types of models. It was not adequately included in RELAP4/MOD7, and in order to do a decent job on predicting, we really do have to have the code like this. So we feel like we are getting the same information, and we will be able to develop RELAP to the same level of capability. So I think it is definitely a contribution to our program, and not just helping or subsidizing industry with this. This was not our intention at all, but to go out and get a code which was useable.

DR. PLESSET: Well, thank you, Chuck. I will call for a 10-minute break. So we should reconvene at 10:45.

(Brief recess.)

DR. PLESSET: We've got to reconvene and get back to our agenda.

Mr. Linebarger, you're on.

MR. LINEBARGER: Thank you, sir.

Mr. Chairman, ladies and gentlemen, it is warm in here. If you'll notice, the heat is on so I think the lack of coats is appropos for survival.

(Slide.)

This morning we are going to discuss the results of the LOFT small break test series. I will omit a discussion of L3-5 and L3-5A. You have had a comprehensive review of that particular experiment yesterday. Brian Sheron set the basis for our going into the LOFT L3 series. Regulatory requested that Research not only address certain specific issues such as the pumps-on/pumps-off question; but that we do a survey of scenarios -- accident scenarios -- and, as he characterized them, the system continuously depressurizing, the system pressurization stabilizing later in the transient just above the secondary pressure, and repressurization.

That was the genesis of the LOFT L3 series that we're currently conducting. I will give you a review of the progress of the results today.

Next slide, please.

(Slide.)

Following the introduction, which includes these licensing concerns specifically, and the progress, we will look at the results. Now Licensing was concerned about three areas. First of all, the general scenario of these types of transients. Secondly, how well the codes predicted the transient's signatures -- that is, the data that was produced from the transients. And in particular, the recovery methods. Such questions as the efficacy of the steam generator; did we see any noncondensible influence? Did ECC play a role as far as natural circulation and steam generator efficacy? Voiding in the core? What about operator action, such as secondary feeding and bleeding, and these sorts of things, were these effective?

These are the things that we will address this morning as we discuss the results of these particular tests. Then after that, we will draw some conclusions.

Next slide, please.

(Slide.)

In determining what tests we were going to run and setting the break size for these particular tests, we went to the Westinghouse calculations. Now the Westinghouse, we're scaled to a four-loop Westinghouse plant -- Trojan is the particular plant. We looked at these calculations and we saw an interesting transition between the 2-inch and the 1-inch break sizes characterized in meters, as you see them on the handout.

You see that in all of these areas -- that is, depressurization; decay heat, the way in which it is dissipated in the system; the ECC, the necessity for EEE; and core uncovery -- that there's a transition when you go between the 2-inch and the 1-inch break size.

We thought that we would run, then a 1-inch break size which would put us, again, on the lower line, as you see it. But instead of running a simulated 1-inch break size, we decided to use the 4-inch break size. The reason was that Licensing had some audit calculations, performed at INEL for the Westinghouse system, and these calculations showed that the 4-inch break size was the most severe particular transient. So that's what motivated us to choose the 4-inch and the 1-inch break sizes.

(Slide.)

As for our progress to date, the only test you see up there that we have not performed, as you know, is L3-6. That will be used in conjunction with L3-5 to address the pumps-on/pumps-off issue.

We have conducted two experiments in which we had a simulated 4-inch break size, L3-1 and L3-5. L3-6 will be the same.

We have essentially two experiments -- L3-2 and

L3-7, in which the simulated break size was approximately one inch. When I say "approximately one inch," I am including L3-2 in which we had an additional break in the system, unanticipated, which made it a bit larger through a portion of the transient.

I am also including L3-5A. That's the tag-end, as you know, or the ancillary part of L3-5 in which we isolated the break. There was no break during L3-5A. However, we were able to study the same sorts of details at the end of L3-5 that we did at L3-2 and L3-7.

As you see, we also looked at operator actions, steam generator feed-and-bleed, in most all of these experiments, and you will see -- because we're going to concentrate on Experiment L3-7, and I'll tell you why -- that during L3-7 we had operator action on both sides of the system while we were bleeding the secondary system and driving down the system pressure temperature; later on in time, as the temperature and pressure decoupled because we subsequently later had to, in addition, exercise the power-operated relief valve in order to depressurize the system. So we have looked at a spectrum here of operator actions.

Next slide, please.

(Slide.)

As for our results -- and I put this just for a comparative basis -- when you look at the same scenario that

I put for the Westinghouse transient, you see that essentially it is very comparable in all areas except core uncovery. During our 4-inch break test, we do not uncover the core as is predicted to occur in that Westinghouse system.

Now we are going to concentrate, as I said, on the 1-inch breaks -- the line you see acrcss the bottom -- for three reasons. One, you've already seen L3-5 and -5A -- particularly L3-5 -- and you've seen the calculations as compared to the data, and you have seen the characteristics of the 4-inch break.

Secondly, we want to emphasize the fact that we tried to exploit our uniqueness. We are nuclear; that's quite obvious. But we also have a size uniqueness, at least at this juncture. At this point in time, for instance, the Semiscale test facility has not been able to conduct tests with a break size of this magnitude simply because their heat losses to the environment are too great. Now they are working to correct that deficiency or that particular problem so that they can do that, but at this time we are forging ahead in this area and because of our size we can conduct such a test.

The third reason I think it's important to look at these in particular is, these are the more probably event. Seventy-eight percent of the penetrations in the primary coolant-- to the primary coolant boundary in the pressurized water reactor that Westinghouse makes, and to which we're scaled, 75 percent

300 7TH STREET, S.W., REPORTERS BUILDING, WASHINGTON, D.C. 20024 (202) 554-2345

of these are one inch or less in size. In the CE System 80, 64 percent are one inch or less. So we're dealing with the more probable event.

Next slide, please.

(Slide.)

In looking at L3-7, which is the most characteristic of the one-inch break experiments which we have run, here is what I call the "signature" of the experiment. This is the first place we look to determine what went on in the experiment. That is, the primary system depressurization history.

Here you see the chronology as these events occurred. The reactor scrammed. The pumps tripped. The HPIS on low pressure. Then the upper plenum saturated and took over as the system pressurizes. Later, we turned off HPIS.

Now we intervened in this particular experiment to do that in order that we might reduce the system inventory below that which would normally occur in order that we might look more clearly at some of the transfer modes in the steam generator, in the system inventory, and that was lower in value.

Later on, we started the secondary feed-and-bleed, and you will see that secondary feed-and-bleed now tends to expedite the recovery process. You see the depressurization increases; the secondary pressure will, as the primary system pressure follows the secondary system pressure. You see later on that the HPI comes on. We turned that back on. Shortly after we turned the HPI on, the system inventory started to rise because the HPI flow exceeded the break flow.

The accumulators came on for a short time during the experiment. I don't show them turning off, although they went off at about 8600 seconds. The accumulator was essentially not needed to recover from this transient.

Notice that this is a very convergent process. What I mean by that, it is very important that once an abnormal situation is sensed, and in getting to a stable end condition, the reactor and all of its system proceed in a convergent fashion. Meaning, that the system not go through a more severe situation than it was originally in; and that this proceeds very smoothely and very convergently.

You will notice that after we isolated the break, we went into system subcooling, and then the pressurizer actually started to refill before the upper unit refilled. There was a great deal of thermal non-equilibrium in the system.

And then later on in time, while we were still exercising the secondary system in the feed-and-bleed mechanism, the primary system was bled at the PORV in order to control pressure.

The next slide, please.

(Slide.)

Again, a quick look at the manner in which the codes were able to predict these results. I draw your attention to the fact that this is a post-test calculation. However, our pre-test calculation was not much different. I think that you'll see that we do a much -- the codes do a much better job of predicting this transient than we did on a pre-test basis for L3-5 and -5A. I think there are two important reasons why.

First of all, this experiment is not break-dominated, and we know that predicting the break flow is one of the problems we are having in L3-5 and -5A.

Secondly, and I think to put yesterday's discussion in context, it should be recognized that the predictions that you saw yesterday were pre-test calculations. I think we can do much better on our post-test calculational basis on L3-5 and -5A.

So you see that we do a -- the codes do do a good job. Late in time, we're having a little trouble. You'll see that late in time, the repressurization, when there is significant thermal equilibrium in the system, right now the calculations are not following. We're investigating the reasons why. We've taken one crack at trying to improve this; this has not improved it, and we're looking at it as far as the modeling is concerned and we're looking into this further.

At this time, if I may -- Dwayne, would you bring the movie up? You have seen a static picture of the experiment. I am going to show you a dynamic picture of the experiment. We had a copy of the film made, and it was wound backwards, and Dwayne has been in the back fitting it, so we would not have to stand on our heads and invert our eyes to make something of this.

(A movie is shown.)

This is a film. It is made from the actual data that we took during the experiment. It is an in-house production. That is, it is computer generated, and the input for the particular movie is developed by our analysts, and the computer animation is all done by our computer section in-house. The film is under development -- that is, we are going to continue to make improvements on it, and in fact in the Water Reactor Safety Meeting next week I hope to have an improved version.

It has multiple use. We use it for analysis purposes. We use it to show groups such as this what went on during the experiment, so you can get a visual picture, somewhat of a better visual picture of what occurred.

The utilities are interested in it because it may be very -- the operators say to them: You're showing us computer-generated stuff. What happens in a real experiment? What happens in a real reactor? And they're anxious to get -- and we have requests from the utilities already for these films, as well as our thermocouple film in the large breaks. So they can show the reactor operators exactly what occurs in a real reactor.

So there is one difference -- there is one error at the end of this particular movie. It doesn't show the accumulator coming on, and I will point that out when we get there.

You are going to see four hours of information in four minutes. So recognize that you're not going to be able to get everything that is occurring the first time through.

Can you see that in the back?

VOICES: No.

(Pause.)

It will be more heavily annotated in the final version. This is the system. The pressure supression system will not be in the system when it's actually running. You see the secondary in green (indicating), and this is the core in red (indicating).

This indicates that the temperature is taken from the thermocouples from the five hottest rods in the core, and the pressure is taken from the upper plenum. We are starting off now at T-0 with the break. You will see the break is -- the fluid is exiting the break at this particular time. The pumps have tripped off. They are starting to coast it down.

When this goes off, they coast it down. You will see that the shrink is occurring in the secondary side. We are continuing to depressurize as the liquid level depletes in the pressurizer at this time. The liquid continues down into the system. Then the upper head is the next portion of the system to void. It comes down to the level of the hot leg piping. We've put the level of the hot leg piping down here so it corresponds to the cooling piping. But then the cold leg starts to void, again before the hot leg completes its voiding, because these locations are draining back into the hot leg area of the system.

We are at 44 minutes -- 47 minutes into the experiment. The cold leg, you can see, is void at this particular location. We still have fluid in the system at this particular time. You will see that the HPI now is off. That went away from the system. We have taken the HPI away from the system.

Continuing into the mass out the break will not continue to feed the system with the mass. Now the hot leg is voided. The hot leg did not completely void during the experiment. You will see that of course the fluid did not even get down into the lower plenum, even though we have HPI off. So for this sized break for a single failure, the core uncovery problem just doesn't exist.

Now the HPIS comes back on. The accumulator is

300 7TH STREET, S.W., REPORTERS BUILDING, WASHINGTON, D.C. 20024 (202) 554-2345

on for a short time, as I indicated to you. It should go off a little bit later in the transient, and it does not. You will see that the fluid now starts to rise in the system as the HPI and the accumulator are both feeding fluid back into the system, and we are well on our way toward recovery.

You will see later in time here that the pressure will start to increase. See how it's starting to increase? We've isolated the break. I don't know what's coming in -- that may be something on the film itself. We had a copy made of this.

(Pause.)

You can see that we've isolated the break, and we have repressurized the system. We are continuing the feed-and-bleed. That's why these two valves are indicated here. We are seeing that the fluid rises in the system, filling all the high points. We are getting to the point that the operator is going to want to control that pressurizer.

And you can see that they are in the process still going at this particular time. Now it's coming down. So they open the PORV and that's what this indicates. The PORV comes on in time with the event, off, and then on for a time again. You can see that now the system is responding to the loss in pressure on the primary side, and when you turn that off it continues to pressurize again.

Okay, you can turn that off. That gives you a dynamic picture of what occurred during that particular experiment.

We are in the process of getting the same information together on L3-5 and L3-5A, our last experiment. We will then be able to give you our best understanding, a visual picture of the fluid distribution in the system during the pumps-off case, which we think will be instructive and helpful to us, also, from an analytic point of view.

Okay, the next slide, please.

(Slide.)

As for the recovery mode itself, the first question is: When is the steam generator really needed as far as four-inch and one-inch break size is concerned. To look at that, I refer you to experiment L3-1, which is also a four-inch break.

Here you have a plot of primary system pressure versus time. We have two calculations. One is the RELAP5 calculation with the steam generator; and the other is the RELAP5 calculation without the steam generator. You can see that of course the system depressurizes and recovery can be effected, even though the steam generator is not in the system.

So we have validated the fact that in fact in a four-inch break the steam generator is not needed. However, it is a different story in the one-inch break experiment.

As you can see, this is a plot of pressure versus time in L3-7, the experiment that you just saw dynamically. As you see, the primary system pressure stays above the secondary system pressure throughout the transient, and it proceeds. Thus, the steam generator was an effective heat sink throughout the experiment. In fact, in the 5- to 6000 second time regime, about half of the energy is leaving the system through the break, and the other half is being taken up by the steam generator.

So we see that the steam generator is needed in the one-inch break to remove the decay heat. Also note the efficacy of a steam generator feed-and-bleed. You are controlling the secondary system pressure without opening an additional break on the primary side during this particular time. And then because there is good thermal communication in the steam generator, the primary side, or the primary system pressure follows it and, as a result, recovery is expedited. So this process of recovery is a very convergent and a very smoothe, if you will, process.

Next slide, please.

(Slide.)

As for the natural circulation mechanisms that we have referred to, natural circulation flow is of course the flow that is needed in order that the heat may be dissipated in the steam generator that is generated in the system.

This slide is the slide of L3-7 reactor vessel fluid temperatures and velocities. On the left is fluid temperature; across the bottom is time in seconds; and then on the right you see velocity in meters per second.

The top curve shows the velocity measured by a turbine meter in the upper plenum just above the level of the fuel itself. Then you see the upper plenum fluid temperature, a lower plenum fluid temperature, compared with saturation temperature.

The "10" you see on the left shows that we were in single-phase natural circulation up until about 400 seconds. At that time, the saturation chronology started to evolve within the system. The saturation chronology means that various portions of the system went into saturation as time proceeded through the experiment.

As a result, there is a gradual transition from single-phase to two-phase natural circulation, and it's a very stable, gradual process. This saturation chronology I think is important. We show that in the large-break series of course the saturation chronology determined what happened in the system with time. And then again in the natural circulation phenomena is again.

You can see at 400 seconds a rise in the velocity measured at the upper plenum fluid velocity, due to two things. First, you're increasing the volumetric flow of the fluid at that particular time. Now you have some slip in the fluid because there is liquid and gaseous phase and the turbine is sensitive to a slip in the fluid.

Then by 1300 seconds, the lower plenum is saturated and we are in pure two-phase natural circulation throughout the system. So you see that it is a stable and gradual process as it develops.

(Slide.)

In the next slide you see the overview of the entire transient. This is a plot of the reactor vessel fluid temperatures -- upper plenum and lower plenum -- compared to saturation, which of course tracks the system pressure throughout the experiment.

On the left is temperature, and along the bottom is time in seconds. You can see that we start out initially in single-phase natural circulation. We evolve into two-phase natural circulation. Then we reversedly go back to one-phase natural circulation. So we see, again, a very convergent process; we see a very stable process; there are no rapid diversions or changes in this -- excursions in the temperature or pressure profiles in the system.

We see a very natural evolution from single-phase to two-phase and then back into single-phase, as the natural circulation process is indeed reversible.

I haven't shown you the natural circulation process as far as the experiments we have run is reestablishable. That was shown yesterday by Keith Condie when he showed you L3-5. We isolated the break. The secondary system pressure was below the primary system -- rather, the converse. That the primary system pressure was below the secondary system pressure.

We ran the experiment in order to show that in fact natural circulation would be reestablishable, and to look at the various cooling modes, and we showed that that occurred in that particular experiment.

I haven't discussed in detail the modes that we have seen as far as the occurrence of reflux. We have seen the reflux mode during two of our experiments. You saw one yesterday when Keith showed you the L3-5 experiment.

It was also predicted to occur in the codes during L3-5A we could not measure reflux during a significant portion of that transient. We measured a bit of reflux as natural circulation was being reestablished, and the codes indicated -- at least in our pre-test calculations -- that reflux would not be a dominant mode.

In experiment L3-2 we saw the same indications that we saw in L3-5, which lead us to believe that reflux occurred. However, during L3-2, we had an additional break. So what I'm saying is, the only time we've been able to establish reflux in our system and been able to measure it, there has been an additional influence on the flow other than just the flow induced from natural circulation. We had break-flow induced flow, as well as the natural circulation induced flow. Those are the only times we've been able to generate reflux that we could measure, at least.

(Slide.)

In conclusion, the natural circulation occurs in the single- and two-phase modes. It is stable in and between modes, as far as we've been able to measure in our transients. We've seen it be reversible. We've seen that it's reestablishable. We have not seen it deterred at all by the influx of ECC into the system, or reactor vessel voiding, and there has been some limited reactor vessel voiding during experiments. As far as noncondensibles are concerned, in the four-inch break size the nitrogen entered the system long after the steam generator is not needed, if it enters at all. We let it enter our experiment because we delayed LPIS entry just to see what effect the nitrogen would have.

As far as the nitrogen in the one-inch break size, the accumulator really isn't needed. If it comes on, it comes on for a very short period of time, and you never get to the point where the nitrogen enters the system. So we've not seen, at least in the single-failure experiment breaks, that nitrogen will be a factor as far as the noncondensibles in the system.

DR. ACOSTA: Excuse me. You mean the nitrogen does not get into the system.

MR. LINEBARGER: Yes, sir. In the one-inch break size, sir, the nitrogen does not get into the system. In the four-inch break size, the nitrogen does get into the system later in the transient. However, the steam generator is no longer playing an influence, or having a role in decay heat removal. So there's no natural circulation to disturb.

DR. THEOFANOUS: Okay, but that's not what this schedule says. I think you should say it is "not relevant," but not that it is "not deterred."

DR. ACOSTA: Yes. It's not a conclusion.

MR. LINEBARGER: Yes; that's true. That's it. That's why I wanted to quanitify it beyond that; yes.

DR. ACOSTA: But when you say "not deterred by," that's a pretty strong statement. And since there has been in the past so much on that particular issue, you ought to rearrange that.

MR. LINEBARGER: If you'd like better terminology, I'll work on that terminology. I understand your point, and it is well taken. So I appreciate the comment.

Is there anything else in that area?

(No response.)

MR. LINEBARGER: As for the licensing conclusion --

(Slide.)

-- we have seen that the PWR and LOFT scenarios are comparable. I have tried to word this very carefully, the next one:

That is, that the calculations we feel do predict the dominant transitions and associated phenomena in the proper time sequence. As you saw in L3-5 and L3-5A, we are having a little bit of difficulty in the time scale and the magnitudes. However, even though we are off that way, we are seeing the proper transitions due to the proper phenomena. But as I say, I think we can do a much better job in L3-5 and -5A in a post-test mode.

The recovery process is convergent, as we have observed it.

The next slide, please.

(Slide.)

The steam generator is an effective heat sink.

Then of course you have the secondary playing its proper role, and the operator initiated steam generator feed-and-bleed does expedite the recovery process. As as I indicated to you earlier, you can even superimpose that you control the primary system pressure with the PORV late in such a transient.

DR. THEOFANOUS: I think again it might be open to interpretation. You want to be careful, especially when you say "licensing conclusions."

MR. LINEBARGER: Okay.

DR. THEOFANOUS: Beginning with your first viewgraph, it says: "LOFT small break results."

MR. LINEBARGER: Yes.

DR. THEOFANOUS: That is the title of your talk.

MR. LINEBARGER: Yes.

DR. THEOFANOUS: And I assume, then -- and I think it is a reasonable thing to assume -- that your conclusions refer to that topic you are discussing.

MR. LINEBARGER: Yes.

DR. THEOFANOUS: So when I see the statement that "PWR and LOFT scenarios are comparable" and the calculations predict definite transitions and associated phenomena in the proper time sequence --

MR. LINEBARGER: Yes.

DR. THEOFANOUS: -- I think I am allowed to kind of take that conclusion as applying to all small breaks. And I want to know whether you feel comfortable in this point to make that general conclusion, or whether you want to make that conclusion only with respect to the one-inch and the four-inch break.

MR. LINEBARGER: When I say "LOFT small break results," I am alluding only to those results that we have obtained to date, and only to those sizes that we've looked at. However, I can say in general that I think they're fairly characteristic of break sizes in that regime. But I am only referring, obviously, to the results that we've obtained to date and the sizes that we have looked at.

DR. THEOFANOUS: Yes, but you have obtained results also that are not very well predicted. I think that on the basis of what results you've showed us in these presentations today, maybe you could make this statement, in which case you have to qualify it because of particular kinds of breaks. But I do think it is dangerous to extrapolate and say for all small breaks, or for all previous breaks, which you have done.

MR. LINEBARGER: Well, that's why I tried to qualify it. First of all, I qualified the remarks. Let's look at the wording. It says: "predict dominant transitions and associated phenomena in proper time sequence." If you look at the details of the calculations in L3-5 and L3-5A, you will find that to be true. We are --

DR. THEOFANOUS: I don't think you are able to tell which are the dominant phenomena. If you can tell me -- if you are able to say which are the dominant phenomena -- that's the kind of discussion we had yesterday -- if you have concluded, before anyone besides you knows what are the dominant phenomena, and to what extent they are dominant, then I would like to see it someplace written.

MR. LINEBARGER: Well, that's why I'm saying --

DR. THEOFANOUS: And tell me where it's at.

MR. LINEBARGER: I'm afraid I don't understand. I'm saying here: This is my interpretation based on what I have observed in the L3-5 and the L3-5A calculations versus the data. I have looked at the break flow, which is the problem that we were having. As I look at the differences between the systems -- for instance, the Semiscale system and other large and small breaks that we ran, the 4-inch break -- I believe that I do have an understanding of what are the dominant phenomena. And we are seeing, as we look at the break flow for instance, that it has the same general characteristics as the calculated -- as the experimental break flow.

It does not have the proper magnitude, but it does predict the transitions. That is, when the system saturates, it predicts the fact that the system is going to go into saturated break flow.

Now we are looking at exactly why did the calculations -- trying to look at, in a post-test mode, why the calculations did differe from the data as far as they did. And I think that we will find that in the break geometry, because w. had this six-foot pipe that was about a 1-1/2-inch diameter pipe leading into a 14-inch diameter pipe, we are seeing some interesting transitions there that the codes are not picking up. So that we are not properly predicting right now on a pre-test analysis basis, we are not properly predicting the conditions in front of the break.

DR. ACOSTA: Is this true for the one-inch break now?

MR. LINEBARGER: No, sir. I'm talking strictly about the four-inch break.

DR. ACOSTA: So in this presentation here, your break flow must be all right.

MR. LINEBARGER: Yes, sir. It's very accurate in this one.

DR. ACOSTA: It would be nice to see plots of that, as we saw plots of the four-inch break yesterday.

MR. LINEBARGER: Right. I didn't show that. The pressurization is so accurate, I haven't shown that, but it is--

DR. THEOFANOUS: I don't -- I don't --

MR. LINEBARGER: You're so right. We are off on the break flow. There's no question about it.

DR. THEOFANOUS: I don't want to belabor the point. I have just one final comment I want to make. It troubles me. I think it goes back to what we heard earlier from the Chairman of the Subcommittee.

We kind of have a tendency -- and I think it is understandable and it's human -- to emphasize all the good [illegible] about what results we're getting and what you learn from things. I would hope that in a meeting like this, we would like to see -- or at least I would like to see -- a little bit more balanced presentation of problems as well as achievements. And if you were here yesterday -- I didn't see you yesterday --

MR. LINEBARGER: I was here.

DR. THEOFANOUS: You were here yesterday? And we had a full discussion with the NRR staff and the other people trying to decide whether some of these disagreements were significant or not significant.

Now you're telling me, if I read very carefully your second statement and the first statement, you are not really in very bad shape. So that's your opinion, and other people in this room seem to disagree with that. That's all I want to say.

MR. LINEBARGER: My point is that -- forgive me if I haven't -- I tried to word this carefully, because I agree with your point that we have a lot of work to do in the L3-5 and L3-5A. There's just no doubt about it.

I did want to point out, however, that we are at the disadvantage, for instance, when you saw the Semiscale results, that they have had a great deal of time to go over this and do a lot of post-test analyses. You have seen -- we have shown you everything we've got. We've laid everything on the table, and we've shown you exactly what was predicted. We have not tried to hide anything.

What I'm saying is, I think that, as I looked at the codes, that there's real hope in a post-test anaylsis mode, if I may put it that way, I don't think they're as bad as they were shown to be yesterday.

DR. THEOFANOUS: I'll go along with that.

MR. LINEBARGER: And you have to look at it much more clearly in a post-test analysis mode, and I think we are converging on the reasons as to these differences. But you are exactly right.

For instance, in L3-7, as I showed you, we're not predicting that the temperature stratification and the repressurization of the system properly late in time. We've got to work on that particular problem. That's the only one that really has popped out as being obvious -- an obvious deficiency there. But these deficiencies must be corrected; yes, sir.

DR. WU: So following this up, would you quantify -- further quantify the conclusions in terms of some other parameters? Like regardless -- Well, suppose you take the break size for being granted, and then it would still be the same conclusion that would hold regardless of the location, any like location in addition to the shape, and so forth?

MR. LINEBARGER: A very good point. And I think what we'll show in L3-5. Because in L3-1, which was the four-inch break, we did a pretty good job, as you saw, predicting the transient depressurization.

DR. WU: Yes.

MR. LINEBARGER: Now we go to the intact loop. We take about a 1-1/2-inch ID pipe and we put it on to an 11-inch ID pipe. We put our orifice down about 7 feet -- which is characteristic of an instrument line or something like this displaced from the primary system, but does penetrate the primary system, and now we have some problems. And I believe that when we try to predict fluid conditions as they mix coming out of that hot leg -- cold leg pipe into that T, and then as they transition down to the break, that we're not doing that properly.

And as I looked at the data, I can show that we're not doing that properly. And we don't understand it, yet we're in the process of comparing those particular things. So that does say that we are sensitive to break location; yes, sir. I think that's a good point.

DR. PLESSET: Could I make an optimistic deduction? That we really don't need any more small-break LOCA experiments in LOFT; that we can fix up some of these little holes and gaps with separate-effects experiments. I liked your optimistic conclusion in that sense, until it was criticized a little bit. Would you say that?

MR. LINEBARGER: I would have to think about that. I would have to see whether certain of these things are principally system effects, or that we can simulate them in a separate effects facility.

DR. PLESSET: Okay. Well, I'd like to know what your conclusion is.

MR. LINEBARGER: Yes, sir.

DR. PLESSET: Okay. Thank you.

MR. LINEBARGER: That concludes my remarks, sir.

DR. PLESSET: Very good.

Well, I think we can go on to our next presentation, Operator Intervention.

(Slide.)

MR. MEYER: Good morning. I will personally welcome the opportunity to present a critique of a topic which is new to the LOFT research program, and is actually new to the entire lightwater reactor community, and which a lot of us feel is extremely important to the future of nuclear energy in the United States.

The work I will report on is funded under the LOFT augmented operator capability program. It is part of the programs funded by the Operational Safety Research Branch of the Nuclear Regulatory Commission. As such, it relates. There are other programs, such as severe accident sequence analysis, and plant status monitoring to the LOFT program.

(Slide.)

As has been discussed at great length, and as you all know, LOFT --

DR. PLESSET: How big is that part of the program?

MR. MEYER: Excuse me?

DR. PLESSET: How big is that part of the program, roughly?

MR. MEYER: In fiscal year '80, it was slightly under a million dollars.

DR. PLESSET: I was just kind of curious.

MR. MEYER: What I am reporting on is principally the fiscal year results. LOFT is deeply involved, as you know, in experiments involving operational transients and small-break loss-of-coolant accidents. It has early on recognized that in these types of accidents, the operator plays a dominant role. So two questions were addressed early in trying to develop a research program which could produce useful results with respect to the operator's role.

The first question of course is: What is the operator's responsibility?

The second question is: How does he exercise that responsibility? What does an operator do?

The first question is actually answered rather simply. The operator has the final responsibility for reactor safety. The automatic protective systems that are on the United States' reactors initiate protective action primarily in those cases where protective action must be initiated within a relatively short time period, such as within ten minutes. Beyond that, the operator must terminate the accident and he must restore the plant to a safety condition.

It is obvious, for example, that you cannot walk away from a plant with the emergency core cooling system operating at full flow.

The second question is: How does the operator exercise that responsibility? That is much more difficult to answer. As a matter of fact, from the limited research we have done, we have found very little that actually documents and defines just what an operator does, and how does he do it.

There is considerable information on the plant itself, and the plant's behavior. I think this sometimes is not recognized. It exists in this form -- typically in this form. And as a matter of fact, this type of information is used in the training programs.

Operators receive classroom-type instruction in theory of plant behavior, but this is not the type of information that he uses in operations. It's not available in the main control room.

(Slide.)

By the way, the previous slide that you saw was pressurizer level and primary plant pressure behavior during a loss-of-coolant accident.from an experiment conducted at LOFT.

(Slide.)

The corresponding form of the information that's in the main control room at the present time is in the form of individual meters.

(Slide.)

Here you have the corresponding pressurizer level and plant pressure. With this type of information, the operator primarily operates according to two principles. The first principle is one that is known as "operation according to rules." There is a training program of rules with respect to when he should initiate charging at what are emphasized. In other words, at what pressurizer level, what meter indications charging to the primary coolant system should be initiated.

The second rule involves the development of skills. The operator develops skills on observing these individual meters and, for example, following the scram the operator will develop a skill in recognizing the amount of shrink that occurs in pressurizer level.

These skills are limited, however. He cannot develop skills in relating the behavior of pressurizer level versus time, for example, to the behavior of steam flow versus time; and it also corresponds to a third parameter, the behavior of primary plant pressure versus time. There are just too many individual meters to look at.

This is the reason why there is a heavy renewed emphasis on the training programs using more advanced simulators, because this is the type of behavior, operating behavior, based on rules and skills that you train an operator by trying to duplicate those type of meter indications so that he develops a learned response to specific rules and skills that he can exercise watching individual meters.

(Slide.)

This type of behavior for this type of study, Dr. Rasmussen's classification of operator behavior is quite useful. And what I've been talking about at maybe too much length is operator behavior based on skills and rules, which might be classified by lower-level behavior as compared to behavior based on knowledge.

Now why the interest in behavior based on knowledge? Training programs which emphasize behavior based on rules and skills work very well when the exact scenario can be duplicated, but if the plant behavior or the series of events and failures that occur were not those that were addressed by the training program, the operator needs to fall back on his knowledge of plant fundamentals -- theory, if you want to call it that.

As a matter of fact, in a discussion yesterday with Tom Pointer, the Manager of Operations on LOFT, Tom pointed out that an experienced operator who has not gone through refresher training of theory tends to forget original knowledge of plant theory, or the fundamental theory of plant behavior if he doesn't use it.

It was for this reason that the number one recommendation under the technical section of the President's Commission on Three Mile Island was that information should be added to the main control room to permit the operator to use his theory, his knowledge of fundamental plant behavior.

(Slide.)

That recommendation is being implemented by Regulatory primarily by NUREG-0696, which exists in draft form at the present time. And by the way, it has generated considerable dialogue between Regulatory and the utilities and the NSSS vendors. 0696 directs that by January 1982 a safety parameter display system should be operating in the main control rooms of all pressurized water reactors.

It directs that the primary function of the system is to help operating personnel make quick assessments of plant safety. It further states that it is desirable that this system be sufficiently flexible to allow for the future incorporation of advanced diagnostic concepts.

The vagueness of this second statement is in itself an indication of the need for research to at least define what the objectives of this type of system is.

(Slide.)

Now as most of you know, LOFT has an operational system installed and operating, what I would call a developmental model of this type of system. I use the term "developmental model" specifically because it is a term well established in control systems, in human engineering in the aerospace industry.

Why do they use it? And when do they use it? You use it when you are trying to design a fundamentally different type of system -- particularly a system which has a strong man-machine interface. In other words, where the relationship between your system -- particularly a control system -- and the operator is extremely important.

Well, all of those terms apply here. Implicit in 0696 is the assumption that the President's Commission recommendation will be followed, and that this system will be computer based. That's a brand-new concept to the control room.

Second, it is new in what it's going to do. Terms such as "safety state vector," as Nick mentioned, "safety parameter display," are brand-new terms. They're not final. They're in the process of adjustment as people learn what this type of system might be able to do.

That is the reason why early in our program we emphasized the installation of a developmental model in LOFT. Now what does the aerospace industry get out of the developmental model? They install it in as near to a realistic environment as possible. The end objective of developmental model trials are to teach the engineers and the scientists so that they can then sit down and write firm, final design requirements for these types of systems.

(Slide.)

This is a block diagram of our system. We think it's analogous to the systems which will be installed and backfitted into the presently operating plants. As a matter of fact, there is a system very similar to this that has been installed by EPRI in September at the SNUPS training simulator for testing.

Over here (indicating) is the LOFT data acquisition system, which is very analogous to the process computer systems on the front line plants, the ones that have the best process computer systems.

The art term for the system, by the way, I might mention, is "operational diagnostics and display." That term is different from the "safety parameter display system," and itself reflects what a changing environment we're in. It's different because we picked that name and installed our system before 0696 was written.

The system itself consists of a small computer interfacing with an interactive color terminal. The primary purpose of this computer is to store past data from the plant, to store directions -- to store hard, documented type of data such as operating procedures, operating limits, and so forth, and to generate a graphic form to display that information.

An interactive terminal is used because the purpose of the computer is not to dominate the operator; the purpose is for the computer to serve as the servant to the operator.

(Slide.)

This is what the interactive color terminals look like at an early stage of installation in the LOFT control room.

(Slide.)

0696 further goes on to state that for each mode of operation of the reactor plant, a single primary display format, designed according to human-factors principles, shall be routinely displayed. So we have given attention to what type of format is really useful for displaying plant safety status for various operating modes.

(Slide.)

We have had the cooperation of Dr. Danchak from the Hartford Graduate Institute on this work. Dr. Danchak, by the way, was the designer of the mimic diagram of the type of displays that are used in the so-called "advanced control rooms" from the nuclear steam supply vendors.

What he did was go back to the mathematical literature on graphic forms for displaying multi-variant data, and then he selected nine types of graphic formats which show some promise for displaying the overall plant safety status.

Of these, I will show you five of them in the time we have available.

Now the first type, Mike terms the "circular profile." It's very similar to the star profile used for the Japanese, and I believe Westinghouse. The difference is that not only is the value of the parameter indicated on the radii, such as feed flow in this case, but the area within the inscribed lines is filled in. The purpose of that is to see whether or not pattern recognition principles will be useful in this problem of station.

By the way, I might state what I think is a fair remark that we feel that there has been an early selection from an arbitrary selection which sometimes occurs naturally, on a certain particular type of format for displaying this type of information, such as the star diagram.

Now what you will see is the various formats used with actual data from an operational transient on the LOFT plant. The transient we selected was the loss of feed flow. In the limited amount of displays I can show you, what you might do is watch "feed flow" --

DR. ACOSTA: What was the normal operation?

MR. MEYER: Pardon?

DR. ACOSTA: What would be normal operation?

MR. MEYER: Well, this display should be normalized, ideally, so that you normalize these parameters to power, you might obtain a very simple pattern that looks like a circle. In the amount of working time we had, we weren't able to normalize this to that degree.

DR. ACOSTA: Okay.

MR. MEYER: But all the displays are normalized to plant power. Plant power is not shown on the pattern itself; it's printed out separately.

(Slide.)

Obviously the loss of feed flow is a very easily recognizable change in this pattern. The operator, probably on some of these displays, can also obtain some quantitative information as he watches the change in pattern slowly occur, and I will point that out on some of the later formats.

(Slide.)

The second type of display is very similar. It's called a "five-fold circular profile." The difference is primarily a matter of how you shade in the -- draw in the circumscribing lines. The value of the parameter is again the radii. So you again might watch feed flow.

(Slide.)

This is an obvious change in pattern. And again, probably some quantitative information can be obtained for watching the pattern slowly change after the abrupt loss of feed flow.

(Slide.)

The third type of display is one that seems quite attractive to us. By the way, the thing that we are constantly trying to avoid is this business of sucumbing to something that looks obviously attractive to an engineer. We are not at the stage where I really could promote one particular type of these displays, because we have not run any operator tests on them under any sort of realistic conditions.

On this one, the principle is that if you normalize your parameters and everything is normal, there is no deviation shown and essentially all you will have is a narrow line to let the operator know that the computer is still awake. In this particular case, again you might watch, if I can find it on here, secondary feed flow.

DR. CATTON: Where are you going to get your operators?

MR. MEYER: Excuse me?

DR. CATTON: Where are you going to get the operators that will interact with you?

MR. MEYER: That is a very big problem. LOFT has a team of operators. They are very useful in this sort of thing, because they have some understandin of what we are trying to do. In other words, they have some understanding of the basic objectives of the research program.

DR. CATTON: So they are, from what I gather, sort of supertrained. And that won't be the operator that would necessarily be using this.

MR. MEYER: That is one of the biggest questions that we are addressing for the next year's work: How to get objective data on this subject. The testing we can get with LOFT operators hopefully can give us some insight to plan objective test programs, perhaps like EPRI and Westinghouse are doing with the SNUPPS training simulator.

(Slide.)

Again, feed flow loss is clearly obvious because it goes off scale as far as deviation, and a color change is thrown in, too, to make it even more evident. In this case, I have a third graph to show what happened a few minutes later. The steam generator level was initially a little bit high, so it is colored in high and yellow.

(Slide.)

You drop after a few minutes. It dropped down to the normal operating band, showing a smaller deviation, but also displaying the green color.

(Slide.)

This type of display is really similar to the circular profile. The only difference is the linear format. Again, you might watch feed flow and steam generator level.

(Slide.)

Loss of feed flow is very obvious. The steam generator level decrease is read off somewhat quantitatively.

(Slide.)

The final of the five displays I will show looks like an enunciator panel, and it is related to the enunciator panel. The enunciator panel, however, is a binary type of device. It has two colors, only white and red. The concept here is to extend on that and use an array of colors or a spectrum of colors to try and portray a pattern of overall safety status.

(Slide.)

Again, the loss of feed flow is quite evident. I'm not sure why Mike picked blue as the color here, but it is evident by the changes in color.

(Slide.)

All right, the next section of this presentation I would like to present to you for your comments and criticisms has to do with a different type of approach we use. That is, we used our experience with conduct of the loss-of-coolant experiments to try and identify types of diagnostic information that might be useful to the operator. We used the operators very definitely to tell us what type of information might be useful. But it is strictly an attempt to identify some concepts and, as I said previously, to put them on the developmental model, get some experience in something that's close to a real environment.

(Slide.)

The first type that was identified is termed an "operating map" and it is used extensively by the aerospace people. It is a basic type of display, for example, that the space shuttle used. In this case, the green rectangle indicates the normal operating pressure and temperature limits for the reactor for power operation. The little black dot there is the current operating state of the reactor at this time. In other words, it is operating at this time, I believe, at about 75 percent power.

(Slide.)

Here a small-break loss-of-coolant experiment is being conducted. The pressure temperature behavior of the primary coolant system is plotted out on the map.

(Slide.)

At later stages in this loss-of-coolant experiment or accident, the fact that the primary coolant system has now reached the condition where the boiling in the primary coolant system is quite evident, because the operating point, which is the white dot for current operating state has been following the saturation line.

(Slide.)

I am not going to try and show you all of the concepts we displayed, or tried to identify. I would emphasize that we tried to identify concepts. We haven't worked out a good, logical relationship between the concepts.

The second type addresses the question of operation of primary coolant pumps and the primary coolant system pressure boundary itself under an accident condition. In this particular case, the limits for the suction head requirements for the primary coolant pump are shown by the two lines at the bottom. The fact that the plant is in this state as far as pressure injector is concerned tells the operator that he doesn't have the designer's recommended NPSH for the pumps, and that his flow problems will be reduced.

(Slide.)

This type of diagram, the mimic diagram as I said, is the basic type of display that is used in the advanced control room. Obviously it is something that would have to be used by the operators, because it basically tells us his flow patterns, his temperature distributions, and his valve conditions.

(Slide.)

All right, this one had a slightly different origin out of the LOFT program. Due to, as Nick mentioned, our emphasis on operating LOFT according to the same rules that the commercial reactor people do, we developed a section in the plant operating manual that would permit the plant to shut down safely, even though there was a worst-case occurrence of design-basis event -- the loss of commercial power simultaneous with an earthquake, et cetera. The object for the use of this part of the manual is to assist the operators to maintain cooling water flow to the primary coolant system, and specifically to the reactor vessel.

Basically the way he would use it in such an unlikely condition would be if he obtains information on what equipment is not operable, like he might have lost the void water storage tank because it's empty, or its flow path to the vessel is no longer available because a valve has lost electrical power. He marks out what equipment is not available. The remaining flow paths are then evident by a path that doesn't have a red cross in it.

In addition, a priority has been pre-established as to which is the best flow path to use, and he picks the highest priority number, and that is his recommended flow path according to the plant operating manual. It is a straightforward step to consider computerizing such a thing.

(Slide.)

This is not a CRT display. This is in the form of a specification for programming a CRT display. In this particular case, the MPBWST is indicated in red. The unavailability of this cross-tie is indicated. Those are manual valves that are normally shut. The loss of power to the A pump is indicated, and then the computer would take prestored information in the computer and recommend, by using a bold-line display what his best available flow path is.

In addition, flow paths which are still possibly available are shown in a distinctive shade, in this case brown.

(Slide.)

Okay, this is an example of learning from operators, which we feel it must be emphasized in this type of program. We engineers and scientists have too long talked to ourselves -- in fact, most of us don't even know how to talk to operators.

This was my attempt to address the problem of what does the operator need to know for long-term decay heat removal, and we're reduced to the stages of maybe a couple of hours or a day after initiation of a loss-of-coolant accident.

We tried it out on the operators, and Tom's operators are pretty polite, so it took me awhile to learn that they thought it was useless.

(Slide.)

So his shift supervisor sat down and understood what I was trying to get at. He understood the problem. So what he did was pose a scenario for himself. He imagined himself in this situation, and he's on the night watch, and Tom walks in unexpectedly early the next morning and starts asking some questions.

So what the shift supervisor did was define the questions that he would have to answer. Starting at the top, they are questions with respect to the status of the primary coolant system of the reactor. Of course you won't have cladding temperature on a commercial plant, but you'll have other temperatures. The flow rate to the primary coolant pumps, if they're operating at pressure.

The next question -- and I won't try and cover all of this -- is the source of cooling water flow for the reactor. This display, by the way, is not complete. A lot of this information is not filled in.

In addition, the next point or question he might be asked is: Which pump are you using for flow to the primary coolant system? When the display is complete, the operating pump would be indicated probably in blue, and the flow rate would be printed out.

The computer's task in this really is pretty simple, after it has generated its display, because the answers that operators want quite often are pretty simple and direct. So what the computer next would calculate in answering the operation manager's question was: What level do you have in this tank you are using? And how much longer can you continue to operate that way before you have to change lineup?

Now this concludes this series of displays. The final display I would like to show you is one that I think best illustrates what can result from a combination of a team of operators and research engineers that actually can get their hands on a small power reactor, water reactor, that can be placed in unusual type conditions.

I would like to give you a little bit of background. During one of the small-break experiments, some interesting behavior was noticed on the pressurizer level instruments, and the resulting analysis of that data. This was attributed -- well, let me put it the other way around. This was associated with some interesting small changes that were occurring in some thermocouples within the reactor vessel.

(Slide.)

It resulted in this type of display. This type of graphic display was then developed. The thermocouples of interest for what I am going to show you are above the core. They're on the core support structure. The thermocouples on the core cladding I've shown down here, but I won't address them in explaining what this display did.

The deviation bar chart actually was used. As you remember, that's one of the five types of displays, or the nine types of displays actually, that Dr. Danchak has recommended for consideration.

In this case, however, zero deviation means that the core support structure temperature is at the same temperature as the saturation temperature in the system. So what the computer does is receives data on what the primary coolant system pressure is. It calculates the corresponding saturation temperature and displays that value here. It then constructs a scale showing 100 degrees' deviation where the support structure is colder than the temperature at which there would be film boiling on the surface, and the temperature at which the core support structure would be 100 degrees superheated, or 100 degrees above the temperature at which boiling begins to occur on the metal surface.

This is the condition of these temperature displays during normal power operation. Again, they're around 75 percent power. And of course all of these support structures and the cladding temperature itself is below the saturation -- the temperature corresponding to saturation for that pressure.

(Slide.)

Immediately following the initiation of a small break and reactor scram, the temperatures start to equalize because you've lost your large rate of power generation within the cladding.

(Slide.)

You then reach the point actually where the -- you reach the point at which the saturation temperature is very close to the core support structure temperature. This is really a coincidence, because what is happening is that the primary coolant system is cooling down, and the core support -- and it's now past the point at which the saturation pressure due to the depressurization is just coincidentally equal to the core support structure temperature.

(Slide.)

At this point, something interesting shows up. The plant is continuing to cool down. The boiling is occurring within the reactor vessel. Boiling, therefore, sets a pressure corresponding to the saturation temperature of the liquid. The saturation temperature of the liquid is decreasing.

What it has done is left behind the core support structure, the heavy metal structures up there, which are at a temperature now above the temperature of the liquid. And something interesting shows up, and it is very evident on this type of display.

The core support structure is now at a temperature above the temperature of the liquid. So the only dominant reason why that could occur is that the core support structure up in this region (indicating) is no longer bathed with liquid where it's getting high heat transfer rates at the surface. It's in a steam environment.

(Slide.)

At further intervals of time, this becomes much more clearly evident. From other data, we know that the inference you can make off of this display is correct; that the water level between the liquid and the vapor is someplace in this (indicating) region. This, by the way, is 21 minutes after the initiation of a small break.

The significance of this is that, using the instrumentation that may be available in some of the commercial reactors, using on-line computer technology, and using graphic display of that information from the computer, a type of information that is absolutely not available in the main control room can be made available to the operator.

In this particular case, for 20 minutes into a small-break loss-of-coolant experiment, at the particular time when this threshold is reached of steam formation in the top of the reactor vessel, you can extract information that tells the operator -- can possibly tell the operator that important fact.

DR. THEOFANOUS: May I ask a question here?

MR. MEYER: Yes.

DR. THEOFANOUS: That's very interesting. Are those thermocouples, now, you say, in place for many of the reactors?

MR. MEYER: Yes. It's a typical practice to place some thermocouples on the core support structure.

DR. THEOFANOUS: Can you tell me how are they placed? Can you tell me in a little more detail?

MR. MEYER: I can't answer that question. I suspect that they vary from plant lant.

DR. THEOFANOUS: Are they in some kind of a penetration into the body of the metal? Or are they stuck on the surface? Or how are they attached, or where are they attached? Do you know?

MR. MEYER: Well, we could speak to how these are attached. Somebody else can probably answer that more accurately than I can.

These, from my understanding, are attached to the surface of the core support structures. The purpose was to obtain the information on what happened to the core support structure during a big loss-of-coolant accident, and ultimately to use that information to find out whether there are significant thermal stresses generated. And, for related reasons, I suspect that a lot of commercial reactors have similar thermocouples.

DR. THEOFANOUS: But you don't know how they are attached?

MR. MEYER: No. I wouldn't attempt to answer that question for a commercial reactor.

DR. SOLBRIG: I'll try and get that information to you.

DR. THEOFANOUS: That's very, very, very interesting.

MR. MEYER: I'm not trying to say that this type of equation can be installed in every commercial reactor and hooked up to the thermocouples that they have. That's not the basic point. The basic point is that if you can use the computer in real time, directly connect your instruments you have in the plant, and then do some predesign diagnostic programs, you can give an operator the information that he just doesn't have at present.

(Slide.)

The final couple of slides here just continue to show the same phenomena. And the last one here -- that was the last one -- shows that if the saturation temperature starts rising, this particular phenomena is no longer useful to the computer. Your liquid temperature is now starting to heat up again, and the core support structure is not in a condition where it will be hotter than the liquid or hotter than the steam.

So that concludes the presentation of our diagnostic graphic displays. Referring back to the title of my presentation, which was "Operator Intervention," you can gather that we have placed high priority on providing information to the operator so that his intervention can be based on knowledge and intelligence. And through that means, we will enhance the reactor operator's capability by real-time computer technology, using the computer to manage multi-channel data, and display that data in the form of higher level information, particularly using the CRT colorgraphic media. And through that method, we will provide an opportunity for knowledge-based behavior.

Thank you.

DR. PLESSET: Thank you.

Are there any particular questions?

MR. MATHIS: I have one. Of the concepts that you have shown here, and in particular this last one, do the operators reactor favorably to that, as compared with your operating scheme?

MR. MEYER: Well, we're introducing a completely new technology into the main control rooms. Operators are just as conservative as the rest of us. They have established a way to live with their bosses and assistants, and if you start perturbing that system you give him a problem. So your reactions vary.

However, I thinK one significant point we overlooked, most of the operators are quite a bit younger than I am. They have absorbed computers in the educational system. The type of operator, for example, that will be using this say five years from now, perhaps is finishing high school and he's using a computer. We've found that at LOFT.

We have found that Tom has three operators who are computer buffs. They have their own home computer. And they've taken to it like a duck to water.

MR. RAY: Were these concepts presented at your recent meeting with the industry?

MR. MEYER: Yes. A very similar presentation was given.

MR. KAUFMAN: I might add one comment, or two. One is that, with the advent of some of the regulations that we're seeing now, in fact we have perturbed what's going to be in the control room. So the driving force for the perturbation is already there.

The only studies that I'm aware of how reactor operators might respond to these sorts of things, there were some psychological studies done associated with the Halden project where they did have people study the reaction of operators, and then try to correlate that with their backgrounds and experience. And they came up with similar kinds of conclusions: That people who had been introduced at a fairly early age, and particularly in high school, to computer technology had very little difficulty in adapting to it.

Those that had come out of fossil plants and out of the paper industry and some of the heavier industries, rather resented the introduction of these kinds of techniques, and in fact did everything possible to ignore them.

MR. MEYER: If you think of your own experience with the interactive terminals, you will get the same sort of guidance we think we should follow with respect to what you're talking about.

The worst problem an analyst has in using a computer is where the computer is forcing him to wait, forcing him to do dull chores like typing out messages. So our planning for future work has to put more emphasis on providing an interactive terminal where the operator can use a language that's instinctive. For example, point to a component if he wants detailed information on it, rather than using the keyboard.

MR. MATHIS: One other question. On your mimic display and so forth, on one of those cases you showed plant conditions such as valves open, closed, pumps operating. How do you get that kind of input to the computer?

MR. MEYER: That's a very good point. A lot of that information right now is not even available on the LOFT facility, which is heavily instrumented and has a lot of the process instrumentation fed to the data acquisition system. On a commercial plant, that problem is even larger.

That is, however, really recognized by the community, and it's one of the first places that the utilities are looking to with respect to estimating the cost and impact of this type of technology.

MR. MATHIS: Well, that's a condition where, unless you have accurate information, up-to-date, you can really go awry; you can go down the wrong path. That is one I would be suspicious of, until you can find absolute means of knowing what your plant configuration is.

MR. MEYER: Well, in this presentation I tried to emphasize the developmental model itself would provide end-user information that could help people write codes and standards for this.

Another part of our problem is -- I think it relates to your point which is very well taken -- that is, we are looking at computer systems which use, say, a simple model for the plant, including the valves, the temperatures, the flows, and the pressures. So that in the event a limit switch on a valve, for example, is stuck, the valve is actually shut when it's indicating "open," the computer can sense this from other information such as the loss of flow, or the incorrect relationship between Delta T and individual pressure readings.

DR. ZUDANS: On this development that you showed, you are, I assume, building on such experience as Halden, and you're not starting from scratch?

MR. MEYER: That's the first thing I did was read all of the Halden reports. I am not ashamed to mention that we copied every useful technique we could find in their program.

DR. ZUDANS: Good for you. Do you have any comments with respect to the German power plant that is installing this system?

MR. MEYER: Well, the German -- if I understand your question, and I'm not as familiar with the German technology as I should be --

DR. ZUDANS: No, no. They're using the Halden system.

MR. MEYER: They're oriented towards the diagnostics and surveillance concept. The concept there is to emphasize fairly sophisticated programs for the computer to sit by itself and do fairly sophisticated analysis of the plant's status, and it's particularly directed towards determining the route cause in case an accident occurs. Our program is a little bit different.

MR. KAUFMAN: But one of the things that we have noted about that effort to develop a diagnostic equipment is exactly the point that was brought up earlier. Many of the diagnostic schemes that are being talked about and are used then as models when people think about implementing some of the regulations and some of the items in the Action Plan are very vulnerable to error propagation, particularly the diagnostic systems that the Germans have are based principlally on fault tree techniques and have a very high rate of error propagation for such things as erroneous valve indicators and that sort of thing.

And I am sure they are aware of that. I have attended presentations where these issues were discussed, and again we have no solution for that. And the concern, of course, is that an operator will use these in the course of responding to some of his more nomally benign or small-break transients, and cause a severe safety problem.

MR. MEYER: I think I can also add that we are trying to avoid helping to develop a situation where the utility operator is stuck with too high a dependence on the computer. We're trying to develop an interactive situation, where the computer is being used -- and I think it's been used in some of the Halden personnel -- the computer becomes a transparent window in the process. It provides a lot of information, but if the window becomes cloudy through their computer failure, that's obvious immediately to the operator.

DR. ZUDANS: And a final question: Can you attach any time scale when this might show up in the commercial power plant?

MR. MEYER: Well, I can only give my personal remarks. I know that this approach has a high level of concensus in the LWR community. There are utilities that we interacted with, particularly on the Technical Advisory Group of the EPRI program, that grabbed on to this concept long before the President's Commission's report came out and prepared similar equipment.

They are now stuck with that equipment, because we are now addressing the question of how do you write code standards and specifications for licensing that type of equipment?

DR. PLESSET: One last question?

DR. CATTON: I was just interested in whether or not you are interacting with the people who are putting together Reg Guide 197?

MR. MEYER: I'm not personally interacting. There may be others in the program who are.

DR. CATTON: Because whatever, as far as I can tell, will be specified in that Reg Guide is going to feed the equipment and systems that you're trying to put together.

MR. MEYER: Well, what I said wasn't quite right. I specifically mentioned Dr. DeSalvo, Operational Safety and Research Branch. He is our primarily link in what we are doing and what the people are doing who are writing 197.

DR. CATTON: Have you looked through 197?

MR. MEYER: I have not personally reviewed it the way I would like to.

DR. PLESSET: Well, we will have one last question from Dr. Wu. I think we've worked over 197 all right.

DR. WU: I will take a minute or so. Do I understand it correctly that you are taking the deterministic approach in taking the diagnosis and also trying to find out the information display for the operator?

MR. MEYER: As I said, I welcome critiques because we're in a new area, and terminology is a big problem. If I understand what you mean by "deterministic" versus the diagnostic approach, I would say we're probably not doing that.

What we are trying to emphasize is using the operators' intelligence and training to do the analyzing, to use the computer to do the dog work, to take data -- which there's a tremendous amount of data available -- and put it in a useful form. In other words, essentially use the computer the way you guys would use an engineering aid.

DR. WU: Well, actually my point is: Suppose the operator feels that he doesn't have that complete input, that there is some value and a plus/minus some error, and they look at the propagation of the error and see if it might lead to different solutions, that type of --

MR. MEYER: Are you referring to instrument error in specific? Instrument error -- how you deal with instrument error, as Nick mentioned in his opening remarks, is something that we are plotting an approach to.

DR. WU: Error being one, and there could be some information not available at that moment. For example, the water level in the pressurizer, and so forth, or the wrong valve.

MR. MEYER: That's an extremely important area, and we feel, again, that we can contribute by identifying the way these types of problems show up. For example, the mimic diagram I showed you had a pressurizer level represented by shading in the lower part of the pressurizer according to that level. In an actual experiment that was run, the pressurizer level dropped down to the point where the instrument tube block in the lower end had reached its lower range. So it continued to feed the same zero level reading to the computer.

However, "zero" for the instrument is not an empty pressurizer. So from that time on, although the operator knew that the pressurizer was empty, there was still a little amount of water shown remaining in the pressurizer. That is the type of thing that really it is hard to identify if you're doing just paper and pencil study.

DR. PLESSET: Well, I think we will have to -- this is, after all, an ECCS Subcommittee and not a control room design subcommittee, and we could work this over somewhere else.

I thank you. It's interesting to hear about this program.

So let's recess until 1:15.

(Whereupon, at 12:07 p.m., the meeting was recessed, to reconvene at 1:15 p.m., this same day.)

- - -

AFTERNOON SESSION

(1:15 p.m.)

DR. PLESSET: I think we can reconvene and go into the first item on our afternoon agenda, "LOFT Experimental Program and Testing Sequence," by Mr. Harvego. Is that the way you pronounce it?

MR. HARVEGO: Yes, sir.

My name is Ed Harvego. The topic of my presentation is the LOFT Experimental Program and Testing Sequence.

(Slide.)

Specifically I will be talking about very briefly the LOFT testing accomplishments, where we've been, and the direction that we're now going in. I will also talk about three new test series which we've developed which we feel will substantially increase the benefits that can be derived from the LOFT experimental program.

I will talk about the LOFT testing sequence, and then the continued planning effort which we have undertaken to ensure that LOFT remains responsive to the needs of the nuclear community.

(Slide.)

This slide shows the test series within the LOFT experimental program for quite some time now. To date, we have concluded a total of 18 experiments. 7 of those were non-nuclear. As alluded to earlier, we have concluded two large non-nuclear break incidents. We feel there is still work to be done in this area, primarily looking at ECC bypass phenomena, and the early rewet that occurred in the two LOFT experiments.

However, LOFT has changing emphasis of this program over the last year, and has concentrated primarily on small breaks and operational transients. To date we have completed five small-break experiments which Dr. Linebarger talked about earlier this morning, and we have completed four operational transients.

Dr. Solbrig talked about that earlier this morning also. Three of those experiments were run over a period of three weeks this last month.

(Slide.)

DR. PLESSET: Let me ask a question to see if I got it on that previous slides on the large-break series, that L2. You said you were going to plan some tests to investigate the "rewet phenomena"?

MR. HARVEGO: We feel there is other work that can be done in that area, yes.

DR. PLESSET: Fine. How would that be done? How would you do that? Have you made any plans yet?

MR. HARVEGO: Primarily looking at different operating conditions -- for example, pump operation -- and how that might affect the fluid conditions; also, looking at different power levels, and ways in which that might affect the fluid conditions that could lead to differences in behavior.

DR. SOLBRIG: Could I make a comment?

DR. PLESSET: Yes. Please.

DR. SOLBRIG: The primary reason for the rewet that we observed is due to the cavitation behavior of the pumps during the 2-1/2 to 5 second time period after a LOCA starts. The pumps, during the pumps-running experiment, they basically had finished cavitating at about 5-1/2 seconds.

Now with the pumps off, with the power turned off, you will get less of that type of behavior than in the pumps-on case. Now if one of the pumps is in a stuck rotor, or a --

DR. PLESSET: Chuck, I was talking about the large-break tests. I thought that's where he -- Are you talking about that, too?

DR. SOLBRIG: Yes.

DR. PLESSET: Oh, good. All right.

DR. SOLBRIG: In the large-break LOCA, the reason we got rewet was because at 2-1/2 seconds at the break, you transition from single-phase to two-phase flow.

DR. PLESSET: This is to simulate a pump coast-down? Or were the pumps shut off?

DR. SOLBRIG: The pumps were operating --

DR. PLESSET: Oh, they kept running?

DR. SOLBRIG: In L2-2 and L2-3, the pumps were left running.

DR. PLESSET: Which is atypical, really, isn't it?

DR. SOLBRIG: Well, it depends upon your assumptions if you continue to have off-site power, it's okay. Anyway, the decrease in flow out the break occurred because two-phase hit the break location at 2-1/2 seconds.

Now the pumps were cavitating, and it took them 5-1/2 seconds to completely cavitate. So they were no longer putting water into the reactor vessel.

Now the difference between the input and the output during this time from 2-1/2 to 5-1/2 seconds caused an upsurge of flow through the reactor core and caused the rewet to occur.

So now what we're looking at is: Under what reasonable conditions would you not have this insurge into the core? Now if you have something like a stuck rotor instead of, you know, the pumps just coasting down, we're looking at the conditions under where we would inhibit -- realistic conditions or reasonable conditions under which we would inhibit this core rewet or insurge into the core.

DR. PLESSET: Now suppose you had loss of off-site power so that the pumps are coasting down? What would you think that would do to this effect?

DR. SOLBRIG: That would decrease that effect considerably. However, we don't think that that alone would be enough to stop it. In other words, we've looked at that calculation already and we still get some excess insurge. So we're looking at other conditions, like assuming that one of the pumps is inoperable, or has a stuck rotor. So we're looking at other conditions where we would basically not have -- to see if there are any conditions under which we would not have this insurge into the core at 2-1/2 seconds.

DR. PLESSET: Let me translate this to a PWR. How would you translate this early rewet to a PWR?

DR. SOLBRIG: Well, basically the calculations that we did on Zion indicated that our results would basically apply, and because of the large pumps that they have in Zion and the fact that the cavitation characteristics we believe are fairly similar to that of the LOFT pumps, we feel that the results we got in L2-2 and L2-3 would apply to a PWR, and they also could be translated to a PWR.

We also feel that by investigating the right conditions in LOFT, this could also be translated to a PWR.

DR. PLESSET: Chuck, suppose I were to ask you about this phenomenon in a 12-foot core, or a 14-foot core? Tell me what would happen then?

DR. SOLBRIG: Yes. The phenomenon seems to be independent of distance of the core. It's not really --

DR. PLESSET: Now this is from a computer analysis?

DR. SOLBRIG: This is from the experimental results at LOFT.

DR. PLESSET: How could you tell what would happen if it was 12-foot?

MR. KAUFMAN: The answer to his question, we did run the computer sensitivies to look at the effect of the core length.

DR. PLESSET: But Chuck was talking about some experimental backup for that.

MR. KAUFMAN: But the upsurge is experimental.

DR. SOLBRIG: Yes, but the upsurge occurs so quickly that all the thermocouples basically are read at about the same time.

DR. PLESSET: But this is for, say, a five-foot core.

DR. SOLBRIG: Five-and-a-half foot. But it's --

DR. PLESSET: I'll give you another half-foot.

DR. SOLBRIG: It's not the reflood phenomenon that you would think about in the normal reflood situation.

DR. PLESSET: No, I understand. But you feel confident that you would get this to the midplane of, say a 14-foot core. There are such in the world.

DR. SOLBRIG: Yes.

DR. PLESSET: That's a foot-and-a-half above the top of your core.

DR. SOLBRIG: Right. I think that, within a second, that certainly within a second the same thing would occur throughout the entire core.

DR. PLESSET: That's based on experimental evidence, or calculated?

DR. SOLBRIG: Well, it's calculation, but I also think that it's verified due to the fact that all of our thermocouples, even in the shorter core, rewet basically at the same time. In other words --

DR. PLESSET: It couldn't have been instantaneous, Chuck, over the whole 5-1/2 foot.

DR. SOLBRIG: I think it was within a tenth of a second.

DR. PLESSET: Let's say ten milliseconds.

(Laughter.)

DR. SOLBRIG: Within a hundred milliseconds.

DR. PLESSET: A hundred milliseconds. All right.

Fine. Thank you. I think there were some questions from --

DR. THEOFANOUS: Just to clarify the point, because I think relates earlier on to what Nick Kaufman mentioned as one of the key conservatisms demonstrated by

LOFT.

I would like to know, Chuck, because you said maybe there is another factor besides pumping that could be causing that. Would you explain that a little bit more?

DR. SOLBRIG: I said that we think the pumps are responsible for that phenomena, but just turning off the pumps, assuming the loss of off-site power, would probably provide still an insurge into the reactor vessel. We are looking at other conditions that might inhibit that insurge into the reactor vessel, such as assuming one of the pumps has a locked rotor or a broken shaft.

DR. THEOFANOUS: This insurge is due to what?

DR. SOLBRIG: The insurge into the reactor vessel is due to the fact that at 2-1/2 seconds the temperature from the core, the higher temperature at the core, has reached the break -- the fluid in the core has reached the break, and we're now in a two-phased flow, so the flow rate decreases considerably. However, the pumps have not cavitated very much yet at 2-1/2 seconds. They are still effectively pumping and putting as much fluid into the vessel as they were previously.

DR. THEOFANOUS: So it's still related to the pumps. I was trying to clarify how much of that is due to the pumps, so that is something else.

DR. PLESSET: Well, he's ascribing it essentially to the pumps.

DR. THEOFANOUS: Oh, okay. And do you think that you will be able to take credit, or you hope to be able to take credit for this phenomenon for actual LOCA calculations?

DR. SOLBRIG: Yes. If we can show that under all conditions, under all reasonable conditions this insurge will occur, I think definitely we should take credit for it.

DR. PLESSET: That's for a core at 102 percent power when this thing starts.

DR. CATTON: Do you see this insurge in Semiscale?

DR. SOLBRIG: Yes, we did. In the Semiscale results, we had powered the rods in such a way as to make sure that we were following the predicted temperature time profile. So that what happened is we pumped excess power into the Semiscale experiments because we were trying to simulate. This is one of the realistic aspects of the LOFT facility.

DR. PLESSET: Well, let me ask you about one other thing. You don't have the right kind of steam generators, and the height relationships are not quite right, or I guess they're pretty far off. Okay? What about steam binding?

DR. SOLBRIG: I really think the first insurge is fairly independent --

DR. PLESSET: Yes, but now I'm past -- I've got this insurge up to the top of a 14-foot core. You told me it would do it. Now I'm going on with this, and I'm getting some drainback, I'm getting a lot of steam being generated in the upper plenum. Tell me what happens then. Do we need this upper plenum test facility? You're telling me "no," I think.

DR. SOLBRIG: The drain -- If we were to prove that we could in fact remove a significant amount of heat under all reasonable conditions --

DR. PLESSET: By this --

DR. SOLBRIG: -- by this phenomena, I would agree with you. If in fact -- because this first insurge really removes a significant amount of stored heat in the--

DR. PLESSET: Oh, we grant that. We just wanted to be sure it would happen for a 14-foot core at 102 percent power, initially, and so on and so forth. And you are pretty optimistic, I guess.

DR. SOLBRIG: Well, it's really too early to tell if we can observe this effect. That is to say, if we can find conditions under which this insurge will not occur. We are in the process of doing the calculations right now to see if we can find that.

So if we determine that in fact under all conditions that this insurge or upsurge will occur, then I would agree with you. My own personal feeling is that there is probably a 50-50 chance that we would find conditions such as that where the upsurge will not occur.

DR. PLESSET: You will find conditions under which it does "not" occur?

DR. SOLBRIG: Yes.

DR. PLESSET: Well, but you've got me confused, now. I thought you would be convinced that it would always occur.

DR. SOLBRIG: It will always occur, we feel, with the pumps running.

DR. PLESSET: Oh.

DR. SOLBRIG: With the pumps turned off, it will also occur to some degree, but not as much as with the pumps running. And I think, however, there are probably some other conditions under which it will not occur, or it will be severely inhibited.

DR. THEOFANOUS: Now I'm confused, because when the pumps are off, how could that happen, when the pumps are off?

DR. SOLBRIG: It takes about 14 to 17 seconds for the pumps to coast down.

DR. THEOFANOUS: So when you say "off," you really mean power to the pumps, not the actual running?

DR. SOLBRIG: Yes.

DR. THEOFANOUS: Oh, okay.

MR. KAUFMAN: Let me address in a little different sense, when we saw the hydraulic phenomenon in L2-2 and L2-3 there was obviously a considerable significance. We set about to say: If there's a condition where we've seen it, can we find a condition where it won't occur? And that is a way of verifying that we truly understand the phenomenon in some detail. So we set about to try to find the conditions via computer calculations under which we could predict that we wouldn't get an insurge.

If we can find such a condition, then the next question is whether there is a reality to that kind of a condition. We put in the test plan an allowance for a test to verify that if we can locate that kind of behavior to run an experiment to try to cause it to exist, so we're in the process of trying to seek out whether there is a worst situation, and if we can find one, then to try to cause it to be --

DR. PLESSET: Yes, Theo?

DR. THEOFANOUS: Couldn't you find out just by analysis, with a good pump model that has in all of the important aspects of pump coastdown, couldn't you find out if the pump was off, the power was off, and if the pump was coasting down, whether still it will be able at this instant in time to move that much fluid through a 12-foot core?

MR. KAUFMAN: We've looked at it enough I think that we probably can say that if the pump is just coasting down as a result of the loss of power, we're okay, we'll still get an insurge.

DR. THEOFANOUS: You still get enough into it?

MR. KAUFMAN: The next question is: If you degrade the coastdown by, for example, seizing the shaft, or some other terrible thing, can you then disturb that insurge?

And then once you find the conditions analytically under which that would happen, does it bear any relationship to reality?

DR. THEOFANOUS: Can you refer me to some calculations that you have done for coasting down which show that you still get this insurge simply by coasting down? Is it in a report, or something like that?

MR. LINEBARGER: It isn't in a report -- Yes. Yes, it is, as a matter of fact.

DR. THEOFANOUS: Which report?

MR. LINEBARGER: It is in our Zion calculations. I believe we have a pump coastdown there. I can't give you a report number, but I will get that for you.

May I amplify a little bit, because --

DR. PLESSET: Well, are we quite finished with that point before we go on?

DR. THEOFANOUS: Yes, if he will give that to me.

DR. PLESSET: Unless Acosta had a question?

DR. ACOSTA: I am a little bit puzzled by the air of mystery surrounding these calculations. Surely these are standard hydraulic calculations in which the pump is operating in its normal four-quadrant mode. What's the mystery?

DR. SOLBRIG: It's the cavitation of the pump which is at issue, and that's not a standard four-quandrant calculation.

DR. ACOSTA: Cavitation performance is a part of pump performance.

DR. PLESSET: You don't need LOFT to do this, is what he's trying to tell you, Chuck. You don't need LOFT for that.

DR. ACOSTA: Well, I must say, this puzzles me. I thought pump performance was well documented, including all aspects of two-phase flow, multipliers and cavitation.

DR. SOLBRIG: No.

DR. ACOSTA: What is mysterious about the cavitation performance of these pumps?

DR. SOLBRIG: The cavitation performance, I guess is not the central issue. The four-quadrant curves only apply to single-phase flow. The two-phase flow models are included on top of that in the current models where we basically assume that there is a cavitation effect as a function of void fraction to the inlet of the pump.

This changes from pump to pump, depending upon the specific speed of the pump, as I understand it.

DR. PLESSET: Be careful, Chuck. He knows a lot about pumps.

(Laughter.)

DR. ACOSTA: You're about to get some other questions, if you're going to say that, because the two-phased performance properly does -- has to reflect all of the cavitational experience. After all, cavitation is really a very low-quality two-phase flow experience.

DR. SOLBRIG: Yes. It is a low-quality two-phase flow experience in general. We're using the word "cavitation" to represent two-phase flow throughout the region from zero to one --

DR. ACOSTA: Yes, you used it; I didn't.

DR. SOLBRIG: Pardon me?

DR. ACOSTA: Do you have these pumps well documented insofar as all of this two-phase work goes, or not? Is that the problem you are addressing here?

DR. SOLBRIG: No --

DR. ACOSTA: Otherwise, people know how to make hydraulic calculations for two-phase flow multipliers, as well as cavitation. I just don't see what -- I don't see why this problem should arise now.

DR. SOLBRIG: Well, we're saying we believe our computations with regard to the two-phase cavitation behavior of the pumps. What we're looking at right now is whether under conditions of stuck rotor, or shaft seizure, whether we will get a different flow rate through the core, and we're doing this analytically. Then after we observe that analytically, then there is still a question with regard to the heat transfer and the rewet characteristics which are open for question as to whether the core will really rewet with the particular minimum flow rates that we will observe through the core.

DR. THEOFANOUS: But I take your answers to mean that it will. Your previous assumption is that here it is, that the calculations have been done, and have indicated that even with pump coasting down, still you predicted very well?

DR. SOLBRIG: Right. That's not pump seizure, though.

DR. ACOSTA: But you have data related to pump seizure already in your two-phase flow maps. There are two-phase flow maps properly done. We have that data, and we can exercise this computation.

DR. SOLBRIG: Yes. We just haven't done the computations.

MR. LINEBARGER: Well, excuse me. We have, Chuck. We really have. We have not documented, but it's something I've been involved in. We have done the pump coastdown, the pump staying on, and all the pumps blocked. In all instances, the core gets some cooling.

However, there's a change in the mechanism between the pumps coasting down and the pumps staying on, the pumps blocked. You get the insurge positively up through the core in the coastdown and the pumps on, in the pumps blocked case, you actually get the reverse flow through the core sufficient to cool the core. You don't get as much cooling influence as you did with the positive core cooling.

What we then did is artifically forced the core in the flow to zero, backed out what the pump characteristics would be in order to induce this type of zero or non-flow at the core inlet.

Now the question is: Is there a situation where you could actually reasonably induce this sort of pump behavior? And we aren't ready to answer that right now.

DR. ACOSTA: The implication is, there are more large-break tests coming because of this? Is that it?

MR. KAUFMAN: The implication is, in looking at the series we've made allowances that there might be.

DR. PLESSET: Yes, Theo?

DR. THEOFANOUS: I guess I don't understand how you would -- suppose you found that condition that you are talking about. I don't know how you would go about making a connection between this particular condition in LOFT and a pump behavior in a reactor? Because you're talking about the partial, now, partial seizure problem, or some kind of partial degradation of the actual shaft movement there. How can you dig anymore into this problem in this fashion? I don't know how it would lead you to something that you can use for actual application.

MR. KAUFMAN: I think you're forgetting that what Ed said is that we had made allowance in the plan to run one more large-break experiment, if we thought it was worth running a large-break experiment. We are not telling you today that we should or shouldn't. We're saying that we're studying the issue, seeing whether there's a need and a value in doing that.

DR. PLESSET: Well, all right --

DR. CATTON: When might that be, that test?

DR. PLESSET: Well, they're not sure they're going to do it. But if they do do it, you would like to know ab ut?

DR. ACOSTA: But when they say "allowance," though, there's got to be a schedule.

MR. KAUFMAN: And when I look at "allowance," we thought out, the earliest would be a year from now, a year-and-a-half from now, it looks like we would have systems that would be capable of coping with it.

DR. PLESSET: Well, we would be also interested, after that test, in a further description of this kind of study that Linebarger and Solbrig have mentioned. That would be of interest to us, right?

DR. THEOFANOUS: Right.

DR. ACOSTA: Very much.

DR. PLESSET: So we find everybody saying "yes," so let's plan on it. Let us know when you're ready.

Okay, sorry to interrupt you so severely, but that was an interesting point.

MR. HARVEGO: I didn't mean to make large breaks such a large part of the presentation.

DR. PLESSET: No, no, they're with us, still.

MR. HARVEGO: Well, in addition to these test series, as I mentioned before, we have added three new tests to the LOFT testing program which we feel would be beneficial.

(Slide.)

These are the L8, the L9, and the L10 series. The L8 series are defined as our "severe core transient experiments." In these experiments, we will start with some initial core uncoveries at relatively low power decays, and gradually progress in severity until ultimately we expect to run experiments where we will get fuel damage.

DR. THEOFANOUS: Could you define that a little more precisely?

MR. HARVEGO: I will be getting into that. Why don't I just outline these, and I will talk specifically about what we're going to do there.

DR. THEOFANOUS: What do you mean by "damage," I want to know. How far do you envision running that?

MR. HARVEGO: We will not run, at this time at least, any experiments that will violate the geometric integrity or envelope of a fuel assembly. We want to be able to remove the fuel assembly intact after that. We don't want pieces of it floating around, or put it in such a condition that it can't be removed as a single unit.

The L9 test series are anticipated transients with multiple failures. In this case we're going to be looking at common-cause, or common-mode failures which either have a high probability of an occurrence, or potentially severe consequences.

And in the L10 series, we have defined as our "override plant protection mode." In this case we will be looking at different override capabilities such as automatic system depressurization, which might be used in commercial reactors to bring them from any given upset condition to a safe shutdown.

There are a number of factors that go into the test sequence, in developing the test sequence.

(Slide.)

As in the past, we are concerned about instrumentation requirements and facility modifications. As we get into the more severe transients, we are going to be concerned about the complexity of the operating requirements, the test severity, and the fuel availability. The fuel availability may be a factor as we get into the fuel-damage experiments.

And then finally, LOFT is a real system. It produces real results, and it has potential real consequences if an experiment is uncontrolled, so in these experiments the safety analysis would play a big role in both the planning of the experiments and the test sequence.

(Slide.)

We do have a detailed test sequence to give to you. It is being copied right now and will be available before the end of this meeting. I don't really plan on going over that anyway, since I think it is pretty self-explanatory. What I would like to do is just go through the various phases of our testing, identifying or kind of characterizing the testing sequences, the phases.

The first phase, which is the phase we're in right now, can generally be characterized as experiments designed to resolve a specific licensing issue. That's the L3-5/L3-6 question of correct pump operations following a small break.

We also plan on running experiments to qualify our system and the code for the more severe transients we will be running. An example of these are the three operational transients that Chuck alluded to earlier this morning, in which, although they were very mild, we feel we're going to get a lot of information from those in terms of the system operation and our ability to predict the phenomenon that occurred.

We will also do our initial core uncovery experiments at relatively lower power decay. In this case, we're evaluating our instrumentation, and particularly our ability to measure liquid levels within the vessel. We will also use these experiments to help us assess the margins in our safety analysis.

And then finally, we do plan at least one experiment where we're going tO look at LOFT typicality. The experiment we have in mind is the simulation of the Arkansas Nuclear-1 cooldown transient which occurred during their startup. So this will give us some information on the abiltiy of LOFT to simulate what could happen during an operational transient in the commercial reactor.

MR. ETHERINGTON: In Item two, who takes the lead? Is it the experimenters who would like to do it? Or is it the code people who know what information they need?

MR. HARVEGO: On this first case, I think there is information to be gained from both to need this particular information. There were some limitations, when we first tried to uncover the core, and obviously our safety analysis is a big factor. We've got to know where the liquid level is.

In addition to that, this initial core uncovery will give us some information on in-core heat transfer, heatup rates, and that type of thing. So this is an experiment that is needed by both the code people and our systems people.

MR. ETHERINGTON: Is there ever a conflict between the two interests?

MR. HARVEGO: Well, it would say that if it becomes a question of facilities, the facility people would have the last say, since we cannot violate certain --

MR. KAUFMAN: I think the answer to your question is "yes." Often and frequently, and in fact that is what really is the process of desk planning, that each of the forces, each of the factions representing their interests and concerns.

MR. ETHERINGTON: I wasn't too happy with the answer. I would think that in this respect the LOFT should be a service facility to the code people.

MR. KAUFMAN: I would like to always be of service, but at the same time I feel I am constrained to the realities of the nuclear facility --

MR. ETHERINGTON: Oh, yes. I don't mean that they should do anything that's dangerous, really.

MR. KAUFMAN: And also constrained by, again, the realities of the hardware. I can only subject it to certain degrees of severity and extremes. But within those kinds of boundaries, yes, indeed, we do try to service the people that are making the requests, and particularly at NRR.

DR. PLESSET: Well, I would be personally less concerned about preserving the integrity of the facility if I get something good out of it, but that's a question of attitude regarding the facility. At a lot of experimental facilities, you do run a lot of risks, and in so doing you're going to get some benefit.

MR. KAUFMAN: Well, certainly there is constantly a process of making that judgment.

DR. PLESSET: Yes.

(Slide.)

MR. HARVEGO: Continuing on in the second phase of the LOFT testing, we plan to get into investigating or doing only experiments that will probably be run to look at the integral effects, the coupled effects of fuel behavior and integral system thermal-hydraulics in a real system such as LOFT.

During this phase, there are a couple of potential large-break experiments that we could run utilizing pressurized fuel. We are considering running one experiment at the fuel pressure at 300 psi, corresponding to gengolite fuel pressures, pressure, and another experiment at 700 psi corresponding to endolite fuel pressures. In these large-break experiments, in the one case we do not expect to get any fuel damage. In the second case, under worst-case hydraulic conditions, we do expect to get fuel damage.

So these experiments will be designed to, first of all, assess the fuel damage criteria, and determine the margins of safety in current fuel rod designs. During this last experiment --

DR. CATTON: Excuse me. Is this the DNBR-type experiment that you're referring to?

MR. HARVEGO: No.

DR. CATTON: No? Okay.

MR. HARVEGO: This is alluding to the worst-case hydraulic conditions that we're looking into.

DR. CATTON: Fine.

MR. HARVEGO: In this last experiment, since we do expect to get some fuel ballooning and rupture, we will have operational the LOFT automatic isotope detection system so we'll be looking at the release, transport, and deposition of fission products both within the primary system and in the blowdown suppression tank. Again, these will be under very realistic conditions of an integral facility such as LOFT.

DR. THEOFANOUS: Can I ask you, what are those phases? Are they chronologically and not overlapping areas of experimentation?

MR. HARVEGO: Yes. What I tried to do is look at the total testing sequence and I characterized the phases and we progress --

DR. THEOFANOUS: But are they chronologically oriented? And are they non-overlapping?

MR. HARVEGO: Yes. Yes. The answer to the question is: They are chronological.

DR. THEOFANOUS: So I take it then we are someplace now between Phase I and Phase II?

MR. HARVEGO: No, we are in Phase I. The Phase I were the initial experiments where we begin to get initial core uncovery.

DR. THEOFANOUS: So when do you plan to initiate Phase II?

MR. HARVEGO: I would say Phase II will be around the end of 1981, the beginning of 1982.

DR. PLESSET: Starting. Was your question answered?

DR. ACOSTA: It would be nice to know what chronological dates go with this.

MR. HARVEGO: The dates are really dependent on a number of things, one of those being the level of funding that we get. But I would say that each of these phases is probably on the order of a year long. I'm assuming we're running something on the order of six or seven tests a year, as we currently are doing.

MR. KAUFMAN: The basic concept is to look at several series at once, and try to do an analysis to try to do an analysis to try to establish the value in those series. We have laid out a tentative test plan to these phases, but we are relucant to say that is "the" phasing, or "the" dates, because in fact as you well know that is a function of how the analysis finally flanges up, what the funding levels are, and, as I mentioned earlier, we've tried to keep the kind of program that as we learn we can add something, or that in fact as we find it has no value we take it away.

In fact, in the last year we have done both. We have added an experiment and deleted one.

MR. HARVEGO: Continuing on, then.

(Slide.)

The third phase of our testing will get us into the series nine testing. These are the multiple failure transients. One of the aggrevating events we would be looking at in these tests is the failure to scram following some given plant upset condition or initiating event. So we do expect these to be relatively severe in nature.

We will also be running some controlled core-damage experiments -- such as a slow heatup, or something like that -- to look at potential ballooning, fuel rod ballooning, and their potential effect on core blockage.

DR. PLESSET: Sullivan isn't here, but maybe Brian knows. I thought there were a lot of separate effects measurements of this last one there -- fuel ballooning, blockage. Is that right?

DR. SHERON: I believe Oak Ridge --

DR. PLESSET: Yes, and at Westinghouse, with NRC support; right? And the Germans. A world-wide effort. What do we need LOFT for in this connection? I'm just trying to learn. I'm not criticizing.

MR. KAUFMAN: What we are suggesting here is that we are examining a test that might add information there. We are working with the thermal fuels behavior program, and indeed we, both within LOFT and thermal fuels behavior, are looking to see if there's a valuable test that we can do that would add information to the issue of fuel ballooning. Again, I don't want to imply that we will, or are going to run all of these tests.

We have tried to identify areas where we might contribute, so that the planning people can analyze and decide whether there is value for us to run that kind of experiment.

MR. HARVEGO: When we talk about these particular types of experiments, in many cases we're only talking about one experiment, or two experiments, if they're necessary. The possible necessity for using LOFT would fall into, again, this coupled effects, understanding fuel behavior and the feedback effects it might have on thermal-hydraulics in an integral system.

We will utilize Semiscale and PBF to help us in this area. But again, since those are separate-effects test experiments, we are looking to LOFT as a potential combining or integral effects experiment.

MR. KAUFMAN: And particularly in this regard, we did have a meeting scheduled with NRR to seek their advice as to whether this kind of thing would be worthwhile. Unfortunately, it had to be deferred --

DR. PLESSET: Well, I think you might get some hints in that direction from us, if you want them, but let me not prejudge the situation.

Theo?

DR. THEOFANOUS: I believe in this area LOFT can help.

DR. PLESSET: You do?

DR. THEOFANOUS: Yes.

DR. PLESSET: How?

DR. THEOFANOUS: I think that in these kinds of experiments with a system the size of LOFT, and with actual fuel, it will provide very useful and I think very much needed information which other separate effects tests have only I guess with sufficient extrapolations and interpolations between different tests --

DR. PLESSET: You think this will give you stuff that the SLAG core test facility won't give you?

DR. THEOFANOUS: I like to think in terms of the actual fuel. I like to think in terms of the size of the system and the system feedback. There are industry questions concerning, for example, the distribution of the ballooning. Is it going to be all in one place? Is it going to be distributed? How is it going to be distributed?

Well, we have some answers to those things --

DR. PLESSET: You think the nuclear aspect of the fuel is important for that?

DR. THEOFANOUS: I think so, yes.

DR. PLESSET: Why?

DR. THEOFANOUS: I can tell you why, too, in case you're interested. The gap between the fuel and the cladding is something not very well understood and not very well defined, and how the fuel itself responds under accident conditions -- is it giving out fission gases and so on. Maybe it will crack. All those matters are important.

DR. PLESSET: I understand that, Theo. But do you think the overall system interaction is important for this kind of detail?

DR. THEOFANOUS: I think from the point of view of the size of the bundle.

DR. PLESSET: You want a big bundle.

DR. THEOFANOUS: Yes. And we don't have that.

DR. PLESSET: Well, that's true, but -- Well, anyway, it's interesting speculation.

Did you want to make a comment?

DR. CATTON: I was just going to comment that there's the simulated fuel pins that the ~~ermans are working with that have gap conductants --

DR. THEOFANOUS: But they are simulated.

DR. CATTON: I believe it's completed UO2 with the heater in the center. The only thing it simulates is the method of heating, so you miss the fission gas.

DR. THEOFANOUS: Not only that, but also you --

DR. CATTON: I thought the question of full-length ballooning was pretty well resolved?

DR. PLESSET: There's so much available. Well, anyway, there is some division of opinion here. Why don't you go on.

MR. HARVEGO: At this point in our testing program, we feel we will have quite a lot of information to digest --

(Slide.)

-- so the fourth phase of our testing would involve going back and picking up a number of tests that we feel have a relatively high priority, but at this point haven't been scheduled because of more immediate needs. Therefore, during Phase IV, we plan on going back and looking at the L4 test series, which are the alternative ECC injection concepts, looking at the efficiency of existing ECC injection systems, and also potential new ECC injection concepts. During this period, we would also complete the steam generator tube rupture experiments. Right now, these are planned as large-break experiments in which the steam generator tube rupture occurs just prior to reflood, and the potential effect of binding on the characterics would be looked at.

After finishing this part of the experiment, then we would continue on with the severe core damage experiments, and we would also look at the override capabilities.

(Slide.)

This would be the last phase of our testing. We will look at override capabilities, progress to the severest of the core damage experiments that we've run to this time, and then, because we would expect to get quite a lot of fission product release in these experiments, we would look at things like boundary conditions for investigating future investigations into the containment integrity. And, as Nick Kaufman alluded to earlier, we expect to learn a lot about requirements and conditions for facility cleanup.

This test series that we've proposed we feel is fairly progressive, but it also has some drawbacks in that in many respects it is irreversible. The facility cleanup problems become more difficult as we progress in the severity of transients, and also ensuring the instrumentation integrity becomes more difficult. Therefore, we want to be sure we are progressing in a logical sequence and not leaving anything out as we progress through these testing modes.

Therefore, we do have a continued planning effort underway.

(Slide.)

Basically what this involves is, first of all, identifying the testing needs as we gain more experience; reassessing our current test plan; and then modifying the test plan to reflect the additional testing needs.

(Slide.)

As an example of what I am talking about, we have taken an initial look at potential alternate ECC injection tests that might be run in series 4. Out of these we have selected four of them which we feel have a relatively high priority. These four (indicating) currently will be the first two on our testing sequence. Therefore, we feel we need to go back to reassess the tests that we've defined, look for weaknesses in the schedule, and if possible replace other tests with these two additional tests; or possibly add these tests, combine these tests with another test such as we did for the L3-5 and L3-5A test. Or, if necessary, simply add these to the testing sequence.

(Slide.)

I mentioned identifying user needs as an important part of the planning. That's just really one of the aspects that we have to consider.

In addition to identifying the user needs, we also have to consider various user interest levels. The importance of the tests, so that we can prioritize the testing sequence. We also have to define the LOFT testing capabilities, and then try to match the LOFT capabilities with the needs of the users. We feel we understand the capabilities of LOFT, and in fact are expanding those capabilities.

(Slide.)

The areas where we have difficulty are in identifying users and what their needs are, and then matching these needs to the particular LOFT testing capabilities.

One of the approaches we have taken is shown here. I recognize that this is a little difficult to read, but it is in the handout and I would just like to show you what it is and give you some indication of how it might be used.

This interest matrix basically made up of -- across the top here we've identified potential users of LOFT. Along the side we have identified what we feel are the LOFT testing capabilities. Basically these consist of understanding the cause and consequences of plant upsets. To these, we've also added LOFT as an off-normal operator training facility, and also as a potential equipment qualification facility.

Down in the lower right-hand corner we have identified various interest levels, as "strong," "high" and "medium," with "strong" being the red circles. When we filled this chart out -- so it would be subjective in nature and it would be different depending on who filled it out, but we feel there is something to be gained from this.

For example, we believe that we are responsive to the needs of the NRC in terms of their safety concerns, as indicated by the large number of red dots under the NRC column.

Another thing that might be derived from this chart is the fact that user interests have appeared to be shifting from the consequences of large breaks to the causes of a variety of upset conditions that might lead to another TMI. That's indicated by the red circles and the blue squares following the human errors and component failures as a potential cause for a variety of upset conditions.

Another approach similar to this that we have taken is to develop a similar chart for identifying users and trying to match up the user needs with specific tests. We are in the process of computerizing this. In this case, we have gone to specific documentation and looking at I&E bulletins, NUREG reports, vendor reports, ACRS transcripts, trying to identify specific needs that each of the tests would meet.

We believe this is going to be very helpful to us in, first of all, identifying weaknesses in our testing sequence, and also as we progress into the detailed planning of the tests, to ensure that we do in fact run the experiment that would provide the most benefit to the nuclear community.

(Slide.)

Now in the way of planning for the multiple-failure transients, we are using an event tree approach. Basically this consists of going to Reg Guide 1.70, Chapter 15, looking at all of those transients, and grouping those transients into transients which exhibit similar behavior.

We are also looking for transients that exhibit unique behavior, and then combining all of these and then coming up with a limited number of recommended transients that might be investigated in LOFT that would cover a full range of potential operating conditions or plant responses.

We have made some progress in that area.

(Slide.)

This slide shows some transients that we have selected for initial analysis. These transients were selected basically because of their potential high risk -- "high risk" being the risk being defined as "probability times consequence."

We have selected a total of nine experiments here, or initiating events. We have initially looked at the loss of all AC power, and we are now beginning to look at the inadvertent opening of the steam generator valve and uncontrolled rod withdrawal.

We plan on doing some detailed event trees. From these event trees, to perform calculations of specific event sequences to try and quantify the results in terms of magnitude and timing. From these, then, we would select the tests that we would propose to run in the LOFT facility.

(Slide.)

So in conclusion, we feel that the testing sequence that we are developing, or continuing to develop, is being optimized both from the standpoint of facility utilization and from the standpoint of addressing the needs of the various users.

Our current program we feel, in terms of the tests we have selected, will exploit the uniqueness of LOFT. That is a major objective. We will also maximize the usefulness of the data that is obtained from LOFT.

However, as I mentioned, because the testing sequence that we're proposing is irreversible in many respects as we get the most severe transients, we have a continued planning effort underway to systematically address the various needs of the users, and to be sure that LOFT does remain responsive to the needs of the nuclear community.

That concludes my presentation.

DR. PLESSET: Fine. Thank you very much. Don't go away.

You said you had a more detailed listing of the scheduled test series.

MR. HARVEGO: Yes.

DR. PLESSET: Could we get that?

MR. HARVEGO: Yes.

(Distributes document.)

Basically this goes through identifying specific tests that we would like to run.

DR. PLESSET: And it gives a sequence time?

MR. HARVEGO: It gives the sequence, and it also gives some specifics as to the test objectives and what they are. I believe it's fairly self-explanatory.

DR. PLESSET: Yes. Okay.

MR. HARVEGO: And in certain cases where we question the need for an experiment, we have also identified that on the testing sequence.

DR. PLESSET: Are there any questions?

Yes?

DR. ZUDANS: I got more time to think about this water insurgence. Can I ask a question on that matter?

DR. PLESSET: Sure. Yes, if you want.

DR. ZUDANS: It was stated that if the pumps go down all around, then you do observe insurgence. It was also stated that if you block the pumps, then you have a reverse flow. When you said "block the pumps," did you mean stopping them from running? Or blocking the flow passage?

MR. LINEBARGER: Stopping the rpms. Seize the pumps.

DR. ZUDANS: Not the flow passage. Seize the pumps?

MR. LINEBARGER: Seize the pumps.

DR. ZUDANS: So that means that there is definitely a state --

MR. LINEBARGER: Sir?

DR. ZUDANS: That there is therefore definitely a state where there's a zero flow through the core, because you looked at the two extremes; right?

MR. LINEBARGER: Yes, sir.

DR. ZUDANS: If you could reverse the flow, then that means there is a zero-flow state.

MR. LINEBARGER: Yes, sir. That's why we did --

DR. ZUDANS: And you want to find out at which flow rate, and whether that's physically conceivable.

MR. LINEBARGER: Yes, sir. And then we backed out the pump calculations that go with that. It looks as if the pumps are running about half-speed, as if, let's say you seize two and two are running. Whether that's probable or not, I don't know.

DR. ZUDANS: That means that if you had an early pump trip, that they started coasting down and you scrammed later, you would be in that situation.

MR. LINEBARGER: Perhaps.

DR. ZUDANS: I think it's an interesting question and probably important.

DR. PLESSET: Well, we already agreed, I think, Xenon, that we were going to get more lengthy discussion of just these points, as soon as they're ready.

Are there any other questions of Ed before we let him go? It was a very good presentation, and we thank him.

(No response.)

DR. PLESSET: If not, then we will go on. I think that we have some remarks by Harold Sullivan and Brian Sheron. I will let them decide which order they want to go in.

DR. SHERON: He's first.

DR. PLESSET: Oh, Sullivan's first?

(Laughter.)

DR. PLESSET: He's a little shorter, so you can tell him, Brian, I guess.

(Laughter.)

(Slide.)

DR. SULLIVAN: This is just sort of program. I don't think there's anything really new about the things that I'm going to tell you.

You know that the operational transients have been a part of the Research program. Before TMI occurred, the high priority item was large-break LOCA. Not only was it a high-priority item, it dominated the whole research effort.

Right after TMI occurred, we changed directions to look at small breaks, and now operational transients. We think it was a significant move to move in that direction. We are moving in a direction that the accidents are much more probable. We are using risk assessment and probablistic to look at the scenarios that we're going through.

You have heard both Semiscale and LOFT tell you about the small breaks and the operational transients that are going to be performed in the future.

LOFT has completed four operational transients and Chuck told you about those today.

The Semiscale facility has completed a station blackout, and probably the most severe case of the station blackout in which you let the fluid boil all the way down into the core, you get a heatup, and then they were going to recover and they're planning on repeating that.

The NRR staff, we have the input from them, and they support our small-break and transient research efforts. What we are trying to do is provide a data base for the code assessment process. We are going in that direction. It is probably going to be a little bit different than you've seen us operate in the past. We have done a whole bunch of parameter studies on the large breaks, such as looking at a range of power.

These transients will probably be more in looking at a station blackout, a loss of feedwater, a complication to the loss of feedwater, so they probably won't be the case of looking at a number of small changes, incremental changes; they will be much wider cases of looking at cases of transients.

We are looking at the probabilistic aspects of the accident to help guide us in the areas that we intend to go in the future.

So I think as you see our programs in the future, they will be a much wider spread of transients that we are looking at.

DR. PLESSET: Thank you, Harold.

While I don't feel that I know particularly in real depth regarding a program of Semiscale and LOFT, I feel I do know more about that than I do about the code development and the code assessment program. That is behind seven veils, or maybe more, as far as I can tell.

(Laughter.)

DR. PLESSET: And I don't expect you to take those veils aside right now, but I think that is one area where we need to get a little better picture of what is happening and why, because it has been some time, and some millions and millions of dollars. And it would be nice to know if we are making any progress. That is one thing that I thought I would mention to you.

There are other areas, but this is one that comes up very obviously as an important part of the Research program that we are supposed to review, actually it's twice a year. It doesn't change that much, but --

(Laughter.)

DR. PLESSET: -- any help you can give us on this would be appreciated.

DR. SULLIVAN: I think probably what we should do is just plan to have a subcommittee meeting in which we address the whole code area. It has gotten more complicated in the fact that we are developing transient codes now also instead of the larger codes.

DR. PLESSET: In December? I guess it should be well before the end of December, really. Can you do that?

DR. SULLIVAN: I will take the message back, but I think we can.

DR. PLESSET: Well, Andy will talk with you, but that would be a help to have that.

Who knows about the 3-D program? That is not a small item, and I must confess that -- and I'm just bringing it up now because you're there standing up, and I can get you on the record.

DR. SULLIVAN: Right. Again, I think we probably ought to address that. The 3-D program, and also the code development program are pretty large items.

DR. PLESSET: Well, that doesn't make them less important, I would think.

DR. SULLIVAN: No, the only thing I was indicating was that probably I couldn't address them now.

DR. PLESSET: No, I didn't expect you to. I was just kind of preparing you for the future.

(Laughter.)

DR. PLESSET: You have some other messages to carry back, too, and I thought I would just add that to the list.

DR. SULLIVAN: Yes, we have quite a list from yesterday.

DR. PLESSET: I know you have a good memory, and we rely on you.

Was there anything else that you wanted to tell us?

DR. SULLIVAN: I would just, if there are any questions I would be happy to answer them. Basically, it's the 3D program and the code development. I think you have been through the fuels area in Washington in some detail.

DR. PLESSET: No, I don't want to get into that. Let somebody else worry about that. Just the things I mentioned.

Oh, yes, TLTA. We mentioned that yesterday.

DR. SULLIVAN: Yes. We owe you a response to the three items that we're considering.

DR. PLESSET: Yes.

DR. SULLIVAN: And I think by December we would probably be able --

DR. PLESSET: That would be very helpful. I might mention that I think our November meeting is where they will have a chance to see the Brown's Ferry scram mockup, if anybody is interested. Isn't that right? Did we arrange that yet?

(Pause.)

We have a hope that in connection with our two-day December meeting that you're going to come to, aside from the meeting but at the same time -- that will be in San Jose -- that some of the members and/or consultants might be interested to see the mockup of the BWR scram system about which there has been a lot of talk. I thought I would mention that.

DR. CATTON: What were the dates for that meeting?

DR. PLESSET: December 10th and 11th.

Well, anyway, thank you, Harold.

DR. SULLIVAN: I will turn it over to my colleague.

DR. CATTON: I have one question, Harold, before you go.

DR. PLESSET: Sorry.

DR. CATTON: Having heard about LOFT and Semiscale, what aspects of Semiscale don't properly simulate a nuclear system in a way one cannot circumvent through analysis?

DR. SULLIVAN: If we are going to run transients in which the system returns to power -- you know, even though the rods are in -- there are things like overcooling transients. Some of those are severe enough to --

DR. CATTON: Up to where you don't get scrams, and so forth.

DR. SULLIVAN: You know, we don't --

DR. PLESSET: Overcooling could do it, really -- alone, could do it, to return to power. Sufficient overcooling.

DR. SULLIVAN: Right.

DR. CATTON: Well, but that's not a high amount of power. They could simulate that electrically.

DR. PLESSET: Oh, yes. I didn't mean to disturb your question. I was just explaining his comment.

DR. SULLIVAN: Those kinds of transients would be very hard for us to get the nuclear feedback effect.

DR. CATTON: You couldn't do it through a computer and control the heating?

DR. SULLIVAN: We could, but you don't know if you've got the physics right. And that's the same problem that the BWR facility has, that all of the transients that are overpower transients, there is a rather large void feedback, and we can sense the things like temperature and try to derive some parameters to feed to a computer in and put it back in, even. That seems to be possible -- it's very difficult, but possible. But you don't know that you're getting the physics, all the reactivity feedback correctly.

So any of those transients that have excursions to power are very, very difficult.

DR. CATTON: What percentage of the kinds of runs one might want to make fall in that category?

DR. SULLIVAN: I wouldn't --

DR. PLESSET: Did you want to say something?

MR. LYONS: Yes, just a little bit of amplification. Any kind of a transient in which there is a significant change in the axial neutron distribution, we just can't cover.

DR. CATTON: Oh, I understood Harold's answer.

DR. PLESSET: Yes. If we were to try to get you to do it by computer, it would be very difficult to --

DR. CATTON: I can understand the problem. I thought we understood all the nuclear physics we needed to.

DR. SULLIVAN: Well, I'm a nuclear engineer; I'm not a physicist. They tell me that the void feedback, the temperature feedback coefficients are known to some degree of accuracy, and that they can calculate the power distribution in an area.

DR. CATTON: I guess that would bring up the next question, which would be: Just really how well do we need to know these things? Do you need the third decimal point for a particular transient? Or is it sufficient just to find a trend? And that gets you into cost-effectiveness. Do you really want to run LOFT to do an experiment like that when you can get 90 percent of the answer out of Semiscale with some manipulation?

DR. SULLIVAN: The LOFT and Semiscale programs have been an integral package.

DR. CATTON: I understand.

DR. SULLIVAN: And we had always planned on them being an integral package. If you were going to try and separate them out, somehow, we just haven't done that. I think it would be very difficult to try and figure out all the related funding issues --

DR. CATTON: Oh, certainly. I'm not suggesting that you do it. I was just trying to understand.

DR. SULLIVAN: Certainly anything that has to do with the fuel, Semiscale --

DR. CATTON: Yes.

DR. SULLIVAN: And there are some fuel-related problems. One of the -- I guess you went through the pressurized fuel. That is one of the major things that seem to be left from the physics community now: That they feel like that a bundle test with pressurized rods, and simulating the conditions very well, is one of the things that they would like to see done very much. And I don't think there's a better facility to do that in than LOFT; because you don't have to simulate anything, you get the right --

DR. CATTON: Oh, certainly. Certainly.

DR. SULLIVAN: One of the things I was told is that the test facilities that are available to do those fuel problems that are left, that you can do them in electrically heated facilities, but there you can measure the conditions very well. The ones that are nuclear heated, you can't. And LOFT certainly has the instrumentation, by far.

DR. CATTON: Okay. Thank you.

DR. PLESSET: Well, thank you, Harold.

DR. SULLIVAN: Would you like to add anything?

MR. KAUFMAN: No, I think I answered the same question earlier this morning.

DR. CATTON: It was just phrased differently.

(Laughter.)

DR. PLESSET: Brian?

DR. SHERON: I'm not real sure.

DR. PLESSET: Well, just give us a rousing sendoff.

(Laughter.)

DR. PLESSET: Take the microphone, Brian, if you're going to stay there.

DR. SHERON: I will have to. I don't have any viewgraphs for my closing remarks.

DR. PLESSET: That's fine.

DR. SHERON: I would only want to point out, I guess, that right now we are planning on putting together a comprehensive user need letter for the LOFT program from NRR. Granted, it hasn't been initiated yet; we just finished up on Semiscale, and we haven't had time to do it.

DR. PLESSET: When do you think you might have it? Don't be optimistic.

DR. SHERON: Our plan was to have it by mid-February. I realize that may be a little late for providing any input to your report.

DR. PLESSET: Oh, that's all right. Nobody reads that report anyway, Brian, as far as I can tell.

DR. CATTON: Not in this country.

DR. PLESSET: Not in this country, yes; I should qualify that.

DR. SHERON: I think --

DR. PLESSET: Would you like to go over it with the ACRS subcommittee?

DR. SHERON: The proposed user need? I would have no objection.

DR. PLESSET: You don't sound happy about it, either.

DR. SHERON: Well, you know, as you know there is a letter with the Commissioners right now with respect to this selection of -- the start of a panel to evaluate the cost/benefit of LOFT information. I don't know what the status is of that, and I don't quite honestly know how that will factor into our plans in the sense that NRR identified a user need letter which obviously had tests which took LOFT well out into 1985 or so, that that may be totally inconsistent with what ultimately results from these panels, if they are indeed formed.

So I don't really know how that intends to be worked right now. What I could say I guess is that at this time we do certainly encourage and support the shift in priority in the LOFT program from solely a large-break facility to something which looks at the more probable events that occur in nuclear plants, like the small breaks and the anticipated transients.

We do have some concerns regarding prototypicality of LOFT -- although you are going to have that with any facility that is less than full scale. So it is certainly not something that you should fault anyone with less-than-a full-scale facility on. Hopefully these concerns can be resolved through further analysis or experiments.

With respect to some of the uses of LOFT in the licensing process as I would envision it -- and these are, I guess, my own thoughts at this time -- we are coming across, as we get more and more into the review of operating guidelines and operating procedures for accident events, with respect to what the operator is expected to do, or required to do, and I think in many circumstances obviously testing may not be necessary. It may be a clearcut action that we feel could be competently handled by just a confirmatory analysis and computer code.

There are others, however, that may be very amenable to testing in a facility such as LOFT or Semiscale. These are coming up as part of our ongoing reviews of transient accident guidelines and procedures. Examples are -- at least I have one -- is the steam generator tube rupture, probably the most difficult transient we've identified yet with respect to the burden put on the operator to control the plant. Almost all transients that one classically looks at in Chapter 15 puts the most burden on the operator in terms of controlling the plant and trying to bring it to a safe shutdown.

We have identified a number of questions on what is the most optimum way to bring the plant to a safe shutdown, which I'm not really sure can be answered properly by anaylsis alone. We will certainly be investigating the influence of LOFT to help us in that area.

There may be others. I don't have any examples right now in this area. I should point out that right now our present direction is that we are not trying to key guidelines or procedures for operator action to any specific computer analysis -- from the standpoint, we are not saying that because a computer code says you reach the edge of a cliff at 27 minutes, therefore the procedure would say 27 minutes push this button. We are certainly not putting that kind of reliance on it. As a matter of fact, we are -- at least I envision that we are confirming the acceptability of operating procedures and guidelines with analysis, rather than letting the analysis drive the development of the guidelines.

So from this standpoint, our emphasis really would be to confirm with codes, and then perhaps with subsequent experiments.

Then multiple-failure testing I think can be useful. Again, I would just caution that it is something that has to be well-thought-out. We are having great difficulty in terms of how to treat multiple failures. There is an item in the Task Action Plan which is supposed to recommend improvements in single-failure criteria. I don't know who is doing it, or what progress is being made.

It is also very difficult to look at multiple failures due to the permutations alone are staggering. Right now our approach is not to try and identify specific multiple failure sequences, but rather to train the operator to restore functional requirements within the plant; and, primarily, regardless of what happens, keep the core covered, remove decay heat. And then secondary items are like pressure control, inventory control, reactivity control, and the like.

From this standpoint, I think this approach would have to be factored into any multiple-failure type of planning. Also, the need for looking at a specific multiple-failure event would have to be identified from the standpoint of, is it being done just as a multiple failure? Or is there some aspect of the computer code that should be checked out? Is there a concern?

From that standpoint, I guess we would say we would be working closely with Research, and with EG&G, to develop this user need letter. I would anticipate that we would use the summary, the preliminary test sequence as a starting point, and basically compare what we believe our needs are, and ship that over when we are pretty well happy with what we have.

DR. PLESSET: Fine. Well, I think it isn't necessary for me to try to summarize any more than has already been done.

In some respects, as I said before, we certainly profit by coming here. It is an important center for reactor safety research and reactor safety experimentation. I think that one very positive thing I would like to mention is

Brian's document regarding pumps-off/pumps-on and the small-break LOCA. I think that that was a very helpful thing for us to go through, and I appreciate what he's done, and I think he deserves a lot of credit for it. I don't know if you're going to get it, but let me give it to you for this little bit here.

Now as regards the review of the LOFT tests program, we -- the situation right now I think has a lot of uncertain elements in it. This review panel that you mentioned doesn't seem to have been formulated yet. One problem is that the Commissioners need it by the end of this calendar year for it to be of any significant input.

Now maybe they won't have a panel; maybe they'll get one. There are a lot of old faces around that they can dust off and bring in that are more or less up to speed and they could do it without even coming to Washington, having formed their opinions in the past.

But this doesn't necessarily mean that what this panel says, or what the Commissioners say, has any connection with what is going to happen. We have to keep that in mind, because it goes up that Hill, and sometimes things that go up the Hill never come down. That's defying the law of gravity.

(Laughter.)

DR. PLESSET: So that's one remark I would make. Our report to Congress will be completed in beginning February.

Aside from that, I think we have learned a lot at this meeting from the people here. I think that we also are indebted to Harold Sullivan, in addition to Brian, for helping us in our thinking, and packing a lot of messages on his back to take back to Washington.

As I said, we look forward to further meetings with both Brian and Harold, and also with the EG&G people.

With that note, I think that I can adjourn this subcommittee meeting. If you have any complaints or comments, you can give them to me privately, and we will adjourn.

(Whereupon, at 2:33 p.m., the meeting was adjourned.)

* * *

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

ACRS Subcommittee on ECCS

in the matter of: LOFT and Semiscale, Idaho Falls, Idaho

Date of Proceeding: October 23, 1980

Docket Number: __________

Place of Proceeding: Idaho Falls, Idaho

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

JANE W. BEACH

Official Reporter (Typed)

Jane W. Beach

Official Reporter (Signature)

LOFT OVERVIEW

EG&G Idaho, Inc.

POOR ORIGINAL

Kaufman 1

Primary Program Emphasis

- Create an experimental data base reflecting a wide spectrum of accident phenomena and plant states
- Use and evaluate methods for recognition, control, and recovery from accident phenomena

INEL-S-26 739

LOFT Mission

Establish conditions in a nuclear reactor characteristic of accidents postulated for an LPWR to test and develop methods for analytical description, for accident recognition, and for manual and automatic plant stabilization and recovery.

INEL-S-26 136

History — LOFT Program

March 1976 —First of six non-nuclear LOCA tests
September 1977 —Initial fuel loading
August 1978 —Full power operation
December 1978 —First large break LOCA test
March 1979 —Accident at TMI
May 1979 —Second large break LOCA
May 1979 —First non-nuclear small break test
November 1979 —First nuclear small-break test
June 1980 —First operational transient test

INEL-S-28 456

Data Base is Important to Implementation of NRC Action Items

- Emergency training of shift technical advisors and operators (I.A.1.1. I.A.2.1)
- Analysis of small break LOCA inadequate cooling (I.C.1)
- Characterization of coolant inventory and natural circulation (I.C.1)

Data Base is Important (cont'd)

- Emergency procedure upgrade. NSSS vendor review, NRC review (I.C.5. I.C.7. I.C.8)
- Training for mitigation of core damage (II.B.4)
- Development of instruments for accident monitoring, determination of inadequate core cooling (II.F.1. II.F.2)

Data Base is Important (cont'd)

- Location and needed character of coolant system vents (II.B.1)
- Development and appropriate location of post accident sampling system (II.B.3)
- Design of auxiliary feedwater initiation and indication system (II.E.1)
- Degraded core rulemaking (II.B.8)

Other Important Data Base Uses

- Resolve specific NRC concerns (pump on, off)
- Develop analytical methods that characterize plant accident response
- Boundary conditions and perspective to assess separate effects and nonnuclear tests

INEL-S-29 031

Operational Methods Effort Important to Implementation of NRC Action Items

- Control room staffing requirements (I.A.1.3)
- Emergency procedure upgrade. NSSS vendor review, NRC review (I.C.5, I.C.7, I.C.8)
- Establish upgrade plans for control rooms and NRC audit of plans (I.D.1)

9

Operational Methods Effort

(cont'd)

- Training for core damage mitigation (II.B.4)
- Develop instruments for monitoring accidents and inadequate core cooling (II.F.1. II.F.2)
- Developing and upgrading emergency support facilities (III.A.1.2)

INEL-S-29 034

Other Important Operational Methods Uses

- Develop methods for accident recognition and effective responses
- Develop methods for real time information validation during accidents
- Develop preferred courses of accident response

INEL-S-29 032

Associated Mini-Programs

- Development and commercialization of instruments to identify and measure accident environments
- Development of equipment and techniques for post-accident cleanup and reentry
- Development of equipment and methods for snubber and relief valve testing
- Routine field application of automated ultrasonic testing

INEL-S-26 742

Summary

- The LOFT reactor plant has been repeatedly placed into conditions characteristic of accidents postulated for LPWRs and the plant has been successfully stabilized and recovered
- Operators, plant equipment, and emergency systems have performed well
- New instruments, operational methods, and analytical techniques are being developed and tested
- Data obtained have shown some significant conservatisms in calculations and assumptions used in LPWR licensing process

INEL-S-26 135

POOR ORIGINAL

RESULTS OF THE LOFT ANTICIPATED TRANSIENTS EXPERIMENTS

By

C. W. Solbrig

LOFT has recently completed four anticipated transient experiments. These experiments include (1) loss of load, (2) loss of flow, (3) excessive load increase, and (4) loss of feedwater. Each experiment was successfully completed and is briefly described in this presentation. The first part of this presentation describes why anticipated transient experiments are useful, and the second part describes the experimental results.

The anticipated transient experiments were performed primarily to provide a basis for calibrating the computer codes used to predict these types of transients. After the models in these codes are improved enough to describe these transients, they may then be used for predicting the course of anticipated transients with multiple failures, for example, ATWS, experiments to be performed in LOFT. The tests were non-trivial because several important phenomena were not predicted correctly in magnitude or time. These experiments will allow safety analysis report models for these type of transients to be evaluated. Anticipated transients are of interest because they are expected to occur in a power about once per year. The Three Mile Island incident has provided the need for increased simulator capability and which can only be met in the future with computer codes such as the RELAP5 and the RETRAN computer codes. These computer codes will have to represent operational transients, anticipated transients with multiple failures, small break and large break LOCAs and all transients in between. In order to accomplish this, all aspects of the plant must be represented including secondary side models, pressurizer heater and sprays and post-accident heat removal systems. Some of the questions which will have to be answered by these codes include determination of the correct operating procedure for a given situation, verification of current procedures, tech spec changes and information required in training programs for both operators and technical advisors in power plants.

LOFT is uniquely qualified to perform such experiments because it has most of the systems representative of a large nuclear plant. Small electrically heated systems usually do not represent multiple ECC trains, secondary side components and single failure proof components. In addition, small systems have large relative heat losses. Experiments which are performed in actual nuclear plants can be very helpful but the amount of information or instrumentation available in such a plant is usually insufficient for code verification. In addition, experiments performed in powerplants are usually not very severe and, therefore, do not test all aspects of the code adequately.

The dominant phenomena in these transients are related to primary coolant system (PCS) pressure response and the availability of a heat sink. The mass of the PCS is initially unchanging or increasing. The average temperature of the PCS results from the overall energy balance and determines the average specific volume. Changes in average specific volume determine pressurizer level which in conjunction with the automatic pressure control systems determines the PCS pressure.

The course of each transient was predicted prior to the experiments with the RETRAN computer code. Evaluation of the experimental results indicates the LOFT system response to these transients is not severe and that the LOFT automatic pressure and level control systems can effectively deal with the challenges issued by these transients. At all times during the experiments core cooling was sufficient to maintain the fuel rod cladding temperatures below the saturation temperature of the coolant. The operators were able to understand the course of the transients and respond appropriately in real time to return the plant to a stable controlled situation. Comparison of the experimental results with the RETRAN calculations revealed the major phenomena were predicted in the proper sequence, however, the magnitudes of some phenomena were not precisely calculated. Further analysis has shown the differences between the calculations and the data to come from the following sources: (1) steam generator secondary side feedwater and steaming flow rates, (2) pressurizer spray and heater operation, (3) thermal nonequilibrium between the pressurizer vapor and liquid during insurges and outsurges, and (4) main steam control valve leakage.

In summary, LOFT experiments have provided information useful to the understanding of anticipated transient behavior. The ability of the plant automatic control systems and the operators to recover the plant during transients not compounded by additional failures has been observed to be satisfactory in LOFT. Comparison of currently used analytical methods with the LOFT results has shown a generally good transient characterization with areas for improvement noted.

REFERENCES

1. R. P. Jordan, LOFT Experiment Operating Specification Non-LOCE Baseline Test Series L6, Rev. 1, October 5, 1980.

2. C. D. Keeler, Best Estimate Prediction for LOFT Nuclear Experiments L6-1, L6-2, L6-3, and L6-5, EGG-LOFT-5161, October 1980.

3. D. L. Reeder, Quick Look Report on LOFT Nuclear Experiment L6-5, EGG-LOFT-5165, June 1980.

4. D. L. Reeder, Quick Look Report on LOFT Nuclear Experiments 6-1, L6-2, and L6-3, EGG-LOFT-5270, October 1980.

LOFT ANTICIPATED TRANSIENT

EXPERIMENTAL RESULTS

C. W. SOLBRIG

EG&G Idaho

LOFT

#1

ACCOMPLISHMENTS

- LOFT PERFORMED FOUR ANTICIPATED TRANSIENTS.
    - LOSS OF LOAD.
    - LOSS OF FLOW.
    - EXCESSIVE LOAD INCREASE.
    - LOSS OF FEEDWATER.
- THREE WERE PERFORMED IN ONE WEEK.
- EACH MET THE EXPERIMENTAL OBJECTIVE.

#2

OUTLINE

- WHY ANTICIPATED TRANSIENT EXPERIMENTS ARE USEFUL.
- WHAT EXPERIMENTS LOFT HAS PERFORMED.
- EXPERIMENTAL RESULTS.

EG&G Idaho

LOFT

NEED FOR ANTICIPATED TRANSIENTS

- PROVIDE A BASIS FOR ATMF (E.G., ATWS).
- THE TESTS ARE NON-TRIVIAL. PREDICTIONS COULD HAVE BEEN BETTER.
- THE ADEQUACY OF MOST SAR ANALYSES HAVE NOT BEEN VERIFIED.
- SIMULATORS ARE GOOD ENOUGH FOR SET POINTS BUT NOT ATMF.
- ANTICIPATED TRANSIENTS PROBABILITY IS HIGHER.

EG&G Idaho

LOFT

REACTOR SIMULATORS

- SIMULATORS OF THE FUTURE WILL REQUIRE DIGITAL COMPUTER CODES AS A DRIVER.
- CODES DO NOT REPRESENT ALL ASPECTS OF NORMAL OPERATION AND ANTICIPATED TRANSIENTS.
- SECONDARY SIDE MODELS, PRESSURIZER HEATERS AND SPRAY, ETC., MUST BE IMPROVED.
- AT, ATMF, SMALL BREAKS, LARGE BREAKS, AND ALL TRANSIENTS IN BETWEEN MUST BE REPRESENTABLE.

EG&G Idaho

LOFT

#5

OTHER REGULATORY ISSUES

- SIMULATORS CANNOT ANSWER OPERATION QUESTIONS.
- MANY QUESTIONS ARE POSED FOR THE OPERATION OF A NUCLEAR PLANT - BY PLANT OPERATIONS.
- TECH SPEC CHANGES.
- VERIFICATION OF CURRENT PROCEDURES (E.G., VALVING OUT ECC SYSTEMS AT 1000 PSI).
- TRAINING PROGRAMS - OPERATORS AND TECH ADVISORS.

EG&G Idaho

LOFT

#6

EXAMPLE - DIESEL GENERATOR LOADING TEST

- OPERABILITY CHECK REQUIRED OF DIESEL GENERATOR.
- THE HPI MUST BE BLOCKED FOR A SHORT TIME.
- MINIMUM DOWNTIME IS DESIRED.
- CAN THIS CHECK BE PERFORMED DURING HOT STANDBY CONDITIONS?
- IS THE PLANT ADEQUATELY PROTECTED AGAINST SMALL BREAK?

EG&G Idaho

LOFT

#7

LOFT MUST PERFORM SUCH EXPERIMENTS

- SMALL SYSTEMS DON'T HAVE REPRESENTATIVE EQUIPMENT:
  - MULTIPLE ECC TRAINS.
  - SECONDARY SIDE COMPONENTS.
  - SINGLE FAILURE PROOF COMPONENTS.
- SMALL SYSTEMS HAVE LARGE HEAT LOSSES.
- A LARGE PLANT SUCH AS ARKANSAS NUCLEAR ONE OR SEQUOYAH DOESN'T RECORD ENOUGH INFORMATION - ARE NOT SEVERE.
- LOFT IS THE ONLY FACILITY CAPABLE OF PERFORMING REALISTIC EXPERIMENTS - OTHERS PERFORM CONSERVATIVE EXPERIMENTS.

#8

LOFT ANTICIPATED TRANSIENT EXPERIMENT OBJECTIVES

- PHENOMENA UNDERSTANDING.
- THRESHOLD DETERMINATION.
- AUGMENTED OPERATOR PROGRAM.
- ENGINEERED SAFETY FEATURES/PLANT CONTROL SYSTEMS.
- CODE ASSESSMENT.

EG&G Idaho

LOFT

#9

TRANSIENT CHARACTERISTICS

- COOLANT INVENTORY CONSTANT OR INITIALLY INCREASING.
- PCS ENERGY BALANCE IMPORTANT.
- PCS PRESSURE IS A FUNCTION OF ENERGY BALANCE AND PRESSURIZER DYNAMICS.

LOFT ANTICIPATED TRANSIENTS CONSIDERATIONS

- BASELINE LOFT DATA.
- PCS AND SCS INITIATING EVENTS.
- CODE ASSESSMENT.
- MINIMUM SCHEDULE IMPACT.

EG&G Idaho

LOFT

LOFT ANTICIPATED TRANSIENT EXPERIMENTS COMPLETED

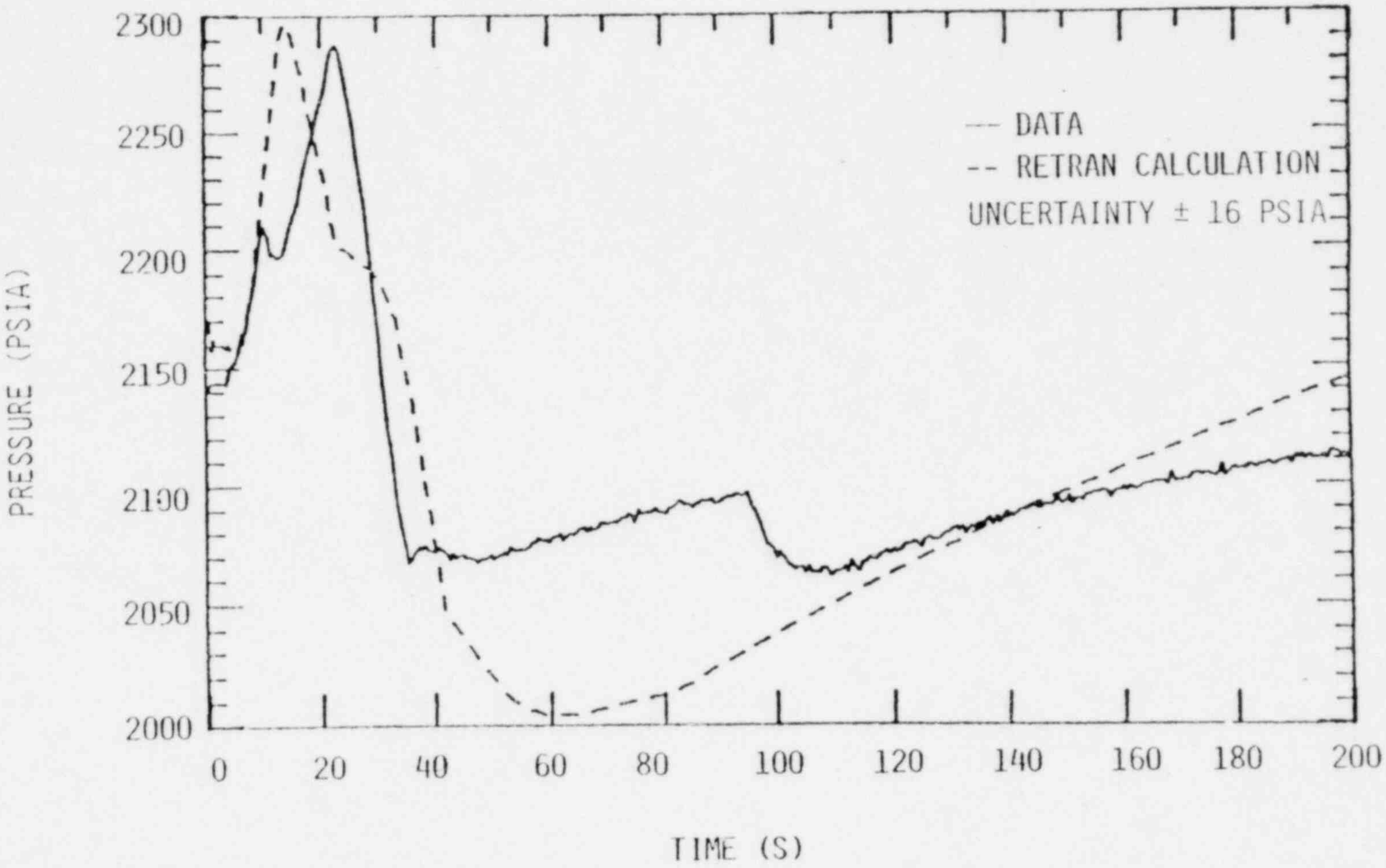
- LOSS-OF-STEAM LOAD.
- LOSS-OF-FORCED PCS FLOW.
- EXCESSIVE LOAD INCREASE.
- LOSS-OF-FEEDWATER

EACH WAS PREDICTED WITH RETRAN

EG&G Idaho

LOFT

MEASURED AND PREDICTED PRESSURE RESPONSE TO LOSS-OF-STEAM LOAD

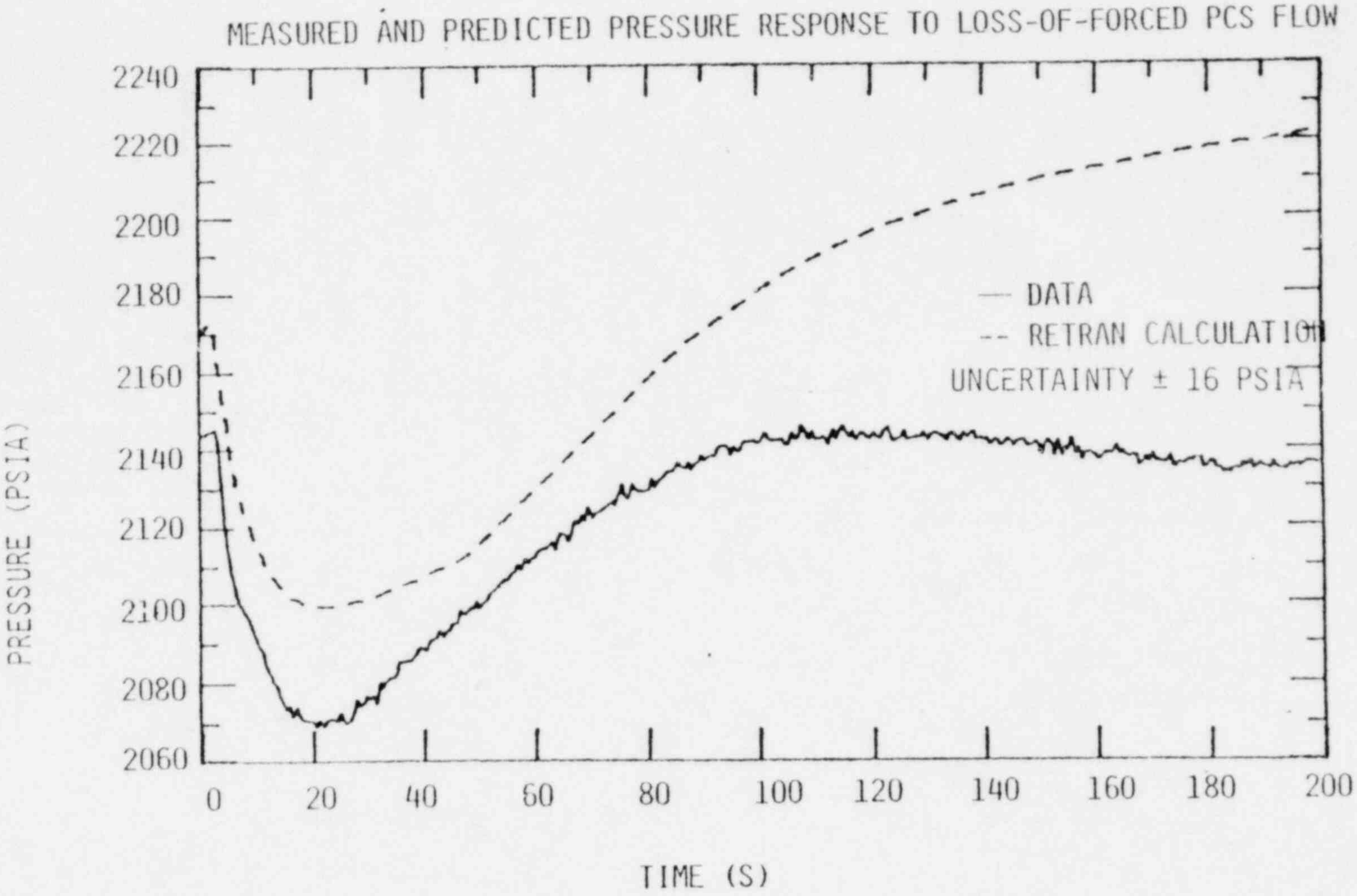


IMPROVEMENTS FROM L6-1

- SPRAY HEAT TRANSFER COEFFICIENTS.
- CONDENSATION EFFECTS.
- PRESSURIZER HEATER MODEL.

EG&G Idaho

LOFT



MEASURED AND PREDICTED PRESSURE RESPONSE TO LOSS-OF-FORCED PCS FLOW

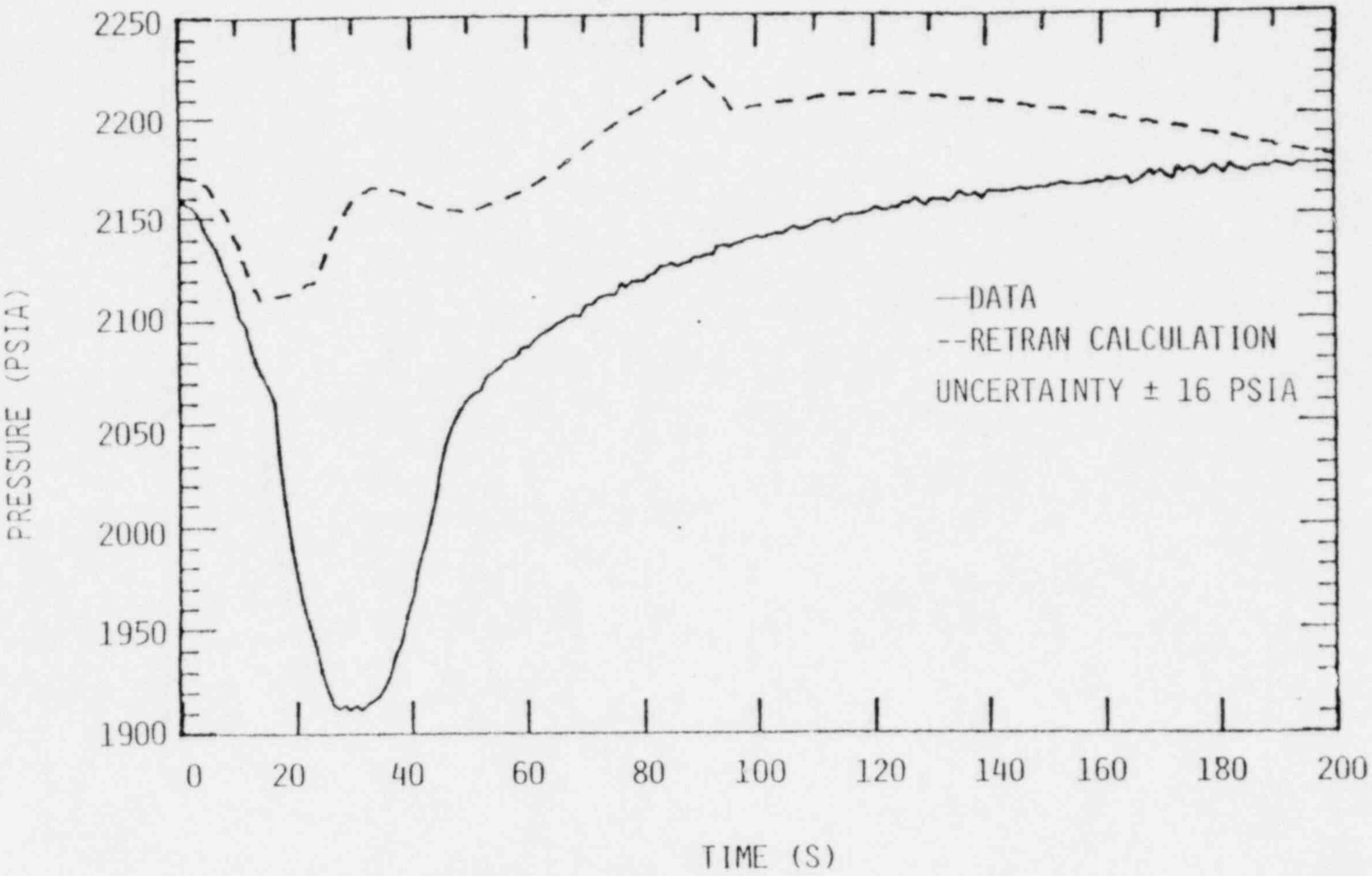
IMPROVEMENTS FROM L6-2

- NON-EQUILIBRIUM MODEL IN PRESSURIZER

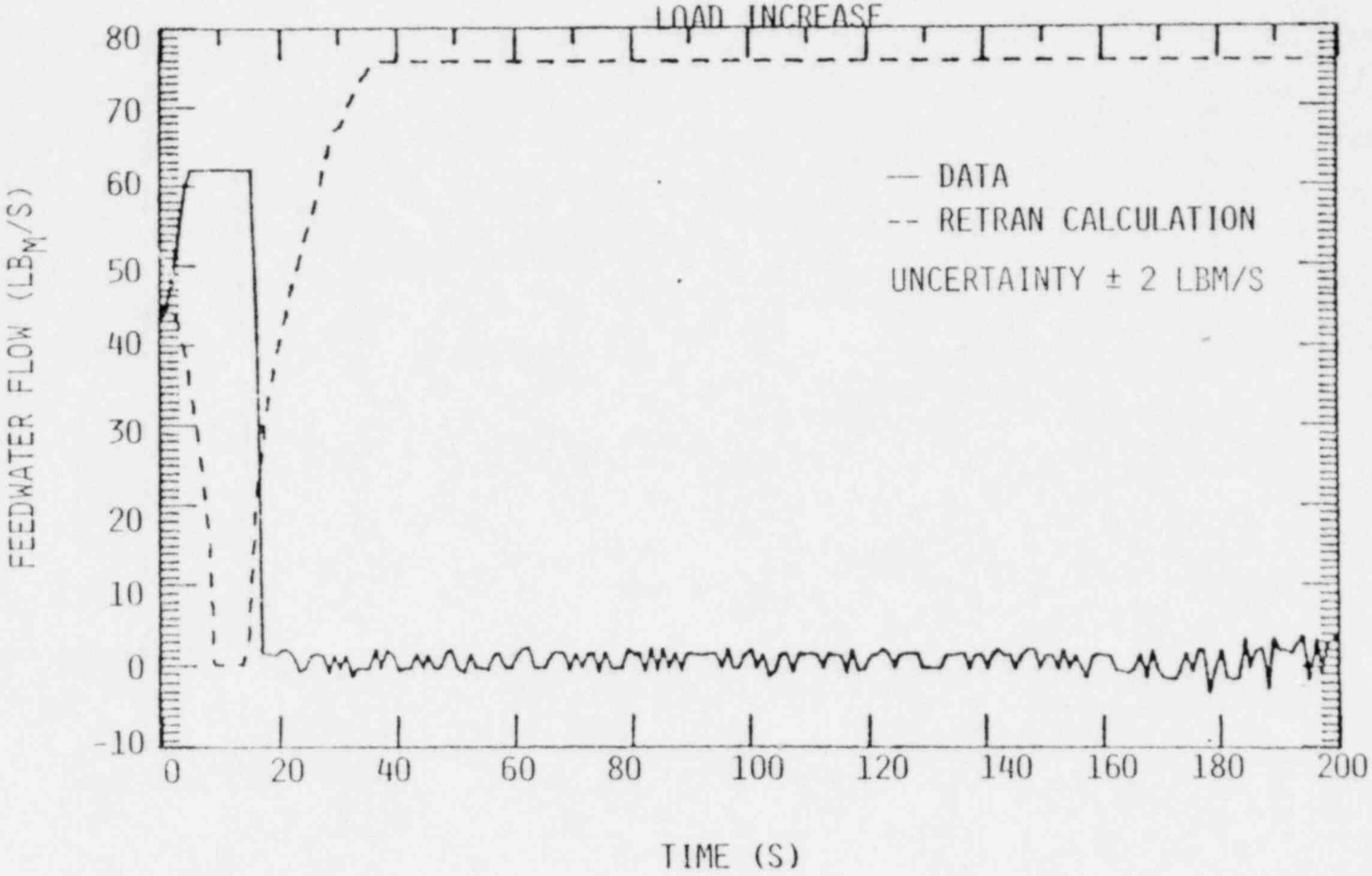
EG&G Idaho

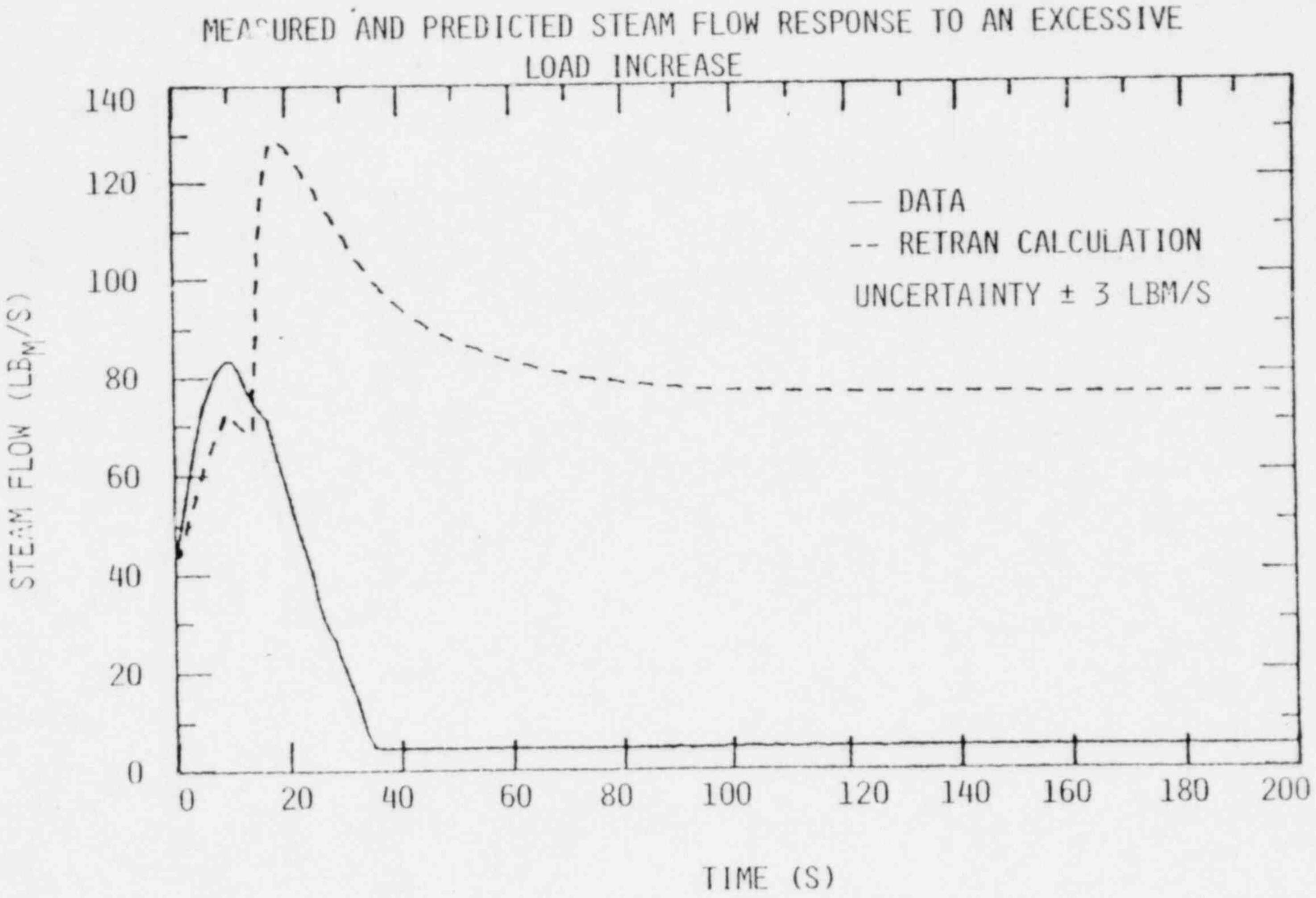
LOFT

MEASURED AND PREDICTED PRESSURE RESPONSE TO AN EXCESSIVE LOAD INCREASE



MEASURED AND PREDICTED FEEDWATER FLOW RESPONSE TO AN EXCESSIVE LOAD INCREASE





MEASURED AND PREDICTED STEAM FLOW RESPONSE TO AN EXCESSIVE LOAD INCREASE

IMPROVEMENTS FROM L6-3

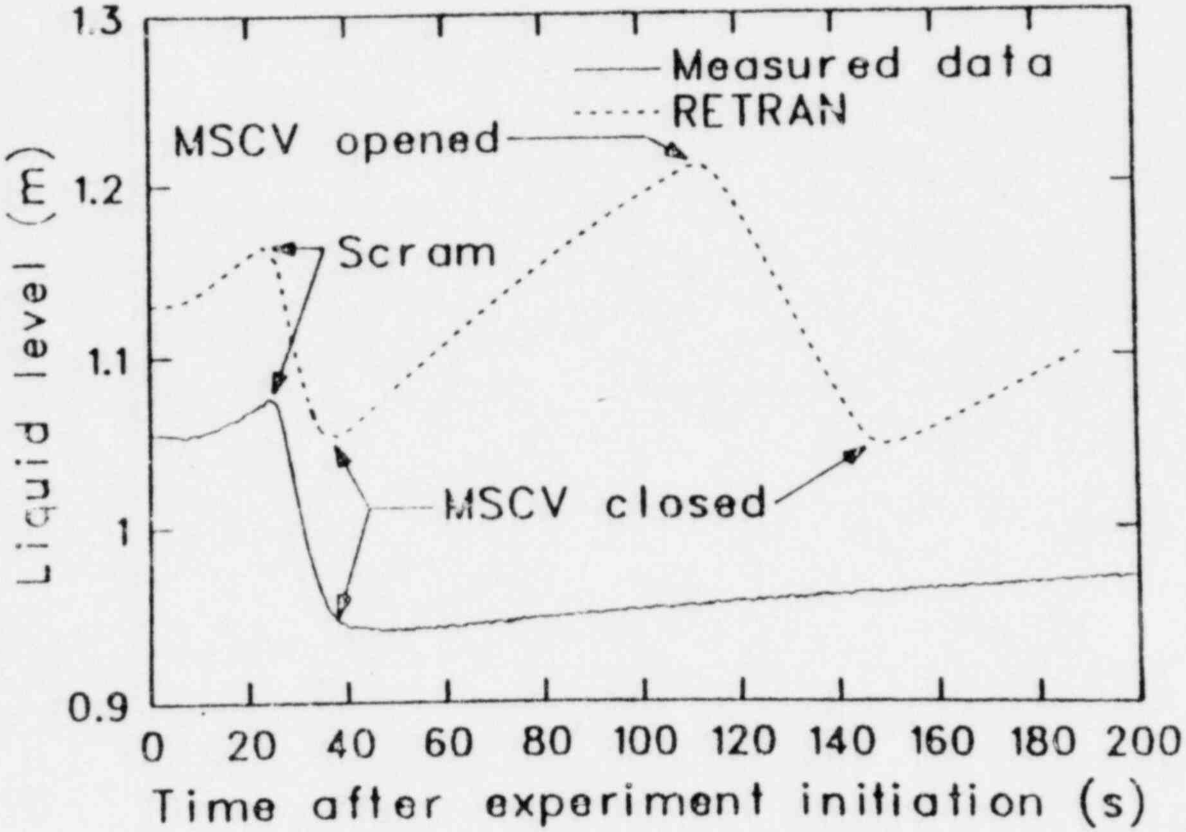
- SCRAM.
- FEEDWATER CONTROL.
- SECONDARY STEAM FLOW.

EG&G Idaho

LOFT

PREDICTED AND MEASURED PRESSURIZER LEVEL

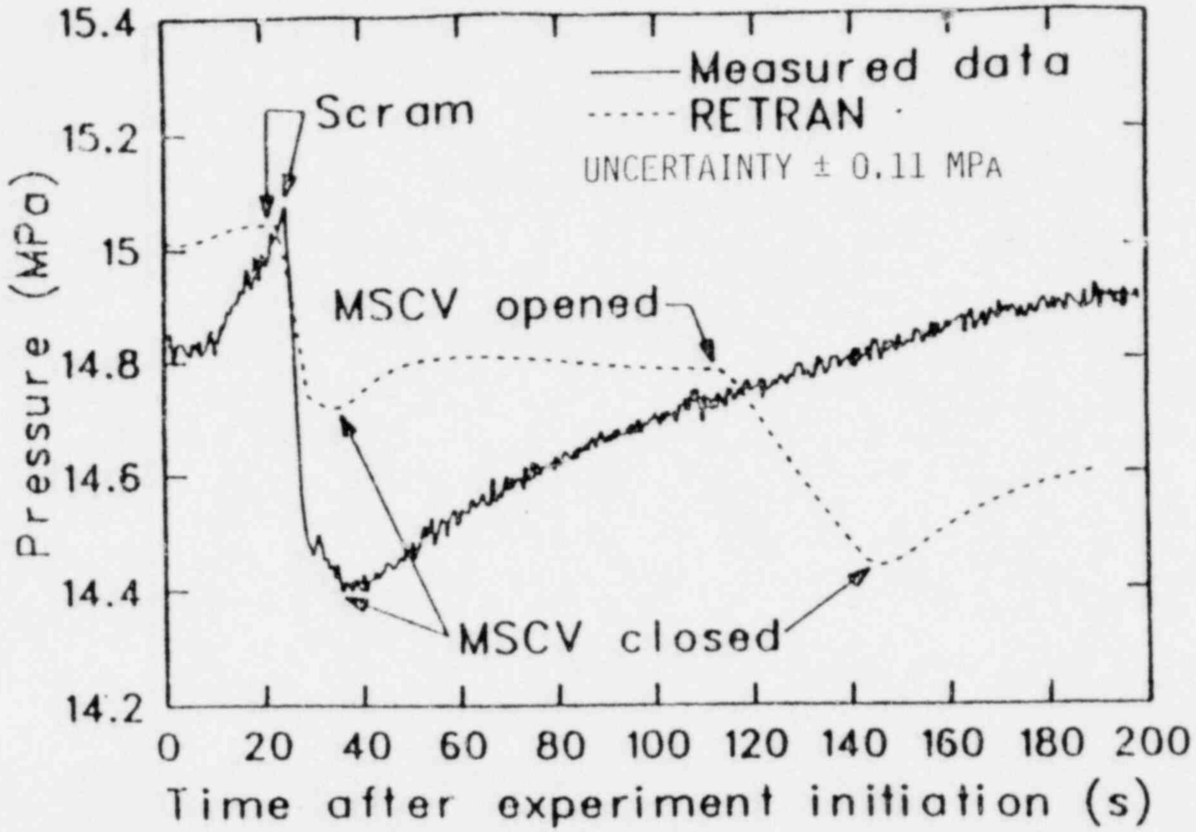
RESPONSE TO LOSS-OF-FEEDWATER



UNCERTAINTY ± 0.07 M

MEASURED AND PREDICTED PRESSURE RESPONSE

TO LOSS-OF-FEEDWATER



IMPROVEMENT FROM L6-5

- DECAY HEAT.
- SECONDARY SIDE LEAKAGE.

EG&G Idaho

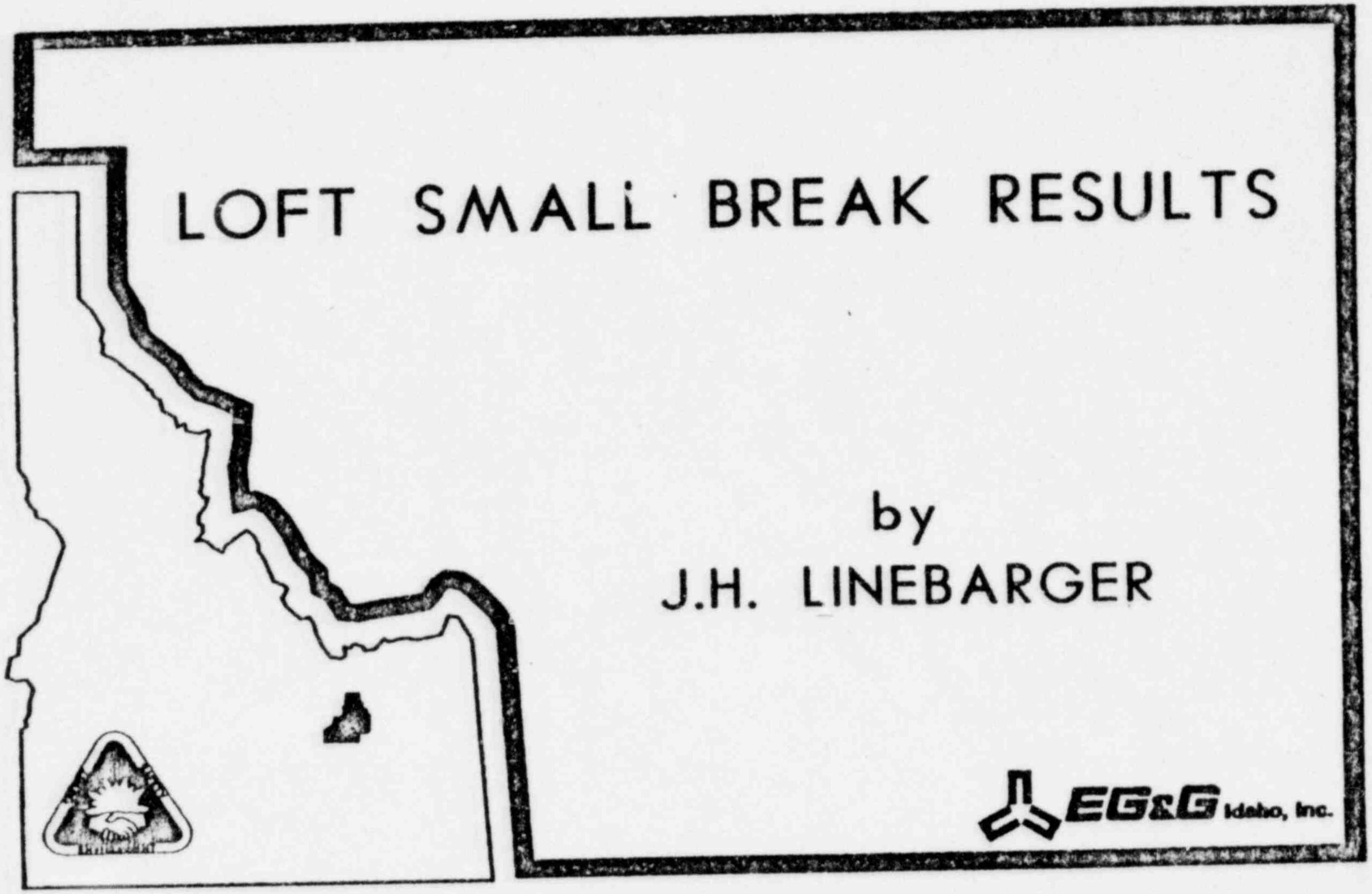
LOFT

CONCLUSIONS

- FOUR SUCCESSFUL EXPERIMENTS WERE PERFORMED.
- CURRENT MODELS IN SAR'S SHOULD BE VERIFIED.
- RETRAN CAN PREDICT TRENDS AND EVENTS.
- SEVERAL AREAS FOR IMPROVEMENT HAVE BEEN DETERMINED.
- ESF/PPS AND OPERATOR ACTION WAS EFFECTIVE.

EG&G Idaho

LOFT



LOFT SMALL BREAK RESULTS

by

J.H. LINEBARGER

EG&G Idaho, Inc.

CONTENTS

INTRODUCTION
- LICENSING CONCERN
- PROGRESS

RESULTS
- SCENARIO
- TRANSIENT SIGNATURES
- RECOVERY

CONCLUSIONS

JHL-1

SMALL BREAKS (SINGLE FAILURE)
WESTINGHOUSE PREDICTIONS

COLD LEG BREAK SIZE	PRESSURE	DECAY HEAT	ECC	CORE UNCOVERY
≥~ 0.05m	CONTINUOUS	BREAK	HPIS AND ACCUMULATOR	PARTIAL
≤~ 0.025m	STABILIZES ABOVE SECONDARY	S.G. AND BREAK	HPIS	NONE

JHL-3

REVIEW OF PROGRESS

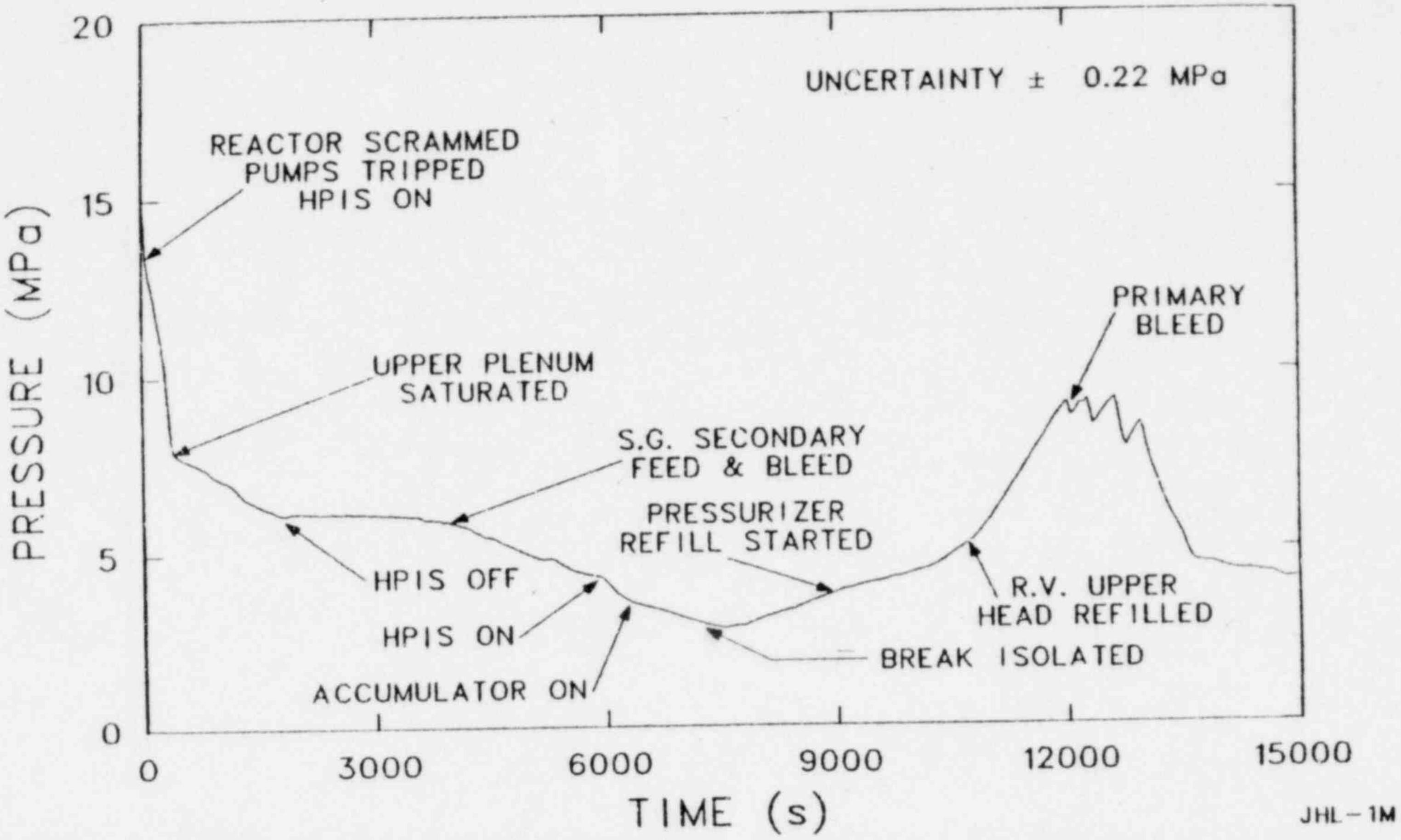
COLD LEG BREAK SIZE (SIMULATED)	EXP.#	OPERATOR ACTIONS
0.10m	L3-1	S.G. F AND B
	L3-5	PUMP OPERATION
	L3-6	
0.025m	L3-2	S.G. F AND B
	L3-7	
	L3-5A	

SMALL BREAKS (SINGLE FAILURE)
LOFT EXPERIMENT RESULTS

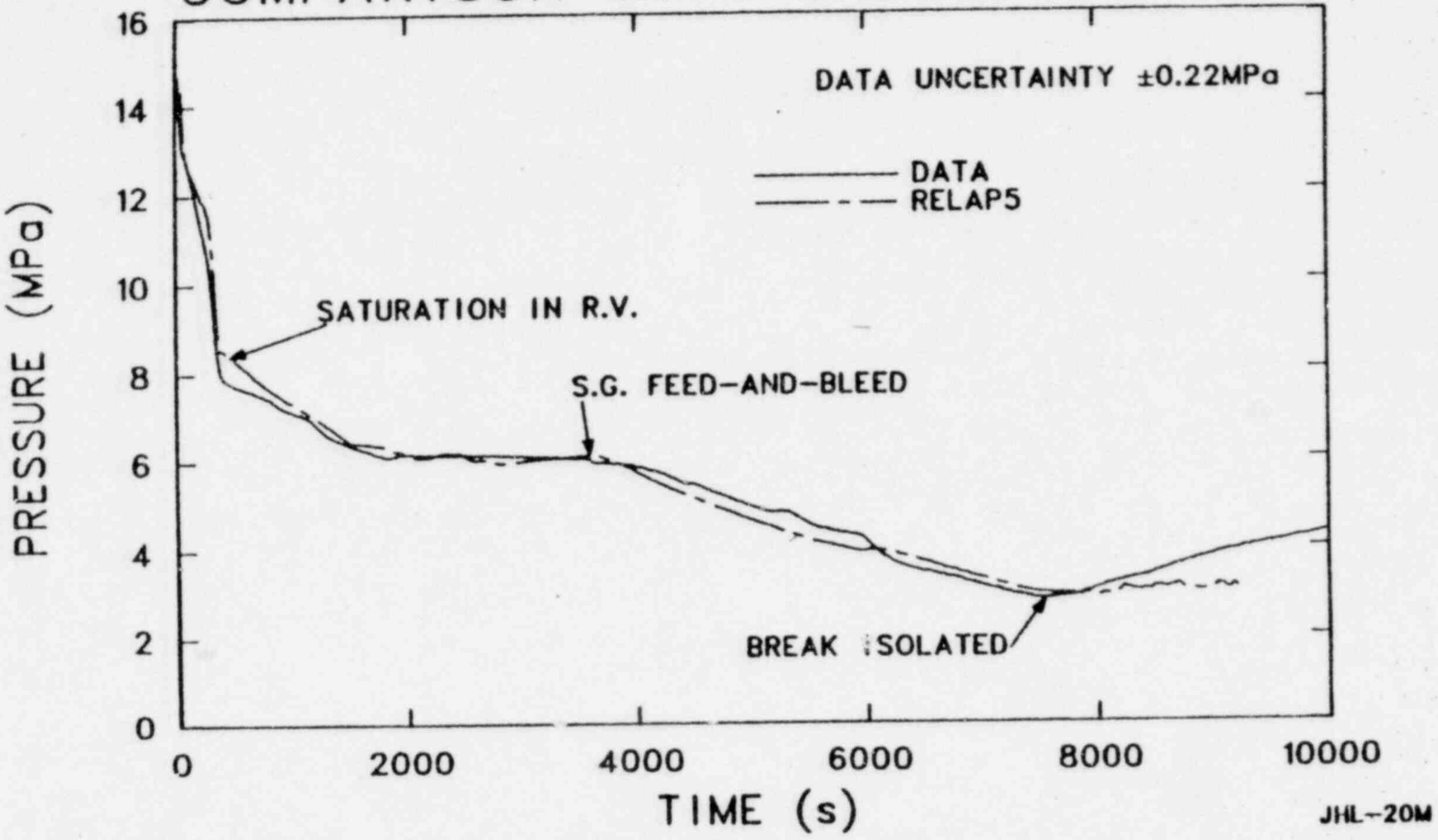
COLD LEG BREAK SIZE (SIMULATED)	PRESSURE	DECAY HEAT	ECC	CORE UNCOVERY
0.10m	SHORT STABLE PERIOD-THEN CONTINUES	BREAK	HPIS AND ACCUMULATOR	NONE
0.025m	STABILIZES ABOVE SECONDARY	S.G. AND BREAK	HPIS	NONE

JHL-5B

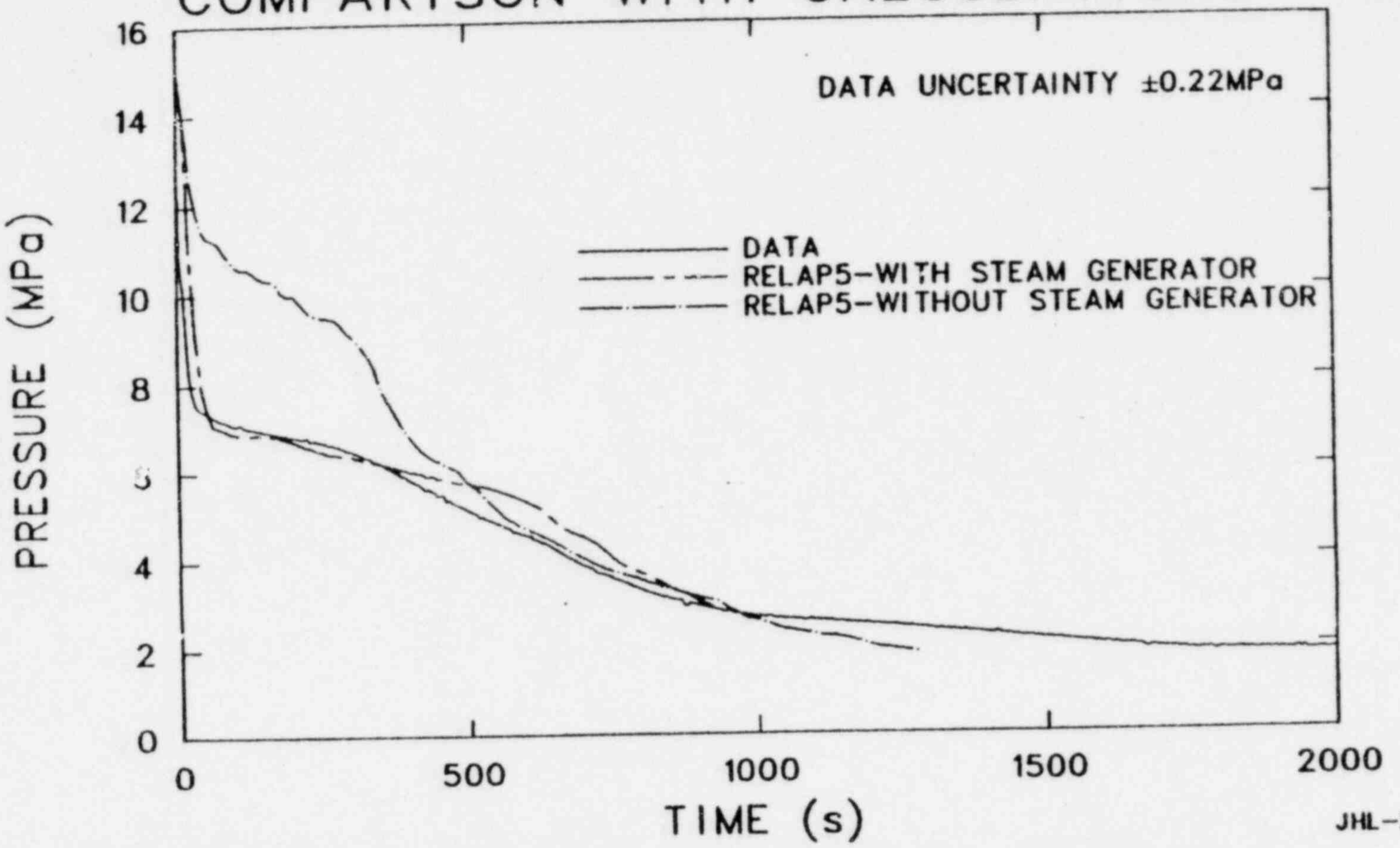
L3-7 PRIMARY SYSTEM PRESSURE



L3-7 PRIMARY SYSTEM PRESSURE COMPARISON WITH CALCULATIONS

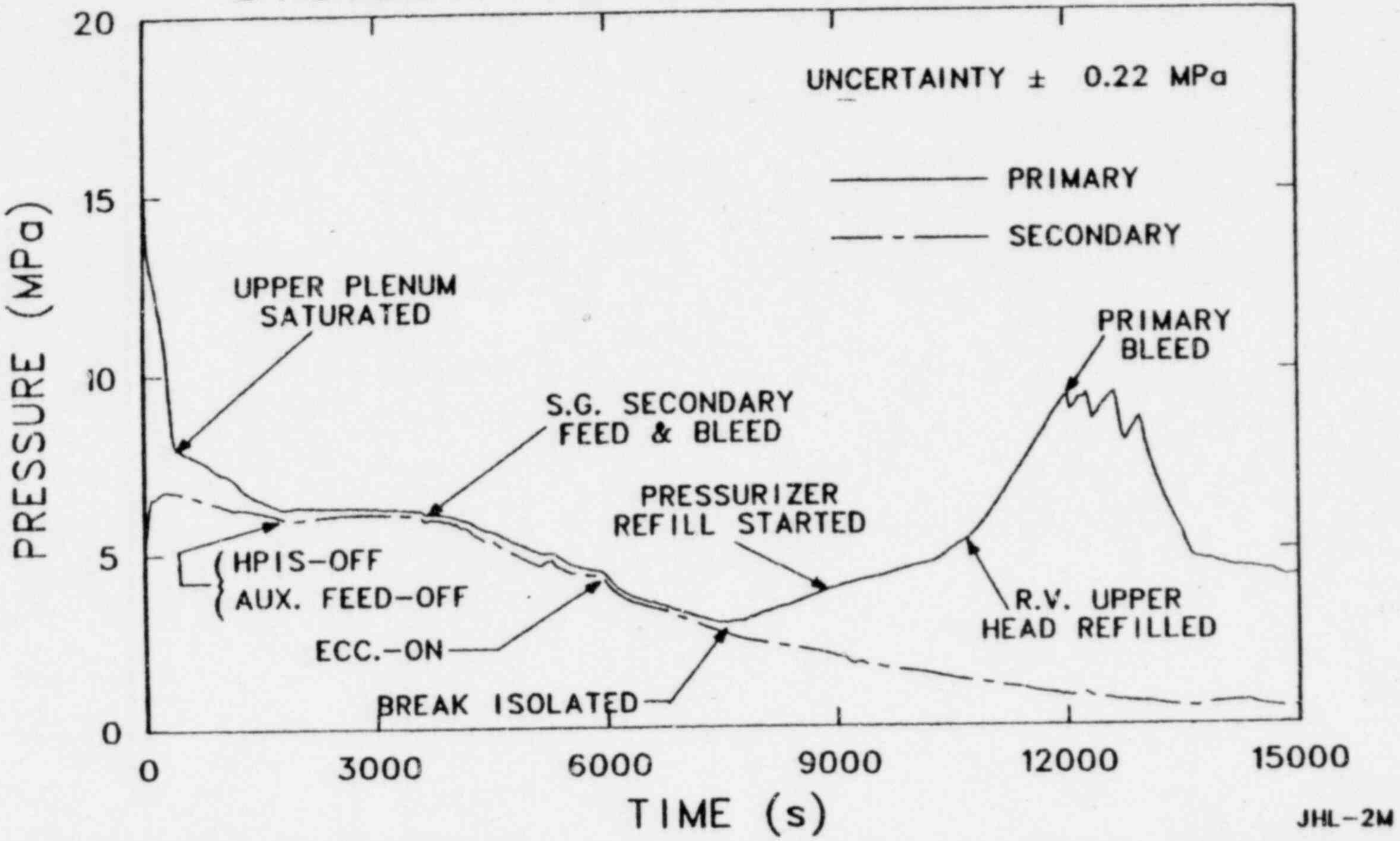


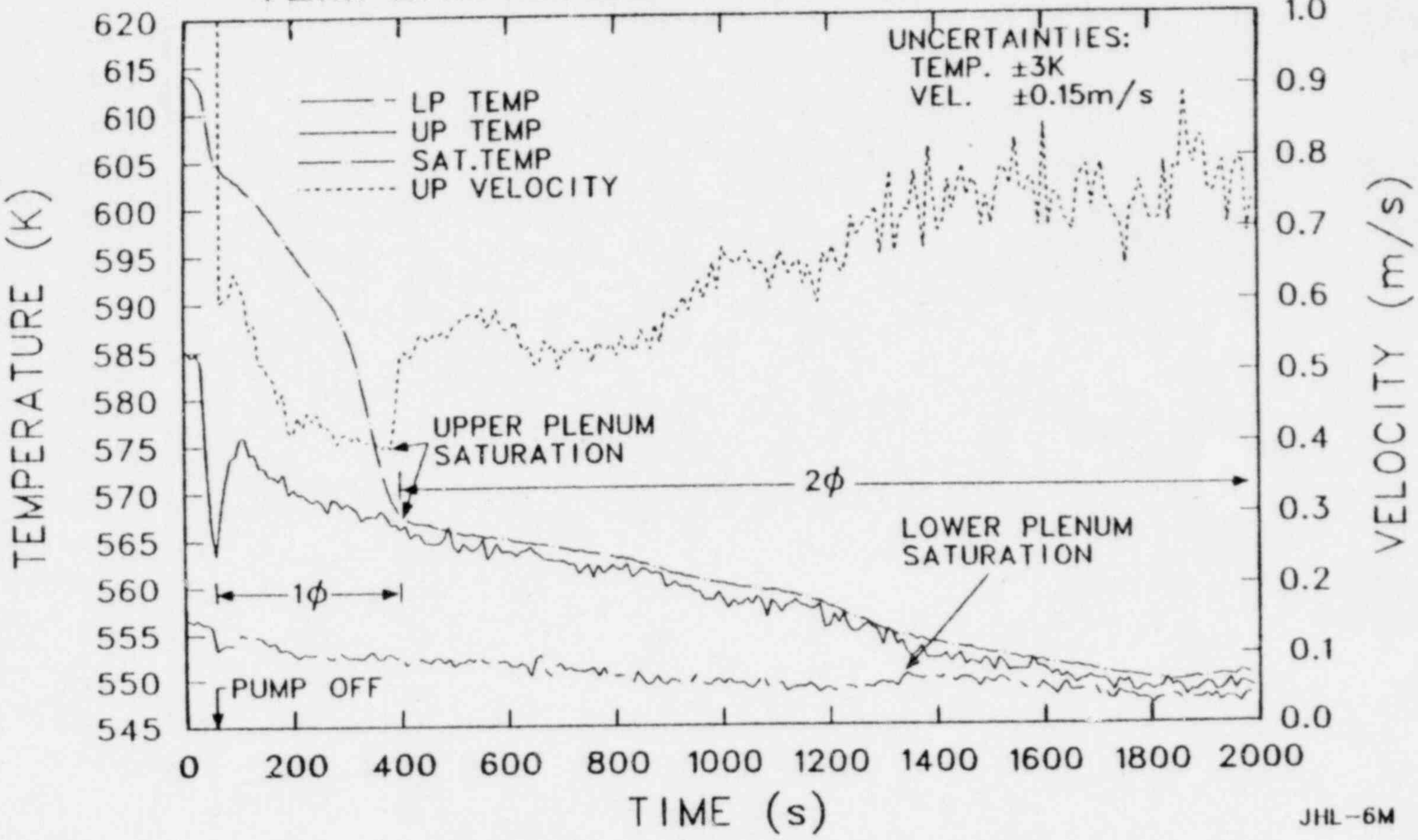
L3-1 PRIMARY SYSTEM PRESSURE COMPARISON WITH CALCULATIONS



JHL-24

L3-7 PRIMARY AND SECONDARY SYSTEM PRESSURE COMPARISON

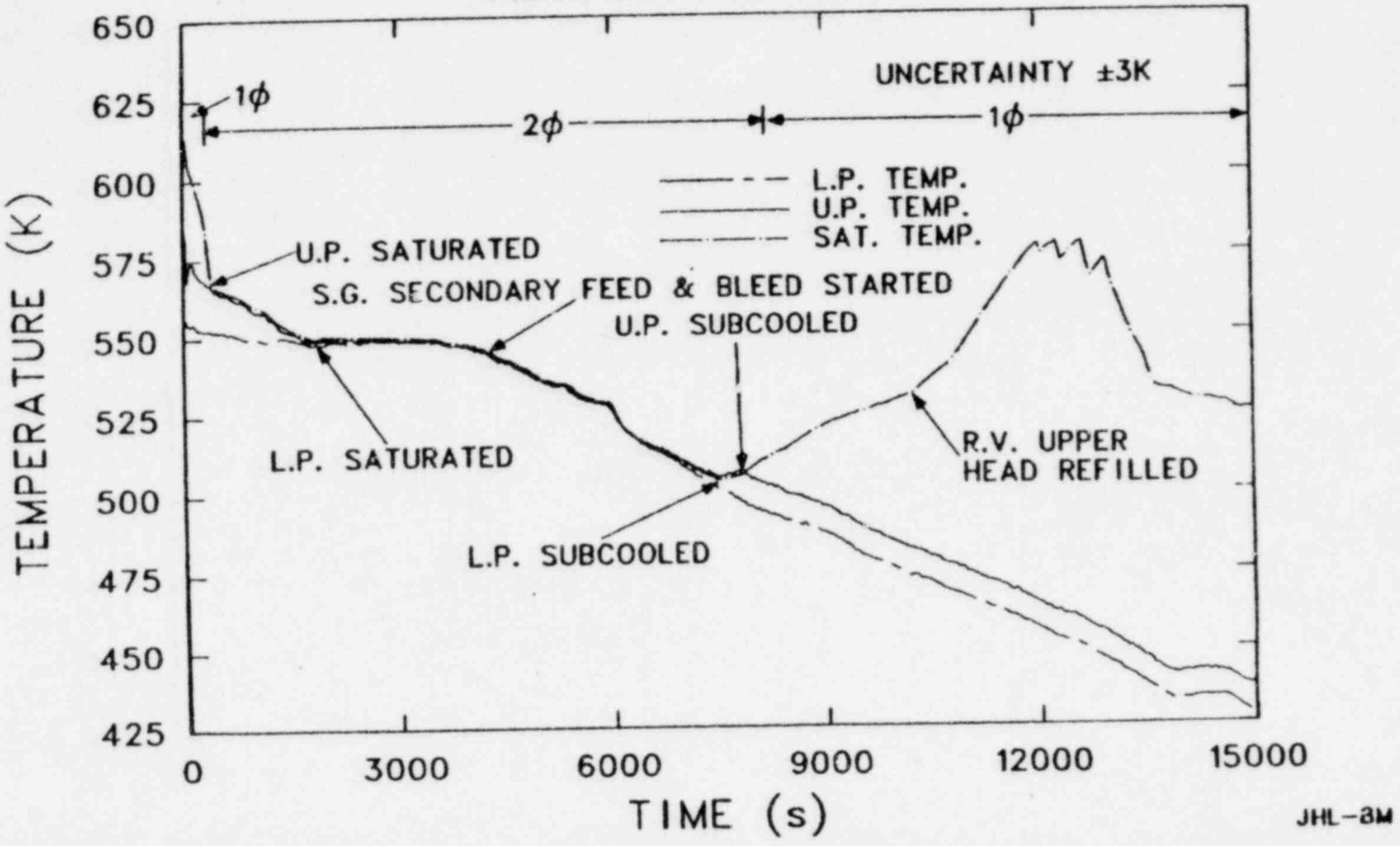




L3-7 REACTOR VESSEL FLUID TEMPERATURES AND VELOCITY

JHL-6M

L3-7 REACTOR VESSEL FLUID TEMPERATURES



NATURAL CIRCULATION CONCLUSIONS

- OCCURS IN SINGLE AND TWO-PHASE MODES
- STABLE IN AND BETWEEN MODES
- REVERSIBLE
- REESTABLISHABLE
- NOT DETERRED BY
  ECC
  R.V. VOIDING
  NON CONDENSIBLES

JHL-10

LICENSING CONCLUSIONS

- PWR AND LOFT SCENARIOS COMPARABLE
- CALCULATIONS PREDICT DOMINANT TRANSITIONS AND ASSOCIATED PHENOMENA IN PROPER TIME SEQUENCE
- RECOVERY PROCESS IS CONVERGENT

LICENSING CONCLUSIONS (CONT'D)

- STEAM GENERATOR IS EFFECTIVE HEAT SINK
- OPERATOR INITIATED STEAM GENERATOR SECONDARY FEED-AND-BLEED EXPEDITES RECOVERY

JHL-9A

OPERATOR INTERVENTION IN TRANSIENT TESTS
LOFT
AUGMENTED OPERATOR CAPABILITY PROGRAM

LOFT

EG&G Idaho

Meyer

LOFT

AUGMENTED OPERATOR CAPABILITY PROGRAM

THE ENHANCEMENT OF THE OPERATOR'S CAPABILITY TO MAINTAIN AND RESTORE PLANT SAFETY BY THE USE OF REAL-TIME COMPUTER TECHNOLOGY APPLIED ACCORDING TO END-TO-END SYSTEM ENGINEERING PRINCIPLES.

LOFT

AUGMENTED OPERATOR CAPABILITY PROGRAM (CONTINUED)

OBJECTIVES

A. DEVELOP ADVANCED DIAGNOSTIC GRAPHIC DISPLAYS

B. DEVELOP REAL-TIME COMPUTER TECHNOLOGY

LOFT

AUGMENTED OPERATOR CAPABILITY PROGRAM (CONTINUED)

OBJECTIVE A

DEVELOP ADVANCED DIAGNOSTIC GRAPHIC DISPLAYS FOR THE LWR NUCLEAR STEAM SUPPLY SYSTEM (NSSS) WHICH:

- DISCLOSE ACTUAL NSSS SAFETY STATUS
- RECOGNIZE AND IDENTIFY EVENTS
- RELATE THE EVENT TO OPERATIONAL PRINCIPLES
- ASSIST THE OPERATOR IN MAINTAINING INTELLIGENT CONTROL

LOFT

AUGMENTED OPERATOR CAPABILITY PROGRAM (CONTINUED)

OBJECTIVE B

DEVELOP REAL-TIME COMPUTER TECHNOLOGY FOR PROVIDING SAFETY-RELATED INFORMATION:

- DATA ACQUISITION
- DATA INTEGRITY
- COMPUTER NETWORKING
- SOFTWARE VALIDITY
- SECURITY
- USER ORIENTED INTERFACING

THE LOFT AUGMENTED OPERATOR CAPABILITY PROGRAM

D. A. Hollenbeck, E. A. Krantz, G. L. Hunt, and D. R. Meyer
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, Idaho 83415

ABSTRACT

The outline of the LOFT Augmented Operator Capability Program is presented. This program utilizes the LOFT (Loss-of-Fluid Test) reactor facility which is located at the Idaho National Engineering Laboratory and the LOFT operational transient experiment series as a test bed for methods of enhancing the reactor operator's capability for safer operation. The design of an Operational Diagnostics and Display System is presented which was backfit to the existing data acquisition computers. Basic color-graphic displays of the process schematic and trend type are presented. In addition, displays were developed and are presented which represent "safety state vector" information. A task analysis method was applied to LOFT reactor operating procedures to test its usefulness in defining the operator's information needs and workload.

NOTICE

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights. The views expressed in this paper are not necessarily those of the U.S. Nuclear Regulatory Commission.

Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research under DOE Contract No. DE-AC07-76ID01570.

INTRODUCTION

A near consensus has been reached on the need to apply state-of-the-art technology to the safe operation problems of a commercial light water reactor (LWR) under upset or faulted conditions. The two major elements of this technology are: (1) computer technology and (2) functional analysis of operations.

COMPUTER TECHNOLOGY

Under off-normal operational conditions, the operator in a nuclear power plant is presented with an enormous amount of information which must be collected, processed, and evaluated in order to make appropriate control decisions as to whether the plant can be restored to normal operating conditions or should be shutdown.

Under emergency conditions, the active area of the control panel and the volume of raw data can exceed the saturation point of the operator. This data is presented to the reactor operator without prioritization in a short period of time. Yet, the operator needs more, not less, information concerning the status of crucial plant systems. Thereby, a dilemma exists in balancing a recognized need to reduce operator data overload against a perceived need by the operator for more data. This dilemma can be resolved by the use of computers to reduce raw information to significant information which can be displayed in recognizable form.

An Operational Diagnostics and Display System (ODDS) has been designed for use with the Loss-of-Fluid Test (LOFT) reactor at the Idaho National Engineering Laboratory. The ODDS is presently being evaluated during small break (loss-of-flow) tests conducted on the LOFT reactor. The ODDS will improve the operator's capability for making correct and timely control decisions.

LOFT is a scaled-down version of a commercial pressurized water reactor (PWR) (one sixty-fourth size). It is felt LOFT resembles a commercial PWR in man-machine factors which permits evaluation of computer-based graphic displays for their potential use in commercial LWR applications. The LOFT man-machine factors representative of typical LWRs are shown in Table I.

TABLE I
LOFT MAN-MACHINE FACTORS REPRESENTATIVE OF TYPICAL LWRs

---

1. Reactor Facility
    - a. Nuclear Steam Supply System
    - b. Main Control Room
    - c. Automatic Protective Systems (RSS, ECCS, CIS)
    - d. Instrument and Control Equipment
2. Operational Framework
    - a. Technical Specifications
    - b. Operating Procedures
    - c. Operating Crew
    - d. Training
    - e. Maintenance Practices

DESIGN CONFIGURATION

The hardware components of the LOFT Operational Diagnostics and Display System (ODDS) are shown in Figure 1. The ODDS consists of a central processing unit (CPU), asynchronous multi-line controller (AMLC), memory unit, disk storage unit, magnetic tape unit, and display terminals. The CPU is a PRIME 550, a machine near the upper end of the performance range of minicomputers. The system is configured with 512 kilobytes of main memory and possesses two kilobytes of high speed cache memory to speed program execution. Both on-line and off-line storage capability are provided for the data files and programs. Three cathode ray tube (CRT) terminals provide an interface with the various users and user interaction with the system. The CRTs are RAMTEK devices interfaced with the PRIME by serial lines and are capable of graphics in eight colors. The same type of serial interface used with the CRTs is also used to connect the PRIME 550 with the LOFT Plant Log and Surveillance Subsystem (PLSS) computer through which data are dynamically acquired.

Initially, The ODDS has been configured to take advantage of the existing LOFT PLSS, a system built around a MODCOMP-IV computer already used to acquire plant information from process instruments in order to provide historical plant log and real-time monitoring functions. The software design approach with respect to data acquisition was to view the data as being comprised of two types: analog and event.

Analog data acquired by the PLSS are routinely buffered so a data point representing an average of several seconds of data for each analog channel is available for processing or presentation. Data transmitted from the PLSS to the PRIME are updated every five seconds. All analog data have been converted to floating point, engineering unit values before being sent to the ODDS.

Event data are discrete data which relate to a physical condition such as a breaker switch or valve position. They are updated to the ODDS every two seconds.

In keeping with the design approach of separating the event and analog data, each type of information is passed over a different physical line by an independent PLSS-resident program and is acquired by an independent program on the ODDS. Complexity of the communication process is kept to a minimum by use of a serial interface with all data transmitted at 9600 baud (bits per second).

Programs resident on the ODDS acquire data from the communication lines, reformat the data, and place the data into storage files on a disk storage unit. Analog and event data are each stored into circular files of approximately 10 hours duration. These data files may be spooled to tape for off-line storage and subsequent retrieval for replay purposes.

A package of display-oriented software exists which accesses the circular disk files and creates the various color displays seen by the user on the CRTs. At the heart of the display package is a set of routines known as the graphics display library. The application programs constructing the various displays all use the graphics display library.

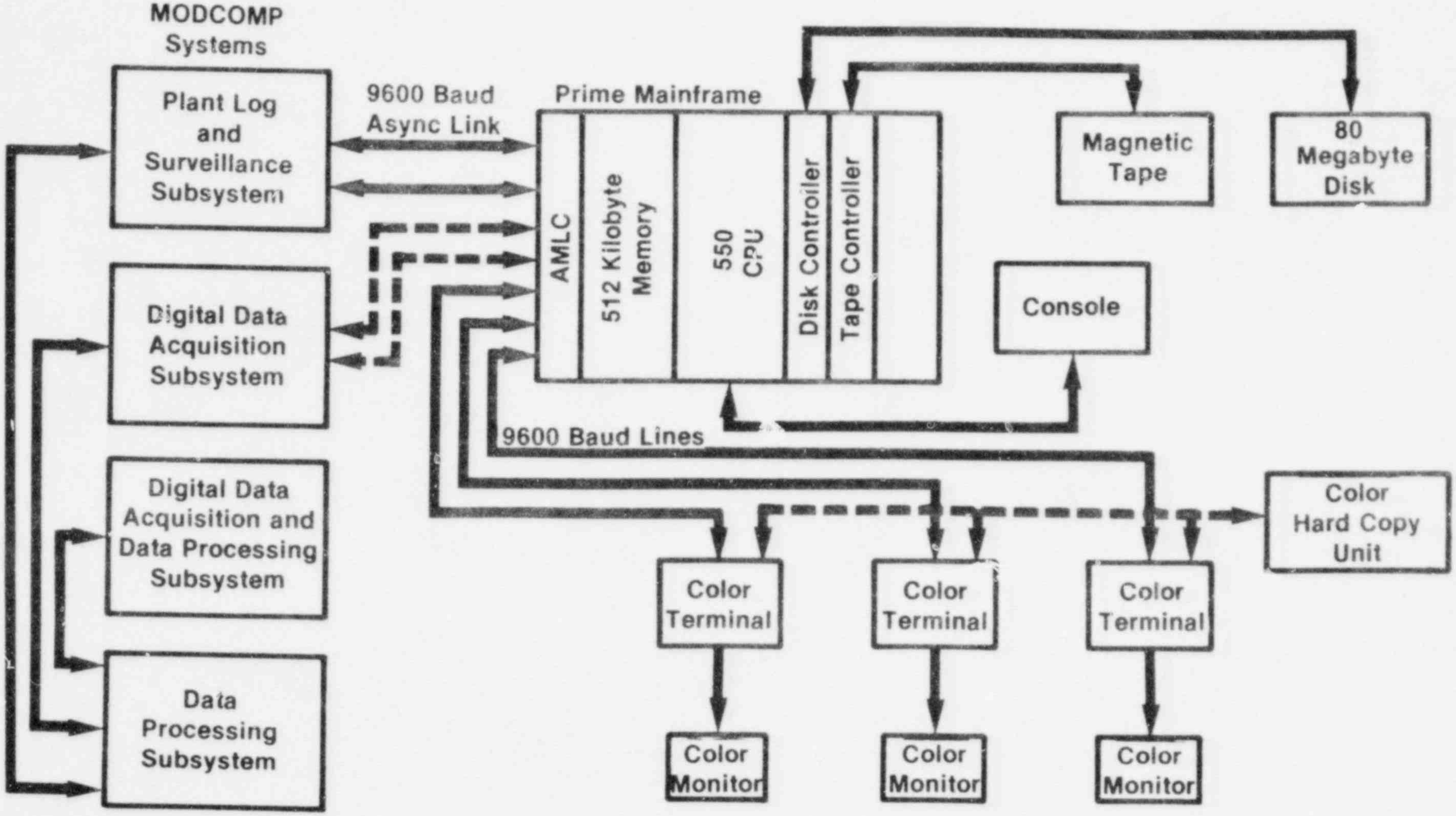


Figure 1. Operational diagnostics and display system (ODDS).

Expansion and enhancement of the software capability is planned. Some items under consideration are: (1) increased data update rates, (2) increased data base to support additional instrumentation, and (3) numerous new applications in the display program package.

BASIC DISPLAYS AND TREND INFORMATION AVAILABLE

A demonstration set of color graphic displays has been implemented on the LOFT ODDS. These displays were chosen to encourage immediate use of the ODDS by the reactor operator. Status-type displays were implemented first to get the ODDS into service rapidly (diagnostic or other complex programs take longer to design and implement). The general criteria used for the selection of LOFT displays were:

a. Displays should present information which is frequently used by the reactor operator during normal reactor operation,

b. Displays should also be of potential use in following the course of a small LOCA (loss-of-coolant accident) or operational transient,

c. Status-type displays should be implemented first,

d. Information should be presented in an integrated fashion to support specific plant evolutions or operation of crucial plant systems,

e. Displays should present information in formats which are complementary to those presently available for the conventional process instrumentation in use at LOFT, and

f. Baseline displays should use information derived from process (non-experimental) measurements.

The demonstration displays can be grouped into two sets: process schematics and status or trend plots. Process schematics exist for the primary coolant system, secondary coolant system and emergency core coolant system. These displays are simplified schematic diagrams with parameter values and component status (e.g., valve position) shown at the appropriate locations on the diagrams. Initial conventions are established for the representation of component status through the use of colors (e.g., pump on or off, vessel level) and symbol shape (e.g., valve open or closed).

Status and trend plots generally show three types of information: (1) present status of one or more crucial plant parameters, (2) recent past history of these parameters, and (3) operating limits for these parameters appropriate for the mode of operation for which the display was intended. Demonstration displays of this type include:

a. Plant heatup (actual vs technical specification limits)

b. Plant cooldown (actual vs technical specification limits)

c. Pressure vs temperature (hot leg conditions vs power operation limits)

d. Minimum pressure vs temperature (cold leg conditions, including pump operation limits)

e. General X-Y plot (any two parameters).

Typical demonstration displays of process schematic, safety state vector, and trend information available on the LOFT ODDS are shown as Figures 2, 3, 4, and 5. Small-break LOCA data from Experiment L3-2 are displayed.

Each of the baseline displays exists in two versions: a "control room operator" version and an "engineering" version. Each version of each display can be called up for viewing on any display terminal either by typing a simple mnemonic (e.g., "PCS" for the Primary Coolant System process schematic) or by pressing a special function key on the terminal keyboard. The control room operator displays have fixed formats and parameter ranges, and display only current data. The engineering displays allow the user to alter such features as the scaling of plots or the indicated status of components; they also allow the replay or display of historical information stored in the computer. This information base includes several hours of the most recent plant data as well as data from previous LOFT tests.

A number of limitations of the present display capabilities are recognized at this time. Some of the more significant ones are:

a. Development of display hierachy and structure has just begun; consequently the present displays are related only through the training and experience of the plant operator.

b. Nuclear industry standards for the use of color, symbology, and other display conventions for such systems have not been established.

c. Some information desired for the demonstration displays is not part of the available data base. (Over 60 status and parameter values have already been added to the LOFT data acquisition system to support the baseline displays.)

d. The displays can be regenerated at will by replaying historical data; however, no simulation capability presently exists to allow varying indicated plant status from that which actually occurred during LOFT operation.

FUNCTIONAL ANALYSIS OF LWR OPERATIONS

Task analysis is being used to determine the operator's information needs during normal and emergency operation of the LOFT facility. Task analysis is a systematic method for analyzing the operation of a system by (1) breaking the operation into its component parts and (2) extracting useful information concerning the operation of the facility. Task analysis is performed in four steps. First, the overall characteristics of system operation are examined to define relevant operating modes of the system and potential transfers between modes. Second, procedures are developed for

POOR ORIGINAL

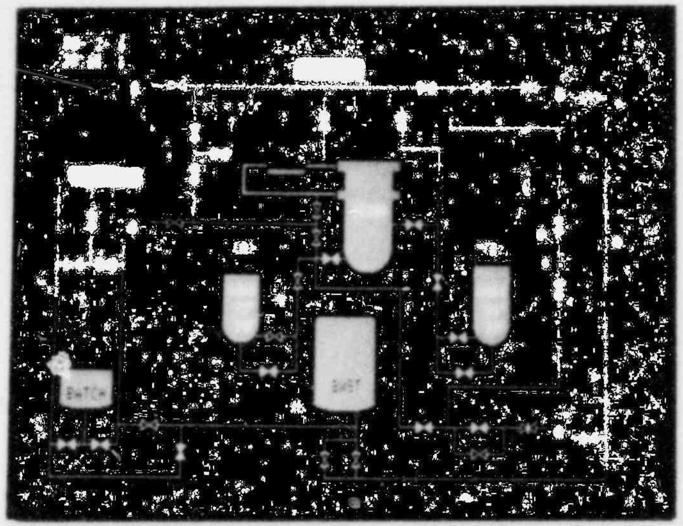


Fig. 2. Emergency Core Cooling System

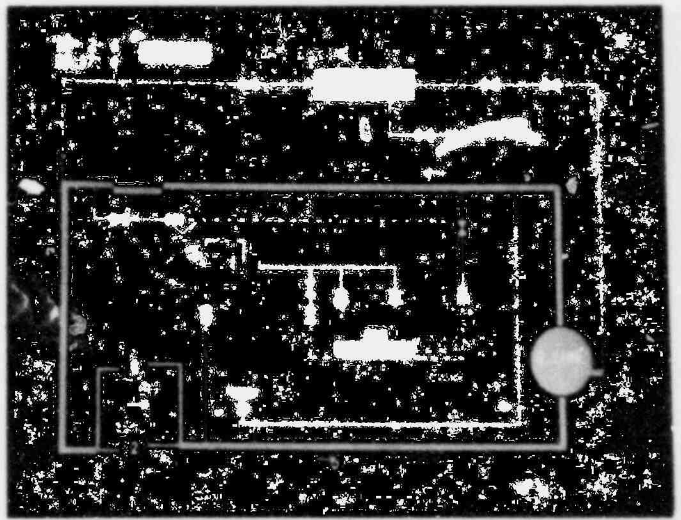


Fig. 3. Primary Cooling System

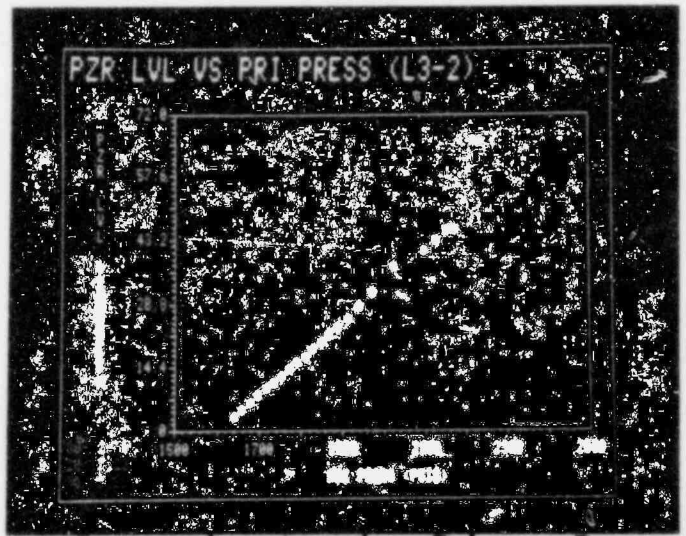


Fig. 4. Pressurizer Level vs Primary Pressure

Fig. 5. Pressure-Temperature Curve

Note: The figures show actual data taken during LOFT small-break test L3-2. The figures are arranged left to right in time sequence.

each mode-to-mode transfer; the LOFT plant operating manual is being used as a basis for this step. Third, each procedure is flow charted to illustrate the operator's decision points and the potential paths through the procedure. Fourth, a tabular form is used to list information from the flow chart including: (1) required decisions, (2) information required to make the decision, (3) source of the information, (4) time available to act, (5) feedback associated with the correct action, and (6) alternative actions available if a malfunction occurs.

The results of LOFT task analyses are used: (1) to make recommendations to improve existing procedures and (2) to make recommendations for the design of CRT displays to be implemented on the ODDS. Representative results of this type of analysis are discussed in Reference 4.

CONCLUSION

The LOFT ODDS was placed in operation in January 1980 and was used by the reactor operators in conducting the LOFT L3-2 small-break test in February 1980. The ODDS is being readily accepted by the LOFT reactor operators as an aid in controlling the plant. Although only a limited number of baseline displays of process schematics and trend information are available at present, computer-based graphic displays are expected to gain acceptance in the future as a useful source of information to assist the reactor operator in his decision-making processes required for normal and off-normal reactor operations.

Functional analysis of operations appears to be as applicable to the LWR operational safety problems as to other modern man-machine control problems. Functional analysis and computer-based graphic technologies are being developed for the LOFT program to permit this unique facility to be used as a workshop and test bed for LWR operational safety problems.

REFERENCES

These references were used as definitions of where reactor operator capabilities should be augmented.

1. NUREG-0585, October 1979, TMI-2 Lessons Learned Task Force Final Report.

2. Report of the President's Commission on the Accident at Three Mile Island, John G. Kemeny, Chairman, October 1979.

3. NUREG/CR-1270, Vol. I, January 1980, Human Factors Evaluation of Control Room Design and Operator Performance at Three Mile Island-2, Final Report.

4. W. R. Nelson, "Response Trees for Emergency Operator Action at the LOFT Facility," 1980 ANS/ENS Meeting on Thermal Reactor Safety, Knoxville, TN, April 7-11, 1980.

RESPONSE TREES FOR EMERGENCY OPERATOR ACTION AT THE LOFT FACILITY

William R. Nelson
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, ID 83415

Presented at

ANS/ENS Topical Meeting on Thermal Reactor Safety

Knoxville, Tennessee

April 7-11, 1980

RESPONSE TREES FOR EMERGENCY OPERATOR ACTION AT THE LOFT FACILITY

William R. Nelson
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, ID 83415

ABSTRACT

A technique for assisting nuclear plant operators during emergency conditions has been developed and implemented at the LOFT facility. The technique is based on "response trees". A response tree is a diagram showing the modes available for responding to an accident and the relative desirability of each. A procedure using response trees is a central reference which directs the operator to specific procedures for responding to the accident. Benefits of the technique include 1) it facilitates efficient operator response, 2) it encourages operator familiarity with all accident response modes, and 3) it applies to many accidents, including common mode and multiple failure events.

INTRODUCTION

Following the onset of an accident which disables equipment used for normal reactor cooling, the first priority of the nuclear plant operator is to ensure that the reactor core is covered with water and that adequate cooling water flow is established. During this time, he must evaluate the situation, determine which emergency procedures apply, find the appropriate procedures, and perform the prescribed actions. Failure to respond quickly and effectively could result in expensive facility damages and potential hazards to the public. A procedure which attempts to streamline this short-term response process has been developed and implemented for the Loss-of-Fluid Test (LOFT) facility.

RESPONSE TREES FOR LOFT

The procedure developed for LOFT is entitled "Loss of Normal Decay Heat Removal Modes." Diagrams called "response trees" have been included in the procedure to illustrate potential modes for cooling the reactor and the relative priority for using each. The procedure is designed to be a central reference point to be used by the operator to determine which specific emergency procedures should be used to respond to the accident.

Figure 1 is the response tree for the LOFT Low Pressure Injection System (LPIS), and Figure 2 is a simplified schematic of the LPIS. The response tree shows all potential cooling modes available using the Low Pressure Injection System. Each cooling mode has five elements: a heat sink, a water source, a pump, a route, and an injection point. Each element may represent many individual components.

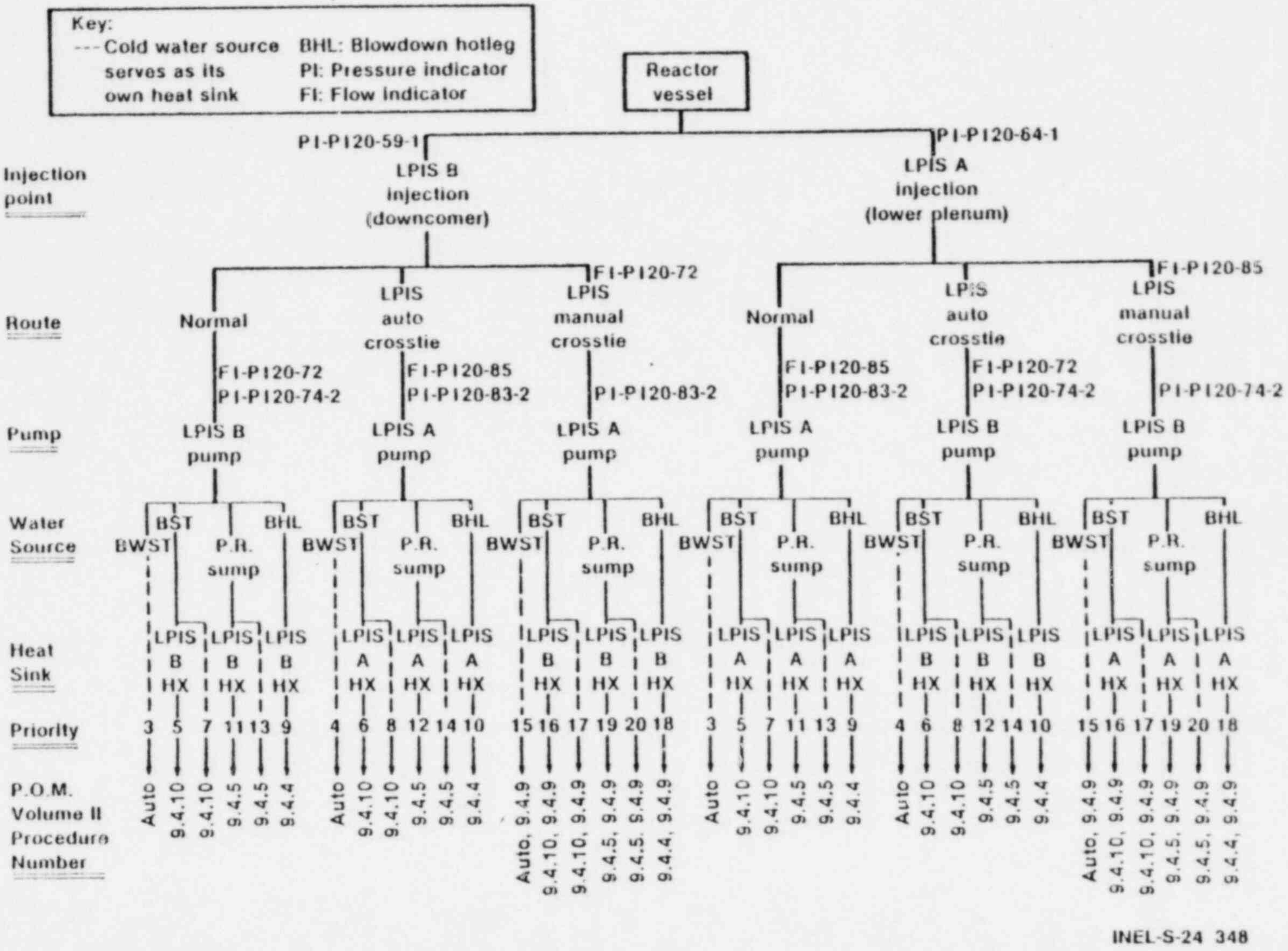
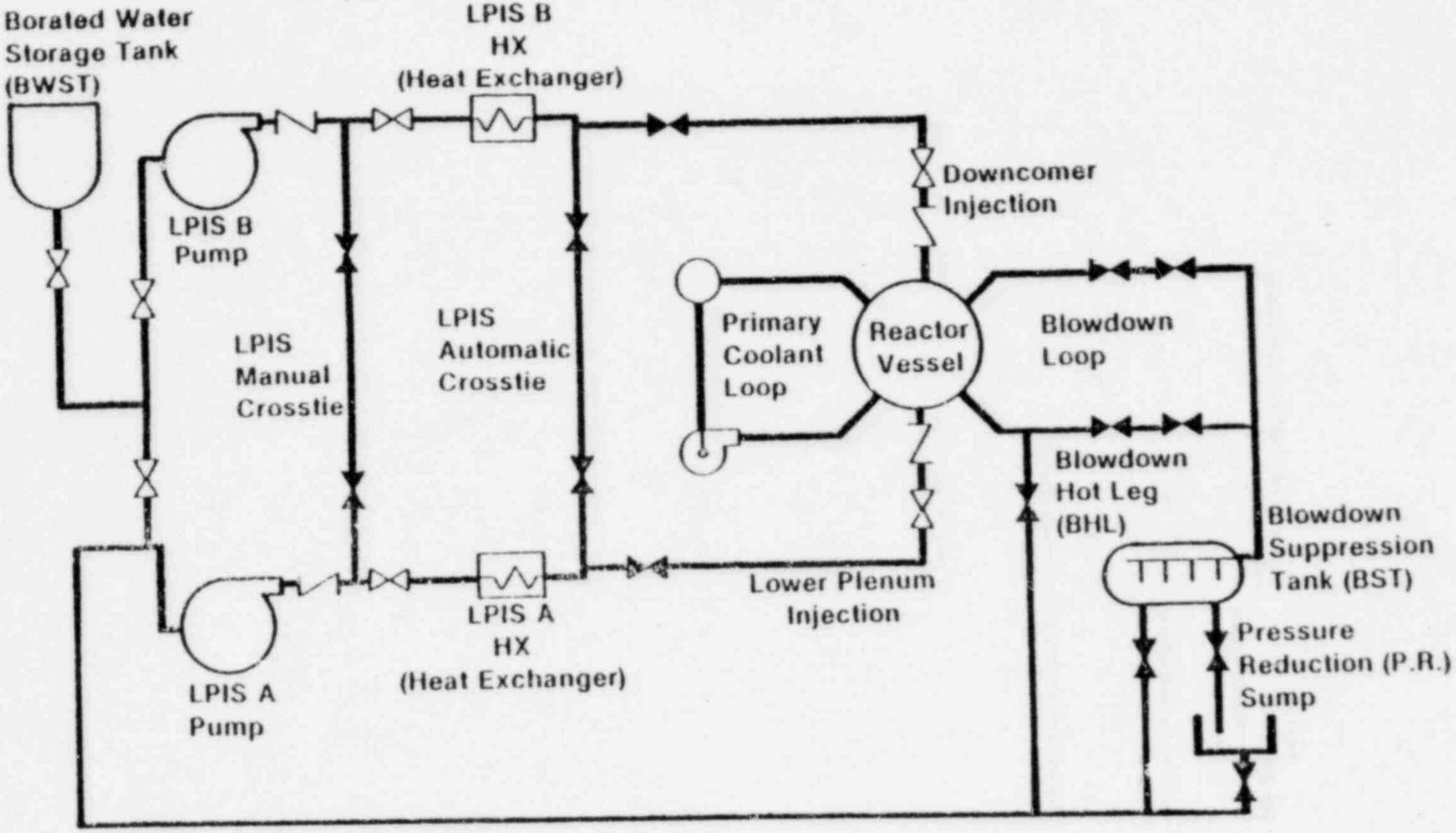


Figure 1. LOFT LPIS Response Tree.



INEL-S-24 346

Figure 2. Simplified Schem[illegible] LOFT Low Pressure Injection System (LPIS).

The five elements are shown on the various levels of the response tree. Each path from the bottom of the tree to the top represents a different cooling mode. At the bottom of each path (cooling mode) is listed a priority number and a reference to the appropriate procedure(s) in the LOFT Plant Operating Manual (POM). Priority numbers were established by evaluating the relative desirability of the cooling modes in terms of cooling effectiveness, difficulty of implementation, and other similar considerations. Cooling modes with small priority numbers are most desirable, and cooling modes which may be initiated automatically are so labeled. At appropriate points on each cooling mode are listed pressure (PI) and flow (FI) instruments which can be used to monitor the performance of the cooling mode.

USE OF THE PROCEDURE

Following the onset of the accident, the operator immediately refers to the procedure to determine an appropriate course of action. Using his current knowledge of system status, he crosses out or otherwise indicates any components which he knows to be disabled. He then does the same for all priority numbers of cooling modes which require the use of a disabled component. Next, he selects from the remaining cooling modes the one(s) with the smallest priority number, refers to the listed procedure(s), and performs the prescribed actions. For example, if LPIS pump A fails to start, a pressure indicator in the downcomer injection line indicates that flow is not reaching the reactor vessel, and the Borated Water Storage Tank (BWST) is empty, he selects the cooling mode with priority number 6, refers to POM procedure 9.4.10, and performs the appropriate actions (see Figure 3). As time progresses and other components are disabled or restored, he continually updates the response tree to ensure that the optimum cooling mode is being implemented.

COLOR GRAPHICS DISPLAY

A color cathode ray tube (CRT) display is being developed for this procedure in conjunction with the LOFT Augmented Operator Capability Program. Figure 4 shows the display as it will look for the example accident. Unavailable components will be shown in magenta, available components will be shown in dark blue, and the recommended cooling mode will be highlighted with double-width lines in cyan. A computer will be used to monitor system status, evaluate the response tree, and generate the correct CRT display for the recommended response.

ADVANTAGES OF THE TECHNIQUE

The following strengths have been noted in the development and implementation of this technique at the LOFT facility:

- It provides a systematic method for identifying all potential cooling modes, establishing their relative priority, and displaying this information for operations personnel.

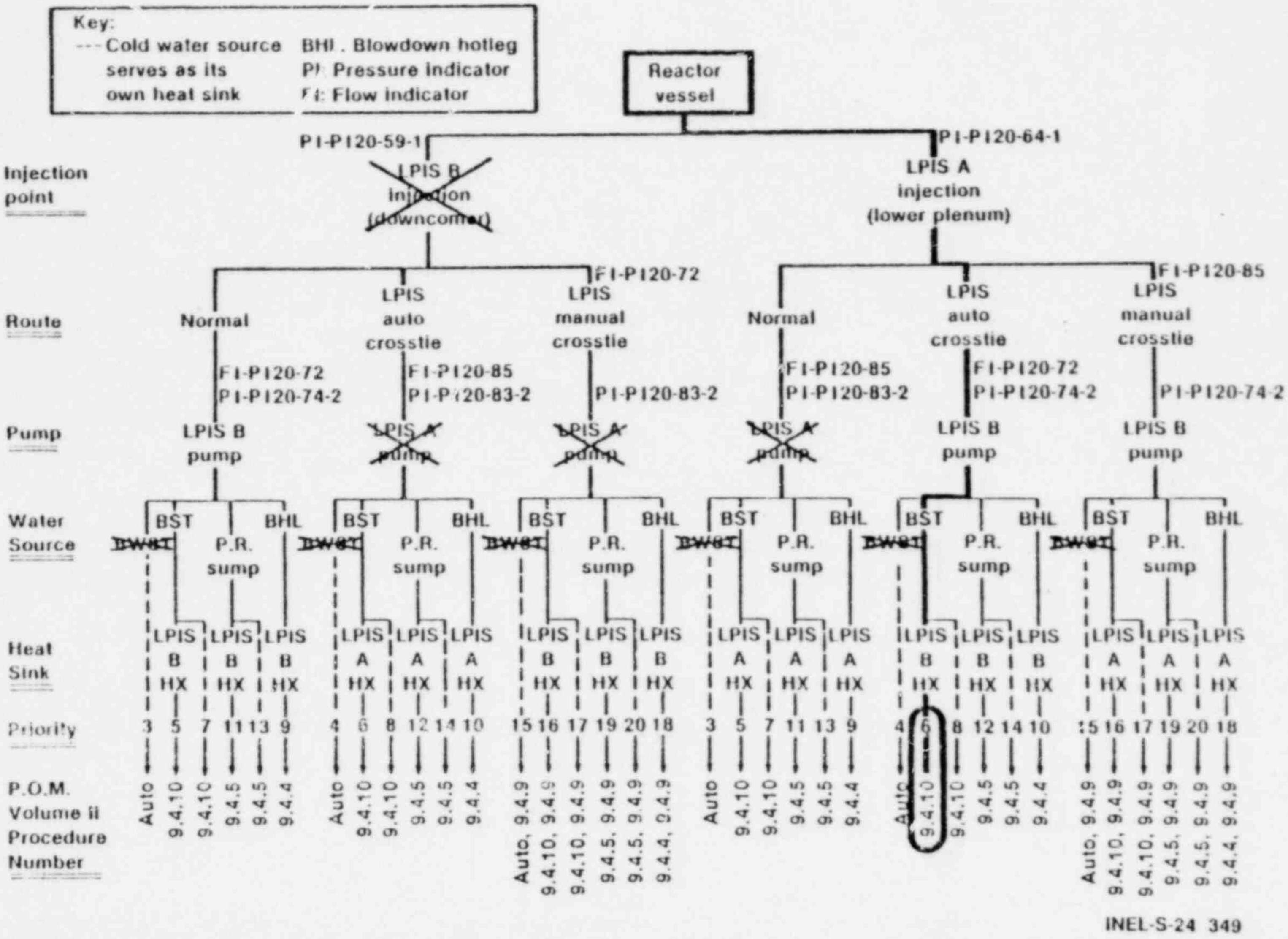
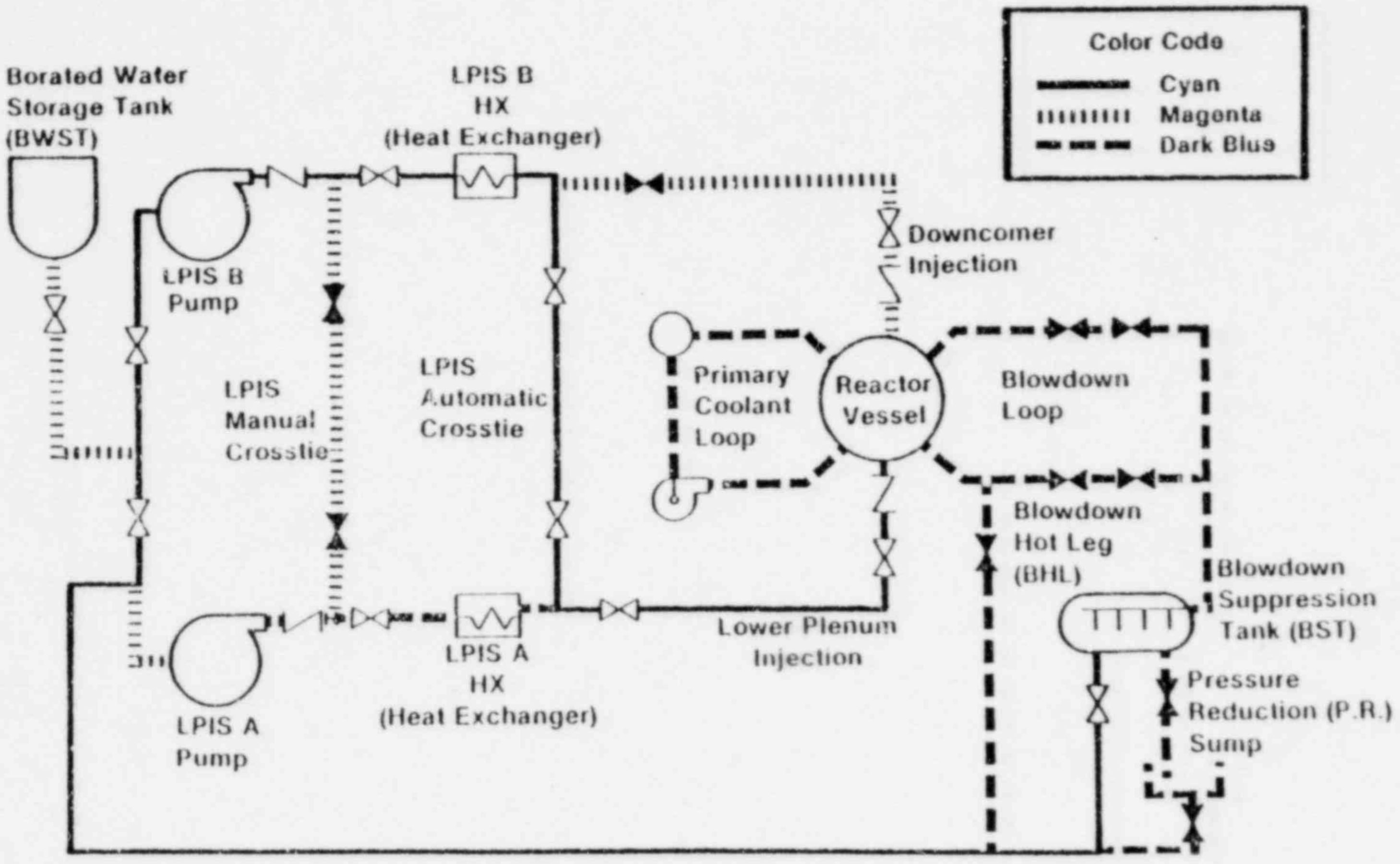


Figure 3. Choice of Cooling Mode for Example Accident.



EL-S-24 346

Figure 4. Color CRT Display for Example Accident.

- Rather than requiring the operator to refer to the entire POM for an applicable procedure, it provides a central point from which he is referred directly to the correct procedure.

- It improves operator familiarity with all potential modes for cooling the reactor and the interrelationships between plant systems and components.

- It is relatively simple and inexpensive to implement.

- The trees are easily modified if facility modifications occur.

CONCLUSION

The use of this technique for accident response can provide the immediate actions necessary to bring the system under control. Sophisticated fault-isolation techniques could then be used to determine the exact cause of the accident and optimize the ultimate recovery of the facility. Thus, response trees could prove to be an important element in responding effectively to nuclear reactor accidents.

ACKNOWLEDGMENTS

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, or any of their employees, makes any warranty, expressed or implied, or assumes any legal liability or responsibility for any third party's use, or the results of such use, of any information, apparatus, product or process disclosed in this report, or represents that its use by such third party would not infringe privately owned rights. The views expressed in this paper are not necessarily those of the U.S. Nuclear Regulatory Commission.

Work supported by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research under DOE Contract No. DE-AC07-76ID01570.

The author is grateful to Mike Clark for his suggestions concerning the format of the CRT display.

LOFT Experimental Program and Testing Sequence

E.A. Harvego

INEL-S-29 291

Harvego (

Content

- LOFT testing accomplishments
- New test series
- LOFT testing sequence
- Continued planning efforts

INEL-S-29 389

LOFT Test Series

- L1 Non nuclear
- L2 Large break series
- L3 Small break series
- L4 Alternative ECC
- L5 Intermediate breaks
- L6 Anticipated transients - operational transients
- L7 Steam generator tube failure

New LOFT Test Series Objectives

L8 series - severe core transients

- Investigate transients resulting in core uncovery and ultimately fuel damage

L9 series - anticipated transients with multiple failures

- Perform experiments with high probability of occurence or severe consequences

L10 series - override plant protection mode

- Determine override transient that can shut a reactor down safely under all conditions

Factors Influencing LOFT Test Sequence

- Instrument requirements
- Facility modifications
- Operating requirements
- Test severity
- Fuel availability
- Experiment safety analysis

INEL-S-29 388

LOFT Testing Sequence
(Phase I)

- Resolution of licensing issue
- Code and system qualification
- Initial core uncovery experiments
- LOFT typicality in simulated LPWR upsets

LOFT Testing Sequence
(Phase II)

- Coupled effect of fuel behavior and integral system thermal-hydraulics
- Fuel damage criteria
- Release, transport, and deposition of fission products under very realistic conditions

LOFT Testing Sequence
(Phase III)

- Multiple failures (common mode/cause)
- ATWS
- Controlled core damage
- Fuel ballooning/core blockage

LOFT Testing Sequence
(Phase IV)

- Efficiency of ECC systems
- New ECCS concepts
- Steam generator tube breach

LOFT Testing Sequence (Phase V)

- Override capabilities
- Severe core damage
- Containment integrity
- Facility cleanup

Continued Planning Effort

- Identify testing needs
- Reassess current test plan
- Modify test plan to reflect testing needs

L4 Series - Alternate ECC

Experiment ID	Description	Priority
L4-1	Accumulator ECC injection into upper plenum	High
L4-2	LPIS injection into upper plenum; (scaled to two-loop W plant)	High
L4-3	C.L. ECC injection with B&W vent valve simulated	High
L4-4	All ECC injection into lower plenum	High
L4-5	ECC injection into intact loop pump suction	Medium
L4-6	All ECC injection into lower plenum. Accumulator set point = 1000 psi	Low

INEL-S-29 386

Aspects of Planning Approach

- Identify user needs
- Determine user interest levels
- Define LOFT testing capabilities
- Match LOFT capabilities to user needs/interests

Potential Interest Levels for Users of LOFT Information and Results

POOR ORIGINAL

LOFT Test Purpose	Type of Test or Operation	General Public	NRC	ACRS	State Regulatory Agencies	Vendors AEs, and Consultants	Utilities an Related Study Groups	Universities	ANS and other Professional Groups	Foreign	DOE
Understanding cause of upsets and accidents	• Human error and component failure										
Understanding course of upsets and accidents	• Increased secondary heat removal • Decreased secondary heat removal • Decreased primary flow • Reactivity/power dist. anomolies • Increased coolant inventory • Decreased coolant inventory • Radioactive release • ATWT • Loss of support systems										
Understanding consequences of upsets and accidents	• Plant availability • Fission product release • Negative public response										
As an off-normal Training center	• Training for recovery from severe upset • Optimize emergency response procedures										
As an equipment qualification facility	• Qualification of equipment • Development of new equipment										

Interest Level
Strong
High
Mild

14

Event Tree Evaluation

- Group transients exhibiting similar behavior
- Identify unique transients
- Recommend specific transients covering range of possible plant responses
- Prioritize transients (severity, probability, uncertainty)

Potential
High Risk Transients
(Probability X Consequence)

Reg. Guide 1.70 Category	Transient
1	Inadvertent opening of steam Generator Valve*
1	Large steam line rupture
2	Loss of all AC power^T
2	Feedwater pipe break
3	Decrease in reactor coolant flow
3	Reactor coolant pump seizure
4	Uncontrolled rod withdrawal*
4	Rod ejection accident
6	Steam generator tube failure

* Detailed event tree analysis required
^T Event tree analysis complete

INEL-S-29 288

16

Conclusions

- Testing sequence optimized
- Current program plan designed to exploit uniqueness and maximize usefulness of LOFT
- Continued planning systematically addresses needs of nuclear community

INEL-S-29 385

LOFT TEST SEQUENCE (PRELIMINARY)

Harvego

PLANNED LOFT TEST SEQUENCE (SEPTEMBER 1980)

(The Current Sequence Is Being Reassessed Using Criteria Developed From The Long Term Planning Strategy. While The Short Term Test Sequence Is Not Expected To Change, Long Term Test Plans May Be Modified As A Result Of This Assessment)

TEST ID	INITIAL POWER LEVEL (MW)	INITIAL CORE ΔT °F	COMMENTS
L3-5	50	35	Small break (2.5%) intact loop cold leg --- pumps off.
L3-5A	Add on to L3-5		Investigate primary system recovery utilizing steam generator.
L6-2	37	25	Loss-of-power to primary coolant pumps.
L6-1	37	25	Loss-of-steam load (closure of MSIV's).
L6-3	37	25	Excess load increase (cooldown transient).
L3-6	50	35	Small break (2.5%) intact loop cold leg --- pumps on. Pumps tripped at end of experiment to measure water remaining.
L8-1	Add on to L3-6		Core uncovery without ECC at low decay heat level.
L9-1	50	35	Loss of all feedwater (multiple failures) with scram on high pressure; PPS setpoints representative of LPWR (PORV challenged.) Mild ATWS.
L3-3	50	35	Small cold leg break (0.16%) HPIS flow approximately equal to break flow. Dry steam generator secondary. Determine the boundary between break heat removal and PORV heat removal. Needs further justification.
CV Leak Test			Required test of containment leak integrity.
L6-7	50	65	LOFT typicality to Arkansas Nuclear One startup test.
L9-2	Add on to L6-7		Rapid cold water accident, upper plenum voiding.

PLANNED LOFT TEST SEQUENCE (CONTINUED)

TEST ID	INITIAL POWER LEVEL (MW)	INITIAL CORE ΔT °F	COMMENTS
L5-1	50	65	Intermediate size break (accumulator line). Determine if large break and small break models continue to predict intermediate break results. Also check out liquid level device.
L8-2	Add on to L5-1		Core uncovery at high decay heat level. Reflood with degraded ECC capability. May be the same as L5-1.
Whole core Changeout			F1 center bundle at 350 psi (BOL). Large peaking factor if only CB changed.
L2-5	16 kw/ft	65	Worst prototypic hydraulic conditions in core. Investigate fuel behavior at BOL fuel pressure (no fuel damage expected).
Replaces CB F1 with F2			F2 will be pressurized to 700 psi.
L2-6	16 kw/ft	65	Same as L2-5 with 700 psi fuel pressure (EOL). Fuel damage and fission product release expected.
Replaces F2 with unpress A1			Only minimal fuel damage experiments can be done until F1 is examined for damages.
L5-2	16 kw/ft	65	Intermediate size break on hot leg. Pressurizer surge line. Needs further justification based on L5-1.
L6-4	16 kw/ft	65	Uncontrolled rod withdrawal at power. Investigate worst case moderate frequency accident.
L9-3	16 kw/ft	65	ATWS. Loss-of-Feedwater is initiating event. (Multiple failures.)
L9-4	16 kw/ft	65	ATWS. Loss of offsite power is initiating event. (Multiple failures.)

PLANNED LOFT TEST SEQUENCE (CONTINUED)

TEST ID	INITIAL POWER LEVEL (MW)	INITIAL CORE ΔT °F	COMMENTS
Put F1 Bundle Back In			F1 inspection completed and fuel is assumed not damaged.
L8-3	16 kw/ft	65	Small break with slow core heat up (1°F/min). Uniform clad swelling and blockage of flow channel. Investigate potential initiating events. (Candidate: Loss-of-Feedwater.)
Replace F1 With A3			
L7-1	16 kw/ft	65	Large break with S.G. tube ruptures at start of reflood/refill (>25 tubes ruptures). Provides upper bound of envelope on effect of ruptures. Critical number of tube ruptures resulting in extreme core temperatures expected to be between 10 and 25 based on Semiscale results.
L7-2	16 kw/ft	65	Large break with S.G. tube ruptures at start of reflood/refill (<10 tubes ruptured). Provides a lower bound of envelope on effect of ruptures. L7-3 should be inserted if possible which has critical number of ruptures.
L4-1	16 kw/ft	65	200% cold leg break. Accumulator injection into U.P. Investigate topdown core quench. Applicability to UHI plants.
L4-2	16 kw/ft	65	200% cold leg break. U.P. LPIS injection. Investigate W two loop plant phenomena.
Replace A3 With Press F3			
L8-4	16 kw/ft	65	Severe core damage. Investigate potential initiating events. (Candidate: Loss of offsite power.)

PLANNED LOFT TEST SEQUENCE (CONTINUED)

TEST ID	INITIAL POWER LEVEL (MW)	INITIAL CORE ΔT °F	COMMENTS
Whole Core Changeout			F4 Center bundle.
L10-1	16 kw/ft	65	Override test. Override of L8-3 transient.
L10-2	16 kw/ft	65	Override test. Override of L8-4 transient.
L8-5	16 kw/ft	65	Severe core damage. Investigate potential initiating events. (Candidate: Steam line rupture.)

OPERATIONAL TRANSIENT RESEARCH

. HAVE BEEN PART OF RESEARCH EXPERIMENTAL PROGRAM

. LARGE-BREAK HAVE HAD HIGHER PRIORITY BEFORE TMI EVENT

. COORDINATED RESEARCH PROGRAM INVOLVING LOFT, SEMISCALE AND SEPARATE EFFECT EXPERIMENTS

. LOFT HAS COMPLETED FOUR (4) OPERATIONAL TRANSIENT EXPERIMENTS

. SEMISCALE HAS COMPLETED STATION BLACKOUT EXPERIMENTS

. NRR SUPPORTS OPERATIONAL TRANSIENT RESEARCH

. PROGRAM TO PROVIDE A DATA BASE FOR CODE ASSESSMENT

Sullivan