NUCLEAR REGULATORY COMMISSION

T-Ø883

In the Matter of: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

255TH GENERAL MEETING

DATE: July 10, 1981 PAGES: 140 - 347

AT: Washington, D. C.

TRO4
s
1/0
Delete:
B. White

TRØ4
delete
B. White

ALDERSON REPORTING

400 Virginia Ave., S.W. Washington, D. C. 20024

Telephone: (202) 554-2345

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

255TH GENERAL MEETING

- - -

Nuclear Regulatory Commission
1717 H Street, N.W.
Room 1046
Washington, D.C.

Friday July 10, 1981

The 255th meeting of the Advisory Committee on Reactor Safeguards was convened at 8:30 a.m.

MEMBERS PRESENT:

J.C. MARK, Chairman
P.G. SHEWMON, Vice-Chairman
C.P. SIESS
D.W. MOELLER
M. BENDER
W. KERR
M.W. CARBON
W.M. MATHIS
D.A. WARD
D. OKRENT
J.J. RAY

DESIGNATED FEDERAL EMPLOYEE:

R. MAJOR

ALSO PRESENT:

J.M. JACOBS, Secretary

Present for the NRC and Industry:
Mr. Clark
Mr. Hukill
Mr. Novak

Mr. Arnold
Mr. Broughton
Mr. Croneberger
Mr. Behrle
Mr. Tedesco
Mr. Phillips
Mr. Chestnut
Mr. Rogin
Mr. Chow
Mr. Stolz
Mr. Chisholm
Mr. Tadani
Mr. Broughton
Mr. Croneberger

P R O C E E D I N G S

(8:30 a.m.)

MR. MARK: The meeting will now come to order. This is a continuation of the 255th meeting of the ACRS.

During today's meeting the Committee will hear presentations on and discuss the restart of TMI Unit 1, discuss future ACRS activities, the application of the TMI-2 Action plan to the Pilgrim Station, and will continue preparation of ACRS reports.

A portion of the session of the Safety Research Budget may be closed as necessary to discuss information the premature release of which would be likely to significantly frustrate Committee action. Also, some portions of the session on preparation of our reports may be closed if necessary to discuss privileged information related to the matters being considered.

Mr. Richard Majors is the Designated Federal Employee for this portion of the meeting.

We have not received any written statements or requests to make oral statements from members of the public regarding today's sessions.

A transcript of portions of the meeting is being kept and it is requested that each speaker first identify himself or herself and speak with sufficient clarity and volume so that he or she can be readily heard, which means trying to avail oneself of the microphone.

We will now begin the meeting, and the first item today is to hear the state of discussions on the possible restart of TMI-1. I understand that some of the Met Ed people, who were coming down extra-fast by private plane, haven't arrived yet, but are expected to be here shortly. But I believe we have the basis for starting off.

Dade, do you want to take this over?

MR. MOELLER: Thank you, Mr. Chairman. I wanted to provide the Committee with the report of the Subcommittee's view. The Subcommittee held a meeting on June 25 and 26 to review modifications made to the plant and to consider their application for restart.

This was the third such meeting at which these issues had been discussed. A previous Subcommittee meeting had been held on January 31st and February 1, 1980, in Middletown, Pennsylvania; and again, we had a Subcommittee meeting on November 28 and 29, 1980, here in Washington, D.C.

Attending the latest Subcommittee meeting were William Kerr, Charles Mathis, and Harold Etherington, in addition to myself. And we had at this Subcommittee the following consultants, who assisted us in the review: I. Catton, W. Keyserling, W. Lipinski, and Z. Zudans.

As background for the current review, I wanted to note that the Committee in its interim or status report that it entered on December 11, 1980, mentioned five items that they wanted to closely follow, and you may want to give these attention during today's meeting:

One was the status of reliability assessments of the plant as modified;

Two was the instrumentation for assessment of inadequate core cooling;

Three was the instrumentation for monitoring the position of the pressurizer PORV;

Four was thermal mechanical effects of high pressure injection on the reactor pressure vessel integrity for a small break LOCA with no emergency feedwater flow;

And the fifth item was the consequences of D.C. power failure.

Also at our Subcommittee meeting, we not only addressed each of these items, but we addressed the matter of the additional remarks in our December 11, 1980, letter relative to the need for studies on hydrogen control and filtered venting containment systems.

In the way of background, I might also mention that TMI-1 is being treated as an operating reactor for purposes of restart. It is not being treated as an NTOL.

In terms of the Subcommittee meeting, the latest Subcommittee meeting, I wanted to mention several items.

One was the instrumentation for the assessment of inadequate core cooling. This situation, as you know, is a generic one, and the problem for TMI-1 is much the same as it is for other plants. The TMI group has reviewed the full range of possibilities and weighed the advantages and disadvantages of each. However, they oppose such instrumentation, specifically an RPV liquid level gauge. But the NRC's position is that such will be required of them on a schedule which they will tell you, I'm sure, this morning.

The Subcommittee members quizzed the NRC staff regarding the decision to require a higher PORV set point and a lower trip set point, the end of that being to reduce the frequency of PORV openings. Again, this is a generic matter which is still under dispute. And we still were unable to obtain any real information on a quantitative estimate of whether this does reduce risk and by how much.

Another question that came up in the Subcommittee meeting was the setting of the containment purge valves, which are set at 30 degrees for regular operation. And then the staff -- or the tech specs are written to say that 4 psig containment pressure will be one of the signals for containment isolation. Several of our consultants questioned whether a containment with the purge valve set at 30 degrees would ever reach 4 psig, and therefore was this particular signal of any value in terms of containment isolation.

The TMI-1 group has responded to most of the studies requested in our December 11, 1980 letter, except those in the additional comments. That is again, hydrogen control and vented filtered containment. In general, we concluded that a more positive demonstration of progress should be encouraged. If a problem is generic, the Licensee appears at times to be satisfied to leave the resolution to others, namely to groups like EPRI or the vendors. And we tried in our Subcommittee meeting to point out to them that we desired more action independently on their part in resolving these issues.

Our review of the human factors aspects of the control room left some items unresolved. Some instruments upon failing will read at mis-scale. Our consultant thought that was not a good policy. Certain dial settings are inconsistent with conventional approaches, meaning that the needles move in different directions than what would normally be expected. Improvised labels are to be controlled administratively. We weren't sure that that was the best approach there.

We felt that security needed special attention, in view of the continuing activities to decontaminate Unit 2. A comprehensive review of the security matters at TMI-1 had been done independently by the Los Alamos National Lab, and the Licensee is moving to implement these recommendations. We covered this in a closed session during the Subcommittee meeting and we did not plan it to be on the agenda today for the full Committee.

Emergency planning we reviewed, particularly the drill held early in June, which the reports we received indicated that it was quite successful. FEMA has conducted an independent assessment of these aspects and is providing information to the NRC staff on emergency matters. They have a plan for a startup test program at TMI-1 and it's comparable, with some differences, to that which you have already reviewed for Sequoyah and other plants.

I would like also to bring to your attention one other matter which was discussed at the Subcommittee meeting, and that is the House Committee report which, after an in-depth study of the TMI-2 accident, indicated that it was their belief that the Licensee willfully withheld information on the severity of the TMI-2 accident from federal and state officials. The report, which you have also seen, by Ed Abbott indicated a somewhat similar conclusion.

Now, the report of the NRC I&E group, on the other hand, as I understand it, attributes the lack of information to a number of causes, but ascribes a predominant amount of it to the confusion that accompanied the situation.

In planning for the restart of TMI-2 and whether we would issue a favorable letter on the Licensee's request, we have to look at this matter. Now, the Licensee has made a number of organizational changes and has issued a directive to their staff which was read to us during the Subcommittee meeting indicating and requiring that if an accident should occur in the future that they are to provide state and federal officials with complete and accurate information.

So our task today as I see it, among many other things, will be to assure ourselves that steps taken by the Licensee are adequate to assure a free flow of accurate information should TMI-1 restart and should at some time in the future some type of an accident occur.

MR. OKRENT: Is it our task somehow to arrive at some judgment as to what was the situation around the time of the TMI-2 accident? Is that in issue here?

MR. MOELLER: Speaking personally, a personal opinion, I do not feel qualified in terms of being a lawyer, a criminal justice, or whatever it would be, to make that decision. However, I do believe firmly that it is our responsibility to assure ourselves that whatever the situation was, that steps have been taken to correct it.

MR. OKRENT: Does the staff feel that from their point of view that a decision with regard to just what transpired or did not transpire during the TMI-2 accident is somehow a go or no-go situation or bears directly on their view of restart of TMI-1?

MR. MOELLER: Tom Novak?

MR. NOVAK: Tom Novak from the staff.

Certainly the staff has reviewed the event in much detail. There is a comprehensive report and position by the Office of Inspection and Enforcement. Those views represent the views of the staff.

It's my opinion that as we have gone through the TMI-1 restart that it's a separable issue in the sense that we have been asked to judge the adequacy of the management of the organization that would operate TMI-1, the emergency planning and all other features necessary to cope with a possible accident, and to ensure ourselves that those provisions that are being provided by the Licensee satisfy the requirements of the Commission and the requirements that the staff believes are a correct interpretation of the Commission's order for restart.

So in my view this issue is not directly appropriate for decision on the restart of TMI-1. I believe that there has been a lot of work, and I think in the Subcommittee meetings this has been voiced by the staff, in trying to understand the conditions and the events that occurred immediately following the accident. And we have tried to make our views as clear as we can and they are there for the Commission to appreciate and to use as they feel appropriate.

I don't think I have any other comments to make at this time.

MR. MOELLER: Dr. Plesset had a comment, and then Mr. Bender.

MR. PLESSET: I was going to wait until you finish the report.

MR. MOELLER: I have finished it. The one item remaining is I'll call on the other Subcommittee members who were present.

MR. PLESSET: I will make very briefly a few comments. I am sure that you followed, as did the staff, how the action plan was implemented at TMI-1. I just wanted to say, I don't believe that just implementing the action plan means the plant will operate safely. I think there is too much thought given to that as a cure-all. That's one remark I was going to make.

The other one is, I couldn't care less about an analysis of filtered vented containment at TMI that management would bring in. I don't think that's particular of concern as far as safe operation of the plant goes.

My point I want to make is, I think that the accident at TMI-2 has put too much blame on operators. I think the basic point -- and this has been made in some reports and reviews of the subject by the management -- I think the central point, more important than the other things I have just mentioned, is how is the management there? Is it really going to be competent, because if it isn't it's a very serious thing to let these people operate a nuclear plant.

If it's been fixed, that's worth knowing. If it hasn't been fixed or has weaknesses, I think we have to know that, and I think this is a central issue. I don't think we want to get into legalities. That's my point. All we're concerned about is is this group of people fit to run a plant, because aside from any changes that might have taken place the answer would ordinarily be expected to be know, unless there's been really drastic changes, and one would want to have a very definite demonstration of this. And I should think both the Committee and the staff would need to be reassured very strongly on this point.

That's all I wanted to say.

MR. MOELLER: Thank you.

Mr. Bender?

MR. BENDER: I wanted to address the matter of communications, because it seems to be the main thing on this particular plant operation. I don't think it's unique to TMI-1. In rehashing the circumstances at TMI-2, it hardly seems to be appropriate.

It seems to me that it's a two-way street. Undoubtedly we need to make sure that the Applicant, the Licensee, knows what his obligations are and that probably ought to be explored some if it hasn't been. On the other hand, I haven't learned enough about the capabilities on the other side to know that the information provided is and will be used knowledgeably.

There's some suggestion that a great deal of information should be imparted to the state authorities and the NRC. We have had some searching discussions as to how the NRC should use the information which it receives and how much it should receive. Now, I find a total lack of understanding as to whether there is an NRC position on it, whether the Commissioners have sanctioned the knowledge that the NRC is requiring. And I have some doubt as to whether the state officials have the kind of competence that would enable them to deal with detailed information on the operating status of a plant.

And so I think we need to think some about how we will respond to a question of adequate communications.

MR. MOELLER: Thank you.

Let me call upon Mr. Kerr or Mr. Mathis. And I guess Mr. Etherington is not here. Mr. Kerr, did you have any comments?

MR. KERR: I would say that we have heard from the Applicant in the number of meetings that you have mentioned, a considerable amount of detail about the reorganized management. It involves both new people and new organizations, and that is part of our consideration.

It seems to me that efforts have been made both to improve the organization and to add to the capability of both management and the operating organization, and I would judge that these changes represent significant improvements.

MR. MOELLER: Thank you.

Mr. Mathis?

MR. MATHIS: I might make just one additional comment that follows up on what Mike had to say. In our review in the Subcommittee meeting there were discussions concerning the ability of the state people to know enough to ask the right questions during an emergency, and I think there is an admission on their part, as you pointed out, that there is a deficiency there and they've got some work to do, too. I think that's all I have to add.

MR. MOELLER: Thank you.

Dr. Okrent?

MR. MARK: You mentioned that the containment isolation, I believe it was, was set at 30 degrees.

MR. MOELLER: The purge valves.

MR. MARK: Do they close at 30 degrees? Is that Fahrenheit or Centigrade or what?

MR. MOELLER: No, that's the angle of the setting.

MR. KERR: Zero degrees is closed, I think. 30 degrees is 30 degrees open.

MR. MARK: Okay.

MR. SHEWMON: Is it half open?

MR. MARK: That's about half of it.

MR. KERR: The idea is that they will be able to close against the flow that would result in an accident if they're set at this point, whereas they probably might not be able to close if they were set at a more open position, I think. That's my understanding.

MR. MOELLER: Mr. Okrent?

MR. OKRENT: I guess I have a different view of what requirements it's appropriate to place on the technical capabilities of people within a state organization responsible for managing emergency measures. I don't think it's appropriate that they be asked to be experts in each and every hazard that exists in society.

It seems to me that people who are running the facility, whether it's an LNG facility or a facility that makes dioxin for nuclear reactors, that people who run it should be able not only to assess the status of the plant, but also to explain in language intelligible to the layman what the potential hazards are and what might occur and so forth, so that those people who have the responsibility for taking emergency action have a basis for decisionmaking.

If in fact a group running the facility is unable to provide this information, that's where the deficiency lies and it has to be remedied.

MR. BENDER: I think Dr. Okrent is taking issue with the point I made and I would want to respond to it. Of course it's important that the organization running the plant has knowledge enough to provide to the public some understanding of accident circumstances. But it is a public governmental authority that issues orders to the public, and it can't do it without adequate knowledge.

And I don't care how capable the operating organization is to develop information. If the recipient is not adequately knowledgeable to use the information, he is likely to lead the public in the wrong direction. And I think we cannot assume that all the burden has to be on the operator.

MR. PLESSET: I would like to side with Dave on this. I think you should not expect political persons who have been successful in running for office, or their appointees, to make these judgments. What it boils down to is in my opinion, and I think Dave is saying somewhat the same thing, it devolves upon the management. And beyond that, there has to be some confidence and trust in that management. And if you don't have that, you're really in trouble and you're not going to get some officials in the government to act responsibly if they don't have that confidence.

The technical decision has to be made by the management of the facility, whether it's an LNG or nuclear plant or whatever.

MR. BENDER: It's not up to the management of the facility to decide whether the public is in sufficient jeopardy to recommend that they move from their living quarters somewhere else. That's the job of the public authorities.

All the organization running the plant is report the conditions. The interpretation of it is a public responsibility. And I can't accept the fact that because an individual is politically designated that he doesn't need competence. And I think that message needs to be communicated back to those that are concerned about it.

MR. SHEWMON: Dade, I too could take sides on this and feel reasonably strongly. But I think we would be better if we got on and we could argue this in a letter or something.

MR. MOELLER: All right. You have in your folder

-- does someone know what section -- tab 8, an agenda for today's review of this restart matter. The first item on that agenda is review of new management organization and the resultant increase in staffing. This will be a presentation by the Licensee.

Before I call on them, let me ask, does the staff have any comments?

MR. NOVAK: On your summary, you mean?

MR. MOELLER: Yes.

MR. NOVAK: I don't believe so, no.

MR. MOELLER: Thank you. Then we'll move to the Licensee, and for the review of the first item we will call upon Mr. Robert Arnold.

MR. ARNOLD: Thank you very much, Dr. Moeller. The company appreciates very much the opportunity to be before this Committee.

I think with regard to some of the immediately preceding discussion I would like to make a couple of comments. First of all, that the company does not feel that it's their job to, I guess, make the judgment on the issue that the Committee was immediately discussing. However, we have made the internal commitment and have communicated that externally as well, that our intent and our sense of our responsibility is to be fully forthcoming, fully straightforward in our communications with both the state and federal agencies. And we believe that we have set up our organizations and trained our people such that that will in fact happen.

I think that we have recognized since very shortly after the accident that in terms of the restart of Unit 1 that the capability of the management, of the organization, and the confidence of others in that capability was an absolutely essential ingredient for authorization to restart Unit 1. And we do believe that we have demonstrated that we have been responsive to that concern.

In terms of the presentations, Mr. Philip Clark will start off with the review of management organization. We would ask the Committee's indulgence in rescheduling the next item to immediately after the break. The person we had intended to make that presentation became ill last night and so is not available this morning, and we would like a little bit more time to readjust on that. But other than that, we are ready to proceed in accordance with the sequence of the agenda. And Mr. Clark will make the opening presentation and will chair our remaining presentations.

MR. MOELLER: We have noted that and we will simply postpone the one item.

MR. CLARK: Starting before the TMI accident, GPU management recognized that nuclear power was different and we had started to revise our organization to deal with nuclear stations. That plant was set back by the accident and then accelerated. And I want to show you today the results.

(Slide.)

This slide shows some of the evidence that we incorporated in the organization of the staff. First we had set up a full-time organization dedicated solely to nuclear generation. That organization was put in effect and recognized in our licenses with NRC in September of last year and has been operating ever since.

It is operating today as a nuclear group. It is intended that it be a nuclear corporation. That change from group to corporation requires approval by SEC, Pennsylvania PUC and New Jersey BPU, and the NRC. We have two of those four and we expect the others this summer.

But even without being a corporation, we are operating as a group with the organization staffing as I am going to describe it to you. We have increased -- and I will show numbers later -- the onsite technical and management resources. We have set up strong central technical control at the plant, with continuity from the design through the construction and the control of the technical configuration of the plant during operation. Prior to the accident that responsibility changed hands with time.

We have put full-time on-site management for operation and maintenance of each plant, and we have taken the support activities, administration, engineering, and radiation protection, and put those separately in the organization so that the plant management can devote themselves to the operation and maintenance.

We have set up an independent nuclear assurance division, with training, quality assurance, and a nuclear safety assessment department independent of the operating organization and the engineering. We have pooled the resources that were applied to TMI-1 into Oyster Creek and in our case, very importantly, which were applied to the Forked River plant. We have pooled them all and augmented them substantially since the accident, and we are proceeding to develop and implement personnel policies and procedures which are appropriate for nuclear power.

And as an example of that is the requirements we are establishing for periodic requalification of radiation protection technicians and other classifications, so that we are able to continually assure that those people are up to date and qualified to perform their duties.

(Slide.)

Before I show you the organization, I want to show the purpose of the GPU Nuclear Group as we have promulgated it to the organization. That purpose is: first, to manage and direct the nuclear activities, provide the required high level of protection for the health and safety of the public and the employees; second, consistent with the above, to generate electricity in conformance with the rules and regulations. This is an attempt on our part to make sure that the organization understands unequivocally where the organization is with regard to the priority to be placed on protection of public health and safety.

(Slide.)

This next chart shows the organization of GPU Nuclear. The office of the president, with Bob Arnold and me. We have a vice president on site for operation and maintenance of each of the nuclear plants. We have a vice president for technical functions, in whom resides the technical control of the plant configuration, the technical content of the procedures, the technical content of the training. We have a vice president for nuclear assurance, who has the training and the nuclear safety assessment function, and that means that since the accident we have a vice president of communications. Before the accident there was one public relations media person working on TMI. Today there are on the order of 30. In addition, the professional level of the communications staff is vastly increased.

We have a vice president for radiation and environmental control, reporting to the office of the president independent of the operating staff, to provide the kind of quality aspect of the radiation protection. We have a vice president of maintenance and construction to deal with the ongoing modifications, the craft labor, the buildup for outages, and relieve the onsite plant management of the responsibility of carrying out those outages, so they can concentrate on the operation and maintenance of the plant.

We have a GORB, general office review board, which is a safety review board reporting to the office of the president. It has both inside and outside members. There are roughly five members of that 11-man group who are not employees of the organization, but come from outside, from a variety of backgrounds and disciplines. And we have established a full-time chairman for the GORB and provided staff support to see that that function can operate effectively.

There are 12 boxes on his chart at that level. Five of them designated by these dark lines are new to GPU since the accident. Five of them are shifted from construction, which was Forked River, to operational activities. Two of them before the accident had responsibility for nuclear plus generation, and their scope has been narrowed to nuclear activities only. We feel at this level of the organization there is a good deal of new blood and people from outside with a new perspective.

There is a many-fold increase in the amount of management attention at that level applied to the operation and maintenance of our plants.

MR. BENDER: Mr. Clark, before you take that off, the two people that came in to take over the management of TMI-1 and TMI-2 which you show as new to the organization, what was their background?

MR. CLARK: Mr. Hukill, who is here today, spent over 20 years in the Navy in the nuclear business. I think importantly he not only was the CO on the ship, but he spent several years in headquarters, where he was involved in establishing the training programs, monitoring the performance of the operational aspect of the fleet. So he not only was exposed to the system; he then was involved in directing and managing parts of the system, which I think is very important to provide the needed inside.

Mr. Hovey was with General Electric for a while, involved in their nuclear activities, including Vallecitos. And most recently before he came to us he was with AGNS as the manager of the plant that they were designing and constructing for waste processing; and I think is very knowledgeable particularly in the technical things involved in the liquid processing waste management, and I think has a very good background for TMI-2.

MR. BENDER: Fine. Just to complete the picture, there is a new vice president for maintenance and construction. What was his background?

MR. CLARK: Mr. Manganaro was in the Navy for probably over 25 years, including a tour as commanding officer of shipyards which were doing overhaul and maintenance of nuclear ships. And I think he has a very good background for the kind of activity he is involved in for us.

MR. BENDER: Thank you very much.

(Slide.)

MR. CLARK: There are a number of ways to try to capture what capability you have in the organization, and this chart is an attempt to show some of them. Within the organization -- and this is as of December 30th, 1980; the numbers are at the end of last year. We look at the total staff level of 1947. As of June we had 2149. I don't have a breakdown of how they're spread, but there would be increases in essentially all of these numbers by about ten percent, with a total of 200 added.

We had in the organization at that time 416 technical professionals. Those are degreed people in science or engineering. Amongst those people they had 5,000 years of professional experience and 3100 years of nuclear experience. On the average, they had seven-plus years of nuclear experience.

I think another measure over and above the technical background is the operating experience. We have 119 people with a senior reactort operator license or equivalent, by which we mean a Navy engineering officer of the watch qualification.

I think importantly from our standpoint and yours, we see a pretty good spread amongst the three plants. There are technical professionals, 35 to 40, in the staff of the plant for operation and maintenance. I think we also see a good deal of operating experience in things like technical functions, with 35 SRO-licensed people or equivalent, some also in nuclear assurance. So that in addition to the technical expertise, we have the operating experience and background spread through the organization, and we think it's important to have both of those elements.

For comparison, before the accident the total staffing applied to TMI-1 an TMI-2 both onsite and offsite is about 630 people. Today applied to TMI-1 we have onsite 650 people for TMI-1 alone, and offsite it's a little hard to judge. The support people keep changing, but 170 or more people.

So in terms of the total resources applied to the operation and maintenance of these plants, we think we have made major increases in the total staffing most importantly we think in terms of the t nnical background and in terms of management capability, which prior to the accident was not applied to the operation and maintenance of the plants.

MR. BENDER: Mr. Clark, the numbers are impressive. How do they compare now with a similar complement of people that might be at other plants, equivalent to TMI-1 during the startup stages?

MR. CLARK: We're looking very carefully at that and it is very hard to get good numbers, because even in our case, in addition to those numbers, we have applied full-time to our plants on the order of 1200 contractors. And they are onsite or they are dedicated offsite resources.

The mix between employees and contractors, we have been moving and intend to continue to move to bring more in-house. But when you start comparing with anybody else, it's very hard to find out what they have in both categories.

Let's take guards. Amongst the three plants we have 220 security people. A lot of people contract that and they may or may not show up in those numbers.

Given those qualifications, however, it's our belief that our staffing is well on the high side of the staffing for people in an equivalent position. We have looked at the '79 surveys by EEI and other people and we are ell above those numbers. It's our sense that other utilities are also increasing their staffing, but that we are above it.

We do get some input from our outside members of the GORB, people, some of whom may be known to you -- Bill Lowe, Clu Raddis, et cetera. Their sense expressed to us, as I understand it, is that in terms of the total management and staffing, we are well on the high side of other companies.

MR. BENDER: Thank you.

MR. MOELLER: In terms of the 1947 as of December or the 2149 as of June, are these all full-time employees?

MR. CLARK: Yes, sir, full-time employees assigned to the Nuclear Group. And because we are not a corporation, some have a Met Ed payroll or Jersey, but they work for us.

MR. MOELLER: But they are all full-time, your people?

MR. CLARK: Yes, sir.

MR. MOELLER: Okay.

MR. CLARK: That was all that I had prepared.

MR. MOELLER: Any additional questions for Mr. Clark?

MR. WARD: Would you tell us or will someone tell us a little bit more about how the review board functions and interacts with management?

MR. CLARK: Yes. We have a full-time chairman who is an employee. We have three general office review boards, one for each of the three plants. It's a slightly different composition because at TMI-2 we are looking for chemical processing expertise, a little different mix at Oyster Creek, for example.

They have as consultants to them representatives from the NMSS suppliers. Each board meets for about a day and a half every three months. There are formal minutes kept of the meetings. There are formal recommendations from the board to the office of the president, which we respond to formally.

They are provided with staff support from our nuclear safety assessment department here, so that they have the capability, which I think this Committee would recognize is necessary, to have an ongoing continuing effective process, rather than an episodic kind of review that you would have.

They are quite active. Our employees are either division level, which is this level, or the level below it. We have put high-level people on it and we think very competent people from outside.

Does that respond to you?

MR. WARD: Yes, thank you.

MR. MOELLER: Any other questions?

(No response.)

MR. MOELLER: Well, thank you, Mr. Clark.

The next item on our agenda, then, if we postpone the item shown, will be to move into emergency preparedness. And for that we will call upon the NRC staff, who in turn I guess will call upon FEMA for a report. Harley Silver?

MR. NOVAK: Steve Chesnut will introduce our presentation.

MR. MOELLER: Thank you.

MR. CHESNUT: Good morning. My name is Steve Chesnut. I have served as the team leader for the TMI-1 emergency preparedness for the past year, and I wish to report that we have completed our review of the TMI-1 on-site emergency plans in accordance with the new emergency planning rules and the new restart order, and we have recently completed the last item on that restart order, which was the conduct of a test exercise of the TMI-1 emergngcy plan. That was conducted on June 2.

There was one excercise involving not only the licensee but also the state government as well as four of the five counties within the plume exposure emergency planning zone around TMI. The results of that exercise were that the licensee as well as the state and counties demonstrated an adequate preparedness, based not only on that exercise but on the review of the existing plans as well.

York County did not participate in that exercise. They have agreed to or we are in the process of forming up plans for an exercise involving York County, which is scheduled for late August of 1981.

I will also be here to answer any of your questions with regard to the communications you talked about earlier, the provisions with the licensee's plan and also coordination with the state. Also Mr. Robert Jaske from

FEMA, who is the Acting Director of the Radiological Emergency Preparedness Division of FEMA, will be also available to answer questions in those areas as well.

MR. MOELLER: Could we have a statement from Mr. Jaske, did you say was his name?

MR. CHESNUT: Yes, sir.

MR. MOELLER: Simply telling us what FEMA's conclusions were in a sentence or two.

MR. JASKE: FEMA did, of course, participate and evaluate the accident -- or I mean, excuse me, the exercise, and we did submit a statement of findings to the NRC staff on June 16th, which I believe should be available to you. We concur in Mr. Chesnut's statements that the state demonstrated adequate capability to respond to an accident.

Our report on the accident noted 72 deficiencies, which did not detract from our overall conclusion of adequacy. These deficiencies are principally the type of deficiency that can be cured by additional drilling and sub-exercise refinements of such things as communications and state and county coordination, which due to the newness of the plan to all the parties did have some rough spots.

We are working actively with the participants in the counties and in the state to resolve any of those details, which we did note in the formal report.

MR. MOELLER: Thank you, Mr. Jaske.

Mr. Carbon, did you have a question?

MR. CARBON: I think it is pretty much answered, Mr. Chesnut. You are from the NRC staff?

MR. CHESNUT: Yes, sir. I am from the Division of Emergency Preparedness.

MR. MOELLER: Mr. Bender.

MR. BENDER: Mr. Chesnut, in sending up this team to deal with emergency response, did the NRC or FEMA either one attempt to establish some qualifications for the people that were responsible for emergency management?

MR. CHESNUT: Are you referring to in the review of the planning or for the NRC's own response to an emergency?

MR. BENDER: In reviewing the program, I guess I have in mind that the NRC has some responsibilities, the state has some and FEMA has some responsibilities. In order to have an effective team, I would presume there would be some requirements for experience, some training capabilities, as well as some practice exercises.

Does that kind of set of requirements exist?

MR. CHESNUT: Yes, sir. For people who are reviewing plans, we have all undergone in-house training on the standards.

MR. BENDER: I am not talking about the planning reviewers, although it is good to know they have capabilities too.

MR. CHESNUT: In I&E there is a separate branch called Instruments Response Branch, and they have been responsible for improving NRC's own emergency preparedness. We are in the process of hiring full-time duty officers who are trained to respond in initial hours. In addition to that, for many of the exercises that have been conducted of the many licensee plants around the country, the NRC's instant response teams have periodically participated in some of those drills and exercises.

MR. BENDER: Well, that is a partial answer but it's a long way from being a complete answer. I know that some drills have been carried on. I think the question still is what kind of qualifications are required for people that have to manage such emergencies, and do we have such qualifications and are the people in charge now being judged as to whether they meet them?

MR. MOELLER: Mr. Bender, you are referring not only to NRC or FEMA but to the state or local people.

MR. BENDER: I'm talking about the whole emergency response team. I really don't know what it is. I wish I did. But in the absence of knowing, I am hoping that the planning or review organization is doing something to take it into account.

MR. CHESNUT: Well, I think we are. What we have done is we identified the many functional areas which needed to be addressed in an emergency. Within the NRC there is a lot of expertise in each of those areas. So in many cases it is a matter of developing standard procedures and having people with the right expertise to address the problem, and that is what we are doing. We are testing the coordination of all those individuals.

At the state level, for instance, FEMA does perform reviews of the ability of the state to respond. That includes some of the training and qualifications of those individuals. The state of Pennsylvania, for instance, has a technical branch, the Bureau of Radiological Protection, where they have several nuclear engineers and health physicists to digest the technical information from the licensee and make a recommendation for protective action to the state, which would be the Pennsylvania Emergency Management Agency or the governor.

Those agencies respond to all kinds of emergencies, from tank car collisions to radiological disasters, so they are used to applying emergency resources. But they use the Bureau of Radiation Protection, for instance, as a technical response branch in those areas.

Those, of course, are reviews. We have no set standards for each state to have a certain individual with certain qualifications able to respond.

MR. MOELLER: Mr. Shewmon has a question.

MR. SHEWMON: This could come from Mr. Chesnut or the licensee's response, but I am so ignorant of this I don't really know whether we are talking about something like the QA audit or somebody goes in and asks a certain fraction of those involved what they do, or at the other extreme of my experience at a grade school fire drill where everybody gets up and walks out in the yard.

Could you tell me briefly whether the populace is involved in this, whether it is an audit or whether they indeed physically transported people out of areas to see how it worked?

MR. MOELLER: Go ahead, Mr. Chesnut.

MR. CHESNUT: I think you were referring mainly to an evacuation order. There are more facets to emergency planning than conducting an evacuation. We normally don't have evacuations of the local populace. These drills usually go to a point where an evacuation would be a decision and the state actually deploys its resources to effect it, such as the state police or traffic control or something of that nature.

MR. SHEWMON: The exercise was a drill down to the level of the people who would be managing and carrying it out.

MR. CHESNUT: Yes, sir.

MR. SHEWMON: Thank you.

MR. MOELLER: Any other questions for Mr. Chesnut?

Thank you very much.

We will move on, then, to the licensee and request your comments on the drill.

MR. CLARK: Mr. Rogin, the Director of Emergency Preparedness, will make that presentation.

MR. KERR: Mr. Jaske, excuse me. You mentioned, I believe, 72 deficiencies that were noted in your report?

MR. JASKE: Yes, sir.

MR. KERR: Is there any number of deficiencies that would have convinced you the plant was inadequate, or does it have more to do with the seriousness of the deficiencies rather than the number?

MR. JASKE: Well, the basic plan is quite workable, and the deficiencies I mentioned are those we discovered in actually performing the exercise itself. I was talking about deficiencies related to the exercise objectives, which are sometimes refinements of the plan or sometimes do not always include every element in the plan, as we have just discussed, the full evacuation carried out.

I don't know whether you have had an opportunity -- you know, you are free to examine.

MR. KERR: I can answer that. I have not examined the list. But the deficiencies were, then, not deficiencies in the plan but it is in its execution?

MR. JASKE: That's correct.

MR. KERR: So you judged the plan to be acceptable.

MR. JASKE: That's correct.

MR. KERR: Thank you.

MR. MOELLER: Mr. Ray.

MR. RAY: Before Mr. Jaske leaves, could you tell us how that 72 defiencies is split from the viewpoint are assignable to the licensee?

MR. JASKE: Well, FEMA's judgment of the state and local effectiveness concentrated almost entirely on the state and local government. The only place where we noted any difficulties was in some of the discussions between the utility and the state about an accident classification, which we have taken care of by being sure that the accident category descriptions are more clearly identified.

There was some confusion over the nomenclature. We made quite a point of that with the state. The Governor's Office was involved in a misapplication of nomenclature at one point in the exercise, and we highlighted that as a problem that needs solution, and I assure you that that will be solved.

But essentially, 95 percent or more of the deficiencies relate to the state and local government. We did not make an evaluation of the licensee except in that one case where there was communication.

MR. ROGIN: I am the Manager for Emergency Preparedness for GPU Nuclear Group. I would like to say a few words perhaps to answer some of the questions with regard to the extent of the exercise and what our observations were with regard to its utility and its ultimate result.

We were evaluated by the staff in three different areas, one of which certainly was our ability to implement the plan effectively should an emergency occur, and the other two were related to our ability to prepare for such an implementation.

One was with regard to our own ability in house to plan for and develop a reasonable scenario and conduct exercises which would provide some training opportunities for people, and secondly with our own ability to self-evaluate our performance and to identify those things which we felt needed additional attention and effort on our part in order to assure that we were maintaining an appropriate level of emergency preparedness.

The staff has indicated to us in their results that we had done those jobs satisfactorily. We had identified the major -- there were no major shortcomings or deficiencies indicated -- but there were some areas which needed additional attention, and we had identified those as part of our own self-critique and taken or initiated the appropriate actions to follow up on those, and those areas will be a subject of additional attention by the staff.

So overall we felt that the exercise was one which gave us a very good opportunity to implement the full scale of our plan and to coordinate with the state and the local governments to assure that we could in fact execute and implement a coordinated response to an emergency.

Just to give you some feel for an answer to the one question with regard to the magnitude of the exercise.

(Slide)

It was indicated in general that it went to the point where it tested right down to the local population, but we didn't actually evacuate anyone. But these are major events which did occur as part of the exercise. We had full accountability of personnel on the site for both units.

We actually had a search and rescue scenario. We called in an off-site fire department and ambulance services to respond, and in fact during the course of the event on June 2nd we also conducted our Radiation Management Corporation supervised annual radiation health physics drill, which called for us to take a simulated contaminated person and transport them to Hershey Medical Center, and then we will get a full, formal critique on that, including the Hershey Medical Center part of it.

We also went through contamination drills. As I indicated, we had a fire drill with several companies off site responding, and these are all people, by the way, who would normally be our first line response if we had an actual event, and all these people had been participants in previous drills on site, including actually taking some of them inside the protected area on a couple of occasions to practice our security processes.

We had the state police not only in terms of assistance of traffic control, but they also provided helicopter assistance for population notification in the surrounding areas. So in essence we activated all of our response organizations. We went all the way through a scenario to general emergency declaration and recommendation by the GPU Nuclear management that the state should consider an evacuation within the plume exposure pathway, and the state then followed up and began the process of evacuating but did not in fact evacuate people.

We had five counties in the risk area which participated. The fifth one, York County, was mentioned, and we expect to have an exercise with them in the not too distant future. Some specific dates are being discussed right now. We also had several municipalities in each of the counties, so we feel right from the control room all the way down to the local township or municipality level, the whole system of communications and implementation for emergency response was exercised. There was very little simulation that occurred.

We did not evacuate the entire site. We took a selected number of people once we had the accountability drills completed and took 15 people out of Unit 1 and actually took them through the evacuation process to get a feel for that, and of course we didn't have a real contaminated person so we simulated that.

But that gives you a fair flavor for the kinds of things that occurred. We have already begun discussions as a result of the FEMA and NRC feedback to us on the drill. We are working already with the state in several areas in discussing how we can make arrangements to improve upon some of those areas where we identified shortcomings.

And finally, I would only mention that the company has entered into a maintenance agreement with a consulting firm that we have used in the recent past for the next 18 months to two years, and they will be assisting the counties and local municipalities through the good offices of Pennsylvania Emergency Management Agency, working with them as an extension of their office to assure the upgrading of the emergency plans of all the counties and local municipalities within the risk area.

So we feel we have had an exercise which clearly exercised all of the various aspects of our plan and gave us some areas which we wanted to direct some attention to but did not identify any area which was significantly deficient or would impact upon our ability to manage a real emergency.

MR. BENDER: Could you tell us a little bit about what kind of advice and instructions you received from NRC and from the state during this exercise?

MR. ROGIN: In terms of the preparation for or the actual exercise?

MR. BENDER: The actual exercise. How were they involved?

MR. ROGIN: It might be better if I asked Mr. Arnold, who served as the actual emergency support director and communicated directly with the site director of operations of NRC to answer that question because the communications were personally between the two people.

MR. ARNOLD: The way in which the scenario developed, I think, was such that the company was evaluating the conditions of the plant in providing information to the state and the NRC people who were participating in the drill as to plant conditions. There were, as the situation was simulated to become quite serious, several conversations between myself, the NRC senior person participating in the exercise who was at the emergency off-site facility with us, the nuclear engineer from the staff, the Department of

Environmental Resources, who was also there, and Mr. Tom Jeruski, who was head of the Radiation Protection Department or Bureau of Radiation Protection, as to what the plant situation was and whether protective action would be required.

So there was dialogue on that, and I think keeping abreast of all parties as to what the company's understanding of the situation was. I can't recall any specific kind of inquiry or advice as to how we may need to proceed differently than we were that came out of those discussions.

MR. BENDER: Let me pick a few items out of the scenario. One of the things that is shown here is a post-accident sample of the reactor cooling system. That presumably came from some decision that there might be a fuel failure of some sort. Is that an automatic thing or --

MR. ARNOLD: Yes, it is. We have to demonstrate that we can provide within a certain time -- I think three hours is what we are committed to -- that we can obtain a reactor coolant sample in the situation where there had been fuel damage. So that is part of the institutionalized response to a reactor tri

MR. BENDER: Did the state or the NRC ask when are you going to do that?

MR. ARNOLD: I did not receive that inquiry, to my knowledge, but they would have been aware from their information, their participants in the control room as to whether or not that action was under way, and I am quite sure that the plant people took the initiative in following through on that procedure, to tell you the truth.

MR. BENDER: Let me just ask about one other item. The last item on the scenario says the reactor cooling system is on natural circulation moving to decay heat, and I don't know what the postulate was at that time as to whether nuclides were in the systm or not. What kind of communication is involved in telling people about the status of the system at that time? What did you have to tell them? What did they ask?

MR. ARNOLD: In the emergency off-site facility we have status boards, and although we can't dummy in the signals for an actual situation, we can call on the CRT for various parameters. So those that were germane to the establishment of natural circulation or demonstration that natural circulation was occurring were available to the NRC participants and state as well as myself.

In the direction that we were going for removal of decay heat, that of going to natural circulation having steam to the atmosphere, was identified to the state as the plant situation degenerated, so that I could ask Mr. Hank Hukill, who was the emergency director in the control room, if there was any initiation from the state or NRC people as to what we should be doing differently, but there certainly was not from the off-site facility.

Mr. Hukill.

MR. HUKILL: I am Director of Unit 1. I was the emergency director during the exercise. We had the NRC, our senior representative on the site, right in the control room with me. He had a man on the emergency phone with Bethesda. He asked me several questions which I don't remember what they were now, but I reported to him what we were doing and he asked on several occasions what we were doing.

We were also in the control room in continuous communication with the state Bureau of Radiation Protection. They asked several questions concerning what our readings were and what our off-site readings were and what our recommendations were until that really passed to Bob Arnold at the off-site control center.

MR. BENDER: I was just trying to think about trying to reconstruct the TMI-2 incident in asking people, well, how did it go and how do you report how the event transpired and what people remembered, and I think it is kind of interesting to recognize that here in just an exercise it is hard to remember all the things that were going on. You might think about what kind of recordkeeping you are supposed to have during one of these things.

MR. HUKILL: We do have a log that was kept. I have a separate log keeper that just walked behind me during the whole drill and writes down every single question asked me and every decision made. So I could go back to that and find out what I had been asked, but I can't remember it all.

MR. ARNOLD: I think your point is well made, and I would like to identify that we have for the emergency support director a tape recording. We tape all of his communications.

MR. MOELLER: Mr. Rogin, I think because of time we are going to want to move along on this. Mr. Carbon has a question.

MR. CARBON: Yes. How would you notify the general public in a real emergency, and what kind of timing do you anticipate, what amount of time to evacuate a certain group of people? How fast does this take place and how is the evacuation carried out, just briefly?

MR. ROGIN: With regard to notification, we have to notify principally two people to start the scheme, that is, with regard to the public. The first is Dauphin County and the second is Pennsylvania Emergency Management Agency. PEMA then picks up notification of the other four counties , and then each of the five counties in turn has its own notification process.

PEMA also notifies the Bureau of Radiation Protection, who then closes the loop back to the licensee, and then from that point on, our communication is between the Bureau of Radiation Protection and the licensee, and then they in turn take care of any future updates, notifications and reclassifications of accidents, except in the case of a general emergency. And when we go to a general emergency, we have an exception to the case and both the licensee and the state go directly to all five counties.

The timing on that -- the requirement is 15 minutes from declaration of the first classification of the event. My feeling is we would probably take care of the first part of that, which is the state and the county, within several minutes, and I think the state has certainly come in well below the 15 minutes, in any case. So within 15 minutes the public down to at least the county level has been notified and determinations were being made.

With regard to how much beyond that the notification ought to go, with regard to notification of the general public we are installing a system now which is the siren system backed up by some other means, and the decision on that really resides with the state. When they determine that the general public needs to be alerted to an event, then they sound the alert.

With regard to the evacuation, if course it is a variable. Our studies conducted some time ago indicate that on the best case we could evacuate the entire ten-mile EPZ in five to ten hours, and the latitude demonstrates the difference in degree of mobilization of the state police and others at the time evacuation is ordered. It could go up to as much 13 hours, depending on poor weather, improper mobilization or poor mobilization, lack of timing and so forth.

MR. CARBON: Thank you.

MR. WARD: The exercise apparently lasted seven or eight hours, is that correct?

MR. ROGIN: Just a little better than ten, sir.

MR. WARD: Could you tell me the day of the week and the time of day?

MR. ROGIN: I think it was Tuesday. I'm sure it was Tuesday, and we started about 5:30 in the morning because we wanted to go through a shift change, so we had a change of shifts before we were actually at the highest classification.

MR. WARD: So the final parts of it were during daylight hours.

MR. ROGIN: Yes, sir. The evacuation orders came about something like 12:30 or 1:00. The general emergency declaration was like 12:30, and by the time we gave everybody a chance to exercise the state requirements and so forth, we terminated right around 3:00.

MR. WARD: Do you believe that is a good test of a situation you might have in weekends or night-time hours, for example?

MR. BOGIN: I think it does because we started out -- of course, there are always certain artificialities in an exercise. People did know the date of the exercise although they had no idea of the time. That was required for a variety of reasons. But we started out on the back shift. We had to go through a shift change. We had to bring in people early for the various governmental agencies within the state and counties.

Clearly there might be a little variation on the theme if it was totally unannounced and so forth, but I think the system is such that all our previous tests to that system, including call-outs in the evenings and so forth, we have proven that our system is responsive. We really think we could do it just as well on Saturday now as we could on Tuesday morning.

MR. WARD: Thank you.

MR. MARK: It was mentioned, I have forgotten exactly by whom, of Mr. Gerusky, the state radiation protection officer, being involved in some aspects of the exercise. Now, he expressed himself, I think, as less than fully satisfied with the communications, depth of and so forth, that he was involved in in April '79. Has he expressed himself recently on his feelings, let's say, about the company's present relationship or status with respect to the office he is responsible for?

MR. ARNOLD: Perhaps it might be more appropriate to address that question to Mr. William Dornsife, who is a nuclear engineer on Mr. Gerusky's staff and who is here today, if he wouldn't mind responding.

MR. MARK: Well, I remember one time he was rather disappointed or critical or something. Has he changed his opinion is what I am really trying to find out.

MR. ARNOLD: I think Mr. Gerusky may be in more of a position to respond to that than the company, although I have no reason to think that Mr. Gerusky would not tell this committee that he was not fully satisfied with the interaction of the company at this time.

MR. DORNSIFE: I am the state's nuclear engineer and the one that was at the EOF during the June 2nd exercise, and also was the nuclear engineer during the TMI accident, so I can see a different perspective from the two things, so I could comment either on that improvement in the situation or also in the communications.

First of all communications-wise. We did have some procedural deficiencies in communications particularly with the operational data to the BRP, and we found from previous drills with this licensee and other licensees in the Commonwealth, and also, of course, with the real accident, that probability the majc-'ty of times our protective recommendations will be either partly if not highly based on operational status of the plant. So we feel that knowing the operational status of the plant is very important.

There was a problem with the direct line that was set up that was used for radiological information because of some physical limitations with getting information over that line. There was a deficiency in the early exercise. However, by going to the EOF and interfacing directly with the licensee and people at the EOF has certainly, in my opinion and I'm sure Mr. Jeruski's also, has cured that deficiency and has gone a tremendously long way toward having the state understand the things we didn't fully understand durig the TMI-2 accident.

Having the state and the NRC at the EOF and having information available is a large step forward, in our opinion.

MR. MARK: Thank you.

MR. MOELLER: Thank you.

Thank you, Mr. Rogin.

The next item on the ag- da is the discussion of the proposed reliability study and its objectives, and the licensee will respond on that.

MR. CLARK: Mr. Keaton, the Director of Systems Engineering in our Technical Functions Division, will make that presentation.

MR. MOELLER: Let me remind Mr. Keaton that we do want to cover it thoroughly, and yet please note that we have to keep moving along.

MR. KEATON: I will certainly try to keep this very short, and in fact I have only one Vu-graph, which may help to hold the time down.

I would like to remind some of the members of this committee and perhaps inform some other members that previously we have done a probabilistic risk assessment study on one of our plants, but that plant was Oyster Creek rather than TMI. That study was started early in '78, and the first cut at it was finished early in 1979.

The occurrence of the accident directed the attention of people who had been involved in that to other things for quite a while. We have now resumed the work required to complete that study and we expect to have the final results later this year.

(Slide)

Basically what we are intending to do for Three Mile Island unit 1, as shown on the slide, is an analogous study to that which was and is being done for Oyster Creek.

Our incentives for doing it are, one, to understand the factors that lead to the risk to the public, and also the factors which lead to availability or unavailability of the plant.

The process that we intend to go through will be similar to that we used for Oyster Creek. It is a thorough and comprehensive risk assessment based on getting a detailed understanding of the systems, conducting event trees and fault trees, and using both industry data for failure rates and then augmenting those by actual experience at TMI where that experience is available.

We intend to try to cover not only random failures but common mode, common cause type of events, and also external hazards. The reasons why we are interested in doing this are that we find that the results from such a study can be used in a very meaningful way in our decision-making process, particularly with respect to allocation of capital resources, because we frequently face a situation of should we put money to improve safety on this area or on this area.

As we stand today, we have to make those decisions based on engineering judgment. We think that with the results of the study that we can in some cases quantify the answers to those questions and augment the engineer judgment to where those are hopefully better decisions.

We intend to have the specification prepared for this study in the third quarter of 1981. We don't believe that we have the resources in house to do the study ourselves, so we do intend to go out for bid. We intend to commence the study toward the end of the year. We visualize that at least the initial go-through is probably on the order of a 12-months study, and our experience with Oyster Creek indicates there will be some follow-up studies required.

MR. MOELLER: Are there any questions for Mr. Keaton on this?

Roughly what amount of budget are you talking about? What does such a study cost, a ballpark number? Does anyone know?

MR. KEATON: It is on the order of several hundred thousand dollars.

MR. MOELLER: That's fine. I wanted to have an idea. Okay, thank you very much, then.

Mr. Silver, do you have any comments on this?

MR. ARNOLD: Dr. Moeller, the Oyster Creek study has cost several million dollars at this point.

MR. MOELLER: Thank you.

MR. NOVAK: The staff has no comment.

MR. MOELLER: We will move on to the next item, which is the new instrumentation for detection of inadequate core cooling. We will have the licensee give the report on that.

MR. CLARK: Mr. Keaton will make that presentation.

MR. MOELLER: All right. He will do that one, too. Thank you.

MR. KEATON: Excuse me. I had misunderstood the order and I thought the staff was going first or I would have been better prepared.

As I indicated to the subcommittee and would like to reiterate to this group, in previous discussions we have tended to emphasize to you some of the problems that we have found in trying to understand some of the requirements which have been imposed primarily through NUREG-0737. Today I would like to shift that emphasis and concentrate more on what we have been doing and what we have found out and where we think we are going.

We recognize that some of the earlier presentations have left a rather negative impression of our approach to this, and we do continue to have some questions, but I want to indicate to the committee that we are not sitting and doing nothing but we are really trying to make process in this area.

(Slide)

As you know, the requirement that we are addressing is given in NUREG-0737, Section II.F.2 which requires the evaluation of additional instrumentation for detection of inadequate core cooling. As we have pointed out in some discussions with the subcommittee, it is not clear that in fact water level is the most desirable signal, and there may be real reasons to consider alternative signals, which might, I think, provide some different types of information.

(Slide)

The approach that we have been and are continuing to take is basically a two-fold one. One is to try to understand the criteria that might be used for these types of signals, and to that end we have worked with the B&W owner's group in the work that was done primarily by B&W. We have done some in-house consideration of how we might look at operator guidelines with respect to a LOCA or overcooling accident, and what we would do with level instrumentation and level information if we had such information.

And also we early on became aware of the fact that this kind of information might be used for other things than just inadequate core cooling, so we have been trying to consider what the criteria for that would be as well.

With respect to potential detectors, our initial efforts were through the B&W owners groups to evaluate the different systems that have been considered. We followed this with some in-house evaluations of detectors and came to the conclusion that we weren't very satisfied with either of those two, so we have initiated a study by an external consultant to carry out an independent evaluation of the systems that are available and to consider whether there are some other systems that aren't currently being developed that might be useful.

We have agreed to work with Penn State University in a proposal they have developed for the use of neutron detectors for water level inside a vessel, and we are also aware of and expect to evaluate the EPRI study which is currently in progress and is due to be available in about October.

Our approach, then, has been based upon the results of these two paths to decide that one of the systems is useful and should be installed, in which case we would propose to install it, or if that is not the case, to support the development of other systems and alternates, such as, for example, an inventory system.

(Slide)

In understanding the uses that might be made of water level or related instruments, we have looked at the various types of events that we think this information might be considered for. With respect to a LOCA, our first conclusion -- and I believe this is basically agreed to by the NRC staff -- is that the required operator actions can be taken based upon the existing instruments, and we think this is very important because if that were not the case, we would be in a situation of needing to install something before we could restart the reactor.

But we found and the NRC staff testified that the required operator actions can in fact be based on the instruments we now have. In the case of a LOCA, about the only real use we have identified for water level is it can be used as a confirmatory signal, that the operator could look at it to back up the signals that already exist in assuring that he was taking the correct actions.

The suggestion was made at the subcommittee meeting by one of the members or consultants that another possible use of void fraction would be to use it as the basis for the pump trip criteria. That, of course, is not the case right now. It is based upon the occurrence of 1600 pounds of pressure in the reactor coolant system.

But since really the reason to trip pumps is to avoid getting too large of a homogeneous void fraction in the system, it is in fact conceivable that that might be an alternative and perhaps even preferred criterion for tripping the pumps. Frankly, we haven't done too much work on it in the past two weeks, but we do want to consider it.

With respect to overcooling events we believe that the required operator actions can be taken adequately based upon the instruments that exist. The suggestion was again made at the subcommittee meeting that if we had not a level instrument but a true measure of the mass inventory of water in the system, that in fact then that instrument could be used to distinguish between an overcooling event and a LOCA, because although both of those events lead to a change in level, they do not lead to a change in mass inventory.

But that would be a system that is somewhat different from one which displays level, although obviously they are closely related. It is a matter of how information would be processed and how it would be displayed and whether it would be displayed as a water level or as a mass inventory of water.

With respect to venting the RCS system, as you know, we are committed to install vents at the high points of the loops and at the pressurizer, although not necessarily prior to restart, and the suggestion has been made that use of a water level instrument would be a good method of determining when and whether to vent and also when to terminate the vent.

In this case I think the instrument would be required to be, in fact, a level indicator because there you care not about the inventory of the system as a whole but the extent and nature of voids that would be effected by the venting operation.

We have not yet taken a position on whether that is required for the venting. Frankly, the guidelines for that are under development and we think it is an open issue whether it is really needed.

A similar situation with respect to the bubble in the head, such as the St. Lucie incident. We have new operator guidelines that are currently being developed for how an operator should respond to that event. We aren't sure whether level in the vessel or inventory in the vessel is a required or desirable signal, and we won't know that until we have gotten the guidelines better understood.

I do want to tell the committee that on our own initiative we decided several months ago that there would be a significant advantage to us, we felt, to have an indication of whether saturation conditions exist at the reactor head, and we do presently have a project under way that will lead to installing thermocouples, monitoring the water at the top of the vessel so that we can look to see whether it is saturated or not. That will give us an indication of whether a bubble could be drawn or not.

Finally, it has been suggested by various people, including myself, that level or inventory or void fraction information might be useful in analyzing after the fact, going back and looking at what really happened during some of the severe transients that might occur.

We had experience with one of those transients, you might remember, in March or April of 1978 at TMI-2 where we had a severe overcooling transient caused by the steam safety valve sticking open, and we were forced to do a considerable amount of evaluation of that event after the fact to address the question of whether the core had been uncovered or not.

We were able to conclude that the core had not been uncovered without having level instrumentation. On the other hand, if we had had something like water level or inventory, we most certainly would have used it.

Turning fairly quickly now to looking at the systems that we believe are probably the frontrunner for level measurement, this slide basically summarizes, .hen, the vessel delta P system Westinghouse is associated with and work has also done been done by EG&G, and of course, the same kind of system is used in water reactors and in our opinion. will give a good measure of water level under reasonably quiescent conditions.

The vendor claims that this system can be used to measure void fraction when the reactor coolant pumps are running. I'm not prepared to refute that, but I can tell you that there's a great deal of skepticism as to how accurate or reliable that type of information is likely to be.

B&W has a hot leg delta P system which has been looked at which works on this same principle. The hot leg level in a B&W system is a good signal in terms of anticipation in that you will see changes in void fraction or changes in level in the hot leg long before it gets low enough to affect the core. So from an anticipatory standpoint, that's good. From the standpoint of meeting the total range coverage, and of course, the hot leg by itself doesn't do that. It takes it down to only about eight feet above the core.

The Combustion Engineering system of heated thermocouples indicates the level at discrete intervals, depending upon the number of thermocouples. We still have some question in our mind as to how well the system really will work in the presence of two-phase mixture. And given a single phase and well-defined boundary between them, I think the system unquestionably will work very well.

But in the case of the two-phase conditions, the design of the detector is supposed to get an equivalent collapsed level, and we have not completed our evaluation of whether we think it will do that or not.

The neutron detector signal, of course, is potentially a very attractive one because it is a non-intrusive detector, and this is a system that we have agreed to work with. The tests that have been done to date, not by Penn State but sponsored by EPRI, indicate that there is good sensitivity to water level as long as it's within about eight feet of the top of the core. If it's more than that, the response to the neutron detectors is difficult to interpret.

Finally, the core exit thermocouples, of course, don't indicate water level as such to keep the core covered. They might be used as some sort of indication of what was going on, once the level has dropped below the top of the core. That would mean trying to take the temperatures and back-calculate to an equivalent level.

So, our conclusion on those is that really, we don't see anything that strikes us as an ideal detector at the moment, although some of them have varying degrees of promise.

(Slide.)

Where we believe we stand at the moment -- and this certainly is not intended to think that we have finished -- is that we have not seen anything that makes us believe that such a system is required prior to restart of Unit 1. And there I think we're in agreement with the NRC staff. We don't need it as an input to safety systems, so far as we know. And as I said earlier, the operator can respond today based upon information that is available today.

We further feel there is a considerable question as to what the criteria for these detectors are. The major issue to us is, is the real detector that we want a water level indicator, a mass indicator or even a void fraction meter? Those are not the same thing, although water level and mass inventory are relatively closely related.

We believe that the information could be used for confirmation. In some cases it would be helpful for later evaluation in some of the longer-term actions such as venting, where we are still developing the guidelines and we might find such information would be very helpful.

As I indicated earlier, the idea of using void fraction as a basis for reactor trip we think merits some further evaluation and we haven't done that yet.

With respect to the detectors, our principal concern is whether a detector that can be used under quiescent conditions can also be used under pump flow conditions where you are transitioning from a level to a void fraction. To us, it is not clear that the existing systems can do that, although there is disagreement on that point, and some other people believe that they can.

We tried to do a rack-up of the existing systems against the existing criteria in NUREG-0737. It was our conclusion that none of those systems meet all criteria, although the staff has indicated in some cases it would be acceptable to them. So we do not feel that we are in the position yet where we are prepared to adopt one of the existing systems without at least doing some further looking at other approaches that might have advantages over the existing systems.

We furthermore are very much concerned with prematurely installing this before we know what system we want, and before we really know how we would use the system if we had it in there. Adding equipment per se to a nuclear plant always has the disadvantage of the complexity of the system in making the operating training and procedures more difficult.

So we want to make sure where we do add equipment that we do it in a fashion where the advantages outweigh the disadvantages, and we need to understand both the hardware and the software that would go along with it better than we do today, before we would be able to make the statement that we thought it was better.

So we plan to continue very actively the evaluation that we have had underway, and plant to continue our interaction with the staff on these questions, and hopefully, will be able to arrive at a conclusion that is acceptable to both of us as soon as possible.

MR. MOELLER: Questions for Mr. Keaton? Mr. Ward?

MR. WARD: Mr. Keaton, you mentioned something to measure the reactor inventory several times. What do you have in mind there?

MR. KEATON: Well, I don't have a system sitting in my back pocket right now. The right kind of a delta P system under quiescent conditions, I think. My concern is with the force flow condition. In principle, it could probably be done with a heated thermocouple system, provided that the data from the reactor vessel were appropriately processed. It would be more than just looking at which thermocouple was covered and providing that some additional data processing was done with respect to what's going on in the loads. And I don't know how that's done. So getting a good level of mass inventory strikes me as being a more complicated problem than measuring the level, and we're still trying to consider how we would do it.

MR. WARD: Thank you.

MR. MOELLER: Any other questions? Thank you, Mr. Keaton. Let's turn to the staff and simply ask for a response or comments that you may have at this time.

MR. NOVAK: Larry Phillips of the staff will make some comments.

MR. MOELLER: Thank you.

MR. PHILLIPS: Good morning, gentlemen, I'm Larry Phillips.

(Slide.)

There seems to be several questions concerning the staff requirements and criteria. As indicated on the next page of your handout, the criteria, we feel, are well spelled-out in the referenced documents; 0660, 0737 and 1.97. We are prepared to make it -- some more definitive statements concerning the conceptual designs which we had been involved rather deeply in the review of. And this staff has taken some positions concerning what is acceptable conceptually.

Minimum instrumentation system. We would require saturation meter for PWR's. It's not required for BWR's. We would require coolant inventory or level above the core, and we are using those rather synonomously. We don't think it's a great problem to get from one to the other, since we're looking for trending. We are not looking for very precise inventory information.

The illustration of Westinghouse, GE and B&W and CE plants are generally put in the perspective of what had been proposed by the vendors for those particular reactors. That is, Westinghouse has proposed a delta P system; CE has proposed a heated junction thermocouple system; B&W has no firm proposal as of yet. However, this does not preclude interchange systems. For instance, a CE system could be put on a Westinghouse reactor or vice versa.

We required -- the next point I think is a rather important position we have taken actually within the last week. In NUREG-0737 we indicated that we might accept core-exit thermocouples as an indication below the core, provided that it was processed in terms of level. CE, in their heated junction thermocouple system, had proposed tying it in with above core level processing, but actually using it as an inversed temperature relationship rather than attempting a direct level display.

They have indicated that level can be backed out of it with more time, but they don't want to do that online. The staff does agree with that approach, and our position at this time is that conceptually, we do accept that with a good processing system which will accomplish the objectives that they have indicated. So we would not require a direct level readout below the core.

Now, carrying that a step further, the same situation would be true of a B&W reactor where, if they had a level monitoring system or inventory monitoring above the core and up into the candy cane, and with the existence of their core exit thermocouples, they, too, could get by without actually going into level below the core if they properly interpreted their core exit thermocouple temperature and tied that in with their above core level display.

And, of course, core exit thermocouples are required for all PWR's, and the staff position at present is that thermocouples are also required for BWR's. And that position is being appealed, but that is the current staff position.

(Slide.)

The next slide that I mentioned just gives the basis of the NUREG documents. We think there is nothing greatly new in this. This was presented at the clarification meetings over a year ago, and we indicated at least at that time that we felt that the Westinghouse delta P and the heated junction thermocouple concepts could be made acceptable, and the staff position has not changed. We are more confident today than we were then. We believe that those systems can be made acceptable, and we expect to take a rather definite -- to complete our review of these systems, the generic systems, by the end of the year.

(Slide.)

Our position on TMI-1 restart is that existing instrumentation with commitment to upgrade for NUREG-0737 is acceptable for restart. We feel that we need evidence of reasonable progress on additional instrumentation before we would agree to restart.

Now, why do we feel there has not been reasonable progress? Well, as I say, over a year ago the staff indicated that here are two systems that have been proposed in the industry that we put a number like over 95% or 99% confidence in that they could be made into acce table systems. We ee no evidence that the licensee in this case has really made any indepth study of those systems. They have given us no submittal with any details as to why either of those systems could not be incorporated onto their particular reactor.

In fact, some of the cases we heard indicated a great deal of misinformation about the systems, and it's our understanding that they have had no real contact or indepth discussions with the vendors of the systems.

If we saw evidence, a good strong argument, that hey, there is something unique about this reactor that makes either of these systems no good per the application, then at that point perhaps we would consider an alternate approach, which is not far enough down the line in development with a less-definitive design while they carry out this development.

At the present time, we feel that we would require them to pick a system and just set out to install that system, unless they can show us reasons why, as I say, those particular systems cannot be used.

(Slide.)

MR. MOELLER: We have a question.

MR. KERR: Do I understand that there now exist two systems which have been tested and demonstrated workable and which the staff would approve today?

MR. PHILLIPS: There are two systems which have been tested to a great extent and which the staff has under review at the present time, and to which that review has progressed to the point that the staff feels that something like -- I guess we would put something like 99% confidence on it; that when our review is complete and with any minor modifications that we might require, that those systems will be acceptable.

MR. KERR: So your answer to my question is no, that there are not existing systems today which are approved by the staff?

MR. PHILLIPS: We have not approved any systems.

MR. KERR: Now, what is it you want this licensee to do?

MR. PHILLIPS: We want this licensee to evaluate those systems in detail and to provide us with -- either choose one of the systems or tell us why they cannot be chosen, or to choose an alternate system which is in a progressive stage of design.

MR. KERR: What do you mean by choose?

MR. PHILLIPS: By choose?

MR. KERR: Yes. Do you mean install?

MR. PHILLIPS: We want them to select a system, proceed to procure it and lay out a schedule for installation. Basically, everything that is on this particular slide.

MR. MARK: Is anyone quite clear what, let's say, the delta P system, in fact, will measure?

MR. PLESSET: I think that the delta P system is a little more straightforward.

MR. MARK: I think it's the heated thermocouple that a lot of people have some reservations about.

MR. KERR: It's straightforward if the system is static.

MR. PLESSET: But I think they have been looking at the dynamic effects. But I thought that there was some question about the heated thermocouple system still that might be serious.

MR. PHILLIPS: I think most of the serious questions have pretty well been resolved. The design currently essentially has the heated thermocouple enclosed in a manner that what they're monitoring at the last level is actually what they are measuring. It's on/off, it has been well tested.

MR. KERR: But these tests have not been in reactors, have they?

MR. PHILLIPS: They have been under reactor conditions.

MR. PLESSET: I was aware of the enclosed feature that you mentioned. I thought there were still questions about them; what they would really measure. But there's no point in pursuing it here.

MR. KERR: Certainly, in the minds of our consultants there are questions about what they will measure.

MR. MOELLER: Mr. Bender?

MR. BENDER: Larry, how many operating plants have been more responsive than this applicant or this licensee?

MR. PHILLIPS: In my opinion, all but one.

MR. BENDER: What have they done differently than this applicant?

MR. PHILLIPS: Well, up to this point at least, this applicant has taken the position that they don't -- essentially, to be blunt, that they don't need a system and they're not going to install one. And this is inspite of staff indication, oh, at least a year and a half ago that that position was totally unacceptable.

I think we've seen more recently a little more movement towards at least indication that well, we are looking at systems harder now; we really are studying the systems. I'm surprised that the emphasis is on the procedures before they get the system or how it's going to be used and the actions and so forth.

MR. CLARK: At some point I would like to comment.

MR. BENDER: I think we would like to know just what GPU's position is on this.

MR. CLARK: GPU's position is that we have been and are seriously evaluating the NUREG requirement to provide improved instrumentation for detection of inadequate core cooling. We feel in a very difficult position on this item because one of the Lessons Learned from the TMI-2 accident, which we fully endorse, is that the utility uniquely has the role of having themselves the competence to balance and design operational characteristics, that we cannot rely on the NSS supplier. And, in effect, say, if he tells us it's okay, that we'll do it, that we have the responsibility and we felt compelled to carry it out, and that in this particular case, the question of how the operator will use whatever instruments and what instruments will, in fact, give him or others the needed information. We feel that that question has to be decided before we add complexity and possible confusion to the plant.

In particular, we feel that the reactor vessel delta P for B&W plants is not shown to be adequate and proper instrumentation. I think we heard today, I for the first time, that apparently, the staff has agreed that reactor vessel delta P is not required in the case of CE if they make a combination of heated core thermocouples and what I will call looped delta P. That's one of the things we have been looking at.

It would be very easy for us to say we commit to put in the system because somebody else says it works. But we don't feel we should do that until we are satisfied that it will work and that we know how to use it. So it is not our position that we will not install instrumentation. I don't believe we have ever said that. I think we have said we are not prepared to commit to reactor vessel delta P because we are not satisfied that that will do the job.

MR. BENDER: Mr. Clark, do you believe something more is needed to determine whether there is adequate core cooling or not in a plant?

MR. CLARK: From everything we have seen, we believe that you can get -- and we will get -- the operator adequate information to deal with LOCA's or other events, with the existing and planned events including saturation meter in the core thermocouples.

I think, however, we agree that additional information would be desirable if there is a way to obtain it and display it to the operator so that, in fact, assists him and doesn't confuse him.

MR. BENDER: Do you believe the regulations require something more than is there now?

MR. CLARK: I believe the regulation requires -- 0737 requires evaluation of additional information, and I believe the staff position is pretty clear up to this point that they require reactor vessel water level. They have defined some criteria for the instrumentation which, for example, says full range.

I think that the continuing dialogue between us and the staff and others is helping improve the understanding of what is needed, and perhaps narrow the difference and lead us to a selection of instrumentation which would, in fact, be more helpful than the initial concepts.

MR. BENDER: Well, this committee is on record I think as saying that there are number of questions that need to be resolved concerning the use of whatever instrumentation is provided. And I'm not aware that the staff has responded to the committee's interest. It offered to meet with the staff and discuss these matters.

Does the staff have a position on the fact that the committee has raised questions?

MR. PHILLIPS: Yes, the staff prepared a response to that very shortly after it came, and I think that -- I don't know why it hasn't gone out yet. I think it got lost somewhere in the general administrative passing through.

MR. BENDER: May I interpret that as meaning that the staff really isn't very much interested in what the committee's opinion is?

MR. PHILLIPS: No, absolutely not. We indicated in the response that we would be happy to meet with the committee and suggested -- we suggested that it be done in the framework of the meeting on operating guidelines and procedures, since most of the questions seemed to be directed in that context.

MR. BENDER: Well, wouldn't it be fair to conclude from the committee's commentary that it doesn't have the same 99% confidence that the staff does that the two systems that have been discussed are well enough understood to make them usable at this time?

MR. PHILLIPS: Yes, I think you probably could conclude that.

MR. BENDER: In view of that doubt, do you believe the committee should sanction the staff's position that one of these systems should be selected by this licensee and plan for installation immediately?

MR. PHILLIPS. I think you should sanction the staff's position that this licensee should get all the available information on the systems and evaluate them for installation in this plant, and at least provide us a complete and accurate story of why he feels that they can or cannot be used in his plant; as evidence that he is making an honest effort and showing real progress towards doing what he can to install an acceptable system.

MR. BENDER: Let's get back to the question I asked sometime ago. What is it that you have gotten from the other operating plants that's more complete than this? What type of information are you getting?

MR. PHILLIPS: Well, there's a large number of plants. I believe 27 or something like that who are installing Westinghouse delta P systems and have committed to have those installed by January. And there are some other plants who have committed to the heated junction thermocouple system.

There are other plants who have indicated that they are studying -- .

MR. KERR: Do you think these will be installed by

January 1?

MR. PHILLIPS: I think they would have been installed by January 1. Right now I would say there's probably enough question concerning schedule that -- .

MR. KERR: I'm not sure what you mean by they would have been. My question is do you think they will be.

MR. PHILLIPS: What I mean by would have been is they could be installed by January 1.

MR. KERR: I guess I don't know how to make my question -- .

MR. PHILLIPS: The systems are available, they will be delivered and are able to be installed by January.

MR. BENDER: Well, the fact that all those licensees have agreed to put something in could be interpreted as blind response to regulatory demands without adequate information. I think the licensee here is raising a question as to whether he doesn't have a responsibility to make sure that the device is a usable one before he commits to it.

MR. PHILLIPS: Why did the licensee here make misrepresentations concerning the staff position, things that were made very clear, I believe some of them, in NUREG-0737 and certainly, in the presentations at the clarification meetings over a year ago.

As I said, what is indicated is just becoming known today; that same thing was said over a year ago at the clarification meetings concerning the use of thermocouples to supplement the above-core monitoring of the heated junction thermocouple -- of core-exit thermocouples, to supplement.

I have heard them say things concerning the heated junction thermocouple system in the Westinghouse delta P systems that make it very clear that they have not studied these systems in any detail or have not conferred with the vendors on this system.

How can you say if the two systems which the staff feels certainly have progressed the furthest and have the best chance, that the licensee has not evaluated these systems in any detail or made any submittals concerning that evaluation; how can we say it's an honest effort?

MR. BENDER: Mr. Clark, could you tell us what you have done?

MR. CLARK: Yes, I think I would like to say first that if we have misrepresented the staff's position, if I have, I apologize. I don't recognize where . h ve done that, but I think part of the problem perhaps is that there has been a fair bit of dialogue without reducing it to writing, and clarifications provided in meetings are subject to misunderstanding.

I think similarly, on our side, while we feel we have been talking to the staff both directly and in these meetings about what we are doing, we concluded at the sub-committee meeting that we ought to reduce that to writing. And in fact, I have here a draft we are unable to complete to submit to the staff outlining more definitively what we have done, are planning to do, the kind of thing we have been presenting orally. And I think we have an obligation and intend to fulfill it to go put that on the record to attempt to clear up some of the misunderstanding or lack of effective communication which appears to have existed.

I think importantly from our standpoint it is an underlying question of whether one should commit to do something until he understands how to use it or not. And it has been our feeling that we should not commit to add some-thing until we have thought out and understood how it would be used. And I gather that we perhaps feel more strongly on that point than the staff does.

MR. BENDER: How much time have you spent with the two potential suppliers of this type of equipment?

MR. CLARK: I'll have to talk to someone to determine that.

MR. MOELLER: While they're getting that information, Mr. Okrent?

MR. OKRENT: No, I don't want to interrupt the flow of discussion.

MR. MOELLER: Okay. Well, let's get this answer and then have Mr. Okrent's question and then take a break.

MR. OKRENT: Again, I think Mr. Bender should pursue his line until he feels he's completed the thought.

MR. MOELLER: Right.

MR. CLARK: We are having some trouble quantifying that. I think maybe -- .

MR. BENDER: I don't need to know it to the nearest hour. Have you been with them for a day?

MR. CLARK: This is Bob Keaton.

MR. KEATON: Maybe I can answer that. We have not spent a lot of time in contact with Westinghouse and Combustion Engineering. I agree with Mr. Phillips on that subject.

We have been concentrating our attention on what to do with information and what information we wanted, rather than concentration on systems that were developed and trying to figure out how to use those systems.

MR. BENDER: Well look, one of the lessons from TMI was that discussions with the nuclear steam system supplier and equipment suppliers is part of the learning process, and there seems to be more introspection in this thing than there is dialogue. And I guess I don't expect this premise that you stated that you're looking at it very carefully, if all you're doing is just cogitating your navel.

MR. KEATON: We have spent a great deal of time with B&W on this question. You had asked me about Westinghouse and Combustion.

MR. BENDER: I'm talking about Westinghouse and Combustion as equipment suppliers. When you go to buy a turbine that matches the B&W equipment, you don't go only to B&W to discuss the equipment; you go to the turbine manufacturer to see what he's going to provide you and you try to match it up with the system. I don't see that kind of discussion going on here.

MR. KEATON: Our initial approach to this, as I think I tried to say, was to approach this through B&W in the same way that the Westinghouse owners worked through Westinghouse and the Combustion Engineering owners worked through CE. And we have had extensive dialogue with B&W and they have not come up with a system that meets the staff requirements and frankly, that meets our requirements. Perhaps we are some derelict in getting together with Westinghouse and Combustion Engineering, and frankly, we do intend to do that, but our initial approach was to work through our own vendor.

MR. CLARK: I would like to add two things to that. First, in terms of equipment capability, we do have the Oyster Creek plant and as a boiling water reactor it does have instrumentation for water level, two-phase flow, et cetera. Mr. Keaton is aware of this instrumentation and its capabilities, and in that sense, we have the CE input.

Second, recognizing that we were not satisfied with our evaluations of the systems, we have already contracted with the man at UCLA to perform for us an evaluation of the equipment systems that are available. So I have no probem with agreeing that we will -- and Mr. Keaton says that he had intended to -- meet with the Westinghouse and CE people. But I think we have been pursuing these other points that I have mentioned and have been addressing the problem, and are not just avoiding it.

MR. BENDER: Well, I know this committee is not totally satisfied that it understands all the capabilities that are needed for such a system, but I have some discomfort with the progress that's been made. I think I have to agree that the staff may have reason for raising questions.

MR. OKRENT: Well, I guess I would like to follow on Mr. Bender's last point. Does the licensee have a proposed schedule for arriving at resolution of the issue? I can be sympathetic with the point of view that they need to know what's in their plant. I can also be sympathetic with the point of view that says at some point, you need to arrive at an answer.

And so, I guess what I would like to know is do they have a time by which they, in fact, expect to come up with some specific recommendation for what they think, (a) is practical; namely, that they can get it; (b) represents the necessary advance beyond the saturation meter; and (c) by when will they do that and why is that a suitable time period.

MR. CLARK: The final step on Mr. Keaton's chart was to perform an evaluation and either select an available concept or decide what further development to pursue. I think that was the choice. The UCLA study is scheduled for completion at the end of this summer. I think we would foresee evaluating that.

The other ongoing work that we have, the information obtained from contacts with the vendors which we intend to make and have been urged here to do and come to a conclusion this fall as to whether there's a system we would pick to install, or what further development we would intend to pursue.

MR. MOELLER: Any other questions on this topic? Well, Mr. Chairman, I would suggest a break until 11:00.

MR. MARK: So be it.

(A short recess was taken.)

MR. MARK: The meeting will resume.

MR. MOELLER: Thank you, Mr. Chairman. The next item on the agenda is a discussion of additional studies to identify possible events which might lead to the loss of both battery trains, and this will be a presentation by the licensee, followed by comment on other studies that the licensee has underway.

MR. CLARK: Mr. Chisholm, the Manager of Electrical Power and Instrumentation, will make that presentation.

(Slide.)

MR. CHISHOLM: We have had an ongoing study at TMI looking at reliability and long-term planning for improvements on the system. For purposes of this report, we focused on responding to NUREG-0666, which was a staff report presented to a subcommittee meeting in January.

The conclusions in that report involve the consequences of common mode failures which could lead to a loss of both batteries. Just to briefly summarize that report, it divided the type of failures into two types. One failure is the kind of failure wherein the batteries become unavailable after you have lost power.

MR. MOELLER: We can't hear you, and go back and repeat, if you will, the last statement.

MR. CHISHOLM: The Type 1 failures are those that become evident after a loss of offsite power, when the power from batteries is demanded and not available. A Type 2 failure is failure of the DC system, which are directly attributable to mistakes in operational maintenance that directly relate to loss of DC power. They are about equal in probability.

The first one depends directly on the particular reliability of the AC offsite power at the plant. The combination of the two has the probability of 6 times 10^{-5} per reactor year which is a very large number, and in fact, represents half of the probability of all accident sequences studies which would lead to core damage.

The report further makes some recommendations, stating that this could be reduced by a factor of 50 if certain things were done such as prohibiting bus ties and improving maintenance test procedures.

(Slide.)

In response to this report, we did make certain improvements, certain changes at Three Mile Island. We have locked open disconnect switches, the main bus tie switches are going to be locked open, the disconnect switches will be locked closed. We have put into place procedures to restrict the use of these switches to cold shutdown.

We found one bus tie in the plant which we studied and decided was not justifiable and that is being removed. We have put in recovery procedures for the electrical system to give the plant people guidance as to when they lose one battery, how they can proceed without jeopardizing the second one.

And we have reviewed the maintenance and test procedures to make sure that there were no precautions in them to preclude these common mode events.

(Slide.)

Going over and using the same kind of approach that was done in NUREG-0666, and using some of their numbers we tried to put a quantitative evaluation of how our improvements would improve the reliability which was included in 0666. And also, we have looked at several things in the TMI-1 plant which were different than the plant studied in NUREG-0666.

And we have concluded that whereas the report had stated that a factor of 50 could be achieved, we think that our plant with its differences and with the changes we are making, we get a reduction of about .003, which is an improvement of about 100. So we feel that we have established the fact that the reliability is much better than that which is concluded by that report and much better than the prediction that could be achieved by the recommendations of the report.

MR. KERR: Before you begin this, did you go through 0666 and see if you agreed with its conclusions? Did you do a detailed review of that analysis?

MR. CHISHOLM: Well, we have not done a quantitative assessment of TMI-1 yet, and as Mr. Keaton -- .

MR. KERR: I'm not making my question clear. Did you go through the report in detail and see if you would draw the same conclusions about reliability of the system analyzed in that report, that the NRC staff drew?

MR. CHISHOLM: Yes, I guess in a general way I would agree with the conclusions of that report.

MR. KERR: How about in a detailed way?

MR. CHISHOLM: The closest I can get to that is we did a quantitative analysis on Oyster Creek on that system, and if we compare the quantitative results of that against not the initial figures of NUREG-0666 of 6 times 10^{-5} events per reactor year where you losed both batteries, but the improvement that we think shows up in the specific differences in the system, I think within an order of magnitude or so we would agree with it.

In other words, the Oyster Creek study -- .

MR. KERR: I'm sorry, I don't understand. I'm either not making my question clear or -- what I'm trying to find out is whether you followed in detail the analysis that they used to get their results and whether you agreed with their results.

MR. CHISHOLM: Yes, we do. If you start out with the system they studied and the assumptions they made, yes, we would agree with it.

MR. KERR: Okay.

MR. MOELLER: Mr. Ray?

MR. RAY: Could you tell me the basis on which you conclude that this plant has more reliable offsite power?

MR. CHISHOLM: Yes. I think there are really two parts to that. First of all, I think that TMI-1's location is close to a major center of the local distribution where the PJM system ties, and right within a mile of the plant is the Middletown Junction Substation, which is the major point in that system. And without going into the details of the number of lines and so forth, PJM at one point did a quantitative study of the reliability of that point and its susceptibility to failure and so forth.

I don't have the number offhand as to what it was, but it was an extremely reliable point.

The other point that should be made is that in NUREG-0666, a large part of the type of failures that they postulate are switching failures after a generator trip. The configuration of the TMI-1 plant is such that the auxiliary transformers are fed directly from the grid, and we do not challenge any switching systems when we trip the generator.

The power is available at all times without switching from auxiliaries to startup transformers. So we feel that the improvement in numbers they used was not quite arbitrary but it was based on some data in the report. They had a five to one uncertainty between the main system which they used and the best system that existed, and I think what we have is close to being about the best system around.

MR. RAY: I'd like to make sure I understand your second point. The fact that the auxiliary power is fed from offsite sources, you mean there's no bank, and therefore, when you lose a unit, you don't have to shut down your auxiliaries in sequence and then come back in to service them? They just continue to run, supplied by offsite power?

MR. CHISHOLM: Yes.

(Slide.)

The aux transformers come directly from the 230 kV bus; whereas, the generator feeds the bus separately, so on a generator trip, it leaves the aux transformers running continuously.

MR. OKRENT: If my arithmetic is right, if I take, say, 6 times 10^{-5} and divide it by 300 I get 2 times 10^{-7}. And the other way, I would get 3 times 10^{-7}. Anyway, I would get a small number.

MR. CHISHOLM: Yes.

MR. OKRENT: When numbers get that small, you have to ask yourself are there scenarios that I haven't included in my earlier thinking because I already had so big a number I didn't need to think about them. Have you gone through that kind of a thought process to see whether you really believe what you have put on these slides?

MR. CHISHOLM: Well, I think we have to the extent that -- the approach we took was to try to just respond to the factors that were pointed out in NUREG-0666. I guess in the future we intend to do a specific analysis of TMI-1 and a quantitative analysis, and we have not done that.

What I did here was to try to show specific improvements in the areas that were pointed out in 666.

MR. KERR: I would judge that he didn't understand your question. Why don't you try again?

MR. OKRENT: I don't know that he didn't understand the question. I'll let you try to explore it if you want.

MR. KERR: My impression was that you were being asked whether there were low probability contributors which would contribute on the level of 10^{-7} or so but wouldn't contribute it at a level of 10^{-4}, which you maybe didn't consider after having bridged this very low number. I didn't hear the response to that.

MR. CHISHOLM: I guess that's true. We did not try to reconsider those things. For example, this study only looked at common mode failures which were the dominant contributors when you had a large number. There may be non-common mode effects that now become significant in addressing that and that was not looked at.

MR. KERR: There might even be low probability common mode effects which haven't been considered.

MR. CHISHOLM: That's possible, yes.

MR. MOELLER: Other questions on this topic? Mr. Ray.

MR. RAY: I don't see in the handout the slide you had that showed the changes in practices and so on. Is there any reason for that?

MR. CHISHOLM: Yes, the reason was I was asked to make this presentation in ten minutes and I wanted to keep the number of --.

MR. RAY: Could you make that available to us?

MR. CHISHOLM: I certainly could, yes.

MR. RAY: The other question is among the recommendations in 0666 NUREG, I saw many tests and maintenance activities. Either I wasn't listening hard enough or you didn't mention that.

(Slide.)

MR. CHISHOLM: I went over it very briefly, but I will be glad to expand on it.

Some of the things that we have initiated are periodic surveillance of the terminal connections on the batteries, and we are doing that in two ways. One is checking, doing a periodic talk test on the connections. The other is we are trying out a thermal surveillance technique that hasn't been put into procedure yet but it's being used.

The other things in terms of procedures, we have written some new procedures, recovery procedures, where if one battery is degraded we have a specific procedure about what the plant people should do to avoid jeopardizing the second one.

As far as the maintenance procedures, they have been reviewed and in cases where there was a possibility of those procedures leading to some kind of damage to the batteries, we tried to put precautions in the procedures so the person that was doing them would be aware of these things and would avoid them.

MR. RAY: How about training the people who are going to do that? Are you going to change your training practices?

MR. CHISHOLM: I'm not responsible for training so I really can't respond to that directly.

MR. RAY: Certainly, the maintenance of the heat of the system, if I could put it that way, should be -- proper maintenance of that and techniques and so on, procedures -- should be a major portion of your training program.

MR. CLARK: Mr. Ross who is the next speaker will try to address that. While we can't address it specifically, I can tell you that we have much increased the level and amount of maintenance training, recognizing just what you say. I can't specifically say whether this point is in it or not.

MR. RAY: But you recognize the need for it.

MR. CLARK: Absolutely.

MR. RAY: How about an inspection of the DC system? Are you going to change your policies and make that more frequent?

MR. CHISHOLM: I think the frequency of our test is -- it's more frequent than the assumptions that were made in NUREG-0666. I don't know the exact numbers, but our testing is more frequent than that.

MR. RAY: Thank you.

MR. MOELLER: Any other questions? Thank you, Mr. Chisholm. Do we have questions or comments by the staff on this topic?

MR. STOLZ: John Stolz speaking. We believe that in general, the TMI-1 DC battery system meets the minimum requirements of 0666 with respect to reliability, and certainly meets three our of four of the recommendations of that report. So we believe that accordingly, it is acceptable.

MR. OKRENT: Excuse me. What do you mean when you say it meets the minimum requirements of 0666?

MR. STOLZ: That is that it met the objective of increasing the reliability to the level stated in that report.

MR. OKRENT: Okay.

MR. KERR: Is there a maximum requirement of 0666 or just one requirement?

MR. STOLZ: I'm not aware of a maximum requirement.

MR. KERR: So it really met the requirement, then.

MR. STOLZ: Yes, sir.

MR. MOELLER: The next topic is the licensee to discuss other studies underway.

MR. CLARK: Mr. Ross as the Manager of Plant Operations will address this.

MR. MOELLER: Thank you. Mr. Ross?

MR. CLARK: I want to address that. He will address the ECCS outage questions. Some of the other questions will be addressed later, by someone filling in for Mr. Wilson.

MR. MOELLER: Fine.

MR. CLARK: In the interest of time, can I suggest we move to the next item? We have a point under discussion here on this.

MR. MOELLER: Fine, let's go into the discussion of additional SCRAM versus PORV actuation.

MR. CLARK: That's Mr. Keaton.

MR. MOELLER: Mr. Keaton, we have listed 20 minutes for this side. I think we will go with the licensee first and then ask the staff for any comments.

MR. KEATON: I don't have any viewgraphs for this portion of the presentation because my answer on this is very quick to give.

We have not done a quantitative evaluation of the change in probability and the associated change in risks associated with the switch and the setpoints of the PORV in the high pressure trip. It was our qualitative evaluation that given the present hardware performances we have seen throughout the industry, that to make the change so that the trip on high pressure occurred before the PORV would be actuated, was qualitatively a direction in the direction of improved safety. And so we concurred with the staff that that change should be made and we have made it.

As a result of the overall plant risk assessment study which I described earlier, we feel that once those numbers are available, that we would be able to use them as a basis for doing something quantitative, but we do not have them at the present time.

MR. MOELLER: Okay. Why don't we move to the staff, then, and ask for their comments on this subject? Of course, the committee has commented on the need for quantification of the risks, or whether this does lead to diminished risk.

MR. CHOW: My name is Ed Chow, I'm with the

Reliability and Risk Assessment Branch in NRC. Today, I would like to talk about operating data on PORV and reactor trips at B&W plants. Basically, I will break down my talk into two parts; one is a comparison of the PORV openings with reactor trips pre-TMI and post-TMI; and my second one covers the number of trips pre-TMI and post-TMI.

So the post part would be the number of PORV openings and the information I will be using is based on NUREG-0667. That's the transient response of B&W designed reactors.

If you just look at it pre-TMI, the number of openings is 149. So the average number of PORV openings per reactor year is estimated to be about five. And if you compare that with post-TMI, as far as we know there are two incidents where the PORV opens with reactor trip. So if we estimate the average number of PORV openings per reactor year, it's about 11.

The second part I wanted to talk about is the number of reactor trips.

MR. KERR: I didn't understand the point you were making. Are you saying that there are simply fewer trips now than there were before, or fewer trips per year or per plant?

MR. MOELLER: As I understood he, included here both the PORV opening and a trip. Is that correct?

MR. CHOW: That is correct. I am talking about a challenge to PORVs. I am talking about a reduction in number of PORV trips after TMI.

MR. KERR: At TMI?

MR. CHOW: After TMI.

MR. KERR: Would you mind repeating what you said, and maybe I will understand it this time. I won't ask for a third time.

MR. CHOW: Just briefly, the number of PORV openings per reactor trips. We were talking about transients where the pressure in the primary coolant system goes up so high so they actuate PORV. It also goes further up to cause reactor trips, and pre-TMI the number is 149.

MR. MOELLER: And that 149 equals five per year for all of our reactors or for what?

MR. CHOW: Five per reactor year. That is for all the B&W reactors.

MR. KERR: And this is an event in which one got both a PORV opening --

MR. CHOW: Both PORV openings and reactor trips.

MR. MOELLER: And that is before the change in the settings.

MR. KERR: Right.

MR. CHOW: And when we compared that with the post-TMI situation where we have a difference in terms of the PORV setpoints and reactor high pressure trips, we have a reduction in the number of PORV openings with reactor trips, and as a matter of fact we have two PORV openings with reactor trips after TMI.

MR. KERR: Okay. I think I understand what you are saying. What I would have been interested in, but maybe you are going to tell me, is the number of times one got PORV openings without reactor trips, and the number of additional trips that have been caused by the resetting. I assume earlier one would have gotten PORV openings which didn't produce reactor trips and now one is getting reactor trips that would have previously been PORV openings.

Are you going to give me data on that?

MR. CHOW: Yes, sir. In the second part I will address the number of trips.

When we look at the number of trips pre-TMI according to NUREG-0667, we have 232 number of trips. So keeping in mind these trips include the number of PORV openings with reactor trips plus other trips, so if we just estimate using the average number of reactor trips per reactor year, it's about 7 or 8, somewhere between 7 and 8.

When we look at the post-TMI data in the year of 1979, the number of reactor trips actually is 45, and if we go further, in 1980 the number of reactor trips is 34. And for this year since we only have data for the first six months, we have 13 reactor trips.

So as far as comparing the number of trips after TMI, we believe there is not a significant increase in terms of reactor trips. Obviously in '79 there were more trips than there were in '80. The reason is due to larger number of startup and shutdowns and also due to the operator lacking experience in using the setpoints.

So based upon our conclusion, we say that with the new setpoints the average number of PORV openings with reactor trips has been reduced, and with the new setpoints the number of reactor trips does not indicate that it has increased in a considerable manner.

MR. BENDER: Aren't you a little skeptical of drawing conclusions from those data? I would be a bit skeptical because I can't see why, unless the conditions for operation have changed, that one wouldn't be getting more trips now, because previously when you wouldn't have gotten a trip and would have gotten a PORV opening, you now should get a trip. I just don't see why if conditions were similar you wouldn't be getting more trips, but I must be missing something.

MR. CHOW: The conditions are different not just in terms of setpoint of change. Also you have in terms of the modification upgrading of other systems involved.

MR. TADANI: Dr. Kerr, I would tend to agree with you. Yes, indeed one would be skeptical to increase the number of setpoints to trip the reactor. You would expect more trips. I don't think we have had enough experience yet to determine. I think we have confidence that the trip frequency hasn't gone up significantly.

I think what Ed is trying to say is on one hand you do see a dramatic reduction in challenges to PORVs, and on the other hand you do not see a significant increase in trips. Only time can tell us.

MR. KERR: If I didn't know any better, I might conclude that was because the reactors were operating only half the time and so you wouldn't get either trips or PORVs opening.

MR. TADANI: You are absolutely right. Other considerations would come in to play. But even if there were a factor of 2 in there, you do see significant reductions of PORV challenges. I think that is the key point.

The other point is, of course, even if you did not trip the reactor, you would need to challenge other systems such as PORV.

MR. KERR: Well, the question that I raised and perhaps others was whether there is greater risk associated with the PORV opening than there is with the trip, and I don't see that one necessarily has to know how many trips occur per year in order to answer that question, but maybe you do.

MR. TADANI: I think there are two ways one could look at that. It is obviously a fairly complex question. One way would be to go to structural people and ask them if the frequency is going up by a certain factor, if you will, would you be concerned? The general response from mechanical engineering people has been they are not concerned about the frequencies that they are seeing.

MR. KERR: I don't think either one of us should try to answer the question here this morning. I just wanted to make certain that you understood the question I was raising.

MR. TADANI: I think I understand it.

MR. KERR: I think qualitatively people have concluded that there is less risk in tripping than PORV opening, and that is the reason for changing setting, isn't it?

MR. TADANI: Yes. Because of the significant reduction, that is to me critical. It is a factor of 50 or so reduction in challenge frequency.

MR. BENDER: When I listened to the statistics, the number I heard only identified PORV openings that led to trip. Now, do we know anything about PORV openings that didn't lead to trip?

MR. TADANI: I think I am not aware of any such situation since the TMI-2 event. Now, I believe there were events like that prior to TMI-2.

MR. BENDER: Are you saying you are not aware of them and therefore there were none, or that there just is no record of any?

MR. KERR: Under the present setting this couldn't occur unless something went wrong because you trip at a lower pressure, don't you?

MR. TADANI: I suppose it is possible, depending on the logic for scram. You may need two out of four pressure channels to trip the reactor, but you only need one failure channel in the pressure channel to open the PORV. I would still expect the reactor would scram even if the initiating event is in the PORV.

MR. BENDER: Well, if the answer is every time the PORV opens now there is a scram, or let's say 95 percent of the cases --

MR. TADANI: That would be my expectation, yes.

MR. BENDER: Okay, fine.

MR. MOELLER: Other questions or comments?

Have you finished your presentation or do you have more?

MR. CHOW: Yes, that pretty much summarizes what I have said.

MR. MOELLER: All right.

Any other questions for Mr. Chow?

Well, thank you.

We will move on, then. Do you want to pick up the item we just skipped?

MR. CLARK: Either way.

MR. MOELLER: Okay, let's pick it up, and that is the ECCS outages. That is, as I recall, primarily simply a reporting of the frequency of occurrences; is that right?

MR. CLARK: We believed it was on the agenda because it was still an open item. Mr. Ross is going to give the status of where we are in closing it out.

MR. MOELLER: Fine.

MR. ROSS: My name is Mike Ross. I will just pretty much give you the same type report I gave to the subcommittee last week, our status in closing out this open item with the staff, and the item is ECCS outage of reporting.

It is pretty much the same as we had reported. Basically we sent a letter to the staff outlining our approach to closing this out. We will go through and review all licensee event reports, tabulate the time that the ECCS-type equipment was involved in this and the amount of time it was out of service. In addition, we will go through and tabulate the amount of time the ECCS equipment is out of service for surveillances.

This time we reported to the staff, and in the report by the end of this month, July, we feel that the staff will accept this as our final submittal and we will close the item.

MR. MOELLER: Have you looked at LERs to see how easily you can pick out this information?

MR. CLARK: Roy Harding from Licensing.

MR. HARDING: My name is Roy Harding. I have been involved in part of the review of the LERs. Some that I have looked at of the LERs from approximately 1975 to 1980 -- some of the LERs have the exact time frames. Others do not but it can be easily inferred., A very good guesstimate can be given as to what that time frame was.

We also have some maintenance record printouts that we can use. Using the date from the LER, we can bounce that) against the date of the LER and determine time frames from that.

MR. PLESSET: Could the staff tell me what is involved on this point, what is the concern?

MR. NOVAK: The staff's concern is to identify if in fact a tech spec which permits ECCS equipment to be taken out of service could then be returned and taken out without any concern for the integrated availability of that system.

MR. PLESSET: What does that mean, Tom?

MR. NOVAK: Well, what we are looking at is the following. We are wondering if in fact a tech spec shouldn't be developed so that in fact it says as you go through an operating year, there should be an assurance that the availability of that system is high enough that it is there most of the time. The way a tech spec is written on ECCS is that if something goes out, you have 72 hours that it may be out, and if you can't fix it, then you have to do something, and then next week that same thing can happen again and there is no history affected, there is no history penalty.

Our approach is there should be some sort of bookkeeping that says the system should not be going out every week and considered to be reliable even though they restore it. So the concern is look at the history of the outage of that system and see if there isn't a criteria that should be established to assure that over a long period of time it is more availabale than unavailable.

MR. PLESSET: So here is a safety system which you allow to be out and the plant continues to operate.

MR. NOVAK: Where, there is a limiting condition of operation. In other words, there is a restoration time permitted. It isn't automatic. You may have a shift, you may have a day, you may have three days, depending on the equipment, to bring it back into an available status.

MR. PLESSET: During which the plant operates?

MR. NOVAK: That is correct.

MR. PLESSET: Are there any special procedures applied?

MR. NOVAK: Generally yes. You check out the remaining systems to make sure they are available.

MR. PLESSET: How careful are they to follow this? That is an important point because this is really not a good situation.

MR. NOVAK: By technical specifications they are required to, and if they fail to perform surveillance test of the sister train, so to speak --

MR. PLESSET: Okay. That is what I wanted to be sure that they had good procedures for this.

MR. MOELLER: Now, the LER that is being done is a generic study. Is it only for B&W plants?

MR. NOVAK: No, it would apply to all plants.

MR. MOELLER: Okay. Any other questions or comments? Fine.

We will move on to the next item, which is a discussion of the effectivenss of the 4 psig containment isolation setpoint, and this is with regard to the 30 degree setting on purge valves.

Why don't we go with the staff first and ask for comments on that?

MR. NOVAK: Peter Hearne of the Containent Systems B ,h will make a short presentation.

MR. HEARNE: With the first valve the licensee is required to meet the first parameters for his isolation signals. He is to sense at least two parameters over a spectrum of LOCA breaks. That is to assure that he senses one parameter for each individual break.

The licensee has chosen 4 psig containment pressure trip as one parameter. That would be mainly used for the large LOCA breaks. He has also chosen the reactor pressure parameter and the reactor trip signal and that would cover all breaks. Each of these isolation signals are redundant. With those two parameters TMI satisfies the requirements for the purge valves.

MR. MOELLER: But is the 4 psig meaningful? Would it ever take place? That is our question.

MR. HEARNE: For the large breaks it would.

MR. MOELLER: For the large breaks it would.

MR. NOVAK: I und rstand the licensee in their presentation will indicate some more information about pressure versus break size.

MR. MOELLER: How did you conclude that it would reach 4 psig and in what time frame for a large break?

MR. HEARNE: It is based on other plants using a single parameter, and it is based on a qualitative judgment that you have an event which will have a maximum of a 4 psi pressure drop across the purge line which is 48 inches in diameter versus a 38 inch source in the reactor coolant. You will have a pressure drop, and if you think about it too long it will not take very long before --

MR. CLARK: We have some analytical results.

MR. MOELLER: Fine. I think that is fine. That is what I was looking for.

Does that complete your remarks?

MR. HEARNE: Yes.

MR. MOELLER: Okay, let's go on.

MR. CLARK: Mr. Broughton, Systems Analysis Director, will make this presentation.

MR. MOELLER: Fine.

(Slide)

MR. BROUGHTON: Before we go into the effect of the open purge valves on pressure, I want to indicate that in the handout there is the isolation signals that are used in TMI-1. For each of the lines to be isolated there are at least two different signals, and in some cases there are three different signals which could cause the valve to shut or a line to be isolated.

The original isolation system on TMI-1 prior to these revisions for restart was basically in isolation on the 4 psig building pressure, and all the items on this list would have isolated on 4 psig pressure with the exception of the purge valve, which would isolate not only at 4 psig but also at high radiation level.

So the areas that are blocked in indicate where diversity has been added to the system as a result of modifications.

(Slide)

In response to the question of what effects did the open purge valves have on containment pressure, during the past two weeks we have made some computer analyses using the CONTEMPT code, which was a code normally used to predict containment response for licensing design basis events. This chart here presents a summary of that information.

The upper line that says "purge valves closed" indicates the maximum containment pressure you would expect to see for a given break size using standard FSAR assumptions for loss of coolant action. The lower line, which is labeled "purge valve open," represents the additional information that we developed during the past two weeks.

I would like to point out that there are three main regions to consider here in discussing the effect of purge valve opening. The first is the region over to the right, in which even with the purge valves open the containment pressure reaches four pounds and continues to rise, so this would be for large break LOCAS, that is, anything greater than .5 square feet. And it is also for a considerable number of small break LOCAS.

The next region to look at is this region here, which is less than about .005 square feet, which are usually termed very small breaks, in which 4 psi is not reached regardless of whether the purge valves are open or shut, and the reason is there is enough containment coolant to be able to remove the energy that is being released.

The third region is the one in the middle, which would be small breaks, which if the purge valves were shut, would cause 4 psig to be reached, but with the purge valves open would not. The maximum break size that falls into that category is on the order of .1 square feet. It actually comes down to about .85 square feet.

MR. MOELLER: Where is the pressure reading taken to actuate these things?

MR. BROUGHTON: This containment pressure refers to the average pressure within the containment.

MR. MOELLER: Are there three or four pressure gauges or how many?

MR. BROUGHTON: I believe there are three pressure sensors inside the building. I believe they are physically located outside the building. But they are part of the safeguard system which has been installed in the plant.

MR. MOELLER: But there are three in there in three different places, presumably -- or the sensors are.

MR. BROUGHTON: They are in three different places. I'm not sure whether they are on different levels or around the building at different radial locations.

MR. MOELLER: And was their location considered in the CONTEMPT code?

MR. BROUGHTON: The CONTEMPT is a homogeneous model so it computes the average pressure that you would see in the system, and in general these transients are slow enough that that is reasonable. That is especially true for the small breaks in which it may take 60 seconds or so to reach 4 pounds.

MR. MOELLER: I was thinking of the small breaks, that's true. I was just wondering if in the larger breaks the pressure gauge was down somewhere where it might sense the pressure increase a little earlier, or at least one of them being where they would sense it as early as possible.

MR. BROUGHTON: The significance of the break at which 4 psi is reached is roughly as follows. For small breaks that are on the order of .08 to .1 square feet, the core remains covered throughout the scenario so the likelihood of gross fuel damage is small.

It is only for breaks on the order of .08 or .1 square feet or larger in which uncovery would be expected to occur and therefore the likelihood of damage in high radiation releases would be greater.

(Slide)

The assumptions that we have used are ones that are pertinent to TMI-1. There are two 48-inch purge lines. In each line, one valve is limited to 30 degrees open, and that produces equivalent line size of about 17 inches.

MR. MOELLER: At the time of the TMI-2 accident, what was the situation? Were the purge valves open at 30 degrees then?

MR. BROUGHTON: At TMI-2 the purge valves were shut at the time of the accident. At TMI-1 the limitation of 30 degrees open has just recently been -- we did not have that restriction prior.

MR. MOELLER: Okay.

MR. BROUGHTON: We have assumed in the analysis I have shown you that the purge valves did not close on any of the other signals that would cause them to. Therefore, the pressure increases that you saw were pressure increases which would exist even with the valves open, and we also assumed that maximum heat removal from other sources was present, which would tend to decrease the pressure estimate.

The results, as we have discussed, are that for large breaks and even small breaks at the larger end of that spectrum, 4 psig is still reached. For the small breaks, 4 psig is not reached, regardless of whether the valves are open or shut. And the conclusion we drew from this is that the 4 psi signal is still valid if the purge valves are limited to 30 degrees open.

The reactor trip isolation signal, which is the primary containment isolation signal, anticipates both the building pressure condition and the safety systems injection for Three Mile Island.

MR. MOELLER: Thank you.

Any questions on this topic, further questions?

There being none, let's move on to Item I on our agenda, which is the control room design review, and that will be a presentation by the licensee.

MR. CLARK: Mr. Broughton will make that as well.

MR. MOELLER: Fine.

(Slide)

MR. BROUGHTON: In reviewing the ACRS Subcommittee transcripts, we felt there were two items we should provide additional comment on. Those items are included in the handout. Dr. Moeller's comments this morning included two additional items which I will also address.

An item brought up at the subcommittee meeting was the concern of the use of dual meter scale markings to aid the operator in determining saturation conditions in the primary. We have reviewed how we will be providing this information to the operator and what we believe the requiremens are. We have tried to summarize those on this slide.

Since the TMI-2 accident, saturation margin has now become a basis for operator action. Prior to the TMI-2 accident that was not the case. Therefore, since we are requiring an operator to be aware of this and take actions baed on it, it is prudent to provide him with a continuous indication of that particular parameter.

If the operator needs to look at both temperature and pressure separately to evaluate the saturation margin, there are several problems with that which might preclude him from acting in a timely manner. First of all, that is not a continuous display; that says he chooses to look at it and go through the process of evaluating.

There are several steps involved in determining what the saturation margin is. That involves that the operator do more than simply look at a value and understand it, and the whole process is subject to some error under stress, which is the time in which an understanding of what this parameter is is the most important. If the choice were between using steam tables to do this and a dual scale meter, it is probably the dual scale meters that would be the preferable choice. However, we believe the preferred method is the direct reading instrument.

At TMI-2 we have two direct reading instruments. One monitors each of the two loops. We have an alarm which would enunciate if the saturation margin reached a predetermined low value, another indication to the operator of what this parameter is. We also calculate saturation margins in the process computer as a backup to the alarms and hardware instrumentationm, and in addition we provide a graphic display of temperature and pressure to also allow evaluation.

MR. RAY: You say TMI-2. You mean TMI-1.

MR. BROUGHTON: I'm sorry. I meant TMI-1.

(Slide)

MR. PLESSET: You don't have a graphic display that indicates whether you are in the subcooled region or not. In other words, there is a saturation line, and the condition of the plant could either be above or below it. You don't have that. You will have it?

MR. BROUGHTON: Yes, we will have a graphic display of pressure versus temperature. It will show the saturation line and the minimum subcooling margin line, and then it will show the plant status with respect to that.

MR. PLESSET: Okay. I wanted it clear. Fine.

MR. BROUGHTON: The second item which was discussed concerns midscale meter failures. I have broken this up into two pieces because the way in which they are being identified to the operator is dependent upon what the real cause of the meter failure is. First of all, in our control room review we have guidelines which say that if a meter fails, that that failure should be evident to the operator without predefining how to make that evident, so there may be several different acceptable ways to identify to an operator that he has a malfunctioning meter.

The concern with midscale meter failures is that if the failure point is near the normal operating point, then it may not be evident to the operator that failure occurred. One of the concerns with the midscale meter failures was evident through the Crystal River 3 event in which the failure of power supplies caused several different key meters to fail in position and provided erroneous information.

With regard to how we would be dealing with that, the Crystal River 3 changes that were made to the plant are the type that will identify to the operator that he has had the power supply loss. He will have meters which are powered separately from the lost buss so he can still monitor the plant properly. He now has procedures which tell him which meters to use, given these power supply losses, and those will be meters that are powered and active.

And finally, the individual meters will have failure points indicated on them which will show where the meter would fail, and it is also keyed to the power supply to drive the meter. So we feel in the case of power supply losses that that is well covered by the changes we have made to the plant.

The individual meter failures -- this could be a failure of either the meter itself or the sensing circuit that would be providing information to the meter. We have two types in use. The first type is the analog meter, which was the real subject of this comment in the control room. In general the analog meters fail at a midpoint on the scale. We will be adding the failure point to those meters so it is evident where the meter failure point is.

The second type of meters -- and the bulk of these have been added since the TMI-2 accident -- are digital meters. They would fail dark or off if the meter power was lost, and if the meter power was still available but the sensor was malfunctioning, it would fail to a reading of zero.

In looking at the analog meters, most of the failure points, even though they are midscale, do fall out of normal operating, so the operator would be alerted to something abnormal by that meter. And in those cases the marking of the failure points should be adequate to indicate that the meter should be checked for operability.

The ones which were of concern are the analog meter failure points which fall in the normal band.

We have evaluated those and considered several things. For example, if there are redundant meters to measure that parameter in close proximity to that failed meter, that is a viable method for the operator to use to see if his meter really is functioning or malfunctioning.

MR. MOELLER: How does he know which one is working?

MR. BROUGHTON: Well, he has, for example, the failure point marked on one. But in the case of one of the redundant ones, an example I have given you is the midscale point on it was 500 pounds, which is the normal setpoint.

So from that you can't tell if that meter is working, but the redundant meter was a wider scale meter so its midpoint was 600 pounds. So in that case, if it failed at 900 pounds the other one reads at 900 pounds but not its failure point.

MR. CKRENT: Is there only one unique failure point for a meter?

MR. BROUGHTON: There is one unique failure point on loss of power for the meter. If there is a malfunction in the sensing circuit somewhere, the failure could be anywhere. But the most common are loss of power related either to the meter or the sensing circuit.

MR. OKRENT: I appreciate that, but I am just wondering whether in this discussion about the failure point you have asked yourself are there situations that in fact lead to something else. I don't know whether this is the right one, but maybe before you lose power you have a surge in power, and as I have had a surge in temperature in my car and now the meter that reads temperature in my car no longer sits at zero, the failure -- the operator -- I'm just wondering if you had thought it all through.

MR. BROUGHTON: When we say failure points here we specifically mean failure points on loss of power, not out of calibration conditions or malfunctions.

Another consideration --

MR. KERR: It is impossible for the sequence described by Dr. Okrent to occur just before a loss of power, for the meter to come back to the failure point even though it had just been up?

MR. BROUGHTON: It certainly would be possible for a mechanical meter to bind at any point even if it lost power, but those are very unlikely failures. Normally the meter is designed such that even with surges in the measured parameter it is adequately protected against physical damage.

So the use of redundant parameters is one way of providing us information in the case of midscale failure.

Another consideration is the length of time required for the operator to identify that he has a failed meter before he gets into some sort of a problem. For example, one of the flow meters in the decay heat removal system may fail near the normal operating point, but that is not one of the primary parameters monitored in that mode, and there are other parameters that would indicate it.

The third item we looked at in evaluating this is the fact that we have added these new digital meters for key plant parameters. Even though they were added for a different purpose, those meters are added to aid the operator in manual control, primarily when the plant is shut down, and it is important if there is some meter failure when he is in manual that that is not known; it is less important if he knows about that in automatic.

So we look at the key process parameters for which he needs good information. In those cases we now have dual displays and they have different failure modes, and therefore our review concludes that while we still have midscale meter failures, the consequence is not one that we need to be concerned about.

MR. MOELLER: What does a digital meter read upon failure?

MR. BROUGHTON: The lights would go out on it if it lost power. If the sensing circuit itself failed and there was an opening in the transmitter, it then sees a low range value, which would cause a zero reading on the meter, which is different than a measured zero on the parameter.

There were two other items which were mentioned this morning in the summary concerning the auto-manual positions on multiple rotary controls. This is an item which was discovered during the NRC walk-through of our control room, and we had also identified it during our walk-through.

The switches in question amounted to four initially, and in further reviewing the way they were marked and what the convention was, it turns out that two of those do meet the auto-manual conventions.

The other two which are different from the normal convention are ones which we hae addressed in our response to the control room review, and our conclusion was that the probability of error while operating these is very low because of the marking of the switch, the function of the switch, the fact that it is very frequently operated and there is very positive feedback to the operator when he goes to operate the switch.

The last item that was mentioned this morning had to do with control of temporary labels, and perhaps our explanation of this during the subcommittee was not clear. 1 The concern was that in many control because of inadequate labeling, operators tend to provide temporary labels so that they can understand the function of various indicators and controls.

Currently we have had no policy at TMI against such temporary labels. There hasn't been any overall plan for the control room to label. As a result of work we are doing, there will be such a plan. The labeling in the system and in the control room will be controlled just like the other aspects of the plant design are controlled, and there will be a prohibition against temporary labels. If a label needs to be changed, it will be changed on the plant as any other engineering change would be made.

These are administrative controls but I think they are adequate in this case, and we don't know of any other approach to take rather than use such administrative approach.

MR. MOELLER: Questions or comments on this presentation?

MR. OKRENT: I have a related question. Could you remind me? Is the computer capability at TMI-1 similar or better or less capable than that you had at TMI-2?

MR. BROUGHTON: The answer is that at the time of restart there will be much greater capability.

(Slide)

I have got one slide here which is not part of your handout which indicates what will be available in terms of the computer at the time of restart. There is a new process computer which will be operable which is a state of the art machine which updates the existing computer at the plant, and there will be additional output devices for the operator in terms of both CRT displays and additional video output devices which will greatly increase the capability in getting information from the computer.

MR. OKRENT: That is enough.

MR. MOELLER: Would the staff comment on the control room and human factors aspects? Do you have any comments?

MR. PLESSET: Or on the previous item, the containment setpoint.

MR. MOELLER: All right. They had commented earlier on that. Go ahead.,

MR. RAMIREZ: Ray Ramirez from the Human Factors Engineering Branch.

We have pretty much accepted the proposals that have been given to us by TMI people as an interim corrective actic . We have to be published in the near future NUREG-0700, which will address all of these items for the longer term. What they have done in the interim is make at least as good as some other plants that are operating.

MR. KERR: So that there is a possibility that within a couple of years they will have to redo the control room?

MR. RAMIREZ: Not the control room entirely, but those items which were discussed during the last half-hour.

MR. MOELLER: So you are developing a manual or a guide on this, and you say at the moment they are as good as the other plants. Could they have been better than the other plants? I mean with all the effort they are putting into it, why don't have them be better?

MR. RAMIREZ: I guess to make it clear, it would make them better than similar plants operating with similar equipment.

MR. MOELLER: They will be better? Okay. That is good to hear.

Mr. Kerr.

MR. KERR: I have no additional questions.

MR. MOELLER: Thank you.

Then we will move on to Item J, which is the startup test program, a presentation by the licensee on their objectives.

MR. CLARK: Mr. Behrle will make the presentation.

MR. BEHRLE: The handout you are receiving is the one I presented to the ACRS Subcommittee on June 26th, and based on the material in that handout, the subcommittee requested I present a discussion on the objectives of the test program and a comparison of low power and natural circulation program with Sequoyah.

At TMI we do our test program consistent with testing to verify proper installation modifications, functional testing to verify design adequacy, and integrated testing to verify proper core performance and plant response.

The basis for test selection that we used is to consider various programs and requirements. We looked at normal reviewing test requirements, at modifications that were made to the plants.

(Slide)

We looked at the initial TMI-1 test program. We looked at the natural circulation testing that is being performed in other NTOL plants. We took a look at Reg Guide 1.68. We also then considered various aspects of plant re-initialization, operator training, procedure verification, surveillance testing, and we looked at plant transient analysis verification.

Some of the key documents that we used to come up with our test requirements are listed on this slide, and they are basically centered around the commitments made in the TMI-1 restart report.

(Slide)

NUREG-0680, which is the safety evaluation to that restart report, NUREG-0737, the clarification of TMI Action Plan requirements, and then the draft Reg Guide for light-water refueling startup test, which we normally adhere to following a shutdown.

(Slide)

Our integrated test program consists of hot functional testing where we verify some aspects of modifications at normal operating temperature and pressure prior to taking the reactor critical. It consists of zero power physics testing where we verify core performance at zero power, zero nuclear power. It consists of the low power natural circulation test program that we have employed for operator training, and the power installation test program that we used to verify adequate plant steady state and transient events.

In comparison with the Sequoyah program, Sequoyah was initially required to perform ten tests using simulated decay heat, which is basically keeping the reactor critical at about 3 percent power, and then subsequent NTOL plants were required to perform eight of these tests using simulated decay heat, and then one, which was the boron mixing test, was permitted with real decay heat following some period of core exposure.

In looking at our program compared to Sequoyah, the first test item is to establish stable, natural circulation. That is included in our program and each shift will participate in or witness a transition to natural circulation.

The second item was to establish natural circulation with simulated loss of off-site power. This is not included in the restart program because we did do this verification during the original startup in 1974. We did perform a full-fledged loss of off-site power test.

MR. MOELLER: Is that particular item -- you say it has already occurred so there is no need to repeat it, but it seems to me you test both equipment and people. Will your people know how to respond?

MR. BEHRLE: That type of situation is one that we do conduct operator training on at the B&W simulator in Lynchburg, so all the operators are exposed to that type training.

MR. MOELLER: Go ahead.

MR. BEHRLE: And in this case we will need to repeat it for each shift and repeat it a number of times, anyway.

MR. CARBON: What is the difference between 1 and

2, actually?

MR. BEHRLE: You mean item 1 and item 2? In item 1 where you establish natural circulation conditions, we trip the reactor coolant pumps and then go to predetermined level, which is 50 percent of the operating range in the steam generator with your emergency feedwater system. And you will see your flow decaying, you will see your hot leg temperature go up probably about 25 or 30 degrees, and your cold leg temperature drop to saturation conditions in the secondary side of the steam generator, and the operator will watch and verify this transition from a forced flow situation into natural circulation condition.

Now, on a total loss of off-site power, again your reactor coolant pumps would trip but your diesels would come on and load the emergency feed pumps on.

MR. CARBON: Fine.

MR. BEHRLE: I guess to answer your question, I don't really see that much difference in terms of what really happens in the plant.

Item 3 was natural circulation with loss of pressurizer heaters, and we have included that in our program. We will shut off the pressurizer heaters under natural circulation flow and verify that for at least a two-hour time period. We can maintain adequate saturation margin.

Item 4 was the effect of secondary side isolation on reactor coolant flow, and although we have not included that in the restart program, that was more or less verified on the B&W nuclear steam supply system during the TMI-2 accident.

The fifth item is natural circulation flow at reduced reactor coolant system pressure, and we really don't have an auxiliary spray that operates under high pressure conditions. So to determine effects on natural circulation flow and reduce pressure, we just extend item 3 for some period of time until we get the reactor coolant system pressure at about 1800 pounds.

Item 6 is the cooldown capability of makeup and letdown systems, and we do have that included in our program. Item 7 is simulated loss of all on-site and off-site AC power. We have part of that item in our program, and the part that we do perform is to verify the independence of the heat sink: in other words, the emergency feedwater system, the independence from AC power. We do not shut off all AC power on the primary side because we want to maintain the reactor coolant pump seals, and this will not cause damage to the seals.

(Slide)

Item 8 was to establish natural circulation from stagnant conditions, and the NRC has deleted this requirement from all the NTOL plants subsequent to Sequoyah, so it is not in the TMI-1 restart program.

Item 9(a) was forced circulation cooldown, and we do incorporate this into our program.

Item 9(b) was boron mixing and cooldown, and we have not included boron mixing in the restart program because that was verified on the B&W NSSS following the Unit 1 accident.

Two additional items were added that were not included in the Sequoyah program. The first of those is to verify plant natural circulation procedures by adequate operator guidance to prevent overcooling ss the steam generator level increases from 30 inches to 50 percent in the operating range.

The second test is to determine the lowest level in the steam generator that sustains natural circulation without emergency feedwater flow, so the way we do this is to use main feedwater flow to provide a certain inventory of water in the steam generator. We trip the reactor coolant pump and block the emergency feedwater from entering the steam generator and just maintain main feedwater to see how low we can get in the steam generator level and still sustain natural circulation.

Are there any questions?

MR. MOELLER: Questions for Mr. Behrle on this?

Yes, Mr. Shewmon.

MR. SHEWMON: I would like to bring up one matter and I am not at all sure you are the one to answer it, but it seems to me about two years ago you were the people to first discover what I would refer to as stagnant line cracking, where you had a splurge of things of that sort. First there was a question of whether you had 100 of them, and then after the NDP procedures got in better shape, it turned out to be only dozens of them or something.

Could you tell me where that situation is and what has been done?

MR. BEHRLE: I don't believe I can respond to that.

MR. CLARK: Mr. Kronberger will respond to that question.

MR. SHEWMON: Okay, thank you.

MR. CLARK: He is scheduled to make the next presentation. Perhaps he could just incorporate that. You don't see it on the agenda. It is the one we skipped.

MR. MOELLER: Is that okay, Paul?

MR. SHEWMON: Thank you.

MR. MOELLER: Let me quickly ask if the staff had any comments on the startup program.

MR. NOVAK: I would like to say we have not completed our review of the licensee's startup program. It is in process and we will be getting together with the licensee in the very near future to discuss the comments that we have.

MR. MOELLER: Do you see any real problems in this area?

MR. NOVAK: Nothing I think that cannot be resolved, but we do indeed have some questions.

MR. MOELLER: Surely.

Okay, go ahead with the next presenation.

MR. CLARK: The next item, which is the status of some ongoing studies, we ask a little bit of forbearance. Mr. Wilson was going to make that presentation and he got food poisoning last night, and we have tried to put together a presentation by Mr. Kronberger.

MR. KRONBERGER: What I would like to do is cover quickly three different areas that were raised during the subcommittee meeting two weeks ago. One was how we are involved in general technical problems from industry's standpoint, specifically what if anything we are doing in the area of hydrogen control and filtered vented containment, and what our activities are in some of the other generic licensing issues.

What I would like to do very quickly is discuss how we see the various technical problems being categorized and what our involvement is. Basically we see three general areas of technical problems: those which are industry generic, those which are generic to the B&W 177 size plants, which is TMI-1, and those which apply to TMI-1 as plant-specific problems.

(Slide)

Organizationally if you take those three categories you have the industry-generic issues which are being handled as industry tasks, which are licensing and regulatory affairs. The degraded core rulemaking would be an example of that.

Our other areas of involvement are those which are being worked, whether industry-generic or B&W-specific, through the EPRI activities. Likewise B&W owners group activities following the same two basic types of categories with some examples down here, which isn't intended to be all-inclusive of certain technical issues which are being addressed organizationally throughe of these.

And then the other ones, which are GPU tasks which are being handled by our engineering projects group and fall in the categories are shown there.

(Slide)

First the extent of our involvement, again trying to run through this quickly. This represents GPUN direct membership on the various EPRI and B&W owners group activities. This indicates the numbers of people and in certain cases chairmanship of certain committees which our people are involved with. I show that simply to indicate that we ar⸗ involved not necessarily more than everyone lse but we do have some substantial involvement on those various industry-wide activities.

(Slide)

Now what I would like to do is jump in this slide -- I'm afraid you don't have it in your handout although the first portion of the slide is on the last page -- to what our activities are on hydrogen control and vented containment. As I am sure you are aware, in TMI-1 we have a large, dry containment which is very analogous to the containment configuration being studied for Zion.

So among the various activities that we are involved with we are trying to keep tabs on what is being done on the studies relative to Zion, relative to the studies that were being done by Sol Levy for EPRI, relative to the studies I know that the NRC has been involved more directly with. We do not have any in-house program directly involved on filtered vented containment.

We have had some studies on the whole area of hydrogen development, which were touched upon in the subcommittee meeting, and such issues as trying to understand more completely the TMI-2 hydrogen burn episode, trying to understand more about hydrogen mixing, trying to understand more about dry containment capability and capacity. But that is the type of activities we have going on in house. We do not have any major program analogous to the one for Zion.

As far as some of the other general areas that were touched upon at the last subcommittee meeting, we did address how we were proceeding on a general problem such as the block wall issue.

On the seismic interaction question. We indicated last time that we were not specifically proceeding for TMI-1 at this time on reevaluating the acceptability of the original design as far as seismic interaction, and at that time it was indicated that this was being done for Diablo Canyon. Basically what our posture is, although we recognize this is an issue to address, recognizing that we are in a relatively low seismic area at TMI-1, we consider it to be a problem although not necessarily the highest priority problem to address right now.

As far as containment flooding, we may have given the wrong impression as to what we have done on the problem that did crop up at Indian Point Unit 2. We did thoroughly evaluate our design vis-a-vis the Indian Point situation. There are some major differences on our plant as contrasted with Indian Point. We have no raw river lines. We only have one raw river line which enters the containment, and that is only actuated when we have an ES signal, and we are worrying about post-accident air cooling.

So we have raw water going to the containment for post-accident cooling, but that is the only one. We have studied deeply other flooding problems outside containment as it relates to river water lines. We have thoroughly studied flooding in containment as a result of our experience in TMI-2 as it involves such issues as the 7901(d) environmental qualification of equipment, so I do think that in the area of flooding we have done more.

MR. OKRENT: With regard to what is sometimes called internally induced flooding, have you looked at the rest of your plant, meaning what is not in containment, to see whether there are important functions which are vulnerable to flooding of this kind, and whether such flooding is physically possible from breaks in equipment? And if so, has there been some kind of evaluation of the chance of a serious accident being caused via this mechanism?

MR. CRONEBERGER: If I could, I'll just try to describe the various types of things we have looked at. As a result of the work we did on high energy line breaks, the concern about potential flooding was very thoroughly analyzed and we satisfied ourselves that we didn't have any detrimental effects associated with those lines.

We have, as a result of the Indian Point issue, investigated where we have river water lines entering the auxiliary structures, and have concluded that essentially the only area where those lines entered the building, which is an isolated heat exchanger vault, that we have very substantial floodable volume which represents what the most severe -- the highest flow rate line, in fact, should break approximately 30 minutes for operation action to terminate those services.

We have looked at large tanks such as -- .

MR. OKRENT: Is that the cooling water? The main cooling water?

MR. CRONEBERGER: No, it's a number of cooling water services which are the open cycle cooling water services for decay heat removal, nuclear services cooling, secondary services lines.

MR. OKRENT: How about condenser cooling?

MR. CRONEBERGER: That had been looked at before as a separate issue which is a possible flooding problem in the turbine building. The turbine building in TMI-1 -- there is provision for basically, the turbine building essentially discharge the water out.

MR. OKRENT: Are you telling me, then, that there is no way that internally-caused flooding could lead to a serious accident? What is it you're telling me?

MR. CRONEBERGER: I'm trying to say that as far as what might be major volumes of water which could enter the plant, these have been looked at in the past. I cannot say that in fact we've done a detailed investigation of the plant and identified all the water sources for the plant and assured ourselves that in fact there is no problem. We've looked at what we consider to be the worst cases.

MR. OKRENT: Well, on one plant that was looked at via some research study, it turned out there was one drum where there was a lot of electrical supplies centralized, and it would have been very awkward to have it flooded. And in fact, it turned out there were at least physically possible mechanisms for doing this. So I don't know whether you have that situation or not, but -- well, a different question.

Earlier, there was a brief discussion by Mr. Keaton concerning plants doing a reliability study on TMI-1 in the coming months or years. Was it contemplated that this study would encompass a look at whether there were safety improvements that met some kind of a cost-benefit effectiveness criterion to see whether, in fact, there might be some things worthwhile doing on TMI-1?

MR. KEATON: Yes, we would try to pick up that type of thing. And I meant to imply that in saying that we would address external hazards which would include things like flooding. And we would expect to look, for example, for the type of thing where there was one area in a plant where, because of all the different types of equipment that are there, that it was a particularly vulnerable area.

MR. OKRENT: My question was quite general. Flooding would only be one aspect that one might consider. I don't want to pursue it now.

MR. KEATON: I think, though, that our general answer is that as part of the study we do intend to look for risks associated with the common location of equipment, in addition to just looking at random failures.

MR. MOELLER: Any other questions for Mr. Croneberger?

MR. SHEWMON: He hasn't gotten to the last point yet. Let me ask a question to bring it up. It seems to me one of the more distressing things about Indian Point was the fact that pipes ran for a lot longer than 30 minutes before anybody thought it was worth going inside to look. And I would be interested in having you reassure me by telling me what procedures you have instituted to make sure that water won't run into the sump for a day or two unbothered or molested.

MR. CRONEBERGER: Are you referring to the Indian Point situation of water inside the containment? We have installed now both the new sump level detection systems plus the building level indication and alarms. So within the plant as modified, the operator should be getting indication which I believe was not available to the Indian Point operators.

MR. SHEWMON: As I understand, they had some indicators like that but the operators knew they didn't work or werent' reliable anyway, so they just ignored them when they alarmed.

MR. CLARK: We would like Mr. Ross from the plant operations staff to address how they would respond.

MR. ROSS: In addition to the changed parts of increased instrumentation Mr. Croneberger talked about, we have reviewed in detail this incident with all operators. We have changed some operating procedures to initiate such things as checks of sump level indicators on a weekly basis to make sure the operators have faith in them.

MR. SHEWMON: If they didn't work, then what happens?

MR. ROSS: At that time, we increase our surveillance time and take action to get something done with it. We recognize the importance of that sump.

In addition to training, I would say our increased surveillance is a major item, and operator awareness.

MR. SHEWMON: Okay. Now let's go back to the last item.

MR. CRONEBERGER: You're talking about pipe cracking?

MR. SHEWMON: Yes.

MR. CRONEBERGER: As a result of improved UT standards that were used, the amount of indications which were identified as inter-granular stress corrosion was substantially reduced from what the first identified number was. To my knowledge, we have now completely repaired all of the indications which we have identified as requiring that.

MR. SHEWMON: So you have no particular idea of why you had this spate of things, but you are sure you have replaced all the things, so hopefully it won't come back for a couple of years, with luck. Is that it?

MR. CRONEBERGER: We have entered into an agreement with EPRI on some testing of the system which had most of the indications, which was one of the spent fuel cooling systems. I'm really not prepared technically to address what that testing program is.

There is some instrumentaton installed; I am just not sufficiently aware of the nature of what that program is.

MR. SHEWMON: My impression is that there were sensitized welds which I guess meet specs in these non-safety grade systems. But there was also more active corrosion in those than most other plants had. And I appreciate the fact that you don't know. My impression is no one else did; at least, the last time I asked about this so I was curious.

MR. CRONEBERGER: The major number of indications occurred in one of the redundant spent fuel cooling lines which happened to be stagnant since the plant went into service. One of the outcomes of such an evaluation obviously, were there, in fact, systems to initiate a practice to move the water through that system periodically, which is being addressed.

MR. MOELLER: Does that complete your presentation?

MR. CRONEBERGER: Yes, sir.

MR. MOELLER: Thank you. Are there any other questions for Mr. Croneberger? If there are none, why don't we ask the licensee if there are any additional remarks or comments that you have at this time.

MR. CLARK: No.

MR. MOELLER: Then could we ask the staff to sort of give us a summary statement on where you stand with respect to the TMI-1 restart?

MR. NOVAK: As the committee knows, we have summarized our review in a series of SER's, NUREG-0680 and its supplements and a variety of others, which I didn't present to the subcommittee at that time. There are still some matters outstanding; among them, the startup test review which I mentioned a little while ago and several other relatively minor matters which will be resolved prior to restart.

For the information of the committee, the evidentiary hearing has concluded yesterday I believe, in fact, and that process is, of course, continuing in proposed findings and presumably, in September, at least that's the current schedule, a recommendation or an initial decision by the Board to the Commission is to establish a general schedule for future events.

I don't think we have any further comments on specific items discussed here today, but of course, we are prepared to answer any further questions the committee may have.

MR. MOELLER: Are there any additional questions for the staff? Max Carbon?

MR. CARBON: I'd like to go back to the emergency preparedness discussion for just a moment. In an emergency, is there anyplace where the NRC resident inspector is not permitted to go? Is he limited in his movement?

MR. ARNOLD: I think I could answer from us that there is no area that he would be restricted from because of him being an NRC employee. He can go in the control room or wherever he wishes, and that is, in fact, where we would expect him to be. The only restriction would be those that apply from the radiological controls program.

MR. CARBON: Thank you.

MR. MOELLER: Any other questions? There being none, Mr. Chairman, I think that concludes the items that the full committee had on the agenda to review at this time, and the remaining item is to poll the committee in terms of their thoughts on this particular implication.

MR. MARK: Are you making a proposal that a letter should be prepared on this?

MR. MOELLER: I would propose that the committee consider writing a letter. Of course, it will contain a number of items that we would want to comment on, so what we would need to know from the members would be, one, do they agree that we are in a position to attempt to write a letter; and secondly, if they agree to that, what should we cover in the letter, what are the key items that we should mention.

MR. MARK: Is there an opinion that we are not prepared or should not attempt to compose the letter on the restart? Plus, I think it depends a little bit on what the letter say.

MR. MOELLER: Certainly.

MR. PLESSET: And I guess that's obvious, and depending on what the letter says I may have some additional comments along the lines that I mentioned in the opening statement.

MR. MARK: It sounds as if the letter could be undertaken. There will be a fair number of items on which comments will be called for. Perhaps including -- .

MR. PLESSET: My additional comments might be entirely unnecessary.

MR. MOELLER: Right. Certainly, we request from Mr. Plesset his thoughts, and we will first attempt to put them in the letter itself, and where we are unable to do so, then he is certainly free to add them as remarks.

MR. KERR: And if they are entirely unnecessary we will try to dissuade him.

(Laughter.)

MR. MARK: I guess that's as far as we can take this at the moment. In spite of the fact that we're all starving to death, I believe we should ask Mr. Tedesco, who has come to address us an hour ago, to tell us a little bit about the staff's picture of the future program. That may not take very long.

MR. KERR: You wouldn't consider a five-minute break?

(Laughter.)

MR. MARK: I would like to thank the delegation from TMI who has given us a pretty good picture.

MR. ARNOLD: We thank the committee very much for the opportunity to meet before you.

(Slide.)

MR. TEDESCO: What we wanted to do was share with the committee where we are on the SER's that we had talked about last month. And this is a very brief summary on the five plants, Susquehanna, Shoreham, Waterford, and Comanche Peak.

What we have here is the date that we contemplate issuance. Now you have Susquehanna, with 40. Shoreham, because of a large number of open items that we and the applicant have decided to defer issuing the SER. Waterford should be down to you today, and Comanche Peak we will issue next week. However, there are also a large number of items on that.

Fermi should be out today, and that has about 23 items. But just to give you a very brief overview of the statements on this SER --.

MR. KERR: What is meant by 25 (staff position 3)?

MR. TEDESCO: Here are a number of open issues that we have identified in our SER. These are the ones that presently exist in the SER; 14 on Susquehanna, 41 on Shoreham, 25, 42 and 23. Here is an expectation that by the time the August ACRS comes around, we will have reduced this 14 by 7. That's an expectation that we would have. And here, in addition to open items, there are positions that we have taken in the SER on a number of items.

That gives you kind of an overview. There are two positions here, and that would give you an analysis of the contents of the SER.

(Slide.)

What we have decided to do is make a recommendation to the committee as follows: that with regard to the SER's that we have published, we would request that the committee consider at its August meeting Waterford, Fermi, Susquehanna, and we've decided to defer Shoreham and Comanche Peak reviews until we are in a position to more precisely close out the open items that we have identified.

MR. SHEWMON: Would it be warping your position too much if you said there were around two dozen or fewer issues at this point in time and you think they will clear up by next month? And if there is over three dozen, you are unwilling to project that?

MR. TEDESCO: I wouldn't want to make that a hard rule, but if you felt that way -- .

MR. KERR: Well, unless I misinterpret these, he's anticipating 23 open items at the time of the August meeting for Fermi-2. Am I misinterpreting that number -- 21?

MR. TEDESCO: Fermi would come out to be something like 21. Actually, Fermi-2 now has 22 items.

MR. SHEWMON: No, he says there's 22 now and he thinks there will be on the order of 2 next month.

MR. TEDESCO: No. Two of them will be reduced by next month.

MR. SHEWMON: Oh. Thank you.

MR. MARK: That's the number that will get covered.

MR. SHEWMON. The last column, just out of curiosity, means what?

MR. TEDESCO: Those are positions we have taken. In other words, we have resolved an issue by showing this requirement. This is our requirement.

MR. PLESSET: So this is not on the list anymore?

MR. TEDESCO: No, sir.

MR. SIESS: You stopped arguing?

MR. TEDESCO: Yes, sir.

MR. PLESSET: So those could be taken out, too? Oh, those are already out.

MR. TEDESCO: So that's the picture we would present to you for next month.

Now, there's also one other item. This afternoon you will be hearing the story about Pilgrim, which is the first construction permit application to come to the committee in some time. The information you will receive is based primarily on the TMI requirements that have been established.

Next month, it would be our plan, if the committee so desires, to also present Allens Creek. Now, the committee may feel that perhaps it is not necessary to go through all these issues with all the plants because we would really feel, from the feedback, whether or not you want to continue to see CP's after they have their review.

Basically, Allens Creek would be very much like Pilgrim except Allens Creek is a boiler, and perhaps the most unique thing there is the size of containment. We have gone through dialogue before with the committee on the requirements, so we're going to leave it to you on how you'd like to handle it; whether or not you want to continue to see the CP's.

MR. KERR: While the committee is deciding that question, can you tell me why there are going to be 20 or so open items on Fermi when we are considering it? That still strikes me as being a fairly large number.

MR. TEDESCO: Ideally, we would all like to see zero, but we are working very hard on these reviews.

MR. KERR: Well, even non-ideally, I'd like to see about five or so.

MR. SIESS: This number game bothers me. Are all the items equally significant, equally important? Maybe you can answer that in two parts. One, are they equally important to you, and do you think they are equally important to us.

MR. TEDESCO: There are a number of items that are common to all these plants that are in the long-term review, like equipment qualification, fire protection, emergency preparedness, some on the control room design, some on emergency operating procedures just by virtue of the nature of review. The plants for construction have not gone far enough for us to -- in other words, you still may have them in the control room panel. We just can't really say it's all finished.

MR. SIESS: Some of those are things that there's no argument about? You just haven't had a chance to know what they're doing, but you suspect that they will be done and you expect that you will see that they are done.

MR. TEDESCO: Yes, sir, before licensing.

MR. SIESS: Some of those would stay open right down to the hearing stage, wouldn't they?

MR. TEDESCO: And they may indeed become license conditions and we may say look, before you go above fuel loading or criticality, you must do this.

MR. SIESS: So what would be the chances on, let's take something like Fermi, and take Dr. Kerr's criterion that he would like to see five. How long do you think we might have to wait until you got it down to five?

MR. TEDESCO: Okay, I can tell you on that, we have asked the applicant when he would be able to provide us the necessary information that we would need to write off on these residual items, and as far as Fermi goes 9 of them should start to come into us around August 1st, and then the balance of them start to come in next year.

MR. SIESS: Want to wait that long, Bill?

MR. TEDESCO: No, they haven't started going through the hearing, but this is the type of review going on.

MR. KERR: Well, this is sort of a committee decision. I don't know -- .

MR. SHEWMON: These are coming in because they are contested and we know there are going to be hearings, is that right?

MR. TEDESCO: Yes, sir, and we're minimizing the impact on the availability.

MR. SHEWMON: I appreciate the political pressure for that. What criteria have you used to say yes, I think these are far enough along so that things will be revised. Or, sorry, everything will be accommodated by the time the hearing is over, which I presume is something you have approximated.

MR. TEDESCO: We have to satisfy all these issues to satisfy the regulations.

MR. SHEWMON: The question is what criteria have you used to say these are now ready to move on?

MR. TEDESCO: These have been 96 or 97% reviewed with that point. When you say the SER today, you're talking about 800 pages of material that represents a rather significant amount of review effort on our part. And the items I'm identifying here are not going to be doubling that. There will be a number of items of a residual nature, and we're confident that we can handle them, that we have enough experience on these matters to deal with them.

MR. KERR: And the Fermi SER will be available tomorrow?

MR. TEDESCO: I got it last night, it went into final correction typing this morning. It should be run off either late this afternoon or tomorrow you should have it.

MR. SIESS: Bob, you've been in this business a long time. I've got a suspicion that we are seeing SER's now with lists of open items that three or four years ago we would never have seen listed as open items. Is that right?

MR. TEDESCO: Yes, you are. You're also seeing -- I have experienced a much greater depth of review; depth and breadth, than we have seen in the past. The items after TMI have opened up.

MR. SIESS: I'm saying what we used to be concerned about, what I would call open controversial items where there was still considerable discussion going on between the staff and the applicant with questions as to how and whether they were going to get resolved. But now we are getting these long lists of things that are just not yet resolved, although there is every expectation they will be. And we have written letters in the past and said we expect it to be resolved in a manner satisfactory to the regulatory staff. And I suspect there were items never called to our attention because you didn't write SER's that way.

So I think we need to keep our perspective on what we're doing when we review and maybe write a letter or on 23 "open" issues. I don't think it's any departure.

MR. MARK: I think it would really meet your point and help us size this up a little better if one knew of those 23 whether there are 18 of them that are just not ready for discussion but there is no debate or any expectation that there will be a difference of opinion by the time the last documents are written.

MR. KERR: Bob, is there anybody in the review process who looks at the questions that go out to an applicant to see if some of them are patently ridiculous and ought to be squelched? Or does anybody who thinks he has a question in the review process get to send it out to an applicant?

MR. TEDESCO: We try to screen the questions as much as we are able to. It's an imperfect system but we do try to do that.

MR. KERR: But there is some effort?

MR. TEDESCO: Yes.

MR. KERR: Some of the questions I have been seeing lately have just influenced me to ask that, because at least from my limited perspective, I couldn't see why some of them were being asked.

MR. TEDESCO: We are trying to present to the committee on the reports that we see -- you will notice that we have cut Shoreham, realizing that there were too many open items to deal with it.

MR. SHEWMON: Would you try to go back to follow up on Mr. Siess's question and rank these with regard to those that you feel are controversial and how many of them still involve an active argument, or whatever words you want to use to paraphrase what he was saying?

MR. TEDESCO: All of them will involve discussions with the applicants. None of them will come out clean because there is a lot of subjective judgment how and when to demonstrate qualifications.

Fire protection is another. Fire barriers, sprinkler system and manual versus automatic; you're bound to get involved in discussions that are not necessarily of complete agreement initially.

Emergency preparedness -- we are involved with FEMA now. That adds another dimension to our whole review. The I&E bulletin, 79 bulletin, dealing with the consequences of failure, control room design, staffing and management. In many instances we are dealing with new organizations who have not even gone into the whole context of providing an adequate operating staff. They are very subjective issues that we're dealing with.

MR. SIESS: Mr. Chairman, one thing I guess that's bothering me is the emphasis on the open items in almost a legalistic fashion. I suspect that there are many items that are closed items already reviewed by the staff that we probably ought to devote as much attention to or more in an ACRS review as there are open items. Just because an item is closed it doesn't mean that the ACRS doesn't need to look at it.

So the emphasis on the open items I think is obscuring what we should do.

MR. KERR: I would say on the contrary, I want the items to be closed so we will look at all of them equally, because we don't look at the open items at all, in a sense, because we don't know what position the staff is going to take on them. It's that that causes me concern.

MR. SIESS: In the previous review -- I don't know what we're doing now because we haven't done one in a while. We used to focus a fair amount of attention on open items so we know what to say about them later.

MR. KERR: We don't know what the staff's position is going to be on them.

MR. BENDER: I think it makes a difference whether we're reviewing the plant at the construction of the operating stage. I think a construction license in general tended to focus on the things as to what was going to be designed into the plant, and we started out at the construction permit stage.

At the operating stage it seems to me that what's important is to be sure we have an understanding that the operational commitments of the licensee are sorted out. And I think we need a pattern. I believe that's what the real problem is right now; that each one of these guys is coming in and trying to establish his own precedent and we don't have any pattern of agreement to work from. And I think that's why we can't get them all through at one time.

Once an issue has been resolved for one guy, everybody knows that well, the staff has now established its position and they'll all conform to it, because what one agrees to, they all have to agree to.

But it's going to take a while to get these first two or three through the mill, and then I think the others will move much faster.

I don't know when the fire protection system will get sorted out. The lawyers will take care of that. But some of the others will get sorted out.

MR. MARK: What's your pleasure?

MR. OKRENT: My pleasure is lunch.

(Laughter.)

MR. SHEWMON: Why don't we go ahead and try to take them unless one of the subcommittee chairmen feels strongly that we shouldn't?

MR. PLESSET: I agree.

MR. MARK: I agree.

MR. OKRENT: Does that mean a three-day meeting instead of a four-day meeting?

MR. SIESS: That's what I want decided.

MR. TEDESCO: We'd also appreciate your comments.

MR. PLESSET: Let me ask about that. Is that the Allen's Creek containment question or more than that?

MR. TEDESCO: It's the first boiler for a CP and they're obligated to provide a hydrogen control system.

MR. PLESSET: So it's more than a Mark III containment.

MR. OKRENT: It's more than the question of loads and blowdowns.

MR. PLESSET: We're going to have to hear it sometimes because there are some others down the line.

MR. TEDESCO: We will be issuing an SER on Allen's Creek within the next few days, so that might be another item for your agenda.

MR. OKRENT: Has the Commission adopted a role in the NTCP business?

MR. TEDESCO: No. They have Revision 1 to represent the staff requirements.

MR. OKRENT: And it's like what you represented to the Commission?

MR. TEDESCO: Yes, sir.

MR. MARK: Well, I would suggest that we adjourn until five after 2:00.

(Whereupon, at 1:05 p.m. the meeting recessed for lunch, to reconvene at 2:05 p.m. the same day.)

MR. MARK: Let's reconvene.

MR. BENDER: As has been discussed previously at this meeting, the committee has been asked to evaluate whether it wants to write out a separate report on the Pilgrim Unit 2 Construction Permit, which the committee reported on sometime ago, in November of 1975 and October of 1977.

Since that time, of course, the TMI II accident occurred, and as a result of that accident the staff has generated some new requirements for construction permits, and they are outlined in NUREG-0718 Revision 1.

The subcommittee, consisting of Jerry Ray and myself, met with the applicant here in Washington on July 8, and at that time we heard a preview of what is being planned in response to the NUREG and post-TMI requirements. I think Jerry and I were favorably impressed by the plans for Pilgrim 2. On the off-chance that the committee might find it desirable to write a report, Dave Bessette, who was the Designated Federal Employee at the subcommittee meeting, prepared a draft statement which the committee might want to look at and decide whether it wants to act upon it.

The applicant and the staff are here to make a presentation. What I can say about the post-accident actions is that they have resulted in the Pilgrim 2 organization beefing its quality assurance capability somewhat. Its organizational plan as previously established was in conformance with what the staff was expecting already. It is instituting a PRA program. It has responded to a number of review requirements concerning TMI II types of accidents.

I think you will learn more by the presentation that has been planned by the staff and the applicant. So I suggest that we begin right away with that part of the presentation.

MS. ADENSAM: Gentlemen, and the staff and the applicant, as the chairman has indicated, met with the Pilgrim 2 Subcommittee on July 8 to discuss the TMI related requirements for the Pilgrim 2 construction permit application. During our introduction, the staff really outlined how we had been handling the near term construction permit and manufacturing license review of TMI related requirements. The subcommittee asked that we briefly address this review approach to the full committee.

On March 30th, 1981, Mr. Denton established a near term construction and manufacturing license dedicated review team to review the applicants' responses to TMI related requirements, published it for comment in the Federal Register as a proposed rule on March 23rd, 1981, along the organizational lines shown here in the first vugraph.

I would like to point out that the dedicated review team did include members from I&E, and the Emergency Preparedness staff. They remained in their respective branches, where they had access to total branch expertise, and the usual technical and management review of their work.

The team's first assignment was to develop detailed criteria for their review of applicants' responses to the requirements of the proposed rule. Most of these were presented to the applicant at a meeting on April 8, 1981.

The review schedule that is being used is shown here. The meetings in the third week are extremely intensive, and to work require a total cooperative commitment from the applicant as well as the staff. The applicant made available not only technical expertise, but individuals empowered to speak for the utility, and to sign written commitments to amend the PSAR.

We are happy to report today that there has been excellent cooperation on both sides, and few issues have had to be resolved after the SSER input was provided to the Division of Licensing.

We are currently into our third review with the dedicated review team. We are going to be issuing the Allens Creed SER supplement. Technical review is expected to be completed on the floating nuclear plant application by mid-July, with the supplement to be issued about a month later.

As you see from this vugraph, we can keep the dedicated team busy through August. If Black Fox does not insist on deferring their submittal until November 3, and submits earlier, we should be able to use those services up to October.

The dedicated review team is also committed to review issues related to their hearing efforts, otherwise it will probably be disbanded by the end of the year or earlier, if they are not needed for TMI reviews.

The subcommittee also asked that the staff address the following question on the applicant's organizational changes. One word is important and the change was the new organization that the staff wants to see.

Mr. Dominic Vassalo, Chief of the Licensee Qualifications, will address these questions.

MR. VASSALO: In response, the first question was, were there any important changes from the previous organizational plan? The only important change that we are aware of was in the area of nuclear operations. Previously they had one organization. They have now split that into Nuclear Operations, and Nuclear Operations Support, but that is primarily concerned with the operational activities. Other than that, there were no other organizational changes.

We reviewed this previously, and as a matter of fact, it was a contention in the hearing, and we had previously prepared testimony on it. The same vice president is in charge. The same project manager for the Pilgrim 2 Project. So we have not seen any drastic changes. I think they have had some staff changes, but the organization is appropriate for this activity.

Primarily because they have a nuclear oriented organization, that is the organization with a high level manager responsible solely for nuclear activities. As a matter of fact, Pilgrim did this early on, I guess, in 1975 and 1976, when we reviewed it. I might say that they were very early because now, in reviewing operating licenses and other construction permit applications, we look to that as being right type of organization.

In other words, that there be an organization which is responsible and dedicated to nuclear activities, and not diluted by having other responsibilities. So, therefore, we think this is the organization that is appropriate and will be capable of undertaking the design and construction activities.

MR. BENDER: Thank you.

Ms. Adensam, do you have anything further?

MS. ADENSAM: We had not planned any other presentation.

MR. BENDER: If not, why don't we proceed with the applicant's presentation.

MR. BUTLER: I am Robert Butler, the Pilgrim 2 Project Manager for Boston Edison. I have been in this capacity since projection inception in 1972. We are pleased to appear before ACRS today.

This slide shows that the subcommittee has requested that we briefly present information on the more important topics shown on the vugraph, which were contained in our PSAR amendment, and address the NTCP requirements, which has now been reviewed by the staff. We plan to follow this agenda.

May I have the next slide please?

A brief history and current status of the project is shown on this vugraph. The major points to focus on are the CP licensing process, which has been underway since December 1973, the advanced status of design and equipment deliveries, and the post-TMI licensing progress with a date in 1981.

The Atomic Safety and Licensing Board's partial initial decision of February of this year identified TMI requirements and emergency planning as the only remaining issues for Pilgrim 2. It is Boston Edison's desire to proceed expeditiously to now obtain ACRS concurrence, and proceed with final hearings.

Boston Edison and our contractors are currently evaluating the schedule and cost for Pilgrim 2. Key factors in the evaluation include the impact on schedule and cost of changes resulting from the TMI requirements and recent industry experience.

The major uncertainty in the schedule for issuance of a construction permit is the lack of a final rule on TMI regulated requirements, and the uncertain effects this could have on the duration of the Pilgrim 2 hearings, and subsequent adjudicatory actions.

However, we anticipate the earlier schedule for issuance of a CP is the spring of 1982. Beyond that, typically nuclear construction durations are running between six and eight years.

I have with me today members of the Boston Edison nuclear organization, as well as nuclear engineering staff, as well as our contractors, Bechtel and Combustion Engineering.

On organization, the subcommittee has requested that this presentation be brief since our current organization, as Mr. Vassalo said, has essentially been in place since 1975, and has been presented to ACRS previously.

Boston Edison has established a separate nuclear organization reporting at the executive level to the Vice President Nuclear. The organization has responsibility for the company's nuclear activities, which include operation of Pilgrim 1 and the Pilgrim 2 Project.

Within the organization, I as the Pilgrim 2 Project manager have the full time management responsibility for Pilgrim 2 design, licensing, procurement, and construction. As seen on the vugraph, other Boston Edison organizational elements, QA, Nuclear Engineering, and Nuclear Operations Support, report independently to the Vice President and provide services to both Pilgrim 1 and Pilgrim 2. Nuclear Operations is responsible now for Pilgrim 1 operations, and later for Pilgrim 2 operations.

Prior to the start of construction, the nuclear organization has maintained an in-house staff equivalent of about 20 full-time engineers and managers on Pilgrim 2 work. Consultants have been and continue to be utilized to supplement the Boston Edison staff.

Currently the nuclear organization in support of Pilgrim 1 and 2 has about 100 personnel on the management and technical staff, having an average level of nuclear experience of about eight years. This does not include 41 currently approved help requisitions, indicating growth in this organization.

MR. KERR: What fraction of the nuclear experience came from employment with Boston Edison?

MR. BUTLER: What percentage?

MR. KERR: Yes, or what fraction, roughly, half of it, two-thirds, 20 percentD?

MR. BUTLER: I would guess between 25 and 50 percent.

MR. KERR: Thank you.

MR. BUTLER: During the construction , the nuclear organization staff on Pilgrim 2 is estimated to increase fromm approximately 39 to 244. The vugraph indicates the breakdown of this complement.

The next slide, please.

The division of responsibility of Boston Edison, Bechtel, CE, and GE is defined in contractual and procedural documents. Boston Edison is responsible for the overall design and construction operations of Pilgrim 2, including conformance to regulatory requirements.

Boston Edison's nuclear organizational meets these responsibilities by providing management oversight to principal contractor activities, obtaining Federal licenses and permits, approving basic design criteria, releasing selected design documents, and authorizing the expenditure of funds.

Combustion Engineering is responsible for design and fabrication of the nuclear steam supply system. General

Electric is responsible for the design and fabrication of the turbine/generator. Fitting between these, Bechtel is responsible for the balance of plant, including engineering, procurement, construction, and QA/QC for their scope.

Additionally, Bechtel is responsible for design interface control among Bechtel, CE, and GE, and between Bechtel and its subcontractors.

Boston Edison manages and evaluates the performance of its contractors by review and approval of design criteria, design and procurement documents prior to purchase or construction. All these activities are performed under written procedural control and periodic audit.

MR. KERR: Excuse me, but is there anything that has changed relative to that slide as a result of TMI-II?

MR. BUTLER: Not directly as a result of TMI-II.

MR. KERR: I am not looking for changes if they are not there.

MR. BUTLER: No, there are none.

The Quality Assurance Manager reports directly to the Vice President-Nuclear, and with his staff is responsible for establishing and maintaining adequate QA controls. Construction oversight of contractor performance will provided by the Pilgrim 2 Project staff and the construction QA group.

The Project Construction Manager will report to the Pilgrim 2 Project Manager. Boston Edison QA provides construction oversight to the construction QA group which is responsible for monitoring the QA aspects of sight construction. The construction QA group will interact directly with the principal contractors' sight organization and with Boston Edison home office QA organization.

Boston Edison QA staffing on Pilgrim 2 is estimated to increase from four at the start of construction to a peak of nine. Bechtel on-sight planned staffing levels are 56 in the QC group and six in the QA group at the peak of construction. These estimated staffing levels will be continuously reviewed to insure adequate project coverage.

Procedures for construction management and control will be approved for use prior to the start of each construction activity.

Since project inception, operations oriented personnel have been utilized to review design. Nuclear Operations Support provides this function.

An important operations aspect is the feedback of industry operating experience which was covered in some detail at the subcommittee meeting.

Boston Edison's single nuclear organization will greatly facilitate the transition from construction to Pilgrim 2 operation by maintaining continuity of personnel in the development of operating procedures, review of test results, training, and technical support.

Engineering and management personnel involved in the design and construction phases will be encouraged to transfer to available positions in operations or in support of operations.

Plant pre-operation and start-up testing will be accomplished by an integrated start-up organization of Edison, Bechtel and CE personnel, to be managed by Boston Edison.

The Boston Edison Executive Office exercises top level management oversight by approval of operating budgets, capital authorizations, periodic project status reviews, including quarterly QA status reports, and reports of the quality assurance review committee, a committee of QA project operations and engineering personnel to determine the adequacy of the QA program and its implementation.

Further executive oversight will include setting policy for future activities, and participating in Pilgrim 2 executive review meetings on approximately a semi-annual basis. Executive review meetings are attended also by top level management of the principal contractors, enabling the executives of all three companies to be periodically informed of project planning, status, and problems.

This concludes my presentation on organization and management. Are there any questions?

MR. WARD: Would you show that again? Where did you say the operations review committee fits in?

MR. BUTLER: The quality assurance review committee?

MR. WARD: Okay, the quality assurance review committee.

MR. BUTLER: The quality assurance review committee is a committee chaired by the Quality Assurance Manager, but with representatives of the QA organizations reporting to him, representatives of the project reporting to me, representatives of engineering reporting to the engineering manager, and representatives of Operations Support. So it is a committee which has multi-discipline and organizational orientation.

MR. WARD: It reports to the Vice President?

MR. BUTLER: That reports to actually to the Vice President and to myself. They provide their reports for us as management advisory reports. We are not members of that committee.

MR. BENDER: Are there further questions?

(No response.)

MR. BENDER: Why don't we go to Mr. Ashkar. Thank you, Mr. Butler.

MR. BUTLER: The next speaker is Mr. Ashkar of the

Nuclear Engineering Department, who is responsible for PRA. Mr. Ashkar will also cover briefly ATWS.

MR. ASHKAR: My name is Jim Ashkar, and I am with the Nuclear Engineering Department. I am responsible for the PRA Program.

Boston Edison will perform a plant/site specific probabilistic risk assessment, and submit a PRA report to the NRC delineating the results within two years after the CP. The PRA report will present the site/plant specific risks in terms of probability of frequency curves for various health effects for both the base plant design submitted in the PSAR, and for the revised design as modified by the PRA Program.

Boston Edison will utilize intermediate products from the PRA to allow early identification and investigation of potential improvements in order to facilitate coordination with construction schedules.

From the following slides, I will cover the program objectives, organization, program flow, the program elements, and the PRA Report.

MR. MOELLER: Can you tell us who is doing this, are you doing it in-house, or a certain contractor?

MR. ASHKAR: I have a slide for that area.

The objectives of our PRA Program are to seek design improvements in systems affecting the reliability of accomplishing core and containment heat removal which, (1) contribute to significant risk reduction, (2) represent practical applications of demonstrated engineering technology, and (3) do not excessively impact on the plant construction, start-up schedule, or upon plant cost.

MR. KERR: Mr. Ashkar, what are you going to reduce? What significant risk reduction compared to what?

MR. ASHKAR: What we intend to do is to look at our base plant design as it is described in the PSAR as a basis, and look at improvements that we may incorporate compared to our base plant design.

MR. KERR: How do you decide how much improvement you need?

MR. ASHKAR: What we will be using are other reports, industry standards, and guidelines that are available to us to facilitate our judgment in making those improvements.

MR. KERR: I don't know of any industry standard that tells you how much risk is acceptable.

MR. ASHKAR: That's right. I meant other reports that have been made, other probabilistic risk assessments to determine how risk is for operating plants currently, and other guidelines, such as the ACRS Safety Code. AIF has similar documents, and I think EPRI has similar documents also, although they are far from final.

MR. KERR: Thank you.

MR. ASHKAR: Secondly, we would seek to quantify the merit of design improvements that we would incorporate. The PRA Program, like other project programs, will be managed by Boston Edison Company. PRA activities will be performed and reviewed by engineers who are experienced and highly qualified in risk assessment methodology. Boston Edison, Bechtel, and CE resources will be supplemented by consulting organizations in this regard.

Design alternatives will be developed by Engineers in the qualified design groups presently responsible for design, primarily at Bechtel and Combustion Engineering.

High level independent review of the entire PRA program will be performed by separate senior level oversight groups. The design decisions will be made by the current project management team, who will have the added benefit of risk quantification provided by the PRA Program.

The next slide summarizes major activities of our PRA Program and show the interrelationships. Results of other PRA programs will be used to identify issues which have the potential to compromise the expected reliability of core and containment heat removal systems. We will develop practical resolutions to those issues.

For issue resolutions requiring design or other modifications, alternative modifications will be developed, including alternate core and containment heat removal system designs.

We will use the results of generic safety studies and PRA studies of other plants in conjunction with operating experience feedback to establish a revised design.

The plant/site risk assessment will be performed on the revised design, and will include an expression of risk reduction over the base plant design as documented currently in the PSAR.

MR. KERR: When you refer to the base plant, I think you say it is the one described in your PSAR.

MR. ASHKAR: That is right.

MR. KERR: So it was a plant that was designed X years ago, where X is equal to what?

MR. ASHKAR: Up to 1974, 1975.

MR. KERR: And you have concluded that the risk associated with the operation of that plant is too great, and that it needs to be reduced?

MR. ASHKAR: I don't know that we have come to that conclusion since we have not conducted a risk study on the base plant. We think that by doing a risk evaluation, we may be able to identify areas for improvement.

MR. KERR: I guess we will learn as your program develops how you decide what improvements are needed.

MR. ASHKAR: The next slide shows our PRA Program plan and schedule, and the establishment of our organization responsibilities and management control. The schedule will reflect use of intermediate PRA results coordinated with the construction schedule. This will insure that the beneficial changes resulting from intermediate PRA products can be implemented early in construction, without an excessive impact on project cost.

In my next slide I will address the key elements of our plant/site risk assessment, which will feed back as intermediate products in the design process. In general, the approach taken will reflect contemporary methodologies.

In the preliminary analysis, we will assure completeness of scope of initiating events by developing a master logic tree to identify the range and nature of inititing events, which could lead to unacceptable consequences. We will also select the most significant primary and support systems for further analysis at this time.

Next, in developing plant event sequences, we will consider system responses and interactions during allowable plant operating modes. For example, failure during cold shutdown operation will be considered an initiating event of our study. We will establish a plant specific data base covering component failure rates, test and maintenance data, and the frequency of initiating events.

Because we will not have detailed designs available, we will necessarily need to make design assumptions, and select appropriate corresponding component failure characteristic data. We will subsequently verify agreement between the final design and these design assumptions. These verifications will insure that the assumptions in developing the data base and used in the analysis are consistent with the final design.

This documentation step will also contribute to configuration and control, and in turn provide reliable information for developing surveillance and the maintenance program.

In the analysis of external caused failure, we will estimate the frequency and consequences of significant failures, which are externally initiated by earthquakes, fires, and the like.

In the analysis of system failures, we will compare key system fault trees and event trees, which will reveal intra-system and inter-system failure mechanisms. This will not be limited to primary safety systems, but will include support systems as well.

Based on operating experience feedback and other PRA results, we know that problems in the support system may also result in common cause failures. These potential problems would be revealed through key system fault trees and event trees, and related analysis. Problems identified would be intermediate products fed back into the design process.

We will also assess how human interaction and the environmental conditions affect the availability of systems. This will be done utilizing the best available technologies consistent with contemporary PRA methods. Environmental effects will include radiation, and other adverse environmental conditions, especially for those components that must be operable for long period of time, and maybe inaccessible for maintenance under accident conditions.

The uncertainty in input data used in quantifying plant event sequences and other analyses will be carried through the analyses and expressed as part of the final results. By establishing the basis for the uncertainty, we will determine how the uncertainty affects the final results and document this in the PRA report.

MR. KERR: I am trying to understand how these generalizations, which sound reasonable, are going to be applied to your specific plant. For example, are you going to include the containment system in this study?

MR. ASHKAR: What we hope to do, as I mentioned earlier, is to review other PRA reports that are particularly applicable to our plant type, and identify systems and assumptions procedures that we feel those results would indicate may be beneficial.

MR. KERR: Let me take you through a couple of examples. Suppose you look at your containment system, and decide that by doing something or other to it you can reduce the risk by a certain amount. You have to make a decision on the containment, I assume, fairly early on if it is going to be built into the system. On the other hand, maybe you can do something to some other part of the system which is pretty far down the line, and would reduce the risk a lot more.

How do you decide whether to make the change in the containment, which you have now looked at, or wait and look at something else, or are you going to improve everything?

MR. ASHKAR: In each case, when we are considering an alternative that may reduce risk, we will also develop cost information.

MR. KERR: But you have to do some of the studies first. It seems to me the containment one has to be done and completed fairly soon, otherwise you can't start construction.

MR. ASHKAR: Right.

MR. KERR: Indeed, I think you might start construction before you finish the total study, the total PRA study.

MR. ASHKAR: That is correct.

MR. KERR: How are you going to know whether it would be better to improve the containment or to improve something else? What sort of decision process are you going to use? I am trying to understand the decision process that you are going to use.

MR. ASHKAR: In addition to the decision process, we are scheduled to do the PRA activities within the intended construction schedule. The way we have at this point planned to conduct the PRA would be to first utilize the benefits of other studies that are available immediately.

Secondly, it would be to go through and identify outliers, which we are going to discuss, very early in our program, and get through the event sequence analysis as early as we can to try and identify the issues we are talking about.

We think that in the first few months of the study we can get to that phase, and give substantial clarity to the rest of the process.

MR. KERR: I don't understand how you are going to make your decision, but perhaps I will understand better after you get to the end of the process.

MR. ASHKAR: To continue, outliers indentified in the plant and containment event sequence analysis phases will be intermediate products fed back into the design process. When outliers have been dispositioned, we will develop a list of critical items which delineate the relative importance of components to reliability of core and containment cooling.

The results of the PRA study will contribute --

MR. KERR: How does one identify an outlier?

MR. ASHKAR: What we have tried to do at this point would be to quantify event sequence frequencies, and compare those with other reports that are expected for those frequencies to see how some event sequences are particularly unusual.

The results of the PRA study will contribute to defining importance to safety as required by 10 CFR 50 Appendix B, criteria 2. At this point, those multiple failure sequences which should be considered in the control room design review will also be identified.

MR. WARD: Jim, do you expect that that critical items list is going to correlate with what you call your Q list? Is it going to be the same as the Q list, do you think?

MR. ASHKAR: We expect that it will be a larger list, and better guidance not only in safety systems, but in other typical non-safety systems.

MR. WARD: So you are looking for -- When you say, identify outliers, for example, you are looking at things not only the safety items, but reliability items.

MR. ASHKAR: The balance of plant, things that would contribute to initiating events.

MR. WARD: But just from the safety standpoint, or any economic standpoint?

MR. ASHKAR: In the scope of the study, we are only going to be looking at safety issues, event sequences that would lead to unacceptable off-site doses.

MR. RAY: I think that it would be clearer for us if you defined what an outlier is. What do you consider an outlier?

MR. ASHKAR: What I would consider an outlier is the frequency for an event sequence, frequency of occurrence which is unusually high compared to those we expect from other studies, such as WASH-1400, Indian Point, etc. I think we are beginning to see some level of convergence with respect to frequencies of some of these events, something on the order of a half order of magnitude, or an order of magnitude different would be something that would raise a red flag to me, anyway.

MR. RAY: Something that is so significant and important, has such widespread effect that it warrants early attention, and possibly action, and that kind of thing?

MR. ASHKAR: Yes.

MR. RAY: The criticality you are talking about is not so much, the way I interpret your remarks, and I could be corrected if I am wrong, not so much with respect to operation of a plant, so much as reducing the risk that is inherently built into the plant.

MR. ASHKAR: Correct.

The next slide please.

The PRA report will be prepared documenting the PRA activities, including methods used, important assumptions to be verified, data sources, typical analyses, and treatment of uncertainty. It will include results such as expected core melt frequencies, containment failure frequencies, both of which we used to identify outliers, identification of those dominant sequences contributing to risk, and risk curves for both the base plant design, that is the PSAR design, and the revised design that we will ultimately end up with.

The relative contribution of individual design improvements of core and containment heat removal reliability will also be presented in the report at that time.

Our report will describe applications of the PRA results which are planned in subsequent phases of the project, such as development of a preventative maintenance program, development of surveillance testing programs, and the development of emergency procedures, and operator training.

MR. WARD: Do you know about what this program is going to cost?

MR. ASHKAR: I know what others have cost, somewhere between one and two million dollars, closer to two I would say.

MR. SHEWMON: You have said earlier, and I missed it, you were asked who was going to conduct this, and I missed the answer.

MR. ASHKAR: What we have already initiated is a project team made up of representatives from our prime contractors, Bechtel and Combustion Engineering. Also we will solicit the aid of a risk contractor consultant to provide the preparation of the detailed plans and procedures for control, and will also do major portions of the reliability and quantification phases. Boston Edison will also be involved in program guidance, direction, and control, and also will provide operations review input. Our approach will be a team approach, utilizing a variety of contractors.

MR. SHEWMON: Could you tell me how or whether your system might respond to or include possibilities of in-house sabotage or fire?

MR. ASHKAR: We are planning to utilize the available technologies for including fire analysis. We have not well developed this part, or our approach to sabotage with regard to PRA. That is the best answer I can give you at this point.

MR. SHEWMON: Are there well developed technologies that you gone into for treating fires?

MR. BENDER: There are methods. There are areas that are a threat, and how they might involve safety. I don't know that it has been done probabilistically, but I don't see any reason why it couldn't.

MR. ASHKAR: There are some techniques that we have investigated, and they are not well developed or sophisticated, but they are approaches. We will try to get a handle on sabotage.

MR. SHEWMON: I guess in both of those you are likely to end up with physical distribution of flames, and I don't know whether that comes into PRA at this point or not. That is part of my question.

MR. BENDER: I think it would be unfair to ask Mr. Ashkar to be too explicit on how he is going to do this thing. The plan is not all that well developed.

MR. ASHKAR: I have a brief statement that I would like to make with regard to ATWS. Even though this is not a particularly post-TMI issue, I would like to bring you up to date on where we stand and our intention to include treatment in the PRA program.

Pilgrim 2 probabilistic risk assessment program will include anticipated transients without scram events. Boston Edison and its principal contractors have been actively participating in various activities related to ATWS.

Although ATWS was not a design basis event for Pilgrim 2, the Pilgrim 2 has been included among the designs considered in the generic ATWS analysis performed by Combustion Engineering, and on behalf of NSSS owners groups. These analyses demonstrated the capability of the existing program to designs to mitigate the consequences of an ATWS event.

Boston Edison is involved in, and closely follows the rulemaking process for the resolution of the ATWS issue. The Pilgrim 2 does not preclude future incorporation of any of the ATWS plant modifications which have been addressed thus far in the various proposed rules for ATWS resolution.

This concludes my presentation.

MR. KERR: Have your efforts or considerations had any influence on the pressure relieving capability of your primary system?

MR. ASHKAR: The generic studies that have been done by Combustion, which we reviewed, would indicate that some advantage would be provided by the addition of relieving capability to the system. We have not incorporated it in our PRA program as we initiated it.

MR. KERR: You will incorporate into your plant additional pressure relieving capability?

MR. ASHKAR: We have not conducted our PRA program and made conclusions with regard to that yet.

MR. BENDER: As I understood from our previous conversation, the plant design does not preclude the use of additional relief valves. You have not made a decision yet on whether they are needed, and you are waiting for the PRA study to do that.

MR. ASHKAR: That is right.

MR. KERR: I think you would make a considerable contribution to the industry, because the ACRS tried to do that a long time ago and failed. So I am looking eagerly forward to your results.

MR. BENDER: Can we move to the next subject?

MR. BUTLER: The next speaker is Mr. Ron Jagels, the Bechtel Project Engineer, who will cover the topics of hydrogen control and degraded core rulemaking activities.

MR. JAGELS: I am Ron Jagels, the Bechtel Project Engineering for Pilgrim 2.

This afternoon, I will discuss the agenda items dealing with hydrogen control, and degraded core rulemaking.

The Pilgrim 2 containment is a pre-stressed, closed-tension, steel-lined, concrete structure with a free volume of approximately 2.5 million cubic feet. The Pilgrim 2 design will satisfy the requirements for degraded core hydrogen prescribed in NUREG-0718, and summarized on this slide.

First, a hydrogen control system will be provided which is capable of handling hydrogen generated by the equivalent of 100 percent fuel clad metal water reaction. The containment and associated systems will provide reasonable assurance that uniformly distributed hydrogen concentration will not exceed 10 percent.

The existing containment design exceeds the requirement of a minimum containment design pressure of 45 pounds gauge.

Pilgrim has performed analyses on the containment pressure effects due to hydrogen. Results of these analyses were presented to Dr. Okrent's subcommittee in February of this year, and we have recently revised and updated these analyses to include the effects of the distributed ignition system.

The more analyses used flammability data which account for steam dilution effects, and also take credit for containment heat sinks, such as structural steel, and the steel liner. The analyses also assume that one containment spray train was available for containment heat removal.

The results of the analyses based on these previous assumptions show that for an initial hydrogen concentration of 10 percent, the peak containment pressure is less than 69 pounds, the containment test pressure, and also that the containment liner temperature will not exceed the design value.

Pilgrim 2 has also committed in PSAR Amendment 43 to incorporate the results of industry and NRC research and testing into the final selection and design of a hydrogen control system.

Additional analyses will also be conducted which will include a consideration of various accident scenarios that have different steam and hydrogen release rates.

Within two years after issuance of the construction permit, Pilgrim will submit to the NRC for review the design details of the hydrogen control system that has been selected.

In summary, Pilgrim 2 has a large, dry containment that has considerable margin to withstand accidents that generate large quantities of hydrogen. The preliminary analyses show that the containment pressure from the degraded core hydrogen can be controlled to less than the test pressure of 69 pounds.

Are there any questions on hydrogen control?

(No response.)

MR. BENDER: Why don't you go on to the next agenda item.

MR. JAGELS: The next agenda item is the degraded core rulemaking activities.

The Pilgrim 2 has been monitoring and participating in the degraded core rulemaking activities. We believe that these are generic rulemakings and should not be litigated in individual proceedings. At the conclusion of the rulemaking activities, the Pilgrim 2 project will be subject and will comply with the outcome of the rulemaking.

The resolution of these rulemakings is not imminent, and it is likely that it will be several years before they are resolved. Pilgrim 2 has made commitments, and we have planned activities which we feel will significantly reduce the potential effect of these rulemakings on the project.

First, as I have just described, we have committed to install a hydrogen control system capable of handling 100 percent of the fuel clad metal water reaction. This requirement is more stringent than the 75 percent metal water reaction included in the interim rule on hydrogen.

We have committed to the installation of a three-foot diameter spare containment penetration which could be used for a containment filter vent system, if the requirement to install such a vent is the outcome of the rulemaking.

A plant specific risk assessment will be performed. This objectives of this assessment include seeking design improvements in both the core and containment heat removal systems.

In addition, the project will continue to monitor and participate in the on-going rulemaking activities.

The effect of these commitments, we feel, will increase the likelihood that the Pilgrim 2 design can accommodate the outcome of the degraded core rulemaking.

MR. KERR: Mr. Jagels, when you refer to a 100 percent metal water reaction, is the metal to which you refer just the cladding?

MR. JAGELS: That is correct.

MR. OKRENT: As a point of information, the containment as it is currently designed can take what negative pressure?

MR. JAGELS: I do not recall that from my memory, Dr. Okrent. I have the figure with me in the PSAR.

MR. BENDER: Can you get somewhere within a few pounds of it?

MR. OKRENT: We will take a full vacuum.

MR. BENDER: Will take a full vacuum, or a half.

MR. OKRENT: We will take 14.7 p.s.i.

MR. JAGELS: I prefer checking it, if I might furnish it later.

MR. BENDER: Why don't you do that, and we will move on.

MR. BUTLER: The next speaker is Mr. George McHugh, Boston Edison Project Engineer, who will take the topic of the control room design approach.

MR. McHUGH: My name is George McHugh, Boston Edison Project Engineer responsible for balance of plant activities, which include the development of the control design.

Our approach to control room design has changed substantially since the accident at Three Mile Island. I intend to advise you today of our current status, explain our pre-TMI work, and our current approach as outlined in Amendment 43.

First, our current status. We have not yet fabricated or purchased our main control boards. In fact, in Amendment 43 we have committed to provide control board layouts for NRC staff review prior to fabrication or revision from the board.

We are currently formulating the details of our program, which will involve state-of-the-art analytical techniques, the application of accepted human factor principles, and the use of mock-ups to verify the effectiveness of the layouts, using operations oriented analyses.

My next slide outlines the work we did before the accident at Three Mile Island. We did develop a preliminary control room design, including preliminary control board layouts, well prior to the TMI II accident.

The design was developed and reviewed by personnel who were involved in our then state-of-the-art safety sequence analysis, which identified information and devices required in the main control room to respond to the transient and accident conditions analyzed.

The results of the safety sequence analysis was documented on Safety Sequence Diagrams, SSDs. Even though it was a specific objective of the SSD program to identify control room inventory, the analysis did impact the design.

Starting with Bechtel-CE proposed control board layouts, a preliminary allocation for function, indications, enunciations, and controls was established based on a review of the operation of plant systems.

Using P and ID, preliminary system descriptions, and control logic diagrams, this review was performed with operations personnel who had actual PWR operating experience, and we used human factors guidelines available at that time, 1976-1977 such as Dr. Allen Swain's Zion work in 1975, and the EPRI-Lockheed study in 1976. Although there was no direct use of a human factors consultant, we did use available guidelines.

This resulted in a preliminary control board layout based on an operating cost to be formulated during the design process.

Our pre-TMI effort provided valuable insight into the development of our post-TMI approach. In other words, we were able to focus more sharply on TMI experience. Since TMI, we have collected enough information to recognize the need for an improved approach, and are formulating the details of our program.

As outlined in Amendment 43, our program calls for an update of the safety sequence analysis to reflect changes in design since 1973, advances in design development, use of the latest SSG methodology, and analysis of the more comprehensive scope of iniating events.

This analysis will establish the inventory of information required in the control room to respond to transients and accident conditions analyzed. As stated earlier today, our PRA Program will provide a master logic tree to establish the scope of events requiring analysis. The PRA will also be used to determine the multiple failures that must be considered in the control room information format for inventory.

Even more important than the quantity of information required is the manner, configuration and format in which it is presented. In that regard, our program calls for developing a new preliminary design based on the direct input of human factors principles and data, which I will discuss in a minute in more detail on the next slide.

Once the new preliminary design is developed, we will conduct an operations oriented analysis to determine operational efficiency.

The design process which results will of necessity be iterative, and will involve a team approach in program development analysis, revised design development and review, involving individuals experienced in operations, systems analysis, human factors engineering, architectural engineering, and control room design.

The results of our program will be submitted to the NRC staff prior to fabrication of the main control boards, and the plant specific simulator.

My final slide shows the program flow I have just described. I have highlighted on the left the human factors input, which we intend to obtain from expert consultants, and have also separated analytical tasks from the design flow shown on the right.

Shown on the left is the direct human factors input in three phases. First, in the initial program establishment, including early control and design, and planning activities. Human factors input to the program establishment will improve development of two types of guidelines:

(a) guidelines to perform activities in subsequent phases, for example, functional allocation analysis, analysis of operator tasks versus capability, and how to determine the information required for assigned tasks;

(b) design guidelines which will include development of color, lighting and acoustical environmental design criteria, criteria for information display, hardwires versus CRT displays, the board profile establishment, and a preliminary control room configuration plan.

The next phase involves the safety sequence analysis with PRA input, and the human factors guidance functional analysis.

Human factors input will also be obtained in the identification, selection, and arrangment of the indications, enunciations, and control on the main control boards to accomplish the functions allocated. This will accomplished through a link analysis at the major subsystem level to allocate areas in the overall layout, without addressing hardware.

Hardware selection will be made considering the human machine interface. Enunciation philosophy will define the use of hardwires versus CRT displays. The resulting new preliminary design will be reviewed against the configuration plan resulting in preliminary control board layout.

In preparation for the next phase, human factors criteria for operational procedure. will be developed. Draft procedures will be written, and control room work stations will be designed.

Finally, operational sequence diagrams will be prepared for tasks to be analyzed in phase three.

The third phase will involve details reviews of specific groupings of devices as the operations oriented analysis proceeds using full-scale mock-ups. This analysis will be performed by individuals experienced in systems analysis, operations, human factors engineering, and control room designs.

The output will be a finalized set of main control board layouts and a report, both of which will be submitted to the NRC staff for review. Following staff review, we will fabricate both the main control boards and the plant specific simulator.

It is our overall intent to obtain a

state-of-the-art control room which will assure that ability of the control room operations personnel to prevent anticipated transients from developing into accidents, and to cope with accidents should they occur.

Are there any questions?

MR. BENDER: Are there any questions?

MR. WARD: George, you said that in the part of the analysis you are going to have operations people take part in that. Are those people who would be in the Pilgrim 1 organization, or have experience there? Who are those people?

MR. McHUGH: As Mr. Butler said, we have the Nuclear Operations organization responsible for the operation of Pilgrim 1, and there is a Nuclear Operations Support Department, which supports both Pilgrim 1 and Pilgrim 2, and includes seasoned operators with the PWRs and the BWRs experience. We will be drawing on them, and additional consultants from other utilities as necessary to meet our judgment of what is adequate experience base.

MR. BENDER: Are there other questions.

(No response.)

MR. BENDER: Thank you, Mr. McHugh.

Let's go to the last two items.

MR. BUTLER: The last speaker we have is Mr. David Bryant, the Boston Edison Project Engineer, who will cover the topics on inadequate core cooling instrumentation, and safety/relief valve testing.

MR. BRYANT: Ladies, and gentlemen, as Mr. Butler said, I am David Bryant, and I am the Project Engineer at Boston Edison for the NSSS.

With respect to the instrumentation to detect inadequate core cooling, the commitment we have made has two key elements.

First, we will install a primary coolant saturation meter. This type of device is being installed in operating plants and provides a continuous read-out of the margins of saturation conditions.

Second, we are going to perform a study to determine what additional instrumentations will be included in the Pilgrim 2 to detect inadequate core cooling. This type will include both reactor water level instruments and core exit thermal-couples. We have committed not to preclude either of these devices from the design.

We will submit the results of our study to NRC after prototype testing, and before we procure the hardware for Pilgrim 2.

I might add that the operating plants and the near-term operating license applicants are installing such systems, and I expect that we will have feedback from actual operating experience from these systems before we have to install the hardware in Pilgrim 2.

This is all I intended to say on this subject. Are there any questions?

MR. BENDER: Go ahead.

MR. BRYANT: The testing of primary safety valves and relief valves, and design of the piping and support.

Go to the next slide.

The next slides summarizes the commitments made by Pilgrim 2 in the PSAR. First Pilgrim 2 will implement the results of the testing being performed on behalf of the industry by the Electric Power Research Institute, EPRI.

Just for clarity, when I say relief valves, I mean power operated relief valves, PORVs.

Also Pilgrim 2 will design the discharge piping and the supports for these valves for the loads from the design basis transients, and accidents.

These commitments meet the NRC's requirements in NUREG-0718.

The next slide please.

This slide summarizes the program being run by EPRI. First, it is designed to comply with NUREG-0737, which is the task action plan for the operating plants and near-term operating license applicants. The program involves testing 10 PORVs and nine safety valves. It happens that the specific PORV and safety valve being supplied by Combustion for Pilgrim 2 are included in the test program.

The program is designed to envelope the operating conditions on all the plants that are represented by the program. Testing is currently underway and the final output of the program is scheduled for July of 1982.

The block valve portion of the program is still under development. Testing has no begun on block valves.

The next slide.

This is an EPRI slide that summarizes the major outputs from the valve test program. First, there will be reports on each of the valve tests which are being done at three facilities. The tests will demonstrate the valve's performance under the design basis conditions.

Second, there will be a report to document the selection of which valve were tested, and adjusted by the applicability of the test results to all the various plants.

Third, reports will be provided to substantiate the test conditions and to show how these conditions envelope the various plants.

These reports will also deal with the effects of as-built piping. In the case of Pilgrim 2, of course, our piping exists only on paper, so that we can redesign our piping, if necessary, to utilize the conclusions from the tests.

Fourth, there will be a report documenting the recommended computer programs to calculate the hydro-dynamic loads used for design of piping and supports under the design basis conditions. This code is expected to be suitable for use on new projects such as Pilgrim 2.

Thus, the EPRI valve test program is expected to provide us all that we need to meet our commitments in the PSAR on valve testing, except that it will not provide the specific calculations of the piping loads. We will do those calculations ourselves on the project, utilizing the outputs from the EPRI program, and utilizing information on our own plant transients.

Furthermore, all the EPRI effort is currently scheduled to be completed in late 1981 or early 1982, which is well before the needs of Pilgrim 2 for this information.

The last slide please.

In summary, the EPRI program is currently underway on power operated relief valves and on safety valves. The Pilgrim 2 valves are included in the EPRI program. Pilgrim 2 will utilize the results of the EPRI program as they become available, which will allow us to meet our commitments.

The block valve program is currently under development.

Ladies, and gentlemen, that concludes my presentation. Are there any questions?

MR. BENDER: Are there questions?

(No response.)

MR. BENDER: Thank you, Mr. Bryant.

MR. BUTLER: This concludes the applicant's presentation. We do have an answer to Dr. Okrent's question, and Ron Jagels will give it.

MR. JAGELS: Dr. Okrent's question was the design pressure of the containment on the other end of the spectrum. The answer is, the containment is designed for a minus 3 p.s.ig., or 11.7 p.s.i.a. on the low end.

MR. BUTLER: I would like to thank the committee for your attention to our presentation, and for interrupting your unfinished business to let us in pretty much on schedule.

MR. BENDER: Does the staff have anything further comments to make?

MS. ADENSAM: No, we haven't.

MR. BENDER: Mr. Chairman, we have a couple of alternatives that we can follow here. At one time, we thought that we might write a letter on this subject, but it doesn't seem that the content of the presentation indicates any great need for a letter specifically on this plant.

I suspect that we have heard the picture of what will be done on most construction permits, and how the staff will review. They have a pretty effective program. I would suggest that we reflect in the minutes of the committee's the further story on Pilgrim 2.

Does that sound adequate for a construction permit purposes, taking into account the TMI 2 considerations. We could notify Mr. Dircks in some way that this is what has been done, and not prescribe a formal letter at this time.

MR. MARK: I think that sounds appropriate for the case. What we have learned is that the Pilgrim 2 group has taken a very forceful look at the TMI 2 action plans and consequences, and adjusting their thoughts to them, and is committed meet, I guess, all of the major technical requirements of the staff. Just how they will meet them, they have time to go into that.

I think what you suggest is right. The minutes should reflect that we have seen. The old CP letter of the committee, those things never depreciate. The comment is that the new requirements are being followed with good intention.

Are there any differences, or opinions from other committee members?

(No response.)

MR. MARK: If not, I suggest that we stop the recording at this point of the record.

I would like to thank the Boston Edison people for putting this together for us.

(Whereupon, at 5:40 p.m., recording of the meeting was discontinued.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

____

in the matter of: ACRS/255th General Meeting

Date of Proceeding: July 10, 1981

Docket Number: ____________________

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcri thereof for the file of the Commission.

ANN RILEY

Official Reporter (Typed)

Ann Riley

Official Reporter (Signature)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

____

in the matter of: ACRS/255th General Meeting

Date of Proceeding: July 10, 1981

Docket Number: ____________________

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Patricia A. Minson

Official Reporter (Typed)

Patricia A. Minson

Official Reporter (Signature)

GPU NUCLEAR

MAJOR ELEMENTS

- FULL TIME ORGANIZATION DEDICATED SOLELY TO NUCLEAR GENERATION
- INCREASED ON-SITE TECHNICAL AND MANAGEMENT RESOURCES
- STRONG CENTRAL TECHNICAL CONTROL
- FULL TIME ON-SITE MANAGEMENT FOR PLANT OPERATION AND MAINTENANCE - - WITH SUPPORT IN ADMINISTRATION, ENGINEERING, RADIATION PROTECTION, AND OTHER AREAS BEING PROVIDED SEPARATELY
- INDEPENDENT NUCLEAR ASSURANCE DIVISION - ENCOMPASSING TRAINING, QUALITY ASSURANCE AND A NUCLEAR SAFETY ASSESSMENT DEPARTMENT
- POOLING OF RESOURCES FOR SUPPORT OF SEVERAL GENERATING STATIONS
- PERSONNEL POLICIES AND PROCEDURES APPROPRIATE FOR NUCLEAR GENERATION

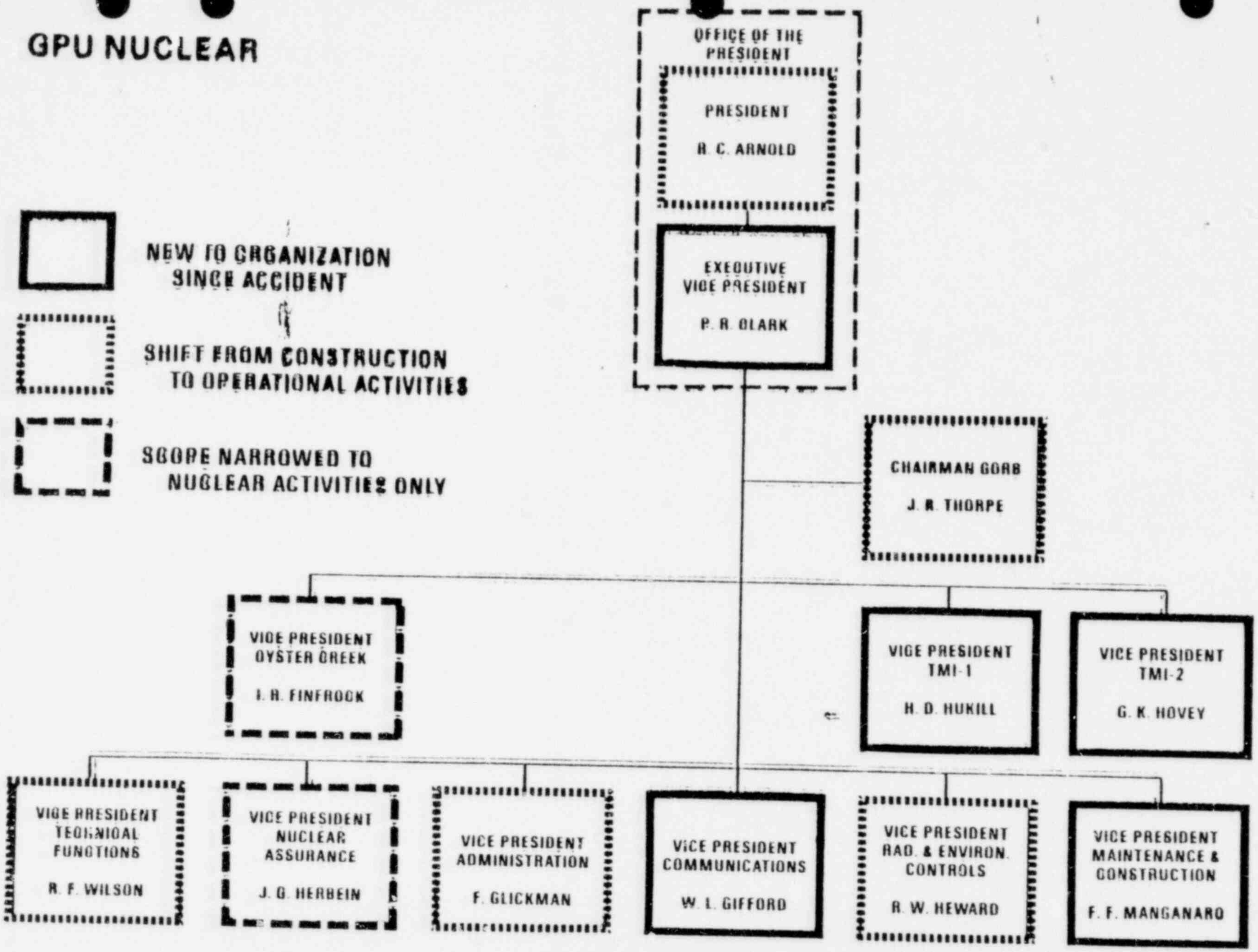
GPU NUCLEAR GROUP

PURPOSE

MANAGE AND DIRECT THE NUCLEAR ACTIVITIES OF THE GPU SYSTEM TO PROVIDE THE REQUIRED HIGH LEVEL OF PROTECTION FOR THE HEALTH AND SAFETY OF THE PUBLIC AND THE EMPLOYEES.

CONSISTENT WITH THE ABOVE, GENERATE ELECTRICITY FROM THE GPU NUCLEAR STATIONS IN A RELIABLE AND EFFICIENT MANNER IN CONFORMANCE WITH ALL APPLICABLE LAWS, REGULATIONS, LICENSES, AND OTHER REQUIREMENTS AND THE DIRECTIONS AND INTERESTS OF THE OWNERS.

GPU NUCLEAR


NEW TO ORGANIZATION SINCE ACCIDENT
SHIFT FROM CONSTRUCTION TO OPERATIONAL ACTIVITIES
SCOPE NARROWED TO NUCLEAR ACTIVITIES ONLY
OFFICE OF THE PRESIDENT
PRESIDENT
R. C. ARNOLD
EXECUTIVE VICE PRESIDENT
P. R. CLARK
CHAIRMAN GORB
J. R. THORPE
VICE PRESIDENT OYSTER CREEK
I. R. FINFROCK
VICE PRESIDENT TMI-1
H. D. HUKILL
VICE PRESIDENT TMI-2
G. K. HOVEY
VICE PRESIDENT TECHNICAL FUNCTIONS
R. F. WILSON
VICE PRESIDENT NUCLEAR ASSURANCE
J. G. HERBEIN
VICE PRESIDENT ADMINISTRATION
F. GLICKMAN
VICE PRESIDENT COMMUNICATIONS
W. L. GIFFORD
VICE PRESIDENT RAD. & ENVIRON. CONTROLS
R. W. HEWARD
VICE PRESIDENT MAINTENANCE & CONSTRUCTION
F. F. MANGANARO

GPU NUCLEAR

SUMMARY

A-	TECHNICAL PROFESSIONALS	416
B-	PROFESSIONAL EXPERIENCE	5039
C-	NUCLEAR EXPERIENCE	3153
D-	SENIOR REACTOR OPERATOR LICENSES OR EQUIV.	119
E-	TOTAL STAFF LEVEL	1947

OFFICE OF THE PRESIDENT
- PRESIDENT
- EXECUTIVE VICE PRESIDENT

A - 2
B - 50
C - 42
D - 1
E - 5

CHAIRMAN GORB

A - 1
B - 32
C - 25
D - 0
E - 2

VICE PRESIDENT OYSTER CREEK

A - 40
B - 426
C - 299
D - 23
E - 293

VICE PRESIDENT TMI-1

A - 34
B - 377
C - 291
D - 21
E - 300

VICE PRESIDENT TMI-2

A - 37
B - 379
C - 281
D - 24
E - 254

VICE PRESIDENT TECHNICAL FUNCTIONS

A - 178
B - 2315
C - 1294
D - 35
E - 259

VICE PRESIDENT NUCLEAR ASSURANCE

A - 61
B - 838
C - 502
D - 12
E - 214

VICE PRESIDENT ADMINISTRATION

E - 395

VICE PRESIDENT COMMUNICATIONS

E - 29

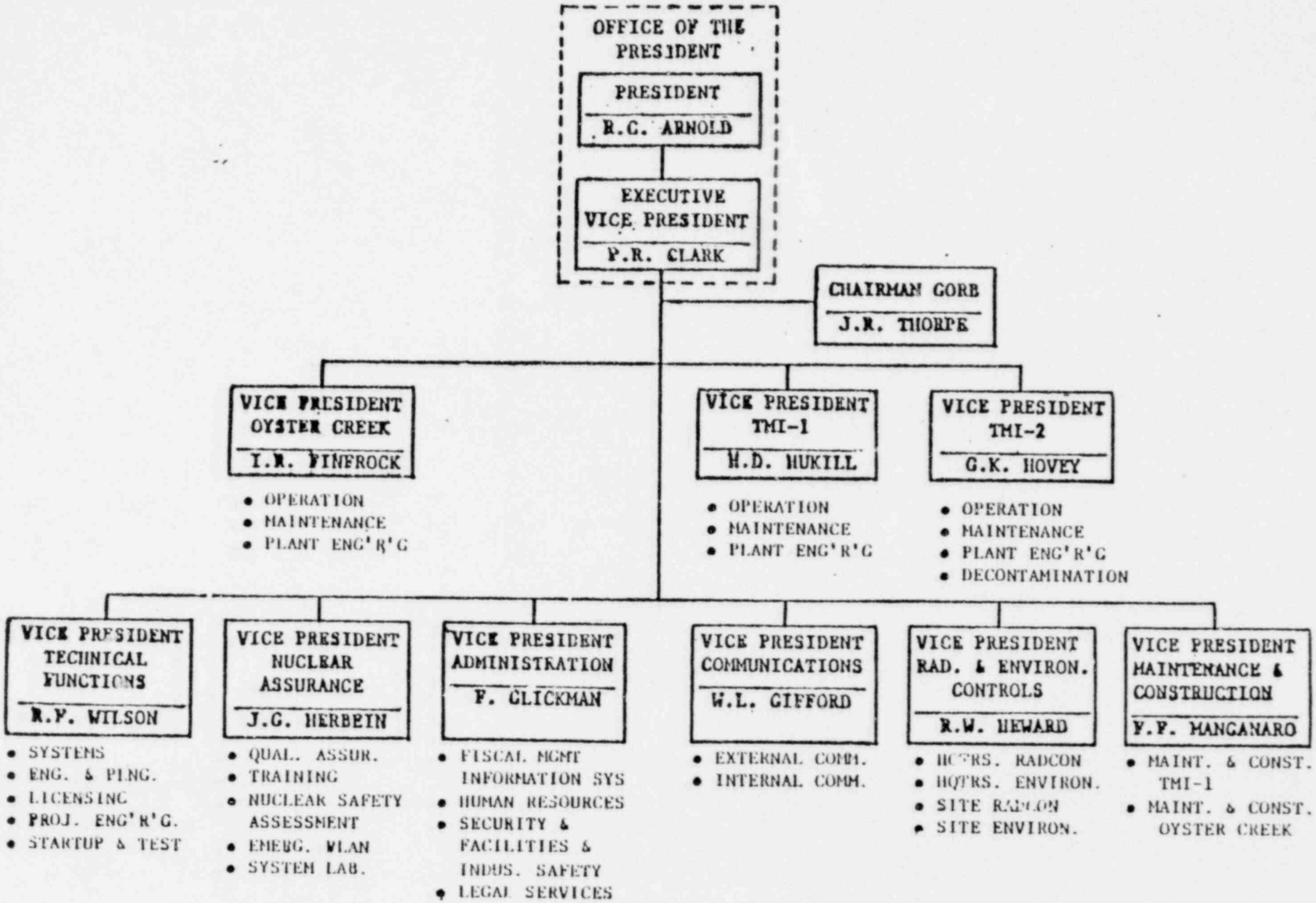
VICE PRESIDENT RAD & ENVIRON. CONTROLS

A - 52
B - 405
C - 284
D - 1
E - 180

VICE PRESIDENT MAINTENANCE & CONSTRUCTION

A - 11
B - 212
C - 135
D - 2
E - 16

GPU NUCLEAR


OFFICE OF THE PRESIDENT
PRESIDENT
R.C. ARNOLD
EXECUTIVE VICE PRESIDENT
P.R. CLARK
CHAIRMAN GORB
J.R. THORPE
VICE PRESIDENT OYSTER CREEK
I.R. FINFROCK
• OPERATION
• MAINTENANCE
• PLANT ENG'R'G
VICE PRESIDENT TMI-1
H.D. HUKILL
• OPERATION
• MAINTENANCE
• PLANT ENG'R'G
VICE PRESIDENT TMI-2
G.K. HOVEY
• OPERATION
• MAINTENANCE
• PLANT ENG'R'G
• DECONTAMINATION
VICE PRESIDENT TECHNICAL FUNCTIONS
R.F. WILSON
• SYSTEMS
• ENG. & PLNG.
• LICENSING
• PROJ. ENG'R'G.
• STARTUP & TEST
VICE PRESIDENT NUCLEAR ASSURANCE
J.G. HERBEIN
• QUAL. ASSUR.
• TRAINING
• NUCLEAR SAFETY ASSESSMENT
• EMERG. PLAN
• SYSTEM LAB.
VICE PRESIDENT ADMINISTRATION
F. GLICKMAN
• FISCAL MGMT INFORMATION SYS
• HUMAN RESOURCES
• SECURITY & FACILITIES & INDUS. SAFETY
• LEGAL SERVICES
VICE PRESIDENT COMMUNICATIONS
W.L. GIFFORD
• EXTERNAL COMM.
• INTERNAL COMM.
VICE PRESIDENT RAD. & ENVIRON. CONTROLS
R.W. HEWARD
• HQTRS. RADCON
• HQTRS. ENVIRON.
• SITE RADCON
• SITE ENVIRON.
VICE PRESIDENT MAINTENANCE & CONSTRUCTION
F.F. MANGANARO
• MAINT. & CONST. TMI-1
• MAINT. & CONST. OYSTER CREEK

PURPOSE

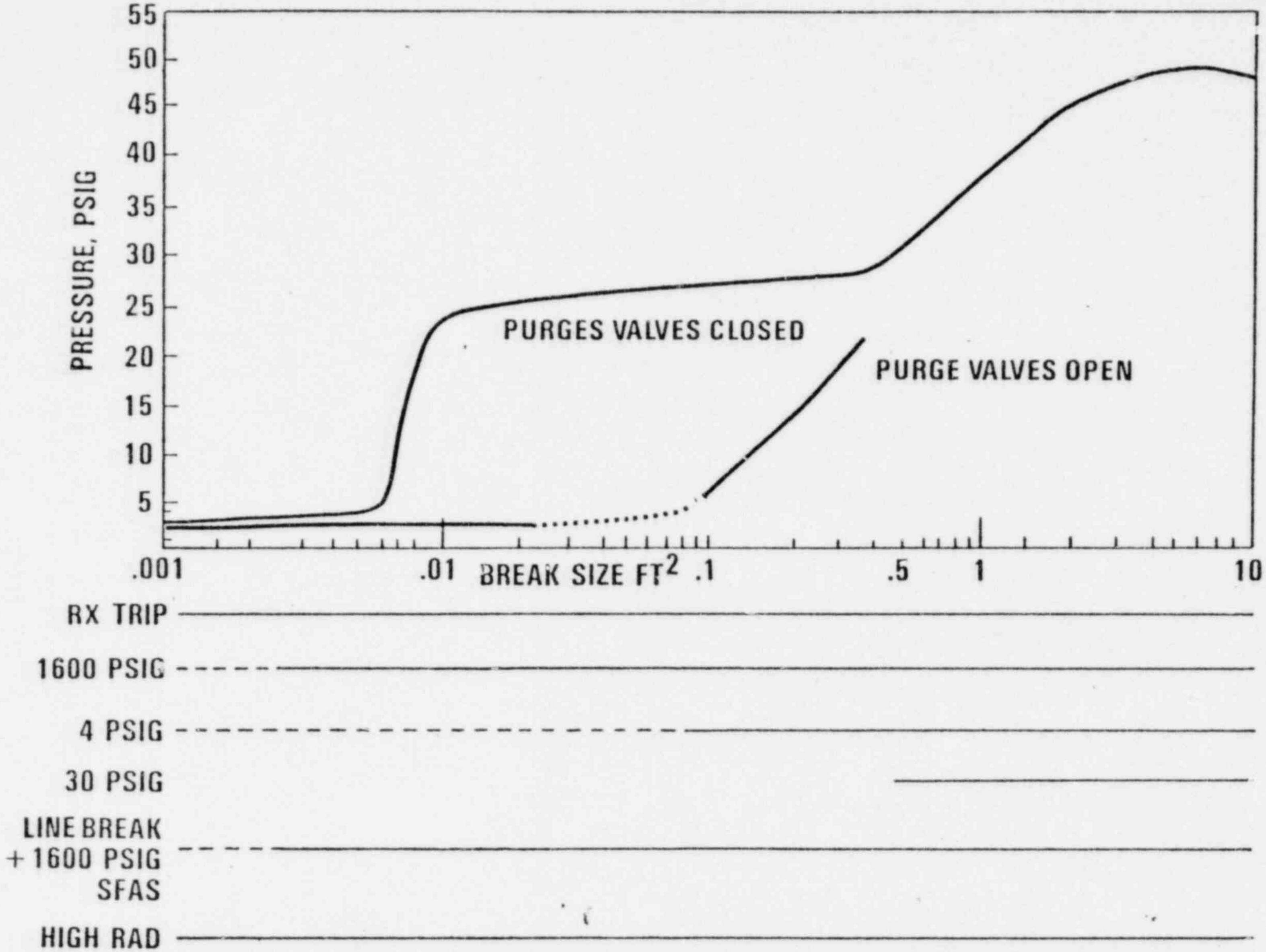
- Summarize Containment Isolation Signals
- Discuss Effect of Open Purge Valves on Containment Pressure

TMI-1 CONTAINMENT ISOLATION SIGNALS

	LINES	REACTOR TRIP	1600 PSIG SFAS	4 PSIG BLDG. PRESSURE	30 PSIG BLDG. PRESSURE	1600 PSIG STAS & LINE BRK	HIGH RADIATION LEVEL
PREVENT TRANSFER OF RADIATION OUTSIDE CONTAINMENT	R.B. SUMP						
	RCDT						
	RCS SAMPLE						
	R.B. PURGE						
	RCS LETDOWN	MU-V3	MU-V2A, B	MU-V2A, B MU-V3			ALARM
	OTSG SAMPLE						
	DEMIN WATER						
	CORE FLOOD SAMPLE & N_2 LINES						
FACILITATE STABLE SHUTDOWN	R.B. AIR COOLERS						
	RC MAKEUP						
	CONTAINMENT AIR SAMPLE						
MAINTAIN REACTOR COOLANT PUMP SERVICES	R.C. PUMP * SEAL RETURN						
	NSCCW						
	ICCW						

*IN ACCORDANCE WITH GDC–55

EFFECT OF BUILDING PURGE VALVES ON PEAK PRESSURE



CONTAINMENT PURGE VALVE EFFECTS ON BUILDING PRESSURE

ASSUMPTIONS:

- 2 48" Purge Valves Limited to 30° Open (17" EQU. DIA.)
- Purge Valve Closure on Reactor Trip Fails
- Purge Valve Closure on High Radiation Fails
- Maximum Heat Removal from Fan Coolers & Heat Sinks

RESULTS: (CONTEMPT)

- 4 PSIG is Reached for LOCA's Causing Core Uncovery (>.085 FT^2)
- 4 PSIG is not Reached for Very Small LOCA's with Purge Valves ~~Open.~~ Closed.

CONCLUSIONS:

- 4 PSIG Signal is Valid with Purge Valves Limited to 30° Open
- Reactor Trip Isolation Signal Anticipates ESAS and Building Pressure Signals for LOCA.

DUAL PRESSURE/TEMPERATURE SCALES

- Saturation Margin is Now the Basis for Operator Action
- Indication should be Continuously Monitored and Displayed
- Evaluation of T + P by Operator to Determine Saturation Margin
  - Is Not Continuous
  - Involves Several Steps
  - Involves Mental Processing
  - Subject to Error Under Stress
- Dual Scale Meter Appears Preferable to Steam Table
- Preferred Method is Direct Reading Indication
- TMI-1 Uses
  - 2 Saturation Margin Meters
  - Low Saturation Margin Alarm from Either Loop
  - Process Computer Calculation of Saturation Margin
  - Graphic Display of Pressure-Temperature Plot

MIDSCALE METER FAILURES

- Power Supply Loss (Crystal River 3 Event)

  Loss of Power Alarms & Indications

  Parameters Independent of ICS/NNI

  Plant Control Procedures

  Meter Failure Points Indicated & Keyed to Power Supply

- Individual Meter Failures

  Analog Meters: Failure Points Noted on Meter

  Digital Meters: Fail Dark or Zero

  Most Analog Meter Failure Points Outside Normal Band

  Analog Meter Failure Points in Normal Band have been Evaluated

  Digital Meters Added for Key Parameters, Including Those Used for Manual Control Post Trip

NUREG 666 Conclusions

Type 1 Failures:

Unavailability of Multiple Batteries on demand — 4×10^{-4}/demand

Assuming loss of off-site power occurrence at 0.22/year, unavailability of multiple batteries on demand. — 9×10^{-5}/Reactor Year

Type 2 Failures:

Unavailability of Multiple Batteries due to Test, Operational and Maintenance errors. — 6×10^{-5}/Reactor Year

Combination of Type 1 and Type 2 Failures represents a contribution of 50% of core damage probability for all accident sequences studied.

Contribution of DC Power Failure to core damage probability can be reduced from 50% to 1% by:

- Prohibiting certain design and operational features such as bus-ties
- Augmenting test and maintenance activities
- Incorporate requirements for staggered test and maintenance activities

TMI-1 System Improvements

Battery Disconnect Switches to be Locked Closed.

Bus Tie Switches to be Locked Open.

Substation Distribution Bus Tie to be Disabled.

Procedures Reviewed and Upgraded.

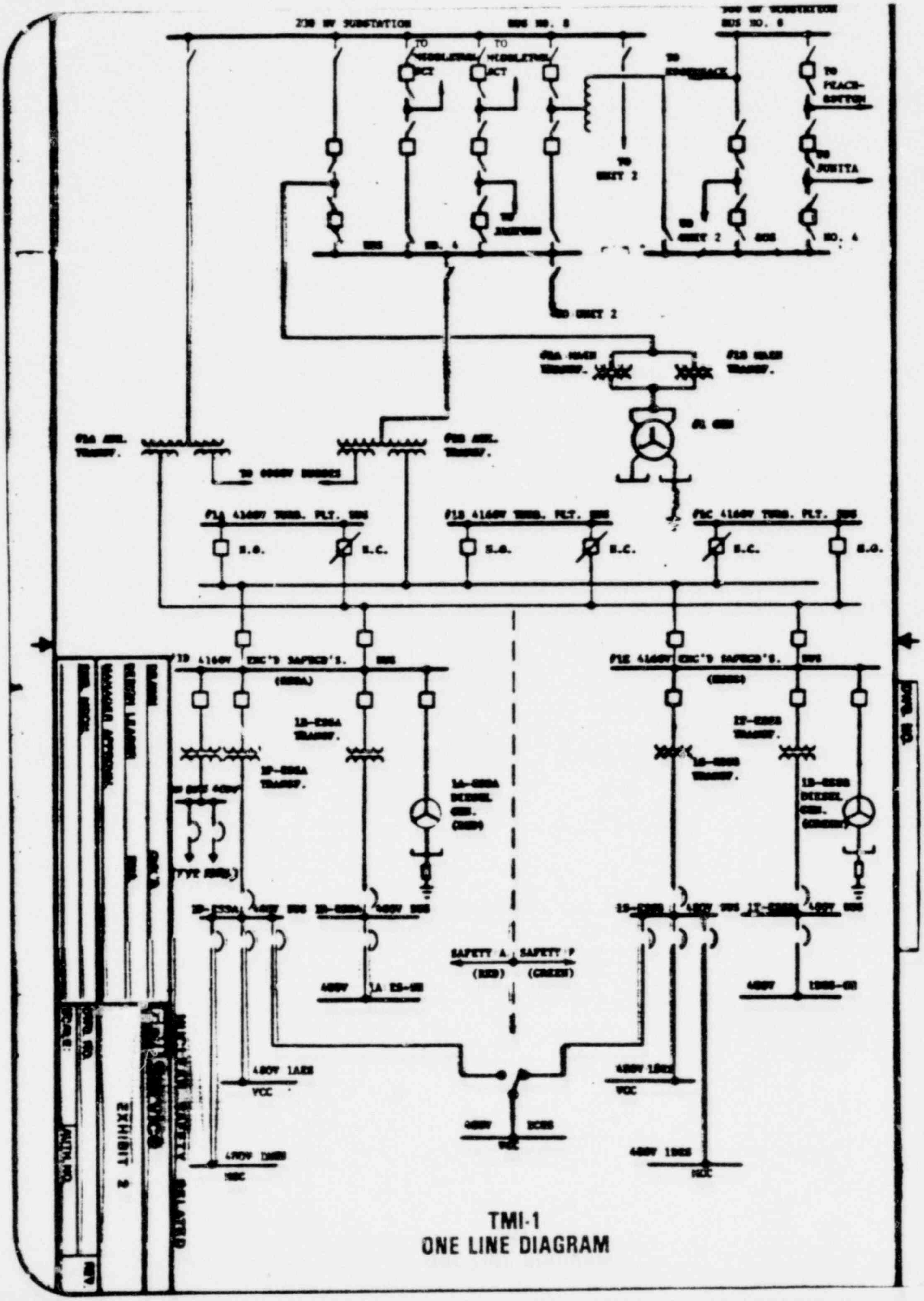
- Surveillance of Terminal Connections
- Upgrading of Battery Discharge Test Procedure
- Restrict Use of Battery Disconnect Switches and Bus Tie Switches to Cold Shut-down
- Disable Bus Ties between 230 KV Switchyard Dist. Panels
- Recovery Procedures Written for Loss of a DC Supply

Significant TMI-1 Plant Differences from NUREG 666

Stand-by Battery Chargers

More Reliable off-site Power

	Relative Reduction In System Unavailability		
	Type 1 Failures	Type 2 Failures	Combined
System Studied in NUREG 666	1.0	1.0	1.0
TMI-1 System at Restart			
1. Restrict use of bus tie to cold shut-down	1.0	0.001	
2. Stand-by Battery Charger	0.03	0.2	
3. More Reliable off-site Power	0.2	1.0	
Combination of 1, 2, & 3	0.006	0.001	0.003
Other TMI-1 Restart Improvements			
Improved Surveillance	0.03	1.0	0.5
Improved Maint. and Test	0.1	0.1	0.1



TMI-1
ONE LINE DIAGRAM

INDEX

ACRS Presentation
June 25 - 26, 1981
W. H. Behrle

ACRS PRESENTATION

JUNE 25 - 26, 1981

W. H. BEHRLE

TEST PROGRAM SCOPE

Testing Included in the Unit 1 Restart Test Program Consists of:

- Construction and Startup Testing of Newly Installed Components and Systems Installed by the Unit 1 Restart Program.

- Construction and Startup Testing of Modifications to Existing Plant Systems and Components by the Unit 1 Restart Program.

- Testing of Selected Non-Modified Systems as Determined by Startup and Plant Operations.

- Testing of Selected Maintenance Projects as Requested by Maintenance.

- Selected Integrated Testing During Plant Startup And Power Escalation to Assure Proper Plant Operation and Transient Response Following the 1979 Extended Outage.

BASES FOR SELECTION OF TESTS

- NORMAL REFUELING TEST REQUIREMENTS
- MODIFICATIONS MADE TO THE PLANT
- TMI 1 INITIAL TEST PROGRAM
- NATURAL CIRCULATION TESTING PERFORMED AT NTOL PLANTS
- REG GUIDE 1.68 TESTING PERFORMED AT NEW PLANTS
- PLANT REINITIALIZATION CONSIDERATIONS
- OPERATOR TRAINING CONSIDERATIONS
- PROCEDURE VERIFICATION CONSIDERATIONS
- SURVEILLANCE CONSIDERATIONS
- PLANT TRANSIENT ANALYSIS VERIFICATION

MAJOR DOCUMENTS CONSULTED

- NUREG - 0578 - Lessons Learned (Short Term)
- NUREG - 0585 - Lessons Learned (Long Term)
- I&E Bulletins - 79 - 01 and 05
- August 9, 1979 Shutdown Order
- TMI #1 Restart Report (Thru Amendment 25)
- NUREG - 0660 - TMI Action Plan
- NUREG - 0680 - Safety Evaluation Report (of TMI #1 Restart Report)
- NUREG - 0694 - TMI Related Requirements for NTOL's
- NUREG - 0737 - Clarification of TMI Action Plan Requirements
- Reg Guide 1.68 - Initial Test Programs for Nuclear Power Plants
- Draft Reg Guide for LWR Refueling and Startup Tests

TEST PROGRAM ORGANIZATION/CONTROL

THE TMI #1 RESTART TEST PROGRAM IS PERFORMED IN AN ORGANIZED FASHION BY FORMALLY APPROVED DOCUMENTS. DETAILED NUCLEAR SAFETY RELATED TEST PROCEDURES AND RESULTS ARE APPROVED BY A TECHNICAL REVIEW/APPROVAL GROUP.

- o TEST MANUAL AND INSTRUCTIONS -

  1) ESTABLISHES THE STARTUP AND TEST GROUP TO PREPARE, PERFORM AND DOCUMENT THE TEST PROGRAM.

  2) ESTABLISHES THE AUTHORITY OF THE TECHNICAL REVIEW/APPROVAL GROUP AS THE CENTRAL APPROVAL AND COORDINATING BODY.

  3) DEFINES RESPONSIBILITIES OF VARIOUS PARTICIPANTS AND ORGANIZATIONS TO THE TEST PROGRAM.

  4) PROVIDES INSTRUCTIONS FOR TEST PROCEDURE FORMAT AND CONTENT, PREREQUISITE LISTS, TEST ENGINEER'S LOG, TEST BRIEFINGS, ETC.

- o MASTER TEST INDEX (MTX) - IDENTIFIES ALL TESTS REQUIRED FOR RESTART. LISTING INCLUDES CONSTRUCTION, FUNCTIONAL AND INTEGRATED PLANT TESTING.

- o TEST SPECIFICATION - IDENTIFIES TEST SCOPE AND ACCEPTANCE CRITERIA FOR ALL FUNCTIONAL AND INTEGRATED PLANT TESTS.

TEST PROGRAM ORGANIZATION/CONTROL (CONTINUED)

- TECHNICAL REVIEW/APPROVAL GROUP - COMPOSED OF REPRESENTATIVES FROM:

  1) ENGINEERING

  2) PLANT OPERATIONS

  3) STARTUP AND TEST

  4) NSSS SUPPLIER

  5) QA

  THIS GROUP REVIEWS AND APPROVES SAFETY RELATED TEST PROCEDURES AND RESULTS.

NON-MODIFIED SYSTEMS

- o Instrument Air
- o Secondary Services Closed Cooling System
- o Secondary River Water System
- o Nuclear Services River Water System
- o Circulating Water System
- o Main and Auxiliary Steam System
- o Condenser Air Removal System
- o Condensate System
- o Penetration Pressurization System
- o Fluid Block System
- o Penetration Cooling System
- o Intermediate Closed Cooling Water System
- o Gaseous Waste Disposal System
- o Reclaimed Water System

Selection of Non-Modified Systems for testing was based on "Important to Safety" considerations. For example, failure of Condenser Air Removal System (Loss of Vacuum) causes shift from Turbine Bypass Valves to Atmospheric Dump Valves.

LIST OF HFT's

1) Checkout Incore Thermocouples (Mod)

2) Set Main Steam Safety Valves (Surv)

3) Test Main Steam Safety Valve Acoustic Monitors (Mod)

4) Verify proper operation of Pressurizer Heater Level/Pressure Interlocks, Spray Valve Flow/Pressure Interlock (RI)

5) Determine Pressurizer heat losses and ability to control pressure and saturation margin on one(1) Heater Bank (RI/Mod)

6) Lift the PORV and determine adequate response of Elbow Taps, Acoustic Monitor, Tailpipe Thermocouples and Manual Switch (Mod)

7) Run Steam Driven Emergency Feed Pump on recirculation and verify it does not overspeed and comes up to rated speed in less than 30 sec (Mod/Surv)

8) Perform Diesel Generator Loading Test combining ES with Motor Driven Emergency Feed Pump Auto Start (Mod/Surv)

9) Check agreement of various Non-Nuclear Instrument Channels as a function of RCS Temperature/Pressure (RI/Mod)

10) Perform HPI Functional Test to verify Cavitating Venturis and high capacity Makeup Valve (Mod/Surv)

11) Verify proper operation of Tsat Meter as a function of RCS Temperature/Pressure (Mod)

12) Verify ability of RB Coolers to maintain RB Temperature less than design (PMT)

13) Verify operability of RCS High Point Vents (Mod)

14) Verify ability to sample RCS at normal operating temperature and pressure with long-handled tools in an acceptable time period (Mod)

15) Verify acceptable RCP operating parameters as a function of RCS Temperature/Pressure (RI)

16) Flow balance the Intermediate Cooling Water System, as required (RI)

17) Take thermal expansion readings on hangers/supports as a function of RCS Temperature (Mod)

18) Perform RCS leakage measurements (Surv)

19) Perform CRD drop time measurements (Surv)

20) Perform DH-V22A/B, CF-V4A/B & 5A/B Leakage Surv Test (Mod/Surv)

ZERO POWER PHYSICS PROGRAM

At zero nuclear power, 2155 psig and 532°F RCS conditions, perform the following physics tests in accordance with normal refueling testing procedures:

1) Control Rod Withdrawal
2) Deboration to Critical
3) Determination of Sensible Heat and NI Overlap
4) Perform Reactivity Measurements
5) Determination of all Rods Out Boron Concentration
6) Determination of All Rods Out Temperature Coefficient
7) Integral Control Rod Worth Measurements
8) Shutdown Margin Verification
9) Temperature Coefficients
10) Ejected Control Rod Worth Measurement

LOW NUCLEAR POWER TESTING
INCLUDING NATURAL CIRCULATION

1) Perform correlation between Out of Core Detector indication vs. Heat Balance Power as a function of T_{COLD} (3% Power)

2) Verify auto start of Emergency Feedwater Pumps and OTSG Level Control at 30" on Startup Range upon loss of both Feedwater Pumps and demonstrate adequacy of flow indication (RM-13B) (3% Power)

3) Verify ability to control level with the new Manual Loader Station (RM-13D) (3% Power)

4) Verify adequate air supply to Emergency Feedwater Control Valves (EF-V30A/B) and Turbine Driven Emergency Feed Pump Steam Control Valve (MS-V6) for 2 hours with loss of Instrument and Backup Instrument Air (RM-13H) (3% Power)

5) Verify smooth transition to Natural Circulation Flow with OTSG Level Conrol at 50% on Operating Range upon loss of all 4 RCP's and demonstrate adequacy of flow indication (RM-13B) (3% Power)

6) Determine effect of loss of Pressurizer Heaters on Saturation Margin (3% Power)

7) Determine effect of SG Level on Natural Circulation Flow (3% Power)

8) Verify that OP 1102-16 (RCS Natural Circulation Cooling) provides adequate guidance to prevent overcooling as OTSG Level Control setpoint changes from 30" on Startup Range to 50% on Operating Range (Following 40% Power Trip)

LOW NUCLEAR POWER TESTING INCLUDING
NATURAL CIRCULATION (CONTINUED)

9) DETERMINE LOWEST LEVEL IN OTSG THAT SUSTAINS NATURAL CIRCULATION FLOW WITH NO EMERGENCY FEEDWATER (FOLLOWING 100% POWER TRIP)

POWER ESCALATION TESTING

Following the Low Power Test Program, escalate power to the 15%, 40%, 75% and 100% power plateaus in steps and perform the following testing, as indicated:

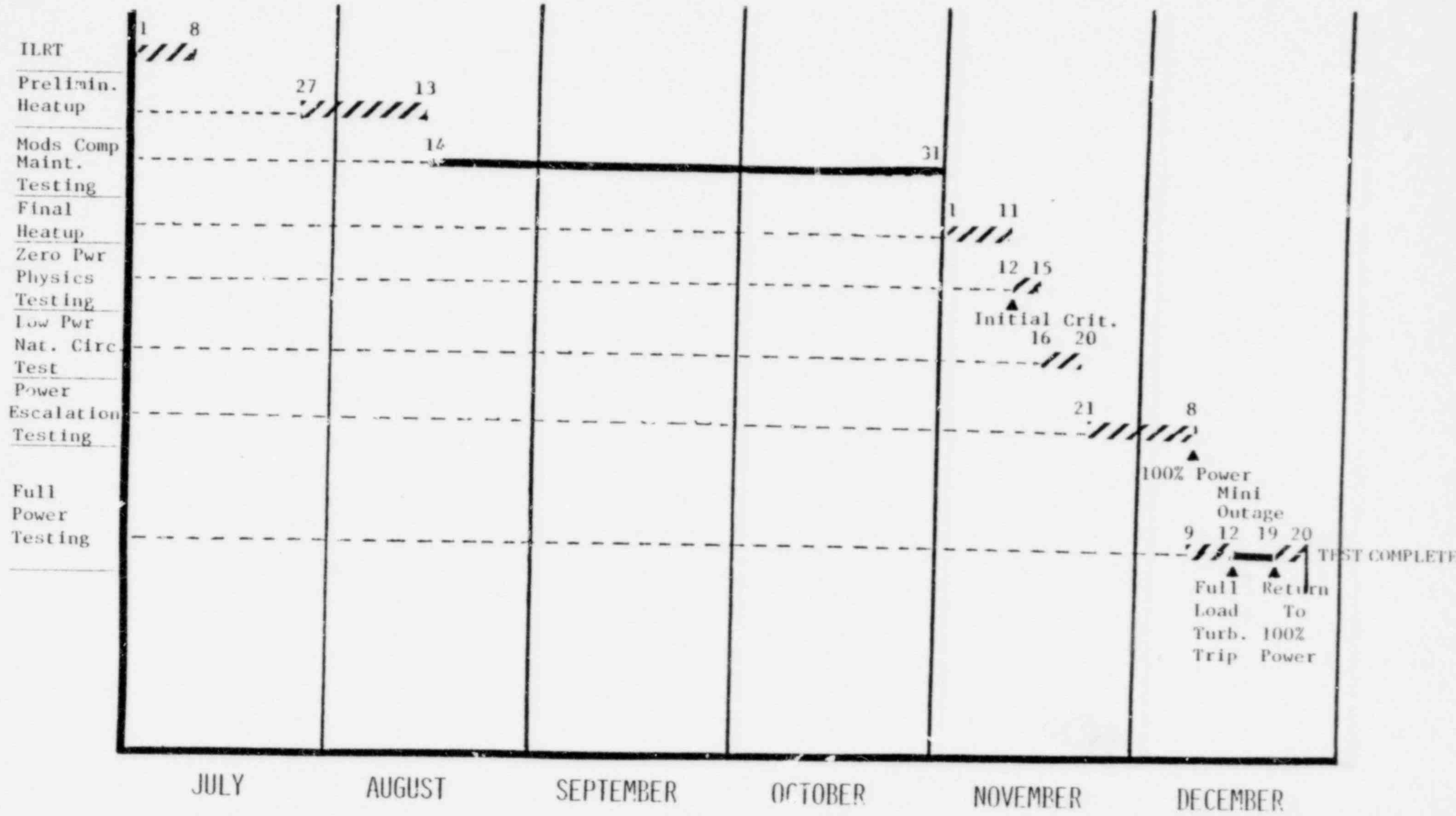
1) Perform Turbine Overspeed Surveillance Test (0-12%-0) (Surv)
2) Perform Nuclear Instrumentation Calibration at power (15%, 40%, 76% and 100%) (Surv)
3) Perform ICS Tuning at power (20%, 40%, 80% and 95-100% and at minor levels between majors)
4) Perform Turbine Bypass Valve Testing at power (% open vs. steaming rate and correct bias from ICS on Turbine Trip) (0-15%)
5) Perform Heat Balance Surveillance Testing and verify computer program as required (15%, 40%, 76% and 100%)
6) Perform Feedwater System Operation and Tuning (40%, 76%, 100%)
7) Perform Turbine Generator Operation and Testing (15%, 40%, 76%, 100%)
8) Perform Power Imbalance Detector Correlation Test (40%) (Surv)
9) Perform Unit Load Transient Testing (40%, 80%, 100%)
10) Perform Unit Load Steady State Testing (15%, 40%, 76%, 100%)
11) Perform Saturation Monitor Checks (15%, 40%, 76%, 100%) (LM-1)
12) Perform Incore Thermocouple Checks (15%, 40%, 76%, 100%) (RM-4)
13) Perform Core Power Distribution Testing (15%, 40%, 76%, 100%) and verify computer programs.
14) Perform Reactivity Coefficients Testing (100%)
15) Verify RCP Flow (100%) (Surv)
16) Perform loss of both Feedwater Pumps (40%) (RM-3) (Verify letdown isolation and bypass - RM-5)
17) Perform loss of one(1) Feedwater Pump runback (100%)
18) Perform Turbine Trip Test (100%) (RM-3)
19) Perform CRD misalignment runback (76%)
20) Check thermal expansion/hanger settings of systems affected by power operation (15%, 40%, 76%, 100%)

TEST PROCEDURE REQUIREMENTS

Test Type	Number Required*	Written	Approved	Performed
Non-Modified Systems	15	14	0	0
Modification Functional	29	19	10	3
Hot Functional	14	12	0	0
Low Power/Power Escalation	16	3	0	0
	74	48	10	3

*Based on review/planning of 57 modifications

TMI-1 RESTART INTEGRATED SCHEDULE



COMPARISON OF TMI #1 RESTART LOW POWER TEST PROGRAM

WITH SEQUOYAH LOW POWER TEST PROGRAM

10 Tests required by NRC for Sequoyah
8 Tests required with simulated Decay Heat and
1 Test required with real Decay Heat for all other NTOL's

Sequoyah	TMI Restart
1) Establish stable N/C conditions	1) Included in TP 700/2 - each shift participates/ witnesses.
2) Establish N/C with simulated loss of off-site power	2) Not included in Restart Program. Was verified during actual loss of off-site power in initial startup in 1974.
3) N/C with loss of Pressurizer Heaters	3) Included in TP 700/2 - Low Power Natural Circulation Test Procedure
4) Effect of OTSG secondary side isolation on N/C flow	4) Not included in Restart Program. Single OTSG N/C flow was verified on B&W NSSS on TMI #2
5) N/C flow at reduced RCS pressure	5) No auxiliary spray is available on TMI #1. Any performance of this item in TP 700/2 will be a continuation of item 3.
6) Cooldown capability of Makeup/Letdown	6) Included in TP 651/1 - Intermediate Cooling System Flow Balance.
7) Simulated loss of all on-site and off-site AC power	7) Secondary side heat sink (Emergency Feedwater) availability without AC power is demonstrated with forced RCS flow in TP 700/2 as part of EFW Modification Testing. Primary side response would be no different than Loss of Off-Site Power Test in initial Startup Program and would cause loss of seals to RCP's which could degrade seal life.

Sequoyah	TMI Restart
8) Establish N/C from stagnant conditions	8) NRC deleted this requirement from all NTOL's subsequent to Sequoyah. It is not included in the TMI Restart.
9a) Forced circulation cooldown	9a) Included in TP 700/2 - Low Power Natural Circulation Test Procedure.
9b) Boron mixing and cooldown	9b) Not included in Restart Program. Boron mixing and cooldown were verified on B&W NSSS on TMI #2.
	10) Verify that Plant Natural Circulation Procedures provide adequate guidance to the operator to prevent overcooling as OTSG level changes from 30" on Startup Range to 50% on Operating range.
	11) Determine the lowest level in the OTSG that sustains Natural Circulation flow without Emergency Feedwater flow.

TMI-1 PROBABILISTIC RISK ASSESSMENT

- PURPOSE
  - Evaluate Risk to Public
  - Evaluate Plant Reliability & Availability
- TECHNIQUES
  - Detailed Event Sequence Diagrams & Event Trees
  - Identify Critical Systems
  - Systems Analysis for Failure Paths (Fault Trees)
  - Quantification Using TMI-1 Data where Available
- SCOPE
  - Random Failures
  - Common Mode
  - Common Cause
  - External Hazards
- USES
  - Identify Changes Significant to Public Risk
  - Identify Areas for Reliability & Availability Improvements
  - Relative Benefits of Changes
- SCHEDULE
  - Bid Specification -- Third Quarter '81
  - Commence Study -- Fourth Quarter '81

REQUIREMENT

NUREG - 0737 SECTION II.F.2

"LICENSEES SHALL PROVIDE A DESCRIPTION OF ANY ADDITIONAL INSTRUMENTATION OR CONTROLS (PRIMARY OR BACKUP) PROPOSED FOR THE PLANT TO SUPPLEMENT EXISTING INSTRUMENTATION (INCLUDING PRIMARY COOLANT SATURATION MONITORS) IN ORDER TO PROVIDE AN UNAMBIGUOUS, EASY-TO-INTERPRET INDICATION OF INADEQUATE CORE COOLING (ICC)."

"THE EVALUATION IS TO INCLUDE REACTOR-WATER-LEVEL INDICATION."

GPUN APPROACH TO
EVALUATION OF WATER LEVEL MEASUREMENT

- Define use and develop criteria
  - Participated in B&W Owner's Group Evaluation
  - In-house evaluation vs operator guidelines
  - Considering uses other than operator action
- Evaluate Potential Detectors
  - Participated in B&W Owner's Group Evaluation
  - In-house evaluations
  - Sponsoring study by consultant of possible methods
  - Cooperating in University proposal re neutron detectors
  - Will review EPRI evaluation (due October 1981)
- Select Appropriate Action
  - Install available detector(s)
  - Support further development
  - Define alternate approach

POSSIBLE USES OF WATER LEVEL AND RELATED INSTRUMENTS (FOR B&W PLANTS)

LOCA RESPONSE

- REQUIRED OPERATOR ACTIONS CAN BE BASED ON EXISTING INSTRUMENTS.
- LOOP WATER LEVEL MIGHT BE USED AS CONFIRMATION/BACKUP.
- VOID FRACTION MIGHT BE USED AS NEW PUMP TRIP CRITERION.

OVER-COOLING RESPONSE

- REQUIRED OPERATOR ACTIONS CAN BE BASED ON EXISTING INSTRUMENTS.
- RCS INVENTORY COULD BE USED TO CONFIRM NOT A LOCA.

RCS VENTING (POST ACCIDENT)

- OPERATOR GUIDELINES UNDER DEVELOPMENT.
- WATER LEVEL IN HOT LEGS MAY BE USEFUL.

RESPONSE TO BUBBLE IN HEAD

- NEW OPERATOR GUIDELINES UNDER DEVELOPMENT.
- USEFULNESS OF VESSEL WATER LEVEL NOT YET DEFINED.

POST TRIP EVALUATIONS

- USEFULNESS OF SPECIFIC INSTRUMENTS DEPENDS ON SCENARIO.

LEADING CANDIDATES FOR LEVEL MEASUREMENT

METHOD	DEVELOPER	COMMENTS
VESSEL ΔP	WESTINGHOUSE EG&G GE	DIRECT LEVEL MEASUREMENT UNDER QUIESENT CONDITIONS. INDICATES "EQUIVALENT" LEVEL FOR 2 PHASE, LOW FLOW CONDITIONS. DIFFICULT TO INTERPRET WITH FORCED FLOW.
HOT-LEG ΔP	B&W	SIMILAR IN PRINCIPLE TO VESSEL ΔP. GOOD "ANTICIPATION" BUT NOT FULL RANGE.
HEATED T/C'S	CE ORNL (EG&G)	INDICATES LEVEL AT DISCRETE INTERVALS. RESPONSE VS QUALITY OF FLUID MUST BE KNOWN. REQUIRES APPROPRIATE PENETRATIONS IN REACTOR HEAD.
NEUTRON DETECTORS	EPRI (PREVIOUSLY) PSU (POTENTIALLY)	NON-INTRUSIVE DETECTORS. TESTS INDICATE SENSITIVITY GOOD WITH WATER LEVEL WITHIN 8 FEET OF TOP OF CORE.
CORE EXIT T/C'S	?	MAY BE ABLE TO CORRELATE TO WATER LEVEL IF BELOW TOP OF CORE.

CURRENT GPUN CONCLUSIONS
RE WATER LEVEL OR ALTERNATE MEASUREMENT

- NOT REQUIRED PRIOR TO TMI RESTART
  - NO NEED AS INPUT TO SAFETY SYSTEMS.
  - REQUIRED OPERATOR ACTIONS CAN BE PERFORMED BASED ON EXISTING INSTRUMENTS.

- CRITERIA FOR DETECTOR NOT YET CLEAR
  - WATER LEVEL VS VOID FRACTION VS INVENTORY NEEDS FURTHER STUDY
  - ADDED INFORMATION MIGHT BE HELPFUL FOR CONFIRMATION OR LATER EVALUATIONS.
  - ADDED INFORMATION MIGHT HELP GUIDE LONGER TERM ACTIONS (E.G., VENTING).
  - USE OF VOID FRACTION AS BASIS FOR PUMP TRIP NEEDS EVALUATION.

- NO "IDEAL" DETECTOR HAS BEEN IDENTIFIED
  - FORCED FLOW VS LOW FLOW/STAGNANT POOL IS A PROBLEM.
  - EXISTING SYSTEMS DO NOT MEET ALL NRC CRITERIA.
  - NEW APPROACHES SHOULD BE CONSIDERED.

- PREMATURE INSTALLATION IS INAPPROPRIATE
  - MAY ADD UNNECESSARY COMPLEXITY
  - COULD BE MISLEADING UNLESS USE IS CAREFULLY DEFINED.

- GPUN WILL CONTINUE ACTIVE EVALUATION OF CRITERIA AND DETECTORS.

II.F.2 - INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (ICC)

MINIMUM INSTRUMENTATION FOR ICC MONITORING SYSTEM

ICC INSTRUMENTS (SEE NOTE 1)	REACTOR TYPE (SEE NOTE 2)			
	WESTINGHOUSE	CE	B&W	GE
SATURATION METER	REQUIRED	REQUIRED	REQUIRED	NOT REQUIRED
COOLANT INVENTORY (LEVEL) ABOVE CORE	REQUIRED	REQUIRED	REQUIRED*	REQUIRED
COOLANT INVENTORY (LEVEL) WITHIN CORE	DESIRABLE	NOT REQUIRED**	NOT REQUIRED**	REQUIRED
CORE EXIT T/Cs	REQUIRED	REQUIRED	REQUIRED	CORE T/C REQUIRED*

* NO FIRM DESIGN PROPOSED BY VENDOR.

** NOT REQUIRED PROVIDED THAT CORE EXIT THERMOCOUPLE INFORMATION IS PROCESSED, RECORDED, AND DISPLAYED IN AN ACCEPTABLE MANNER TO FACILITATE INTERPRETATION OF CORE COOLING CONDITIONS IN CONJUNCTION WITH ABOVE-CORE LEVEL INSTRUMENTATION.

NOTE 1; LEVEL INSTRUMENTATION MUST BE TESTED AND EVALUATED FOR LARGE BREAK LOCA SURVIVABILITY AND POST LOCA OPERABILITY.

NOTE 2: REQUIREMENTS ARE BASED ON REACTOR VENDOR PROPOSED INSTRUMENTATION, INTERCHANGEABILITY OF INSTRUMENTATION SYSTEMS IS ACCEPTABLE.

BASIS FOR STAFF POSITION
ON ICC MONITORING SYSTEMS

INSTRUMENT	REFERENCE FOR REQUIREMENT
SATURATION METER	NUREG-0660 NUREG-0737 R.G. 1.97
COOLANT INVENTORY (LEVEL)	NUREG-0660 NUREG-0737 (CLARIFICATION ITEM 6) R.G. 1.97 *
CORE EXIT T/Cs	FOR PWR: NUREG-0737 (II.F.2 ATTACHMENT 1) R.G. 1.97 FOR BWR: LASALLE SER R.G. 1.97

* ICC INDICATION RANGE

BWR - FROM BOTTOM OF CORE SUPPORT PLATE TO LESSER OF TOP OF VESSEL OR CENTERLINE OF MAIN STEAM LINE

PWR - FROM BOTTOM OF CORE TO TOP OF VESSEL

STAFF POSITION - ICC INSTRUMENTATION FOR RESTART

* EXISTING INSTRUMENTATION WITH COMMITMENT TO UPGRADE PER NUREG 0737 IS ACCEPTABLE FOR RESTART

* EVIDENCE OF REASONABLE PROGRESS ON ADDITIONAL INSTRUMENTATION (REACTOR WATER LEVEL) IS REQUIRED

Criteria to Show Evidence of Reasonable Progress on Additional Instrumentation (Reactor Water Level)

1. Selection of a level measurement system concept or an equivalent system for development
2. Definition of the development program and scheduel for development and procurement of the selected system
3. Evidence of a tangible commitment to performance or participation in the appropriate test programs to execute the defined development program
4. Justification for the concept selected if it results in significant schedule delays
5. Contingency plans and schedule for procurement of an alternative concept
6. Appropriate analyses to incorporate the water level status information into the guidelines for operator actions with respect to ICC

Pogin

1981 ANNUAL RADIATION EMERGENCY EXERCISE SCENARIO

THREE MILE ISLAND

UNIT I

JUNE 2, 1981

EXERCISE EVENTS

DECLARATION OF EMERGENCY

NOTIFICATION OF OFFSITE AGENCIES

EMERGENCY ANNOUNCEMENTS

ACTIVATION OF ONSITE ORGANIZATION

ACTIVATION OF OFFSITE ORGANIZATION

FULL ACCOUNTABILITY

RCS POST ACCIDENT SAMPLE

SEARCH AND RESCUE

*EVACUATION OF PERSONNEL

MONITORING AT ASSEMBLY AREA

*CONTAMINATED PERSONS ARE DECONTAMINATED

EMERGENCY MEDICAL ASSISTANCE TO CONTAMINATED-INJURED PERSON

OFFSITE MEDICAL RESPONSE

RESPONSE TO FIRE AT CWP HOUSE

STATE POLICE HELICOPTER NOTIFICATION AND TRAFFIC CONTROL

MAJOR SCENARIO EVENTS

T (MIN)

T (MIN)		
30	1.	PRIMARY TO SECONDARY LEAK INDICATED BY INCREASING COUNTRATE ON CONDENSER OFFGAS MONITOR.
140	2.	STEAM LINE RADIATION MEASUREMENTS INDICATE LEAK IN BOTH STEAM GENERATORS.
170	3.	WASTE GAS COMPRESSOR SEAL FAILURE ALLOWS BUILDUP OF AIRBORNE RADIOACTIVITY IN AUXILIARY BUILDING.
220	4.	STEAM GENERATOR TUBE RUPTURE INDICATED BY STEP CHANGES IN COUNTRATE ON THE OFFGAS MONITOR AND MAKE-UP FLOW RATE.
320	5.	AUXILIARY OPERATOR IS INJURED AND CONTAMINATED WHILE INVESTIGATING MAKE-UP PUMP PROBLEM. OFFSITE MEDICAL ASSISTANCE.
380	6.	FIRE IN CIRCULATING WATER PUMP HOUSE. OFFSITE FIRE ASSISTANCE.
390	7.	FIRE CAUSES LOSS OF ALL CIRCULATING WATER PUMPS AND CONDENSER VACUUM REQUIRING STEAMING TO ATMOSPHERE TO CONTINUE COOLDOWN. MAJOR FUEL FAILURE OCCURS.
425	8.	OFFSITE POWER IS LOST. "B" DIESEL GENERATOR PICKS UP VITAL LOADS.
UNTIL TERMINATION	9.	REACTOR COOLANT SYSTEM IS ON NATURAL CIRCULATION REMOVING DECAY HEAT BY STEAMING TO ATMOSPHERE.

GPUN Problem Categorization

Problem or Issue:

— Industry Generic

- Broad applicability in industry
- Generally long-term
- GPUN has no unique expertise or resource
- Regulatory position defined or undefined
- May require major resources

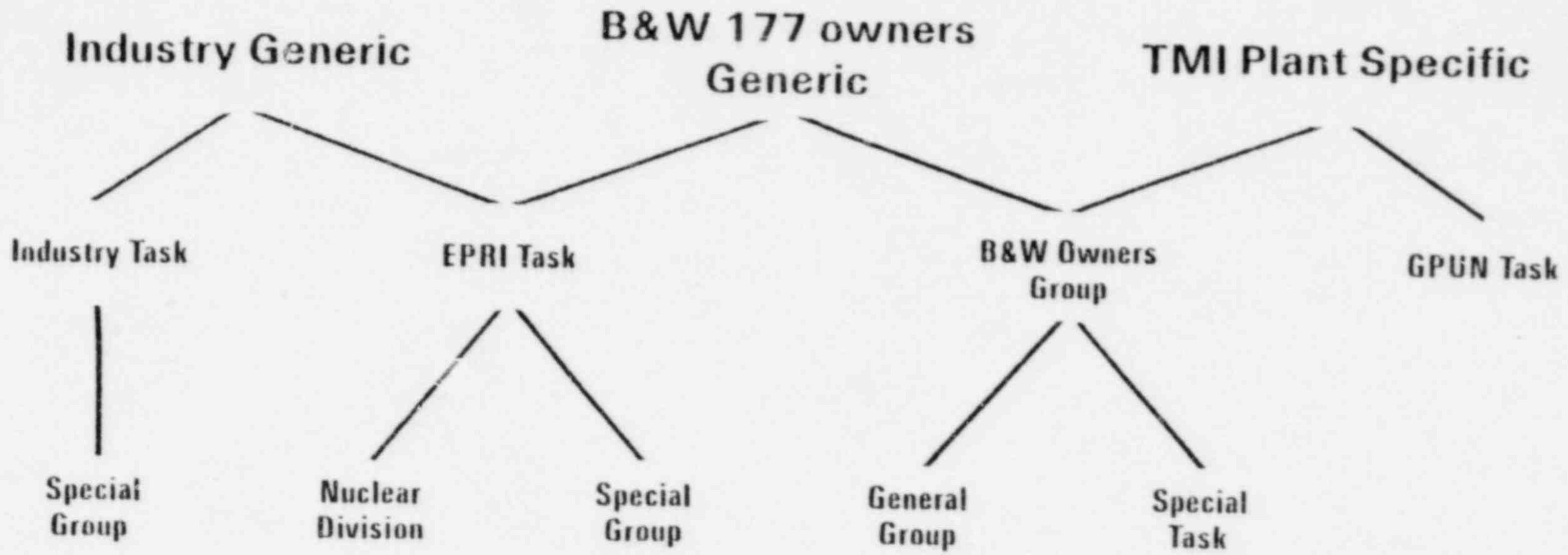
— B&W 177 Owners Specific

- Applies mainly to B&W plants
- Combined resources may expedite solution
- Near or long-term
- Owners + B&W have unique capability
- Regulatory position defined or undefined

— TMI Plant Specific

- Specific to TMI
- Near, intermediate, or long-term
- Regulatory position generally defined
- Generally little or no benefit to group approach

Problem Solution



GPUN Responsibility	Licensing & Regulatory Affairs	Projects & Tech Specialists	Tech Specialist	Engr. Projects & Licensing	Tech Specialists	Engr. Projects
Example	• Degraded Core Rule Making	• Transient Analysis	• Valve Test Program • Steam Generators	• PV Water Level	• Steam Generators • Asymmetric LOCA Loads	• TMI Lessons Learned • GPUN Mgmt. Directives

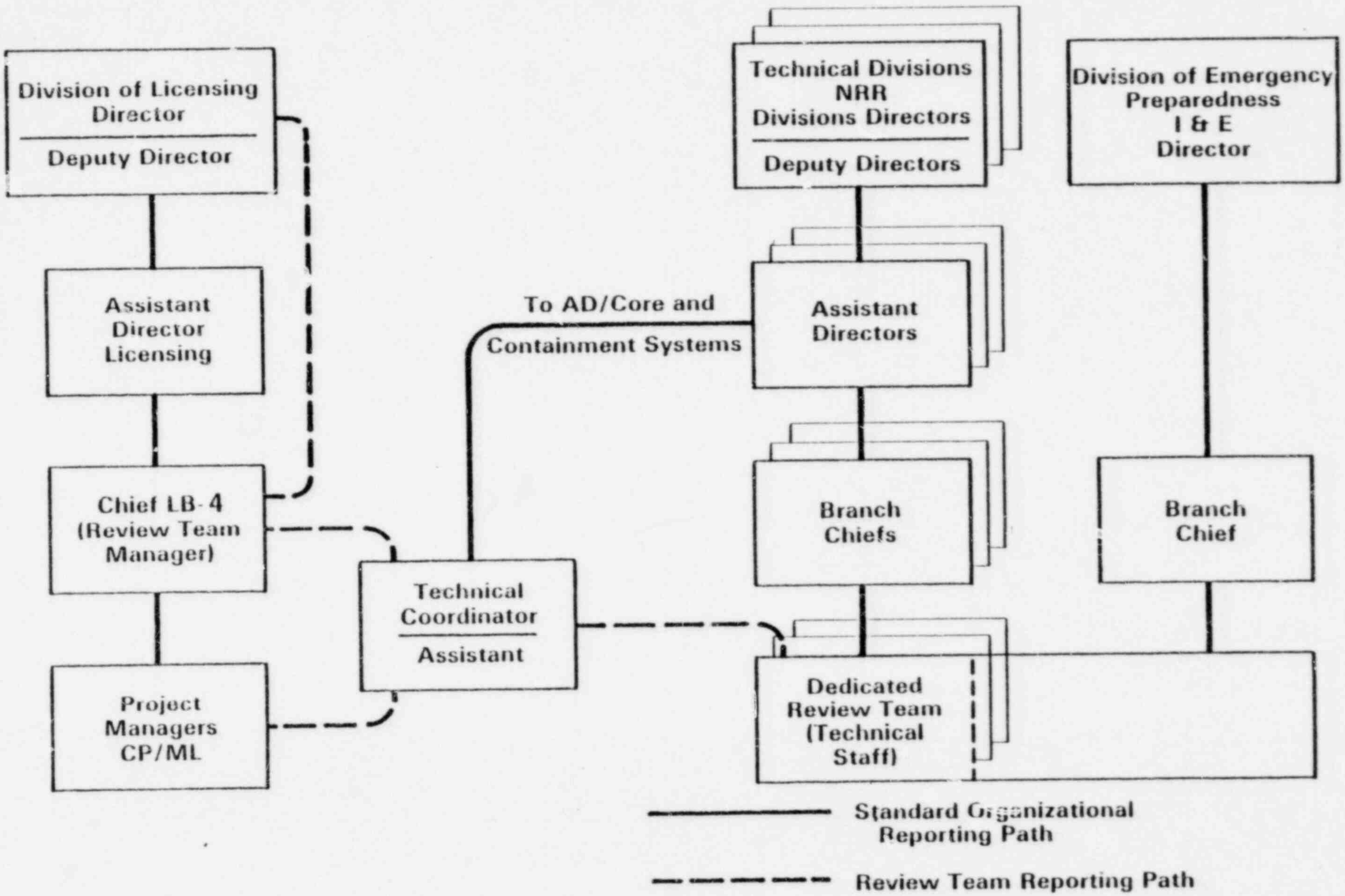
GPUN - Industry Participation

	GPUN
EPRI - Nuclear Power Div. Comm.	Member
• Safety & Analysis Task Force	Member
• Engineering & Operations Task Force	-
• System & Materials Task Force	
- Fuel Cycle	Member
- Plant Materials	Member
- Pressure Boundary Subcommittee	Member
• Owner Groups	
- BWR Pipe Cracking	Member
- PWR Safety and Relief Valve	Member
- Steam Generators	Member
• Code Development and Tech. Advisory Group	5 Members
B&W Owners - Executive	Member
• Operator Support	Member
• ATWS	Member
• Asymmetric LOCA Loads & Analysis	Member, Chairman
• Electrical Equip. Qual.	Member
• Inservice Inspections	Member
• Steam Generator	Member
• Reactor Pressure Vessel	Member, Chairman
• TMI-2 Follow up	Member

Hydrogen Control - Vented Containment

- Industry Generic Issue
- Issues are complex, regulatory positions being developed
- Work, Studies in progress
- GPUN has no special expertise, follow current Industry/NRC work

CP/ML REVIEW TEAM ORGANIZATION



CONDUCT OF REVIEW

MILESTONE	TIME
APPLICANT'S INITIAL SUBMITTAL RECEIVED	--
ADDITIONAL INFORMATION REQUIREMENTS IDENTIFIED BY REVIEWERS	SECOND WEEK
MEETINGS WITH APPLICANT, AS NEEDED, TO RESOLVE ISSUES	THIRD WEEK
APPLICANT'S REVISED SUBMITTAL RECEIVED	FOURTH WEEK
DRAFT SSER INPUTS TO TECHNICAL COORDINATOR FOR REVIEW	FOURTH WEEK
SSER INPUTS ISSUED TO DL	FIFTH WEEK
DL ISSUES SSER	APPROXIMATELY EIGHTH WEEK

NTCP PLANT REVIEW SCHEDULES

PLANT	SUBMITTAL RECEIVED	REVIEW COMPLETED	SSER ISSUED
PILGRIM 2	04/06/81C	05/08/81C	06/81C
ALLENS CREEK 1	05/01/81C	06/05/81C	07/81E
FNP 1 - 8	06/17/81C	07/17/81E	08/81E
SKAGIT 1 & 2	08/03/81E	09/04/81E	10/81E
BLACK FOX 1 & 2	11/02/81E	12/04/81E	01/82E
PEBBLE SPRINGS 1 & 2	N/S	N/S	N/S
PERKINS 1 - 3	N/S	N/S	N/S

BOSTON EDISON
PILGRIM STATION UNIT 2
ACRS MEETING AGENDA
JULY 10, 1981

· PROJECT STATUS & ORGANIZATION	R.M. BUTLER
· PROBABILISTIC RISK ASSESSMENT PROGRAM	J.W. ASHKAR
· ATWS CONSIDERATION	J.W. ASHKAR
· HYDROGEN CONTROL	R.E. JAGELS
· DEGRADED CORE RULEMAKING ACTIVITIES	R.E. JAGELS
· CONTROL ROOM DESIGN APPROACH	G.M. McHUGH
· INADEQUATE CORE COOLING INSTRUMENTATION	D.A. BRYANT
· SAFETY/RELIEF VALVE TESTING	D.A. BRYANT

STATUS OF PILGRIM #2

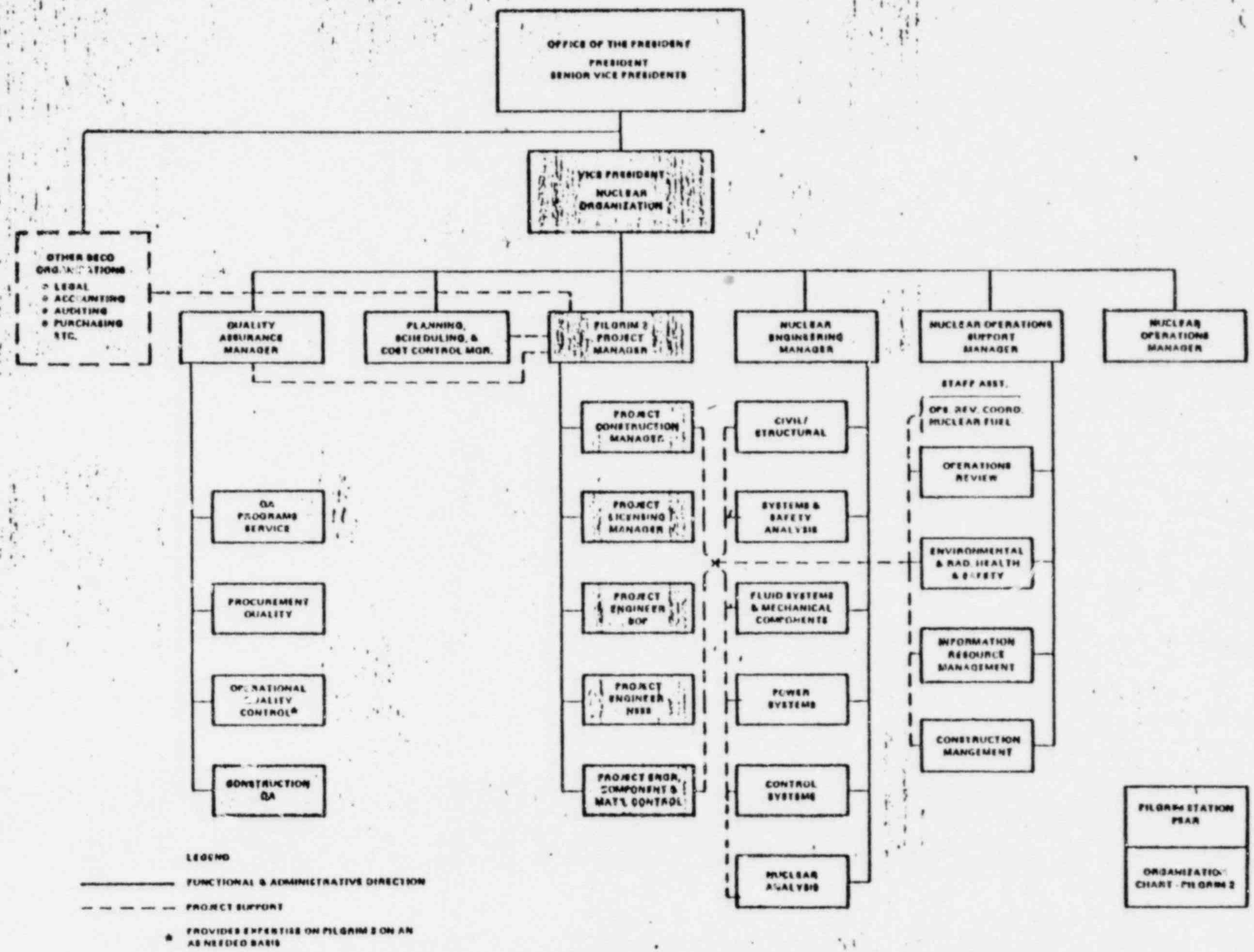
• MAJOR CONTRACTS	1972
• PSAR DOCKETED	12/73
• NRC SER	6/75
• ASLB HEARINGS STARTED	10/75
• ACRS MEETINGS/REPORTS	11/75 & 10/77
• NPDES PERMIT ISSUED	8/78
• ASLB PARTIAL INITIAL DECISION	2/81
• DRAFT NTCP RULE (NUREG-0718) - TMI	3/81
• PSAR AM. #43 - TMI	4/81-5/81
• NRC SER SUPPLEMENT - TMI	6/81

• DESIGN COMPLETION	
NSSS	90%
BALANCE OF PLANT	65%
TURBINE GENERATOR	100%
• DELIVERED EQUIPMENT* IN STORAGE	$175 MILLION
• TOTAL EXPENDITURES TO DATE	$300 MILLION

*INCLUDES: RPV, SG, RCP, T/G, PRESSURIZER

ACRS S/C
RMB-1

RMB-2



PILGRIM STATION PSAR

ORGANIZATION CHART - PILGRIM 2

PILGRIM STATION UNIT #2

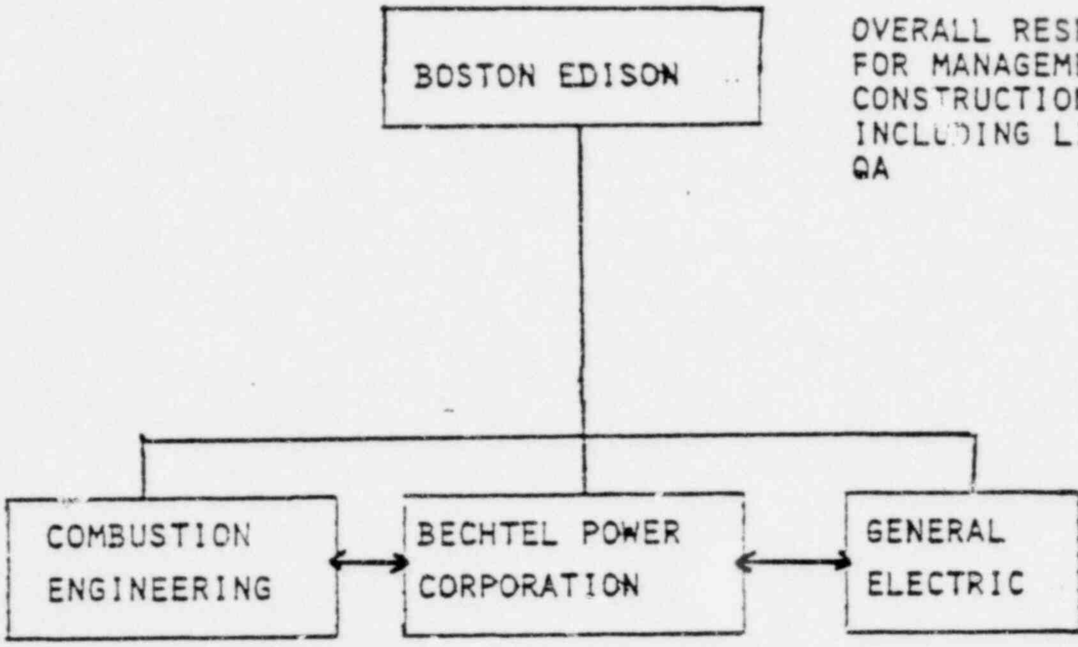
MANPOWER ESTIMATE DURING CONSTRUCTION

(IN EQUIVALENT NUMBER OF MEN)

	START OF CONSTRUCTION	YR. 1	YR. 2	YR. 3	YR. 4	YR. 5	COMMERCIAL OPERATION
NUCLEAR OPERATIONS	0	4	40	80	120	180	200
NUCLEAR OPERATIONS SUPPORT	2	2	3	3	3	3	4
NUCLEAR ENGINEERING	21	24	24	19	20	19	18
PILGRIM 2 PROJECT	10	22	22	22	22	22	12
QUALITY ASSURANCE	4	6	7	8	9	9	8
PLANNING, SCHEDULING & COST CONTROL	2	3	3	3	3	3	2
TOTAL NUCLEAR ORGANIZATION PERSONNEL SUPPORTING PILGRIM UNIT #2	39	61	99	135	177	236	244

ACRS S/C
RMB-3

PILGRIM 2 - ORGANIZATION



OVERALL RESPONSIBILITY FOR MANAGEMENT, DESIGN, CONSTRUCTION, OPERATIONS INCLUDING LICENSING & QA

COMBUSTION ENGINEERING

- NSSS DESIGN & FABRICATION
- NSSS QA/QC
- INITIAL CORE FABRICATION

BECHTEL POWER CORPORATION

- BALANCE OF PLANT (BOP) DESIGN, PROCUREMENT & CONSTRUCTION
- INTERFACE CONTROL
- BOP QA/QC

GENERAL ELECTRIC

- TURBINE/GENERATOR DESIGN & FABRICATION

PRA PROGRAM
SUMMARY

- PLANT/SITE-SPECIFIC PRA
    - Objectives
    - Organization
    - Program Flow
    - Key Elements
    - PRA Report

PRA PROGRAM
OBJECTIVES

A) Seek Design Improvements To Increase Core And Containment Heat Removal Reliability

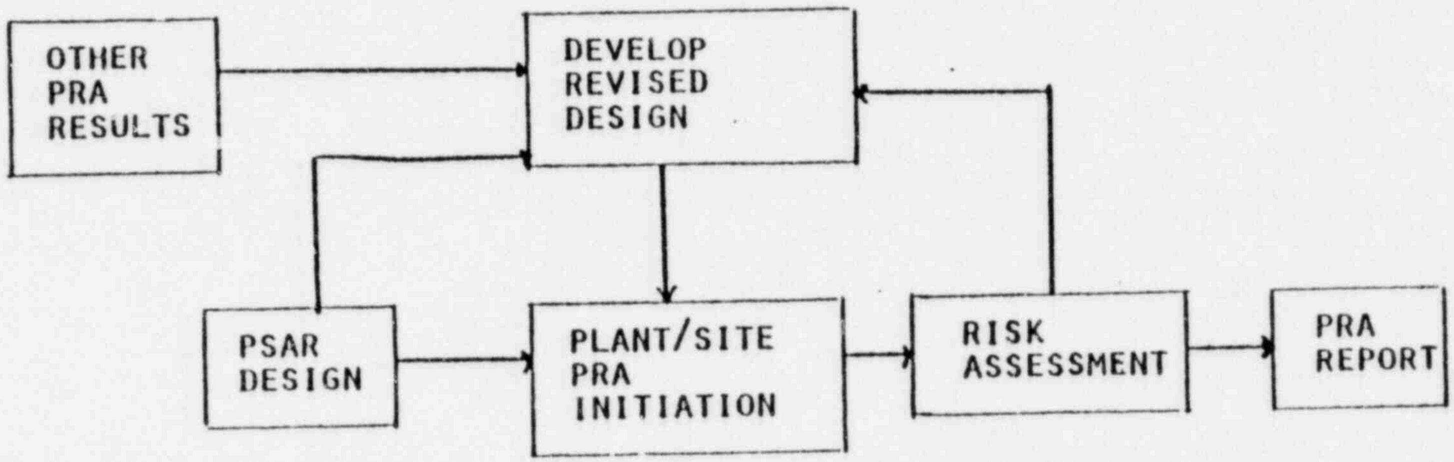
- Significant Risk Reduction
- Demonstrated Engineering Technology
- Acceptable Cost And Schedule Impact

B) Quantify Merits Of Design Improvements

PRA PROGRAM
ORGANIZATION

- PRA ACTIVITIES - EXPERIENCED, HIGHLY QUALIFIED
- DESIGN ALTERNATES - DESIGN GROUPS PRESENTLY RESPONSIBLE
- INDEPENDENT REVIEW - SEPARATE SENIOR LEVEL OVERSIGHT
- DESIGN DECISIONS - CURRENT PROJECT MANAGEMENT TEAM

PRA PROGRAM FLOW



PRA PROGARM - KEY ELEMENTS

- Preliminary Analysis - "Master Logic Tree"
- Plant Event Sequences - Allowable Plant Operating Modes
- Data Base - Verify Assumptions
- Externally Caused Failures - Frequency & Consequences Quantified
- Key System Fault Trees
  - including support systems (common mode failure)
  - human interaction
  - environmental effects
- Treatment of Uncertainty
- Feedback Into Design
  - outlier identification
  - critical items list
  - "importance to safety" 10CFR50, App. B
  - multiple failure considerations

PRA REPORT

- Probabilistic Risk Assessment
  - METHODS
  - ASSUMPTIONS
  - DATA SOURCES
  - TYPICAL ANALYSIS
  - TREATMENT OF UNCERTAINTY
- Results
  - CORE MELT FREQUENCY
  - CONTAINMENT FAILURE FREQUENCY
  - DOMINANT SEQUENCES
  - RISK CURVES - HEALTH EFFECTS
  - SYSTEM RELIABILITY IMPROVEMENTS
- Future Applications

PILGRIM 2 CONTAINMENT
(1200 MWe PWR)

- PRESTRESSED CONCRETE
- FREE VOLUME: 2.5 X 10^6 CU. FT.
- DESIGN PRESSURE: 60 PSIG
- TEST PRESSURE: 69 PSIG

DEGRADED CORE HYDROGEN REQUIREMENTS

(NUREG-0718)

- 100% METAL WATER REACTION
- H_2 CONTROL TO LIMIT H_2 TO $< 10\%$
- ASME III DIV 2, CC-3720; 45 PSIG MIN.

PRELIMINARY ANALYSIS

ASSUMPTIONS

- H_2 CONTROL METHOD - IGNITERS
- STEAM DILUTION EFFECTS CONSIDERED
- HEAT SINKS CONSIDERED
- ONE SPRAY TRAIN

RESULTS

- CONTAINMENT PRESSURE $< P_{TEST}$ (69 PSIG)
- CONTAINMENT LINER TEMP $< T_{DESIGN}$

HYDROGEN CONTROL COMMITTMENTS

- INCORPORATE INDUSTRY AND NRC RESEARCH/TESTING
- ADDITIONAL ANALYSES
- SUBMIT H_2 CONTROL DESIGN TO NRC 2 YRS POST-CP

POST-ACCIDENT SAMPLING

- SAMPLE STATION ACCESSIBLE WITHIN 1 HOUR
- POST-ACCIDENT SAMPLES FROM:
  - REACTOR COOLANT SYSTEM
  - CONTAINMENT SUMP
  - CONTAINMENT ATMOSPHERE
- SHIELDING OF SAMPLE/ANALYSIS FACILITIES TO LIMIT DOSES TO $<$ GDC 19
- POST-TMI CHANGES:
  - SHIELDING
  - SYSTEM ARRANGEMENT

DEGRADED CORE CONSIDERATIONS

- HYDROGEN CONTROL SYSTEM
- 3 FT SPARE CONTAINMENT PENETRATION
- PLANT SPECIFIC PRA
- MONITOR INDUSTRY/NRC EFFORTS
- PARTICIPATE IN RULEMAKING

CONTROL ROOM DESIGN APPROACH

SUMMARY

- REVISED APPROACH
- CURRENT STATUS
- PRE-TMI WORK
- POST-TMI APPROACH

CONTROL ROOM DESIGN APPROACH

PRE-TMI WORK

- Safety Sequence Analysis
- Bechtel/CE Proposed Layouts
- Review Operation Of Plant Systems
- Human Factors Guidelines
- Preliminary Layouts

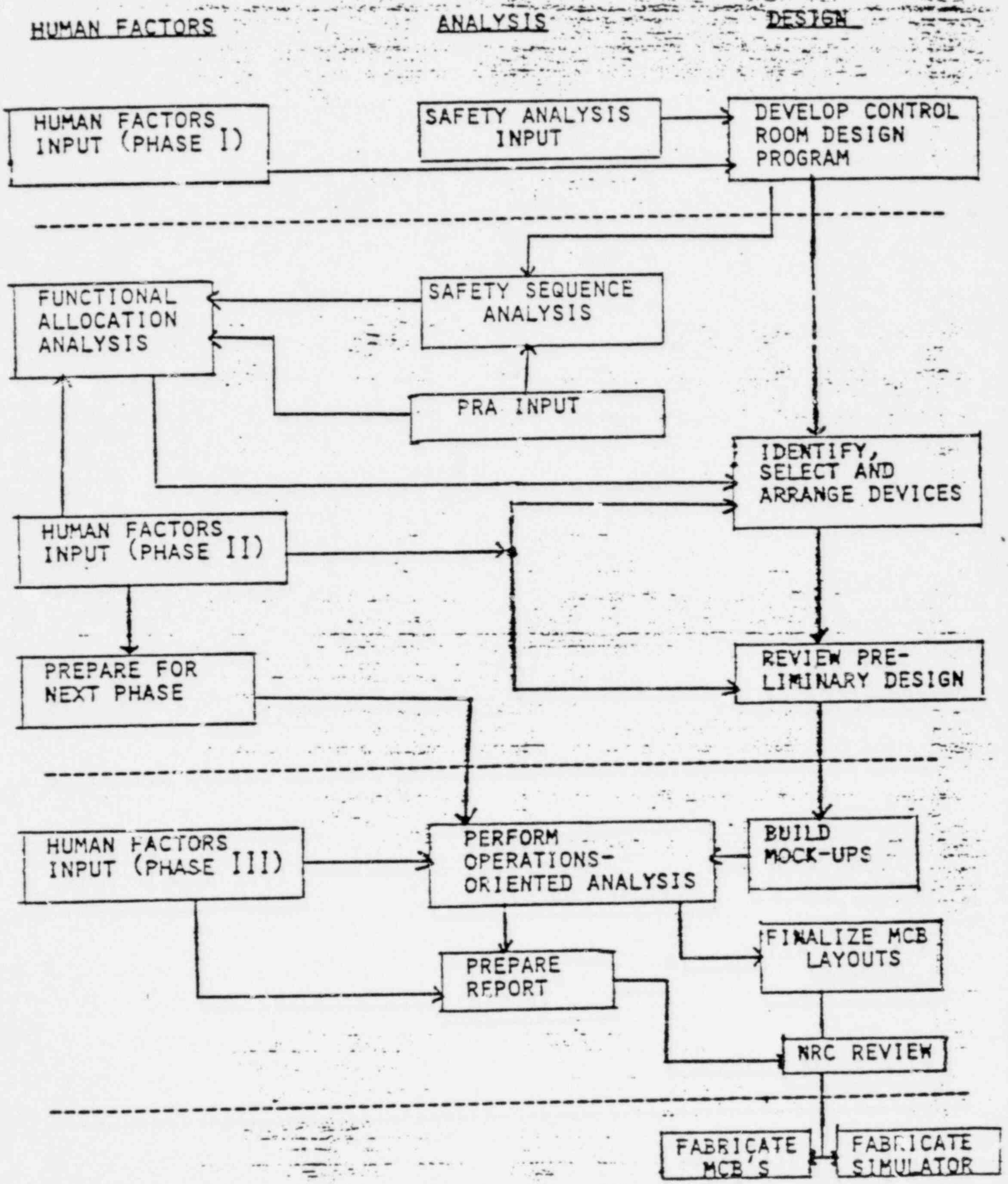
CONTROL ROOM DESIGN APPROACH

POST-TMI APPROACH

- Initial Activities
  - Data Collection
  - Program Formulation
- Implementation
  - Update SSD's & PRA Input
  - Provide Direct Human Factors Input
  - Conduct Operations-Oriented Analysis
- Team Approach
  - Program Development
  - Analyses
  - Revised Preliminary Design Development
  - Design Reviews
- NRC Review Prior to Fabrication

CONTROL ROOM DESIGN APPROACH

POST-TMI PROGRAM FLOW



PILGRIM 2 RESPONSE TO II.F.2 REQUIREMENTS FOR ICCI

- A PRIMARY COOLANT SATURATION METER WILL BE PROVIDED
- AN INVESTIGATIVE STUDY TO IDENTIFY APPROPRIATE ADDITIONAL ICCI WILL BE PERFORMED
    - RESULTS OF HJTC PROTOTYPICAL TEST
    - RESULTS OF APPLICABILITY TO PILGRIM 2 OF C-E GENERIC RESULTS

BOSTON EDISON COMPANY

PLAN FOR TESTING OF

SAFETY AND RELIEF VALVES

AND DESIGN OF PIPING AND SUPPORTS

FOR PILGRIM UNIT #2

SUMMARY OF COMMITMENT

PILGRIM 2 WILL:

- IMPLEMENT RESULTS OF INDUSTRY (EPRI) TESTS ON RELIEF, SAFETY, PORV BLOCK VALVES
- DESIGN DISCHARGE PIPING AND SUPPORTS FOR ALL LOADS FROM DESIGN BASIS TRANSIENTS AND ACCIDENTS

SUMMARY OF EPRI PROGRAM

- Designed To Comply With NUREG-0737 Item II.D.1 (PWR's)
- Testing 10 PORV's And 9 Safety Valves (Including Exact Pilgrim 2 Valves)
- Envelope Conditions On All Plants Represented
- Testing Underway Now
- Final Output By July 1982
- PORV Block Valve Program Under Development

EPRI PWR SAFETY AND RELIEF VALVE TEST PROGRAM

SUMMARY OF FOUR PRINCIPAL OUTPUTS:

1. SAFETY AND RELIEF VALVE TEST REPORTS FROM THE MARSHALL, WYLE, AND CE VALVE TEST PROGRAM.

2. A REPORT DOCUMENTING THE SELECTION OF THE RELIEF AND SAFETY VALVES TO BE TESTED AND JUSTIFYING THE APPLICABILITY OF THE TEST RESULTS TO VALVES UTILIZED IN OPERATING PLANTS AND PLANTS UNDER CONSTRUCTION.

3. REPORTS PROVIDING JUSTIFICATION OF THE SET OF GENERICALLY LIMITING FLUID CONDITIONS UTILIZED TO ESTABLISH THE VALVE TEST CONDITIONS BASED ON EVALUATION OF PLANT FSARs. JUSTIFICATION FOR OTHER TEST PARAMETERS WILL BE PROVIDED INCLUDING JUSTIFICATION THAT THE PRINCIPAL EFFECTS OF AS-BUILT RELIEF AND SAFETY VALVE DISCHARGE PIPING ON VALVE OPERABILITY HAVE BEEN ACCOUNTED FOR.

4. A REPORT DOCUMENTING A CODE FOR COMPUTING HYDRODYNAMIC LOADS FOR RELIEF AND SAFETY VALVE DISCHARGE PIPING UNDER STEAM AND WATER CONDITIONS.

NOTE: EPRI SLIDE

SUMMARY

- EPRI Program Underway On PORV's, Safety Valves
- Pilgrim 2 Valves Are Included In EPRI Program
- Pilgrim 2 Will Utilize Results As They Become Available
- Block Valve Program Under Development