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270TH GENERAL MEETING

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
270TH GENERAL MEETING

- - -

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

Thursday, October 7, 1982

The Committee met, pursuant to notice, at 8:30 a.m., PAUL G. SHEWMON, Chairman of the Committee, presiding.

PRESENT:

ACRS MEMBERS:

PAUL G. SHEWMON, Chairman
JEREMIAH J. RAY, Vice Chairman
J. CARSON MARK
MILTON S. PLESSET
CHESTER P. SIESS
ROBERT C. AXTMANN
DADE W. MOELLER
MYER BENDER
WILLIAM KERR
MAX W. CARBON
FORREST J. REMICK
DAVID A. WARD
JESSE C. EBERSOLE
HAROLD W. LEWIS
DAVID OKRENT

M. NORMAN SCHWARTZ,
ACRS Professional Secretary

RAYMOND F. FRALEY,
Designated Federal Employee

ALSO PRESENT:

R. AULUCK
T. DODS
R. M. NELSON
D. W. MAZUR
W. C. BIBB
R. G. MATLOCK
J. D. MARTIN
C. M. POWERS
D. L. RENBERGER
J. W. KLINGEHOEFER
C. MORGAN
MR. JONES
E. DIETERICH
M. FLEISHMAN
B. SHERON
E. JORDAN
P. POLK
R. MATTSON
MR. FROEHLICK
MR. ROSENTHAL
MR. BOUCHEY
D. TIMMONS
T. NOVAK
A. SCHWENCER
MR. HODGES
MR. POWERS
MR. KOBICKI
MR. GASTON

P R O C E E D I N G S

MR. SHEWMON: Good morning.

This is the first day for you people, the third day for us, of the regular 270th meeting of the ACRS. As soon as I get my notes here, I'll tell you the rest.

During the meeting we will hear reports and discuss the WPPSS, often known as the WPPSS Nuclear Plant 2, the modeling of small break LOCA's, and reactor pressure vessel thermal shock, PTS, also anticipated ACRS activities.

The items to be discussed on Friday and Saturday will be listed on the schedule for the meeting which is posted on the bulletin board in the back of the room.

The meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act and the Government in the Sunshine Act. Portions of the meeting may be closed as necessary to discuss proprietary or otherwise privileged information. Mr. Raymond Fraley on my right is the Designated Federal Employee for this portion of the meeting.

A transcript is being kept, and you would help the young lady a great deal if you would announce who you are and speak loud enough so that she can hear you without getting red in the face.

We have received no written statements or requests to make oral statements from members of the public regarding today's meeting.

I guess almost everybody here now was here for the meeting with the RSK for the last day.

That went well. One item that may be of interest or will be of interest to all of you is that I expect and plan to finish the activities of this meeting by Friday evening. I hope we can keep a quorum large enough to do that. The California contingents are quitting the middle of Friday afternoon, so we won't be helped greatly, although Milt will leave his two letters to finish up after he's gone.

(Laughter.)

So what I could do with that, if anybody has letters for this meeting I would like to have them tonight. I hope we will stay on a little bit tonight and try to get over the reading of at least the WPPSS letter. And since Mike may be bringing in the other one Friday morning, I'm not sure we can do PTS, but Milt's I would like to get a first reading of today.

Now, this item on the SER-SAR -- that's not in the folder any more?

MR. FRALEY: Well, it's been passed out.

There is a handout with your folders which is not an item scheduled for discussion as part of this meeting, but it has to do with Staff activities toward the development of revised SER's and SAR's. If you recall, some time ago you designated Mr. Gasky as your representative for this negotiation and we are now in the middle of it.

He would appreciate any comments you care to make regarding the material in this handout. It's called "Improved SAR's and SER's". It won't be discussed around the table, but please give him your comments as individuals and he will try to factor them in.

MR. SHEWMON: One other item. On the SAR's, there are some suggestions on changing those. I would like to request that if you have some opinions on how they might be made more useful to one and all, to please mention it to Ray or to me or to Myer.

Mike drafted for us an earlier letter that we sent out, and the Commission has come back with a letter from the EDO which is in your handout. We've been discussing just what it is we want to send back up. I would appreciate your input on that.

MR. FRALEY: I was just going to say that the Staff is moving toward trying the new improved SER, but the work on the new improved, revised SAR format is kind of hung up on type.

MR. SIESS: I head you say "revised and improved". I didn't hear "shorter".

MR. SHEWMON: If you think along with many of us, that's what they mean.

MR. SIESS: I would like to think that that's what they think also.

MR. SHEWMON: Maybe we can convince them with your winning ways.

Now, one other item, which I will try to remind you of tomorrow morning, there is a fire drill here Friday after lunch and anybody caught in the halls will be ordered down the stairs and out. But this room will be a sanctuary. So if you don't show up Friday afternoon and try to work someplace, you will be part of the fire drill. So keep that in mind.

MR. KERR: What happens if there really is a fire during that period? Do we stay here?

MR. SHEWMON: Our executive secretary will take care of us.

I think that completes my announcements, and with that I will turn it over to Dr. Plesset to take care of the WPPSS presentation, or lead us through it.

MR. PLESSET: All right, thank you, Paul.

We will get agendas for that shortly. Well, let me just go on anyway. That will be passed around in a minute.

The WPPSS 2 Subcommittee met September the 2nd and 3rd. The morning of the 2nd was a tour of the facility. The afternoon and the following day was a meeting with the Applicant and the Staff.

You've got it, Ray. We're all supposed to have one.

MR. FRALEY: I'll pass it around.

MR. PLESSET: As you know, the plant has been the subject of a lot of delays and difficulty, so some time was spent discussing the organization and the schedule. But we seem to be moving along rather well now.

Is it all right now? You must be alarmed at the size of this book being passed around. Our part will be just the first portion of it. The rest of it includes what we went through at the Subcommittee meeting.

The members of the Subcommittee were Jesse Ebersole, Jerry Ray, and Dave Ward, Carson Mark; and we had consultants present, Ivan Catton and Walt Lipinski. Dr. Lipinski is here again today, for reasons that will become clear in a moment.

We had some discussion of the open issues. One was just mentioned casually during our Subcommittee meeting and had to do with the cable distribution system. But this seems to be more of an item that we had anticipated, and at our request Walt Lipinski met with the Staff and the Applicant on this a couple of days ago, and he will be available to answer any questions you might have regarding the cable distribution problem.

I don't think I need to give any more in the way of a general introduction, except to call on the other members of the Subcommittee to see if they would like to add anything. Carson?

MR. MARK: No.

MR. PLESSET: Jerry?

MR. RAY: No.

MR. PLESSET: Dave Ward?

MR. WARD: No.

MR. PLESSET: Jesse?

MR. EBERSOLE: I thought one of the more interesting aspects of the meeting was the disclosure that the post-trip power was substantially higher than the other BWRs that we had heard about, which leads to maybe a new look at the ATWS problem. As you know, in a BWR ATWS the critical thing is: What are we going to do in the brief interval of time after we have tripped the pumps? And how much time is that? We're already down to just a few minutes on some of the older plants, and here with the substantial increase in power I'm not sure but what our time is really cut to an inadequate level.

MR. PLESSET: I think we can get a little statement on that a little later from Walt, who went to a meeting on this subject. So he is here for a double purpose, now that you reminded me of that.

That brings up another point. It was apparent to us at the meeting that the comparison of the characteristics of this BWR-5 Mark II with other plants, BWR-5 had a table which was not right. They were supposed to send us another version of that, but I didn't get one. Did anybody else get one?

MR. SCHWENCER: It's in the handout that the project manager will hand out.

MR. PLESSET: Oh, we did get one just a few seconds ago. But that is a good point that Jesse mentioned and I think we should come back to it at the proper time.

Yes, Carson?

MR. MARK: It was also, I believe, observed, I guess at the meeting, that although the estimates for loss of offsite power, for example, are once every $100 million years, something of that sort, the estimates for earthquakes which would shake down the poles were once in 10^4 years.

MR. PLESSET: There was some delay in getting to our Subcommittee meeting because of the geoscience review. As far as I can tell, it seemed to be a reasonable situation.

As you know, the site is a very good one as far as population distribution goes. It is very low. You might want to keep this in mind.

Well, I think we might go to the Staff. Who will speak for the Staff?

MR. AULUCK: I will. Good morning. My name is Roger Auluck. I'm the project manager for the WPPSS 2 facility. It's a BWR-5 Mark II containment. It's a freestanding steam containment enclosed in a reinforced concrete biological shield wall.

Before I discuss the status of the outstanding issues, I will briefly go through some of the important dates on this project.

(Slide.)

In August 1971, Washington Public Power Supply System filed an application to construct and operate this proposed Project No. 2. The Commission reported the results of its preconstruction review prior to construction in a safety evaluation report in September 1972. The construction permit, CP-PR-93, was issued in March 1973.

In March of 1977, the Supply System tendered an application for the operating license for WNP-2. After the acceptance review, the application was docketed in June of '78. The final environmental statement for the OL stage was issued in December of 1981. The safety evaluation report for the OL stage was issued in March of 1982. The first supplement on this project was issued in August of '82, and we had the Subcommittee meeting in Richland, Washington, the 2nd and 3rd of last month.

Applicant's estimated fuel load date is September of '83.

(Slide.)

Now I will go over the outstanding issues. There were 16 or 17 of them. We have received information on about five of these issues and they are currently under review. Item 21 is resolved, so that will not have to be discussed. I will go over each one of these issues one by one.

(Slide.)

Internally generated missiles: The FSAR did not identify all the missile sources and their effects inside as well as outside of the containment. Applicant is proposing to submit the final report some time this month, so we will address this in a later supplement.

(Slide.)

Tornado missile protection for diesel generator exhaust: The Applicant believes that the probability of a tornado of sufficient velocity to lift large, heavy missiles almost a thousand feet to half a mile away and plug the diesels is extremely small. We are reviewing the Applicant's position in this area and will report the results in the supplement.

(Slide.)

Turbine missiles: In view of the current experience, the Staff proposes to emphasize the turbine missile generation probability in our reviews of that turbine missile issue, and eliminate the need for analysis of strike and damage probabilities, given an assumed turbine failure rate. The Staff is in the process of revising the appropriate regulatory guide and the branch technical position and the standard review plan.

MR. MARK: On issue number 2, turbine missiles or tornadoes, what is the significance of that thousand feet away?

MR. AULUCK: The requirements were they have to design for missiles from the height of 30 feet above the plant grade. The 1,000 feet is the first hill, which is about 30 feet below the level of the tornado exhaust elevation.

MR. MARK: So that's the nearest place from which a missile could get to that height and reach the target?

MR. AULUCK: Yes.

MR. EBERSOLE: May I ask a question about this topic. I believe it was at this project that we suggested that the Staff look at the electrohydraulic or other systems that controls the overspeed potential for the plant in the context that it strongly resembles the GE scram dump volume problem. The fundamental way this works is by zero energy dumping of hydraulic pressure, which then actuates certain functions to trip the main steam -- the stop valves.

I believe there are the usual residual problems -- the volume into which they dump the oil guaranteed to be empty, analogous to the scram dump volume problem. And in short what can one do to examine the potential chance of having a 180-percent missile with its orientation the way it is, which would then be a very potent missile indeed?

It may be you are already analyzing for the

180 percent missile velocity. I don't believe it was clear at the time of the meeting whether you are doing this.

MR. AULUCK: Yes. We have John Shifkins from the Staff who can deal with that.

MR. EBERSOLE: The similarity between the dump volume case on the scram system and the electrohydraulic dump system is rather striking. I think it bears some investigation.

MR. AULUCK: Yes, I think we have a person here to address that later on in the presentation.

MR. MOELLER: Item number 2 again. You are concerned with the exhausts. Is there any concern about the diesel intakes?

MR. AULUCK: No.

MR. MOELLER: They're protected.

MR. AULUCK: They're protected. They are on the lower elevation.

(Slide.)

The next one is electrical equipment qualification: We have received the requested information on environmental qualification and seismic, and we are proposing to do our environmental audit towards the end of this month and the SQRT audit in November.

MR. SHEWMON: Mr. Auluck, my guess is that if you go through all two dozens of these things and give us a chance to comment on them, that we are going to take a long time. Could you go to the ones where there has been a change since the Subcommittee meeting or where you think there is particular difficulty in reaching the solution?

MR. PLESSET: Thank you, Mr. Chairman.

MR. AULUCK: That will cut short my presentation quite a bit, and I will be happy to do that.

MR. SHEWMON: We'll find other questions to fill it, don't worry.

(Laughter.)

(Slide.)

MR. AULUCK: Essentially there has been -- let me put the list back -- since the Subcommittee meeting there is no change, except item 21 is resolved, and we have added -- well, we did mention cable separation criteria at the Subcommittee meeting, and I can address that a little bit more if you'd like.

(Slide.)

Based on NRR and Region V concerns on cable separation criteria and implementation of these criteria, the Applicant has recently established a task force to deal with these concerns. This arose during a Region V routine inspection of the facility. What they saw was, it is very difficult to verify what is on the plant. It's not that it is not in conformance with the criteria, but it is difficult to verify. And that's how our dialogue started with the Applicant.

To do that, the Applicant has formed a task force, and as we understand one of the items that the task force is looking at is the preparation of the electrical separation design guide document. What that will do is step by step articulate how the separation criteria were translated into installed cables. This should help them as well as us in walking through the plant and tracing the cables.

We had a meeting yesterday in that direction and we are still working on it. The Applicant hopes to complete the guide by the first week of November. So our next item will be after we receive the guide or any more information.

MR. EBERSOLE: Reg Guide 1.75 provides a nominal sort of a protection from electrically induced fires into electrically operated equipment. It does not provide for exposure fires, although there has been created Appendix R, which to a large extent will fix the errors that might occur.

In the course of your walkdown examination, do you carefully verify that the requirements of Appendix R for independence from the normal electrical system, which is separated, is in fact in place and we have an appropriate degree of independence for the Appendix R functions? Is that a part of this walkdown?

MR. AULUCK: Yes. Well, this plant is before the 1974 docketed plants, so they do not have to meet Reg Guide 1.75 criteria.

MR. EBERSOLE: Well, that even makes it more important that the Appendix R requirements be better than ordinary.

MR. AULUCK: They do meet Appendix R requirements for fire protection.

MR. EBERSOLE: You do validate that in this kind of a walkdown? Is that correct?

MR. AULUCK: That will be in this walkdown.

MR. NOVAK: This is Tom Novak. I think we ought to at least clarify that they are separate, in my mind at least. We have Region V people here today who can answer questions regarding specific inspections with regard to cable separation.

There is a separate activity. There is a separate inspection module that you can apply to the plant in terms of inspecting it against Appendix R. I think there is a lot of similarity, but I think it's handled separately.

MR. RAY: It's confusing to me that at this stage of the game it's necessary to take five paces back and prepare a memorandum recording how things were installed. It seems to me that instructions to the contrary as to how to install them initially and installation records should guide the NRC inspectors. Are these absent?

MR. AULUCK: No. Usually the NRC inspector looks at the FSAR and he is able to perform his inspection. But here the FSAR was very confusing and could be misunderstood.

MR. NOVAK: Mr. Sternberg of Region V is here, who is responsible for the supervision of those inspections. He's going to make some comments later and will address the inspection activity.

There is no specific inspection module that I know of that an inspector would use to inspect for cable separation.

MR. RAY: Tom, I guess what I'm really asking, is such a document for this purpose necessary in every plant? Has it been?

MR. NOVAK: Dan, did you hear him?

MR. RAY: Has such a document prepared by applicant been necessary in every project in the past?

MR. STERNBERG: No, it was not prepared.

MR. SHEWMON: Please identify yourself and use the mike.

MR. STERNBERG: My name is Dan Sternberg of the Reactor Projects Branch in Region V. Such a document has not been generally required at other facilities.

MR. RAY: It still eludes me as to why in this specific case it is necessary. Will you enlighten me on that? Or will you when you have your opportunity at the podium?

MR. STERNBERG: I'd be glad to take the time now, sir.

MR. PLESSET: Make it short if you can.

MR. STERNBERG: Normally what is inspected against is the FSAR. It's normally a somewhat straightforward verification, a review of what the Licensee has done to confirm that the plant is built in compliance with the FSAR, and then a selected sample is looked at independently by our inspector.

When that was attempted at WNP No. 2, we could not confirm that the Licensee had done such a verification and we were unable to do it on our own because the criteria appeared to be somewhat muddled.

When we sought clarification from the Applicant, we ran into the same problem. They appeared to be equally unclear as to what the requirements were.

So we surfaced the issue back to NRR. The FSAR was revised. The revised FSAR chapter still lacks specificity that would enable us to do a complete indepth inspection, and that is pretty much what prompted the meeting that we held yesterday.

MR. EBERSOLE: May I ask a question? I'm beginning to get the notion that the FSAR is used as a guide document for examination of critical details, and my impression has always been that it is a highly generalized and a totally inadequate document to run an investigation, typical of investigating acquiescence to Reg. Guide 1.75.

Surely you must have better documentation, including table schedules, which trace the individual history of every cable, where it went through the penetration, down which trays, et cetera, et cetera, far beyond the capacity of the SAR. As a matter of fact, there is as much documentation in that as there is in the whole FSAR. Don't you have these papers?

MR. STERNBERG: Yes, sir. They are utilized. The FSAR is the overall guidance that specifies what general ground rules are used. Then you use the detailed items you spoke of to conduct the in-depth inspection.

We couldn't get from the FSAR to anything in the detailed documents that clearly indicated what cables should be in what trays and which ones shouldn't be in such trays, and that's basically the area. It's a somewhat complex term, but it's not so much related to the separation between the divisions, the safety divisions, but a term called "associated circuits," which is discussed in Reg Guide 1.75. That pretty much was the focal point of all of this discussion.

MR. EBERSOLE: The associated circuits?

MR. STERNBERG: Yes, sir.

MR. MOELLER: In the list of outstanding issues, you have item 23, the control room design review. I realize of course that that is directed primarily to human factors review. Did the Staff take into consideration in its review of the control room for the plant a recent letter that the ACRS submitted to the Chairman on the subject of control room habitability?

MR. AULUCK: I think so, but we have Dick Froehlick from Human Factors here.

MR. MOELLER: I know the Subcommittee discussed it, but I'm curious as to how much the Subcommittee actually used the report.

MR. FROELICH: Dick Fralick.

No, we do not address as part of the human factors review control room habitability. That is generally done by the fire protection group.

MR. MOELLER: I realize that you don't generally, but did anyone do it in this case? Did they take the Committee's report and go down it item by item and check out this particular plant's control room in comparison to the comments that the Committee made?

MR. FROELICH: Our branch has not done that. In fact, our review is not scheduled until April of 1983. We will probably do a preliminary review some time in late November or early December.

MR. MOELLER: Is there someone here from the Staff who can answer my question?

MR. NOVAK: I think I'll answer that for you. Certainly we will consider the Committee letter in the remaining review. I think we can respond in a separate memo back to the Committee in what areas this design satisfied those requirements, what areas it didn't, and identify the differences.

MR. MOELLER: That will be very useful.

MR. PLESSET: Go on, Mr. Auluck.

MR. AULUCK: The concern raised at the Subcommittee level was Table 1.2 of the SER. We have corrected that. This is in your backup handout.

(Slide.)

There is a minor change here. LaSalle's power level is the same as WNP-2's.

On page 2 --

(Slide.)

-- the old numbers for maximum heat flux and average heat flux were in error. They were taken out of the wrong document.

MR. PLESSET: Okay, go on.

MR. AULUCK: Those were the main items that the Subcommittee raised.

MR. PLESSET: Are there any other questions?

MR. AULUCK: I'd be happy to answer any other questions.

MR. SHEWMON: With Nine Mile Point having to replace all of its recirc lines now, several years in, I'm curious as to what this plant has done or how they met the requirements on controlling the stress corrosion cracking. Can you tell me whether they have got special stainless steel in their piping or have committed to inspect because they don't, or what?

MR. AULUCK: I think I would defer to the materials group.

MR. SHEWMON: We can wait until a little bit later. Maybe they can answer it. But I would like to hear a short discussion of it before the day is over, if I could.

MR. MOELLER: Mr. Chairman, in comparing this facility to other similar facilities, in several instances it's been compared to LaSalle. One of the ACRS Fellows extended the comparison into some other areas, including the gaseous rad waste system and the liquid rad waste system. The preliminary data or the data I have show a factor of 10^5 difference in the absorption coefficient for krypton and xenon and a factor of 5 difference in certain aspects of the liquid rad waste system, particularly with respect to chemical wastes.

I would be glad to give you a copy of this comparison, both to you and the Applicant. And if someone could, just briefly before the meeting is over, tell us why.

MR. AULUCK: We'd be glad to do that.

MR. MOELLER: Thank you.

MR. CARBON: Do you expect any particular difficulty in resolving these outstanding issues, or do you view them as fairly normal?

MR. AULUCK: Fairly normal. We're just waiting for the information.

MR. EBERSOLE: I think the full Committee might be interested in item number 10. They haven't heard what the issue really is. What was found is that the designs here have incorporated multiplexes in the root heat sink system for this plant, which is a rather unusual feature.

One might argue it's an unnecessary complexity and perhaps a complexity that contributes to unreliability. I think the Applicant will argue the opposite view. But that in essence is the issue. You might comment on where you stand on this evaluation.

MR. AULUCK: We are asking the Applicant to perform some kind of an electricomagnetic interference test, either in the lab or the field, or justify why they should not do these tests. We have a Staff member from the group to --

MR. KERR: I'm sorry? Did you say an electromagnetic interference test?

MR. AULUCK: Yes.

MR. EBERSOLE: The system could be ordinarily or electromagnetically controlled, Bill, but it's a fancy system. The need for such a relatively sophisticated system I guess is what is in question, along with its unreliability.

MR. KERR: Do we have some evidence that it is or is not unrealiable.

MR. EBERSOLE: No, I don't have anything, just the general knowledge that it is complex.

MR. KERR: Who is going to discuss that?

MR. AULUCK: Jack?

MR. ROSENTHAL: My name is Jack Rosenthal. I am in the Instrumentation and Control Systems Branch.

WNP-2 utilizes a multiplexer system for control of the service water system and the standby service water system. This is the first 1-E application of multiplexers of this nature at a nuclear power plant, although there are lots of non-1-E applications of the multiplexer.

We thought it appropriate to do an in-depth review of the application of this first of a kind. When we wrote the SER, we had not performed sufficient review, so we left it as an open item. Since that time we have received manuals by the vendor of the equipment on the level that we would normally receive. We have reviewed that documentation. We have met with the Licensee on two occasions. We've had information that's gone back and forth.

We had several concerns and those concerns are resolved, with the exception of the susceptibility of this -- of the electronics to electromagnetic interference, and we will pursue that with the Licensee.

MR. KERR: Is there something that would lead you to believe that this system is unusually susceptible to electromagnetic interference, as compared with other communications systems?

MR. ROSENTHAL: There's a 10-magahertz carrier and phase shift signals going back and forth on that carrier. You've got a very complex parity-checking scheme, horizontal, vertical checking back and forth. We think that there is some contention that the equipment would be more susceptible to EMI than the sort of hard-wired stuff that we normally see.

On the other hand, that very complex parity scheme connection tells them when their problem arises long before an accident occurs.

MR. KERR: Excuse me. I don't understand what you meant when you said there's some potential for, whatever the phrase was you used. Is there some reason that you think it is especially susceptible to EMI? Is there some experience someone has had or is it that you just don't know that it isn't?

MR. ROSENTHAL: This class of equipment, low signal level equipment TTL, C-Morse type logic, low level signal stuff, should be the sort of thing that's susceptible to EMI. In other areas where we have seen a computer-type application of this sort of electronic applications, we've requested licensees to do EMI testing and we've established thresholds for that testing, and we've attempted to ensure that that equipment would not be susceptible to that environment.

It is the same class of equipment as the analog parity units that are provided by General Electric at the front end of the reactor protective equipment. That was tested for EMI. The multiplexers were not. We had a similar concern at ANO Unit 2, and again that was tested for EMI.

So it is just an extension of that sort of testing this sort of electronics in a different application.

MR. KERR: I'm trying to understand why it is that testing is necessary if there have already been a number of tests that apparently have demonstrated that either it is or is not susceptible. Does each separate installation require a field test? I'm not trying to be critical. I'm just trying to understand what it is we are looking for.

MR. ROSENTHAL: In this case the packaging is very important to its susceptibility to radiated fields. In this case we would require the testing.

Where some vendors are now going towards generic testing which will be performed in a lab, we may be able to bound the susceptibility of that product line and not require individual testing.

MR. EBERSOLE: May I ask, did you raise the question fundamentally, why did you multiplex this? What advantage did you get versus what disadvantage did you accrue? What's the fundamental logic of using a sophisticated and delicate system against a rather primitive function, not fast in need of response, with rugged signals that you could use, crude equipment but effective? Why do we have all this lace refinery on a system that ought to be fundamentally simple and sound and immune from such wierd effects as transient spikes?

MR. ROSENTHAL: They did it to save money.

MR. EBERSOLE: That's the common answer, I think.

MR. SHEWMON: Are you against modern technology?

(Laughter.)

MR. EBERSOLE: Not when it always seems to uniformly lead to an inferior product.

MR. SHEWMON: Whether or not it is inferior remains to be seen.

MR. EBERSOLE: Well, I think there is some question about that.

MR. KERR: I would like to ask Dr. Lipinski if he has any comments on this issue.

MR. LIPINSKI: Effectively, what Mr. Rosenthal has said is precisely the case, because we're talking signals probably in the high volt level, 10-milliamp loop. The Applicant can correct me. I haven't looked at the details of their system.

But if you string these cables out over a long run and run them adjacent to the power cables, then you can induce pulses from the power cables into the signal cables, and it's very important, because here you're talking about a pulse system and any spurious pulse or pulses that are introduced can give you erroneous information at the receiving end. It is very important that you analyze the system and test it to make sure that it's not susceptible to an introduction of spurious pulses.

MR. ROSENTHAL: May I offer why we ultimately and somewhat reluctantly did accept the system? There are two considerations:

One, the system is on-line and is constantly tested. There's subtesting diagnostics, there's a very complicated parity arrangement. Signals are sent back and forth all the time. So it seemed to us that we should know when that system has failed. And it is still in its standby mode when it is self-revealing that it has failed, and that is a very nice feature of the more elaborate electronics signals.

The second thing is that the multiplexer is part of -- is a component of the standby service water system. There are tech specs on the standby service water system which are to some extent punitive. The Licensee is required to keep both trains of standby service water operable. These are relatively -- I don't remember the number of hours -- a small period of time in which one of the two can be inoperable, and then he's got to shut down.

MR. EBERSOLE: On that score, that is analogous to the airplane pilot who must land as soon as he feels his landing gear has failed.

MR. ROSENTHAL: Yes. But in this case you have a standby system which in a standby mode the multiplexer is self-revealing. You haven't had an initiating event that requires that that system operate, and yet that system is revealing should it be faulted.

So just to finish this thought, we thought that here was an application where, if the system was exhibiting particularly unreliable behavior, it would be self-revealing to the Applicant; and too, that the conditions for tolerating the unreliability are financially punitive to shut down the plant ultimately. Hence, should the equipment in operation prove unreliable there would be a large reason for changing the equipment out. And that was the second consideration.

MR. EBERSOLE: Well, from a practical point of view --

MR. PLESSET: Jesse, I think we're getting the picture pretty well. Let's let Walt make one final remark. I think we should move on.

MR. LIPINSKI: Reference was made to parity checking. If there's a single bit out of place, parity checks will detect this. But parity checks do not detect multiple bit failures. Their probability is lower, but they are still a source of a problem for such a system.

MR. PLESSET: I think we can go on, Mr. Chairman. Any other comments?

(No response.)

MR. PLESSET: Well then, we'll excuse you, Mr. Auluck.

MR. AULUCK: Thank you.

MR. PLESSET: I think our next presentation is from the Region V office.

MR. STERNBERG: I don't have any slides. My name is Dan Sternberg. I'm Chief of the Reactor Projects Branch in Region V with responsibility for both the operations and construction inspection program at WNP-2.

The complete construction inspection history was described to the Subcommittee in December. There have been no substantive or significant changes at all. What I would like to do is just give you a brief review of where we currently stand and what we perceive to be the problem.

The regional construction inspection program has essentially been completed at WNP-2. There are a few remaining inspection modules that we have yet to accomplish. They should be finished in the next two months and we don't really foresee any problems. That's the routine inspection that's done at all construction facilities.

There are, however, some outstanding items that have yet to be inspected and closed out. These open items normally are just a catch-all area of things that have been identified and yet not corrected. The most significant area that yet has to be inspected are what we call the 50.55(e) reports, or the construction deficiency reports.

These are submitted by the Supply System when things are identified during the construction program. Normally corrective actions are identified and then our inspectors review the corrective action to see that it was appropriately taken care of.

Right now we have about 60 of these reports that have yet to be closed out by our inspectors. Most of them are not significant in any way. Two of them have received a significant amount of both NRC and Applicant attention. They are what we call the anchor bolt and grout issue, where there was a problem with a large number of safety-related pipe supports and supports of other kinds and the anchor bolts associated with them.

The next one of those significant 50.55(e) reports centers on the electrical cable separation issue, primarily in the area of associated cables. We have had some problem in closing out 50.55(e) reports at WNP-2, in that final reports that we have received from the Applicant stating that actions have been completed or closed out or were planned to be closed out, when our inspectors looked at it in detail it turned out that really there was still some significant amounts of work to be done and that it really wasn't ready for closeout. That area or that particular problem is still being reviewed, and we are looking for an aggressive corrective action on the Applicant's part so we can close these items out or identify if there is any significant work that yet needs to be done.

One other significant open item relates to our verification of the reverification program --

MR. KERR: Excuse me. I guess I dont quite understand that last statement. If one puts it in plain English, are you saying you get reports saying that the problem has been corrected and it hasn't been?

MR. STERNBERG: Yes, sir.

MR. KERR: Is that because the people writing the reports don't know what's going on?

MR. STERNBERG: We've seen examples of that. We have also seen examples where it was stated that at some time in the future a program would be established to achieve some end goal, and it had been presumed apparently that that action was undertaken, when in reality when we got to look at it it hadn't yet been undertaken and the corrective action hadn't been completed. It is those types of things.

MR. KERR: So in a sense there might have been some ambiguity in what was being claimed, or at least some unsatisfactory English?

MR. STERNBERG: Yes. I'm not suggesting that there was any intentional falsification or misstatement. It's just that in these complex issues -- I might just take a moment to indicate one area where WNP-2 has represented a significant problem for our routine inspection program.

Historically, they have used multiple contractors to achieve a given functional area. For example, mechanical installation is normally done at a nuclear power plant by one contractor. At WNP-2 we ran into situations where there were three, four or more contractors in the electrical area and similar things. So that we have had to perform multiple inspections.

Normally, in for example safety-related welding, if we conduct an inspection program in welding we will review the procedures used, the qualifications of the welders, the NDT techniques used, and when we have done it then we can feel confident that all work done at the site that's safety-related is done in accordance with that type of system. Here we run into multiple ones and we have to go in and inspect each of the individual contractors.

It really has been a significant drain on our resources. Additionally, it tends to confuse the issue. We put it to bed with one contractor, and there appears to have been, at least in the past, a problem of communications, where a problem identified with one contractor did not get communicated to other contractors doing similar work in different areas of the site. So that's also been a source of some of the 50.55(e) problems.

As I say, the significant item back in the 1979 time frame related to problems with the lead electrical contractor. A stop work order was issued and as a result of that a verification program was undertaken by the Applicant to go back and look at all the work done to ensure that there were no problems. We have yet to complete our review of that reverification program and the results of that program. So that is probably in the top three of the open items that we still have to review.

Okay, let me just give you a few more pertinent things that are going on. Right now Region V, with the assistance of a group of people from our Region I office in Pennsylvania, have a special independent measurement van at the site doing an independent review of radiographs of welds. We're doing our own radiographs. We're reviewing radiographs that have had potential problems and questions raised.

This is a new part of our inspection program. It's designed to give us added assurance that both the

Licensee's records are adequate and that there are no serious problems. That review right now is in progress and we really do not have the results of that particular effort.

The preoperational test inspection program, which I am also responsible for, has not really gotten under way. It is not really possible for us to do it right now, in that the Licensee is not really doing very much in the preop area. There have been very few systems turned over from construction to the startup organization.

So right now I think what that translates to is there's a very ambitious schedule in front of us as inspectors over the next 9 to 12 months if we are indeed going to be able to review, as we normally do, the complete preoperational test program. So that is one item.

Then the last thing I would like to bring out is, in response to regionalization and other manpower constraints we in the region have reorganized to place both the construction and inspection program under one office. That is where I'm at right now, just having recently taken over the facility.

MR. PLESSET: Max, did you have a question?

MR. CARBON: Yes.

What was it that's at the heart of the anchor bolt and grouting issue? Just briefly, what is it?

MR. STERNBERG: The problems were identified by the Applicant and the corrective actions taken did not fully correct the problem, and subsequent problems in the same area occurred.

The grout issue was one where, under certain anchor plates there is an amount of grout that is installed as a load-bearing pad. The grout basically was crumbling. So there has been an aggressive campaign to try to identify all of the areas where that potential grout problem might occur.

The anchor bolts, they just basically said, we didn't have good control over the installation and quality of the anchor bolts and we have to replace, or at least inspect and replace all those requiring it. And that was a massive job and it didn't go particularly smoothly.

MR. PLESSET: Any other questions?

(No response.)

MR. PLESSET: I don't think there are any other questions, so we will go on. I believe we now go to the Applicant for his presentation. And Mr. Nelson, is that your understanding?

MR. NELSON: My name is Roger Nelson, I'm the Manager of Project Licensing for WNP-2. I guess the first thing I would like to do is follow up with what Dr. Plesset mentioned earlier about the handouts we ha- for you. It is the entire subcommittee presentation with the very first portion which is marked "Full Committee" is the only portion we'll be discussing today, which is basically a summary of the subcommittee work.

What I have is the agenda as I've laid out and the list of presenters that are going to give the summary work related to the subcommittee work. So I don't really have anything else to say except maybe answer a few questions. I think right now we could we could just get on with the business, and I would like to introduce our Director of Operations, Don Mazur.

MR. SHEWMON: Sir, before you sit down, where can I get an answer to questions about stress and material control in the stainless steel loops, and also, your ability to take care of oxygen? I'll do it anyplace on the program you tell me that I should put it.

MR. PLESSET: Do you have that now?

MR. NELSON: We are involved with a lot of work related to intergranular stress cracking being done by GE and a number of groups.

MR. SHEWMON: That's comforting. Have you learned anything from it?

MR. NELSON: The answer is yes. We have with us Doug Timmons. Maybe he can answer that question more directly.

MR. TIMMONS: My name is Doug Timmons. We have been involved with the issue of stress corrosion cracking on WNP-2 specifically. In the past we were aware of the problems at Duane Arnold. The stress intensity factor calculations and so forth are being done on the plant.

I am now trying to get more specific information on the plant that we can provide to you later today.

MR. SHEWMON: Do you have the ability to reduce oxygen after you have changed fuel and started the plant or before you start up? Do you bleed air out in any way out your circuits?

MR. TIMMONS: I do not believe we have a separate oxygenation system.

MR. SHEWMON: Do you have a de-aeration system or an operating procedure which would tend to encourage it?

MR. TIMMONS: We will get that information, also.

MR. SHEWMON: Thank you.

MR. PLESSET: Thank you, Mr. Nelson.

MR. MAZUR: Good morning. We appreciate the opportunity to further demonstrate our capability and performance toward our request and application for a 40-year full power operating license. We are prepared to answer any and all questions.

Before I get started I would like to pass along an apology on behalf of Mr. Ferguson. In his recent recovery from bypass surgery and the rapid-fire action of this project, he just failed to make it on the gate time to get here. However, we do have in the audience with us his boss, a member of the Executive Board, Mr. Karl Halvertson, sitting in and listening to our performance.

With that, I would like to get started.

(Slide.)

The Supply System is a municipal corporation formed to operate and generate electricity and build the plants. We have no responsibility for marketing the power or the transmission capability. That falls within the purview of the BPA. We're basically an all nuclear facility and rapidly transitioning from a construction mode to an operating agency.

With that in mind, Mr. Ferguson undertook a review of the organization put together years ago in behalf of emphasizing construction toward that transition of operation. He brought in the INPO organization, consulted with other CEOs at other utilities, and in keeping within clear lines of responsibility and delegation of authority moved to an organizational change within the last couple of months, formulating a director of operations function which I head up, including all of the necessary responsibilities for putting the plants online, operating them and maintaining them. In addition, maintaining the corporate nuclear safety review board function as well as the technical specialist assigned for plant verification.

(Slide.)

The numbers you see in the vu-graph represent the manpower strength assigned to those functions. Starting at the top, the technical specialist function for plant verification under John Honnencap, who reports directly to Mr. Ferguson, giving him the assurance at the time we load fuel as appropriate if so done.

There is then the Nuclear Safety Review Board which is operative now and has onboard key members of the staff and online responsibilities and does have some outside consultants and specialists who support that function. It will be meeting from now on. It will function along tech specs, plant performance, plant deviations and so forth.

On down to the operations function, where we have both the finishing of the construction of the plants under the program directors in the case of number 2, as well as the others, and the power generation function to maintain the plant, and the engineering function to support the operation and transition of the plant under technology.

(Slide.)

The types of people down to the third level and fourth level within our organization representing nuclear experience look something like this, averaging something close to 20 years. Starting at the top with Mr. Ferguson and on down to the plant training manager and so forth. I feel very fortunate to have built a team like that and it is very strong.

MR. MOELLER: How many of these years are experience working on the WPPS plant, or are they actual experience at other plants that are operating?

MR. MAZUR: I have two previous nuclear plants online ahead of this one. I have two years within WPPS, so the rest is outside. Mr. Ferguson has a very comparable type of experience. There are others in our organization who have something like BWRs under their belt.

MR. MOELLER: Fine, thank you.

MR. CARBON: Specifically, what does Mr. Martin, the Plant Manager, have?

MR. MOELLER: Mr. Martin, stand up and tell him exactly. I keep forgetting.

MR. MARTIN: I'm Jerry Martin, the Plant Manager. I've been with the supply system for four years. So the balance of the time since 1960 has been totally in the operation of nuclear plants. It would be a balance of 18 years, and just prior to the supply system experience I was with the Commission in Atlanta for 18 months, and before that, the balance of those years was with the General Electric company on startup crews at Millstone, KKM in Switzerland, GKN in Holland and the Browns Ferry project.

MR. RAY: Mr. Mazur, back on your organizational chart, off to the left is a block that's entitled "Licensing and Assurance." Is that where QA resides?

MR. MAZUR: Yes, sir.

MR. RAY: So it's not under your direction?

MR. MAZUR: It is not. It's totally independent.

(Slide.)

The final slide I want to go through is a very important feature of any organization. Mr. Ferguson works for the Executive Board. The Executive Board has specifically delegated him all matters related to safety in the operation of the plant. That delegation has been even expanded to include industrial safety matters. The essence of that delegation is full authority to stop/cease work to take any necessary action, regardless of the monetary consequence to provide for the safety of the public and employees in the plant. That delegation does exist and he does exercise it.

MR. MARK: Your Nuclear Safety Review Board, I believe you mentioned had the heads of each line?

MR. MAZUR: I said it was made up from membership of the line management disciplines.

MR. MARK: Then you mentioned those outside consultants. Do they have a clear way to express their opinions, or are they only there to answer questions?

MR. MAZUR: Opinions, sir.

MR. MARK: So they can bring to Mr. Ferguson's attention something that they think should have that, even if the line management people don't think so?

MR. MAZUR: I'm confident that --

MR. BOUCHEY: I'm Don Bouchey. The outside members act as full members on the board, voting members.

MR. MARK: I was just worried about the word "consultant."

MR. BOUCHEY: That means we pay them.

MR. MAZUR: If there are no other questions, I'd like to introduce Bill Bibb, the Director of Power Generation, responsible for the operation of this plant and all others.

MR. AXTMANN: Where does training come on your organizational chart?

MR. MAZUR: Under power generation. Mr. Bill Bibb will take care of that in detail.

MR. BIBB: Good morning, my name is Bill Bibb.

(Slide.)

I'm Director of Power Generation for the supply system. In that job, I have responsibility for all the operating plants in the supply system, including the startup of WNP-2. I have been -- I am here today as the officer in the company responsible for safe and efficient operation of our plants, and in that position, I have been relieved of other responsibilities so that full attention may be devoted to the safe operation of those plants.

Just a little bit on the experience. I have been in the business for 28 years. A list of my experience -- prior to coming with the supply system was with the General Electric Company in startup and operations of a number of boiling water reactors around the country and around the world.

Now, under the objectives of the supply system, our objective is that WNP-2 be built and operated safely and satisfy all state and federal requirements. The central theme of the power generation organization is focused on quality of performance. In order to satisfy that requirement, it is necessary that we have qualified people and that we have an adequate training program to satisfy that need.

The supply system has in place a selection process for new employees designed to insure that the people hired to fill those positions meet all the basic qualification requirements. Those people are then brought onboard, trained to meet all the requirements for licensing on our power plant.

A third training program has been established and is functioning and is on schedule. We've also initiated the process of INPO accreditation of that training program, to demonstrate that the program is working.

The existing WNP-2 plant staff represents more than 1800 man years of nuclear power plant experience, of which over 600 man years are boiling water years of experience. Most of the staff at WNP-2 are well along the way of the training program. Forty-three or thereabouts of the required 46 licensed people on the plant have already been certified at the RO or SRO level through the training program of simulators and what have you.

Also, all of them have been through the basic training such as psychological training, examinations, basic radiological safety and those types of things.

MR. CARBON: Would you say just a word about what you mean by psychological training?

MR. BIBB: Let me correct myself. I didn't mean psychological training. I meant psychological exams. Psychologically it is a basic function of hiring the people. We believe the economy of operations and safety are best served through a strong quality program, and that well-trained, qualified people are the best assurance of meeting that goal.

I would like to take just a minute now to walk through the organization that we have in power generation at the upper level. I will start from the left and tell you a little bit about it. We do have two operating plants and they are both under one plant manager, Mr. McDonald. There is a small plant called the Packwood plant that is a hydro unit, 30 megawatts. It runs well and keeps a lot of light bulbs going.

We also have the Hanford Generation Project which is a unit of 750 megawatts, uses waste steam from the old N-reactor. It, too, runs well and is online most of the time. So we feel real good about it. As a matter of fact, at the beginning of this year it passed the 50 billion kilowatt hour mark.

Coming on across the chart, we in my organization have a group called generation services. They are just that. They provide non-destructive examinations. We have a standards lab, labor services and those things that are needed to help at the plants that need not necessarily be on the plant staff.

Additionally, we have a central organization, a training group under Mr. Stickney. His job is with a few people in the central organization to provide policy and guidance for training those on the plant staff reporting to the plant manager at each of the sites. He assists in other kinds of training such as basic or generic training to get people into entry level training, and those kinds of things.

I also have at Unit 2 the startup manager reporting directly to me. That is a recent change. In the past, they were reporting to the construction people, and now they report to the operations people. That is under Mr. Haffleback. He has about 80 to 85 people total, 65 of which are systems engineers. They are well-trained people within the supply system, as well as having experience on several other boiling water type plants in the startup area. So I feel real good about the startup staff we have at Unit 2.

The next one is the WNP-1 plant. We have the plant manager there. I won't say much about that one. That plant is in a slowdown mode at this point.

WNP-2 plant -- you heard Mr. Martin speak. He's here and he will talk to you a little more about the specifics of that, and we will get into that some more.

We also have a WNP-3 plant which is a pressurized water reactor plant. That's about a little over 60 percent completed.

Quickly, let me run through the support organizations. You can see in power generation we have some activities. There are a number of other activities that I will quickly run through to show where we get some of the support.

There is an organization in the company that provides sort of an overview in radiological chemistry, training, those kinds of things. Additionally, they supply security in its entirety at each of the plants. Emergency prepareiness planning and so on.

There's a licensing and assurance director. Mr. Glasscock, who is here today I believe, represents that organization. They provide the independent quality assurance overview and policy. Licensing is also under that particular group.

MR. REMICK: Mr. Bibb, in the QA area, do you use outside consulting QA audits in addition to your own internal QA?

MR. BIBB: I could ask Mr. Glasscock to address that, if you want to get specific.

MR. GLASSCOCK: I'm Bob Classcock from the supply system. We have had the opportunity -- at the present time, we are doing a self-initiated evaluation in accordance with the INPO guidelines. We have had to utilize the INPO system will performing the audits. We may do additional audits if we feel it's necessary.

MR. MOELLER: You were talking about training. Do you have a separate training building and a separate training staff? How is that done?

MR. BIBB: Yes, we do. We have the central training policy and then at each site, we have the specific training people.

MR. REMICK: One other question on training.

You mentioned you had 43 operators that were certified. Does that mean company certified? They have not taken their RO or SRO exams yet?

MR. BIBB: No one in our organization has done that. We will do that about three months before fuel load. They have all certified at BWR simulators at other places in the country. We do have a simulator coming ourselves.

MR. REMICK: So your certification is on a simulator. What simulator did they use?

MR. BIBB: We used the GE one in Oklahoma, and we've also used the one in Ohio.

MR. MOELLER: What percent of the time does a typical person spend in training? What do you anticipate after the plant goes into operation?

MR. BIBB: Our total training program is just under three years for the average person coming in.

MR. MOELLER: I mean after they are part of the working staff. I presume they'll have refresher training. Roughly what percent of the time will they spend?

MR. BIBB: Probably about 20 percent of their time. Jerry, is that about right?

MR. MARTIN: We have the six-shift rotation plan for them to go into refresher training on the training rotation shift.

MR. REMICK: Would the question come up about the use of STA? Would you use the shift concept or what?

MR. BIBB: At this point, we're committed to do that.

MR. REMICK: The STA?

MR. BIBB: Yes.

(Slide.)

Getting off with the support for the plants --

MR. RAY: Mr. Bibb, just one question. I notice you mentioned support services. I noticed it was a separate organization, parallel with Mr. Mazur's. Do they have, for instance, radiological chemistry support activities for on-site people?

MR. BIBB: No, they do not as a rule, although they may on occasion. Their role primarily is to provide an overview between our operations of the plant and our engineering, which provides criteria so that our people implement engineering criteria.

These people are in a support role for a check and balance type situation where they may have a few people; say, two or three experts, in the field of radiology, okay? They would check on our people to see that we are following the procedure that we are meeting the criteria.

MR. RAY: So in a sense, it's an auditing and consultant service?

MR. BIBB: Essentially, yes.

MR. RAY: How about your health physicists on site?

MR. BIBB: Same thing.

MR. RAY: To whom do they report?

MR. BIBB: To my people; Mr. Martin, specifically.

MR. RAY: Not to Shannon?

MR. BIBB: That's correct.

(Slide.)

MR. MARTIN: I want to leave you with a few thoughts. Picking up on some of the things that Mr. Mazur said, the WNP-2 plant is nearing completion. The hydrostatic test was completed this past August successfully.

The plant test program is under way and is well-staffed with trained engineers. We have about two-thirds of the total systems in the plant provisionally accepted and about 26 or 28 systems that are turned over into the pre-operational test program. There are a total of 101 engineered systems in the plant, to give you a base of reference.

By the way, as I say, that program is moving well. We are into the flushing and testing on many components at this point. The plant staff is essentially complete on WNP-2. All the key positions are filled except one. The training program is working and it is on schedule. Accreditation of our training program is in process.

The plant procedures manual is nearing completion. It is targeted for the first of the year or shortly thereafter. Top managers, as Mr. Mazur pointed out, in our company have substantial nuclear experience. We believe we have good people and a strong program that will put us in a complete state of readiness before the scheduled fuel load date of September of 1983.

With that, if there are no questions, I would like to turn it over to the Program Director for WNP-2, Mr. Robert Matlock.

MR. SHEWMON: Mr. Chairman, is this a contested license?

MR. PLESSET: I believe not.

MR. SHEWMON: Why is it they are in there so far before startup?

MR. PLESSET: You are asking the same question I have asked repeatedly and I never got an answer to it.

MR. SCHWENCER: All Schwencer, Licensing.

At the time the Staff started its review on this project the construction completion date was December of 1982 and the Staff sent its staff to work on the basis of meeting that and it being in a position not to slip in a way that would impact the plant.

Once the review process was begun, we felt we should continue. At one time we felt this was going to be a contested case, but that did not develop.

MR. PLESSET: Does that answer your question, Mr. Chairman?

Go ahead.

MR. MATLOCK: Good morning, Mr. Chairman and members of the committee.

I am responsible for completing the construction on that plant. I am responsible for the site. I am also responsible for effecting an orderly transition at fuel load in September of 1983.

The WNP-2 chronology goes something like this. Engineering was started just prior to 1970. The construction permit was received in early '73, and construction began and proceeded in a somewhat orderly fashion up until the first part of 1977. That was the time at which the reactor vessel was set.

About that time, troubles began to beset the WNP program. In the '77 time frame, the construction contractor was faulted for workmanship problems and that sort of thing continued to be compounded until significant quality problems were identified in 1979 in the sacrificial shield wall, which we have now resolved.

I have noted in red the breath-catching period of time at WNP-2 in the spring of 1980. In July, as a matter of fact, the plant was shut down. It was shut down by the supply system. All safety-related work was stopped. There were quality problems for essentially the rest of that calendar year and that time was spent putting our act back together, looking at the quality assurance and management systems, trying to get back into the situation where we could get back into work on WNP-2.

We have done that. We are back on track and at that time, in this period of time, the plant was 86 percent compete. Last month it was 92 percent complete, and we are constructing and tracking to a fuel load date of September of 1983.

I want to spend the majority of my time talking about the way in which we have recovered from past quality problems that beset Number 2, and provide that as the basis for our confidence that when we start up we are going to have a good plant.

(Slide.)

MR. RAY: Just a brief question. Would you tell us, without going into much detail, what the major change in contractors was? What was the key change?

MR. MATLOCK: The principal change was to bring in a major contractor, Bechtel Power Corporation. The essence of that change was that it was a new previously-unassociated contractor with experience and with the resources to breathe new life into the project.

MR. RAY: Does that organization now have overall control?

MR. MATLOCK: Yes, they are in charge of the erection contractors. That organization is also the systems completion contractor.

The problems we encountered were simply that quality was not being achieved. Management actions up until the early part of 1980 were not successful in turning the project around, and there was a substantial backlog and an increasing backlog of problems.

In the recovery process we shut down and went in and revised our procedures and management control systems in such a way that we could assure quality for future work. This was worked out with Region V. We also committed to go back and develop a quality verification program. That program was addressed, looking backwards at safety-related construction that had already been completed, and we verified that it had been done in an appropriate way.

The restart program, the scope included quality Class 1 and/or seismic category 1 component structures and system. It included the evaluation and review of contractors' QA programs. It also resulted in some substantial changes. I estimate there was something on the order of 100 manyears of effort involved in that over a nine month period and many procedures were modified or rewritten.

(Slide.)

Some other program improvements we made at that time to strengthen the organization was consolidating everything at the site under the program director -- operational startup, construction, everything under the program director, who reported directly to the managing director.

That was carried on for some time until we got this project moving and we reorganized to a more traditional form. As I mentioned, we hired Bechtel and the new blood. The new thoughts were very helpful in getting us back on track. We reassigned undivided responsibility for engineering to the architect-engineer, whose responsibilities had been somewhat diluted prior to that time, and he was assigned responsibility just for engineering support.

We also went into an aggressive program of reducing deficiency backlogs.

MR. KERR: What is meant by "reducing the backlog to within new performance measurement limits"?

MR. MATLOCK: Well, part of the restart program was to define what our performance guidelines in the future were going to be. We set some limits on the number of outstanding deficiency documentations we could have at any one time. It reduces our backlog to that particular performance limit.

MR. KERR: Thank you.

MR. MATLOCK: Then this additionally means a lot to me personally because I am in charge of the project. Subsequently, after we had restarted in August of last year, we terminated the then-piping mechanical contractor and reassigned that work to a new contractor -- in this case Bechtel Power Corporation. That was ASME work. It was Section 3 work at that time.

That change-out process forced an entire, complete review of all code-related documentation because there was a change in code responsibility. We went through the entire set of documentation that existed at that time for the piping mechanical contract. This was quite an undertaking, but it was certainly worthwhile.

MR. WARD: Mr. Matlock, Mr. Stoneberg of the Region V office alluded to a problem that the inspection people had due to what he said were a multiplicity of contractors and gave the example for mechanical contractors welding safety grade equipment. They did not look at the procedures and qualifications of one contractor, but then they did not have confidence that another contractor would have communicated to him resolution of problems and that sort of thing.

Was that something that was characteristic of the pre-1980 stop work and has been corrected since then?

MR. MATLOCK: The answer to the first part of the question is yes. It was characteristic of the work prior to the stop work. The answer to the second question was we have, in my view, reduced that but we have not completely eliminated it. That was characteristic of contractors with several responsibilities.

It is my view that we have reduced the significance of that problem rather substantially.

MR. STONEBERG: There has been a reduction. I only bring it out as an example of why we were challenging the inspection program for Region V.

(Slide.)

MR. MATLOCK: For the quality verification program, this was to guarantee that when we got back to work we were going to do things right. The quality verification program, as we described it, was a program to look backwards. Its scope included a review of, again, Class 1 and seismic category 1 documentation and hardware that had been previously installed.

It included inactive -- that is, closed out -- contracts, complete purchase equipment contracts for equipment that had already been purchased. We looked at the competed part of incomplete systems. We also undertook special tasks. We looked at the qualification of people. We looked at the adequacy and disposition of past deficiencies, and we looked at some procurement techniques that were carried out in the past.

This program -- the quality verification program -- is now 85 percent complete and is scheduled to be done in the final report in March of next year.

MR. CARBON: Could you give me just a short example? Suppose you were worried about or were wondering whether some bolts had been sunk in concrete. How do you verify that they withstand the forces they are supposed to?

MR. MATLOCK: We have the guy that is responsible, not for the problem but who worked on the solution. Doug, would you address that?

MR. TIMMONS: Doug Timmons with the supply system.

Because of the problems we have had at our site, we have had an overall program to put the issue to bed. To answer your specific question, we have gone back into the documentation, records and so forth to see what type of documentation exists for the various contractors.

We have looked at the engineering criteria for the installation of the anchor bolts and seen how that was translated into the contractor's procedures to see if he was implementing what he should be implementing, and then we looked at the documentation for the back end of the process.

Where we have found difficulties or what I would call suspect installations, we have gone back on occasion and will be again shortly on one specific contractor and performed a pull test program on anchor bolts. In other cases, we can confirm for some specific contractors, due to the time frame in which they installed their safety related or seismic Category 1 anchor bolts, that the paper trail is whole. We have done a sampling of his work and have been satisfied that he did it correctly.

Our program has been a mix, depending on the history and the problems associated with the contractors. Some contractors we reviewed to see yes, he did his job properly. Others we have had to go into a pull test program and so forth, depending upon the specific nature of the contractor.

MR. SHEWMON: There is a specific force that these should be able to withstand, and you can pull nondestructively to that force?

MR. TIMMONS: That is correct.

MR. SHEWMON: Or hopefully nondestructively, but at least can test them.

MR. TIMMONS: It is non-destructive. There have been tests performed for representative cases in the field where we can draw a direct relationship between pull test and movement of the anchor bolt, and that is a direct relationship which you can draw to failure, and that is the relationship we rely on in these pull tests.

MR. SHEWMON: I am glad you are doing some of that. Occasionally some members of the committee try to point out that generating enough paper on a quality assurance may not always guarantee quality.

MR. MATLOCK: The quality verification program was results-oriented. It is not done yet, but we have some indications from what we have done so far. We have found no new problems. We have not found another set of generic problems that we heretofore had not uncovered or identified.

We also have now developed confidence in the disposition of past technical deficencies -- that is, deficiencies in the past when they were resolved seem to have been resolved adequately. This program so far has given us an additional degree of confidence in the adequacy of past work.

It is also accomplishing its primary intended task. It is verifying past work and causing corrective action to be taken in some areas where we have found the need, such as the grouting program and the anchor bolt program.

Now one additional word on documentation.

(Slide.)

Since our problems unfortunately are not unique to the industry and there has been words said about missing documentation and that sort of thing on plants, we do not have a missing documentation problem. There are not massive parts of our documentation that are missing -- maybe a few -- but we have been able to recover those documents by going back to the suppliers. The documentation we have now we found to be generally in compliance with the codes and once we have got it in order, it is adequate.

Also as part of this program, we confirmed weld quality by review of all the ASME installations by a review of all of the ASME radiographs. Some 3,000 welds have been reviewed and reevaluated.

(Slide.)

We are back working. The project is moving to completion. The programs for quality verification and quality control are in place and they are part of the everyday process. We are 92 percent complete on construction. A quarter of the systems of the 101 systems are turned over. Two-thirds of them are provisionally accepted. That is, the hardware is in the possession of the Supply System. Forty percent of the rooms within that facility are either turned over to us or are in our possession on a provisional basis.

We have hydrostatically tested the reactor vessel and associated piping. We completed that successfully last month. We recently filled the wet well for pooled cool flush of the associated systems. We are tracking to a fuel load date of September of 1983. That is tight. That's not a contingency schedule, but it is achievable. I have no reason to believe that that date is not achieveable.

(Slide.)

Now I would like to spend a few minutes on the plant verification and design verification issue.

About a year and a half ago, one of the things we committed to internally in the organization was to establish an acceptance review process. The process was for the purpose of establishing a basis, a well-documented basis, for our own use in saying that when this plant goes on line it can be operated safely and is technically adequate. And, of course, with our history on number 2 there was a special clause that was directed at finding the quality defects that may be in number 2 and correcting them.

That led to an overall program that we call the plant verification program. The elements of that program include design requirements and design verification. It also includes construction of the quality verification program and performance test and startup, and the operating envelope review.

(Slide.)

The approach of course is to develop a plan and track it. It has a couple of other features. One is that it is overviewed and run out of the managing director's office. Dr. John Honnecap reports directly to Bob Ferguson and is in charge of this program. That gives it independence. We also are utilizing an outside, independent technical auditor to review, counsel, recommend and report on the progress of this program.

(Slide.)

MR. MATLOCK: In summary, for the construction program it is my very strong feeling that we have an experienced, and have now an experienced design and construction organization who are dedicated to completing this project and doing it right. We have resolved or are resolving quality problems that may have occurred in the past. We have in place now the means, controls and verification means to ensure that the design is correct and that the plant is built in accordance with the design.

We also have in place an orderly transition program from construction to operation.

Now, if there are no other qustions, I would like to introduce Jerry Martin. He is the individual who will be on the point here shortly in about a year, and he will talk about planned operations.

MR. PLESSET: Maybe this is the time to consider a break, if that is agreeable, Mr. Chairman.

MR. SHEWMON: I was about to remind you.

MR. PLESSET: So thank you, Mr. Matlock.

Mr. Chairman, I hope that the Applicant can answer the questions raised about -- we are not recess at the moment. There is also a question about ATWS.

I am going to suggest that we arrange for a brief presentation by Dr. Lipinski, who has participated on our behalf in a meeting with the Staff and the Applicant on that point, if that is agreeable. We can have that at any time in the next interval.

MR. SHEWMON: Well, we have picked up 15 minutes.

MR. PLESSET: So why don't we have a recess until 10:30.

(A brief recess was taken.)

MR. PLESSET: Mr. Martin, I think we are ready for you to begin your portion of the program.

MR. SHEWMON: Could we get people to sit down and stop talking and so on, except for Mr. Martin?

MR. MARTIN: My name is Jerry Martin, and I am the plant manager at WNP-2. I have been in this position since June of 1979. The one year prior to that with the supply system, I was manager of start-up and operations.

The subjects I wish to cover this morning are listed here. There are eight topics. I will start with the plant layout.

(Slide.)

MR. MARTIN: Our site is on the Hanford Reservation, which is a 570 square mile reservation that has been controlled since the early '40s as part of the

Manhattan Project. On this 570 square mile Hanford Reservation we have the WNP-2 site, which is 1089 acres. On the WNP-2 site, the power plant complex is shown here with the reactor building, turbine generator building, diesel generator building, radioactive waste building. The plant staff is housed here in the service building. The other major component shown on the site are the six cooling towers, mechanical forced draft, low profile.

Right in the center is the circulating water pumphouse. Down in this area are the spray ponds or ultimate heat sink with the standby service water pumphouses located right at these corners.

That is our basic site layout.

(Slide.)

MR. MARTIN: As plant manager, my responsibilities have been in parallel with the construction effort to hire and develop a staff that will be ready to support the initial pre-op testing of the plant and then obviously be redy to operate the plant at the issuance of the operating license. This staff is shown in this configuration. This came about as the result of a study of the nuclear industry and the other utiliies.

So I have reporting directly to me six basic departments, starting with the maintenance manager who is a man who had previously been with the test and start-up group. He has approximately ten years of experience in the start-up and testing of nuclear power plants and maintenance of the systems associated with nuclear power plants. He has an organization reporting under him that has 638 man-years of nuclear experience.

The training manager reports directly to me. The training manager, Rod Davidson, has 16 years of nuclear experience, including six years with the General Electric Company and six years with the supply system specifically in the area of training.

I will go into a little more detail later on the training organization.

The operations manager, Roger Corcoran, is a degreed individual who has 16 years of nuclear experience, including six years at the La Crosse boiling water reactor. The organization under him has 550 years of nuclear experience.

Adminstrative manager, there are 16 -- excuse me, there are 22 individuals in that organization who prepare and provide us the administrative procedures and clerical support. The administrative manager is also the secretary to the Plant Operations Review Committee.

The technical manager of the technical

organization, they have 24 individuals. Kirk Cowen is the manager, and he has 22 years of nuclear experience.

The sixth department is the health physics and chemistry manager, Bob Graybill. He has 29 years of nuclear experience and was involved in establishing the health physics training program on the Duane Arnold Energy Center and the La Crosse boiling water reactor.

In total, adding up the numbers, we have greater than 1800 man-years of total nuclear experience, of which there are 659 man-years of boiling water reactor experience.

MR. WARD: Jerry, who is responsible for generating the plant operating procedures?

MR. MARTIN: Under the plant administrative manager, physically generating the procedures comes under the administrative manager. However, of course, the Plant Operations Review Committee, of which I am Chairman, and with members of the department managers as required by tech specs. For example, the maintenance manager is a member of the POC, the operations manager, the techncal manager The POC, then, is the committee who approves the operating procedures, and they are physically prepared and typed in the administrative department.

MR. WARD: Who drafts the procedures? Is this done in your operations manager group?

MR. MARTIN: We have 14 volumes, and if it is the specific operating procedures, it comes under the operating manager's department. The operating procedures were written by the shift managers. So those specific procedures come under the operations. If they are administrative procedures, they come under the administrative manager.

MR. CARBON: What kind of experience has Mr. Corcoran had at the technical supervisory level such as shift supervisor or assistant plant superintendent or whatever?

MR. MARTIN: Mr. Corcoran is here. If you permit, I would like him to stand up and address that.

This is Roger Corcoran.

MR. CORCORAN: Roger Corcoran, supply system. I have not been a shift supervisor. I spent several years at the La Crosse boiling water reactor, one year as the plant supervisor on that staff. Other years have been spent in the supply system, hiring the shift supervisors and the operators.

MR. CARBON: Thank you.

MR. RAY: Jerry, getting back to Mr. Ward's question, are surveillance testing procedures also under operations?

MR. NELSON: No, sir, Mr. Ray. They are a technical procedure and under the technical manager, Mr. Cowan.

MR. RAY: Thank you.

The 1800 man-year on-site experience is impressive.

CAn you tell me what proportion of that reflects the changes in the reorganization that took place? Has 1800 man-years been on site since the early years of the project, or was it before the servicing of all the problems that Mr. Matlock talked about?

MR. NELSON: We have been hiring this staff in parallel with the construction project, so I have numbers here that show you how the number goes up to 239. So to answer your question, it has been accumulating in parallel with the construction effort, but it has not come about as any quantum step change.

MR. RAY: As a result of the reorganization or the correction of the problems you have had.

MR. NELSON: No. We have from the start instituted a hiring policy of hiring those with as much nuclear experience, and preferably directly related to boiling water reactor experience as possible.

MR. RAY: Thank you.

MR. CARBON: What specifications are you

setting for the assistant plant manager? Where do you feel you are perhaps weakest in your organizational structure?

MR. NELSON: Your first question, the specifications or qualifications, we are following ANS 3.1, the 1978 draft. We are also keeping up to date with the '81 version of ANS 3.1 which Reg. Guide 1.8 will encompass. So that is the document that we have for specifications.

For example, the plant manager in that is required to have ten years of nuclear power plant experience. We are categorizing the assistant's job in that same category.

MR. CARBON: You personally have extensive start-up experience and so on, but I don't see too much operating experience.

Are you pushing for that in the plant management?

MR. NELSON: Are you speaking personally to the plant manager?

MR. CARBON: Yes.

MR. NELSON: I must clarify that back at Millstone, my experience, for example, I have a Senior Reactor Operator's license at that plant under a full turnkey contract. That required that I personally operated the plant, and under the full turnkey concept, we operated the controls of the plant up through the morning run.

MR. CARBON: Good, I'm glad to see that.

MR. WARD: Jerry, one other question. Maybe you will get to this later, but where do the STAs fit into this organiation?

MR. MARTIN: They fit under the technical department. There are 24 individuals in this organization. They are degreed engineers who are also in our cold license program. So the shift technical advisors are under the technical advisor.

MR. WARD: So they have a dotted line relationship with the shift organization?

MR. NELSON: That's correct. They report in under the operations manager. If we expanded this down to the shift organization, to the shift manager, the shift manager has reporting to him a control room supervisor. The shift support supervisor is then dotted in directly to the shift manager. He actually comes out of the technical department.

MR. WARD: What do you think about the STA concept? Do you expect it to be helpful to you?

MR. MARTIN: Yes, I do. We have had a policy to assure that it is helpful that these individuals not only hold a technical degree, but that they also go through the cold license training so they will have established credibility with the shift managers.

We have five who are now certified at the SRO level. They have completed this training through the Browns Ferry Training Center in Chattanooga, Tennessee.

MR. CARBON: Some organizations seem to be going in the long range direction of melding the shift technical advisors with the shift supervisors and to do away with the separate function.

What is your long range view?

MR. MARTIN: From the start, our long range view has been that we believe very strongly that the shift manager with the 15 to 20 years nuclear power plant experience is what essentially protects the health and safety of the public. We also view that the shift technical advisor is an asset if he is not only a degreed individual but if he also has this operational savvy. So our long range plan was to upgrade the analytical skills of our shift managers, those who had 15 years of power plant experience. When they had completed that program, we sent them to a one-year program, and after a one year academic upgrade, couple that with the longtime nuclear experience, our feeling is that that is our long term approach to upgrade the analytical skills of the experienced shift manager.

MR. CARBON: In that process, will the shift technical advisor disappear then and his capabilities and role be handled by the shift supervisor?

MR. MARTIN: If it is documented and finally approved that the shift manager does contain the analytical skills required, then the need for the shift technical advisor would go away. In the interim, we are committed, however, to the shift technical advisor.

MR. CARBON: Thank you.

MR. WARD: Are you making it a requirement that your shift managers have I think you said 15 or 20 years experience?

MR. MARTIN: That is not a requirement, no. We do have a group of shift managers who overall average is just under ten years. Using that as an example relating the shift technical advisor to an experienced shift manager, we do have close to 15 years experience. We follow the requirements of ANS 3.1. ANS 3.1, 1978, I believe is three years.

MR. PLESSET: Go ahead, Jesse.

MR. EBERSOLE: Mr. Martin, your plant is one of -- it has a unique in classes of plants feature that I just thought it would be informative to peak into.

You have a function at your plant which is semi-automatic blowdown. Prior to getting that function, you have an interval of I think 30 to 90 seconds during which you are supposed to sort out whether you should let it proceed or not.

I think it would be informative for you to tell us when you would and when you would not let that drastic action go to its terminal condition.

MR. MARTIN: Mr. Ebersole, the best way to answer that, I was going to later in the presentation get into --

MR. EBERSOLE: You can get into it later. That's all right.

MR. MARTIN: We address the symptomatic emergency procedures guideline.

MR. EBERSOLE: Also, you have an auxiliary control room, control center from some point that you can operate the plant if your main control room is in disarray.

Would you also comment under which circumstances the operators function if the main control would be so degraded that you would invoke its use?

MR. MARTIN: I will address that. The remote shutdown panel is the subject for the next speaker.

MR. EBERSOLE: Okay.

MR. MARTIN: One last point on this slide. We have the 239, 249 on board supporting the preoperational test program. We have been on shift for over a year. The philosophy is that the operators are the ones manipulating the plant during the test program using the plant operating procedures that are in place as a trial use and also to provide this as a training period for the plant operators.

(Slide.)

MR. MARTIN: Training. I'll move rapidly through this.

We not only have cold license training program, but as you can see, we cover training for non-licensed personnel, including the maintenance people, the test and startup people. We do have training programs to cover these categories.

Yes, sir.

MR. SHEWMON: Would you tell me, as an example, a little bit about your instrument people? I guess I have two things. One, they revolve primarily around what controls you have over certifying that somebody who works on a system is qualified for it. Two examples would be when you are starting up. Another would be if you need somebody at 4:00 o'clock on Sunday morning, does the union contract require that that go to the guy with the least amount of overtime, or how do you certify him?

MR. MARTIN: To address that question, we have a nuclear bargaining agreement. It does not restrict us in the sense of obtaining the necessary qualified individual I need. For example, if at 3:00 in the morning I need an instrumentation control person, the contract does not preclude me obtaining that individual, regardless of seniority.

MR. SHEWMON: And how do you know the one you are calling out is qualified?

MR. MARTIN: We have in our administrative procedure manual a qualification check list requirement. Each individual will have had to have had that qualification completed and signed off by his immediate supervisor and then also concurred in and signed off by the maintenance manager, and those individuals' qualifications are known to the shift manager. We have a posting of those individuals who are qualified to perform the certain categories of critical work.

MR. SHEWMON: And they are to take instrumentation people on board now and working with you, or will you be hiring them on and training them?

MR. MARTIN: They are all hired. The last slide showed we have 76 people in the maintenance department, and they have all been on board for some time. Those individuals, in addition to the mechanics, have been the people who have had the hands-on experience with the hardware since provisional acceptance to the supply system.

We have taken an aggressive role in as soon as construction turnover or provisional acceptance is complete, the plant maintenance personnel, mechanics, instrument and control are the ones who are readying that system for the pre-operational test program.

MR. CARBON: One additional question.

How often is that technician that handles that particular detector requalified?

MR. MARTIN: There is an annual requalification requirement. We have an ongoing retraining program.

MR. CARBON: Annually?

MR. MARTIN: Annually, yes.

(Slide.)

MR. MARTIN: Training -- to cover all of these areas of training, reporting directly to me is the plant training manager, covering all of these plant-specific training areas, cold license, the maintenance training that was referred to. Reporting directly to Mr. Phil Bibb is the headquarters training manager. They do more generic training in support of the total company and the plant staff.

The other category is the specialized radiological training, fire protection training, ALARA training. They report through the Director of Support Services.

This specialized training is in that category. That is the philosophy of our training of the company, plant specific, and generic and radiological training.

(Slide.)

MR. MARTIN: To summarize this subject, we are committed, we believe, very strongly to training. Our company has purchased a plant-specific simulator for each of the plants. We do send our shift technical advisors to the cold license training program. The bottom line is that we feel that we want to accreditate this process, and our managing director, Bob Ferguson, has sent a letter August 6 of this year to Dennis Wilkinson of INPO, to start the accreditation process for our overall training program.

MR. WARD: You expect to have by the time of plant startup the new style system-based plant emergency procedures in place?

MR. MARTIN: Yes, I do. I have that coming up.

MR. WARD: Since you won't have your plant-specific simulator by that time, will you?

MR. MARTIN: We may not.

MR. WARD: How do you plan to train your operators in the use of those procedures?

MR. MARTIN: We plan to do several things. We are going to have the NSSS vendor run through them for verification. Of course, we have been involved for some time now with the BWR Owners Group on emergency procedures. With the review by various utilities that are looking at our procedures, we are going to have the architect engineer plus the NSSS vendor review our procedures. Then we are going to walk through these procedures in our own control room. We also should have the use of our plant-specific simulator at the factory. If we don't have the simulator physically near the WNP-2 site, we will have the opportunity to walk through those things on our simulator.

MR. WARD: You say you are going to walk through them in the control room.

Is that to the benefit of the engineers to make sure the procedures are good, or will each of your six shifts of operators walk through those procedures?

MR. MARTIN: Each of the six shifts of the operators will be walked through. That is the bottom line of that slide.

I think we have just covered the subject of emergency procedures.

Your question is yes, we are going from event-related to symptom-based procedures, and the operator will provide the bottom line. Each of the shifts will go through the training process.

MR. REMICK: You indicated that your STAs will take the cold licensing training program.

MR. MARTIN: That's correct.

MR. REMICK: Are you going to encourage any of those people to become licensed operators or SROs?

MR. MARTIN: Both. I am encouraging them to, and if the opportunity allows, we are going to allow them to sit for the NRC exam.

MR. REMICK: Those who might not sit, are you going toi have them somehow participate in the requalification program? How are you going to maintain their expertise if they are not licensed?

MR. MARTIN: We are set up on a six-shift rotation with four training shifts. The STAs we will have on a rotating shift basis, and they will have an opportunity to participate in the retraining process.

MR. REMICK: Whether or not they are

licensed?

MR. MARTIN: That's correct.

MR. KERR: This will be a retraining process aimed primarily at STAs or primarily at operators or of some combination thereof?

MR. MARTIN: I view the STA has to stay locked right into the operator in order for him to establish credibility with the shift manager, so they will be locked in with the cold license retraining, but not as a hard rule that the licensed operator has.

(Slide.)

MR. MARTIN: To summarize to this point, I feel that we have a well-staffed, complete organization already up and functioning on shift right now, extensive nuclear experience, including BWR experience, and our training program is comprehensive.

The next subject I would like to go onto is emergency procedures -- if there are no other questions, I will go past that.

MR. KERR: I have a question. This may not be the time to answer it. If it isn't, tell me when.

Once a prospective candidate for licensed operator has gone through the training program, what procedure do you plan to use to determine whether you considered him to be a qualified operator?

MR. MARTIN: Let me see if I understand your question.

Once he has received all of the cold licensing training?

MR. KERR: Whatever procedure you plan to give him, what process do you use to decide whether you are willing to turn him loose and let him operate your plant?

MR. MARTIN: The actual on-the-job training that takes place in addition to the classroom training is really the opportunity for his direct supervisor to observe his performance. During the pre-operational test program, for example, that operator has to demonstrate hands-on ability to manipulate the controls and the system. So the process is through immediate, continued observation by his supervisor and review by upper management.

MR. KERR: There is a formal process, or is it just relatively informal in that when he finishes all this, the supervisors get together and say Joe is okay but Jim is not?

MR. MARTIN: To answer your question, it is very similar to the Chairman's question on instrumentation control. We have qualification checklists that have to be filled out. As I stated, his supervisor has to approve that man's performance. Then it has to be concurred in by the operations manager. So it is a fairly formalized program with the administrative controls in Volume 1 of our Plant Procedures Manual.

MR. KERR: Thank you.

(Slide.)

MR. MARTIN: Before I skip over Mr. Ebersole's question on blowdown, we did touch on emergency procedures. If I don't state your question correctly, you were asking the auto-blowdown, how does the operator know how to respond to the need to depressurize the vessel?

MR. EBERSOLE: Well, it has always been a point of argument as to whether you should put a time delay on that or not. What does the human intelligence contribution have over and above an automatic system that would allow it to go ahead and materialize?

MR. MARTIN: Yo are referring to the 120 seconds time delay?

MR. EBERSOLE: Right. What does he know that the instrumentation ought not to know or doesn't know? The only thing I ever heard, he knows somebody has gone behind the board with a screwdriver. You must have a better answer than that.

MR. KERR: Do you understand Mr. Ebersole's question, because I don't, and I want you to explain it to me if you did.

MR. MARTIN: I know there is a 120 second time delay in our auto-blowdown system. I was going to refer, as I deferred the question earlier, to our emergency procedure which says that if we have, for example, a pressure control problem, the way we get into that, we've got the symptom of reactor pressure control and we have the decision then for a need to go ahead and depressurize. I think the key in responding to this is the operator knowing the overall plant conditions, for example, if there is maintenance on that system or, for example, if he has no other high pressure coolant injection system, then he must realize -- and they obtain this understanding through training -- that with no other high pressure system available, the emergency procedure guidelines will lead him to manually depressurize to allow other feedwater systems to come into the vessel to fill the vessel.

MR. EBERSOLE: Well, is the answer that he is not ever to likely intercept an SAR that is coming to completion?

If a semi-automatic -- if he gets a horn blowing, all right, he hears the horn blowing, he knows he is going to have a semi-automatic blowdown in, what is it, a minute and a half, 90 seconds, or is it two minute?

MR. MARTIN: 120 seconds.

MR. EBERSOLE: It is supposed to be a thinking interval. Presumably the equipment is at high pressure, it has no feedwater, or the level says triple low level or something. The critical issue is he knows if he intercepts blowdown, he may ruin the core.

Now, then, what will he do, if anything, ever, to stop a blowdown?

MR. MARTIN: I'm not sure I really fully understand.

MR. EBERSOLE: What will he do to intercept a blowdown in this time interval when the horn is blown? What does he do at that interval? It is a case in point of what does an operator do under duress?

MR. MARTIN: Let me refer, I was going to go to another slide. I could probably answer it quicker from here.

Through our graphics display system, and again, coupled directly with the emergency procedure, he is going to assess the major plant parameters to assess where the plant really is as far as reactor vessel control meaning pressure and level, containment control pressure, temperature and level and drywell control.

So what we expect to happen is that the operator will physically go to the board and through a process using the graphics display system, the new safety parameter display system, this is a touch pad that is going to be installed in the plant. He an merely touch a button here, put in on this high resolution color graphics cathode ray tube display and get an overview of the plant. He will be able to see those major plant parameters I addressed. He will be able to see in bar graph form reactivity, reactor pressure vessel levels, containment integrity, coolant system integrity, and the final display on the overview would be any radiological release.

So, to answer your question, he has this as an aid to overview the plant parameters, to aid him through the emergency procedure guidelines to take the action necessary to overcome this specific event.

MR. EBERSOLE: Mr. Martin, you are waltzing around the problem. You see, this is a safety grade function. It has multiple channel inputs. It is one out of two twice. Everything is fully classified. It is supposed to be a competely valid safety trip.

Now, by what right has he got to go in and mess it up?

MR. SHEWMON: Is there anything here that you are going to hear today that is going to change your mind?

MR. EBERSOLE: This thing is in situ. It is there. I didn't say that he couldn't do it, Paul. I just want to know when he does.

MR. PLESSET: The 120 seconds are up, Jesse.

MR. EBERSOLE: Maybe we had better let it go and let it trip.

MR. MARTIN: You are specifically wanting the logic in the ADS, and I would like to ask Chris Powers to respond.

MR. EBERSOLE: I want the ADS intercept.

MR. POWERS: My name is Chris Powers. There is a specific case in our emergency operating procedures where the operator must interact with the ADS system to prevent its blowdown. That is specifically when he recognizes we are in the ATWS situation. Under that scenario, we want to control reactor water level at the absolute minimum we can. We would like to control it just about the top of the fuel. That reduces the power level at which we are operating and allows the semi-automatic control to shut it down.

MR. EBERSOLE: Is that the only one you know of?

MR. PLESSET: That takes care of it, Jesse. He's got a specific situation.

MR. EBERSOLE: I didn't even know he had that one.

MR. NOVAK: I recognize it is calculated, but give us a couple of seconds. We would like Ralph Hodges to address that.

MR. PLESSET: All right, we will give you that.

MR. HODGES: There are a couple of places in the emergency procedure guidelines which the procedures will be based on which tell the operator to inhibit the ADS. One is the ATWS situation. Another is the level control. The procedure he has running will terminate a decrease in the water level and stabilize the water level. As an example, let's say he has some sort of an event. The HPSI system was out for maintenance, and RCIC system, he has a little trouble getting it started, but he has finally got it started, but if the water level is down below level three, he is reasonably sure, based on past experience with that system, once it's running, that he can turn the level around and get the level restored without getting the level down. He is going to prevent ADS in that case.

MR. EBERSOLE: He has picked up a pump, after the horn started blowing.

MR. HODGES: Yes.

MR. EBERSOLE: So now he has a hope of avoiding a blowdown.

MR. HODGES: Yes. He has to assess the system.

MR. EBERSOLE: I see. Thank you.

MR. PLESSET: Why don't you go on then? That was very interesting, wasn't it, Jesse?

MR. EBERSOLE: I think it was.

MR. MARTIN: I'll go on now where I am going in with the human factors and the control room habitability.

MR. WARD: Could I ask one more question? I hate to beat this one to death, but is that particular -- I guess there are two operations now rather than one.

Is that particular operation simulated faithfully enough on whatever simulator you are using, Browns Ferry, so you believe your operators are going to get training in that?

MR. PLESSET: I'm sure it must be.

Well, let him say it.

Say yes.

(General laughter.)

MR. MARTIN: Yes.

(General laughter.)

MR. MARTIN: The next two topics, control room habitability and human factors, I will just stay on the slide and discuss both of these.

We have a control room that does meet the requirements for control room habitability. On the subject of human factors and emergency procedures, I have already discussed the graphics display system that we have added to the panels to give the operator this added ability to determine the events that are going on in the plant.

What else you see on this chart, we have taken, for example, the reactor water clean-up system and drawn lines of demarcation around it, which includes the instrumentation for that system. We have increased the letter size of the system name. We have put the annunciators related to that system directly above the system, and we have increased the letters on the labeling on all of the panel, and the grouping, to assist the operator in operating the system. We have also included mimicking on the board.

If I go over to the recirculation system, for example, we have two recirculation pumps. We added symbols for the pumps, one here, one here. The flow controlled storage valve is indicated here, with a solid line to its controller. So now the operator has a graphically displayed system which has the lines of demarcation around it and with the added capability of the graphics display system. These are the things that we have done in the area of upgrading our control room.

(Slide.)

Just to put the old panel on, that's what it looked like before we did the changes. There was an array of switches and instrumentation with very small lettering. And in times of confusion just from the man-machine interface, now the operator has a much better chance to operate on that specific system.

(Slide.)

The next subject I would like to discuss is the emergency plan. We have been actively participating in generating an emergency plan. This plan received original concurrence from the state and county, as indicated here, back in 1976. We really only have three major milestones left. The big one is the major exercise which is scheduled for June of 1983. Prior to that major exercise, we have plans for 20 drills in the timeframe of February, March, April to be ready for the major drill in June of 1983.

The reason -- I guess the basis for optimism on our emergency plan is, again, that we are located on the Hanford Reservation and our 10-mile planning zone includes one of the lowest populations in number; around 1300 permanent residents in the 10-mile planning zone.

So these are the reasons that we do not expect any difficulties with the local governments of Benton

County, Franklin County or other agencies, DOE or the state. For example, the state has an active program to support the Trojan Nuclear Power Plant already in place. Again, the state and county plans were concurred with in 1976, and have been upgraded through the review of FEMA and the regional assistance committee in March and December of 1981.

MR. MARK: You have 1300 residents in the 10-mile zone. What's the distance to the closest resident? You must be nine and a half miles, or something.

MR. MARTIN: Distance to the closest resident -- I would like to defer that to Vince Everett, our Manager of Emergency Planning.

MR. SHEWMON: He was holding up five-plus fingers.

MR. EVERETT: That's three and a half miles directly east of the plant, across the river.

MR. MARK: Yes, that's the other side of the river I've never been on. Anyway, you don't have any maximally exposed persons sitting right on the edge of the fence, there.

MR. EVERETT: No, we do not.

MR. SHEWMON: Out of curiosity, would you tell me how your emergency exercises differ from those that are normally held on the site, or does the Hanford site ever hold such things?

MR. MARTIN: I'd like to refer that to Vince Everett.

MR. EVERETT: Hanford Department of Energy does hold major exercises continuously for all its contractors. In fact, they had one just two weeks ago. Ours is not a lot different, actually, it's just bigger because we have more outside agencies involved with us.

MR. MOELLER: Do you take part in the DOE exercises? Is there an integration of your capabilities?

MR. EVERETT: To date, no, we have not. We have acted as observers, critiquers, for DOE. We are involved ourselves as far as communications, where they notify us of the emergency, but to date, that's as far as we've gotten.

MR. MOELLER: I noticed in the final environmental statement that there was a letter from one of the Indian tribes asking about certain things. Are they involved in the emergency planning, or need they be?

MR. EVERETT: The Aquan Indian nation. We met with them and talked with them. They are interested in being involved in the emergency plan. They are an independent nation, so we treat them as if they are a different state. The nearest location is 35 miles from the plant in a predominantly upwind direction.

MR. EBERSOLE: What is the nearest reactor plant to you? How far is it? Is it downwind or upwind?

MR. MARTIN: FFTF would be --

MR. EBERSOLE: About a mile?

MR. MARTIN: The FFTF is approximately two to three miles.

MR. EBERSOLE: Is the meteorology towards you from it?

MR. EVERETT: The meteorology on the Hanford Reservation is as much toward them from us as it is from them to us.

MR. EBERSOLE: Are you prepared for the worst that might happen at the FFTF?

MR. EVERETT: We have procedures in place right now that say that if DOE calls us and recommends evacuation or shelter, we will immediately activate those procedures.

MR. EBERSOLE: That's on the ground of some standard releases from FFTF. I'm talking about the worst, the normal accident case. Or do you have any criteria yet for what that might be?

MR. EVERETT: DOE has analyzed FFTF accidents and established a four and a half mile emergency planning zone for evacuation.

MR. EBERSOLE: To boil it down, are you ready for a Class 9 at FFTF?

MR. EVERETT: We just have plans to evacuate.

MR. EBERSOLE: You have to have handle your plant in a Class 9 FFTF accident.

MR. EVERETT: That's the sodium and everything; the works.

MR. EBERSOLE: Thank you.

MR. MARTIN: The last two topics, one being fire protection summary statement. We do comply with the NRC requirements under the branch technical position APCSB 9.5-1, Appendix A and Appendix R of 10 CFR Part 50.

(Slide.)

MR. EBERSOLE: At this point, I guess to go back to the question -- what criteria do you use to abandon the control room function? There must be some level of degradation including refusal to act on signals that must be spurious.

MR. MARTIN: Mr. Ebersole, Chris Powers is prepared to address the remote shutdown capability --

MR. EBERSOLE: Oh. Another point --

MR. MARTIN: -- right after this presentation.

(Slide.)

The last subject is the MARK II containment. It's a unique, freestanding steel containment. It has an incline bottom and exquenchers. We have been part of the MARK II owners group and have used the hydrodynamic loads. These plant-specfic loads have all been agreed on by the NRC, and the bottom line is that they have been approved.

There have been several major modifications as a result of the testing programs. The plant-specific testing programs on the foreign tests, the plant in Italy, the Tyoko plant in Japan, and also, the testing done in San Jose. We found added stiffeners and downcomer braces, and we've made those modifications.

The vacuum breaker actuation -- we have installed dampening devices on the vacuum breakers and they were inspected during the subcommittee's visit to the site. That concludes my remarks.

MR. WARD: A question on the suppression pool, -- two questions. How do you assure some quality of the water in the suppression pool, and is the pool lined, or what's the material of the pool surface?

MR. MARTIN: The pool is painted. It's a carbon -- Maybe Ed Fredinberg -- would you address the question of the material of the freestanding steel containment?

MR. FREDINBERG: Is the question relating to the coating system?

MR. PLESSET: He's concerned with the broad question of the purity of the water in that pool; how do you keep it clean?

MR. FREDINBERG: I can't address that question.

MR. EBERSOLE: Before we get into it, let me just refine it a little bit. In some recent subcommittee meetings here, we've been finding some rather startling results about the filtration steel filters on the RHR recirculating water after loss of coolant accidents; namely, we found lighter than water insulating materials potentially heavily contaminated, possibly rust, a whole host of things that could interfere with the cooling and lubricating functions on the internal seals in the RHR pumps.

Now, your plant has the rod from the pump system and feed it to the delicate aspects of the pump's seals internals. We found a substantial possibility here that in some designs where ordinary plastic insulation is used, in fact, the filtration systems will plug or feed contaminates into the channels and destroy the seals.

The essence of this is do you have a program in place that addresses the purity of the water, including some period of time after an accident to insure that these rather delicate RHR pumps and spray pumps also as well -- their seals and journals will remain operable in the face of whatever contaminate might go into the water? It would include paint in your case, and in other cases it might include rust. It might include anything that could be determinable as coming out after a major loss of coolant accident.

MR. FREDINBERG: I've been advised that our next speak, Chris Powers, will be prepared to address that.

MR. MARTIN: Mr. Ebersole, on your question of remote shutdown that you asked earlier on the decay heat removal, the RHR operation, there was one other question specifically oxygen control. And now, on this question of purity of the containment water and these subjects, I think it would be appropriate for Chris Powers. I would like to introduce Chris who is reactor engineering supervisor on the staff.

MR. MAZUR: Don Mazur, Director of Operations, Supply System. We are prepared to answer that now. I would like to dispense with it before Mr. Powers begins.

MR. SHEWMON: Mr. Chairman, you are also going to hear from Mr. Lipinski?

MR. PLESSET: Yes, I thought we would do that at the end of Mr. Powers' presentation.

MR. SHEWMON: Can we hear the other from back there, then?

MR. MAZUR: We're prepared to do it right now, sir.

MR. TIMMONS: We are going to answer your question in two parts. One is the type of modifications we made specifically in our plant address the IGS CC concern, and secondly, within Chris Powers' presentation he will specifically address the oxygen concentration. I'm going to speak specifically to our plant modification.

Recognizing the problem that hit the plants, we made specific modifications to our recirculation lines, particularly the 12-inch riser loops that go from the horizontal cross header into the vessel. Those 12-inch risers, which have a 90^{o} elbow within them were removed and solution heat treated to remove sensitization that may have occurred during their original shock welds on all of those internal welds of that riser.

On each end of the 12-inch risers, we also went in and machined a portion of the ID from those and butted in such that when they were installed we would have a low carbon content material there at the weld interface, so we would not have sensitization or minimize sensitization en route.

Also, we used a low carbon grade material insert at the upper weld of the risers with the upper grade filler metal. So this took care of the sensitization problem on the upper part.

On the lower part, again, we did machine out a portion of the ID and laid in a 16-L material that would minimize the sensitization on the bottom part of the riser, and used a low grade weld filler metal there, also. It did leave the welds, not on the riser but on the piece coming out from the horizontal header, in a condition that did not have a low grade material.

Now, the way we've addressed that and other welds within the system which do not have a low grade material in them is that we did commit that NUREG. We stated that we would follow the augmented ISI program on these specific welds.

In addition, some other actions we took to address this issue was the removal of the bypass lines. Also, the removal of the CRD hydraulic return line. These were some specific plant changes we made to address the IGS CC condition.

MR. SHEWMON: When you get to looking at inspection procedures, I suggest you check what's been learned recently at the Nine Mile Point plant. One could summarize it not too loosely by saying that we've proven again that after inspection, the best way to find cracks is to still see where the leaks are.

I think one leak they inspected -- not too much later they started finding leaks and they went back and tried harder with probably different equipment and found that maybe they could find them. It is a very difficult problem for the inspection. The staff's heart is in the right place, but whether the equipment will do it is still not too clear.

Let me ask the staff a different thing. Amongst the things that were listed here, there was no stress control. Does the staff allow or prove any of this yet? The last thing I heard, they were worried about the durability of this, and if there is nobody here who can answer it right now, I can get an answer later.

MR. NOVAK: We have Bill Johnston here. I think he'll try to be responsive.

MR. SHEWMON: As you know, one of the concerns here, or one of the lines of defense is to get rid of tensile stresses on the welds. The Japanese have different stress control procedures. Has the staff approved any of those yet?

MR. JOHNSTON: Bill Johnston, Division of Engineering. If you're speaking of induction heating method for removing the stress and that sort of thing, yes, we've approved it as being used in some other plants. I'm not aware that it's being used in this one. I can give you the names of the plants if you'd like.

MR. SHEWMON: No, I hadn't realized that you were allowing it. Six months or a year ago there was concern about having checked it out.

MR. JOHNSTON: It is approved for use in one of the plants that I'm aware of specifically, that have committed to do it to a number of welds.

MR. SHEWMON: Okay, thank you. That's all I have, then.

MR. PLESSET: Very good, Mr. Powers. Do you want to go ahead?

MR. POWERS: I would just like to add an additional clarifying comment to the information that Mr. Timmons presented to you, concerning oxygen control. We have a program in place now to examine the advisability of controlling oxygen for two main reasons, one of which is to reduce the stress corrosion cracking problem.

We are examining startup techniques in which we can control the oxygen level in the primary loop prior to pressurization. So we have a program in place to establish oxygen levels for guidance in operating the plant.

MR. SHEWMON: One of the areas where this will buy you something is, as you know better than I, BWRs have had a spotty history on radiation exposure. They hold records of having maintenance records with the highest exposures of any plants running. Some people manage to get around this by design and operation and others set records of their own kind. So I hope that you can be in the first category.

MR. POWERS: The topics I would like to cover are the description of our AC power distribution system, our decay heat removal systems, and finally, our remote shutdown system design.

(Slide.)

The slide I have before you now locates the plant in relationship to the power distribution grid system. The plant is physically located right here (indicating). Some of the statistics on the grid size here. The total BPA installed capacity is on the order of 23,000 megawatts. The total grid capacity including all of the utilities connected within the grid approaches 55,000 megawatts.

There are 30 hydroelectric dams that comprise the installed capacity. They are located on the

Columbia River, the Snake river, the upper, middle and lower Columbia River and the Snake River.

I believe we are tied into something like 160 other locations, to 14 other grid systems that connect into the Pacific Northwest grid.

The impression I would like to leave you with before I leave this slide is that we are connected to a very large grid, a very stable grid, hydro-based, and our unit represents something on the order of 2 percent additional capacity to that grid.

(Slide.)

This next slide that I have provides somewhat more detail of the immediate vicinity of the unit. We are physically located here; this is WNP-2 (indicating). To give you an idea, here is FFTF, here are the one and four projects.

My purpose in showing you this slide is to point out, as you will see in more detail in a later slide, the output of WNP-2 is right up to the Howard Ashe station where it connects with a 550 kV transmission.

We have three connections to the 500 kV system. We have one here going to the major load center, towards Seattle. We have one leaving and go east toward the eastern BPA grid, and we have another connection down into the Bonneville area and down into the load center to the south where we export power to California.

In addition to that, we have coming from the Midway substation, which is a major 230 kV switching loop, we have incoming lines to Howard Ashe that provide our startup transformer with its power. In addition to that, we have a backup transformer that is powered from the Benton substation that's connected to the other substation.

(Slide.)

This particular slide focuses on our switchyard. I'd like to put it in perspective. Looking at it from the critical SM-7 and SM-8 buses, which provide force to our systems, we have essentially four sources of power to the SM-7 and 8 buses.

During normal plant operation, we supply 4160 volt power to SM-7 and 8 via our normal auxiliary transformer located here, which is stepped down from the output of our generator.

In addition to that, we have the 230 kV startup power coming in through TRS that feeds through SM-1 or 3 down to SM-7. In addition to that, we have the backup transformer TRV coming from the Benton substation, 150 kV that energizes SM-7 or 8. In addition to that, we have emergency standby diesels that will come on and supply all the loads off of SM-7 and 8.

I would like to stress in summary that we have four sources of power to our critical buses.

In summary, on the electrical power system, we are supported by a large diverse isolable grid which is very strongly hydrobased. Our critical systems have four sources of power. We feel that the loss of offsite power is accomodated for in our plant design. We have priority restoration of power from the BPA. We have onsite emergency diesel generator capability and we have specific operating procedures that direct the operator to manage a loss of offsite power. We feel that our AC power systems are highly reliable.

(Slide.)

MR. EBERSOLE: Chris, would you say the reliability of your power systems are probably defined at the lowest level by the earthquake vulnerability? You know, you had some small numbers like 10^{-8} and 9 and so forth, and we commented that that was inconsistent with the common mode due to failure from earthquake being about 10^{-4}. So one should go back and realistically cut them down to size, or cut them up to size.

MR. POWERS: Our reliability numbers presented at the ACRS subcommittee were based on BPA experience of over 35 years as a function of time without power. They did not include common mode failure of the seismic event.

I do want to point out that the power coming into the switchyard is coming from a number of diverse systems; at least three. We additionally have the capability to backfeed our critical buses back through our own 500 kV system, so we feel that our AC systems are quite reliable.

MR. RAY: Jesse?

MR. KERR: Was that meant to be a response to Mr. Ebersole's question about the effective earthquakes? Because it did not seem to me that it was.

MR. EBERSOLE: I thought it was about the best he could do. I might say, do you know anything about the character of transmission lines in response to earthquake loads? Do they have any design features that look at these loads? You are dependent on transmission lines. I don't know to what extent there are margins in those things. I guess I am still stuck at a realization that you've got about a 10^{-4} or thereabouts reliability of loss of AC power.

MR. RAY: Jesse, I don't think the loop system is unique in this respect. It's the same all over the world and is subject to the effective earthquakes.

MR. EBERSOLE: But it's just the illusion that one sees in the claim of 10^{-9}.

MR. KERR: It may be unique in calculating that and even apparently believing it, but it seems to me the latter would be extremely unfortunate.

MR. RAY: I think at the subcommittee meeting we shifted our belief a little bit.

MR. PLESSET: I have had the fun of watching a transmission line in an earthquake. It's spectacular.

MR. EBERSOLE: Staying up or coming down?

MR. PLESSET: It stayed up, but the fireworks were impressive.

MR. RAY: The convectors will connect. That doesn't mean they will stay up. Structures are fairly flexible, but I would never guarantee particularly these latter structures against an earthquake. They're going to survive some but not all of them.

MR. CARBON: Milt, are the transmission lines supposed to withstand SSE or anything close to that?

MR. PLESSET: I don't think so. Jerry, that's not the case?

MR. RAY: No, there are no standards against which they would qualify in that sense. I suspect that the only -- and even there, I was going to say, but I think I have my neck out -- I was going to say that underground construction is perhaps a bit more reliable, but if you have shifts in the earth you can certainly break cables that way, too. So in the last analysis, it is a hard thing to conceive of a construction that we could qualify against.

MR. SHEWMON: Where are we on the agenda?

MR. PLESSET: We are now toward the end of Mr. Powers' presentation, decay heat removal.

MR. POWERS: I would like to summarize briefly our ability to achieve cold shutdown. Very quickly, our decay heat removal path would be discharging main steam to the condenser with feedwater makeup to the RPV, using the cooling towers as the main heat sink. We would bring the reactor from rated temperature and pressure down to approximately 350°, 135 pounds and go into the normal shutdown cooling mode of RHR, which would use either -- we would circulate primary coolant to the RHR heat exchangers and circulate standby service water on the other side of the heat exchangers, and dissipate that heat energy to either the cooling towers or our spray ponds.

(Slide.)

If for some reason we became isolated from our main condenser, we have two alternative modes in which we can achieve cold shutdown, the first of which is a mode that we call steam condensing in which we take reactor steam directly to the heat exchangers, condense it there using standby service water as a cooling mechanism, and using RCIC to return to the vessel. We would depressurize the reactor again to 350^{o} or 135 pounds and go into the normal model of shutdown cooling operation.

If, for some reason, we were unable to use the shutdown cooling mode, we could go into the alternate shutdown mode, which would involve discharging steam to the suppression pool via the safety relief valves, and using the RHR heat exchangers and the standby service water to cool the pool.

MR. EBERSOLE: Is the first mode using the RCIC suggesting that you can do that at much greatly reduced need for electrical power from the diesels? However, you are dependent on one diesel to run RCIC and have electrical service even though you don't need a big pump? I'm talking about the auxiliary system to keep it cool and so forth. You do need electric pumps in conjunction with RCIC, do you not?

MR. POWERS: The RCIC provides its own cooling.

MR. EBERSOLE: It has its own cooling? It gets environmental controls and so forth?

MR. POWERS: Yes, it does.

MR. EBERSOLE: What you want is RHR exchangers, anyway.

MR. POWERS: Yes. We would need electrical power to supply standby service water.

MR. EBERSOLE: So you're eliminating the direct service water pumps.

MR. POWERS: Yes.

MR. EBERSOLE: I thought you would say you don't need as much electric power, so that mode is for the second one. Maybe you get along on one less diesel. Well, it doesn't matter. You'll eventually have to descend to the second one anyway, so I'll drop the question.

(Slide.)

MR. POWERS: On the decay heat removal systems, we have several diverse means to remove decay heat, and we feel we can keep the reactor shut down safely.

Very quickly, on our remote shutdown system design, here on the left side of the vugraph I have depicted our presently-implemented remote shutdown system capability. We have control of the RHR, DB shutdown removal system, routed through the remote shutdown panel to the control room.

We have the capability to competely disconnect through transfer switches the main control room and its impact on the control of the RHR system. If for any reason we should need to evaluate the control room, we can progress through the remote shutdown panel and bring the plant to shutdown from the remote shutdown panel.

In addition to that, we will be implementing modifications to the ARHR system that provides us a second, diverse, remote shutdown capability controlling the alternate shutdown mode I just described from the location that is independent from the remote shutdown panel. That provides us with two diverse remote shutdown capabilities.

MR. WARD: Does that mean, Chris, that the alternate mode cannot be controlled from the remote shutdown panel?

MR. POWERS: That is correct.

MR. EBERSOLE: Actually, that second system has been put there in response to some pressure I think, has it not, from the NRC to have another channel of remote shutdown apart from the single one you had in the remote shutdown panel over here?

MR. POWERS: That is correct.

MR. EBERSOLE: So you now have a rather dispersed system. Where it says "alternate location" you have instruments and controls at various places.

MR. POWERS: Dispersed in the sense that it is not in a single panel. It is within a room that is quite accessible and communication -- we do not believe communication would be a problem in controlling the plant.

MR. EBERSOLE: What do you use for alternate locations? Do you use voice communications systems?

MR. POWERS: Yes, we do.

MR. EBERSOLE: Voice-actuated?

MR. POWERS: We have two systems. We have a radio system, a walkie-talkie system, as well as a sound-powered telephone system.

MR. EBERSOLE: Is it fair to say the second mode is really a dispatching of instructions to remote operators to do things at other places?

MR. POWERS: Well, actually a number of the controls for safety relief valves, high pressure core spray, some of the critical motor-operated valves that we would need to open will be in a somewhat centralized location so we can actually perform control functions from that location as well as disperse directions to plant operations to effect other actions that are more remote from that particular location.

MR. EBERSOLE: Again, the control room is severable from the decay alternate mode of RHRA? You can shut down?

MR. POWERS: We have no transfer switches at this location.

MR. EBERSOLE: I am talking about from the control room.

MR. POWERS: We have no --

MR. EBERSOLE: That system is vulnerable to hot shorts coming in from the control room.

MR. POWERS: I do not believe it is that susceptible to shorts.

MR. EBERSOLE: I am talking about hot, energized malfunctions such as in the Brown's Ferry case. I have incoming spurious signals from the control room.

MR. SHEWMON: Jess, come on.

MR. EBERSOLE: Let's get off this. It is too detailed.

MR. CARBON: Let me ask a broad question along that line. Is there anything that can go wrong in the control room or at the remote shutdown panel or at the alternate location that would affect the operation of the others? Could something go wrong in the control room that would cause the remote shutdown panel to be inoperable or vice versa?

MR. POWERS: No, there is not.

MR. CARBON: Have we checked that out?

MR. POWERS: Yes. We have evaluated that situation. There are transfer switches that totally disconnect the remote shutdown panel. The controls that are remotely located in the plant are tightly controlled keylock switches. We basically designed the system to provide the capability to achieve cold shutdown by a number of diverse means without interaction.

MR. WARD: Let us see. Mr. Chairman, if I may be excused, this is really Jesse's question, I guess. You cannot really say that for the alternate location. A fire in the control room -- is it not conceivable that a fire in the control room, an exposure fire which damaged some electrical equipment, could render the alternate location inoperable?

MR. POWERS: It is conceivable that that could happen, but again we have the remote shutdown panel to rely on.

MR. EBERSOLE: Let me ask the Staff's position. Do you think it is adequate if you have a fire in the main control room, that that is adequate or do you not ask for diversity or redundancy?

This is sort of a halfway interpretation. We have got one channel of shutdown here which is still subject to all sorts of upsets on the control room. We have one which is separately therefrom. What is the Staff's position now on what you have to do in Appendix R?

MR. ROSENTHAL: My name is Jack Rosenthal.

Our branch insisted that RHR A side be controllable from outside the control room. We did not require the second system. We interpret Appendix R to very specifically require that one system need be operable from outside.

MR. EBERSOLE: Is that on the thesis that the likelihood of a fire is so low that one channel is enough?

MR. KOBICKI: My name is Thomas Kobicki.

Yes. Supposedly what you are saying is correct. We do feel that a single electrically isolated alternate system is satisfactory to comply with Appendix R.

MR. PLESSET: If you will leave it at that, I think Dr. Moeller had a question.

MR. MOELLER: Yes. It relates back to the previous subject. You mentioned the RCIC pump. What are the sources of power for them? You said steam and --

MR. POWERS: We have HPCS on this particular one, which is a diesel-driven third division of the diesel.

MR. MOELLER: It is a dedicated diesel?

MR. POWERS: Yes.

MR. WARD: Chris, could I ask you about procedures for use of the remote shutdown panel? Are those part of the class of procedures that you call your emergency operating procedures, or is it different?

MR. POWERS: They are in our abnormal procedures class, which is a group of procedures which is designed to provide a bridge between normal operating and emergency procedures. We would not necessarily be in an emergency condition on the symptom-based basis, should we have to use the remote shutdown system.

MR. WARD: How will the operators be trained in this sort of operation since you really cannot do it on a simulator?

MR. POWERS: That is correct. We perform a test during our power ascension test program in which that process brings us from fuel load to warranty, where we test the operability of the remote shutdown system to bring the plant from full-rated temperature and pressure to shutdown, which would provide actual operating experience to the operators in the use of that system.

MR. WARD: How do you cover six shifts of operation?

MR. POWERS: Well, we have a training program to maximize the use or the experience we will gain from the startup program in which we will repeat the various segments of the test program to provide each shift with the actual hands-on operating experience of a particular test.

MR. PLESSET: Well, I think we have got to move on. Unfortunately, your presentation was very interesting, Mr. Powers. I presume you are finished.

MR. POWERS: I am, sir.

(Laughter.)

MR. POWERS: Our next speaker will be Duane Renberger, Director of Technology. He will be discussing equipment qualification and the geology and seismology of the two.

MR. RENBERGER: The first topic is equipment qualification.

(Slide.)

As you may be aware, much of the equipment for this plant was ordered before the current IEEE 323 standard was developed, before the NUREG document was issued, and so we have been in a program of a recovering situation here, and obtaining compliance with the purpose of the NRC and industry-developed requirements.

The objectives of our program are stated here -- to confirm safety-related equipment to perform its function under postulated accident and seismic conditions. There has not always been a one-on-one agreement with the NRC. We have participated in committee meetings to work out what it means to implement certain of these requirements and the best way to implement them

We have worked our program maybe a little different from other utilities in that we have managed this program for the plant with our own staff. I have the program manager here with me today. We have undertaken a strong involvement in the industry actions relating to equipment qualification.

By doing it ourselves, we are recognizing the equipment qualification is a program that continues through plant life. You just do not do it once and say it is done, but you have to show that this equipment performs throughout this four-year period.

(Slide.)

So, briefly to summarize where we stand, we have 85 percent of the items seismically qualified at this time. We have made the submittal to the NRC this week that identifies the equipment and its qualification status and provide the basis for the environmental audit the NRC performs.

We will have all equipment seismically qualified by fuel load, which is September of '83. Eighty percent of the 1E items in the harsh environment are qualified. We have made that submittal to the NRC. The environmental audit will take place this month. We will complete the remaining qualification, obviously.

We have submitted the justification for interim operation for NRC approval prior to fuel load, which would allow startup with not all chains fully qualified, but some chains of the safety-related equipment will be qualified necessary to perform the safety functions. We will have all of the 1E items in the harsh environment qualified prior to the second refueling outage.

So this is where we stand on the program. We have made the submittals. We will have the audit very soon. We want the audits to take place when they are scheduled because it is important to us to understand that we are performing in this program in line with the expectations of th NRC.

(Slide.)

Now we will go on to geology. In the geology and seismology area there has been quite a history, obviously, at Hanford. We have the FFTF facility there. This gadget, relocated gadget, site is their supply system, Unit 2, which we are talking about today. Also, our units 1 and 4 are located there.

So there has been a long history of licensing actions over a period of time which saw Part 100 come into being at site 1, which is a mile or so away from site 2, which was granted a construction permit in accordance with those requirements. Nevertheless, it seems like there has been a lot more work done throughout the past two or three years to look at the structures in the Hanford region and to gain a better understanding of the seismic situation at the site.

To briefly orient you, the site is here. The Columbia River bends around here. The Richmond-Kenaway-Pascal area is this right in this area (indicating). The lines on this chart are the principal structures in the region -- not faults but structures. These are ridges.

So much of the work that has been going on by ourselves and the other licensees in the area has been to establish any possible earthquake mechanisms on these structures and to also investigate specific faults.

There is a fault that was identified by our own staff and consultants in a review out here by Gable Mountain that was determined to be capable. We have had to assess the impact of that on the site. There was also a fault found in a bore hole in this vicinity that we subsequently found was not capable. It was that that delayed the FSAR, but that has been resolved at this point.

(Slide.)

A brief look back at the construction permit licensing basis. It was based much like other plants were licensed at the time on the basis of intensity 7, which was an earthquake that occurred some 80 kilometers away in 1936. Assume the nearby structure at Rattlesnake Mountain was capable, even though that earthquake may not have been associated with that structure.

So it was in effect putting it closer to the site by saying it might occur on the structure and there was an increase in the intensity value and then an acceleration to 0.25G with the new mark hall slightly modified.

(Slide.)

What has been done in the operating licensing phase for this project has been to reestablish and reconfirm that .25G is adequate and conservative. By looking at those structures at the site and estimating maximum magnitude by all kinds of methodology and arriving at a deterministic magnitude assessment for the nearby structures at Rattlesnake Mountain, south of the site, and the Gable Mountain structure north of the site. Then those magnitudes resulted in a response spectra that are below the design.

Then there was a site-specific response spectra developed based on a conservative estimate of that 1936 earthquake that I mentioned and bringing it close to the site. Much of this was done in the operating license phase and evaluating the resulting response spectra. Again, it was found the plant response spectra basis was adequate.

The potential impact of small magnitude earthquakes was evaluated again through development of a site-specific response spectra. That was evaluated versus the design spectra and again the plant was found to comply.

In addition, because the site has all of these structures and there are faults in the region, the seismicity is pretty low and the probabilistic evaluation was done for the site mainly for the purpose of a long-range perspective on the potential seismic exposure at the site. So this evaluation resulted in an indication that the probability of exceeding the safe shutdown earthquake was in the range of 10^{-4}, 10^{-5}.

We have actually redone it recently, and it is about one times 10^{-5} per year, exceeding the safe shutdown earthquake. So at this point there are no open items on this topic.

MR. MOELLER: How does the SSE for Unit 2 compare to that for Units 1 and 4, as well as FFTF?

MR. RENBERGER: It is essentially the same.

MR. CARBON: You had in the slide here the probability of exceedance of the SSE was one times 10^{-4} and you say it is more like five. What sort of an error band is there? How accurate is that -- a factor of ten, twenty, a hundred?

MR. RENBERGER: I would not want to say what the accuracy is. That number that is quoted was prior to the reanalysis. We reanalyzed it and took out this fault I said we showed was non-capable. We took that out. That drove the number down

The number you see there is the average -- the potential exceedances in the study. The study carried through uncertainties, so it allows you to look at the 90 percent confidence level and so on -- the 90 percent confidence level, based on the assumptions made in the study. They are assumptions; they are not earthquake facts. They are assuming earthquakes will occur on these structures.

The 90 percent confidence level, I think, is up about a factor of 2.5 above the 3.7, four times 10^5 number. It is a little -- I am not sure. Maybe I did not answer your question.

MR. CARBON: If you did, I do not know what the answer was.

(Laughter.)

MR. PLESSET: Thank you, Mr. Renberger.

MR. EBERSOLE: When you get to this point, you ought to take the final step that says what does that mean in the context of what is the probability of surviving that earthquake with respect to functioning of the shutdown heat removal system.

MR. RENBERGER: Surviving safe shutdown?

MR. EBERSOLE: The one that was worse than that. You just came up with the probability of exceedance. That leaves you floating out in space wondering what does exceedance mean. What does "exceedance" mean?

MR. RENBERGER: In this case you are saying if there is an earthquake larger than the safe shutdown.

MR. EBERSOLE: So am I really looking, as you well know, at the probability of surviving? You did not give me that.

MR. RENBERGER: No, and we have not addressed that. I think that is very difficult. I know there are studies being made to address that subject.

MR. SHEWMON: He addressed what he had to to get his license -- what the Staff requires.

MR. PLESSET: Thank you, Mr. Renberger.

Now, Mr. Chairman, I would like the Committee's concurrence not to have a discussion of security. The Subcommittee did not have one and I would hope that the Committee will concur in passing that over at this time.

MR. SHEWMON: I do not see any waving hands.

MR. PLESSET: Before I turn the meeting back to you, I would like to ask Dr. Lipinski to briefly address a couple of points that he has looked into on behalf of the Committee.

MR. LIPINSKI: On September 3, Mr. Corcoran of the Applicant's staff made a presentation where they discussed how they were going to commission the plant. One of the vugraphs he used was a plot of reactor path power versus reactor recirculation pump flow. The plant was to start at 100 percent power and associated temperatures.

They will gradually reduce power by reducing the recirculation flow until they get to zero recirculation flow and in effect have natural circulation, and the power was shown to be 47 percent. What our agenda does not include was seeing that number about 50 percent higher than the number we had heard in the ATWS discussions. We proceeded to ask questions which could not be resolved at that meeting.

Following the meeting, a meeting was arranged by the applicant and experts from General Electric Company to discuss the ATWS issues. Under ATWS you do not maintain 47 percent power because the water level in the vessel falls. I do not have my notes here, but I think the number was in the range of 30 to 35 percent for the power under an ATWS condition with natural circulation and the recirculation pump tripped off.

So that number is the number that we have heard in the presentation of the ATWS meetings.

MR. PLESSET: That takes care of the cause of the confusion.

MR. LIPINSKI: That was followed up during the meeting. We were not talking about the same event. In one case we were talking about normal operation and the other case we were interested in was the ATWS.

MR. PLESSET: Would you address the other question?

MR. LIPINSKI: On the question of cable separation, I was asked to sit in on the meeting yesterday. Prior to the meeting I did have the letter written by the NRC specifying separation requirements and the applicant's response to the Staff letter as to how they were going to respond to them, and then at yesterday's meeting I got a draft copy of amendment 23 to the FSAR, which has more detail than either of those two letters.

There was one item that concerned me, and that involved the associated cables. Let me redefine it -- non-class 1E cables that are not separated adequately from class 1E or associated cables. In the FSAR the applicant adequately addresses this issue in terms of if they do not meet the separation requirement, do they need to redefine them or they will proceed to analyze these cables and show that they do not influence the class 1E or associated cables.

At this point it is really an inspection effort to verify that the agreed-upon requirements have been met and the plant has been constructed accordingly.

MR. PLESSET: Is that a feasible kind of construction that can be done in a reasonable time?

MR. LIPINSKI: Part of their problem is they have a color code in the control room that has 20 different colors, and balance of plant has 20 different colors, and they are not the same set such that the ability to trace wiring and do comparisons is going to be a very time-consuming task.

Now the guide that the applicant has provided in November is supposed to help make this task easier.

MR. PLESSET: Thank you, Walt. I believe he has been of help to the Committee.

MR. MOELLER: Back to the security item, I agree that the full Committee need not cover it, provided the Subcommittee has in one way or another. Has a member of our Staff or a member of the Subcommittee read the security plan for this facility and are they happy with it and they can relay that message to me?

MR. PLESSET: I plead innocence of knowledge.

Dr. Griesmeyer?

MR. MARK: You said the Staff. Did you mean NRC Staff?

MR. MOELLER: The ACRS Staff or the

Subcommittee.

MR. MARK: I am not aware of any. We will, of course, ask the NRC Staff.

MR. MOELLER: Yes. I would like to do that.

MR. MARK: The response was -- why don't we get it again?

MR. MOELLER: Could we just have that, please?

MR. EBERSOLE: Dr. Plesset, Mr. Chairman, one little residual thing I would like to leave with the Staff, I mentioned earlier the matter of the potential contaminants to the seals and journals, and we did not get to that and it is too detailed to get to that. I would like to add the following.

As you know, in all of these BWRs we inject solution poisoning after the ATWS. I have never heard it yet evaluated as to what that material might do to the ultimate end points of the system, this being only one of that. The system has numerous lines with flow checks and so forth that are potentially pluggable by any chemical that may be temperature sensitive to the remaining solution.

Bill told me that the BWR uses, I believe he said, sodium trisulphate.

MR. SHEWMON: No, no, no.

MR. EBERSOLE: What do you use for post-accident -- sodium panaborate? Is this solution sensitive to temperature? Would you have to trace it? I am looking at the potential of ultimate malfunction of the perimeters of the system -- for instance, of the seals, of the journals, of the instrumentation level devices, at the orifices. Do you follow me? Have we followed on an aspect of what happens at the system perimeters in respect to primary coolant contamination from any source, including sodium panaborate -- also including potential contaminants --

MR. SHEWMON: Why don't we wait for three questions at a time? They have an answer to that.

MR. EBERSOLE: It is a collective answer I want.

MR. SHEWMON: You may not get a collective answer. Some of us can only focus on one or two questions at a time.

MR. BIBB: Mr. Ebersole, we'll get back to you in just a few minutes on this.

MR. HODGES: Wayne Hodges from the NRC Staff.

I don't know what the exact concentration they run, but the concentrate of sodium borate in the standby liquid control system tank, they use a 13 weight percent sodium pan-borate with natural boron in it. And the temperature for starting crystallization for that solution, that 13 weight percent, is about 59 degrees Fahrenheit.

MR. EBERSOLE: So it stays in solution.

MR. HODGES: Yes. Some of the considerations being looked at have looked at going to higher concentrations, so you might get into temperatures where you would have to worry about it, any temperature below say 120 degrees Fahrenheit. So some may be required after an ATWS fix, but at present no heat tracer will be required.

MR. EBERSOLE: That still leaves the other contaminants that we talked about. So the question will still remain.

MR. PLESSET: Pan-borate seems to be well in hand, Jesse. The other is a bit open-ended.

Do you have a comment on that?

MR. TIMMONS: Specific to the WNP-2 project, one of the activities that we've done in the past with General Electric is that they did review the size of the seals and so forth in the various ECCS pump sections to determine what size particle might cause damage or not pass through the system. That particular dimension, which I'm not sure of myself at this moment, was passed to our architect-engineer and included in the mesh sizes. So that specific issue has been addressed.

Also, we do have quality controls on our paint and so forth which we use inside of our drywell and wetwell.

MR. EBERSOLE: You are telling me the screens on the suction are the last filter point for the fluid that goes to the seals internals. Many other plants use refined filters, employing hydroclones which are centrifugal separators. You don't have those?

MR. TIMMONS: Just a moment. Let me -- we'll get an answer for you in just a moment.

(Pause.)

MR. MOELLER: Could we hear the response on security while they're talking on that?

MR. GASTON: I'm Charles Gaston, the NRC Staff. I'm the reviewer for the physical security program in WNP-2. I have reviewed their submittal and visited the site and they do meet the regulatory requirements.

MR. MOELLER: Thank you.

While they're caucusing, I had a follow-up on the two other areas. I wonder if the Staff had a chance to look at the comparison of LaSalle and WNP-2 waste systems, and could either give me an answer now or -- it's not crucial to licensing, but next week would be fine, or some time later.

MR. NOVAK: Thank you. We'll take care of it.

MR. MOELLER: The other one, what was your response on control room habitability?

MR. NOVAK: What we said was we would get back to you by written memo.

MR. MOELLER: That'll be fine. Thank you.

MR. PLESSET: Does the Applicant have any other comment they wanted to make?

MR. TIMMONS: Doug Timmons, Washington Public Power Supply System.

I've been informed that we do have other specific strainers in those systems. We do not have the details. We can provide them in correspondence if you'd wish.

MR. PLESSET: Okay. Does the Staff wish to make a comment?

MR. SCHWENCER: I'd like to make a brief comment. During the ATWs the 30 percent is for a plant with a high flow HPCI system, not a high pressure core spray system, as WNP-2 has. Considering a worst case ATWS event, where you get MSIV closure so you don't have any steam-driven equipment, such as feedwater pumps, the plant is sitting in a relief valve setpoint at about pounds, then the injection capability of the high pressure core spray is only 500 gpm, and you get another 600 RCIC.

The combined flow capability of those two systems is capable of supporting a boil-away power, if you want to characterize it as such, of 11 percent. So the water level in the vessel would drop down to the point, so that the natural circulation flow rate would reach equilibrium level at about 11 percent power at this point, with the high pressure turbine injection system, high flow to high pressures, and you can get the higher power level.

But even the 30 percent tends to be on the high side and sometimes it's more like 20 to 25.

MR. PLESSET: Thank you, Wayne. Let's leave it at that.

MR. SHEWMON: We're going to have a break very soon.

MR. WARD: I have one question I would like to address to the Staff, if I may. This is a plant that has had a difficult history during its construction. I guess we haven't heard much about the reasons for that, but they are apparently related to organization and management during the construction phase.

Back ten years ago, the agency granted them a construction permit and as part of that review for the construction permit there was a review of the construction organization. Has the Staff I guess learned anything from this experience? Is there any indication that there are going to be any changes or modifications of requirements for the construction permit as a result of this experience and perhaps a few other experiences in the construction phase of plants?

MR. NOVAK: I don't think we have focused on whether or not our requirements for the issuance of a construction permit are weak in any sense. I would argue that what we have seen are a variety of reasons why plants -- if the owners of that plant choose to, we'll have a plant that has a number of problems. If you don't use good quality assurance practices, you certainly can end up with a problem.

I don't think the problems fall into any one bin. As I understand the position of the Staff, it just takes continued attention and management dedication. I would think that the independent design reviews in a sense are an attempt to backfit a program to see if there are problems in ongoing plants.

I don't think we can write a regulation that could guarantee a problem-free plant. You're just going to have to depend on the people building it to be dedicated to that point.

MR. PLESSET: Well, Mr. Chairman --

MR. CARBON: One more question, if I may.

Tom, I would agree that you can't do that, but in some of the cases, such as here, where there has been fragmentation of mechanical construction, for example, it would appear that when you have fragmentation like that that probably it is going to lead to some sort of problems.

Is there anything that would strike you as being worthwhile to try and combat a problem like that?

MR. NOVAK: I don't have any real position. I think the point that should be thought about, if there is a lesson to be learned from this application, I don't know that necessarily it is due to failure. I think if you had a good program and a good organizational structure, you could in theory build a plant using several fabricators or so forth.

I think here it was a situation where there was a loss of communication, a loss of management control over the process. It happens when you have a single constructor.

So I still go back and would probably argue that you could look for ways to reduce the likelihood of this occurring, and perhaps inform people and put out information that would suggest that these were the things that the owners and constructors of plants should be aware of.

MR. RAY: On this point of QA and the development of the poor construction program, it seems to me, Tom, that in the incipient stages of the plan for the CP it would be necessary to make sure the utility is not contracting for the QA, but they themselves are involved to a very, very extensive degree. It seems to me that that's the key, Max, to the concerns, and David, that you have voiced a few months ago.

MR. SHEWMON: Now?

MR. PLESSET: Now.

MR. SHEWMON: Do you feel we can write a letter at this meeting?

MR. PLESSET: I was going to say that the Subcommittee thought we could write a letter, but it's up to you now as a Committee to see how you really feel. So back to you.

MR. SHEWMON: Is there anybody amongst us who feels we cannot write a letter at this meeting?

(No response.)

MR. SHEWMON: Okay. Mr. Mazur, we will try to write a letter at this meeting, and we thank you for your coming in.

I would like to declare a five-minute break while we clear the meeting, and then we'll get back to work for a little bit.

(Recess.)

AFTERNOON SESSION

(1:48 p.m.)

MR. SHEWMON: Are we ready for the ECCS LOCA? We are on schedule. Let us begin.

MR. PLESSET: What we have to discuss, the first topic this afternoon, is the problem regarding some test capability for B&W events, in particular to get some confidence regarding our capability for describing small break LOCAs and the details of natural circulation. At our meeting we had Jesse Ebersole, David Ward. We had as consultants Ivan Catton, Rick Schumway and Dr. Zudans.

As you may recall, the ACRS has been concerned with the problem of describing small break LOCAs in B&W-type plants. In the past research reviews that we have written we have strongly urged a more consequence approach to getting validation of the codes for the description of these transients.

Now the view that we have held was that there should be something on the order of a SEMISCALE type facility with a capability of where the tests are being done of a research group having a close association with the experienced personnel in code validation. We felt this was terribly important. It should be very, very useful in the work that has been done in the past on

Westinghouse-type plants. It should help considerably in the validation of the description of the behavior of these plants in the small break LOCAs and other transients.

Now the problem, in my opinion, is even more acute for B&W-type plants for a variety of reasons -- the specific configuration involving the candy cane and the steam generator with the small inventory well. The owners group at B&W have made some efforts in this direction with a facility at a B&W plant called the GERDA facility, which we heard about.

We were not too terribly impressed with this, although the Germans, the RSK and so on were instrumental in getting this plant under way. They seemed very happy with this. It was unclear why, but we talked about it with them yesterday..

They also have a small facility at SRI that is quite small -- almost little more than a table-top machine -- which also had other scaling problems. And between the two facilities they feel that this is adequate. I think I can faithfully report the feelings of the Subcommittee and the consultants that they were dissatisfied with this arrangement, not only with the facilities themselves and the details of the way they were constructed, but also with the fact that we felt no great confidence in having a reasonable code validation arrangement in connection with the tests.

Now Licensing and NRR had considerable difficulty with the licensing of B&W plants, and you will hear about that too, so I would like you to be particularly concerned with those problems and with the facilities that the owners' group is proposing to use to try to resolve those problems.

Now we are not talking about an enormous amount of money -- nothing on the scale of the upper plenum test facility, of course -- but maybe we will hear from the Staff regarding what it might cost. So unless the other members of the Subcommittee would like to add some comments -- Jesse?

MR. EBERSOLE: I have nothing to add to what you said.

MR. PLESSET: Dave Ward?

MR. WARD: No.

MR. PLESSET: If that is the case, I think we can go to our agenda, which calls for -- we will do that later. We will do this first. We also had another topic which we will take up after we get through with this, which is the program that the Staff is proposing in modifying Appendix K, which you will want to hear about.

I am proposing in connection with the first topic that we plan to send a communication, most likely to the Executive Director's office, regarding this question, so you might keep that in mind because we will propose something to you later on.

Is there anything else?

MR. BOENHERT: No.

MR. PLESSET: So let us go back to the agenda here. I will call on Brian Sheron. Will you take over?

I might say by way of introducing Brian he has been burdened with this problem for a long time. It goes back -- how long ago did you write that first report on small break LOCAs for B&W, Westinghouse, Combustion Engineering?

MR. SHERON: A long time. I guess it was December of '79.

MR. PLESSET: That first report was a very good one and I thought it was very worthwhile. He has been enmeshed in this for quite a while.

(Slide.)

MR. SHERON: My name is Brian Sheron. I am Chief of the Reactor Systems Branch, NRR.

Your vugraph package has two sets of handouts in it. The first one is the one I will discuss right now, which is the status of issues related to B&W integral systems test facility. The second is the status of ANS 5.1, decay heat use in the regulatory process, which we will talk about a little later.

(Slide.)

As Dr. Plesset said, we have been examining the behavior of the B&W machine under various transient and accident conditions which can lead to two-phase conditions in the primary system. Based on calculations performed subsequent to TMI-2, the B&W machine behaves sort of uniquely compared to Westinghouse, Combustion plants with inverted U-tube steam generators.

I will discuss in a minute the more detailed phenomena we are talking about, but right now the characteristics of the plant are not well understood and the computer models used to predict these characteristics really have not been verified against integral systems tests.

We have looked at the performance of the plant from the standpoint of how does this affect safety. Why are we not worried? Why do we not shut the plants down? One can go through and look at the phenomenon question and hopefully bound it using what I would call mental experiments -- just basic knowledge of the physics. If one bounds the uncertainties -- in other words, does steam condense or what if it did not condense -- you can still argue that one is not going to perhaps uncover the core unless further mistakes or failures took place which would ultimately lead to a degraded system.

This is what we are concerned about -- that in managing small break accidents and in any transient or accident where the system goes two-phase, the operator plays a very big role not only in guiding the plant by manipulating valves, pumps, et cetera, but by just understanding what the plant is doing and not taking the wrong action, misdiagnosing a system and aggravating the event.

So I think this is really where one can summarize what our concern is. It is a rather intangible type of concern that if one does not really understand the performance of these plants under accident conditions, we feel there is a higher probability that an operator could misdiagnose symptoms, take a wrong action, and aggravate what might be a normal type of cooldown after a transient or accident event.

We have identified the need for experimental data applicable to the B&W design, as best I can remember, in early spring 1981. This is when we really started putting it on paper. We have had a series of meetings since then with B&W and the B&W owners' group regarding the basis for our data needs. We have -- up until at least October of 1981, we were not really reaching any resolution or common agreement on what was actually needed to verify the codes.

In October '81, there was a meeting with the senior NRC and B&W and utility management and as a result of that meeting it was agreed that the Staff and the technical staff of the utilities and B&W would embark on a six-month cooperative study to sort of really fine tune and identify the real issues of concern, and also to determine whether an additional experimental facility was needed to obtain the data.

(Slide.)

I point out that at that time there was no, I guess, GERDA facility that was being put forth to obtain the data. Now the six-month study ended. Although it started in October it ended in June of '82. Again, there was no common consent or agreement between the Staff and the utilities and B&W as to what constituted an appropriate experimental data base for verifying system codes against transients and accidents as applied to the B&W NSSS.

B&W owners proposed at about that time to purchase data from the GERDA facility, which is at Alliance Research facility in Ohio, and also to supply us data from SRI-II, which is Standard Research Institute. There was a facility built by EPRI right after TMI to study the accident. It is a small-scale facility. I guess the vessel itself stands maybe 60 inches in length -- five feet or so. They proposed to buy the data from the German government.

The GERDA facility is being built by BBR, which is the utility -- I am sorry, the vendor in Germany. They would buy that data and verify their codes or assess their codes against this data that they purchased. The GERDA data is proprietary. As I say, it is being sponsored by the German industry.

The NRC Staff, both NRR and RES, took a trip to GERDA in July of '82. We toured the facility, poked our nose around, climbed up and down stairs and the like. Right after that meeting there was a second senior utility and NRC management meeting held in July.

(Slide.)

Before I go on to the results of that meeting, I am going to backtrack here and try and show exactly what this phenomena is that we have been bothered about. This is a schematic of the B&W primary system. As you can see, this is the core, upper vessel, upper head region, hot legs. This is what is called the candy cane or inverted U-bends. This is the once-through steam generator, lower plenum of the generator.

This is the loop seal, cold leg piping, reactor coolant pump, and in here I have not shown it but there are vent valves right here. One is postulating a break in this region. You will note that on here I have shown -- this is not really shown to scale in elevations, but the auxiliary feedwater for the lower loop plant comes in at about this elevation above the core and sprays down on the secondary side here (indicating).

The normal water level is somewhere down in this range during normal operation. This is the pressurizer with a loop seal and, as I said, this would be a small break and conditions of the system just about at the time of the break, which I postulated in the cold leg here.

(Slide.)

Now the system will drain down and at about 300 seconds you will note that the primary system is saturated here. You will note that steam bubbles formed in the core. Okay? The pressure comes down, reaches the saturation point and water starts to flash and the pressure decay decreases. Steam formed in the core is going to collect possibly in the upper head region, as is shown here. It may branch, travel up the hot leg pipes and separate out at the top of the U-bends.

Because the break is in the cold leg, the pressure is slightly lower over on this side. One interrupts natural circulation and a broken loop first, at least according to the B&W calculations. This interruption occurs, as you see, when one fails this U-bend with steam so that the flow cannot continue. You will also note the pressurizer here is drained down to the operator would not see anything on the pressurizer level scale.

(Slide.)

Now if I go out a little further in time to about 600 seconds, you will note that when I interrupt natural circulations in the intact loop as well I have lost all steam generators. Steam generators cannot. It accumulates in the high points in the system.

Because of that, the pressure which has come down is now starting to come up. Okay? One is not removing the decay heat that is being generated in the core. Because of that, the vent valves would open. You are pressurizing up here, and steam flow could then exit to the break.

(Slide.)

Now, if one continues this process of venting steam through the vent valve, okay -- again, this is a break small enough that it cannot remove all of the decay heat in the system. It's something like, I believe it was, 1.35 inch diameter break, which is the equivalent diameter.

Dr. Shewmon?

MR. SHEWMON: Thus is a two-pump system, presumably?

MR. PLESSET: It has four pumps.

MR. SHEWMON: I guess I'm a little confused as to why some of the pumps are not still operating. Your small break is such by definition that the pumps can overwhelm it?

MR. SHERON: The reactor coolant pumps?

MR. PLESSET: He's asking about the main coolant pumps; why aren't they running?

MR. SHERON: One would presume that, due to the two failures in the system, the pumps -- the operators would have tripped the pumps.

MR. SHEWMON: All four pumps or the ones in that line?

MR. PLESSET: All four.

MR. SHEWMON: We're digressing to a different subject right now. You're painting us into a corner, and I guess I want to know whether I want to stay painted in it.

MR. SHERON: If the reactor coolant pumps were running, the calculations which have been presented to the Staff at this time indicate that if they remain running for a certain range ot small breaks, that if these pumps for some reason fail to run later during the event, later during the time, that when they stopped the two-phase mixture which was being pumped through the entire system now would collapse.

Just like at TMI, when they turned the pumps off all the two-phase mixture just kind of separated out. If the water separated out, it would uncover the core to an unacceptable degree. You would heat up past 2200. For that reason, right now the guidance being provided to the operators is essentially that which was recommended in a memorandum from B&W to their customers in July 1979, recommending that they trip the reactor coolant pumps on reactor trip and high pressure, the HPI actuation on low system pressure.

MR. SHEWMON: You are arguing this is the only way they can go and this is downhill slower than the other way?

MR. SHERON: When you say downhill --

MR. SHEWMON: We're sliding downhill. The question is just how well is the slide greased.

MR. SHERON: If the pumps were running, you would not get this pressure increase right here because you would continue to pump.

MR. PLESSET: Paul, the general expectation is you would be losing inventory at a greater rate with the pumps running and that's undesirable.

MR. SHEWMON: Is that greater than what you can feed in? Is the break defined as --

MR. WARD: Why don't you let him finish his argument.

MR. SHERON: As I pointed out, this scenario does not lead to a core uncovery as it stands. As you interrupt natural circulation --

MR. PLESSET: Paul, does that answer your question all right?

MR. SHEWMON: For now.

MR. SHERON: With the interruption of natural circulation, you are pressurizing the system. One could argue that that is just forcing water out of the break. You're draining down. The leak is greater than a high pressure injection flow, so one has a net inventory loss.

Once I drain down the system to where the auxiliary feedwater spray can now contact steam -- you will note before here that when the liquid level was up here steam is not in contact with auxiliary spray. Once I drain down to below the sparger level here, this spray level can now contact the tubes and directly condense the steam.

As I note here, once that happens one starts to depressurize the system. I go into what we call the boiler condenser mode. The steam separates, travels over the U-bend, and condenses in the generator.

MR. EBERSOLE: In the meantime, back at the steam generator everything is held at relief set pressure?

MR. SHERON: Yes.

MR. EBERSOLE: So there's no change over there, all right.

MR. SHERON: At least in the analyses.

MR. EBERSOLE: This is a pure, clean, unaltered system. You haven't done anything funny to it yet.

MR. SHERON: Right.

Okay, the pressure is starting to turn around now because I have exposed a condensing surface.

(Slide.)

Now, this is at a time greater than 3,000 seconds.

MR. OKRENT: Could I understand something again? Would you repeat your argument about the change in regular heat transfer to the feedwater spray?

MR. SHERON: What is happening is that the operator is maintaining the secondary level at a certain point. Right now I think that for a small break of this nature the operator would be instructed to raise the level to about 95 percent of the operating range.

The feedwater spray here, because there is no natural circulation, there would be no steam being condensed in this column. It would be almost subcooled water. As the water drains down, the steam that is being generated in the core and collected up here will move down such that it is below the sparger.

Once that occurs, steam in the tubes is exposed to the sparger water, the auxiliary feedwater.

MR. OKRENT: I understand now. Thank you.

(Slide.)

MR. SHERON: If one carried this out further -- and this is not based on any B&W analysis. B&W turns off their computer code at about this point, right about here when the pressure is coming down, because as the pressure comes down the leak flow decreases with pressure, the high pressure injection flow starts to increase, and one starts to recover the inventory.

Now, what happens is, as the level fills up you again cover that condensing surface, okay. Once you've covered that condensing surface, you no longer condense the steam that could be generated in the core. So you've interrupted natural circulation again. You're back where you were.

You may get a second repressurization of the system. One could argue that as you fill the system up one would condense steam at this interface. We have had a question as to whether indeed one does condense steam at a very rapid rate at this interface due to the buildup of a saturated layer here, which basically insulates this steam from the cold water coming up from the bottom.

If one does not condense the steam very rapidly, you would compress it. The pressure would come up, leak flow would increase, HPI flow would go down. You would eventually turn this around and drain down again, so you expose the condensing surface.

One might postulate that in a limit one could get a number of cycles out here before you eventually refilled the system and got back onto a natural circulation.

MR. WARD: What makes you eventually refill the system? Why doesn't this go on indefinitely?

MR. SHERON: Eventually, as you continually push more colder and colder water as the decay heat goes down, you would eventually, I think, condense more and more steam up here, until you've finally got him to a point where he would spill over.

An operator also, in reality, would perhaps attempt to start a pump to sweep the steam over into the generator. But if one were to cool down on natural circulation alone, one might expect some sort of an oscillating behavior slowly dying out. This would not be the kind of pressure that an operator would have at least very good control over.

You also note --

MR. PLESSET: You also don't know how to calculate that too well, which is another significant point.

MR. SHERON: Yes.

(Slide.)

Now, in terms of the parameters very quickly, this is a pressure trace that I've drawn on each figure.

This is the pressurizer level with time.

(Slide.)

As you can see, that's operated at about 600 seconds. The level starts to come up. This would occur over a period of say 600 to 1600 seconds or 1,000 seconds, which is something maybe about a 17 or 18-minute period you would see a steadily rising pressurizer level.

(Slide.)

This was the calculation by B&W of the level in the hot leg. As you can see, as it drops below this line one would calculate natural circulation to be lost. This is like the bottom of the U-bend. You see, you lose it first in the broken loop, then second in the retact loop.

(Slide.)

This is the general phenomenon that we are concerned about, because this was not really predicted before the TMI-2 accident. The B&W code was modified to predict this phenomenon once -- I think Mr. Michelson was the first person to bring it up. And the revision to the B&W code showed that it did take place. I think the Staff codes are equally as susceptible to criticisms as ever, because they have not been verified as well against any sort of integral system test data.

Getting back to the July meeting, the results were that the B&W owners would furnish a full description of the Geria and SRI-II facilities and data that would be obtained from them would be presented.

The owners and B&W would participate in a task group, which would be chaired by Research, to study the relative costs and benefits of three alternatives for integral systems test data. These are the German Gerda facility as it exists today and the SRI-II facility as proposed by EPRI with no modifications.

The second alternative would be an upgraded Gerda facility. Again, Gerda is about the same scale as SEMISCALE, one-fifteen hundredth. It has a 19-tube steam generator. It's scaled full height. It has one loop, not two loops. It does not have an active pump and it is scaled after the German facility, which is a raised loop plant, not a lowered loop which is the majority of the B&W plants in this country with the exception of Davis-Besse.

The Gerda alternative would be to build a SEMISCALE MOD-5 facility at Idaho. This would be similar to the existing SEMISCALE, except it would be configured after a B&W primary system.

In terms of cost for some of these, the SEMISCALE MOD-5, complete instrumentation, complete facility, I think is somewhere between 20 and $25 million. Upgraded Gerda facility, which would include putting in pumps, putting in a second loop, and perhap upgrading the instrumentation, somewhere between 10 and

$15 million is our estimate.

One could also look at the existing SEMISCALE that could be upgraded. I think that might be in the same ballpark as upgrading Gerda.

This task group is to meet and would hear presentations by EPRI, B&W, EG&G on these various alternatives, and would report back to NRR management on the alternatives what their opinion is as to their ability to get the required data.

(Slide.)

MR. EBERSOLE: Question. The thing that sort of bothers me is that all of this is in the raw system as it presently stands. If you add candycane vents and have a method to use them, you don't get this performance at all. It changes the whole system, because you eliminate the steam void at the top of the candycane.

MR. SHERON: I don't think you do. If I open a vent -- let me just find one of the pictures.

MR. EBERSOLE: I thought it was dedicated toward preserving the solid water configuration.

MR. WARD: No.

MR. SHERON: The vents were put in the high points for TMI Action Plan item 2.B.1, and that was strictly as a way of eliminating noncondensible gas from the system high points.

MR. EBERSOLE: Suppose we look at them from the standpoint of what they do if they are appropriately sized.

MR. SHERON: If you open the vent at this point, you would lower the pressure at this point. That would do nothing more -- it would relieve steam, I agree. But it would cause more water to flash right here (Indicating).

MR. EBERSOLE: Would it not, on the other hand, because of the differential pressure between it and the reactor vessel, permit the filling of that? True, it might be flashing at the orifice.

MR. SHERON: It may well prevent filling during the recovery stage.

MR. EBERSOLE: That's what I mean.

MR. SHERON: We did look. We had calculations. I think I mentioned to the Subcommittee, Los Alamos did four calculations for us on the B&W system looking at various means in trying to induce restoration of natural circulation, including opening of high point vents, pumping pumps, and secondary side depressurization. I think it was those three, and then they looked at a base case.

What they concluded, at least from the

analysis, was that none of the three got natural circulation for you. In other words, when this occurs -- and it occurs because you don't have enough water in the system to circulate at this point --

MR. EBERSOLE: Cannot that water that you do have be supported by the pressure in the main pressure vessel against a vented candycane?

MR. SHERON: It's possible, from the standpoint of the physics. But I'm saying that the calculations did not indicate that.

MR. PLESSET: I don't think that they have received that report as yet, Brian, the Los Alamos report. It was just in preliminary form, and presumably will be available soon.

They made their best effort to try to calculate it. Even so, I wouldn't have 100 percent reliability in the calculations. Either way, it just helps to have them.

MR. SHERON: Again, it's a very unique way to manage an accident, which we don't have much experience on at all, and we don't have computer codes that are verified to even say that opening the vent will work, that we have assurance that indeed that's the way it would behave in real life.

MR. EBERSOLE: Okay.

MR. PLESSET: That is a good point that you have raised. They have looked at it, Jesse.

MR. SHERON: I believe the guidelines -- for example, we have other areas which I will show in a minute. This is just one of a number of -- a small part, I guess, of a larger set of concerns.

MR. EBERSOLE: I guess mainly I'm looking at the choice one must make as to the size and mode of operation of what you put there, and just having stopped off arbitrarily for noncondensibles I think is a little premature before you look at what the potential is for doing other things. There may be a fix here, I don't know.

MR. PLESSET: We're a little bit ignorant, too, of this whole machine, what all these things might do.

MR. SHERON: The licensing issues that are, I guess, related to this whole area are -- obviously, the first one is the 2.K.3.30 resolution. This is the integral system test data needed for Staff resolution of the small break LOCA model upgrade.

Midland. We did put in our SER that we required appropriate experimental data to verify the analysis being used to support the licensing of that plant.

The ATOG review and approval is predicated to the ATOG, are to a certain extent predicated on an understanding of the primary system behavior during these abnormal transients and accidents. We have gone through the review. We are still questioning whether some of the actions or guidance being provided are appropriate based on the system response that an operator would actually see.

High point vents. 50.44, which is I think the interim hydrogen rule, requires high point vents in LWR's. Some of the B&W licensees have requested exemptions to vessel head vents for venting noncondensible gas, relying solely on the candycane vents. The reason is -- I'll just put up this schematic here -- is that if indeed this was a steam valve, the method they would use would be to depressurize the system at a calculated rate. This would expand the noncondensible gas bubble down. It would travel out the hot leg, up the hot leg piping here, and would vent it out of the high point vent here.

The theory -- I guess it sounds pretty simple, but I think in actuality one would want a demonstration that this was even a feasible process where an operator doesn't have a noncondensible gas meter to show where everything is, and he's sort of flying blind on this one. If the depressurization was too fast, one might envision that you might accumulate too much gas up here that couldn't be relieved by a high point vent, and he'll stop the circulation at that time.

MR. EBERSOLE: Brian, it seems with the break where you have it and the physical system the way you have it and under the circumstances here, with the great range of breaks, that the ultimate stable mode of operation that you would seek -- and it would require level gauges, which we have -- is to open the high point vent and have it adequate enough for a steaming rate to simply bring the vessel level down to some point that would clear the output pipe and provide whatever pressure feedwater flow you needed there, which would be progressively lower.

In short, you would settle out -- you would come down at a decreasing pressure over time, but in essence you would have a boiling water reactor in a shutdown mode.

MR. PLESSET: But you lose the inventory as you come down.

MR. EBERSOLE: It would be steam, that's all. You would not be losing liquid. You would simply be simmering through the containment.

MR. PLESSET: You would lose your liquid until you got it down to a certain degree.

MR. EBERSOLE: It's going to do that anyway.

MR. SHERON: I agree, there are a number of ways one could obviously add equipment to these loops.

MR. EBERSOLE: That looks like an uncomplicated way and one that you could believe in.

MR. SHERON: It's possible that no one proposed that to B&W.

MR. EBERSOLE: B&W never listens to any proposals.

(Slide.)

MR. SHERON: Okay. As I said, there is a joint industry-NRC task group established, chaired by Harold Sullivan of Research. They're named TAG, for Test Advisory Group. The purpose of the group is to produce a report that identifies the experimental data needs, identifies experimental and plant data presently available or to become available in the near future -- this might include Gerta or SRI-II; I think they're talking plant startup testing, natural circulation tests, except those are single-phase -- to determine the extent that this data base addresses the experimental data needs, and then to recommend any additional programs that might be needed.

I think the owners will tell you, this will include a cost-benefit analysis.

(Slide.)

As I said, the small break LOCA was sort of one facet of many-faceted problem. This is a more, I guess, complete list of what we believe the technical issues are: interruption of natural circulation on both lowered loop and raised loop plants, a general topic which one would call hot leg bubble dynamics; trapping steam at high points; hot leg flow regimes, a slope flow, bubble flow, what's going on in hot legs; operational transients; ATOG verification, are the steps being told to the operator to take to mitigate certain transients and accidents, are they appropriate, are they based on what we believe is a true system response.

Vessel thermal shock under zero flow conditions. As you know, if one has no natural circulation one doesn't have any flow in the cold leg. Cold HPI water could travel in a more or less unmixed slug towards the vessel. B&W right now calculates that internal flow through the vent valve would prevent enough mixing in the upper downcomer region to heat the HPI water before it hits the lower part of the vessel and the critical vessel welds.

I think we basically accept the flow. We would like to have a little more confirmation of the mixing and perhaps it would be even more beneficial. So I think more credit might be given in terms of the fi temperatures.

Hydraulic stability following an accident. This is the long-term depressurization. Describe the depressurization; can an operator do it?

Break isolation. If one has a break somewhere that could be eventually isolated, one interrupts natural circulation and then refills the system. Well, if you don't have natural circulation due to steam-trapped candycanes and that steam doesn't condense, you would continue to repressurize up until you opened a PORV, and you would have feed and bleed whether you like it or not.

Steam generator tube rupture. In order to maintain level on the secondary side of the generator and not overfill that generator, B&W machines have to continually steam off the faulty generator, which basically says they would continue to leak for a period of time. I think, based on what we have learned over the past few years about an operator's ability to manage steam generator tube ruptures, we would certainly like to see some experimental evidence for the once-through generator under rupture conditions.

Cold leg oscillations. This is a

code-generated problem. We're not real sure yet. I put it down here because LASL calculations showed some large temperature swings in the cold leg near the break, which could potentially affect thermal shock in terms of very cold water and very hot near the vessel. We're not even sure if it's a problem with the code or whether it is real.

Again, effect of noncondensible gases, where they go, how can one get them out of the system, how well do vents work, and the like.

MR. MARK: In connection with a couple of items on that list, what do you use for the heat curve? Do you use Appendix K or do you try to be sensible?

MR. SHERON: I think on most of the experiments that we run it's sort of like a choice of how one wants to run the tests. They can be run either way.

MR. MARK: Fission decay heat is what it is. Appendix K is what Appendix K is.

MR. SHERON: I think on the tests we would probably want best estimate.

MR. MARK: You really try to get the heat input correctly, because otherwise there is no point in making any of these calculations.

MR. SHERON: Right. We're not proposing that any test be of an Appendix K type facility.

MR. PLESSET: They are not bound by Appendix K to the same extent they are for the large break.

MR. MARK: If we're going to talk about cold shock, you better be.

MR. SHERON: No, sir. As a matter of fact, if one wants to be conservative for pressurized thermal shock one would like to run a transient right after startup, when there is no decay heat. And if one looks at steam line breaks, the worst steam line break with respect to thermal shock occurs at zero power.

MR. MARK: You would at least try to be realistic.

MR. SHERON: Yes, sir, we would very much want to be realistic on these.

The status of this test advisory group, the first meeting was held September 16th. NRC presented to the B&W owners the data interests we had, we would like to obtain from the experimental facility. B&W gave us a presentation at the Gerda facility at Alliance. EPRI gave us a presentation of the SRI-II facility at Stanford.

Our conclusions were that we think we saw progress being made and that first report coming out of that group would indeed be a joint report.

MR. SHEWMON: Wha does that mean?

MR. SHERON It would be a report that was the product of both the owners and the Staff as opposed to just a Staff report.

MR. MOELLER: But it will not include the German Gerda data, is that correct? That's proprietary?

MR. SHERON The Gerda data is -- I guess I don't know how I would -- if it is proprietary data, we would have to treat it as such. The German Gerda facility would not complete its testing, I guess, until about the summer of '83, and I would not expect to see the data available until perhaps the fall of '83, if that early.

MR. MOELLER: Help me with that. I guess I just don't have the background.

Did the Gerda facility exist in Ohio and the Germans bought in?

MR. SHERON No. As I understand it, the Germans built, paid money and had the Gerda facility built.

MR. MOELLER: They had it built from scratch?

MR. SHERON Yes.

There was a second Test Advisory Group meeting on October 4th.

(Slide.)

MR. SHERON: At this meeting I was not in attendance so I am taking the word of my staff on this. There was an agreement on the list of phenomena to be addressed by the experimental programs. Each group now, both the NRC and the B&W ownership is assigning priority to the phenomena, and they will address these both from the standpoint of how does Gerda in its present configuration that we are interested in, how would Stanford address it.

The Stanford facility --there is a very small, almost bench-top plexiglass, I believe, facility being built at the University of Maryland under the sponsorship of Research. How does that factor in in terms of getting us the data we want? How would a

Semiscale MOD-5 stack up? This is the complete works.

The B&W owners prefer right now to defer an upgraded Semiscale or upgraded MOD-5 decision until completion of the Gerda and SRI-II test commitment now.

Right now we don't have a technical basis as to why deferral is acceptable. In other words, we feel that if one looks at Gerda, one looks at SRI-II, one looks at the limitations of the facilities, one could say there are certain events, certain types of phenomena that they will address. There are certain types they will not address.

One could take the -- at least the phenomena we know it will not address today and start to determine what do we need to do to get that? How important is it to get it?

So right now there is no real clear basis, I guess in our minds as to why a deferral until the end of these testing programs before one makes a decision to go forward with more, why that is appropriate.

The next meeting of this TAG group is scheduled for November 4, 1982.

(Slide.)

MR. SHERON: Now, if satisfactory progress is made on agreeing on how to verify the research and code matters, then NRR will resolve the licensing issues that were put up on the screen here. We will treat integral systems test data as long term confirmation of the adequacy of small break and other accident analysis methods. This would be similar to the way we have treated, I guess, the ECCS models about ten years ago. As you know, the models were approved with the understanding that confirmatory data would be forthcoming from tests such as LOFT and FLECK.

Yes, sir.

MR. OKRENT: I guess I am just not quite sure what those words mean in this context. If I recall correctly, for the Appendix K and LOCA treatment, there the Staff developed, with the help of the Commissioners, an approach which they were pretty confident had enough conservatisms built in that it covered the unknowns.

Do you think you are able to do this at this stage for the particular matter you have been discussing? Do you have sufficient information at this time to do that?

MR. SHERON I do not think the problem we are trying to address is amenable to a -- I can bound the problem.

MR. OKRENT: I agree. That is why I am asking the question.

MR. SHERON I guess the answer is we have confidence right now that -- we have sufficient confidence to say that we are not going to shut a plant down.

I'm not answering your question?

MR. OKRENT: I'm interested in starting a plant up, in fact, because there is a hold at the moment on ACRS Midland, and I happen to be the Chairman, and I would like to understand when you say we are going to resolve the licensing issue, does that mean, well, for the time being we will accept it in a state where we don't know quite what is happening and we hope over the years we will understand it, or we will understand it enough that we have an approach that we are confident is safe and it is a question of being overly safe or -- again, would you have enough information?

Let me state a concern that I have. I see a growing list of separate scenarios for PWRs, each of which requires a fairly sophisticated understanding by the operator of understanding what is going on in order not to do something wrong, even assuming things are working, and perhaps not only aggravate one scenario, but you may move from one scenario to another scenario that you really do not want to do. And I cannot even tell for the scenario that you are discussing today whether the family of related failures has been looked at enough to know that you know what information you need, let alone know what the operator should be told at these ATOGs. Maybe you have done it, but I haven't heard this mentioned in this treatment.

There are secondary system effects that might go along with this particular primary, or there might be a dual primary system effect, whatever.

So when you say NRR will separately resolve the licensing issue, I find this too vague at the moment to know what it means.

Also, I can't really tell whether the Semiscale facility that you now think you want is what you will really need if and when you look at a family of events.

Have I made my concern clear?

MR. SHERON I think I understand. I hope I can answer it for you or at least attempt to.

We have looked at the ATOG program to a sufficient degree to say that if one goes on the assumption that the computer codes are indeed telling us the truth, or at least the approximate truth, then the steps in ATOG, we would have a reasonably high enough assurance that the operator has the right instructions in terms of what steps to take to properly mitigate the event.

I think to really answer your question, it is a degree of confidence that we have in what we are doing. I think we have enough confidence to say that we understand the phenomena or we can bound the uncertainties in the phenomena to an extent that we don't see, I guess, in our limited vision right now that we are going to get into any trouble. We would like to confirm that in the future, that what we are saying right now and what we are doing right now, that our perception of ATOG, that our perception of the B&W machines is indeed correct.

One would like to do that outside of the licensing process. One would not like to drag it in and look at this as a purely technical issue that one can resolve in a technical arena as opposed to the licensing arena.

MR. PLESSET: Well, I think Dr. Okrent has touched on an interesting point, that with the good Semiscale facility relates to the development of a code validation scheme, along with the tests that you get. This is why we have a fair amount of confidence, for example, in a RELAP 5 description of a transient in a Westinghouse type plant, because there has been enough validation of the code to give us a kind of confidence. And this is the kind of confidence I think Brian wants.

And to require a knowledgeable group working with the validation of the code, together with the test should be at a useful facility.

We had a very good example of that at our subcommittee meeting, a RELAP 5 description of a feed and bleed experiment at Semiscale. It just described the phenomena precisely. It was tremendously accurate. They feel very confident about using this in the full scale plant.

MR. OKRENT: I heard yesterday that you can't just take RELAP 5 and have anyone use it and get results that he would feel confident about.

MR. PLESSET: You heard right, yes.

MR. OKRENT: If that is the case, I don't myself feel like that's a validated code. In the reactor physics area, I consider a code validated, if I can use those words, if I can send it to Argon or Los Alamos or to the University of Michigan and send along a set of cross-sections and expect all of these people to get the same answer in fact, and furthermore, know that when they try to calculate a specific experiment, it is going to fall within a certain range. It is certainly not going to be validated for all reactors in any event. It will have been "validated".

So if you can't ship it arond, it already presents to me a problem that apparently you don't have.

MR. PLESSET: No question about it. But let me say that RELAP 5, for a wide spectrum of transients, is validated enough so that it can be done at other places, provided they have the right kind of machine. This is not true of TRAC, for example, as yet. This is the last stage in getting a code that we can ship off anywhere and they can run it without having to worry about it.

Right now a lot of thought has to go into the running of these codes, a lot of engineering judgment for the most part. They are cutting down on this all the time. This is why we still have to give them more money, because we are not finished.

MR. OKRENT: Right now I am more interested in the issue that we have than the specifics of the code. I guess I won't repeat my concern. I tried to give you more than one or two sentences. I can't tell the basis on which you think you can resolve this in licensing if you have not looked at a broad enough family of events to know that there are not some serious surprises. A small LOCA, for whatever reason, is a common enough or likely enough event that it is not one you would want to have a high chance of confusion by the operator on something that really exacerbates it.

MR. SHERON There are steps, for example, in ATOG. They tell the operator to pump pumps, to open high point vents. The operators are under the impression they are going to restore natural circulation. If they don't, we will have an operator getting confused and saying, gee, I thought this was going to happen.

MR. EBERSOLE: Brian, it says pump pumps or work PORVs. You have no right to do it if you are cornered because these are not devices that are supposed to withstand the environment to which they are going to be subjected.

I ask you this: if you have some doubt that you are going to lose this transfer of function to the secondary side, are you now satisfied you are not trapped into a bleed and feed mode which may work only if you so highly pressurize the cold system that you refuse to do it on those grounds?

MR. SHERON We haven't identified the scenario which leads you to a feed and bleed mode other than perhaps a small break which is subsequently isolated. And that again depends on the condensation phenomenon, how well we do or do not understand it.

MR. SHEWMON: A different question, if I may.

One of the reasons that you feel you don't want to close down reactors is the estimate that the probability of getting into this corner is low enough that it is down with other kinds of improbable events. If you have done this calculation, as you must, as a limiting one with only one HPSI, if you had the second one, does that change it significantly?

MR. SHERON We have done a calculation with Los Alamos for two HPSI's of a single break size. Their code did not show a repressurization phenomenon, but again, they showed a decay heat removal scheme under the longer term which was an internal circulation in the vessel and downcomer via the vent valves. In other words, they just did not get natural circulation as their long term cooling method.

This was -- I guess if we had thought about it for this particular scenario, we might have said, oh, yeah, that's probably what would have happened. But again we looked at it and it was of interest that the natural circulation was not the mode of heat removal for the long term. Again you are faced with a refilling of the system, the condensation or failure to condense the steam bubble and to restore the steam generators as your primary heat removal source as opposed to HPI fluid and recirculation in the vent valve.

That concludes my presentation, if there are no other questions.

MR. PLESSET: Any other questions of Brian?

(No response.)

We have another presentation on, I believe --

MR. KERR: Has serious thought been given for turning a small break into a large break?

MR. PLESSET: That's the direction which Jesse was more or less leading to, adequate PORV. Presumably, the PORV would work, discharging that way, and that would do what you are saying. Right, Jesse?

MR. EBERSOLE: Right.

MR. SHEWMON: That's in the vein that if you've got a cold, we can't treat you. But if you've got pneumonia, we can.

MR. KERR: Exactly.

MR. PLESSET: We're going to have a short presentation by Mr. Dieterick from National SMUD. He is speaking for the B&W owners group.

MR. OKRENT: I'm not sure that's quite the right analogy, Paul.

MR. EBERSOLE: I was going to say that myself.

MR. OKRENT: He might take better the situation where you had blockage in the load to the heart, and the doctor didn't know what to do about that by the methods he had available unless he opened up a new path. I think that is more like what Jesse is talking about, and has a slightly different connotation.

MR. EBERSOLE: Well, of course, it is the means employed by the boilers with great success.

MR. SHEWMON: Please go ahead, Mr. Dieterich.

MR. DIETERICH: Ever since I've been in the business we have joked inhouse about a Class 1 guillotine on the hotleg.

MR. EBERSOLE: We used to use a brass cannon as our model.

(Laughter.)

MR. DIETERICH: My name is Bob Dieterich, I am Chairman of the Analysis Subcommittee of the B&W Owners Group, and I do appreciate this opportunity to speak to you for a couple of minutes this afternoon. We do want our position explained with respect to code benchmarking in this area of small break LOCA analysis.

(Slide.)

The first slide which I've put up on the board here this afternoon is a bit of an over-simplification maybe, but it's one we've used inhouse quite a bit to discuss this issue. I think we have tried to break the concerns down into three major areas. We think of the safety items of keeping the core cooling. We analyzed that area of the phenomena. B&W approved the ECCS code and that's what we're trying to address in NUREG-0737, Item II.K.3.30 concerns.

The second area of concern I think represents the long-term phenomena following a small break. In essence, there are the steam and bubble dynamics in the hot leg, what happens to the bubble. These are problems that the operators are faced with.

We want to look at these phenomena because we think the operator needs all the assistance he can get. We think the simulator models could possibly be improved and so forth. I think Brian did an excellent job this afternoon of describing these concerns which we placed basically in this box here.

There are, of course, -- I guess one thing I forgot to mention was the CRAFT code is no longer the good code to use. The CRAFT code is very costly and time consuming to run. We are expecting to move into the RELAP-5 code for this sort of work in the near future.

There is another box of concerns, other issues out here which came up in the past. Such things as bump the pump, these oscillations that kind of came out of a code TRAC run, the small break LOCA combined with the steam line break and so forth.

(Slide.)

I guess maybe one thing I forgot to mention also is because we do feel these items are separable, we have asked the NRC staff if they would look at these for the resolution of the II.K.3.30 concerns is enough benchmarking of the CRAFT code to get that.

The second slide I have here basically shows how we would tie this program altogether, or propose to tie the problem together. There is near-term testing that's going to be done. That's a given; there's no way around it.

EPRI has committed to do additional testing at the Stanford Research Institute. Because of the interest shown, we have requested and they have agreed to move the schedule for the testing up. It will be initiated around mid-1983. There is, of course, testing to be performed by the Germans at the Alliance Research Center.

We would intend to use in the very near term, over this next year, the results of both the GERDA and the SRI-II testing to benchmark our codes. As I said, we intend to go to RELAP-5 for this long-term sort of analysis. We would hope to benchmark the data from those two tests over the next year with that data.

At the same time, we hope that the NRC would benchmark the TRAC code which is used by them to analyze these sort of events. And as additional confirmation of a small break, we are benchmarking CRAFT against the GERDA test data.

We do, as I pointed out -- I think there was a question that came up before the subcommittee meeting here last week -- we do feel that we can commit to use the RELAP-5 code to tie both the GERDA and the SRI-II testing together to tie them to actual plant behavior.

We feel quite strongly -- and I guess this is the bottom line of our position here today -- it was mentioned earlier that we as an owners group feel that GERDA and SRI data will be adequate. We do not feel this -- we have not promoted this.

Our position principally is that the data that falls out both GERDA and SRI has to be analyzed and evaluated, and that this evaluation can then be used as input to a cost-benefit analysis. That cost-benefit analysis will then dictate the needs for any future testing. I think that future testing could take any course. I think the cost-benefit analysis could show that no additional testing is necessary. It could show that maybe some separate effects testing would be prudent. I think it could show that adding a loop at the GERDA facility would be the most cost-effective way to study interaction between two loops, or it could go to the extreme of saying that a whole new facility should be designed and constructed somewhere.

I think it's premature to say that we as an owner's group have a position on what should be done. We feel strongly that this cost-benefit analysis does need to be done utilizing the data that falls out of GERDA and SRI, and I have Chuck Morgan here from B&W. I think quite a few of you are not aware of what this GERDA test facility is, and if you don't mind, he could spend just a couple of minutes and describe that facility for you.

MR. OKRENT: Before you do that, I can't tell from what you've said about a cost-benefit anlaysis what that really means in this case. How are you going to ascertain the information needs and then the effects of not having such information or improving the available information, and putting it into some kind of cost-benefit analysis. Have you done such a systematic study of this entire question? That this is a defined state of affairs? Is there something you can give me that I can read which will tell me here is just how to go at it, here are the criteria to use and so forth?

MR. DIETERICH: I think, I hope I'm answering your question. I think we know the phenomena of interest as to how important some of the phenomena is. I think there is room for argument. Some people get overly concerned about oscillations. I know from pratical experience that most of this transient behavior of this nature is quite long term, and I don't think the operator will see the plant oscillating up and down like that. I think the response will be slow.

I think the cost-benefit analysis -- what I was getting at -- maybe that's a little benefit. It's arguable. Some people might say it's a big benefit to know this behavior. Some people might say it's a small benefit. But I think we have to evaluate the data to determine where testing can best be done. I'll just use an example.

There's been some discussion over the fact that the GERDA facility does not have densitometers in the hot leg, but it has viewing ports, it has delta p instrumentation in the hotleg, and we think that will provide all the instrumentation that's needed. Maybe our test information will show we need new instrumentation. If so, that takes the weighting against GERDA and puts it another test facility. It's that kind of information that I think will come out of this evaluation of this testing.

MR. OKRENT: Is it your feeling that your group has done or has the benefit of enough analysis of different kinds of scenarios that you know all of the possible combinations of interest that the operator might be confronted with, so there aren't chances of his confusing one for another and so forth and so on?

MR. DIETERICH: I don't think we will ever know that. Appreciating the fact that operating procedures are being completely rewritten, the abnormal accident procedures are being completely rewritten to take a more symptomatic approach to accidents. He doesn't have to know it's a small break. He doesn't have to know it's a steam line break. Procedures are going to have to tell him what to do.

MR. EBERSOLE: And the vessel is chilling?

MR. DIETERICH: And the vessel is chilling, yes.

MR. MARK: Mr. Dieterich, you made a remark about the CRAC code not being the way to approach these problems anymore. I'm afraid I don't know the difference between these codes well enough.

MR. KERR: I think he said the CRAFT code. CRAFT.

MR. SHEWMON: It could still be true that you don't know the difference between the two.

MR. MARK: Yes, it's truer than I thought.

(Laughter.)

Is that because its physics are too detailed, or its arithmetic techniques are too crude that it take so much longer? Do you believe it when you get the answer or not? Something coming from CRAFT as compared with RELAP-5?

MR. DIETERICH: I should let Bob Jones back there answer that. He's our code wizard. Go ahead, Bob.

MR. JONES: I'm Bob Jones from B&W. One of the problems we have with the long-term response is we would expect the dynamics in the hotleg to be non-equilibrium in its nature. The CRAFT code is equilibrium in its formulation, and really is not adequate for that purpose.

MR. MARK: Well, that's a good enough reason to decide to do something else.

MR. JONES: That's one of the reasons. Another one is CRAFT is, indeed, a very slow code relative to existing codes that are becoming available such as RELAP-5, and we're looking along that direction also for a faster and more economical code. Also, because of the number of hours associated with this transient, and since the phenomena is rather localized it's quite possible that a simplistic treatment may be sufficient.

So there are various reasons we are looking at using RELAP-5 and other codes, and in fact, even developing how much detail do we need in doing this, and data from GERDA and SRI-II will be used to help make some of those decisions as far as what level of detail we need in these codes.

MR. MARK: Yes, but I was a little bit caught by the fact that CRAFT is too expensive. What does the calculation cost? One percent or 10 percent of what an experiment costs?

MR. JONES: I don't know, relative to an experiment, but it runs rather slow and it's like 30 to 1 in its runtime ratio. The level of detail we would normally use for Appendix K calculations, and with that discussion of rates, let's just say it's expensive. You're especially looking here at 300 hours on the computer. That is a lot of money.

MR. MARK: Okay. Especially if you don't like the physics that's in it, anyway, that's a very good combination of reasons.

MR. JONES: For the early parts of the transients, an equilibrium code is sufficient, but in the longer term it does cause a great deal of problems.

MR. DIETERICH: I think to answer one earlier question that was brought up about this evaluation, if you just kind of note the timeframes here, GERDA and SRI testing are going to be done by the end of 1984. We would hope to have some evauation ongoing by that time which would feed into this decision-making process, if you will. If you'll notice, this decision-making process could be completed out here in this 1984 timeframe, and that is the timeframe at which GERDA would become available to us from the Germans. They, of course, own that facility and we're not free to go in there tomorrow and start modifying it.

Also, as I understand it, it's about in the 1986 timeframe before testing could be commenced at the semi-scale MOD-5 Idaho. So we feels this work falls in, schedule-wise, with the programs that have been proposed.

MR. MORGAN: I am Chuck Morgan. I have had about 18 years with B&W, either developing programs or working on the verification of programs. I would like to just give a very brief explanation of the GERDA facility and its purpose.

After the TMI incident, Babcock, Brown-Bavaria, which is a licensee of B&W, is building a plant in Germany -- Muelein Kahlerish (phonetic) -- which is a 205 fuel assembly plant which looks very similar. It is a once-through steam generator and has a candy cane and so forth. They had some licensing concerns and they came to us and between both BBR and B&W we did a study of what experimental facilities were available and what we could do.

The result of that was in the fall of 1980 the BBR decided to build the facility at the Alliance Research Center, which is a part of Babcock and Wilcox's Research and Development Division, to build a test facility to provide post-small break LOCA integral effects testing capability. GERDA is an acronym from the German Garradarohr Damphaseuga Enliah (phonetic) -- which means straight tube steam generator testing facility.

I would like to say just a little bit about the Alliance Research Center. It is not anywhere near as well known, perhaps, as some of the other research facilities in the country, but they do have some very good experience. They have got fifteen years experience in testing steam generators of the once-through tube -- once-through steam generator models of the B&W design. They also have quite a bit of experience in building large-scale test facilities.

About six years ago we determined that a lot of the CHF data we were getting from outside contractors was not of very high quality. We had a concern with our 17 x 17 fuel assemblies. The Alliance Research Center designed and constructed a ten megawatt heat flux test facility. I think you realize that is a very large test facility. Even back in those days it was a multi-million dollar test facility.

We have had some outside observers go through there and say it is one of the best in the world. The backup for that, I think, is the high quality data that has come out of that test facility. We can say for sure that the scatter of that data is amongst the smallest -- is the smallest that is available in the industry.

So throughout -- another comment I would like to make is throughout the years I have been involved in code development and so forth there has always been a close cooperation between the experimentalists in the

Alliance Research Center and both the code developers and the analysts using the code at the B&W facility at Lynchburg, and in this case with GERDA we have had the additional input from the BBR engineers and their consultants.

(Slide.)

Briefly I will touch on the GERDA scaling criteria. The major concern the Germans had and I think which most of us had is what are the natural circulation capabilities of the candy cane type of arrangement with a once-through steam generator. So the first criteria was that we maintain all of the elevations. I think again in any scale facility that is not full-scale there is always going to be some compromises that have to be made in the scaling.

This is the rationale that was followed in the development of the GERDA test. The top concern was that we get the elevations scaled. The next was that we scale the important phenomena -- what are the flow regimes and so forth. Quite a bit of study was done using the information from Idaho on their scaling philosophy and other information in the literature.

We tried, although there is not complete data for the large-scale piping and so forth, we tried to preserve what we thought were the scaling criteria that would give the best flow phenomena in the hot leg. The rest of it is volume scale by the ratio of the steam generator tubes. We had 19 for the total number in the plant. This is volume-to-power scaling, roughly the same magnitude of scaling that one would have in a SEMISCALE type of test.

In order to get the irrecoverable pressure losses right, because natural circulation would depend on that, in places where the other scaling criteria required diameters that were too large so that the pressure drop would be too small, we accounted for that by putting orifices in several elevations to get the correct loop pressure drop.

So we got it all done and we get a facility in scale that looks something like that. That "proprietary" does not have to be on there. It is an old slide. It looks like a one-loop SEMISCALE, more or less. This is an existing 19-tube steam generator that we had tested before. That was one reason that we could put this loop together in roughly a year and a half and are now taking some of the initial data.

The general arrangement of GERDA is the steam generator, a simulated core. The upper plenum has a simulated reactor vent valve so that we can look at vent valves. I think over here it is one-by-one. We scale the powers of interest during a small break LOCA. There are several possible break locations, so that we can look at different break locations. There is a high point vent in here.

We can add non-condensable gases. We were very careful in the heat losses from the system because in natural circulation you want to make sure that you are not losing heat from a system so fast that you are maybe condensing some of the steam that would not condense in the actual experiment. So we have guard heating on all the hot legs. We have the HPI, as I said before, reactor vent valve simulation.

We can add auxiliary feed at either high point or low point addition. We have a level control on the system.

(Slide.)

This is a test outline. I will not go through all the details, but I would just briefly say that it starts off with several more or less separate effects tests. We will first look at the steam generator behavior, then run some simple natural circulation tests, then operate it in steady state for the boiler condensor mode so that we can get some data that would be a lit[illegible] easier to analyze before we go to the big things.

Then we look at some separate effects on refill transients. During that we will look at the refill characterization at the high point vents, on and off, and then finally, after going through these, we will look at some composite events where we go through the whole transient from the initiation through the final steady state.

Are there any question?

MR. SHEWMON: If not, we would love five minutes between the two of them.

MR. PLESSET: Yes. We have run a little over. Thank you very much.

MR. MARK: May I ask in this kind of experiment you might imagine them taking a couple of years?

MR. MORGAN: Six months, approximately.

MR. MARK: They cannot be run through on GERDA as it now exists, but only with some modification?

MR. MORGAN: All of those experiments that are on that handout sheet I gave you are with the present system.

MR. OKRENT: Why is the Staff not impressed favorably by your proposal and how do you deal with Dr. Plesset's comment about the fact that B&W plants do not all have the same orientation of steam generator and vessel?

MR. MORGAN: Let me try the first one. I am not too sure about exactly what you mean by the second one.

The first one I think there are varying degrees of concerns. One of the concerns has been the i eraction between the vent valve flow, the annulus and the upper plenum area in the cold leg -- the oscillations. In a one-by-one test, that is not really simulated. Although the vent valve is there, you do not have the other loop. That is one of the concerns.

I am a little concerned that that kind of three-dimensional phenomena is not adequately addressed in any test of the GERDA or SEMISCALE size. I think in order to look at those phenomena you will need a much larger test to adequately investigate.

I think, then, the concern on the generator, some of them being raised loop and some of them being lowered loop, since this test was done primarily with money from BBR, B&W has made some contribution to these tests, but it is primarily a German test facility. The geometric arrangement was for the raised loop plant, which is the type of plant that will be in Germany.

If you really feel that you understand the physics and the code benchmarks the physics, then I would feel pretty comfortable that elevation pressure losses, if I have got the right flow regimes and so forth in the one test, I ought to be able to predict them in the lowered loop, and it is not really that critical to do another experiment.

In the course of the experiments it may turn out that we cannot predict them well enough and we may need to do a lowered loop test. That is one of the reasons why we want to do the testing first. You are always going to have to bridge an integral system test to the plant through some sort of computer code that is verified and the people using the code -- I agree with your point that the people that use the code have got to be the ones that compare it to the experiments to show that both the code and their understanding of the nodalization has been checked out.

You will always have to make that bridge somehow. If I did it with the upper loop arrangement, I would be pretty comfortable in extrapolating that to the lowered loop plant.

MR. OKRENT: It is not obvious to me that that is the direction for extrapolation I would like to go. In fact, I guess I would feel more comfortable if things worked well the other way.

MR. MORGAN: I will not disagree with that, but the facility is there.

MR. PLESSET: Okay, thank you. I think we have gotten the Chairman a little bit disturbed by your running over. Before we go to the last topic, let me just say that what I am suggesting is to try to prepare a small letter for you, Mr. Chairman, and we can discuss that later in the meeting. In the meantime, the members can think these things over and be prepared to modify it, accept it or not, as the case might be.

Shall we go to the last item?

MR. SHEWMON: By all means.

MR. PLESSET: The last item came up and had to do with proposed changes to the ECCS rule. As you have heard, there have been rumbles about changing Appendix K. We have a presentation by Mr. Fleischman from the Staff, who will indicate to us what their present thinking is.

MR. FLEISHMAN: My name is Morton Fleishman. I'm with the Office of Regulatory Research, Division of Risk Assessment.

What I am going to be talking about is the present Staff plans for revision of the ECCS rules. I would like to emphasize that this is still in a very preliminary state. It has not received office concurrences yet. In fact, it's still being circulated within the Office of Research, so it may very well be changed.

Just to give you some brief background on the ECCS rules, many of you recall that after a number of years of meetings and discussions the Commission finally settled on putting out a notice of proposed rulemaking putting forth a two-phased approach. That advanced notice was published in the Federal Register in December 1978. It involved both the phase one and phase two.

Phase one was a short-term approach, which included basically procedural and minor technical changes which would have little impact on the overall level of conservatism of Appendix K and ECCS rules. It involved reanalysis requirements for construction permit applicants and holders, and also for applicants and holders of operating licenses. It involved return to nucleate boiling, steam cooling requirements for flooding rates below one inch per second, and a correction to the transition boiling correlation reference.

Phase two was supposed to be a more comprehensive long-term rulemaking, which would involve a number of changes, including change in the fission product decay heat rate, zircalloy oxidation rate changes, changes to the -- allowing the use of new data, new research data, and also new operating experience. It would involve an assessment of the impact of these changes on the overall level of conservativism.

As a result of the advance notice of rulemaking, we received 25 comments from industry, and I have summarized here the major comments that we received. They involve comments such as the fact that the models should be based on a realistic analysis, that the rules should permit greater flexibility to meet the criteria in the use of new research information.

Some people have recommended that the phase one scope should be expanded to include new decay heat and zircalloy oxidation data. Others suggested that the ECCS should be treated just as any other design basis accident, without all the detail that's contained in Appendix K. Finally, there were some who said that there would be no extensive rulemaking changes, just a reinterpretation of existing rule.

After this notice was published in December, shortly after that we had the Three Mile Island accident. Essentially, work was brought to a standstill. There was a moratorium on any development of the ECCS rules while we were involved with the more serious concerns of resolving the Three Mile Island problems.

This past year, in 1981, interest in revising the Appendix K rules were activated. We made a telephone survey of the previous respondents, of some of the people in industry, who basically reiterated the comments that were previously made.

(Slide.)

As a result of the comments that we received and our present thinking, our proposed actions at the moment involve -- we're proposing to proceed with phase one that was originally discussed in the advance notice of proposed rulemaking. We believe phase one will provide relief from the reanalysis requirements, while allowing the use of some new research data, and it's expected to have no extensive impact on the overall level of conservatism of the ECCS rules.

(Slide.)

As far as decay heat is concerned, we are not going to take any action on that in phase one, because we feel that's a major level of conservatism in the ECCS rule. We don't expect to change that until a thorough evaluation is completed.

Phase two plans are such that the Staff now feels that if substantial changes are recommended to be made in phase two that we will consider an overall revision of the rule, or possibly even conversion of Appendix K to a regulatory guide. So the way we are now, the feeling is that the phase two approach would most likely be revised compared to what was originally considered. We are planning on a recommended phase two approach in early 1984.

Also, concerning the new decay heat standard, General Electric has proposed using the new decay heat standard. Our present thinking -- and if you have any further questions Brian Sheron can give you more detail on that, but right now we're planning to have GE demonstrate that there's an adequate level of conservatism using the new decay heat standard.

The GESSAR -- if we approve of this, the GESSAR would be amended to include the new standard. And if utilities wish, they could request exemptions from Appendix K by referencing the revised GESSAR.

MR. SHEWMON: They could request exemption to that aspect of Appendix K, not all of Appendix K.

MR. FLEISHMAN: That's correct.

MR. KERR: What is the adequate level of concern?

MR. SHEWMON: I could answer that, but why don't you ask the Staff.

MR. FLEISHMAN: Right now, for example, the peak clad temperature that's in the regulation calculated by Appendix K is about 500 to 1,000 degrees higher than what we've calculated using best estimate. And just what would be an adequate level of conservatism, I guess I'm not sure exactly whether it would be 100 or 200 or 500 degrees.

MR. KERR: So you will wait until GE demonstrates it before you know what it is?

MR. FLEISHMAN: I guess we'll wait and see what it shows and whether we agree that that is adequate.

Do you have any comments on that, Brian?

MR. PLESSET: I think it's not fair to press him too hard on this. He is right, right now there is evidently a margin on the order of a thousand degrees F for large break LOCA between the allowed peak clad temperature and what would really occur. That is adequate by anybody's standards

MR. WARD: More than adequate.

MR. KERR: I didn't want to make a big thing out of it. I just thought perhaps if there existed some standard of what an adequate amount of conservatism is.

MR. PLESSET: This aspect of the problem will not come up for quite a while. What they're proposing is something very mild, very gentle, which really makes life a little easier and reduces paperwork. Is that an adequate way to represent it?

MR. FLEISHMAN: That's right.

MR. PLESSET: I think our margins don't change anything really serious in Appendix K. That's what they're going to work on, I guess, for the next couple of years.

MR. FLEISHMAN: That's what we're working on right now. As I say, there are some people in Research who may want to expand the scope of this phase one, but we have not made any firm decision on that. But we did send a memorandum up to the Commission indicating what our plans were. At that time it was agreed that we would just proceed with phase one the way I have just described it.

I would just like to summarize briefly what the proposed changes are that we are considering during this phase one. What I've done here is essentially summarize what the actual ECCS rule revisions would be. I would like to look at the second item first, because that's sort of more inclusive. It would apply to all applicants for and holders of construction permits and operating licenses.

Essentially, it would require that if the calculated peak clad temperature was reduced by more than 20 degrees Fahrenheit that there would be no reanalysis required. At present, any change of plus or minus 20 degrees or more would have to require a reanalysis. We are asking that no reanalysis be required if the calculated temperature is reduced. This is essentially a conservative approach. There is also the restriction that they do not take any credit for this reduced temperature in their technical specifications.

The other item would be that if the calculated peak clad temperature was increased up to 100 degrees Fahrenheit above what's in 10 CFR 50.46, they would have up to one year to resubmit the analysis. The feeling here is that the added risk is small, and our present best estimates are that the actual temperature is somewhere between 500 to 1,000. So we felt that that would be reasonable, to give them a year to come in with their reanalysis.

There was also another item, in which we would essentially define what was meant by a significant change from the calculated peak clad temperature. That's just a minor correction in Appendix K. Furthermore, a clarification for the documentation requirements would be if they had to come in with a reanalysis.

The next item would apply only to applicants for and holders of construction permits. This would essentially say that they do not have to come in with a reanalysis if the calculated peak clad temperature increased up to 200 degrees Fahrenheit above the 10 CFR 50.46 limit.

Primarily, the feeling is that this is not the final analysis, the operating parameters can be adjusted from this analysis, and that generally they could be -- the design could be modified to meet the actual requirements at the OL stage. So the feeling was we could allow them a 200-degree leeway in temperature.

The next item is return to nucleate boiling, which would allow a return to nucleate boiling during the blowdown when it was justified by the experiments. This is essentially expected to be a minor effect. I think they have estimated it would amount to something like a zero to 12 degree change in the peak clad temperature. It's essentially a more accurate way of doing the analysis.

There's another one on steam cooling requirements for flooding rates below one inch per second. Essentially, it would delete the requirement that cooling is by steam only for flooding rates below one inch per second.

The analysis would be based on experimental data with flow blockage. It would essentially eliminate the need for an unmechanistic modification to the heat transfer coefficient which currently exists in the way they do their analysis.

And finally, essentially there would be a modification to the transition boiling correlation reference that's given in Appendix K.

MR. SHEWMON: Allow me to restate your next to the last one. Remember in the fuel cooling business this ballooning exercise we went through a while back. The Staff kept calculating that these balloons would get a lot worse because they couldn't take any credit for the fact that there were liquid drops in the steam, and that was at least one of the larger effects, and therefore Appendix K sort of went the other direction from what experiments kept showing with regard to this.

Presumably, this would bring the predictions of the code somewhat closer to physical reality as we now know it.

MR. FLEISHMAN: I think that is correct.

MR. SHEWMON: If you were trying to explain this to a college junior, what would you say with regard to the delete requirements, doing it by steam only? By steam here you mean only the gas. It's 100 percent quantity, dry steam?

MR. FLEISHMAN: It essentially deletes from the regulation any mention of the one inch per second rate. In other words, the way the regulations currently read, they say if it falls below one inch per second you have to have only steam cooling. That part of the regulation would be deleted, essentially. It would just say that they would actually calculate based on experimental data just what the actual heat transfer coefficient would be, no matter what the steam cooling -- the flow rate was.

Am I correct, Brian, on that?

MR. SHEWMON: Yes.

MR. SHERON: If I could just clarify, I think there may be a misunderstanding. When one takes the less than one inch requirement, the steam cooling requirement -- use the example say with Westinghouse. The computer codes they use would first put in a flux heat transfer coefficient. They would drop their entrainment rate, et cetera.

When the flooding rate dropped below one inch per second, the rule would basically say, oops, you have to switch immediately to nothing but pure steam cooling. A computer code cannot do that if it's based on some mechanistic process.

The way Westinghouse did that was, they vaporized the droplets to steam, pure steam, for steam cooling, the thought being at the time when the rule was promulgated that the heat transfer would be degraded. There would be a penalty. What happened was, because when one vaporizes liquid to steam the velocities became so great that the heat transfer was actually better than the FLEC data.

So in order to account for the fact that the steam cooling fix was not giving results that were in agreement with experimental data, Westinghouse went and put on what they called a dynamic steam cooling model, in which they basically adjust the heat transfer in different locations in the actual rod.

MR. SHEWMON: FLEC is pure dry steam, the actual experimental data. They would actually multiply the heat transfer by one factor, I guess it's less than one, to have it match experimental data. It was just predicting the heat transfer way too high.

MR. SHEWMON: That's enough. Thank you.

MR. KERR: Could you explain that to a junior student?

MR. SHEWMON: I think I could explain what I was going to say to him before.

Thank you.

MR. FLEISHMAN: That's the end of my presentation.

MR. PLESSET: I just have a brief question. What's the Committee's reaction to this?

MR. SHEWMON: Do you think it'll do any harm, Jess?

MR. EBERSOLE: This is not my business. I think it's fine. You're out of my scope a little bit.

MR. SHEWMON: I take it nobody else is getting -- I guess we're favorable.

MR. FLEISHMAN: Thank you very much.

MR. PLESSET: Are we going to have a break?

MR. SHEWMON: I was going to suggest that, in fact insist upon it. A ten-minute break.

(Recess.)

MR. SHEWMON: Are you ready, Roger?

We're about as ready as we're going to get in the next ten minutes.

MR. MATTSON: Do you want me to go ahead?

MR. SHEWMON: Yes, please do.

MR. MATTSON: I'm going to interrupt the usually peaceful portion of your agenda to bring you reactor level indicators. You have some information that was just brought down to you today; not that we expect you to have read it overnight or over your lunch hour. But recalling there is a package somewhere in your stuff there that has that piece of paper on the front. The purpose is a recent request for Commission approval level vessel indicators.

(Slide.)

If you go back into it about two pages, you will find something that says "Actions to resolve issues." About a year ago you and staff had about decided we were not communicating with one another on this water level business. You will recall we had a December Commission meeting -- a January Commission meeting, I recall, where things kind of came all apart, and we got sent back to take a harder look at what we were trying to achieve.

In the course of February and March, we met with a number of industry representatives, had a go-around with the CRGR, came back to you in April and convinced you that we had the thing back on track and had a way to address the concerns that you and the Commissioners had expressed.

Since that time, we have been following that course and as we were getting nearer to completing it, I happened to go back and read the ACRS letter from April which said that although you agreed with the course we were on, you wanted to stay in touch. I had read that earlier to mean that you would probably call us back, but given that you hadn't and we were getting near the end I thought I had better take a few minutes, so Ray squeezed me onto the agenda here.

The rest of this slide summarizes briefly what we did for you and the CRGR to draw down the controversy. The question of ambiguity of instrumentation we had pretty well come to grips with by the time we met with you in April. That is, you can't oversell these instruments; there are times when they are just trending instruments, when there is something dynamic going on. But if you're careful in the way they're designed and the way they're put into procedures and the way you advertise them to operators, then there is a net safety benefit.

That is a conclusion which we continue to share with you with the CRGR. That is probably best summarized to date by a document I think you have just been handed within the last 10 or 15 minutes, which is the CRGR meeting minutes from meeting 19 dated September 22nd. It is about a six-page document in which is summarized the CRGR's final analysis of the inadequate core cooling package. That set of instruments includes the subcooled margin monitor, the core exit thermocouples, and now, the inventory trending monitor. Stating the bottom line of the committee, if you will, in agreement with the NRR staff, that there are significant safety benefits from supplementing that package to include the inventory trending system.

Going on to talk about a lot of cost-benefit work that has been done with the study, or done with the instrumentation in the course of studying it for CRGR, if yo go two slides back from where you are in that package it should be about the fifth page, is a third page from this CRGR letter. There is a summary of the costs associated with the various elements of the inadequate core cooling package. As an aside, an interesting thing happened in the course of the CRGR review of that package. They asked us to not only look at the costs of the various ways to design inventory trending systems; they asked us to look at alternatives for other element in the package -- the core exit thermocouples, the saturation margin, subcooling margin monitor.

We went out to the industry, the owners groups, the AIF. The vendors all supplied cost data with some spread to it. In fact, in some cases the spread was more than credible. You see some troubles on this page; for example, it appears that it is more expensive to forward-fit the subcooling monitoring than it is to backfit it. You have to be careful with the kind of data that you get by a shotgun approach for cost information because one or two high estimates can drive the population to do unusual things on the average.

These things are all explained in a Commission paper that includes a technical report that we did for the CRGR. I think you have been given a copy of that, too. It's dated August 19th. It has in it --

MR. KERR: Roger, in Table 1, are these numbers in thousands of dollars?

MR. MATTSON: Yes. Sorry. It must say that -- yes, $1000 per plant, it says down in note C.

MR. KERR: I saw that. I wasn't sure how to interpret it.

MR. MATTSON: So if you want to know more about this cost-benefit information, it's pretty well summarized in this August report in the Denton to Stello memorandum.

Well, what did we change in the course of the last year? I think we changed a number of things.

Skip two more slides. You can see I'm using slides that will be before the Commission next week when we meet with them on this.

(Slide.)

They are probably more detailed than you need, having gone through this a couple of times since the Commissioners saw it. We still think, as we did a year ago, but with a lot better proof today, that Combustion Engineering system and -- the Combustion Engineering system and the Westinghouse system are acceptable on a generic basis.

One of the things that has changed since a year ago is that we now have decided that we must do a plant-specific implementation review of each of those systems in addition to the generic review that we did. There is a tailoring, if you will, of those systems through tests that are conducted when they are installed through training that is conducted on a facility-specific basis, through the way those things are included in the procedures. There is the question of making sure that everybody who is going to begin to use these instruments has the right flavor for how they are to be depended upon.

I think what we were doing a year ago was trying to convince ourselves and the world that we had an infallible, unambiguous indication of water level, and we have now come to the conclusion that there ain't such a thing, so therefore, the way you use this system is important to its safety benefit.

On the B&W system there's been a lot of discussion about whether the one they have proposed was acceptable or whether it needed to have additional features. The principal thing there was to have the ability to detect a bubble in the upper head. They proposed that wasn't necessary for a variety of reasons we talked to you about last April. The final conclusion was that they have to have that capability.

In the course of the year, we also came to understand how these inventory trending systems work with the pumps running, in addition to how they worked with the reactor coolant pumps off, not the high pressure injection pumps. The CE and Westinghouse systems have been tested both analytically and, to the extent we could, experimentally with pumps running and found them to have quite valuable information for the operation of the plant, sufficient to convince us that not only should B&W be required to install an inventory trending system, but also, one that works with the pumps running.

That is a little more Hawkish position, if you will, than we had a year ago. We were a little uncertain about having the pumps running.

One of the things we learned in this cost-benefit aproach was things like environmental qualifications and seismic design in some plants can cost you money in ways you didn't think about. One plant -- I can't remember its name for sure -- spent on the order of three million dollars just stringing new tables for the thermocouples to make them seismic proof. They had a lot of supports and rigging to put up that they didn't have in that plant. They could't just put them in the seismic trays.

You see little things like that happen in plants where there probably are some shortcuts that could save some money and still give us some confidence that we were adding good instrumentation, and we are willing in systems that have already been installed clearly to have some flexibility in what final requirements they must meet for things like seismic and environmental qualification.

In plants that haven't yet installed equipment, we will stick with the NUREG-0737 design specs which is essentially making the system safety grade. I must say, though, that if somebody came in and told me it was going to cost four or five million dollars for the core exit thermocouples, we would take a look at it.

MR. EBERSOLE: Was that matter investigated as to why -- the reason, I recall a cable about a half a mile long that was going to be a dead circuit was going to cost $304,000 until it was discovered that all you had to do was put a rubber sheathing around it .

MR. MATTSON: It is analogous to that kind of problem, to get high radiation capability for some of this cable they go to mineral insulated cable. Mineral insulated cable doesn't shake very well without breaking; whereas, the old cable would have done fine in a shaking environment.

MR. EBERSOLE: That's an unfair cost increment.

MR. MATTSON: Those are the trade-offs. If you read the words in the Commission paper, there is some nuance that leads to a dialogue.

MR. KERR: Roger, is that the significance of 0737 design specs, that it be safety grade? I see reference to 0737 design specs.

MR. MATTSON: That's what the appendix was in 0737; essentially, the safety grade environment, fully environmentally qualified for seismic.

Another thing that is in here that was not in here a year ago is the ability to coordinate the installation and turning on of this equipment on a practical schedule basis, hopefully in concert with the things in SECY 82-111, the emergency response gear facilities procedures, that were discussed so much and fully integrated as a package by the committee to review generic requirements. So again, this will be done on that same schedule and basis.

MR. KERR: I went back recently and reread NUREG-0839 last week. Is there some reason to think that no matter what the plant is doing, that this design should be in by January 1, of 1983?

MR. MATTSON: Well, that isn't what that says. That is the bullet that says since the B&W people have been reluctant to make any commitment, any of them, to this instrumentation, we are recommending to the Commission that we issue orders to the B&W people asking them to tell us by January 1, 1983 when they're going to install the equipment. That one is up to them. They tell us what it practically takes to procure --

MR. KERR: It says, that they be ordered to conclude their design review and submit detailed insulation procurement schedules.

MR. MATTSON: Schedules by January 1, 1983.

MR. KERR: The only thing they have to submit is the schedule? I thought they had to submit detailed engineering and procurement installation schedules.

MR. MATTSON: A schedule for when they're going to do the detailed engineering procurement and installation. It's probably fair to ask more of them than that, but this doesn't.

MR. KERR: I have no reservations about them being asked to do it. I just wondered why it necessarily had to be the same schedule for all plants. I don't know that all plants may need it.

MR. MATTSON: It's the B&W plants, six or seven, and it's the schedule for when they're going to do the things listed there. We're not scheduling things the way we used to.

MR. KERR: 0839 said a particular group of people never knew what other divisions of NRC were asking that same plant to do, and they operated as if there was only division of NRC. And I am sure you wouldn't be guilty of that.

MR. MATTSON: We fixed that. We have infallible institutional arrangements now. We do much better.

This last slide reiterates something I said before -- that we do a plant-specific installation review, make sure it has been coordinated with the control room reanalysis that people are required to do under SECY 82-111 -- just the utility's portion of that, that they do not have to wait for us to complete our final review, then integration with the operator training.

MR. BENDER: Roger, is the U.S. Regulatory Commission the world leader in insisting upon this requirement, or does it appear in every other country's nuclear power plant?

MR. MATTSON: I believe the Germans told us this week it is already in their plants. I am not sure what the French have done.

MR. BENDER: What kind do they use?

MR. MATTSON: I do not know.

MR. EBERSOLE: Question?

MR. KERR: Smith made that statement during his discussion of what they are doing about Class 9 accidents, but they only have one PWR, so I am not sure --

MR. SHEWMON: They only have one what?

MR. FRALEY: One B&W reactor.

MR. SHEWMON: They mostly have BWR. That is why they never talk about boilers.

MR. EBERSOLE: It was a PWR.

MR. BENDER: Do you understand the risk of misinterpretation of this equipment?

MR. MATTSON: That is something we went into in considerable detail from really two points of view, I think -- one sort of the traditional engineering failure modes and effects analysis -- the how does it behave with the hole there versus a hole there? How does it behave under all the circumstances you can think of -- that traditional engineering try to be as complete as you can. How do you calibrate it? How often do you calibrate it? How many of them are there? What are its design requirements? What conditions can you trust it and not trust it?

That has been very thoroughly done, I think, in the past year by the vendors, by the owners groups, by our contractors at Oak Ridge, by us, CRGR, and even a couple of iterations. That is pretty good.

Then in a sort of non-traditional way GE would not need this thing if a lot of things had not gone wrong. How do you take this as a piece of information along with other pieces of information from instrumentation to follow the course of an accident, if you will, and how do operators auction that information in their mind to make their best judgment overall? What happened to the machine and how do I go from here?

That thinking is new. I think it is fairly good, but it is not as complete as the other kind of thinking. For that reason, we have said go slow. Put this thing in in ways that you know it will work, use it in the ways you know it will work to start with. After you gain confidence, begin to expand the ways that you depend on it.

To think about the next slide back, the one you were looking at, there is a slide that says potential uses. If you go to the emergency procedure guidelines for Westinghouse, for B&W and CE and look at the uses actually being made at this time, only two or three, they are the more dependable ones. Do the operators begin to light these things up and see them in operation and begin to understand them?

Rather than regulate that progression, we are going to probably --

MR. KERR: Where can one find ways in which they are being used?

MR. EBERSOLE: On that subject, I could swear there is a document open on the table at home that proposes using those things to expedite the deliberate mode to save X days in shutdown.

MR. MATTSON: That is one of the early uses people are proposing.

MR. EBERSOLE: That gets to a very heart of the matter. It is there for mistakes that you have made and you do not invite trouble with it.

MR. MATTSON: We did not call this a part of the protection system.

MR. EBERSOLE: Whatever you call it, it is there for as infrequent a need as you can think of. To then drive it into use all the time and depend on it is asking for trouble, is it not?

MR. MATTSON: There are not really many circumstances upon which you depend on it because there are not many circumstances that there is a bubble in the system.

MR. EBERSOLE: You are going to invite the bubble in the system

MR. KERR: Jesse, there is the other viewpoint that says if you use the thing occasionally it is more likely to be more reliable.

MR. MATTSON: What I think you are seeing is that in lieu of tigon tubing kind of jerry-rigged sort of thing.

MR. EBERSOLE: I do not know.

MR. MATTSON: I think that is the issue.

MR. BENDER: I think we can probably take issue with the operator deciding to use it to enable him to do things safely and learning how to interpret the thing is bound to be helpful. I am very skeptical about its uses of it for emergencies for which it was intended, but someone may figure out how to use it right.

MR. MATTSON: One specific use being proposed by the Westinghouse people, we are converging on the question of what to do with reactor coolant pumps in small break LOCAs and other events. Clearly you want to keep these pumps on, if you can keep them on, for events that are not going to damage the pumps and lose the pumps at a time when you wish you had them.

One of those events is steam generator tube ruptures. We learned from Ginna it would be nice to keep the core, instead of getting into the goosy situation of a bubble and natural circulation.

There are about -- as sure as I pick a number it will be wrong, but I think about nine Westinghouse plants for which the high pressure injection capability is insufficient at high pressure to enable you to pick a simple pressure trip point for tripping your reactor coolant pumps that would permit you to say with confidence that those pumps will run for the design basis steam generator.

Those nine Westinghouse plants are the only PWRs in the country where you cannot stick a simple pressure trip point and assure yourself that the pumps will keep running. The level indicator working with the pumps, running, telling you about the inventory trending in conjunction with pressure will be getting a simple signal to the operator that he has a steam generator tube rupture, that he is not degrading his inventory, and he can keep his pumps running. That is a specific use for nine plants.

I think they will be used in almost everybody's steam generator tube rupture procedure.

MR. SHEWMON: Are there other questions for Roger on his presentation?

(No response.)

MR. SHEWMON: It has been very interesting. Thank you.

Okay, who is making the presentation on the next item?

MR. JORDAN: I assume the next person is stress corrosion cracking at Nine Mile?

Phil, come on up and I will introduce you. This is Phil Polk. He is the licensing project manager for Nine Mile Point and will give a very brief discussion of the Nine Mile Point event and then a summary of where we are with regards to the generic implications, the utility's responses, and the Staff's action.

MR. SHEWMON: Would you help him on with the microphone since we are short of staff here?

(Slide.)

MR. POLK: In March of this year Nine Mile was shut down to replace recirculation pump seals, an outage due to leaking seals. At the completion of that outage they conducted a hydro test, a 100-pound test of the hydro system to see if the seals were installed correctly.

A decision was made to get a sample and see what type of water it was. It was found to be primary coolant system water. At that point they removed the insulation and found a total of two cracks in that line. They were pinhole cracks. It was more of a whispering as opposed to a leak.

The next step in the process -- the crack was found in the pump -- the safe end to elbow. This is called the elbow risers, so the safe end to -- well, it was found in the heat-affected zone of the weld material -- two through-wall pinhole leaks. Soon thereafter they found another leak in another safe end, and the investigation ensued.

The next step in the process was to go in and do UT examinations. Niagara-Mohawk was able to show that those two particular safe ends were cracked. Also, concurrent with that, they looked at the UT procedures from the prior refueling outage. They were the remaining ten at that facility. They had not been replaced.

They have to date replaced 24 of 34. These were being inspected in accordance with 0313 at each refueling outage as a service-sensitive piping system. The reason they had not replaced those are because they are difficult to replace. They are below the reactor core and require an extended outage to accomplish.

MR. SHEWMON: Can you tell us the difference between the procedures they used one time when they did not find them and then after they knew they were there how they found them the second time?

MR. POLK: The question that arises from all this is they had done the testing nine months prior and went back and reviewed that testing and the tapes they have of it and they were not able to show any error that they have not seen at Nine Mile prior. Now they saw it.

MR. SHEWMON: With the same equipment and the same procedures?

MR. POLK: Same equipment, same procedures they were able to find it. So that raised the question that we are --

MR. VASSELO: Don Vasselo of the Staff.

I think they raised the gain on that when they saw them. The thing is that they calibrated the instrument on the pipe that they knew had the crack.

MR. POLK: You are getting ahead of the story. Initially there was a gain setting that is a 6dB sensitivity setting. Initially they found cracks, and it was probably a question of religion. When you know you have cracks, they are easier to see. From there they raised the gain setting to 10 dB. At that point it was obvious there was a crack.

The next step in the process of crack determination. They went through an investigative process to try to decide if it was possible for a crack to propagate in a period of nine months. One of the things they looked at were high stress points in the system and the entire recirculation system, where radiation levels were acceptable.

One of those cases was the pump discharge weld right here (indicating). It is a cast stainless steel tube, rolled stainless steel weld, and there they likewise found a crack. That is a high stress area. So at that point they had the benefit of a known crack and they started using that particular weld to calibrate the instrumentation so that they rot only increased the gain but then they had a known crack which they could use as a reference.

After that they went through the entire recirculation system on two loops and finally, on August 6, submitted to us the documen tion of that. I guess the bottom line was that they had found 22 cracks in the primary coolant system.

Based on the fact that they were already down for their estimated one-year outage to replace safe ends and also the fact that continuing testing would result in burning up more UT personnel than were perhaps available, they went ahead with the decision to not only replace the safe ends but to likewise replace the entire primary coolant system.

MR. SHEWMON: Replace the whole primary coolant system? Would you explain what that means?

MR. POLK: Nine Mile Point is a five-loop plant. They have decided to replace the 28-inch piping from the inlet to the outlet and also included in that is smaller branch piping up to the first isolation valve coming off the recirc line.

MR. SHEWMON: Did they have a spare set of that sitting around waiting to be put in, or where did they get it?

MR. POLK: Between April and August they were able to, through procurement activities, locate sufficient 316-L nuclear grade material to accomplish that.

MR. SHEWMON: The mill probably does not have a long hold-waiting time for steel at this point.

MR. EBERSOLE: Are they going to keep the valves and pumps?

MR. POLK: Yes.

MR. EBERSOLE: They are stainless steel, are they not?

MR. POLK: Yes, sir. And I guess they have formally told us the outage will be a year and a half now to accomplish that work.

MR. MARK: You mentioned radiation exposure from the UT examinations. What about radiation exposure for this pipe replacement?

MR. POLK: We are in the process of evaluating that. We have not approved replacement of the recirculation piping. We have approved replacement of the safe end replacement. In the approval for safe end replacement we went through a detailed evaluation of radiation and the bottom line was that it was 2,906 personrem to accomplish this repair. That is just the safe ends.

MR. MARK: That is sort of like the steam generator replacement

MR. POLK: It is a large number, yes, sir. I was going to say the issue here is as much radiological as it was the technical aspects of replacing the piping. We saw that as the major area of concern.

MR. SHEWMON: Will they use any chemical procedures for cleaning up the pipe before they go in to replace it or take it out, or have you gotten into that yet?

MR. POLK: Yes, sir. We did that and maybe I should go through the whole evolution of what has happened to doses since that original approval.

The original approval was made on June 18. Then on August 6 they came in and asked for an expansion of scope and the replacement of the entire recirculation system. In the safe end approval they had decontaminated the entire recirculation loop using the London Nuclear process. That brought the doses down considerably.

They had originally estimated 5 to 6,000 personrem to accomplish the work. The process did not enter the reactor vessel. They put seals in the vessel and essentially put hot taps in the lines at the sectioning discharge so that they could flush the entire system. That, in conjunction with some stagnant flushing they did, brought the doses down.

When they came in and asked for approval for the recirculation replacement, they had earlier revised down the safe end dose from 3,000 to roughly 2,000 personrem.

MR. SHEWMON: This is the first time that that process will have been used on the primary system, is that right?

MR. JORDAN: I think Dresden-1 --

MR. SHEWMON: That is not the Candecon. It has been used on feedwater -- whatever you call these systems -- primary system cleanup units before that. But there the argument was this was not part of the primary pressure boundary. It is just something that you use optionally. Here at least you will have the experience. That is a pump. The pump at least will go back in after this.

MR. FOLK: In the prior refueling, Nine Mile uses Candecon to decon the pump between the pump isolation valves.

MR. SHEWMON: Chet?

MR. SIESS: Why was it so much easier to approve that decontamination than it was the Dresden-1 decontamination, which I am not sure is approved yet?

MR. SHEWMON: At least one of the things is it is a much higher concentration. Perhaps it removes metal more?

MR. SIESS: This or Dresden? Did they pick the wrong process?

MR. SHEWMON: With the benefit of hindsight, the Canadians have used this a lot. Another advantage of the Candecon is the cleanup, at least up to now, had gone into regular resin beds, whereas at Dresden you have got this whole thing of having a lot of waste volume and --

MR. SIESS: But still they approved this in a matter of months or weeks and Dresden was a matter of years.

MR. POLK: There was a NUREG published on this and the position we took with the licensee was that if it extended beyond the recirculation loops and if the fluid entered the reactor vessel then it would open the subject up to an increased review.

MR. SIESS: Why is that -- because it is clad with stainless? The pipes are stainless, or is it not clad with stainless?

MR. POLK: Inside the reactor vessel I am not sure what the boundaries were, but it was my understanding that we had agreed to advise you if the process was used to decontaminate steam generators for decontaminated reactor vessels.

MR. SIESS: I guess I just do not understand why the main recirculation piping is so much less -- either less sensitive or less important.

MR. KERR: Aren't they going to replace it? I thought they were going to replace it all.

MR. SIESS: The Staff has not approved replacing it yet. That is what we were just told, so my next question was what things are you considering in deciding whether or not to let them replace it.

MR. POLK: If I could finish the dose considerations, that was the main consideration.

MR. SIESS: That is one of the reasons, is the doses?

MR. POLK: Yes, sir.

On August 6 they revised down to 2000 person rem. A lot of that was as a result of doses streaming from the primary vessel once the piping has been removed.

For the upper nozzle it is not too bad because the fuel has been removed and the water level has been brought down to here (Indicating). You still have some shielding available in the proximity -- the picture is slightly a misnomer. Most of the workers expect it to be down here. So when they removed the upper nozzles, the dose rates were acceptable. But when you remove the lower nozzle, you have to drain all the way down to here, and the doses they estimated were about a factor of 10 high. So the end result was it did come down to about 2000 at that point. Now they have come in and requested the replacement of the entire loop, and because of the way the work was accomplished, the dose was further reduced to 1500 person rem.

The reason for that is that when you replace a safe end, you have two precision cuts here as well as two welds. When you replace the entire system, you either have one precision cut here, one precision weld, but the second cut and welding would be further removed from the reactor vessel and you can have more shielding.

So the estimate now which we have under review is 1500 man rem for the safe end replacement, and their estimate is 350 person rem for the primary coolant system. The primary coolant system is basically clean at this point.

MR. SIESS: On what basis do you decide that it would be all right to replace that at 150 man rem but it would not be all right to replace it if it costs 1500 man rem? I assume they are replacing it in order to reduce the probability of a rupture of one of those pipes.

Is that the reason for replacing it?

MR. POLK: The Licensee never concluded that it was absolutely necessary to change the piping. They felt it was prudent in that they had replaced sensitized safe ends that needed to be replaced.

MR. SIESS: Prudent in terms of his investment or prudent in terms of the public health and safety?

MR. POLK: I don't believe I can answer that.

MR. SIESS: So you can't answer my other question as to how you would decide whether --

MR. POLK: They had gone far enough with the investigation that the safe ends were known to be a problem, and they were going to be down for a year.

MR. SIESS: A problem to public health and safety or to his reliability?

MR. POLK: The safe ends are definitely a known problem to the public health and safety identified in NUREG-0737 and the requirement would be replacement.

MR. BENDER: Could I just ask a little bit about the occupational exposure?

How does it compare with the potential exposure for steam generator replacement in PWRs?

MR. POLK: I don't think I can answer that. I can answer with respect to Nine Mile's track record over the last ten years in terms of normal exposure. One of the thoughts we had was that the plant on average over the last ten years was subjecting the workers to approximately 2000 person rem and obviously during this year there would be no exposure like that. It was almost as if it were a surrogate exposure as part of the clean-up.

In terms of steam generator replacement, I am not familiar with those numbers.

MR. BENDER: It is just an interesting piece of information.

MR. SIESS: These are some of the same people that will be having the ALARA principles to safety goals.

MR. MOELLER: I was just going to comment on that. Presumably by replacing this, then the doses for the first few years afterwards will be quite reduced.

MR. SHEWMON: It will also increase the amount of inspection they have to do, I suspect, in addition to primary loop activity being down.

MR. MOELLER: A question on the steam generators. The numbers we were given for Surrey was about 2000 person rem to do a plant.

MR. VASSELO: The numbers for the steam generators? I think on the Surrey generators it was fairly close to 2900 for the two -- 2900 each, for each unit that was done. There was a repair program on San Onofre that was somewhere in the vicinity of 3500 rem.

Maybe I will take a crack at answering your question, Dr. Siess, on determining how much exposure you would allow in connection with this particular program, we did perform an environmental impact assessment modeled after the Surrey study, and it was related to the BEIR report and other data to establish a connection with the risk to the worker and the public. So there was a tie-in to the potential for impact on a worker, and we do have that, and it was published as a supporting document to this amendment.

MR. SIESS: Did it involve at any point a reduction in risk to the public? Wasn't that one of the benefits?

I know what the costs are in dollars and man rems and lost power generation, etc., but I am still trying to find out what the benefits are of doing this. I assume there are two sides, costs and benefits.

MR. VASSELO: We didn't, to our knowledge, do it in that particular way for this approval.

MR. POLK: The Licensee did this of its own volition. We did not demand it.

MR. SIESS: That's beside the point. I asked why, and I --

MR. SHEWMON: You got a couple of answers.

MR. SIESS: I didn't get an answer from him as to why the Licensee wanted to do this, whether it was improving his operating conditions, his reliability to the public health and safety or what?

MR. OKRENT: I am mystified. It seems to me cracks are the way you start towards LOCAs.

MR. SIESS: That's what I think, but I didn't get that answer.

MR. SHEWMON: You got it from several of us. Why are you unhappy?

MR. SIESS: Because Niagara Mohawk-- -

MR. SHEWMON: He doesn't deal with Niagara Mohawk. We can write them a letter. I would commit them, if they were doing it, to reduce the risk to the public.

MR. SHEWMON: They certainly want to reduce the exposure to their people intermediately.

MR. VASSELO: For the safe end, we found cracks in it, and we have found all other utilities that have cracks in it to do something about it. So it wasn't from the safe end of their own volition to replace them. They were going to have to do something about them.

All other plants that have had protosynthesized stainless steel have either replaced them or clad them with stainless.

MR. SIESS: Why did you want the safe ends replaced, to reduce risk?

MR. POLK: Sure.

MR. VASSELO: Well because it goes along with having cracks, cracks being opened, and it can result in hazard to the plant to go public.

MR. KERR: Mr. Siess, he said it was in NUREG what was it, 0331? That's why he wanted to do it. It says so in the NUREG.

MR. SHEWMON: Can we now go to Dr. Okrent's question, please?

MR. OKRENT: I would like to understand a little bit more the implications of them having missed these cracks and then having seen them when they knew there was a leak.

I have looked quickly at the notice which says a little bit about it.

What does it mean when --

MR. KERR: Excuse me. Let me try to establish facts, if there are any.

Is it clear that they missed the cracks, or is it possible that the cracks were not there?

MR. SHEWMON: Improbable --

MR. JORDAN: It's possible that the cracks were there. There were some indications in their log, but not of the degree that they classified as cracks. So indications until called by an interpreter are just indications, they were not called cracks. They did use some more sensitive techniques subsequently on the safe ends after they saw the cracks, and they were able to, by UT inspection, see them.

The emphasis -- the reason the Staff is interested and I think the reasons we would like to discuss it further with you is not so much the safe ends but the recirculation piping itself. This is the first U.S. plant that has had substantial cracking in large diameter recirculation piping. KRB did have cracking, but this is the first U.S. plant.

So the lines that were inspected, the welds that were inspected, every inspected weld had cracks in it. I guess the deepest crack was on the order of a half inch in a total wall thickness of 1.3. The other cracks were very, very small, ranging to that magnitude. But that is what upset the Licensee and upset the Staff with regards to the degree of cracking, the extent of cracking.

MR. OKRENT: It would upset me if I tried to figure out the likelihood of having those systems leakproof, given my safe shutdown earthquake or something larger, for example. You can think of all kinds of ramifications.

I guess I am interested in knowing what the --

MR. JORDAN: The technique is exactly the Staff's concern, so let me jump ahead, then, to say that the --

MR. SHEWMON: It is nice, gooey stuff.

MR. JORDAN: Yes, it is.

MR. SHEWMON: Is that a metallurgical term?

MR. JORDAN: The root condition, the grain structure, the surface preparation, the interior surface as far as its roughness and grinding, all of those things contribute to a very difficult inspection. The inspections are geared -- and they meet the standards, they meet the code in this case, but the inspections are geared for perhaps initial construction, looking for flaws in the welding and have not been so geared in the thick wall pipe to looking for cracks, intergranular typew cracks.

MR. SHEWMON: I don't disagree with any of the answers you are giving. I think they are all good answers, but in connection with the pressurized thermal shock, we will have in both the NDT experts on the Staff and some from industry who have been doing in-service inspection for Southwest Research. I think those are people who are more familiar with the details of the code and what the specs are, and I suggest that we save some of these good questions for tomorrow.

MR. JORDAN: Good. I'll appreciate that.

MR. SHEWMON: But with regard to details of what the technique can do and what we should be doing, that is a better forum.

MR. JORDAN: And the technique of whether they use a particular size crystal, a given frequency, a pitch-catch arrangement or different gains and whether they calibrate against a notch or a drill hole or a crack, those all influence the ability of the operator to see it.

The staff's view at this point is the best test is one in which the individuals who are doing the examination have seen cracks. So industry has responded. There is a crack sample from Nine Mile Point now set up at Columbus, Ohio -- I'm sorry -- Batelle Columbus.

MR. SHEWMON: That's in Columbus.

MR. JORDAN: It is in Columbus. EPRI has characterized the cracks. We have staff at the location now, and the utilities are going to bring their representatives through to examine that particular sample to characterize the cracking that is there, compare what they characterize with what EPRI has found, and by that means, normalize the methods that they have been using against this known crack sample.

Now, the actions that we are talking about are for eight plants that are presently in outages or expect to be in outages between now and January 31. This set of plants fortuitously are the older boiling water reactors. They have a longer operating history, the Millstones, the Oyster Creeks, Dresdens, Quad Cities and so on. It is fortuitous in this case that we have that set of plants down.

MR. EBERSOLE: I was just going to ask how many operational years does this plant have? Is this a forerunner of others to come?

MR. JORDAN: It may be.

MR. SHEWMON: How many years?

MR. POLK: Went on line in '69 and has ten full power years.

MR. EBERSOLE: It's not nice to do this every ten years.

Are the new pipes going to be better?

MR. JORDAN: These are 316L that they are replacing.

MR. EBERSOLE: To get to the crux of the matter, why aren't they carbon steel?

MR. SHEWMON: The wrong forum, Jesse. Ask tomorrow if you want to.

MR. OKRENT: Well, I would like to know when the ACRS will hear a more detailed discussion than we are getting today and apparently can get today of the generic implications of this which I think are fairly significant, more than one.

MR. JORDAN: Yes. What I wanted to bring to you is not an evaluation of the generic significance but the present information that we have and the actions that both industry and Staff are taking in order to obtain the answers that you want.

So the actions we are taking is for this set of plants that are down, they will normalize, qualify their inspection program against a technique that finds cracks. The NRC will look over their shoulder during this, agree or disagree with the inspection methodology they are using. Each of the utilities is taking a sample of at least seven welds in these large bore pipes. Monticello has inspected or is inspecting every weld because they have an opportunity in replacing insulation, so we have a very large sample of a plant not quite as old as Nine Mile but with quite a bit of history on it.

So the results of this set of plants will then enable the Staff to be able to make a determination, and industry, as to whether Nine Mile is unique for some reason that we haven't yet fathomed, or whether these other plants have the same problems, and the in-service inspection program that is presently being conducted is perhaps not sufficient.

MR. SHEWMON: Mike?

MR. BENEDER: Ed, having heard a little bit about this problem from some of our international contacts, I am led to ask whether we are in a position to say that the inspection techniques will find the things we are concerned about.

Do you know?

MR. JORDAN: The inspection techniques? That's the reason for this normalization we are going through with EPRI.

MR. PENDER: That's fine. I like that. But I am still left with the point that you have got a number of Licensees that are already going through the inspection process, and I am wondering whether when you get done you will know enough to be able to say, well, the inspection was done, it found something, and it found all it needed to find, or whether you will still be stuck with the problem of saying, well it wasn't really normalized yet.

Do you understand my point?

MR. JORDAN: Yes, and before those plants resume operation they will have done an inspection sampling with a technique that does identify the cracks. So what I am saying is, if they go through, they have done seven welds already, they go to the EPRI facility and normalize their technique and it does not find cracks, then they have to redo that inspection.

GE and EPRI both feel that they have techniques that will find and characterize the cracks. They put more of the emphasis on the operator himself and his perseverance and his understanding of what he is really looking for in terms of cracks, as opposed to not just notches or drillholes.

MR. LEWIS: Just a quick question, which may have been asked when I was out of the room, and tomorrow may be a better forum. But in the report, it says that the technique used was the same that GE recommended, except that the gain was set lower than the GE-recommended technique.

In specifying the kind of ultrasonic inspection that has to be done by each licensee, does the NRC specify it to the level of specifying the gains, the dial settings and so forth? Or is each licensee free to do ultrasonic testing in the way that he prefers?

MR. JORDAN: There is a code requirement, which is loose. Within the code, the licensee is free to do what he chooses.

MR. LEWIS: This is for my own information. So that when somebody tells me an ultrasonic inspection has been performed on a set of pipes or pressure vessels, it doesn't mean a thing; is that what you're telling me?

MR. JORDAN: No, I'm saying with regard especially to thick-wall stainless, with the kind of root geometries they have it is very difficult.

MR. LEWIS: I'm asking for specifics. I remember when we first put the gun detectors in airports during the hijacking craze, we let the airlines set the gains any way they wanted, and they always set the things as very insensitive when there was a big crowd ready to go through the thing, then they set the gain up when there was nobody getting on the airplane. We stopped that very quickly.

But I wonder if one doesn't have a problem of that kind here.

MR. JORDAN: I would not want to destroy your confidence in the other --

MR. LEWIS: My confidence won't make reactors safe or unsafe.

MR. JORDAN: I agree. The Staff still has some confidence in the techniques other than in the thick-wall pipe at this point.

MR. LEWIS: Except that the operator is free to set his own within the code.

MR. JORDAN: Within bounds.

MR. OKRENT: Who is it within the Staff that asks, before something happens like this or the incidents we had where sampling techniques on bolts were really completely inappropriate, who is it that tries to see whether in fact by having adopted something or created some particular standard, that that is really okay? Is my question clear?

MR. JORDAN: In terms of the adequacy of a commitment to a given code or standard?

MR. OKRENT: Sure. You all felt that these safe ends had been inspected in an acceptable way.

MR. JORDAN: The Section 11 requirements --

MR. OKRENT: You now have some reservations as to whether or not this was really --

MR. KERR: I repeat: Do we know that cracks were there that were not seen?

MR. SHEWMON: Yes. They don't grow that fast. It's not a long incubation time and then all of a sudden they go in a hurry.

MR. JORDAN: I would agree.

MR. LEWIS: When you turn up the gain and see something you didn't see before, that's the presumption.

MR. KERR: I was told that they saw these, then they turned up the gain and it was very obvious.

MR. POLK: They saw a leak.

MR. KERR: I thought you said they went around with the normal setting.

MR. POLK: They saw an indication by increasing the gain.

MR. SHEWMON: He talked about indications. One of the problems with -- and we've been over this before on the main pressure vessel, which is an easier case than stainless steel -- the ASME people have a code which allows a lot of things to pass.

MR. KERR: You've convinced me that they don't grow that fast and they must have been there and they missed them.

MR. SHEWMON: Yes. But with regard to the standards people, they are the ones who are responsible for saying, we really have to do more than Section 11, and they have at least one standard out which the industry is now redrafting, which indeed says they have to do some things more.

But stainless steel is generally accepted to be one cut harder, and it's relatively easy if you've got some big hole there where somebody forgot to put weld metal. It's a lot harder if you've got tight cracks which branch repeatedly and one of these paths happens to go all the way through. So you have more noise, a more difficult crack here in this material, and a thinner section.

There are more things against you here than there are in pressure vessels, and what we're going to hear about tomorrow is how reliably can you do it in pressure vessels.

MR. JORDAN: It is fair to say the technology has improved and the inspections over the last five years have become substantially more sensitive than they were earlier.

MR. SHEWMON: But there's still a variant of that question, which is, does the NRC require or even encourage people to use these techniques? Because they can still come back and say, we meet Section 11, and that hasn't been upgraded. So some people may be able to do that. But --

MR. JORDAN: In this particular case we are issuing a bulletin. We go to the CRGR tomorrow and plan to issue the bulletin next Wednesday, which requests, in fact requires, the utility to do this normalization, this set of utilities. Then we'll use that information on the next set of utilities that go into outage after January. So we will be perhaps requiring more stringent testing.

MR. SHEWMON: That has the force of law? That's not optional with them? They have to do that?

MR. JORDAN: Yes.

MR. OKRENT: One of the kind of things I'm trying to ascertain is, let's say that you think you are instituting something that at least represents an improvement for this particular situation, but this having occurred, are you looking at various other kinds of situations in the plant where in-service inspection is also occurring, whether it's a steam generator or whatever, to see if you are not being subject to a similar, not the identical but a similar, problem?

I don't get the sense of that. I always seem to have the feeling that the specific problem is being addressed and a specific improvement is developed, but somehow I fail to get the sense that one looks hard for the possible broader implication.

MR. JORDAN: There are a number of research tasks that speak exactly to this in the UT area as well as tube generator type. That goes along with industry and their research. EPRI has a relatively large facility that is devoted to that kind of research, with the object of feeding back a technique that is more successful in finding flaws.

The staff that I have in Inspection & Enforcement is really looking for generic problems like this one and to apply a specific fix and then make a recommendation for a wider fix where it is appropriate. I appreciate your comment that we should look harder for the wider fix.

MR. EBERSOLE: Do you have any feel for the loss of strength margins these cracks might represent?

MR. JORDAN: General Electric made a presentation to us indicating that -- and I believe I am correct -- that a half-wall thickness circumferential could still withstand the seismic event with normal system loads, so that the fracture toughness of the stainless steel and its ductility would cause it to hold together.

So General Electric pushes very hard the leak before break testing.

MR. JORDAN: Samples are being made all over the country.

MR. EBERSOLE: Pressurization to get an ultimate fix. Cap 'em and pump 'em up.

MR. JORDAN: There are no plans at this point to, but there's a lot of cracked pipe out there that people could experiment on.

I haven't given you my presentation, but I think I've told you everything I know.

MR. SHEWMON: That's been very helpful.

(Laughter.)

MR. SHEWMON: Well, I guess we have run over our time, and I'm sure you will hear from us again.

MR. JORDAN: Yes.

(Whereupon, at 5:05 p.m., the Committee was adjourned.)

* * *

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

____

in the matter of: ACRS/270th General Meeting

Date of Proceeding: October 7, 1982

Docket Number: ______________

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcrip thereof for the file of the Commission.

Jane N. Beach

Official Reporter (Typed)

Jane N. Beach

Official Reporter (Signature)

SUMMARY OF LICENSING STATUS
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

AUGUST 1971	APPLICATION TO CONSTRUCT, NO. 2
SEPTEMBER 1972	CP-SER ISSUED
MARCH 1973	CONSTRUCTION PERMIT ISSUED (CPPR-93)
MARCH 1977	APPLICATION FOR OPERATING LICENSE TENDERED
DECEMBER 1981	OL-FES ISSUED
MARCH 1982	OL-SER ISSUED
AUGUST 1982	OL-SSER NO. 1 ISSUED
SEPTEMBER 1982	ACRS SUBCOMMITTEE MEETING
SEPTEMBER 1983	APPLICANT'S ESTIMATED FUEL LOAD DATE

T1

OUTSTANDING ISSUES

		LICENSEE RESPONSE EXPECTED
(2)	INTERNALLY GENERATED MISSILES	OCTOBER 1982
(3)	TORNADO MISSILE PROTECTION FOR DIESEL GENERATOR (DG) EXHAUST	UNDER STAFF REVIEW
(4)	TURBINE MISSILES	UNDER STAFF REVIEW
(6)	EQUIPMENT QUALIFICATION	OCTOBER 1982
(9)	MODIFICATION OF ADS LOGIC	OCTOBER 1982
(10)	STANDBY SERVICE WATER SYSTEM I&C DESIGN	UNDER STAFF REVIEW
(13)	CONTROL SYSTEM FAILURES	DECEMBER 1982
(21)	CRITERIA FOR TESTING HOT PIPE CONTAINMENT PENETRATIONS	UNDER STAFF REVIEW
(22)	EMERGENCY PLANNING PROGRAM	MAY 1983
(23)	CONTROL ROOM DESIGN REVIEW	MARCH 1983
(24)	ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)	MARCH 1983
(26)	TMI II.E.4.2 (OPERABILITY OF PURGE VALVES ONLY)	OCTOBER 1982
(28)	PIPE BREAK IN THE BWR SCRAM DISCHARGE	OCTOBER 1982
(29)	STEAM BYPASS FROM A STUCK OPEN WETWELL-TO-DRYWELL VACUUM BREAKER	PRIOR TO FUEL LOAD
(30)	HEAVY LOAD HANDLING SYSTEM	OCOTBER 1982
(31)	SPRINKLER AND STANDPIPE SYSTEM	UNDER STAFF REVIEW
(32)	ORGANIZATIONAL CHANGES	AWAITING INFORMATION
(33)	CABLE SEPARATION CRITERIA	UNDER REVIEW

2. INTERNALLY GENERATED MISSILES (SER 3.5.1.1, 3.5.1.2)

THE APPLICANT'S SCHEDULED COMPLETION DATE FOR THE REPORT ON INTERNALLY GENERATED MISSILES (OUTSIDE AND INSIDE CONTAINMENT) IS OCTOBER 1982.

STATUS: AWAITING FURTHER INFORMATION

3. <u>TORNADO MISSILE PROTECTION FOR DIESEL GENERATOR EXHAUST</u> (SER 3.5.2, 9.5.8)

THE APPLICANT BELIEVES THAT THE PROBABILITY OF A TORNADO OF SUFFICIENT VELOCITY TO LIFT LARGE, HEAVY MISSILES ALMOST 1000 FEET AWAY AND PLUG THE DIESEL EXHAUSTS IS EXTREMELY LOW.

THE STAFF IS REVIEWING THE APPLICANT'S POSITION.

<u>STATUS</u>: UNDER REVIEW

4. <u>TURBINE MISSILES</u> (SER 3.5.1.3)

THE WNP-2 HAS A WESTINGHOUSE TURBINE GENERATOR AND ITS PLACEMENT AND ORIENTATION IS UNFAVORABLE WITH RESPECT TO THE REACTOR BUILDING; THAT IS, THERE ARE SAFETY-RELATED TARGETS INSIDE THE LOW TRAJECTORY MISSILE (LTM) STRIKE ZONE.

THE STAFF HAS RECEIVED THE REQUESTED INFORMATION FROM THE APPLICANT.

STATUS: UNDER REVIEW

6. <u>ELECTRICAL EQUIPMENT QUALIFICATION</u> (SER 3.10, 3.11)

ENVIRONMENTAL EQUIPMENT QUALIFICATION AUDIT IS SCHEDULED FOR OCTOBER 1982 AND SEISMIC QUALIFICATION REVIEW TEAM (SQRT) AUDIT IS SCHEDULED FOR NOVEMBER 1982.

<u>STATUS</u>: AWAITING INFORMATION

9. MODIFICATIONS OF AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) LOGIC (II.K.3.18, SER 6.3.6)

THE APPLICANT HAS TAKEN TO POSITION THAT THE CURRENT ADS LOGIC DESIGN, WITH IMPLEMENTATION OF THE SYMPTOM - ORIENTED EMERGENCY PROCEDURES GUIDELINES (EPG'S), IS ADEQUATE. THE STAFF'S POSITION IS THAT THE APPLICANT PROVIDE LOGIC MODIFICATIONS THAT ELIMINATE THE NEED FOR OPERATOR ACTION TO DEPRESSURIZE THE VESSEL FOR THE CASE OF A STUCK OPEN SAFETY RELIEF VALVE OR OUTSIDE STEAMLINE BREAK (WITH FAILURE OF HPCS)

STATUS: AWAITING FURTHER INFORMATION

10. <u>STANDBY SERVICE WATER SYSTEM INSTRUMENTATION AND CONTROL</u> (I&C) DESIGN (SER 7.3.2.4)

THE STANDBY SERVICE WATER SYSTEM IS CONTROLLED USING MULTIPLEXED SIGNALS TO OPERATE ASSOCIATED PUMPS AND VALVES. THE SYSTEM IS REDUNDANT (ONE CHANNEL PER ESF DIVISION), POWERED FROM CLASS IE POWER SOURCES, AND IS SEISMICALLY QUALIFIED.

THE STAFF IS REVIEWING AND DISCUSSING THE UNIQUE FAILURE MODES SUCH AS AN ELECTROMAGNETIC INTERFERENCE, TESTABILITY, AND SURVEILLANCE WITH THE APPLICANT. AT PRESENT, THE STAFF'S POSITION IS THAT THE APPLICANT SHOULD PERFORM EMI TEST EITHER IN THE LABORATORY OR IN THE FIELD AT THE SITE.

<u>STATUS</u>: UNDER REVIEW

13. CONTROL SYSTEM FAILURES (SER 7.7.2.1, 7.7.2.2, 7.5.2.3)

THE MAJOR CONCERN HERE IS THAT IF TWO OR MORE CONTROL SYSTEMS RECEIVE POWER OR SENSOR INFORMATION FROM COMMON POWER SOURCES OR COMMON SENSORS, FAILURES OF THESE POWER SOURCES OR SENSORS OR RUPTURE/PLUGGING OF A COMMON IMPULSE LINE COULD RESULT IN EVENT SEQUENCES MORE SEVERE THAN THOSE CONSIDERED IN THE PLANT SAFETY ANALYSIS.

THE APPLICANT HAS COMMITTED TO PERFORM A STUDY TO DETERMINE CONTROL SYSTEMS FAILURES WHICH COULD RESULT IN PHENOMENA WHICH COULD INTITIATE OR WORSEN A TRANSIENT/ACCIDENT.

THE RESULTS OF THE STUDY WILL BE PROVIDED IN DECEMBER 1982 AND, IF NEEDED, REMEDIAL ACTIONS WILL BE IMPLEMENTED PRIOR TO PLANT OPERATION.

STATUS: AWAITING INFORMATION

21. <u>CRITERIA FOR TESTING HOT PIPE CONTAINMENT PENETRATIONS</u>

THE APPLICANT HAS RECENTLY STATED THAT UNLIKE OTHER MARK II PLANTS IT HAS A FREE STANDING STEEL CONTAINMENT AND THE ABOVE CRITERIA FOR TESTING OF HOT PIPE CONTAINMENT PENETRATIONS IS NOT APPLICABLE TO WNP-2.

THE STAFF IS DISCUSSING THIS WITH THE APPLICANT & WILL REPORT THE RESOLUTION IN LATER SSER.

<u>STATUS</u>: UNDER REVIEW

22. EMERGENCY PLANNING PROGRAM (SER 13.3)

THE APPLICANT HAS FILED EMERGENCY PLANNING PROGRAM FOR WNP-2 ONSITE AND CORPORATE ACTIVITIES ONLY. OFFSITE STATE AND LOCAL ENTITIES WITHIN THE EMERGENCY PLANNING ZONES HAVE NOT SUBMITTED THEIR PLAN.

STATUS: AWAITING FURTHER INFORMATION

23. <u>CONTROL ROOM DESIGN REVIEW</u> (SER 18.0)

THE APPLICANT PROPOSES TO SUBMIT THE CONTROL ROOM DESIGN REVIEW REPORT BY MARCH 1983. THE STAFF WILL REPORT THE RESULTS OF THE EVALUATION IN A FUTURE SUPPLEMENT.

<u>STATUS</u>: AWAITING FURTHER INFORMATION

24. <u>ANTICIPATED TRANSIENTS WITHOUT SCRAM</u> (ATWS) (SER 15.2.1)

THE STAFF PRESENTED ITS RECOMMENDATION ON PLANT MODIFICATIONS TO THE COMMISSION IN SEPTEMBER 1980. THE COMMISSION WILL DETERMINE THE REQUIRED MODIFICATIONS TO RESOLVE ATWS CONCERNS AS WELL AS THE REQUIRED SCHEDULE FOR IMPLEMENTATION OF SUCH MODIFICATIONS.

FOR THE INTERIM PERIOD, STAFF REQUIRES THAT EMERGENCY PROCEDURES BE DEVELOPED FOR AN ATWS EVENT. APPLICANT WILL PROVIDE INFORMATION ON EMERGENCY PROCEDURES IN MARCH 1983.

<u>STATUS</u>: AWAITING INFORMATION

26. <u>TMI ITEM II.F.4.2, CONTAINMENT ISOLATION DEPENDABILITY (OPERABILITY OF PURGE VALVES ONLY - SER 6.2.4.4)</u>

THE STAFF REQUIRES THAT THE PERFORMANCE AND RELIABILITY OF PURGE SYSTEM ISOLATION VALVES SHOULD BE DEMONSTRATED UNDER CONDITIONS SIMILAR TO THOSE EXISTING IN THE CONTAINMENT FOLLOWING ONSET OF A LOCA.

THE APPLICANT HAS NOT SUBMITTED INFORMATION CONCERNING PURGE VALVE OPERABILITY UNDER LOCA LOADS. SUBMITTAL IS EXPECTED IN OCTOBER 1982.

<u>STATUS</u>: AWAITING FURTHER INFORMATION

28. PIPE BREAK IN THE BWR SCRAM SYSTEM (SER 4.6)

NUREG-0803, "GENERIC SAFETY EVALUATION REPORT REGARDING INTEGRITY OF BWR SCRAM SYSTEM PIPING", STATES THAT PIPE BREAKS IN THE CONTROL ROD DRIVE HYRAULIC SYSTEM AND THE RESULTING ENVIRONMENTAL EFFECTS SHOULD BE VERIFIED ON A PLANT SPECIFIC BASIS. THE APPLICANT HAS BEEN ASKED TO RESPOND TO THIS CONCERN. RESPONSE IS EXPECTED BY OCTOBER 1982.

STATUS: AWAITING FURTHER INFORMATION

29. STEAM BYPASS FROM A STUCK OPEN WETWELL-TO-DRYWELL VACUUM BREAKER (SER 6.2.1.8.6)

THIS CONCERN WAS RAISED BY THE ACRS DURING THE APRIL 28-29, 1981, FLUID DYNAMICS SUBCOMMITTEE MEETING. DUE TO THE LARGE ΔP DEVELOPED DURING THE CHUGGING PHENOMENON, THE VACUUM BREAKERS MAY OPEN, AND SINCE THE CHUGGING PHENOMENON IS REPEATED EVERY 2 SECONDS ON THE AVERAGE, THE VACUUM BREAKER MAY BE CALLED UPON TO FUNCTION ON A CYCLIC MANNER. FAILURE OF A VACUUM BREAKER TO CLOSE DURING THIS TIME PERIOD COULD RESULT IN STEAM BYPASS OF THE POOL, THUS JEOPORADIZING THE INTEGRITY OF THE CONTAINMENT. THE APPLICANT HAS INDICATED THAT HE IS PARTICIPATING IN THE VALVE QUALIFICATION PROGRAM AND IS CONSIDERING DESIGN MODIFICATIONS TO RESOLVE THIS CONCERN.

STATUS: AWAITING FURTHER INFORMATION

30. HEAVY LOAD HANDLING SYSTEM (SSER 9.1.5)

NUREG-0612, "CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS," PROVIDES GUIDELINES TO ENSURE SAFE HANDLING OF HEAVY LOADS. THE STAFF HAS ALSO IDENTIFIED NUMBER OF MEASURES DEALING WITH SAFE LOAD PATHS, PROCEDURES, OPERATOR TRAINING AND CRANE INSPECTIONS, TESTING, AND MAINTENANCE.

THE APPLICANT HAS NOT PROVIDED SUFFICIENT INFORMATION TO DETERMINE COMPLIANCE WITH SOME OF THE CRITERIA IN NUREG-0612. ADDITIONAL INFORMATION IS EXPECTED BY OCTOBER 1982.

STATUS: AWAITING FURTHER INFORMATION

31. SPRINKLER AND STANDPIPE SYSTEM (SER 9.5.1.6)

THE LICENSEE HAS INDICATED THAT FIFTEEN FIRE AREAS REQUIRE CABLE PROTECTION TO ENSURE POST-FIRE SHUTDOWN CAPABILITY. TWELVE OF THESE HAVE FIRE LOADINGS OF LESS THAN 1/2 HOUR (LESS THAN 40,000 BTU/FT^2) AND THE LICENSEE PROPOSES TO DEVIATE FROM THE STAFF GUIDELINES TO THE EXTENT THAT THEY REQUIRE AUTOMATIC FIRE SUPPRESSION SYSTEM. SEVEN OF THESE HAVE FIRE LOADINGS OF LESS THAN 1/4 HOUR (LESS THAN 20,000 BTU/FT^2), FOR WHICH THE STAFF AGREES FOR DELETION OF THE AUTOMATIC SUPPRESSION SYSTEM. THE JUSTIFICATION FOR DELETION OF THE REMAINING FIVE IS UNDER REVIEW.

STATUS: UNDER REVIEW

32. <u>ORGANIZATIONAL CHANGES</u>

WPPSS HAS MADE SOME ORGANIZATIONAL CHANGES. ONE OF THE CHANGES IS THAT THE LICENSING AND THE QUALITY ASSURANCE ACTIVITIES HAVE BEEN COMBINED TOGETHER. WE HAVE NOT RECEIVED ALL THE INFORMATION AS YET. AFTER WE RECEIVE THE INFORMATION, WE WILL REVIEW AND WILL REPORT OUR EVALUATION IN A LATER SUPPLEMENT.

<u>STATUS</u>: AWAITING FURTHER INFORMATION

ELECTRICAL SEPARATION

BASED ON NRR AND REGION V CONCERNS ON CABLE SEPARATION CRITERIA AND IMPLEMENTATION OF THESE CRITERIA, THE APPLICANT HAS RECENTLY ESTABLISHED A TASK FORCE TO DEAL WITH THESE CONCERNS. TO DATE THE APPLICANT HAS SUBMITTED CLARIFICATIONS ON ITS DESIGN CRITERIA FOR SEPARATION AND IS PROCEEDING WITH AN AUDIT OF HOW THE EXISTING CRITERIA HAVE BEEN APPLIED BY THE ENGINEERING DESIGN STAFF AND THE INSTALLATION CONTRACTOR. THE APPLICANT IS ALSO PREPARING AN "ELECTRICAL SEPARATION DESIGN GUIDE" DOCUMENT THAT WILL, STEP BY STEP, ARTICULATE HOW THE SEPARATION CRITERIA WERE TRANSLATED INTO INSTALLED CABLES. A KEY INTENDED USE OF THIS DOCUMENT WILL BE TO ALLOW THE NRC INSPECTORS TO PERFORM AN INDEPENDENT ASSESSMENT THAT THE CABLES HAVE IN FACT BEEN INSTALLED IN CONFORMANCE WITH THE DESIGN CRITERIA.

THIS GUIDE IS EXPECTED TO BE AVAILABLE IN NOVEMBER, 1982. THE TASK FORCE CURRENTLY EXPECTS TO BE ABLE TO COMPLETE ITS REVIEW, IN JANUARY 1983, IN TIME TO TAKE ANY CORRECTIVE ACTIONS FOUND NECESSARY. THE APPLICANT IS PREPARED TO EITHER MAKE A BRIEF PRESENTATION OR RESPOND TO COMMITTEE QUESTIONS ON THIS SUBJECT.

WE WILL REQUIRE THIS MATTER TO BE SATISFACTORILY RESOLVED PRIOR TO FUEL LOADING WHICH IS CURRENTLY SCHEDULED FOR SEPTEMBER 1983.

Table 1.2 Comparison of principal design features of WNP-2 and similar facilities

Design Feature	WNP-2	Zimmer	LaSalle	Hatch Unit 2
Rated thermal power, MW	3323	2436	3323	2436
Gross electrical output, MW	1150	883	1122	822
Main steam flow rate, lb/hr	14,296,000	10,477,000	14,166,000	10,470,000
Total reactor core flow rate, lb/hr	108,500,000	78,500,000	106,500,000	77,000,000
System pressure nominal in steam dome, psi	1020	1020	1020	1020
Fuel lattice	8x8	8x8	8x8	8x8
Number of fuel assemblies	764	560	764	560
Number of fuel per fuel assembly	62	63	62	62
Number of control rods	185	137	185	137
Reactor vessel inside diameter, in.	251	218	251	218
Reactor vessel inside height, ft	72.9	69.3	72.9	69.3
Reactor vessel design pressure, psig	1250	1250	1250	1250
Reactor vessel wall thickness, in.	6.75	5.375	6.75	5.531
Number of recirculation loops	2	2	2	2

Table 1.2 (continued)

Design Feature	WNP-2	Zimmer	LaSalle	Hatch Unit 2
Recirculation loop inside diameter, in.	24	20	24	28
Recirculation pump flow rate, gpm	47,250	33,880	47,250	45,200
Number of jet pumps	20	20	20	20
Number of high pressure coolant injection (core spray) loops	1	1	1	1
Number of low pressure coolant injection pumps	3	3	3	4
Number of low pressure core spray loops	1	1	1	2
Maximum heat flux, Btu/ft^2/hr	361,010	354,000	361,000	361,591
Average heat flux Btu/ft^2/hr	145,384	143,900	145,208	145,528
Maximum power per fuel rod length, kw/ft	13.4	13.4	13.4	13.4
Maximum fuel temperature, °F	3435	3325	3325	3435
Minimum critical power ratio	1.24	1.21	1.24	1.30
Total peaking factor	2.49	2.43	2.25	2.49

WNP-2 ACRS FULL COMMITTEE MEETING

OCTOBER 7, 1982

Washington Public Power Supply System
Richland, Washington 99352

821702

-73

OPENING/AGENDA

R. M. NELSON
MANAGER, PROJECT LICENSING,
WNP-2

821760-44A

AGENDA

5 min.	Opening/Agenda	R. M. Nelson
20 min.	Introduction	D. W. Mazur
15 min.	Corporate Organization/Power Generation	W. C. Bibb
20 min.	Construction Management Organization/QA	R. G. Matlock
15 min.	BREAK	
45 min.	Plant Operations • Plant Layout • Training • Personnel • Emergency Procedures • Control Room Habitability/Human Factors • Emergency Planning • Fire Protection • Containment Systems	J. D. Martin
20 min.	Electrical Power Systems/Selected Mech. Systems • Reliability of A/C Power • Decay Heat Removal • Remote Shutdown	C. M. Powers
5 min.	Equipment Qualification	D. L. Renberger
20 min.	Geology/Seismology	D. L. Renberger
15 min.	Security (Closed Session)	J. W. Klingelhoefer

821760-46A

INTRODUCTION

D. W. MAZUR
DIRECTOR OF OPERATIONS

821760-45A

WASHINGTON PUBLIC POWER SUPPLY SYSTEM MISSION

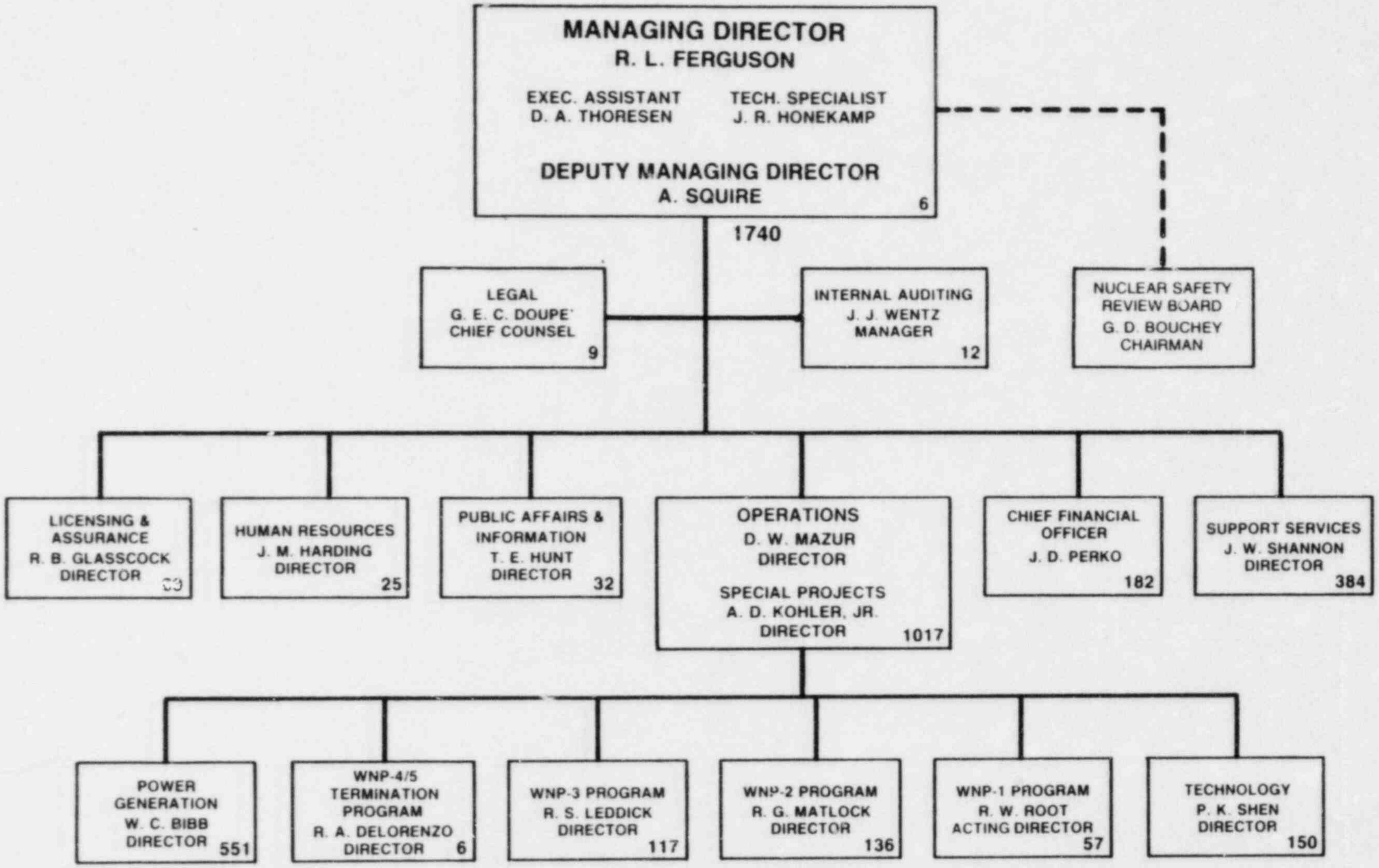
- MUNICIPAL CORPORATION CREATED TO BUILD AND OPERATE ELECTRICAL GENERATING FACILITIES FOR NORTHWEST UTILITIES
- NO MARKETING OR DISTRIBUTION RESPONSIBILITIES
- VIRTUALLY ALL NUCLEAR COMPANY

821650

ORGANIZATIONAL PHILOSOPHY

- CHANGED DIRECTION FROM CONSTRUCTION TO OPERATION AFTER CONSULTING WITH:
  - INSTITUTE OF NUCLEAR POWER OPERATION (INPO)
  - SIMILAR NUCLEAR UTILITIES
  - NUCLEAR UTILITY CEO'S
- CLEAR LINES OF RESPONSIBILITY
- AUTHORITY MUST ACCOMPANY RESPONSIBILITY
- CLOSE COUPLING BETWEEN TECHNICAL, ADMINISTRATIVE AND OPERATIONAL ARMS
- INTERNAL CHECKS & BALANCES (SUCH AS INDEPENDENT AUDIT AND Q.A. FUNCTIONS)
- TECHNICAL OVERVIEW FROM OFFICE OF THE MANAGING DIRECTOR
  - CORPORATE NUCLEAR SAFETY REVIEW BOARD
  - TECHNICAL SPECIALIST

WASHINGTON PUBLIC POWER SUPPLY SYSTEM



NUCLEAR EXPERIENCE OF KEY MANAGEMENT OFFICIALS

Individual	Title	Total Years Nuclear Experience
Mr. R. L. Ferguson	Managing Director	20
Mr. A. Squire	Deputy Managing Director	30
Dr. J. Honekamp	Technical Specialist	22
Mr. D. W. Mazur	Director of Operations	19
Dr. R. G. Matlock	WNP-2 Program Director	21
Mr. C. S. Carlisle	WNP-2 Deputy Program Director	35
Mr. W. C. Bibb	Director, Power Generation	28
Mr. J. D. Martin	WNP-2 Plant Manager	22
Mr. J. R. Holder	Manager, Generation Services	11
Mr. R. R. Stickney	Manager, Generation Training	16
Dr. P. K. Shen	Director, Technology	15
Mr. J. W. Shannon	Director, Support Services	30
Mr. R. B. Glasscock	Director, Licensing and Assurance	24
Dr. G. D. Bouchey	Manager, Nuclear Safety and Licensing	15
Mr. T. E. Hunt	Director, Public Affairs	10

821493

DELEGATION OF AUTHORITY

BOARD OF DIRECTORS

↓

MANAGING DIRECTOR

THE <u>NUCLEAR AND INDUSTRIAL SAFETY</u> POLICY IS TO CONSTRUCT AND OPERATE OUR NUCLEAR FACILITIES WITHOUT RISK TO THE PUBLIC OR EMPLOYEES AND IN COMPLIANCE WITH THE LAW.

MANAGING DIRECTOR'S AUTHORITY:

- CEASE OPERATION OR STOP WORK
- TAKE ANY STEPS NECESSARY TO RECOVER FROM AN ACCIDENT, INCLUDING ANY PROCUREMENT/CONTRACT ACTION
- ANY ACTION NECESSARY TO PROTECT EMPLOYEES OR PUBLIC

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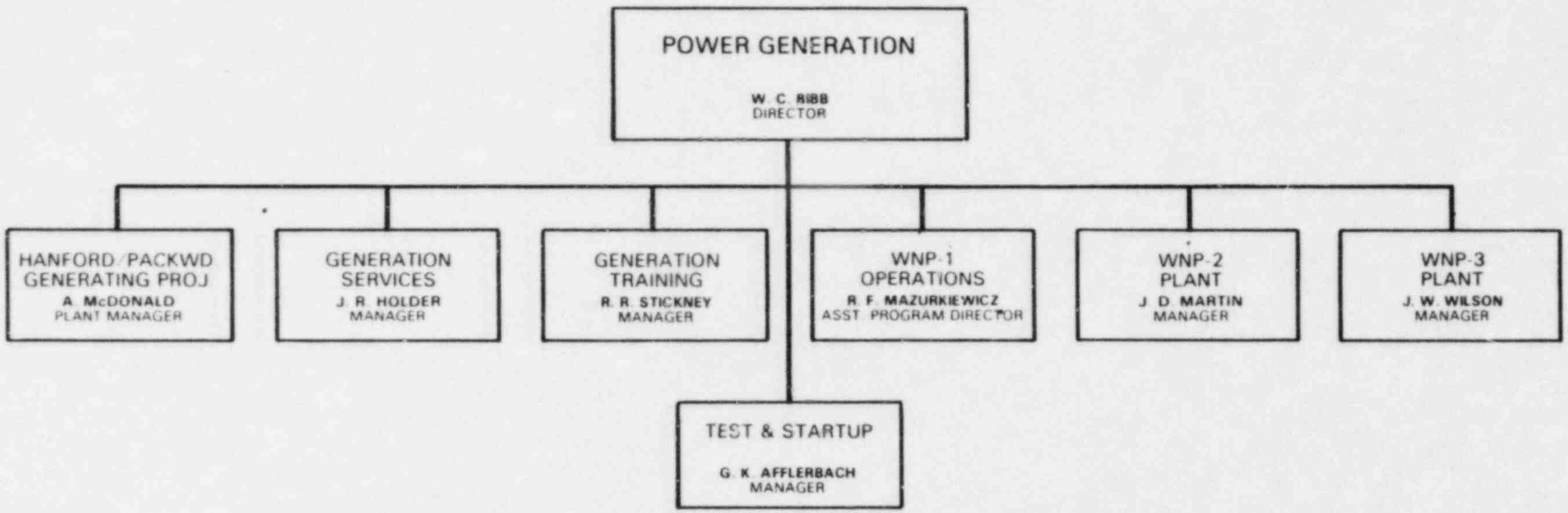
CORPORATE ORGANIZATION/ POWER GENERATION

W. C. BIBB
DIRECTOR, POWER GENERATION

821760-36A

POWER GENERATION ORGANIZATION



PLANT SUPPORT FUNCTIONS

SUPPORT SERVICES DIRECTORATE

- RADIOLOGICAL & CHEMISTRY SUPPORT SERVICES
- SECURITY
- EMERGENCY PREPAREDNESS PLANNING
- INDUSTRIAL SAFETY, INDUSTRIAL HYGIENE, FIRE PROTECTION
- ADMINISTRATIVE SERVICES AND RECORDS MANAGEMENT SUPPORT

LICENSING AND ASSURANCE DIRECTORATE

- INDEPENDENT QA OVERVIEW
- QA POLICY AND GUIDANCE
- LICENSING COORDINATION AND NRC INTERFACE
- OPERATIONAL NUCLEAR SAFETY ASSURANCE

PLANT SUPPORT FUNCTIONS (continued)

CENTRAL SUPPORT FROM POWER GENERATION

- DEVELOPMENT OF OPERATING POLICY
- DEVELOPMENT OF TRAINING POLICIES
- ASSISTS WITH GENERAL TRAINING, SIMULATOR TRAINING, AND COLLEGE TECHNOLOGY/ACADEMIC PROGRAMS
- OPERATING EXPERIENCE PROGRAM/REVIEW (SEE-IN), NOMIS, PPICS, ETC.
- ADMINISTRATIVE SUPPORT
- LABOR SERVICES
- NDE—PROCEDURES, DATA ANALYSIS AND EVALUATION, TECHNIQUE METHODOLOGY, STANDARDS
- STANDARDS LABORATORY

PLANT SUPPORT FUNCTIONS (continued)

TECHNOLOGY DIRECTORATE

- SPECIAL TECHNICAL EXPERTISE
  - WATER CHEMISTRY AND MATERIALS, ETC.
- FUEL MANAGEMENT
  - INCLUDES PLANNING, PROCURING AND LICENSING RELOAD CORES, ENSURING FUEL AVAILABILITY
- ENVIRONMENTAL MONITORING
- REACTOR SAFETY AND CORE ANALYSIS
- ENGINEERED MODIFICATION, INCLUDING CONFIGURATION CONTROL
  - PLANT MANAGER AUTHORIZES WORK
  - ENGINEERING OBTAINS MODIFICATION DESIGN IN ACCORDANCE WITH BASELINE OR APPROVES CHANGE TO BASELINE
  - PLANT AUTHORIZES (THROUGH P.O.C.) WORK AND IMPLEMENTS

821702

PLANT SUPPORT FUNCTIONS
(continued)

CORPORATE NUCLEAR SAFETY REVIEW BOARD

- INDEPENDENT ASSESSMENT OF NUCLEAR SAFETY MATTERS
- MEETS TECHNICAL SPECIFICATION REQUIREMENTS
- UTILIZES OUTSIDE MEMBERS AND CONSULTANTS

821702

SUMMARY

- CORPORATE COMMITMENT TO SAFETY AND OPERATIONAL EXCELLENCE
- TOP LEVEL MANAGERS HAVE SUBSTANTIAL NUCLEAR EXPERIENCE
- ADEQUATE STAFF FOR OPERATION AND TECHNICAL SUPPORT FUNCTIONS
- ADEQUATE TRAINING PROGRAM AND ON SCHEDULE

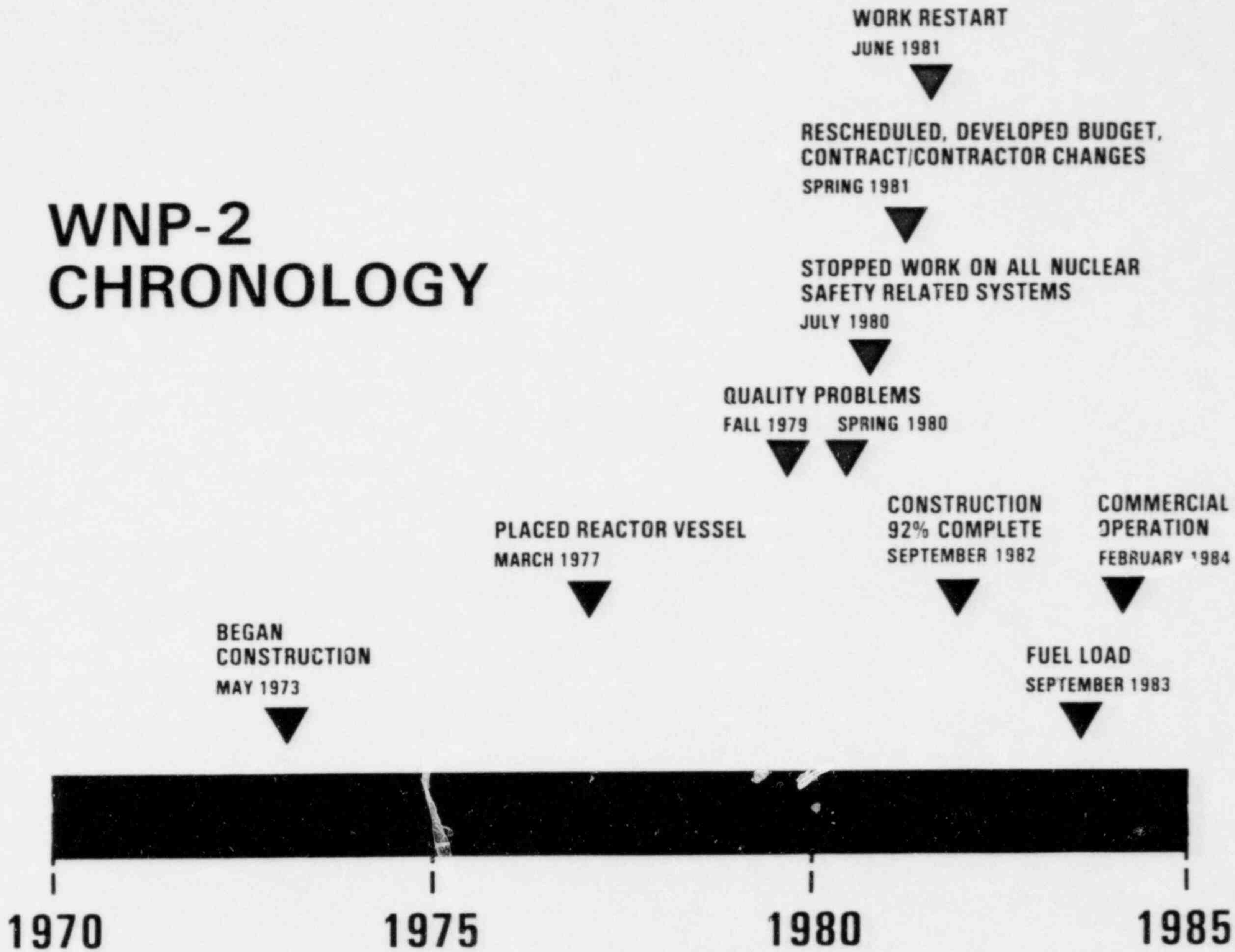
821702

CONSTRUCTION MANAGEMENT ORGANIZATION/QUALITY ASSURANCE

R. G. MATLOCK
PROGRAM DIRECTOR,
WNP-2

821760-37A

WNP-2 CHRONOLOGY



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JULY 1980 STOP WORK

THE PROBLEMS

- CONSTRUCTION QUALITY NOT BEING ACHIEVED
- MANAGEMENT ACTIONS NOT SUCCESSFUL
- BACKLOG OF PROBLEMS INCREASING

THE RECOVERY PROCESS

- RESTART PROGRAM—ASSURE PROPER QUALITY FOR FUTURE CONSTRUCTION
- QUALITY VERIFICATION PROGRAM—VERIFY QUALITY OF PAST CONSTRUCTION

821702

RESTART PROGRAM

- SCOPE INCLUDED QUALITY CLASS I AND/OR SEISMIC CATEGORY I COMPONENTS, STRUCTURES, AND SYSTEMS.
- PROGRAM INCLUDED REVIEW AND EVALUATION OF CONTRACTOR'S QA PROGRAMS, WORK AND INSPECTION PROCEDURES, AND MANAGEMENT CONTROL SYSTEMS.
- CHANGES WERE MADE TO ASSURE COMPLIANCE TO SPECIFICATIONS, CODES AND STANDARDS, AND REGULATORY REQUIREMENTS AND TO IMPLEMENT IMPROVED MANAGEMENT CONTROLS.

OTHER PROGRAM IMPROVEMENTS

- STRENGTHENED PROJECT MANAGEMENT BY CONSOLIDATING TOTAL PROGRAM RESPONSIBILITY UNDER A PROGRAM DIRECTOR REPORTING DIRECTLY TO THE MANAGING DIRECTOR.
- HIRED BECHTEL POWER CORPORATION AS SYSTEMS COMPLETION CONTRACTOR AND CONSTRUCTION MANAGER.
- ASSIGNED THE A/E UNDIVIDED RESPONSIBILITY FOR ENGINEERING IN SUPPORT OF PROJECT COMPLETION.
- REVIEWED AND REDUCED DEFICIENCY BACKLOGS TO WITHIN NEW PERFORMANCE MEASUREMENT LIMITS.
- ADDITIONALLY — REASSIGNMENT OF REMAINING PIPING MECHANICAL WORK TO BECHTEL FORCED A COMPLETE ACCEPTANCE REVIEW OF PAST ASME WORK AND ASSOCIATED DOCUMENTATION DUE TO THE CHANGE IN CODE RESPONSIBILITIES.

821702

QUALITY VERIFICATION PROGRAM

- SCOPE INCLUDED DOCUMENTATION REVIEW AND HARDWARE REINSPECTION OF WORK ACCEPTED/COMPLETED BEFORE JULY 1980
- MAJOR ELEMENTS INCLUDE INACTIVE AND PREPURCHASE CONTRACTS, INCOMPLETE SYSTEMS, AND SPECIAL TASKS
- IMPLEMENTATION BY CONTRACTORS UNDER SUPPLY SYSTEM DIRECTION
- STATUS - PROGRAM IS 85% COMPLETE

821702

QUALITY VERIFICATION PROGRAM (QVP) FINDINGS

- CONSTRUCTION PROBLEMS FOUND BY THE QVP WERE BEING IDENTIFIED BY THE PROJECT IN SPECIAL TASK EFFORTS.
- DEFICIENCY DOCUMENT REVIEWS TO DATE INDICATE THAT PAST TECHNICAL DISPOSITIONS WERE CORRECT.
- EXCEPT AS ALREADY IDENTIFIED NATURE AND NUMBER OF DEFICIENCIES ENCOUNTERED BY QVP PROVIDE CONFIDENCE IN THE WORK COMPLETED BEFORE JULY 1980.
- QVP IS ACCOMPLISHING ITS PRIMARY PURPOSE OF VERIFYING PAST WORK AND CAUSING CORRECTIVE ACTION WHERE NECESSARY.

CONSTRUCTION DOCUMENTATION

- MISSING DOCUMENTATION NOT A PROBLEM
- DOCUMENTATION GENERALLY IN COMPLIANCE WITH CODE AND SPECIFICATION REQUIREMENTS
- DEFICIENCIES RESOLVED USING CODE CASE/OPTIONAL CODE PROVISIONS, PERFORMING ADDITIONAL NDEs, ACQUIRING MISSING DOCUMENTS FROM SUPPLIERS
- CONFIRMED WELD QUALITY BY REVIEW OF ALL ASME RADIOGRAPHS

WNP-2 STATUS/SCHEDULE

- CONSTRUCTION - > 92% COMPLETE
- SYSTEMS TURNOVER - 25% COMPLETE
- SYSTEMS PROVISIONAL ACCEPTANCE - 65% COMPLETE
- SUPPORT SYSTEMS - OPERATIONAL
- ROOM TURNOVER - 40% COMPLETE
- CURRENT ONSITE WORK FORCE - 5400
- HYDRO - COMPLETE
- SPECIAL NUCLEAR MTLS. LICENSE - RECEIVED
- FUEL FABRICATED & STORED
- FUEL LOAD - SEPTEMBER 1983
- COMMERCIAL OPERATION - FEBRUARY 1984

821702

MAJOR ORGANIZATION TRANSITIONS

- OPERATIONAL QUALITY ASSURANCE DEPARTMENT TO CORPORATE Q/A (11/81)
- PLANT OPERATIONS DEPARTMENT TO POWER GENERATION (3/82)
- TEST AND STARTUP DEPARTMENT TO POWER GENERATION (3/82)
- PROJECT QUALITY ASSURANCE DEPARTMENT TO CORPORATE Q/A (4/82)
- ASSUMPTION OF DESIGN RESPONSIBILITY BY THE SUPPLY SYSTEM (ON GOING)
- PHASE OUT OF CONSTRUCTION ACTIVITIES AND TRANSFER OF SITE RESPONSIBILITY TO GENERATION (AT FUEL LOAD)

821702

ACCEPTANCE REVIEW PLANS

. . . DEVELOP DETAILED "ACCEPTANCE REVIEW" PLANS FOR EACH OF OUR PROJECTS WHICH WILL ASSURE A THOROUGH, SYSTEMATIC REVIEW BY SUPPLY SYSTEM PERSONNEL OF OUR NUCLEAR PLANTS PRIOR TO TURNOVER FROM OUR CONTRACTORS FOR COMMERCIAL OPERATION AND WHICH WILL CONSTITUTE A WELL-DOCUMENTED BASIS FOR ACCEPTANCE OF PLANT COMPLETION, SAFETY AND TECHNICAL ADEQUACY.

. . . FOR WNP-2, SPECIAL CONSIDERATION SHOULD BE GIVEN TO ASSURING THAT ANY UNDETECTED QUALITY DEFECTS THAT SIGNIFICANTLY AFFECT PLANT PERFORMANCE OR SAFETY WOULD BE IDENTIFIED AND CORRECTED IN THE COURSE OF OUR FUNCTIONAL TESTING AND ACCEPTANCE REVIEWS.

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PLANT VERIFICATION PROGRAM

821702

PLANT VERIFICATION INCLUDES:

- REQUIREMENTS VERIFICATION
- DESIGN VERIFICATION
- CONSTRUCTION VERIFICATION
- PERFORMANCE VERIFICATION
- OPERATING ENVELOPE VERIFICATION

821702

PLANT VERIFICATION APPROACH

- PLANT VERIFICATION PROGRAM PLAN
  - BASIS FOR CONFIRMATION WNP-2 DESIGNED AND CONSTRUCTED AS COMMITTED
- OVERVIEW OF PROGRAM DEVELOPMENT AND IMPLEMENTATION FROM OFFICE OF THE MANAGING DIRECTOR
- UTILIZE OUTSIDE INDEPENDENT TECHNICAL AUDITOR TO:
  - REVIEW PROGRAM SCOPE
  - AUDIT IMPLEMENTATION
  - ASSURE OBJECTIVITY AND INDEPENDENCE
- TRACK COMPLETION OF PLANT VERIFICATION ACTIVITIES IN PLANT COMPLETION PLAN

821702

ADEQUACY OF DESIGN ESTABLISHED BY:

- EVIDENCE THAT THE BASIC DESIGN PROCESS WAS SOUND
  - QA REVIEWS AND AUDITS OF DESIGN PROCESS
  - EXTERNAL TECHNICAL AUDITS AND DESIGN REVIEWS BY GE, BECHTEL, BRI, AND EDS
  - MANAGEMENT AND TECHNICAL OVERVIEW BY THE SUPPLY SYSTEM
- REQUIREMENTS AND DESIGN REVERIFICATION
  - REVIEW OF THE ENGINEERING RECORD ON A SYSTEM-BY-SYSTEM BASIS FOR ALL SYSTEMS
  - REVIEW OF THE DESIGN REQUIREMENTS FOR ALL SAFETY SYSTEMS
  - DETAILED REVIEW OF THE DESIGN OF THREE SYSTEMS

821702

ASSURANCE OF OBJECTIVITY/INDEPENDENCE IS PROVIDED BY:

- INDEPENDENCE OF REVIEWERS
- FINDINGS REVIEW COMMITTEE
- DIRECT OVERSIGHT FROM THE OFFICE OF THE MANAGING DIRECTOR
- PROGRAM REVIEW AND AUDIT BY OUTSIDE TECHNICAL AUDITOR

821702

CONSTRUCTION PROGRAM SUMMARY

WE:

- HAVE EXPERIENCED DESIGN AND CONSTRUCTION ORGANIZATIONS COMPLETING THE PROJECT.
- HAVE RESOLVED, OR ARE RESOLVING PAST PROJECT CONSTRUCTION QUALITY PROBLEMS AND IMPLEMENTED PROGRAMS TO ASSURE THE ACCEPTABILITY OF HARDWARE PREVIOUSLY INSTALLED.
- HAVE CONTROLS AND VERIFICATION MEANS IN PLACE TO ASSURE THE DESIGN IS CORRECT AND THAT CONSTRUCTION IS IN ACCORDANCE WITH THE DESIGN.
- HAVE PLANNED AND ARE IN THE PROCESS OF IMPLEMENTING AN ORDERLY TRANSITION FROM CONSTRUCTION TO OPERATION (PLANT COMPLETION PLAN).

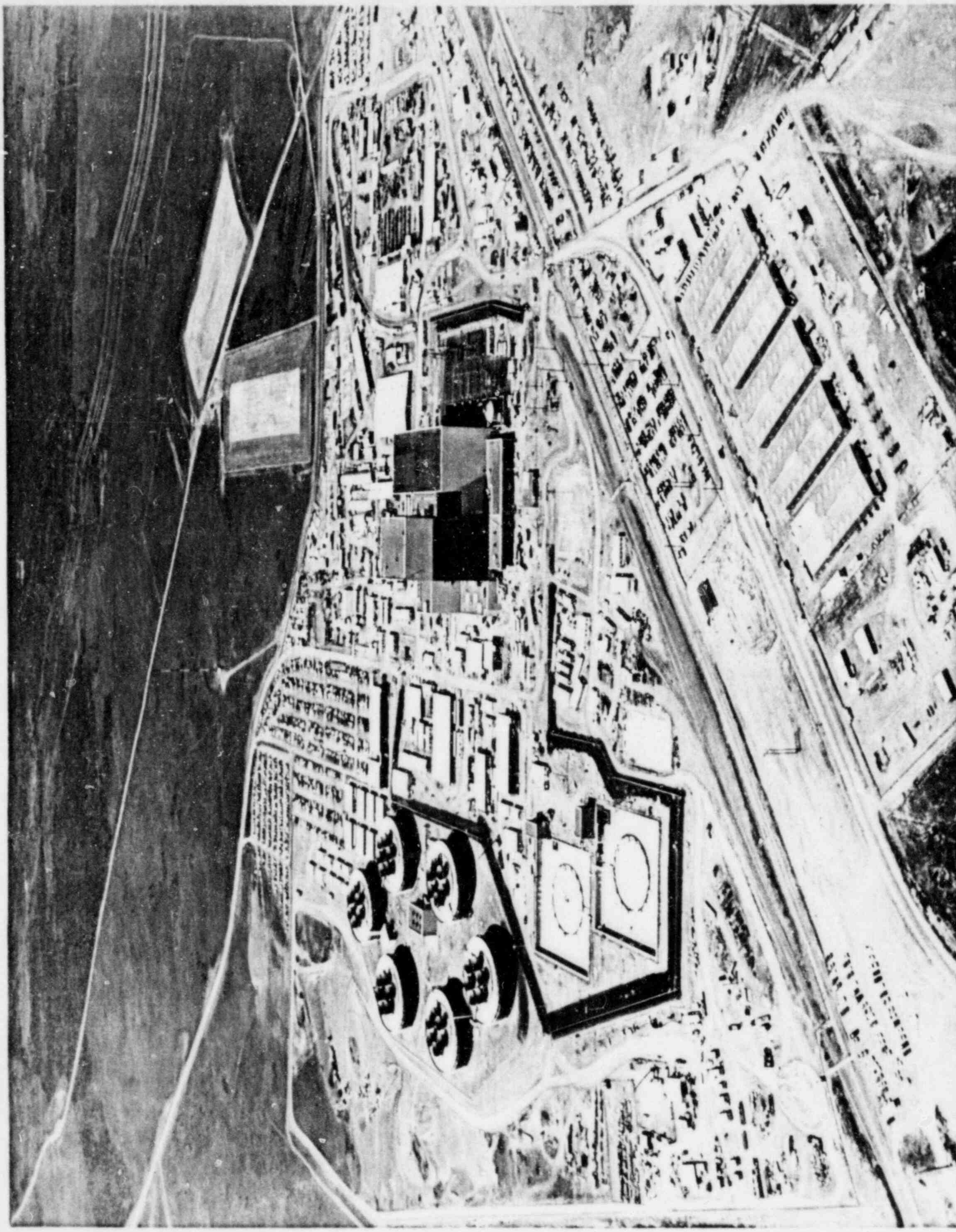
621702

PLANT OPERATIONS

- Plant Layout
- Training
- Personnel
- Emergency Procedures
- Control Room Habitability/Human Factors
- Emergency Planning
- Fire Protection
- Containment Systems

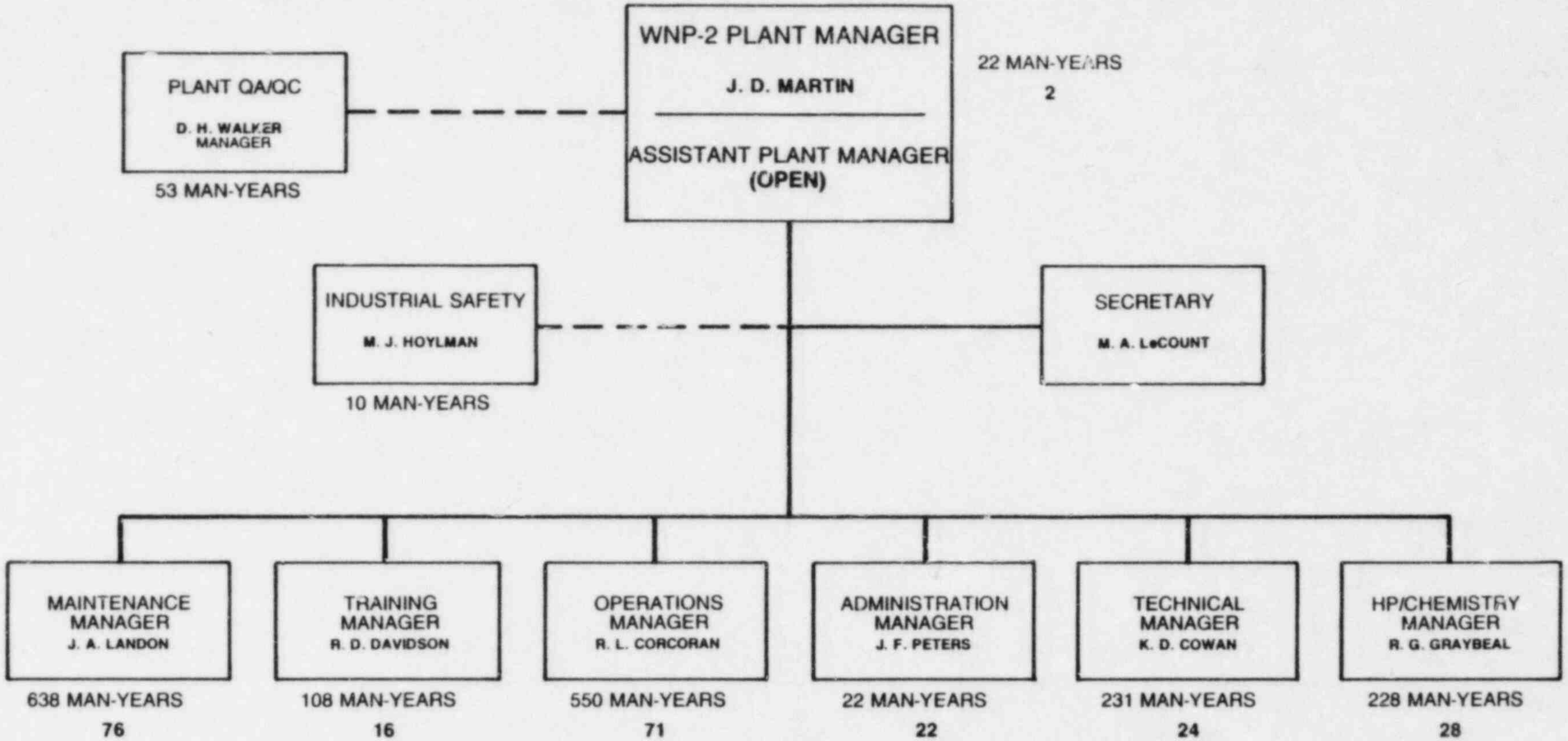
J. D. MARTIN
PLANT MANAGER,
WNP-2

821760-47A



WNP-2 PLANT

NUCLEAR EXPERIENCE



- GREATER THAN 1800 MANYEARS ONSITE NUCLEAR EXPERIENCE.

  OF WHICH

- GREATER THAN 600 MANYEARS ONSITE COMMERCIAL BWR EXPERIENCE.
- **239 OF 240 ON BOARD**

TRAINING PROGRAMS FOR STARTUP AND OPERATION OF WNP-2

- WNP-2 TRAINING DEPARTMENT ORGANIZATION
- PHILOSOPHY FOR PLANT STAFF TRAINING
- GENERAL EMPLOYEE TRAINING
- OPERATOR TRAINING
- TECHNICAL STAFF TRAINING
- MAINTENANCE TRAINING
- HEALTH PHYSICS/CHEMISTRY TRAINING
- TRAINING DEPARTMENT STAFF TRAINING
- STARTUP/TEST STAFF TRAINING

821636

PHILOSOPHY FOR PLANT STAFF TRAINING

DIRECTOR, POWER GENERATION

WNP-2 PLANT MANAGER

PLANT TRAINING DEPARTMENT

- LICENSED OPERATOR SYSTEMS TRAINING
- SIMULATOR OPERATION, MODIFICATION & MAINTENANCE
- STA TRAINING
- STARTUP TRAINING
- MAINTENANCE TRAINING (JOURNEYMAN)
- REQUALIFICATION TRAINING
- NON-LICENSED OPERATOR TRAINING (PLANT SPECIFIC)
- HP/CHEMISTRY TRAINING
- GENERAL EMPLOYEE TRAINING

MANAGER, GENERATION TRAINING

- LICENSED OPERATOR FUNDAMENTALS TRAINING
- COLLEGE TECHNOLOGY PROGRAM
- ACADEMIC UPGRADE
- IN-GRADE MAINTENANCE TRAINING
- NON-LICENSED OPERATOR TRAINING (ENTRY LEVEL)
- POWER PLANT PRINCIPLES
- RESEARCH REACTOR TRAINING

DIRECTOR, SUPPORT SERVICES

MANAGER, HEALTH & SAFETY PROGRAMS

- RADIOLOGICAL PROTECTION TRAINING
- EMERGENCY PLAN TRAINING
- SAFETY & FIRE PROTECTION TRAINING
- FIRST AID & CPR
- ALARA
- RESPIRATORY PROTECTION

821636

SUPPLY SYSTEM TRAINING COMMITMENT

THE SUPPLY SYSTEM IS FIRMLY COMMITTED TO PROVIDING A VIGOROUS AND EFFECTIVE TRAINING PROGRAM. EXAMPLES OF THIS INCLUDE:

- EACH TYPE OF PLANT WILL HAVE A PLANT SPECIFIC SIMULATOR.
- A COLLEGE TECHNOLOGY PROGRAM IS IN PLACE TO IMPROVE THE ANALYTICAL SKILLS OF THE SHIFT MANAGERS & CONTROL ROOM SUPERVISORS.
- SHIFT TECHNICAL ADVISORS ARE ATTENDING THE COLD LICENSE TRAINING PROGRAM AS WELL AS STA TRAINING.
- THE TEST & STARTUP STAFF HAVE PARTICIPATED IN MANY ELEMENTS OF THE COLD LICENSE TRAINING PROGRAM INCLUDING EXTENSIVE SYSTEMS TRAINING & SIMULATOR TRAINING
- SEVERAL SUPPLY SYSTEM COURSES HAVE BEEN EVALUATED BY THE NEW YORK STATE REGENTS AND RECOMMENDED FOR COLLEGE LEVEL CREDIT.
- R. L. FERGUSON LETTER TO E. P. WILKINSON, PRESIDENT OF INPO, DATED AUGUST 6, 1982 TO INITIATE PROCESS THAT WILL RESULT IN ACCREDITATION OF OUR TRAINING PROGRAMS.

821636

SUMMARY

- WELL STAFFED OPERATING ORGANIZATION (STAFFING NEARLY COMPLETE)
- OPERATING STAFF HAS EXTENSIVE NUCLEAR EXPERIENCE (INCLUDING COMMERCIAL BWR EXPERIENCE)
- COMPREHENSIVE TRAINING PROGRAMS PROVIDED FOR PLANT AND PLANT SUPPORT STAFF

821702

EMERGENCY OPERATING PROCEDURES

DEFINITION

PHILOSOPHY

EMERGENCY OPERATING PROCEDURE GUIDELINES

SYMPTOM-BASED

IMPLEMENTATION PLAN FOR EMERGENCY OPERATING PROCEDURES

PREPARATION

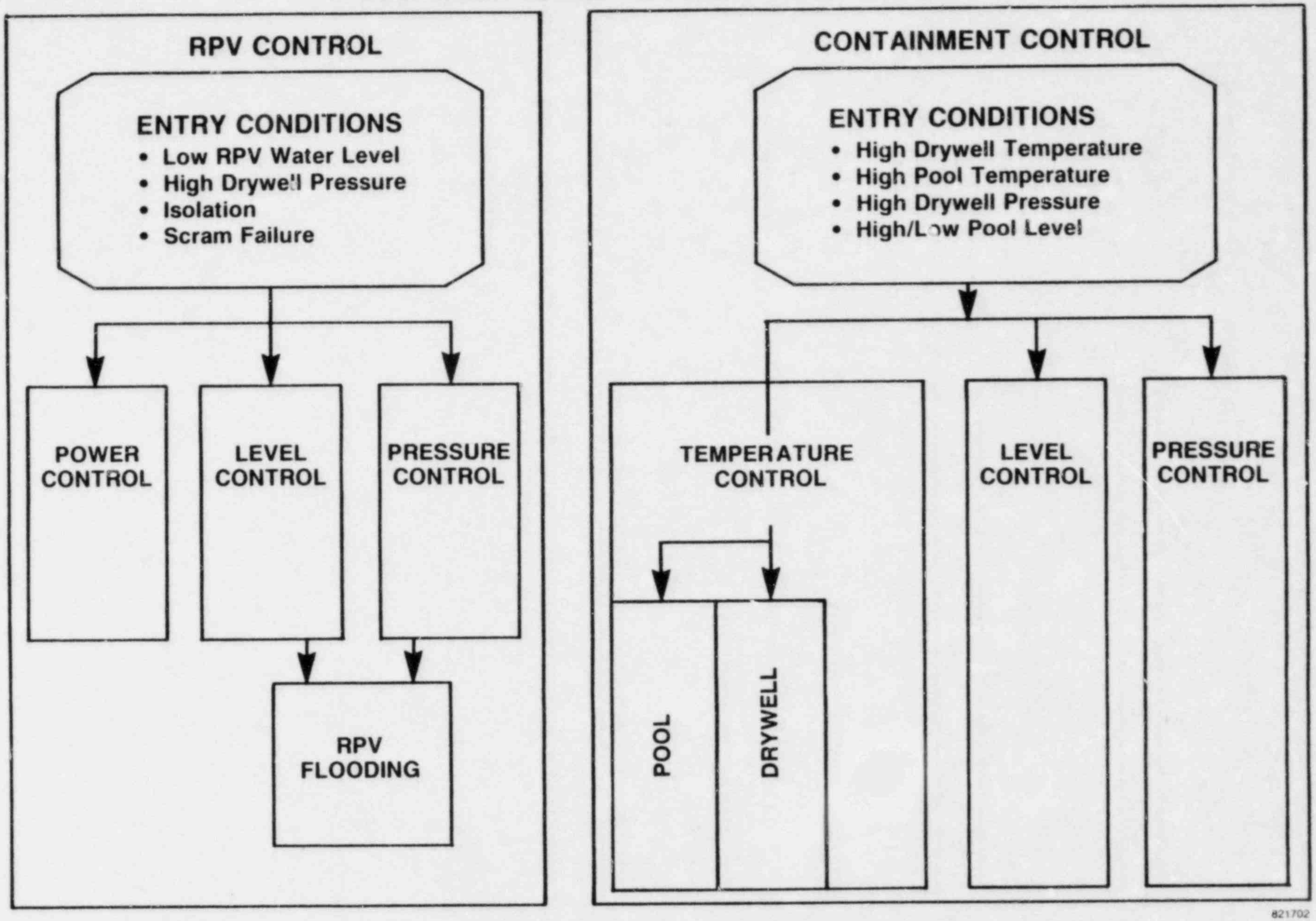
REVIEW

VALIDATION

OPERATOR TRAINING

821702

PROCEDURE ORGANIZATION



CONTROL ROOM HABITABILITY

THE MAIN CONTROL ROOM HABITABILITY SYSTEMS ARE DESIGNED TO ENSURE HABITABILITY DURING ALL NORMAL AND ABNORMAL STATION OPERATING CONDITIONS, INCLUDING 30 DAYS FOLLOWING A LOCA. (PORTABLE BREATHING APPARATUS AND FIVE DAYS WORTH OF FOOD, WATER, MEDICAL SUPPLIES AND SANITARY AND HYGENIC FACILITIES STORED IN CONTROL ROOM)

821050-51A

CONTROL ROOM HUMAN FACTOR IMPROVEMENTS

SUMMARY OF MAJOR AREAS

CONTROL/DISPLAY	• RELOCATION/DELETION OF CONTROLS AND INDICATORS TO IMPROVE OPERATIONAL GROUPING AND ACHIEVE BETTER OPERATOR/PROCEDURE/PANEL INTEGRATION.
ENHANCEMENT	• APPLICATION OF MIMICING AND DEMARCATION, IMPROVED LEGEND PLATE DESIGN, HIERARCHICAL LABELING, AND METER/RECORDER SCALE ADEQUACY TO IMPROVE OPERATOR RECOGNITION AND RESPONSE.
ANNUNCIATOR	• REDESIGN OF ANNUNCIATOR SYSTEM CIRCUITS AND CONTROLS, GROUPING OF RELATED ALARMS, AND UPGRADING OF ALARM WORDING TO IMPROVE OPERATOR ASSESSMENT AND RESPONSE CAPABILITIES

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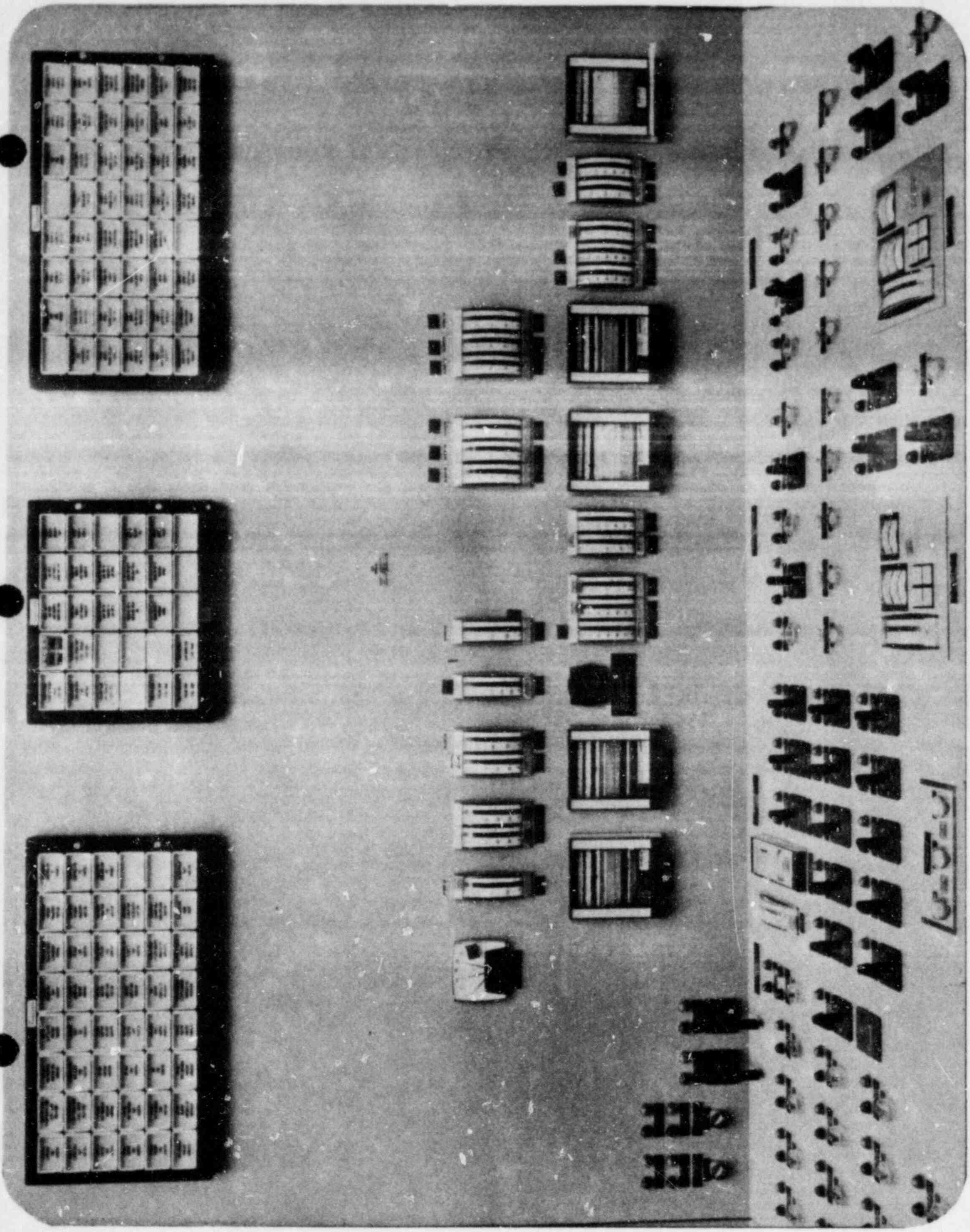

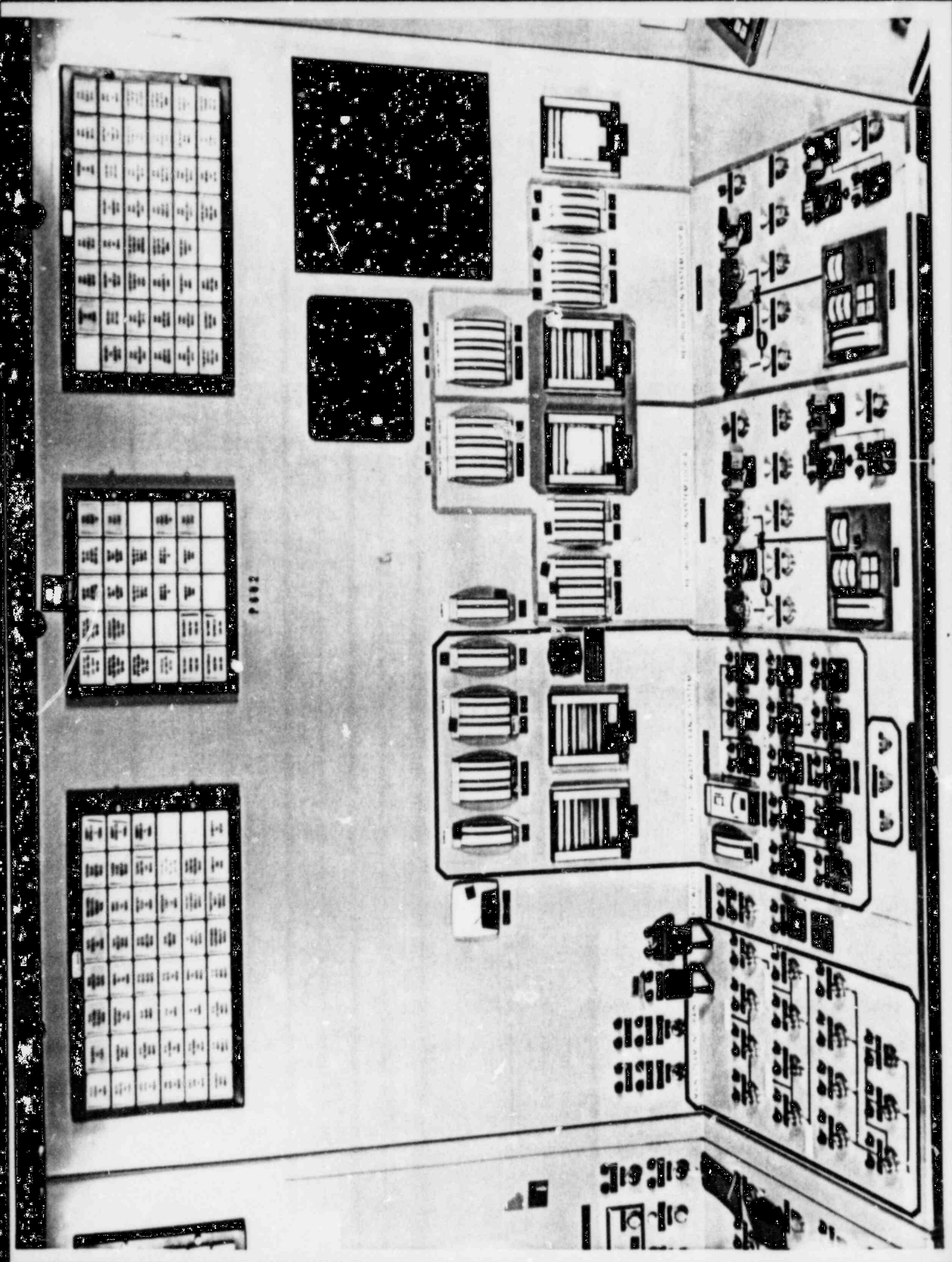
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WNP-2 EMERGENCY PLANNING STATUS

- **WNP-2 EMERGENCY PLAN**
  - ORIGINALLY SUBMITTED TO NRC IN 1976, REVISED MARCH 1981 AND DECEMBER 1981
  - MINOR COMMENTS BY NRC - ALL HAVE BEEN RESOLVED WITH ONLY THREE MILESTONES REMAINING TO BE COMPLETED
- **MILESTONES REMAINING**

  (1) EMERGENCY PLAN IMPLEMENTING PROCEDURES
  - OVER 50% COMPLETE
  - WILL BE SUBMITTED TO NRC MARCH 1983

  (2) STATE/COUNTY EMERGENCY PLANS
  - ORIGINALLY CONCURRED TO BY NRC IN 1976
  - RECENTLY REVISED AND REVIEWED BY FEMA/RAC SEPTEMBER 1982 - NO UNRESOLVED ISSUES OUTSTANDING

  (3) MAJOR EXERCISE
  - SCHEDULED JUNE 1983
  - APPROXIMATELY 20 DRILLS SCHEDULED BETWEEN FEBRUARY AND MAY 1983

ADVANTAGES OF HANFORD SITE EMERGENCY PLANNING

- **HANFORD RESERVATION**
  - LONG HISTORY OF NUCLEAR OPERATIONS
  - LARGE POOL OF TECHNICAL PERSONNEL AND RESOURCES
  - REMOTE SITING
  - ACTIVE DOE EMERGENCY PROGRAM
  - SECURITY CONTROL OVER RESERVATION
- **LOCAL COMMUNITY**
  - LOCAL ACCEPTANCE AND UNDERSTANDING OF NUCLEAR OPERATIONS
  - ACTIVE AND SUPPORTIVE ATTITUDE BY COUNTY OFFICIALS
  - ORIGINAL COUNTY PLAN DEVELOPED IN 1976
  - RECENTLY REVISED TO MEET NEW EMERGENCY PLANNING CRITERIA - FEMA ASSESSMENT FAVORABLE
- **STATE OF WASHINGTON**
  - ACTIVE STATE PROGRAM IN PLACE TO SUPPORT TROJAN NUCLEAR FACILITY
  - FIRST STATE TO RECEIVE NRC CONCURRENCE ON EMERGENCY PLANS (STATE AND COUNTY) IN 1976
  - RECENTLY REVISED TO MEET NEW EMERGENCY PLANNING CRITERIA - FEMA ASSESSMENT FAVORABLE

821760-31A

ADVANTAGES OF HANFORD SITE EMERGENCY PLANNING (continued)

- **SUPPLY SYSTEM**
  - NEW EMERGENCY OPERATIONS FACILITY NEARING COMPLETION
  - NEW ONSITE TECHNICAL SUPPORT CENTER NEARING COMPLETION
  - ACTIVE PROGRAMS UNDERWAY TO DEVELOP COMPUTERIZED EMERGENCY DOSE PROJECTION SYSTEM AND SAFETY PARAMETER DISPLAY SYSTEM
  - SUFFICIENT DEDICATED EQUIPMENT TO PLACE 10 ENVIRONMENTAL PERSONNEL IN THE FIELD QUICKLY PLUS ADDITIONAL EQUIPMENT AVAILABLE
  - HEADQUARTERS LOCATED APPROXIMATELY 10 MILES FROM WNP-2
  - JOINT EMERGENCY INFORMATION CENTER LOCATED AT HEADQUARTERS
  - PRIMARY AND BACKUP COMMUNICATIONS CENTERS (ONE AT EOF AND ONE AT HEADQUARTERS)

FIRE PROTECTION

- WNP-2 COMPLIES WITH NRC REQUIREMENTS UNDER BTP APCSB 9.5-1 (APP. A) AND 10CFR50, APP. R

- FSAR FIRE PROTECTION EVALUATION REPORT DOCUMENTS COMPLIANCE BY A FIRE HAZARDS ANALYSIS FOR EACH FIRE AREA IN THE PLANT

- DEFENSE IN DEPTH HAS BEEN ENSURED BY:
  - FIRE BARRIERS
  - WATER AND GASEOUS FIRE SUPPRESSION SYSTEMS
  - DETECTION SYSTEMS
  - CABLE RACEWAY SYSTEMS PROTECTION
  - REMOTE SHUTDOWN CAPABILITY
  - FIRE PROTECTION/PREVENTION PROGRAM

821760-40A

MARK II CONTAINMENT

I. HYDRODYNAMIC LOADS

- EXTENSIVE MODIFICATIONS WERE MADE TO THE WNP-2 CONTAINMENT AND COMPONENTS IN THE SUPPRESSION POOL
- COMPREHENSIVE TEST PROGRAMS WERE CONDUCTED IN THE MARK II OWNERS GROUP, AND IN FOREIGN TESTS TO UNDERSTAND AND QUANTIFY LOADS
- WNP-2 UTILIZES MANY OF THE LOAD DEFINITIONS AND LOADING CRITERIA DEVELOPED IN THE MARK II PROGRAM
- DUE TO DIFFERENCES BETWEEN WNP-2 AND OTHER DOMESTIC MARK II CONTAINMENTS, PLANT-UNIQUE LOAD DEFINITIONS FOR SRV DISCHARGE AND CHUGGING WERE DEVELOPED FOR SPECIFIC APPLICATION TO WNP-2
- ALL PLANT-UNIQUE LOAD DEFINITIONS FOR WNP-2 HAVE BEEN APPROVED BY THE NRC

II. VACUUM BREAKER ACTUATION

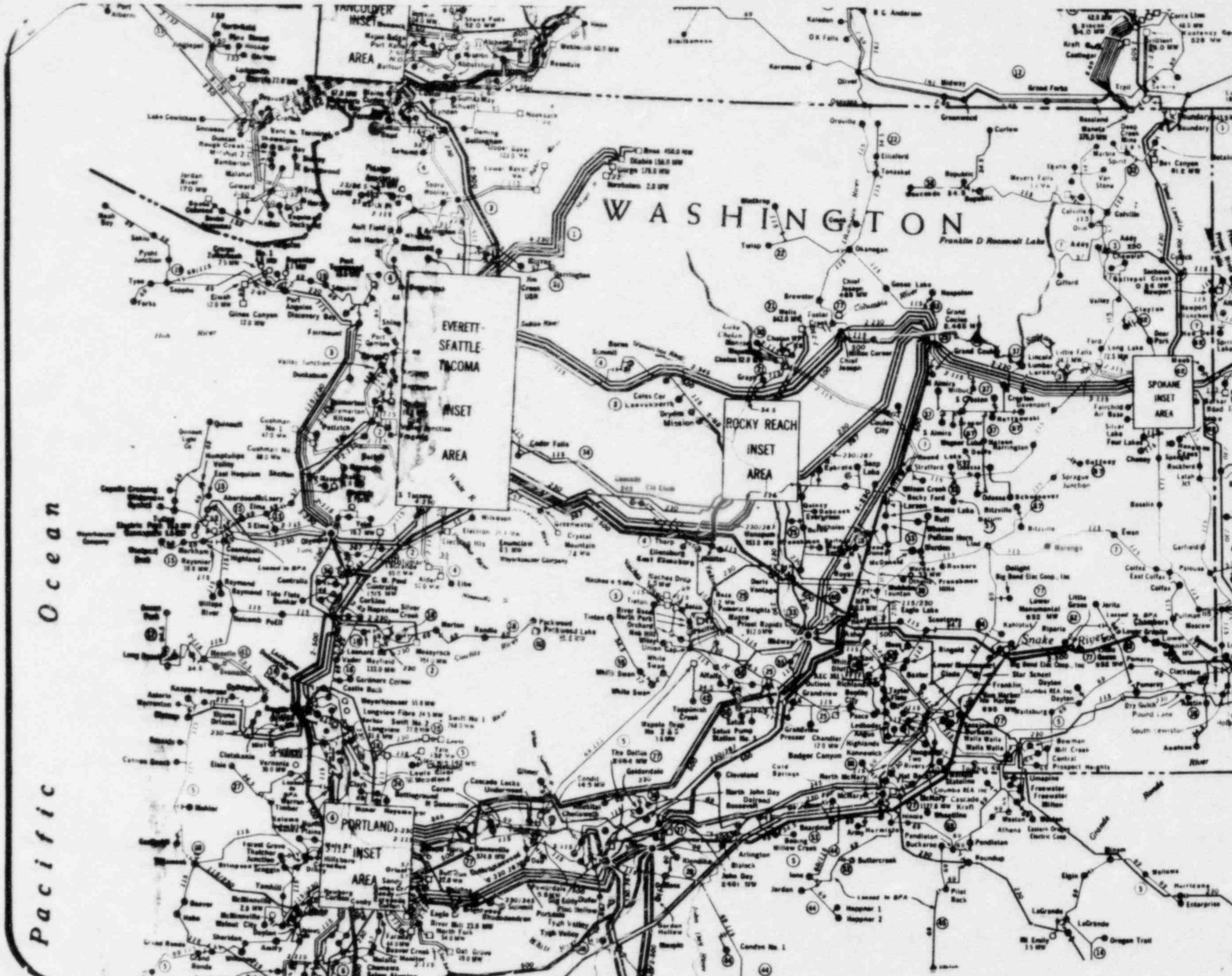
- VACUUM BREAKERS WERE NOT DESIGNED FOR IMPACT LOADS DURING POOL SWELL AND CHUGGING
- WNP-2 WILL INSTALL DAMPING DEVICES TO REDUCE DISC IMPACT VELOCITIES

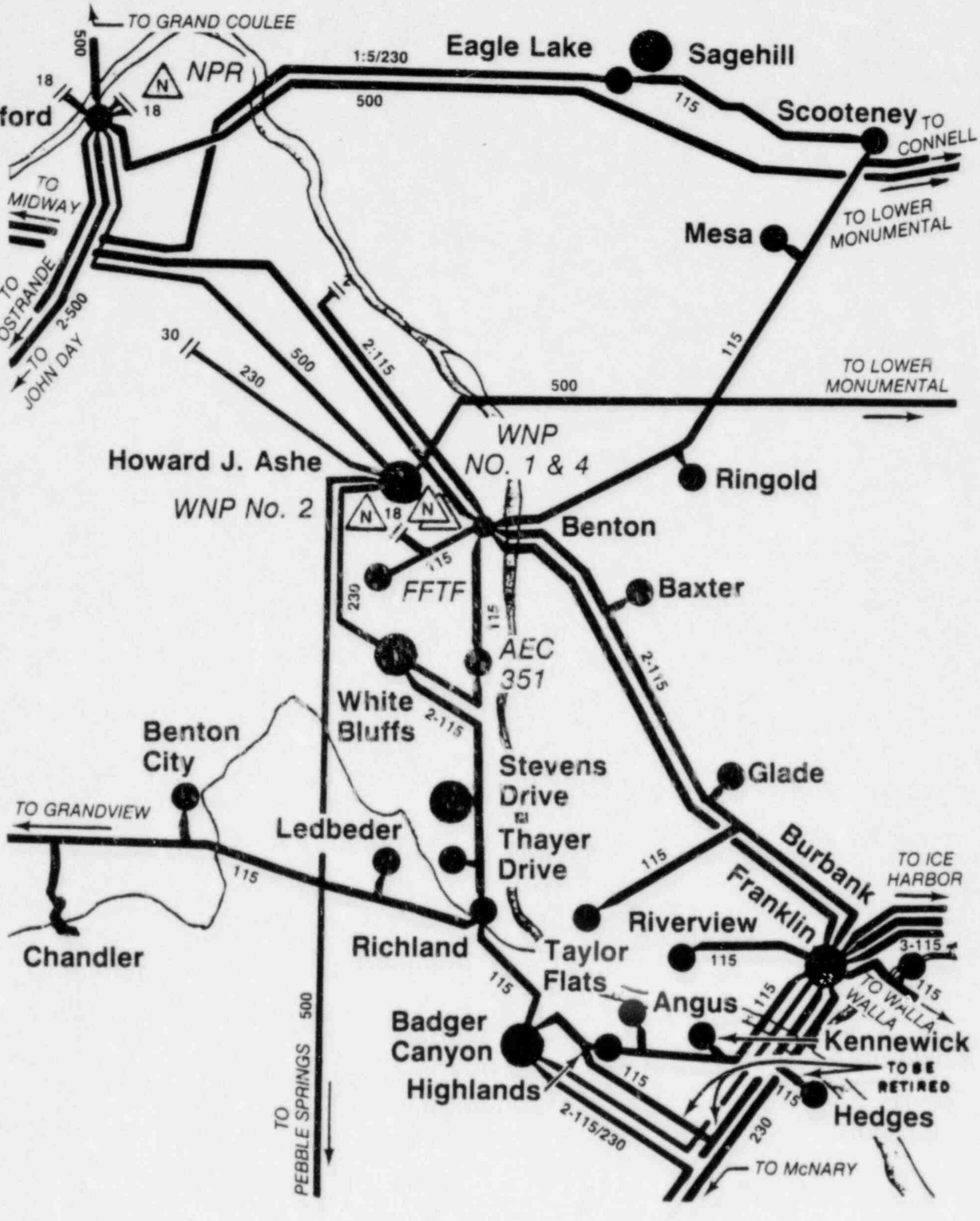
ELECTRICAL POWER SYSTEMS/ SELECTED MECHANICAL SYSTEMS

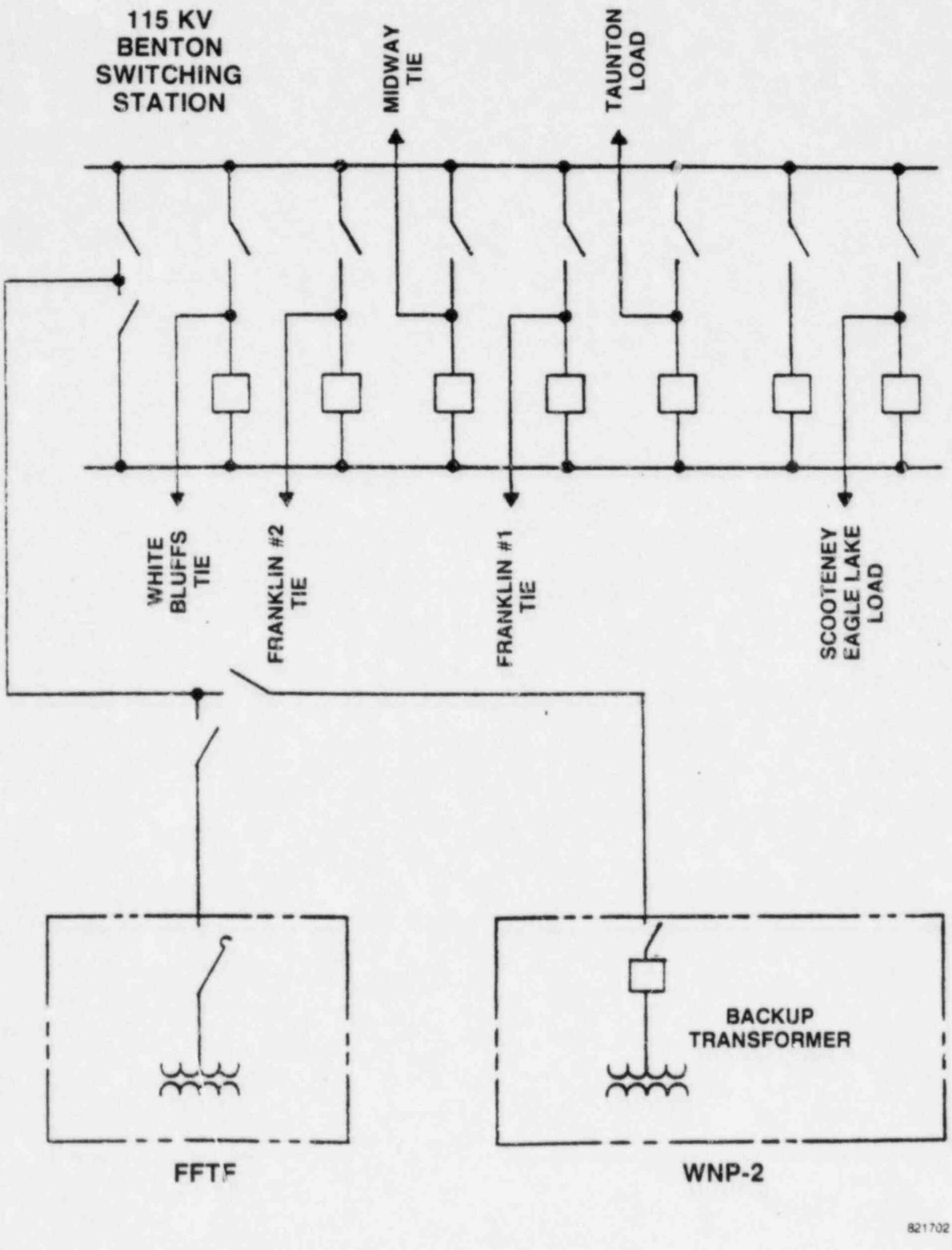
- RELIABILITY OF A/C POWER
- DECAY HEAT REMOVAL
- REMOTE SHUTDOWN

C. M. POWERS
SUPERVISOR,
REACTOR ENGINEERING, WNP-2

821760-38A


WASHINGTON
Pacific Ocean
VANCOUVER INSET AREA
EVERETT-SEATTLE-TACOMA INSET AREA
ROCKY REACH INSET AREA
SPOKANE INSET AREA
PORTLAND INSET AREA
Franklin D Roosevelt Lake
Snake River


TO GRAND COULEE
500
18
18
NPR
Hanford
1:5/230
500
Eagle Lake
Sagehill
115
Scooteney
TO CONNELL
TO MIDWAY
TO LOWER MONUMENTAL
Mesa
TO OSTRANDE
2-500
TO JOHN DAY
30
230
500
2:115
115
500
TO LOWER MONUMENTAL
WNP NO. 1 & 4
Howard J. Ashe
WNP No. 2
Ringold
18
Benton
115
230
FFTF
Baxter
115
AEC 351
2-115
White Bluffs
2-115
Benton City
Stevens Drive
Glade
TO GRANDVIEW
Ledbeder
Thayer Drive
115
Burbank
TO ICE HARBOR
115
Franklin
Riverview
Chandler
Richland
Taylor Flats
115
3-115
115
TO WALLA WALLA
115
Angus
115
500
Badger Canyon
Kennewick
TO BE RETIRED
Highlands
115
115
Hedges
2-115/230
230
TO PEBBLE SPRINGS
TO McNARY

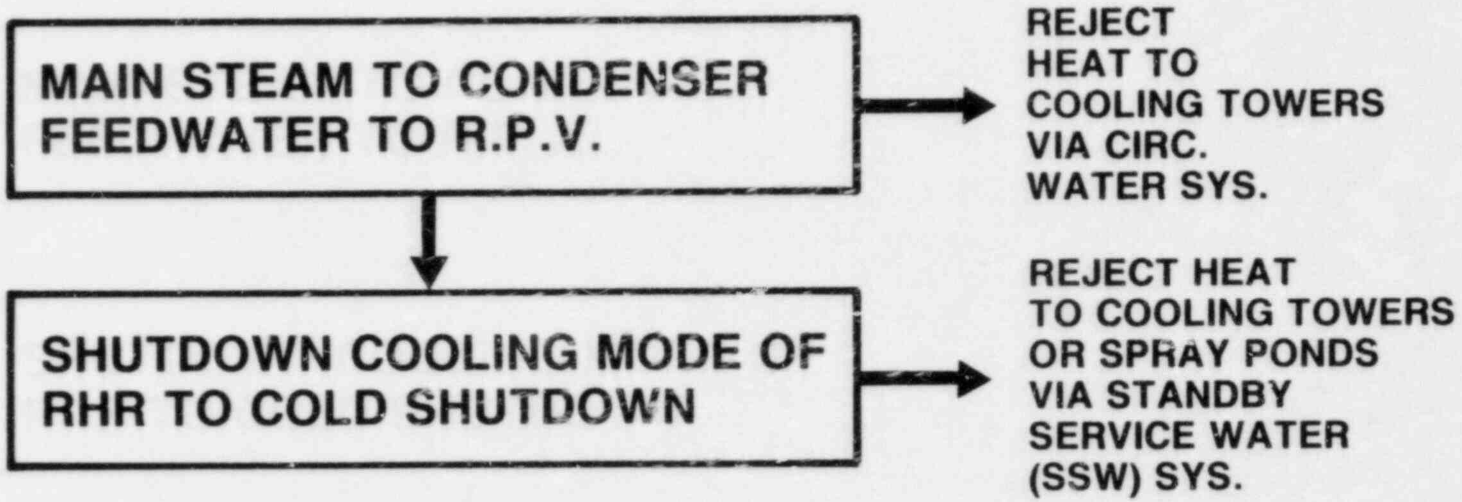

115 KV
BENTON
SWITCHING
STATION
MIDWAY
TIE
TAUNTON
LOAD
WHITE
BLUFFS
TIE
FRANKLIN #2
TIE
FRANKLIN #1
TIE
SCOOTENEY
EAGLE LAKE
LOAD
BACKUP
TRANSFORMER
FFTF
WNP-2
821702

ELECTRICAL POWER SYSTEMS SUMMARY

- WNP-2 SUPPORTED BY DIVERSE, ISOLABLE HYDRO-BASED GRID
- CRITICAL, REDUNDANT SYSTEMS POWERED FROM 4 SEPARATE SUPPLIES
- LOSS OF OFFSITE A/C POWER ACCOMMODATED FOR IN WNP-2 DESIGN
  - — POWER RESTORATION CONTINGENCIES
  - — ONSITE A/C GENERATION CAPABILITIES
  - — EMERGENCY RESPONSE PROCEDURES
- WNP-2 A/C POWER SYSTEMS ARE HIGHLY RELIABLE

DECAY HEAT REMOVAL

(NORMAL)



821702

DECAY HEAT REMOVAL
(RPV ISOLATED FROM NORMAL HEAT SINK)

STEAM CONDENSING MODE

RPV STEAM TO RHR HEAT EXCHANGERS
MAKE-UP RETURN TO RPV VIA RCIC
→ REJECT HEAT TO COOLING TOWERS OR SPRAY PONDS VIA SSW

↓

SHUTDOWN COOLING MODE OF RHR
TO COLD SHUTDOWN
→ REJECT HEAT TO SSW

OR

ALTERNATE SHUTDOWN MODE

RPV STEAM TO SUPPRESSION POOL VIA SRVs
MAKE-UP FROM CST/SP VIS HPCS/RCIC/LPCI
→ REJECT HEAT TO SUPPRESSION POOL AND TO ULTIMATE HEAT SINK VIA RHR/SSW

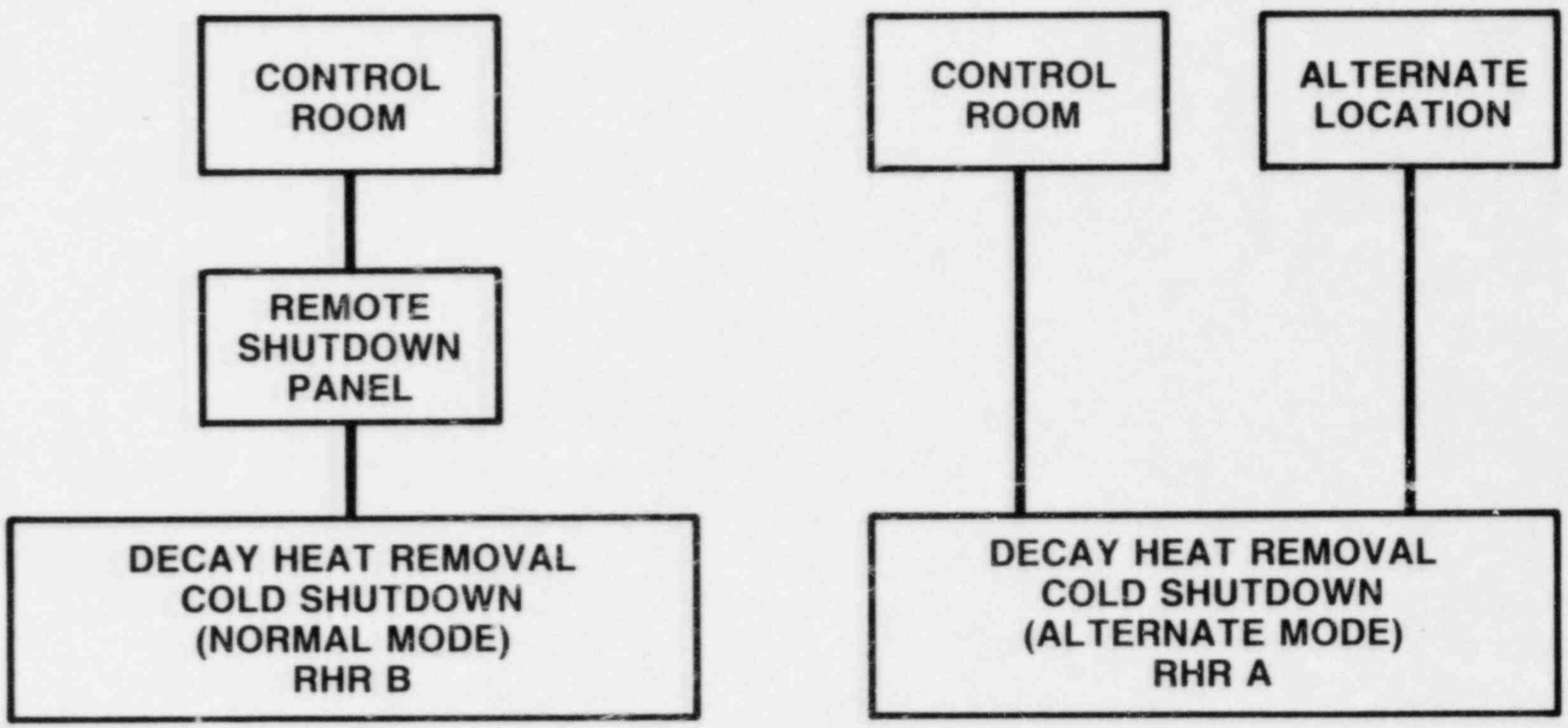
821760-27A

DECAY HEAT REMOVAL SYSTEMS SUMMARY

- SEVERAL DIVERSE MEANS TO REMOVE DECAY HEAT AVAILABLE
- WNP-2 CAN MAINTAIN THE REACTOR IN COLD SHUTDOWN

821760-26A

REMOTE SHUTDOWN SYSTEM



821702

REMOTE SHUTDOWN SYSTEM POSITION SUMMARY

- ALTERNATIVE SHUTDOWN MODE OF OPERATION APPROVED IN LICENSING BASIS
- PROPOSED MODIFICATIONS PROVIDE REDUNDANT REMOTE SHUTDOWN CAPABILITY

821702

EQUIPMENT QUALIFICATION
GEOLOGY/SEISMOLOGY

D. L. RENBERGER
DEPUTY DIRECTOR,
TECHNOLOGY

821760-39A

EQUIPMENT QUALIFICATION OBJECTIVES

1. Confirm that WNP-2's safety related equipment can perform its safety function under all postulated accident and seizure conditions. Where documentation is deficient to establish the confirmation, take the necessary corrective action.

2. Establish the resource & expertise within the Supply System to carry on the work throughout plant life.

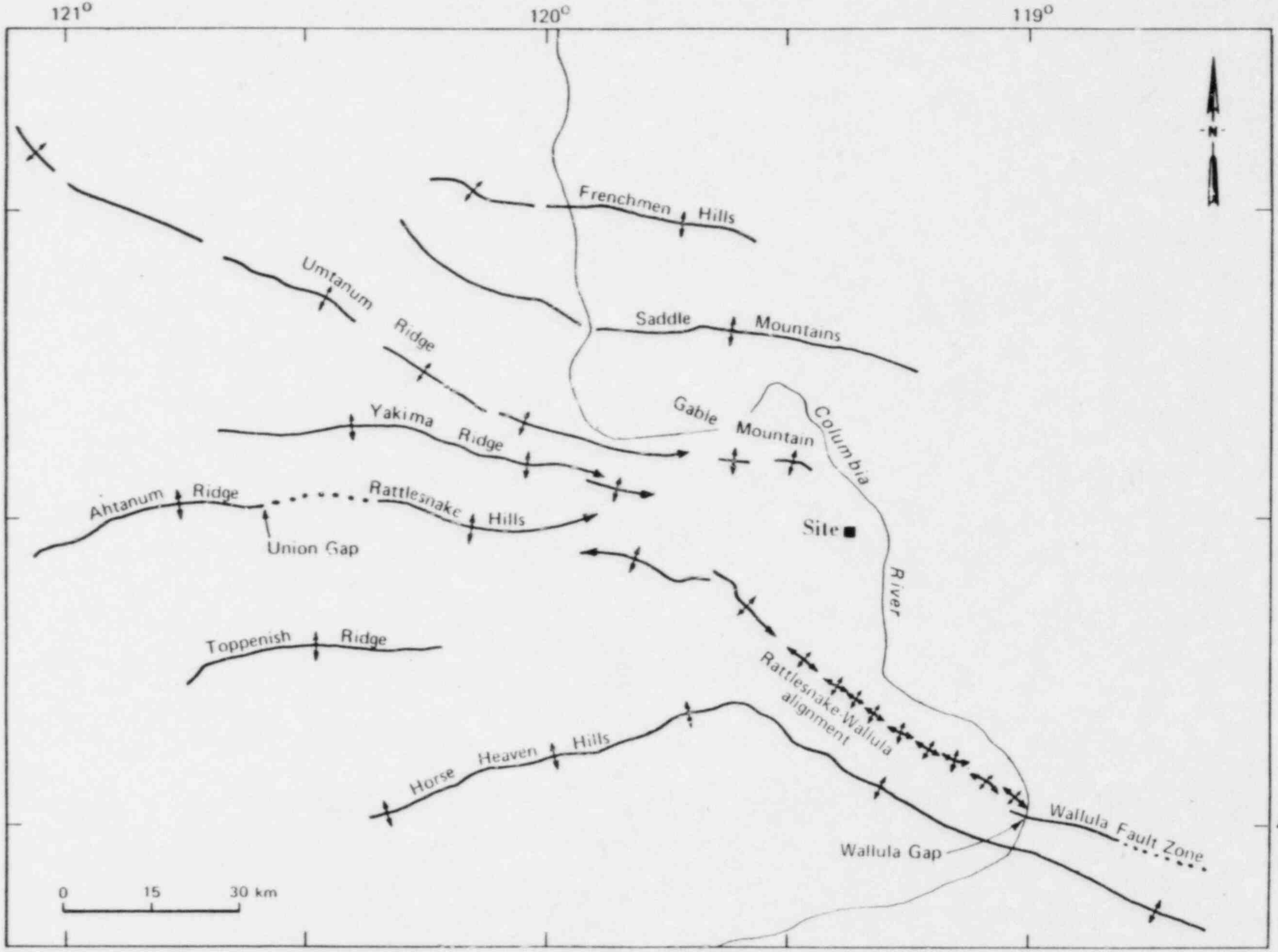
82777

EQUIPMENT QUALIFICATION STATUS

- 85% of items seismically qualified (October 1982 submittal to NRC).
- All equipment seismically qualified by fuel load (9/83).
- 80% of 1E items in a harsh environment are qualified (September 1982 submittal to NRC).
- Remaining 20% of 1E items in a harsh environment are scheduled for qualification (e.g., test, analysis, modification, relocation or replacement).
- Justification for interim operation approved prior to fuel load.
- All 1E items in a harsh environment qualified by November 30, 1985.

821141

CENTRAL COLUMBIA PLATEAU STRUCTURES



LICENSING ACTIONS

1973 -	CP ISSUED
1973 - 1975	INVESTIGATIONS FOR WNP-1 CP (ISSUED 1975)
1975 - 1977	1872 EARTHQUAKE STUDIES AND WNP-4 CP (ISSUED 1978)
1982	WNP-2 OL-SSER AUGUST 1982

821541-12A

CP LICENSING BASIS

- LARGEST HISTORICAL EARTHQUAKE
  INTENSITY (MM) VII
- ASSUME RATTLESNAKE CAPABLE
- FOR CONSERVATISM INCREASE TO INTENSITY (MM) VIII
- DESIGN BASIS 0.25g ZPA WITH APPROPRIATE RESPONSE SPECTRUM

CONCLUSIONS

- Original SSE of .25g confirmed adequate and conservative by:
  - Estimation of maximum magnitude on nearby potential source structures
  - Site specific response spectra based on a conservative estimate of the largest historic earthquake
  - Evaluation of small magnitude earthquakes in close proximity to site
  - Probabilistic evaluation of exceeding SSE considering potential sources within 50 km
- There are no open items

821571

SECURITY

J. W. KLINGELHOEFER

MANAGER,
SAFEGUARDS & INVESTIGATIONS

SECURITY PROGRAM

- DETAILS ARE "SAFEGUARDS INFORMATION" PER 10CFR73.21
- PROVIDES LEVEL OF PROTECTION REQUIRED BY 10CFR73.55 TO RESPOND TO:
  - VIOLENT EXTERNAL ASSAULT
  - INTERNAL THREAT BY INSIDER
- THREE PRIMARY FEATURES
  - SECURITY FORCE
  - PHYSICAL SECURITY EQUIPMENT
  - PROCEDURAL CONTROLS

821050-1A

SECURITY FORCE

- RIGOROUS SELECTION PROCESS FOR SECURITY OFFICERS
- 300 HOURS OF IN-HOUSE TRAINING
- TRAINING PROGRAM APPROVED BY NRC & WASHINGTON STATE
- POSSESSES ALL NECESSARY WEAPONS & EQUIPMENT
- ASSISTANCE AVAILABLE
  - DEPARTMENT OF ENERGY
  - STATE PATROL
  - LOCAL LAW ENFORCEMENT AGENCIES

821050-3A

PHYSICAL SECURITY EQUIPMENT

- CLEARED ISOLATION ZONE AROUND PLANT
- PROTECTED AREA WITHIN PHYSICAL BARRIER
- VITAL AREA BARRIERS & ALARMS
- ILLUMINATION OF PLANT & PROTECTED AREA
- PERIMETER INTRUSION DETECTION
- REMOTE SURVEILLANCE CAPABILITY
- SEARCH FACILITIES AT PROTECTED AREA BARRIER
- IDENTIFICATION BADGING SYSTEM
- CENTRAL & SECONDARY ALARM STATIONS
- REDUNDANT COMMUNICATIONS
- EMERGENCY POWER

821050-4A

PROCEDURAL CONTROLS

NRC APPROVED LICENSING DOCUMENTS

- PHYSICAL SECURITY PLAN
- SAFEGUARDS CONTINGENCY PLAN
- GUARD TRAINING & QUALIFICATION PLAN

IMPLEMENTING INSTRUCTIONS

- PERSONNEL SCREENING
- SEARCH/BADGING/ACCESS CONTROL
- COMMUNICATIONS/RECORDS/REPORTS
- PATROL DUTIES/USE OF FORCE/EMERGENCY ACTIONS
- EQUIPMENT OUTAGES/MAINTENANCE
- WEAPONS HANDLING

821050-2A

1. STATUS OF ISSUES RELATED TO B&W INTEGRAL SYSTEMS TEST FACILITY

2. STATUS OF ANS 5.1 DECAY HEAT (1979) USE IN REGULATORY PROCESS

PRESENTED TO ACRS

OCTOBER 7, 1982

Sheron

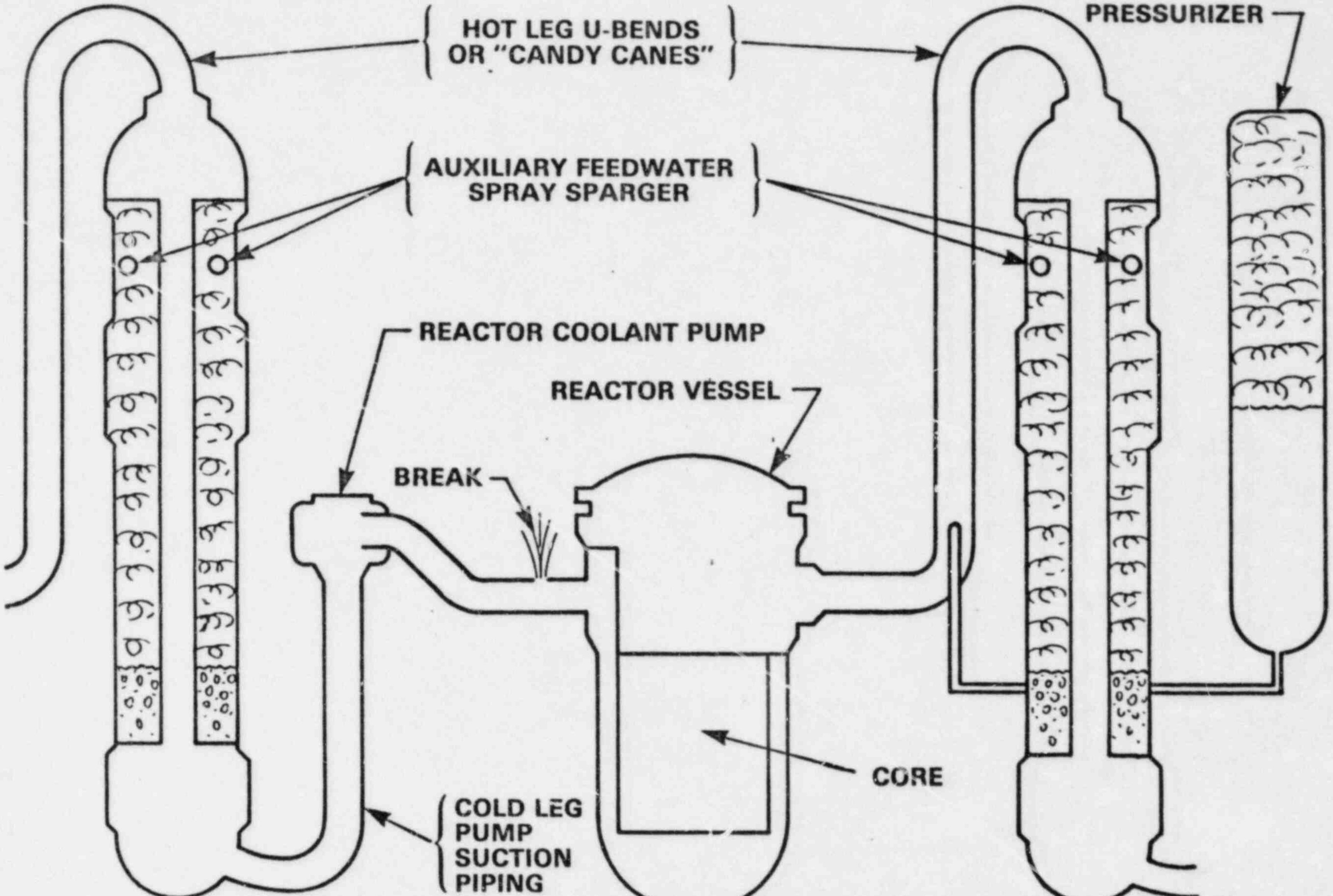
T11

STAFF CONCERN

- o THE DYNAMIC TWO-PHASE THERMAL-HYDRAULIC BEHAVIOR OF NSSSs DESIGNED BY B&W EXHIBITS UNIQUE CHARACTERISTICS.
- o THESE CHARACTERISTICS ARE NOT YET WELL UNDERSTOOD AND COMPUTER MODELS USED TO PREDICT THESE CHARACTERISTICS HAVE NOT BEEN VERIFIED AGAINST APPLICABLE INTEGRAL SYSTEMS DATA.
- o POORLY UNDERSTOOD PLANT PERFORMANCE CHARACTERISTICS DURING TRANSIENTS AND ACCIDENTS COULD RESULT IN INCORRECT OPERATOR DIAGNOSES AND CONSEQUENT ACTIONS WHICH AGGREVATE THE ACCIDENT.
- o CONFIRMATORY INTEGRAL SYSTEM TEST DATA WOULD INCREASE THE LEVEL OF CONFIDENCE IN THE ANALYTICAL MODELS AND THUS THE OPERATOR EMERGENCY PROCEDURES.
- o STAFF IDENTIFIED NEED FOR EXPERIMENTAL DATA APPLICABLE TO B&W DESIGN IN EARLY SPRING, 1981.
- o SERIES OF MEETINGS TOOK PLACE WITH B&W OWNERS REGARDING BASIS FOR STAFF DATA NEEDS.
- o NO RESOLUTION REACHED, CULMINATED IN 10/81 MEETING OF SENIOR NRC AND B&W/UTILITY MANAGEMENT.
- o AGREED TO 6-MONTH COOPERATIVE STUDY TO "IDENTIFY THE ISSUES" AND DETERMINE WHETHER A FACILITY WAS NEEDED TO OBTAIN DATA.

- o 6-MONTH STUDY ENDED IN JUNE, 1982. NO AGREEMENT. (STAFF CONCLUDED INTEGRAL FACILITY NEEDED, OWNERS CONCLUDED ONE NOT NEEDED)
- o B&W OWNERS PROPOSED TO PURCHASE DATA FROM GERDA FACILITY AT ALLIANCE RESEARCH CENTER AND SRI-II DATA (EPRI-SPONSORED) AND SUBMIT TO STAFF.
- o GERDA DATA IS PROPRIETARY DATA BEING SPONSORED BY GERMAN INDUSTRY.
- o STAFF VISITED GERDA IN JULY, 1982.
- o SECOND SENIOR UTILITY/NRC MANAGEMENT MEETING HELD JULY, 1982.

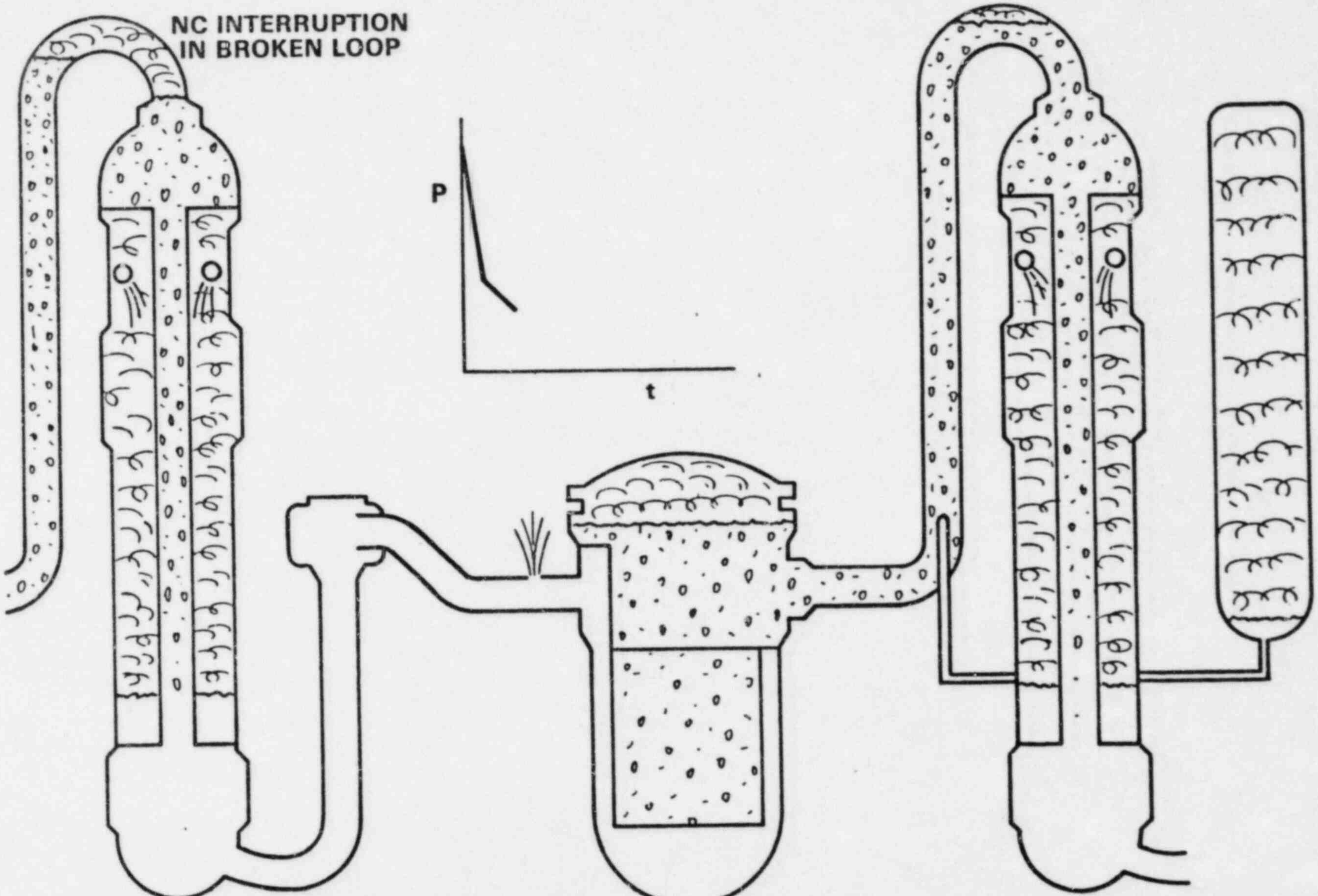
TIME = 0 SECONDS



B&W SBLOCA BEHAVIOR

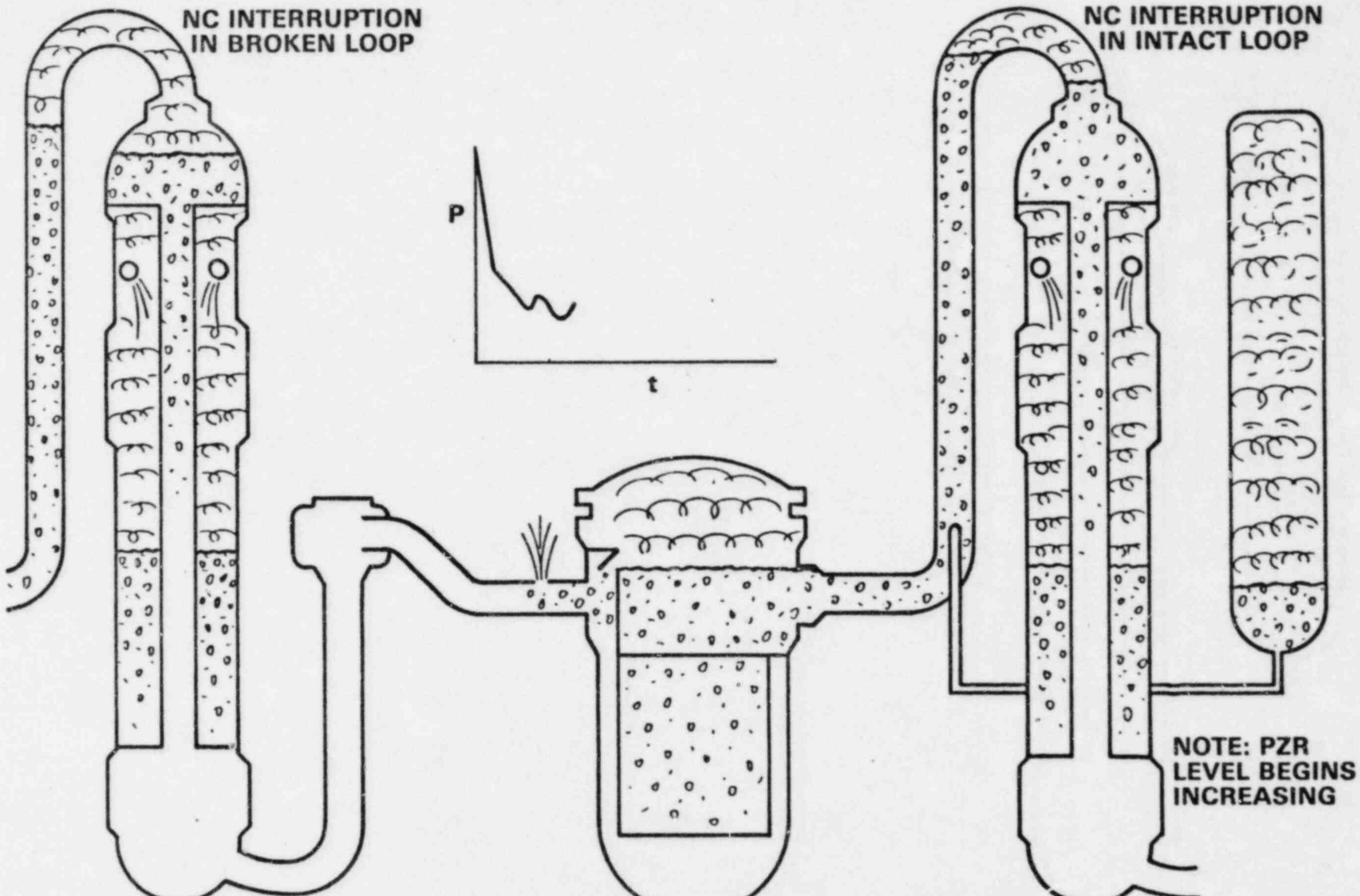
0.01 sq ft (1.35 inch diam) CL BREAK - 1 HPI PUMP

TIME ~ 300 SECONDS



B&W SBLOCA BEHAVIOR
0.01 sq ft (1.35 inch diam) CL BREAK - 1 HPI PUMP

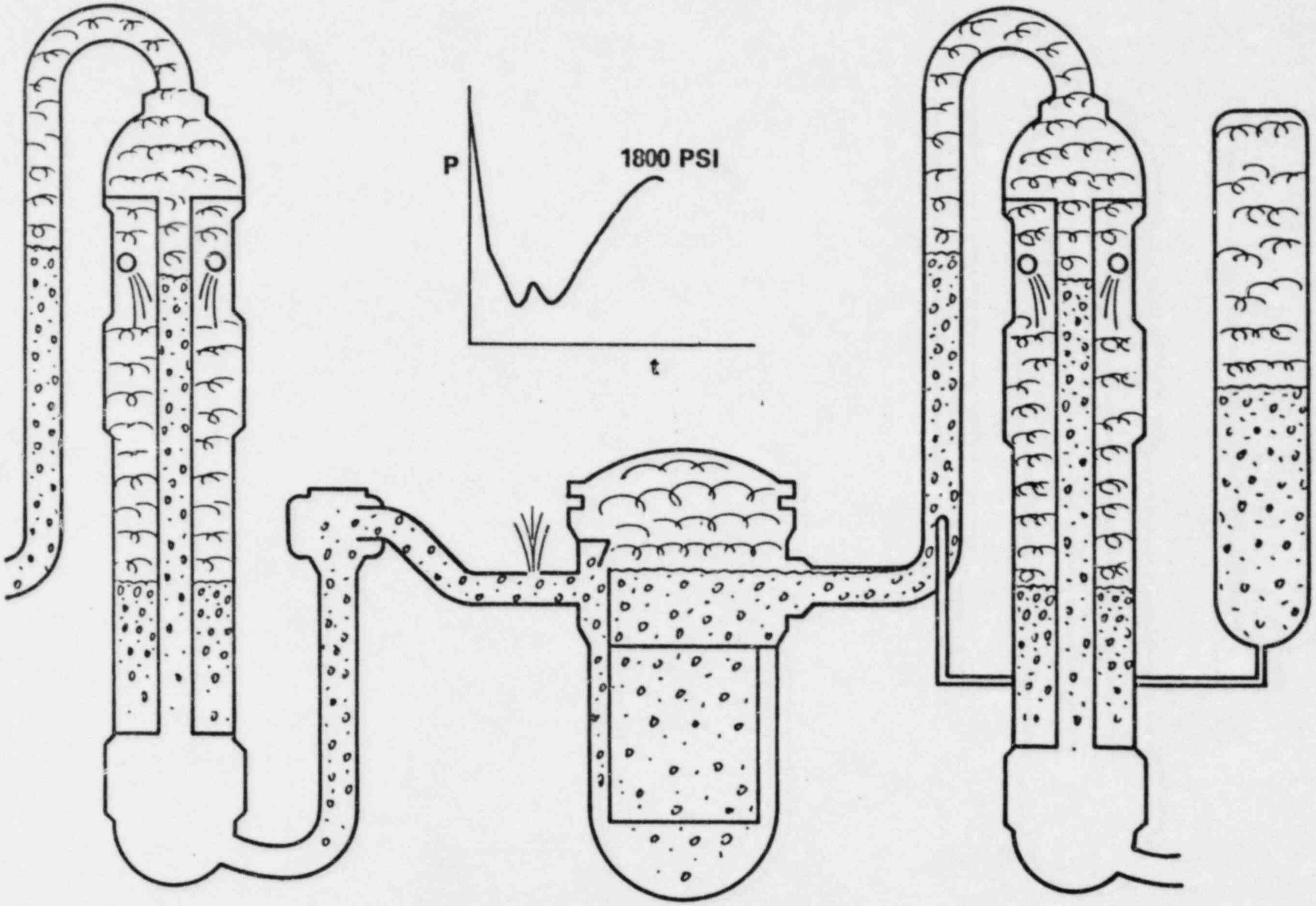
TIME ~ 600 SECONDS



B&W SBLOCA BEHAVIOR

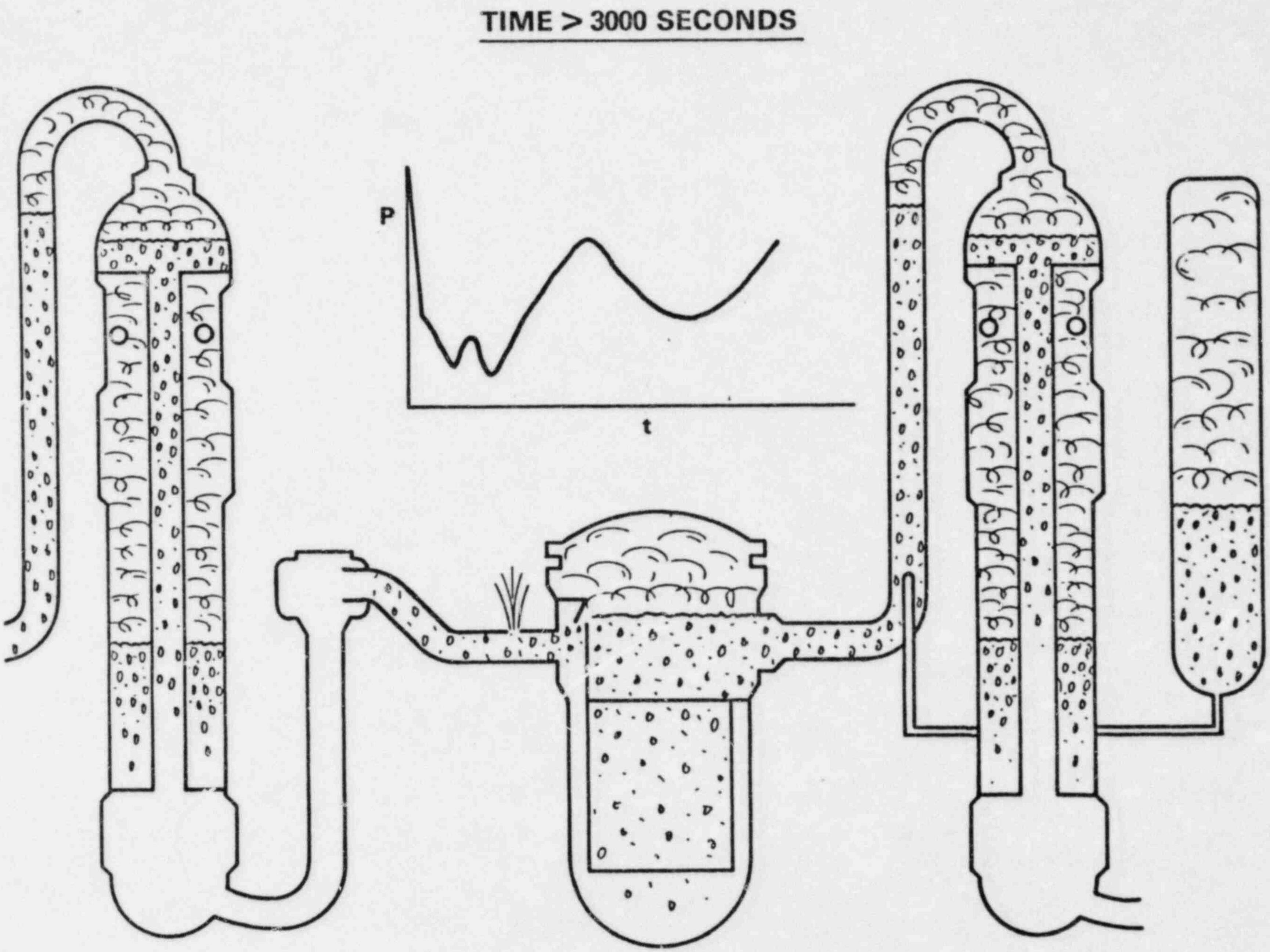
0.01 sq ft (1.35 inch diam) CL BREAK - 1 HPI PUMP

TIME ~ 1600 SECONDS



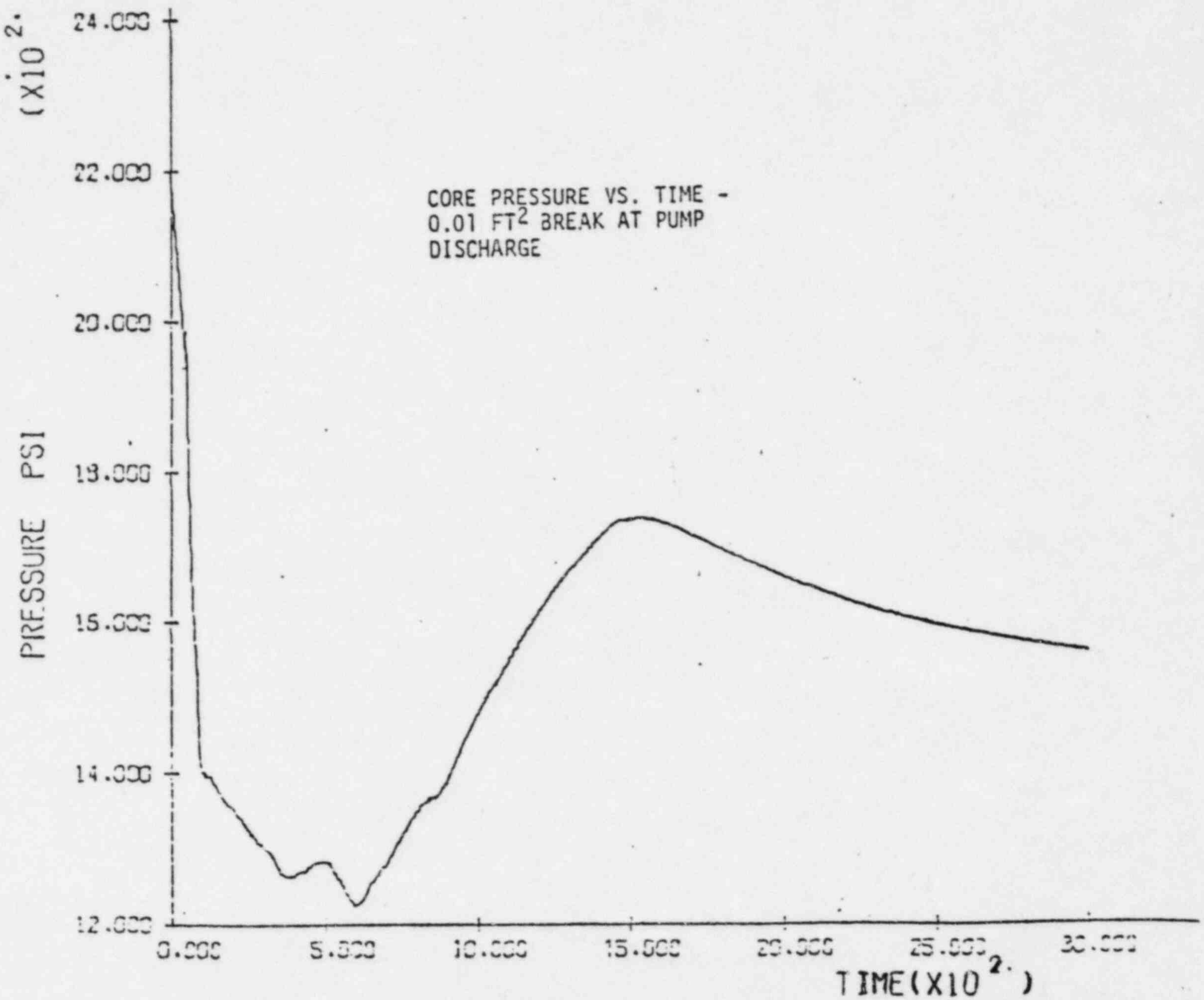
B&W SBLOCA BEHAVIOR

0.01 sq ft (1.35 inch diam) CL BREAK - 1 HPI PUMP

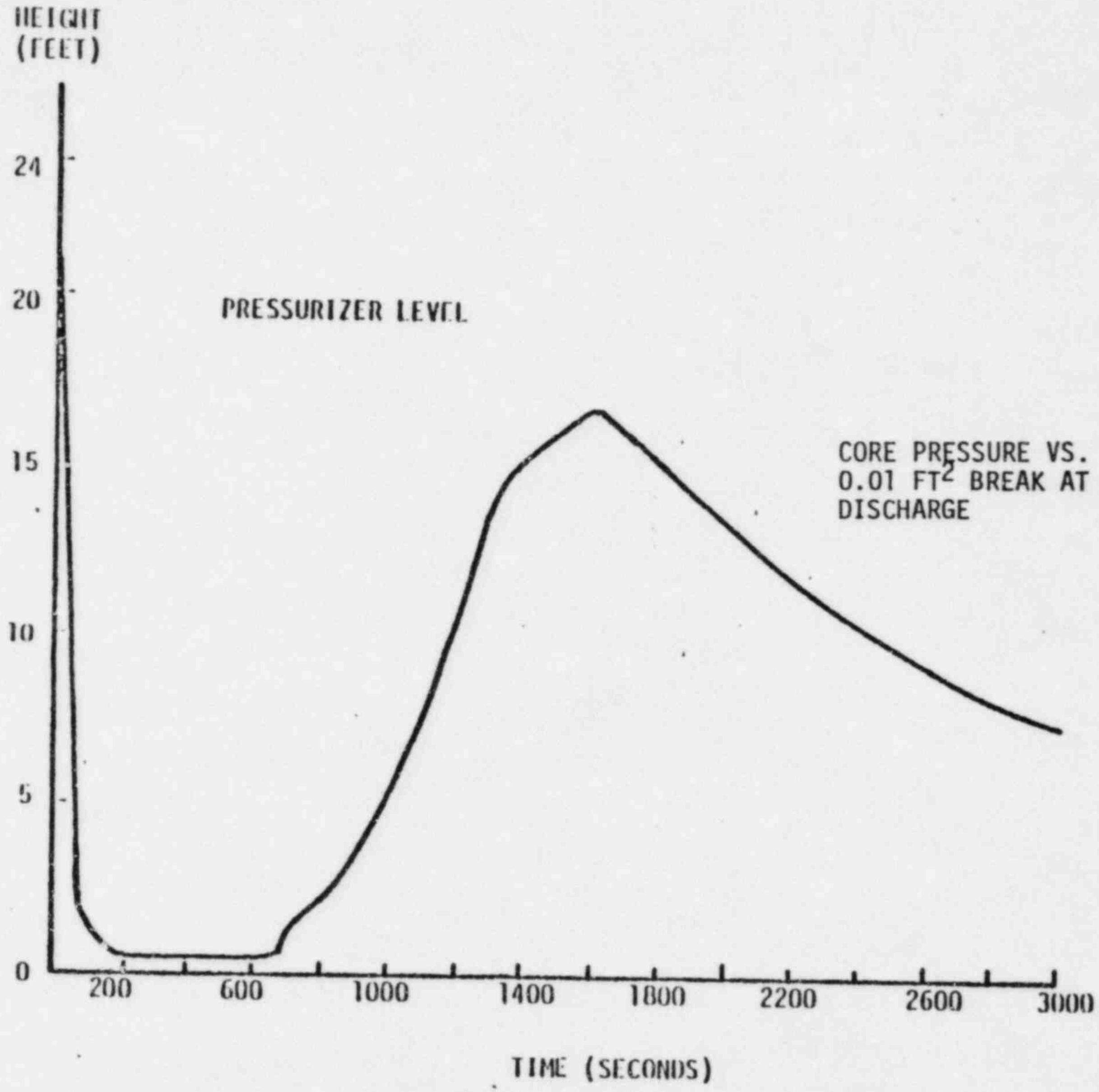


B&W SBLOCA BEHAVIOR

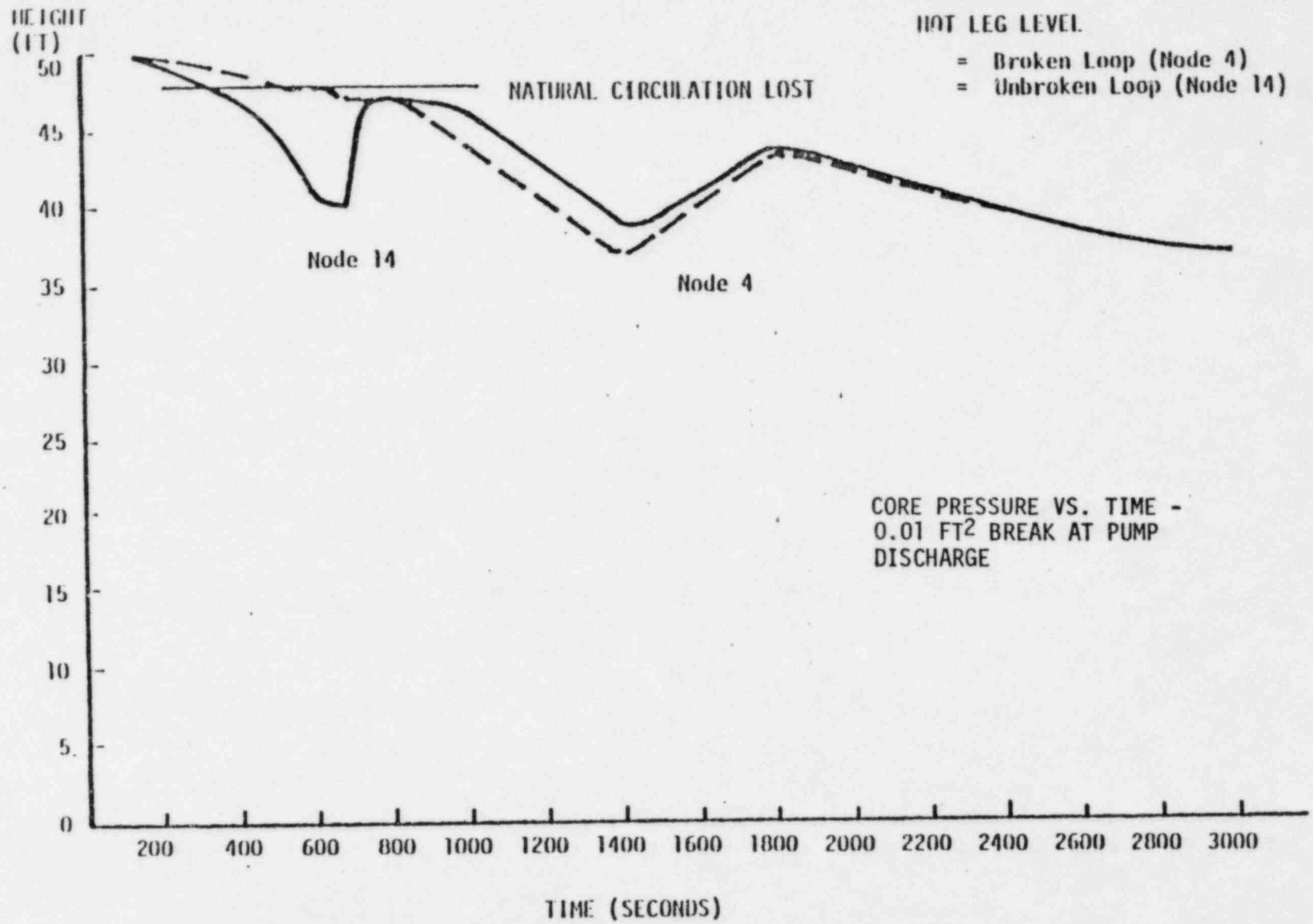
0.01 sq ft (1.35 inch diam) CL BREAK - 1 HPI PUMP



CORE PRESSURE VS. TIME -
0.01 FT2 BREAK AT PUMP
DISCHARGE
PRESSURE PSI
(X10 2.
24.000
22.000
20.000
18.000
16.000
14.000
12.000
0.000
5.000
10.000
15.000
20.000
25.000
30.000
TIME(X10 2.)



CORE PRESSURE VS. TIME - 0.01 FT^2 BREAK AT PUMP DISCHARGE



HEIGHT (FT)
50
45
40
35
30
25
20
15
10
5.
0
HOT LEG LEVEL
= Broken Loop (Node 4)
= Unbroken Loop (Node 14)
NATURAL CIRCULATION LOST
Node 14
Node 4
CORE PRESSURE VS. TIME - 0.01 FT2 BREAK AT PUMP DISCHARGE
200
400
600
800
1000
1200
1400
1600
1800
2000
2200
2400
2600
2800
3000
TIME (SECONDS)

RESULTS OF MEETING WERE:

(1) B&W OWNERS TO FURNISH FULL DESCRIPTION OF GERDA AND SRI-II FACILITIES AND DATA TO BE OBTAINED FROM THEM.

(2) OWNERS AND B&W WILL PARTICIPATE IN A TASK GROUP CHAIRED BY RES TO STUDY RELATIVE COSTS AND BENEFITS OF THREE ALTERNATIVES FOR INTEGRAL SYSTEMS TEST DATA.

- GERMAN GERDA AND SRI-II
- UPGRADED GERDA
- SEMISCALE MOD-5

(3) GROUP WILL MEET TO HEAR PRESENTATIONS BY EPRI, B&W, AND EG&G ON ALTERNATIVES.

(4) REPORT BACK TO NRR MANAGEMENT ON ALTERNATIVES.

RELATED LICENSING ISSUES

- II.K.3.30 RESOLUTION - INTEGRAL SYSTEM TEST DATA NEEDED FOR STAFF RESOLUTION OF SBLOCA MODEL UPGRADE.
- MIDLAND - (STAFF REQUIRED INTEGRAL SYSTEM TEST DATA IN SER).
- ATOG REVIEW & APPROVAL - STAFF WILL REQUIRE INTEGRAL SYSTEM TEST DATA TO CONFIRM ACCEPTABILITY OF CERTAIN EMERGENCY OPERATOR GUIDELINE ACTIONS.
- HIGH POINT VENTS - EXEMPTION REQUESTS FOR VESSEL HEAD VENTS REQUIRED BY 10 CFR 50.44 RECOMMENDED DENIED UNLESS EXPERIMENTAL CONFIRMATION OF ADEQUACY OF CANDY CANE VENTS TO REMOVE NON CONDENSIBLE GASES IN VESSEL HEAD IS PROVIDED.

- JOINT INDUSTRY/NRC TASK GROUP ESTABLISHED AND CHAIRED BY H. SULLIVAN OF RES.
- NAMED "TAG" (TEST ADVISORY GROUP).
- PURPOSE OF GROUP IS TO PRODUCE A REPORT THAT:
  - IDENTIFIES EXPERIMENTAL DATA NEEDS.
  - IDENTIFIES EXPERIMENTAL AND PLANT DATA PRESENTLY AVAILABLE OR TO BECOME AVAILABLE IN NEAR FUTURE.
  - DETERMINES THE EXTENT THAT THIS DATA BASE ADDRESSES THE EXPERIMENTAL DATA NEEDS.
  - RECOMMENDS ANY ADDITIONAL PROGRAMS THAT MIGHT BE NEEDED.

TECHNICAL ISSUES

- o INTERRUPTION OF NATURAL CIRCULATION
  - .. LOWER LOOP PLANT
  - .. RAISED LOOP PLANT
- o HOT LEG BUBBLE DYNAMICS
- o STEAM ENTRAINMENT IN HOT LEGS
- o HOT LEG FLOW REGIME
- o OPERATION TRANSIENTS (ATOG VERIFICATION)
- o VESSEL THERMAL SHOCK UNDER ZERO FLOW CONDITIONS
- o HYDRAULIC STABILITY FOLLOWING AN ACCIDENT (RECOVERY PERIOD)
- o COOLDOWN AND DEPRESSURIZATION
- o BREAK ISOLATION
- o STEAM GENERATOR TUBE RUPTURE
- o COLD LEG OSCILLATIONS
- o EFFECT OF NON-CONDENSIBLE GASES

STATUS OF TEST ADVISORY GROUP

- FIRST MEETING - SEPTEMBER 16, 1982
  - NRC PRESENTED DATA INTEREST
  - B&W PRESENTED GERDA FACILITY
  - EPRI PRESENTED SRI-II FACILITY
  - CONCLUSIONS
    - PROGRESS WAS BEING MADE
    - THE FIRST REPORT WOULD BE A JOINT REPORT

SECOND TEST ADVISORY GROUP (TAG) MEETING
OCTOBER 4, 1982

1. AGREED ON LIST OF PHENOMENA TO BE ADDRESSED BY EXPERIMENTAL PROGRAMS

2. EACH GROUP (NRC & BWOG) TO ASSIGN PRIORITY TO PHENOMENA AND TO RATE FACILITY'S POTENTIAL TO ADDRESS EACH PHENOMENON

   - GERDA
     - PRESENT FACILITY
     - UPGRADED FACILITY
   - SRI-II
   - UNIVERSITY OF MARYLAND FACILITY
   - SEMISCALE MOD-5

3. B&W AND OWNERS PREFER TO DEFER UPGRADED GERDA OR SEMISCALE MOD-5 DECISION UNTIL COMPLETION OF CURRENT GERDA AND SRI-II TEST COMMITMENT

   B&W OWNERS HAVE NOT PROVIDED TECHNICAL BASIS AS TO WHY DEFERRAL IS ACCEPTABLE. THIS IS A NECESSARY INGREDIENT FOR RESOLUTION.

4. NEXT MEETING TENTATIVELY SCHEDULED FOR 11/4/82.

- IF SATISFACTORY PROGRESS IS MADE ON AGREEING HOW TO RESOLVE RESEARCH AND CODE VERIFICATION MATTERS,

  NRR WILL SEPARATELY RESOLVE LICENSING ISSUES. INTEGRAL SYSTEMS TEST DATA WILL BE TREATED AS LONG TERM CONFIRMATION OF ADEQUACY OF SBLOCA AND OTHER ACCIDENT ANALYSIS METHODS.

- THIS APPROACH IS SIMILAR TO THE APPROACH WHICH WAS USED TO ALLOW ECCS EVALUATION MODELS TO BE USED PRIOR TO EXPERIMENTAL CONFIRMATION OF MODELS.

STATUS OF GE APPENDIX K
EXEMPTION REQUEST

- GE SUBMITTED REQUEST TO USE 1979 ANS 5-1 DECAY HEAT ON GESSAR DOCKET.
- PLAN WAS TO PETITION FOR RULEMAKING.
- STAFF IS REVIEWING TECHNICAL ADEQUACY OF 1979 ANS 5.1 DECAY HEAT STANDARD.
- EXPECT TO FIND IT TECHNICALLY ACCEPTABLE.
- STAFF WILL NOT ADDRESS <u>HOW</u> NEW DECAY HEAT SHOULD BE USED IN LICENSING PROCESS DURING TECHNICAL REVIEW.
- IF LICENSEES WISH TO USE NEW DECAY HEAT, EXEMPTION REQUEST TO APPENDIX K SHOULD BE SUBMITTED.
- USE OF SAFER CODE FOR LBLOCA ANALYSIS SHOULD RESULT IN LBLOCA PCT'S $\ll$ 2200°F. PLANTS NO LONGER WOULD BE LOCA-LIMITED.
- STAFF REVIEW ALMOST COMPLETE.

- WHY DECAY HEAT?
  - SAFER EXPENSIVE TO RUN
  - USE DECAY HEAT IN HEATUP CALCULATION ONLY: LESS EXPENSIVE, QUICKER

GERDA

Purpose:

Provide Post—SBLOCA Integral-effects Data for Comparison with Code Predictions of Model Behavior

T15

GERDA SCALING CRITERIA

ELEVATIONS	FULL SCALE MK ELEVATIONS MAINTAINED
PHENOMENA	IMPORTANT SBLOCA PHENOMENA PRESERVED (e.g. HOT LEG TWO-PHASE FLOW BEHAVIOR)
VOLUME	COMPONENTS SCALED BY RATIO OF STEAM GENERATOR TUBES (32,026/19 = 1686)
IRRECOVERABLE PRESSURE LOSSES	ORIFICES USED TO MATCH MK LOSSES

Categorization Of Concerns

II K 3.30 Concerns

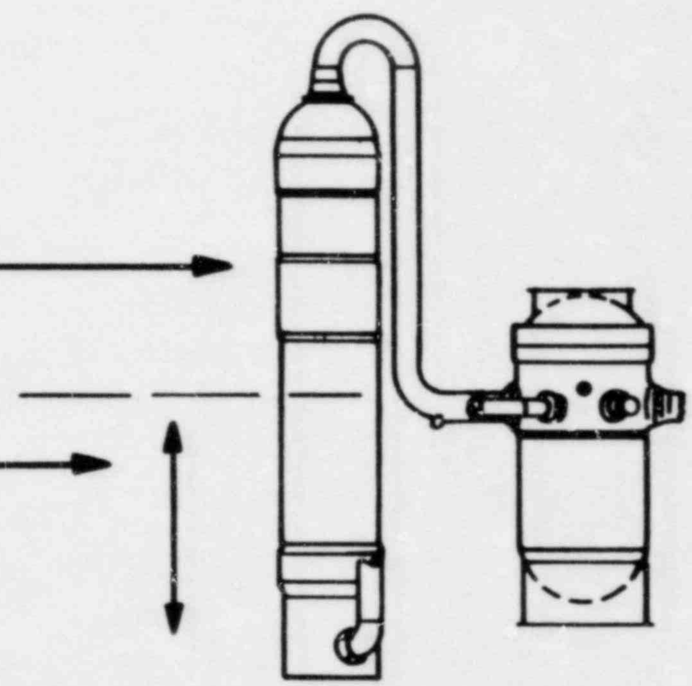
CORE COOLING
- S/G MODEL
- PZR MODEL
- ECCS INJ
- TWO-PHASE PUMP MODEL
- BREAK MODEL
- ETC

OPERATION
- TWO PHASE NATURAL CIRC
- INTERRUPTION OF NATURAL CIRC
- BOILER CONDENSER MODE
- REFILL
- NON CONDENSABLE GASES
- HIGH POINT VENTS
- VENT VALVES

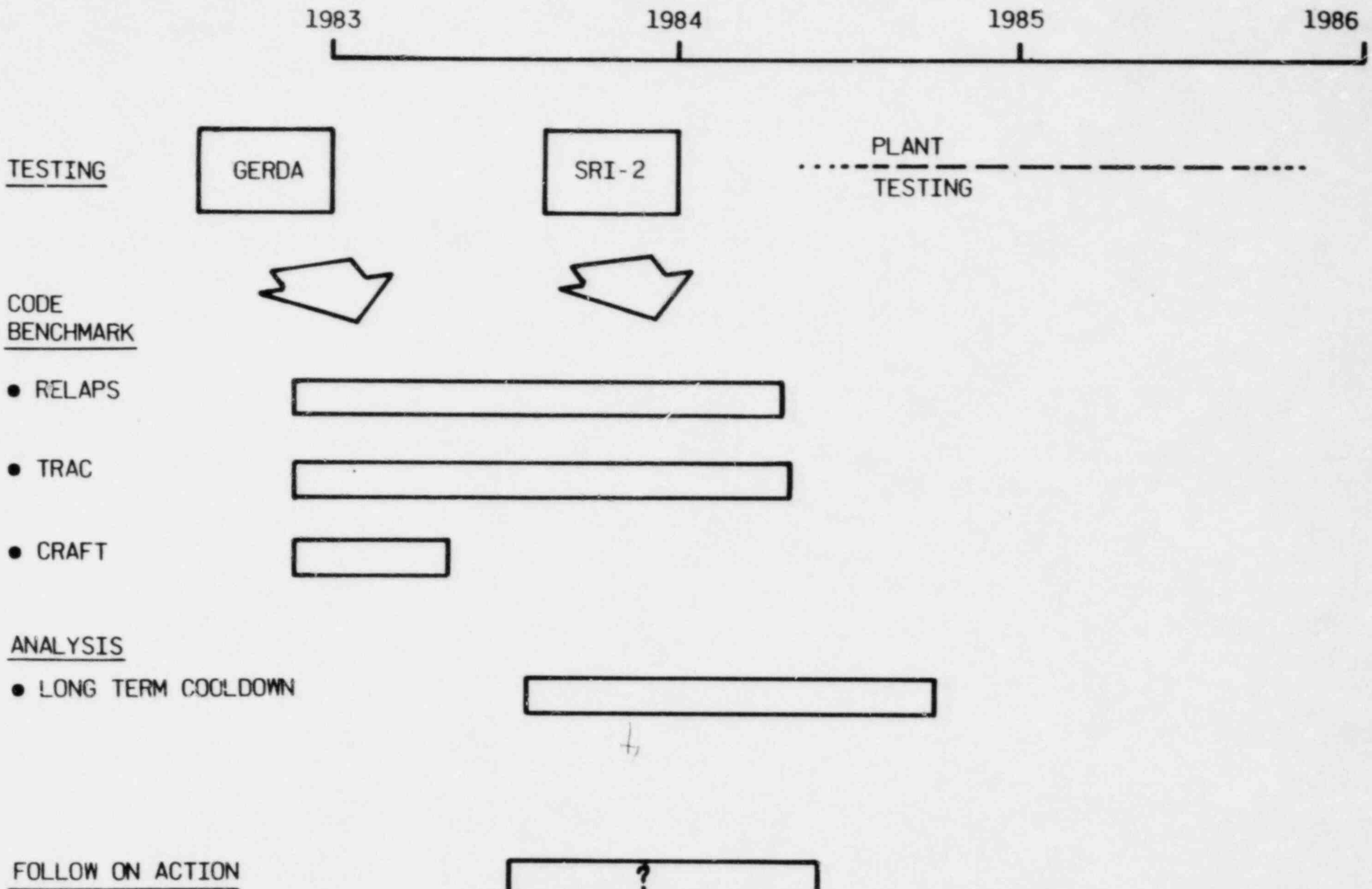
OTHER ISSUES
- COMBINATION SBLOCA AND SLB
- SGTR
- BUMP THE PUMP
- COLD LEG OSCILLATIONS

CRAFT

Best Est Model



INTEGRATED SBLOCA PROGRAM



ARC EXPERIENCE

- 15 YEARS TESTING 19 AND 37 TUBE OTSG MODELS.
- DESIGNED AND CONSTRUCTED 10 MW CHF FACILITY WHICH HAS BEEN ACCLAIMED BY OUTSIDE OBSERVERS "THE FINEST IN THE WORLD."
- HIGH QUALITY QA PROGRAM AS EVIDENCED BY MARK C CHF DATA BASE WHICH HAS THE SMALLEST SCATTER IN THE INDUSTRY.

[illegible]

I. [illegible]

1. SG Characterization (Steady State, Forced Flow).
2. Auxiliary Feedwater (AFW) Effects in Forced Flow.
3. SG Transients (Boiloff, Blowdown, and Refill).
4. AFW Effects in natural circulation.

II. NATURAL CIRCULATION (NC)

5. NC Characterization.
6. NC Transient: NC initiation in a [illegible] and isothermal system.
7. NC cooldown.
8. NC Flow Transients: [illegible] NC following the interruption of forced flow.

III. CONDENSING [illegible]

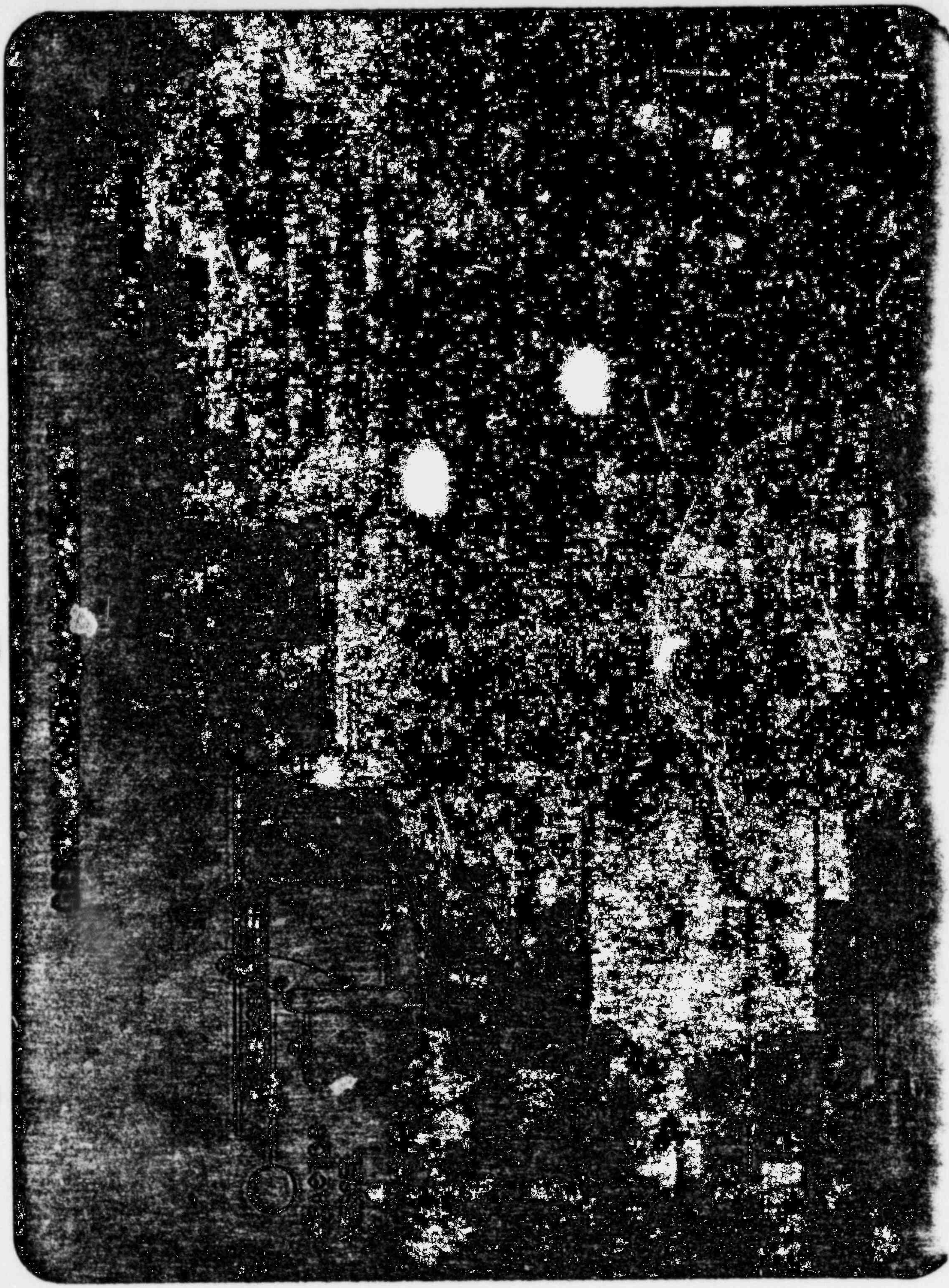
9. BCM Characterization (Steady state without non-condensible gases, NCG).
10. BCM With Non-Condensible Gases (Steady State)
11. NC Transients: Establish BCM after the interruption of NC.

IV. REFILL [illegible]

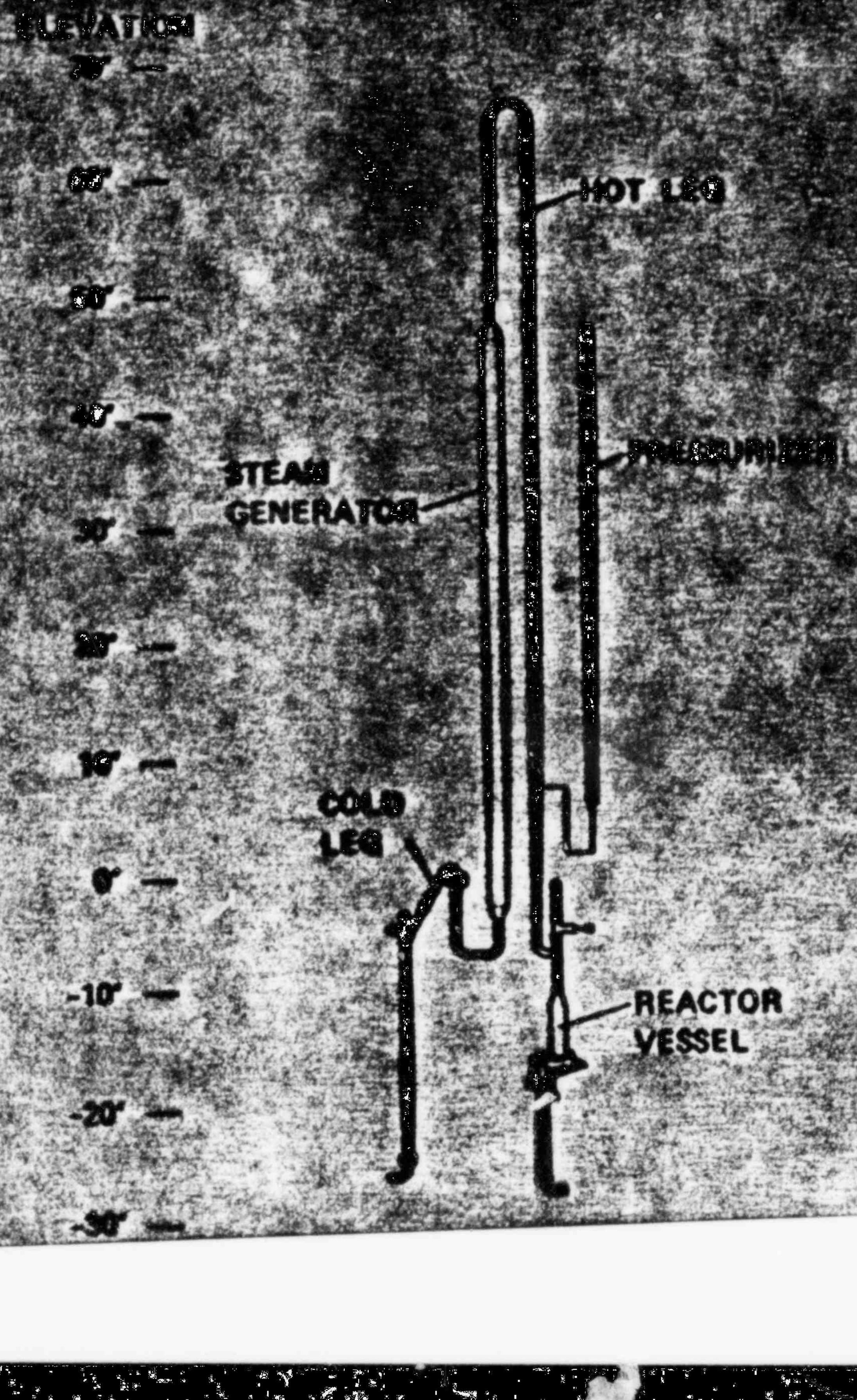
12. Refill Characterization and AFW Effects.
13. High Pressure Injection (HPI) effects on refill (Vary HPI distribution and redundancy).
14. Leak effects on refill (Vary break size and location).
15. NCG effects on refill.

V. COMPOSITE EFFECTS

16. Complete SBLOCA Transient (Including NC, interruption of NC, BCM, refill, cooldown, and depressurization).



GERDA SCALED ARRANGEMENT



ACRS MEETING - OCTOBER 7, 1982

Status of the ECCS Rule Revision

A. Background

On December 6, 1978, the Nuclear Regulatory Commission published in the Federal Register (43 FR 57157) an advance notice of proposed rulemaking on "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Plants" (ECCS rule). In the notice the Commission indicated that it was considering rulemaking to take place in two phases as follows:

1. Phase 1 (Short Term) Changes

This would involve procedure-oriented and minor technical changes to the ECCS rule but would have little impact on the overall conservatism of the rule. The revisions would allow minor changes to the ECCS codes without requiring a complete reanalysis by the applicants and licensees. The technical changes would improve the realism of the evaluation model but not significantly affect the overall conservatism of the ECCS rule. The Phase 1 changes would include the following topics:

(a) Reanalysis requirements for construction permit applications

(b) Reanalysis requirements for operating license applications and licensed plants

(c) Return to nucleate boiling

(d) Steam cooling requirements for flooding rates below one inch per second

(e) Transition boiling correlation reference.

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2. Phase 2 (Long Term) Changes

This would involve a more comprehensive rulemaking action to incorporate new technical data and operating experience into the ECCS analysis and review procedures. The work would include an assessment of the impact of the proposed changes on the overall conservatism of the ECCS rule. The objective would be to preserve an appropriate overall level of conservatism. The Phase 2 changes would include the following topics:

(a) Fission product decay heat rate

(b) Zircaloy oxidation rate

(c) Additional data including that which may indicate the present rule is less conservative than previously believed.

(d) New operating experience.

The Commission invited advice and recommendations on several questions concerning the proposed areas of revision to the ECCS rule by February 8, 1979. Of course, it was only 7 weeks later that the Three Mile Island accident occurred. Twenty five persons submitted comments in response to the invitation which can be summarized as follows:

1. The ECCS model should be based on realistic analysis.
2. The rule should permit greater flexibility to meet the acceptance criteria and to use research information.
3. The Phase 1 scope should be expanded to include the new decay heat and zircaloy oxidation data.
4. Evaluation of the ECCS should be treated as other DBA's without all the detail required in Appendix K.

5. There should be no extensive rulemaking, just reinterpretation of the existing rules.

NRC staff activity on the ECCS rule was severely curtailed as a result of the high priority effort required to respond to the Three Mile Island accident. The matter essentially sat dormant through 1980 and 1981 when it was brought up again in the context of simplifying and streamlining the regulatory process. The NRC staff reviewed the comments received and additionally because of the delay since the advance notice, the staff conducted a telephone survey of the industry. The survey basically reiterated the comments previously submitted by the respondents. Therefore, since the general idea still appears a sensible one, the plans to revise the ECCS rule were reinstated.

B. Plans

1. Phase 1

The staff has recommended that the Commission proceed with the Phase 1 changes as originally described in the advance notice of proposed rulemaking. This will provide a significant measure of relief from reanalysis requirements which do not substantially contribute to safety as well as allow the use of certain recently developed research data that will help facilitate analyses; these changes will not significantly impact the overall conservatism.

The staff has also recommended that the decay heat aspects of Appendix K not be included during the Phase 1 changes. The decay heat curve in use provides one of the major sources of conservatism in the ECCS analysis. Until a thorough evaluation of the conservatism is completed, the requirements are not expected to be changed. The consideration of the new decay heat correlation will be included as part of the Phase 2 program.

2. Phase 2

With regard to the Phase 2 program, the staff has recommended that if any significant rulemaking is initiated which involves substantial changes to present technical requirements, then the rulemaking should consider an overall rule revision or even the elimination of the Appendix K rule (e.g., replace the rule with a Regulatory Guide). Thus, the Phase 2 approach originally proposed will most likely be revised. It is expected that a revised Phase 2 approach will not be ready for Commission consideration until late 1983 or early 1984.

3. Interim Regulatory Procedures

A request has been made by GE to permit the use of the new decay heat standard for current evaluations. The NRC staff is considering an approach whereby the burden of demonstration of adequate conservatism will rest with the nuclear steam supplier. If the staff agrees with the analysis it would recommend that the generic safety analysis report (SAR) submitted by the nuclear steam supplier be amended to include the new standard. Utilities would then be expected to request Commission exemptions, from that part of Appendix K that concerns decay heat requirements, by referencing the revised generic SAR. This approach would be considered as an interim procedure and would only be used pending completion of the more substantive Phase 2 program.

BACKGROUND OF ECCS RULE REVISION

ADVANCE NOTICE - PUBLISHED 12/6/~~82~~ 78

PHASE 1 (SHORT TERM)

- REANALYSIS REQUIREMENTS FOR CPs
- REANALYSIS REQUIREMENTS FOR OLs AND ORs
- RETURN TO NUCLEATE BOILING
- STEAM COOLING REQUIREMENTS FOR FLOODING RATES BELOW 1"/SEC.
- TRANSITION BOILING CORRELATION REF.

PHASE 2 (LONG TERM)

- FISSION PRODUCT DECAY HEAT RATE
- ZIRCALOY OXIDATION RATE
- ADDITIONAL NEW DATA
- NEW OPERATING EXPERIENCE

RESPONSE TO ADVANCED NOTICE ON ECCS RULEMAKING

Private	Utilities	Vendors	Government
3	15	5	2

Major Comments

1. Model Should Be Based on Realistic Analysis

2. Rule Should Permit Greater Flexibility to Meet Criteria and Use Research Information

3. Phase 1 Scope Should Be Expanded to Include New Decay Heat and Zircaloy Oxide Data

4. ECCS Should Be Treated as Other DBA's

5. No Extensive Rulemaking - Just Reinterpretation

PROPOSED ACTIONS

- PROCEED WITH PHASE 1
  - RELIEF FROM REANALYSIS REQUIREMENTS
  - ALLOW USE OF SOME NEW RESEARCH DATA
  - NO SIGNIFICANT IMPACT ON CONSERVATISM
- DECAY HEAT TO BE IN PHASE 2
  - A MAJOR SOURCE OF CONSERVATISM
- PHASE 2
  - IF SUBSTANTIAL CHANGES - CONSIDER OVERALL RULE REVISION OR CONVERSION OF APPENDIX K TO A GUIDE
  - APPROACH TO BE RECOMMENDED EARLY 1984
- INTERIM PROCEDURES ON USE OF NEW DECAY HEAT STANDARD
  - GE TO DEMONSTRATE CONSERVATISM
  - GESSAR AMENDED TO INCLUDE NEW STANDARD
  - UTILITIES REQUEST EXEMPTIONS FROM APPENDIX K BY REFERENCING REVISED GESSAR

SUMMARY OF ECCS RULE REVISIONS

REANALYSIS REQUIREMENTS

FOR CP'S ONLY

- NO REANALYSIS IF ΔPCT $\rightarrow$ 200^{o}F

FOR ALL CP'S, OL'S AND OR'S

- NO REANALYSIS IF ΔPCT $>$ -20^{o}F
- 1 YEAR FOR REANALYSIS IF ΔPCT $\rightarrow$ 100^{o}F
- DEFINES A SIGNIFICANT ΔPCT ($>$ 20^{o}F)
- CLARIFIES DOCUMENTATION REQUIREMENTS

RETURN TO NUCLEATE BOILING

- ALLOWS RETURN TO NUCLEATE BOILING DURING BLOWDOWN WHEN JUSTIFIED

STEAM COOLING REQUIREMENTS FOR FLOODING RATES BELOW 1"/SEC

- DELETE REQUIREMENT THAT COOLING IS BY STEAM ONLY FOR FLOODING RATES $<$ 1"/SEC
- BASED ON EXPERIMENTAL DATA WITH FLOW BLOCKAGE

TRANSITION BOILING CORRELATION REFERENCE CORRECTION

PURPOSE

REQUEST COMMISSION

APPROVAL OF

RECOMMENDATIONS FOR

IMPLEMENTATION OF

TMI ACTION PLAN II.F.2 -

"INSTRUMENTATION FOR DETECTION

OF INADEQUATE CORE COOLING"

T18

ISSUES REMAINING FROM JANUARY 1982 COMMISSION MEETING

*DEMONSTRATE NEED AND USES FOR PROPOSED INSTRUMENTATION

*ALLAY CONCERN ABOUT AMBIGUOUS INFORMATION

*EXAMINE COSTS AND BENEFITS

*INTEGRATE INTO EMERGENCY OPERATING PROCEDURES AND CONTROL ROOM DESIGN REVIEW

*ESTABLISH A RATIONAL SCHEDULE FOR IMPLEMENTATION

ACTIONS TO RESOLVE ISSUES

*FEBRUARY NRC/INDUSTRY MEETING

*COST/BENEFIT EVALUATION OF INVENTORY MONITOR

*PUBLICATION OF GENERIC DESIGN EVALUATION REPORTS

*CRGR MEETING (MARCH)

*APRIL ACRS MEETING

*COST/BENEFIT STUDY OF OVERALL ICC SYSTEM

*FMEA REVIEW

*SECOND CRGR MEETING (SEPTEMBER)

*OCTOBER ACRS + COMMISSION BRIEFINGS

SAFETY BENEFITS

*IMPROVE RELIABILITY IN DIAGNOSING THE APPROACH TO AND THE ONSET OF ICC, AND IN ASSESSING THE EFFECTIVENESS OF RESPONSE TAKEN TO RESTORE CORE COOLING.

*REDUCES CHANCE OF OPERATOR CONFUSION, MISDIAGNOSIS OR ERROR IN RESPONDING TO:

- INCIDENTS OF MODERATE FREQUENCY LEADING TO STEAM BUBBLE FORMATION IN THE RCS, E.G.,

    SG TUBE RUPTURES
    LOSS OF INSTRUMENT BUS OR OTHER CONTROL SYSTEM UPSETS
    RC PUMP SEAL FAILURES
    OVERCOOLING EVENTS
    NORMAL RCS COOLDOWN

- EVENTS INVOLVING MULTIPLE FAULTS

- SMALL BREAK LOCAs

*AIDS EARLY WARNING AND OFF-SITE EMERGENCY RESPONSE DECISIONS

INSTALLED COST (K$/PLANT)

FOR ESTIMATED PLANTS

Design Options

1. Reference Design - meets NUREG-0737 design requirements.
2. Delete all seismic design requirements from reference design.
3. Delete environmental qualification requirements, except seismic, from reference design.
4. Delete single failure design requirements (redundancy) from reference design.
5. Delete Class 1E power source requirement from reference design.

The NRR estimate of costs associated with each design option is shown below in Table I.

Table I

ICC Instrumentation	Fit Status	OPTION $1_{(c)}$ NRR ESTIMATES	$2_{(s)}$	$3_{(s)}$	$4_{(s)}$	$5_{(s)}$	$Range_{(c)}$ INDUSTRY ESTIMATES
Core Exit Thermocouple	BF	2,148	14	35	21	3	648-6,280
	FF	948	15	12	22	5	551-1,250
Subcooling Margin Monitor	BF	325	19	30	30	2	70-500
	FF	658	16	15	30	10	100-1,750
Inventory Trending W/RCS Pumps Off	BF	3,176	9	16	30	2	1,530-5,280
	FF	1,826	4	15	16	2	195-3,694
Inventory Trending W/ RCS Pumps On	BF	240	1	1	8	0	200-280
	FF	200	10	20	50	0	200
Overall ICC Instrumentation	BF	5,889	11	23	26	2	2,488-12,340
	FF	3,632	9	14	22	4	1,046-6,894

NOTE: C- Cost ($1,000/Plant); S- Savings in % (Compared with Option 1); BF- Backfit; FF- Forward Fit.

INSTALLATION AND PROCUREMENT STATUS

OF

INVENTORY TRENDING SYSTEM

(SEPTEMBER 1982)

*WESTINGHOUSE DP SYSTEM - 32 ORDERED

- 8 INSTALLED AND CALIBRATED (2 OLs)
- 2 INSTALLED, FILLED, AND NOT CALIBRATED
- 2 INSTALLED AND NEED MODIFICATION
- 4 INSTALLED AND WILL FILL
- 1 PARTIALLY INSTALLED
- 15 TO BE INSTALLED

*CE HJTC SYSTEM - 21 ORDERED

- 21 TO BE INSTALLED

RECOMMENDATIONS

*CE HJTC AND WESTINGHOUSE DP SYSTEM ARE ACCEPTABLE GENERIC DESIGNS – must also a plant specific implementation review

*B&W DP MEASUREMENT TECHNIQUES ARE ACCEPTABLE IN PRINCIPLE PROVIDED THAT THEY:

- MONITOR COOLANT INVENTORY FROM VESSEL HEAD AND FROM TOP OF HOT LEG TO BOTTOM OF HOT LEG
- ARE SUPPLEMENTED BY INVENTORY TRENDING WITH PUMPS ON; E.G., PUMP CURRENT OR PUMP POWER MONITOR
- MEET NUREG-0737 DESIGN REQUIREMENTS

*FOR THE DESIGN, INSTALLATION, AND UPGRADE OF ICC INSTRUMENTATION SUBSYSTEMS

- NUREG-0737 DESIGN SPECIFICATIONS ARE A REQUIREMENT
- FOR EXISTING INSTALLATIONS SOME DEVIATIONS MAY BE GRANTED WHERE JUSTIFIED AND CONSISTENT WITH EQ RULE

*LICENSEES NOT YET COMMITTED SHOULD BE ORDERED TO CONCLUDE THEIR DESIGN REVIEW AND SUBMIT DETAILED ENGINEERING, PROCUREMENT, AND INSTALLATION SCHEDULES BY JANUARY 1, 1983

*NEGOTIATE PRACTICAL SCHEDULES FOR IMPLEMENTATION ON CASE-BY-CASE BASIS

*PREREQUISITES TO IMPLEMENTATION OF VOID INDICATOR OR INVENTORY TRACKING SYSTEMS

- NRC STAFF REVIEW AND APPROVAL OF PLANT SPECIFIC INSTALLATION AND CALIBRATION SUBMITTAL AND EMERGENCY OPERATING PROCEDURE GUIDELINES FOR THE OVERALL ICC PACKAGE
- INTEGRATION OF THE OVERALL ICC SYSTEM INTO TASK ANALYSIS PORTION OF DETAILED CONTROL ROOM DESIGN REVIEW BY THE LICENSEE
- OPERATOR TRAINING IN OPERATION AND LIMITATIONS OF THE SYSTEM

ACRS Meeting 10/7/82

Stress Corrosion Cracking

Thick Wall Stainless Steel

Recirculation Piping at BWR

Polk

T19

Vo 4 — 7.6

SSINS No.: 6835
IN 82-39

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D. C. 20555

September 21, 1982

IE INFORMATION NOTICE NO. 82-39: SERVICE DEGRADATION OF THICK WALL STAINLESS STEEL RECIRCULATION SYSTEM PIPING AT A BWR PLANT

Addressees:

All boiling water reactor facilities holding an operating license (OL) or construction permit (CP).

Purpose:

This notice is to provide licensees and construction permit holders available information about the degradation of the primary pressure boundary at Nine Mile Point Unit 1 due to intergranular stress corrosion cracking. Recipients should review this information relative to their facilities. If NRC evaluation so indicates, further licensee action may be requested. In the interim, we expect licensees to review the relevance of this information for applicability to their facilities.

Description of Circumstances:

The Nine Mile Point Nuclear Station Unit 1 (NMP Unit 1) was shut down in order to replace recirculation pump seals. On March 23, 1982, leakage was visually detected at two of the ten recirculation loop safe ends during a primary system hydrotest at 900 psig to test the seals. Further visual inspection identified three pin-hole indications and a single ½-inch long axial indication, all of which were located in the heat affected zone of the welds where the safe end joined the pipe.

On March 26, 1982, an ultrasonic examination of the two affected safe ends and one other safe end confirmed the presence of intermittent cracking indications around the pipe's inside diameter. Further ultrasonic examination of the welds joining the pump discharge casting to the riser elbow also revealed cracking in weld heat affected zones on the inside diameter (ID) of the elbows. This was later confirmed by dye penetrant examination.

Because the cracks were confirmed at the welds of the safe ends and riser elbows, the ultrasonic examination was extended to all of the remaining welds in the five loops of the primary system, wherever radiation levels permitted. The results of this examination show ID cracking at a large number of the welds examined.

Two boat samples removed from the area of the through-wall cracks in one safe end were sent to General Electric and Battelle Laboratories, respectively, for evaluation. A boat sample from the crack region of the elbow weld was also evaluated by Sylvester Associates, consultants to the licensee. The results of these metallurgical evaluations concluded the degradation was due to intergranular stress corrosion cracking (IGSCC) in the sensitized region of the welds' heat affected zones. Further metallurgical investigation is being pursued to determine, as far as possible, the probable cause(s) of the problem.

Based on the results of the examinations and investigations to date, the licensee will replace the safe ends and 28-inch recirculation piping in all five loops of the system. Replacement of the branch piping out to the first isolation valve is also being considered; however, no final decision in this regard has been made at this time.

All replacement material will be stainless steel type 316 nuclear grade consistent with NUREG-0313, Revision 1 requirements. The actual replacement will be accomplished in accordance with ASME Boiler and Pressure Vessel Code, Section XI, 1977 Edition and Addenda through summer 1978. Welding will be performed in accordance with Section IX, 1978. Fitup requirements will be in accordance with ANSI Pressure Piping Code B31.1-1977 and Addenda through winter 1979. The replaced system configuration will duplicate the original design.

All ten recirculation system safe ends at NMP Unit 1 had been previously examined volumetrically by ultrasonic techniques at each refueling outage under an augmented inservice inspection program. This was in addition to the ASME code required inservice inspection program applied to other system welds. The augmented program was required because of IGSCC problems experienced with furnace-sensitized safe ends at this and other BWR plants.

It is important to note that the programs conducted under the normal and augmented programs did not indicate a pending problem. Examinations were performed during 1979 and 1981. The procedure employed during the 1981 augmented program for the safe ends was based on ultrasonic test (UT) using the EPRI transducer with a flat calibration block which was stated to be capable of detecting IGSCC at the code required gain or sensitivity level. The procedure differed from the GE recommended procedures in specifying less gain, and differed significantly in the calibration standards and data recording requirements, thus resulting in reduced sensitivity compared to the GE recommended procedures.

After leakage was visually observed on March 23, 1982, a UT examination of the safe ends was performed using the same method employed in the 1981 augmented program. Many safe ends exhibited code "reportable," but not rejectable indications. However, when an ultrasonic sensitivity of 10 decibels above code calibration sensitivity was employed, greater reliability was realized in detecting the presence and full extent of the IGSCC problems with the thick wall piping welds, both at the safe ends and at other locations in the reactor coolant system. The generic implications of the above variances is under further review by the NRC staff.

This IE information notice is to advise licensees of further occurrences of the prevailing IGSCC problem that is under continuing review by the NRC staff.

If you have any questions regarding this matter, please contact the Regional Administrator of the appropriate Regional Office, or this Office.

E. Jordan
Edward L. Jordan, Director
Division of Engineering and
Quality Assurance
Office of Inspection and Enforcement

Technical Contact: W. J. Collins
301-492-7275

Attachment:
List of Recently Issued IE Information Notices

LIST OF RECENTLY ISSUED
IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
82-38	Changes in Format and Distribution System for IE Bulletins, Circulars and Information Notices	9/22/82	All NRC licensees
82-34	Welds In Main Control Pannels	09/17/82	All power reactor facilities holding an OL or CP
82-37	Cracking in the Upper Shell to Transition Cone Girth Weld of a Steam Generator at an Operating Pressurized Water Reactor	9/16/82	All power reactor facilities holding an OL or CP
82-36	Respirator Users Warning for Certain 5-Minute Emergency Escape Self-Contained Apparatus	9/2/82	All power reactor facilities holding an OL or CP, fuel facilities and Priority I material licensees
82-35	Failure of Three Check Valves on High Pressure Injection Lines to Pass Flow	8/25/82	All power reactor facilities holding an OL or CP
82-34	Welds in Main Control Panels	8/25/82	All power reactor facilities holding an OL or CP
82-33	Control of Radiation Levels in Unrestricted Areas Adjacent to Brachytherapy Patients	8/20/82	All Medical Institutions
82-32	Contamination of Reactor Coolant System by Organic Cleaning Solvents	8/19/82	All power reactor facilities holding an OL or CP
82-31	Overexposure of Diver During Work in Fuel Storage Pool	7/28/82	All power reactor facilities holding an OL or CP

OL = Operating License
CP = Construction Permit

BACKGROUND

- o FURNACE SENSITIZED SAFE ENDS LEAKED DURING HYDRO IN MARCH 1982
  - NO CRACKS FOUND IN UT EXAM 9 MONTHS EARLIER
  - IGSCC CONFIRMED
  - BEING REPLACED

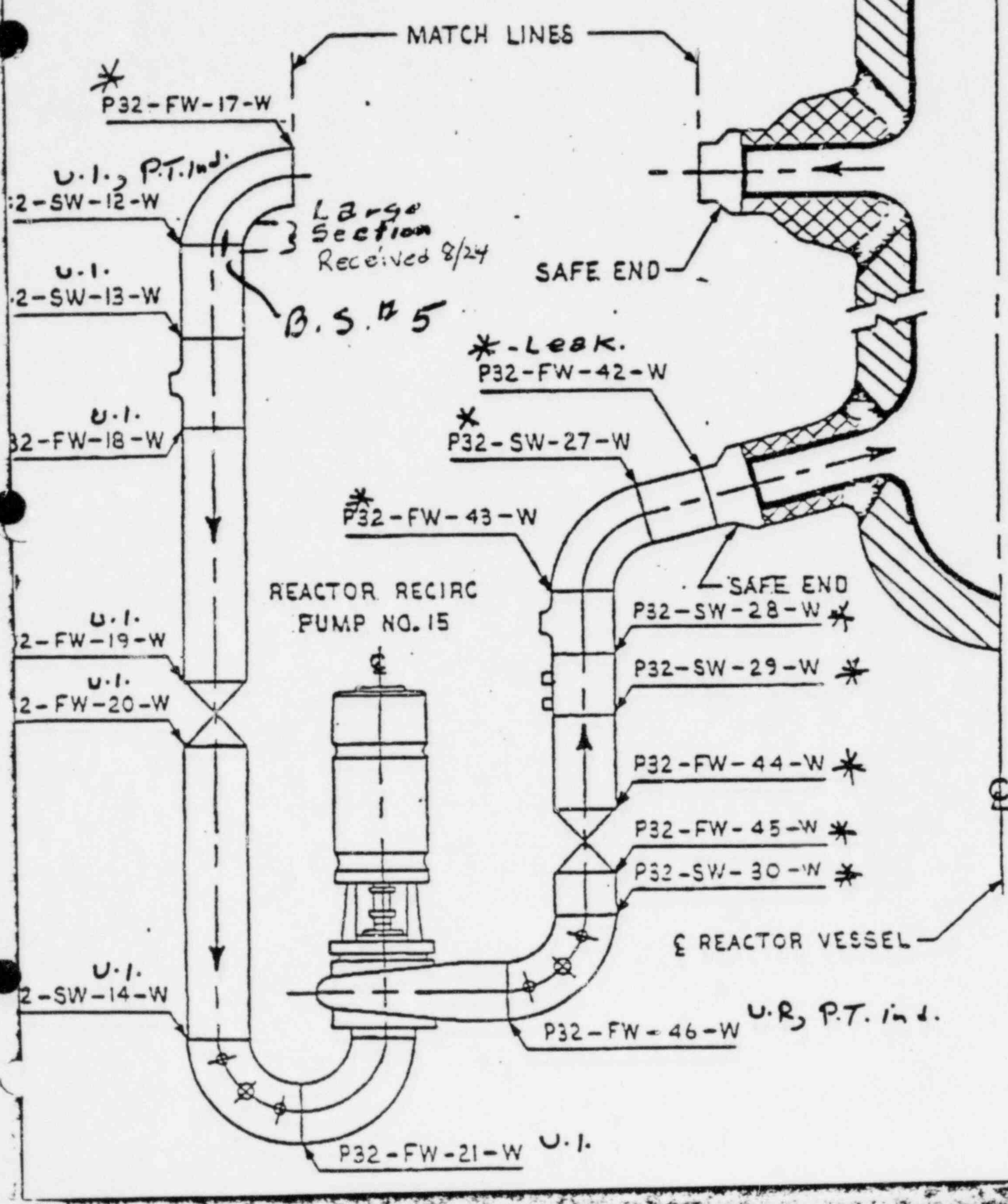
- o INSPECTION OF PUMP ELBOW FOLLOWED - IGSCC CONFIRMED

- o INSPECTION EXTENDED TO 28 INCH DIAMETER RECIRC PIPE
  - ~40% OF WELDS INSPECTED BY UT
  - ~ALL HAVE UT INDICATIONS
  - REPORTED TO NRC 8/82
  - BEING REPLACED

BOAT SAMPLE # 5 & LARGE SECTION LOCATION.

FIGURE 5

RECIRCULATION LOOP NO. 15



MEETING WITH AFFECTED BWR LICENSEES - SEPTEMBER 27, 1982

o AFFECTED PLANTS

PLANTS CURRENTLY IN OR SCHEDULED

TO BE IN A REFUELING OR EXTENDED OUTAGE THROUGH JANUARY 31, 1983

MONTICELLO
BROWNS FERRY 2
QUAD CITIES 1
DRESDEN 2
MILLSTONE 1
HATCH 1
BRUNSWICK 1
OYSTER CREEK
DUANE ARNOLD

PROPOSED STAFF ACTIONS

- o RESULTS OF BWR OWNERS MEETING - SEPTEMBER 27, 1982
  - -- ADEQUATE INSPECTIONS
  - -- UT METHODOLOGY NOT DEMONSTRATED

THEREFORE, PROPOSE FOLLOWING ACTIONS:

- o ISSUE BULLETIN
  - -- DEMONSTRATION OF UT METHODOLOGY
  - -- DOCUMENT
    - -- RESULTS OF INSPECTION AND CORRECTIVE ACTIONS
    - -- BASIS FOR SAMPLING PLAN USED
    - -- EVALUATION OF UT DEMONSTRATION
  - -- REGIONAL FOLLOWUP - NRR ASSISTANCE
  - -- ACTION SHOULD ESTABLISH
    - -- UT METHODOLOGY SUFFICIENT TO DETECT CRACKS
    - -- GENERIC SIGNIFICANCE