NUCLEAR REGULATORY COMMISSION

ORIGINAL

In the Matter of:

METROPOLITAN EDISON COMPANY	)	
	)	DOCKET NO. 50-289
(Three Mile Island Unit 1)	)	(Restart)

DATE: November 11, 1980 PAGES: 5111 - 5287

AT: Harrisburg, Pennsylvania

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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25 North Court Street,
Harrisburg, Pennsylvania

Tuesday, November 11, 1980

Evidentiary hearing in the above-entitled matter was resumed, pursuant to adjournment, at 10:08 a.m.

BEFORE:

IVAN W. SMITH, Esq., Chairman,
Atomic Safety and Licensing Board

DR. WALTER H. JORDAN, Member

DR. LINDA W. LITTLE, Member

Also present on behalf of the Board:

LAWRENCE BRENNER, Esq.,
Special Counsel to the Board

MS. DORIS MORAN,
Clerk to the Board

APPEARANCES:

On behalf of the Licensee, Metropolitan Edison Company:

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THOMAS A. BAXTER, Esq.
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Washington, D. C.

On behalf of the Commonwealth of Pennsylvania:

ROBERT ADLER, Esq.
Assistant Attorney General,
505 Executive House,
Harrisburg, Pennsylvania
WILLIAM DORNSIFE,
Nuclear Engineer

On behalf of Union of Concerned Scientists:

ELLYN WEISS, Esq.,
ROBERT D. POLLARD
Harmond & Weiss,
1725 I Street, N.W.
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On behalf of the Regulatory Staff:

JAMES TOURTELLOTTE, Esq.
JAMES M. CUTCHIN, IV, Esq.
Office of Executive Legal Director,
United States Nuclear Regulatory Commission,
Washington, D. C.

Petitioners for leave to intervene pro se:

STEVEN C. SHOLLY,
304 South Market Street,
Mechanicsville, Pennsylvania

C O N T E N T S

P R O C E E D I N G S

CHAIRMAN SMITH: Good morning.

Before we begin with the witnesses we have several preliminary matters. The Board has circulated to the persons present a draft of our memorandum and order revising emergency planning Contentions. There are some typos in it, but it is substantively correct except for one one reference.

DR. LITTLE: On page No. 8, the reference to York 13 should read, "This Contention was withdrawn. Newberry agreed with the staff's objection that it was repetitive of Newberry Contention, York 3" rather than 13. The second reference is to York 3.

CHAIRMAN SMITH: And the rest is substantively accurate.

This will be served in corrected form Wednesday.

When we return to the office on Friday, Ms. Weiss, we will issue a memorandum and order denying the motion on your Contention 17.

The memorandum and order on emergency planning Contentions is the final ruling of the Board on the Contentions involved up until now. That cleans it all up.

We had indicated that we would issue a list of the surviving Contentions and renumber them and submit those in a form which would be useful to the parties. It seems that we are going to be delayed considerably in doing that, and

it occurred to us that Mr. Zahler seems to be splendidly organized on this subject matter, and we wonder if he could be prevailed upon to take that task over for us, or somebody in your office, Mr. Baxter. He always seems to be ahead of us in what Contentions are in and out, and it seems that -- perhaps he has already done this. Who knows.

MR. BAXTER: I will commit on his behalf.

CHAIRMAN SMITH: Thank you.

Now, it is not just a question of numbering them. Some of the Contentions in their subcontentions require some judgment which the sponsoring Intervenor should be consulted on. But he will be aware of that I'm sure.

Any other preliminary business?

MR. CUTCHIN: Mr. Chairman, on Friday the staff mentioned that it had observed that the Board had some concerns in connection with the staff's not having the appropriate witness available to address concerns which had appeared to the staff to be somewhat broader than the scope of Contention 1 with respect to the Three Mile Island 2 scenario and the coolability of that core.

The staff has available or can make available two witnesses, one of which is coming from Atlanta, and one of which would come from Bethesda. The testimony of these witnesses would to a great degree be in response to what the staff perceived to be the Board's concerns. It may well

somewhat be in the nature of rebuttal to some of the things that came up in other witnesses' testimony, but for purposes of planning, does the Board have a present feeling as to when it may like to hear those witnesses?

CHAIRMAN SMITH: Well, have you given any more thought about whether you can provide a written version of their testimony in advance?

MR. CUTCHIN: That would take a longer time, Mr. Chairman, and I think without going through the transcripts and trying to ferret out the particular Board questions ands prepare a response to them, we would then be perhaps into other issue areas.

We were thinking in terms of trying to make these people available tomorrow, but we need to get the one in Atlanta particularly notified, and it would seem that many of these concerns would actually be relatable to the subject issue that is presently being considered, since indeed Three Mile 2 was a small break LOCA type scenario.

But we could do it either way. We would prefer to put them on live. Then we would just have to see what happens. If other parties felt that they needed an opportunity to prepare to examine those witnesses, we would have to face that, and the Board would have to rule on that on a case by case basis.

CHAIRMAN SMITH: We anticipate, Ms. Weiss has

already indicated she would expect to have some notice. I suppose the risk would be upon the witnesses if their testimony was such it was necessary to call them back.

MR. CUTCHIN: Well, a lot of these questions, of course, were raised by Ms. Weiss and seemed to amble somewhat upon the ambit, and normally she would be under a duty to respond. We just have to play it by ear and see if she is able to look at what is said or listen to what is said and then ask whatever questions she has, or have to have them come back. I think we would have to face that after we hear what they have to say.

CHAIRMAN SMITH: Ms. Weiss, what is your feeling?

MS. WEISS: I don't know. Mr. Cutchin first assumed a duty that we are under. I am not aware that we would be under any duty to be prepared to cross examine a witness who has not presented his direct testimony in writing, and I really think as if we should insist upon seeing it in writing. It is not a question of rebuttal. There wasn't any direct testimony to rebut. And we just think it is significant enough. We are going to see for the first time, because it doesn't appear anywhere else, in any of the staff's testimony, their analysis of how the accident sequence bears on UCS Contention 1 and 2, which is of course grounded in the accident sequence. It is not accurate to imply as if these questins were coming out of left field.

There was a great deal of discovery in what we were interested in. I think we made clear what we were interested in, and we would really insist on our right to have -- since we are going to see that for the first time, to see it in writing.

And I am not looking for a lot of notice, but I think that four or five days is necessary, particularly since we are engaged now in preparing feverishly for the next day.

MR. CUTCHIN: Mr. Chairman, I think she uses the word right. I don't think anyone has a right to see written testimony. That is within the discretion of the Board undser the rules as to whether to require written in all respects. Much of this came up as a matter of Board questions and in response to statements made by Licensee's witnesses.

Of course, the staff, as always, will make an attempt to accommodate whatever the Board decides it wants, but we would prefer to bring these witnesses on as soon as we can and bring them on live.

CHAIRMAN SMITH: Well, you are correct in a limited extent. There is no absolute right on the part of a party to have advanced notice of testimony in exactly the form that Ms. Weiss has requested it, that is, written form. However, there is an overall due process right of the

opportunity to confront the testimony, and if it is of such a nature that some advance notice of it is required, well, that would be subsumed in the due process right.

CHAIRMAN SMITH: Our problem is it is difficult to make a judgment because we don't know what to expect.

Perhaps you want them tomorrow, so even a compromise would not be much help. I was thinking of the possibility of -- I recognize the problems you have about the need to prepare written testimony in detail, and the time and complexity of going over the transcript, but perhaps an outline of what they have in mind might be a reasonable compromise. But even that could not be accomplished in time to have these witnesses here tomorrow, which returns us to where we were, and that is if they were to appear tomorrow and testify live, then we would have to entertain motions from anybody to either make them stay around or return at a later date until the transcript can be examined and a true confrontation afforded after opportunity to consider.

MR. CUTCHIN: We would be willing to run what seems to be a very high risk because I think in the long run it may be more efficient to do it that way, because even if we put in live testimony -- I mean written testimony, it may well be that the Board or the parties raise further concerns, and we keep iterating on this process, or it may

be well just to bring them on live and have all the questions come up and then play it from there.

CHAIRMAN SMITH: Right.

Let's hear from the Commonwealth. Do you have any feelings on it, Mr. Adler?

MR. ROBERT ADLER: No.

CHAIRMAN SMITH: Mr. Baxter?

MR. BAXTER: Mr. Chairman, I do believe that the additional testimony the staff is discussing would be best addressed while we are still in these first three agenda items, and I think it is sort of a circumstance where we should seek some accommodation or expect some from UCS in order to keep the record somewhat cohesive.

My view of the first two issues was whether we can rely on natural circulation or we need forced circulation. The staff's testimony addressed that. UCS is dissatisfied because they didn't explore that in the context of the first 16 hours of the TMI 2 accident, and I think the staff would be justified in resting on the testimony, but they are going to come forth and allow UCS to explore that issue, which they didn't do with their own testimony.

MS. WEISS: Let me make it absolutely clear. I am absolutely dis -- I am not dissatisfied. I am willing to stand on the record the way it is.

DR. JORDAN: Now, I am the one, I think, that

perhaps raised the problem, and my problem particularly concerned -- and I have warned the staff that I wanted their analysis particularly on the testimony of the witnesses that are here today, and particularly their replies to the Board questions, that part of UCS 8 which consists of a number of replies as to how the Licensee plans to deal with the recommendations in the various reports, and I worked to make sure that the staff has seen these replies, how they plan to meet the recommendations, some of which they will say they do not plan to meet for restart, and so that is where my chief problem will lie, and I will be wanting answers, and I would like to see them come in soon, after the Licensee's witnesses have left the stand.

MR. CUTCHIN: I was speaking primarily, sir, about concerns that the Board may have had about the details of the Three Mile 2 accident scenario and the coolability of that core as they relate both to the natural circulation concern of yesterday and to the issues of today, which are in general small break LOCA analyses.

Now, we could bring them on if the Board chooses, and then the Board could allot UCS whatever amount of time it felt was necessaary, the Board felt was necessary for them to prepare, and UCS could tell us then whether they wanted the witnesses back for cross examination.

CHAIRMAN SMITH: Our concern is that it be worked

out with all of UCS's rights preserved. It is a question of efficiency. I think there is a great deal to be said about having the staff's overview of the issue before us early, and you have recognized yourself that it seems to be a rather great risk that they will have to make two appearances.

So actually, what we will be having will be oral testimony which will be reduced to writing by virtue of the transcript, you will be expected to do a workman like job of cross examination based upon what you can do at the time, but upon a reasonable showing that the issues were too complex, or simply that you simply forgot something, I am sure that you represent it accurately, then we will have to entertain a motion to call these people back.

But give the Board a moment to consider it because I would assume that you want to know as early as possible. Somebody has to come in from Atlanta.

MR. CUTCHIN: That is the main reason we need to know early, so we can get him on notice to start making travel plans to be here tomorrow.

CHAIRMAN SMITH: Well, if we can just take a few minutes break, and then we will come back.

Is there anything further you want to say, Ms. Weiss?

MS. WEISS: No.

(A brief recess was taken.)

CHAIRMAN SMITH: Mr. Cutchin, the Board will invite you to present your witnesses tomorrow. I think -- I observed during the break that you were already advising UCS as to who these people were, and that I think would be a good procedure to follow, if you would state as much as possible what you know about the people and about what they are going to tell us.

MR. CUTCHIN: Yes, sir.

The two gentlemen that we will bring tomorrow, one is Bob Martin. He is with the Atlanta Office of Inspection and Enforcement. He was intimately involved in the investigation in the aftermath of the Three Mile 2 accident, and was a supervising editor, preparer, what have you of the document, NUREG-0600, the large, thick orange document that was put out by Inspection and Enforcement.

The other gentleman is Bill Johnston, -t-o-n. He is out of the Bethesda office. He is the Branch Chief of Core Performance Branch. His background is core thermal hydraulics, physics and the like, and he will testify on the core coolability. He was, I believe, involved in the Rogovin investigation, and in doing his work there had to acquire some knowledge of the TMI 2 accident scenario.

And these two gentlemen have a broader knowledge of TMI 2 and core coolability related to that than did the

witness we had on yesterday.

CHAIRMAN SMITH: Will they appear as a panel?

MR. CUTCHIN: Yes, sir. We would propose to offer them at the same time as a panel.

CHAIRMAN SMITH: Okay.

I assume then that much of what we will hear tomorrow has already been published in 0600.

MR. CUTCHIN: I would presume that that is true, but I cannot state that to be a fact.

We will also make an attempt to go through the transcript of Friday and try to propose some of the questions that the Board raised to sort of set the stage.

CHAIRMAN SMITH: Okay.

Anything further, Mr. Baxter?

MR. BAXTER: Do I understand the plan will be that we will begin with these witnesses first thing in the morning and interrupt these?

CHAIRMAN SMITH: I don't think that has been addressed.

MR. CUTCHIN: I didn't hear the question.

MR. BAXTER: Is the proposal to begin with these staff witnesses at 9:00 o'clock tomorrow morning and interrupt this panel?

I have no particular objection. I just want to make sure we understand what the sequence would be.

MR. CUTCHIN: They will be available, I understand, first thing in the morning at the Board's pleasure.

CHAIRMAN SMITH: Another option could be, if it works in better for travel plans, to begin at 1:00 o'clock. There might be a better chance to complete this panel.

But we don't care. Work it out.

MR. CUTCHIN: They will be here tonight is my understanding, so we can put them on, and Mr. Martin has to go, I understand, to Michigan on Thursday. So we would like, if at all possible, to get them off the stand by the end of the day tomorrow.

CHAIRMAN SMITH: I don't think it makes any difference to us.

Anything further?

MR. BAXTER: Yes, sir. I would like to make an inquiry about the Board's plans for hearing schedule during the Thanksgiving week. It is my understanding that the Board had determined to have a hearing beginning at 10:00 o'clock on Monday the 24th and a full day on Tuesday the 25th, but had not decided on Wednesday the 26th, or have you made your travel plans by now?

CHAIRMAN SMITH: Yes. Wednesday will be a typical final day. We will adjourn at about noon. Yes, we will adjourn about noon, yes.

MR. BAXTER: Thank you.

CHAIRMAN SMITH: Any other matters?

Mr. Cutchin?

MR. CUTCHIN: No more from the staff.

CHAIRMAN SMITH: Okay.

I believe now we are ready for your cross examination, Ms. Weiss?

MS. WEISS: We are going to start with questions based on the witnesses' presentation on Friday. Through a mix-up we did not get the transcript of Friday's session, but we prepared some questions anyway, and I think it will probably be all right, although when we get a chance to scrutinize the transcript --

CHAIRMAN SMITH: You have the transcript now?

MS. WEISS: We have it now.

CHAIRMAN SMITH: All right.

MS. WEISS: And Mr. Pollard is going to ask those questions.

CHAIRMAN SMITH: Did you get a copy of the transcript of Friday?

MR. ROBERT ADLER: Yes.

Whereupon,

THOMAS GARY BROUGHTON and ROBERT C. JONES, called as witnesses by counsel for Licensee, Metropolitan Edison Company, having been duly sworn by the Chairman,

resumed the stand, were further examined and testified as follows:

CROSS EXAMINATION

BY MR. POLLARD:

Q Mr. Jones, I will be asking some questions on statements in the transcript.

Do you have a copy?

A (WITNESS JONES) I do now.

Q If you could turn first, please, to page 5059.

DR. JORDAN: Can you wait just a moment?

Mr. Pollard, if you have a few extra documents there, if you could raise your microphone, it might help.

BY MR. POLLARD: (Resuming)

Q On the bottom of page 5059, the next to the last sentence where you are talking about in this analysis we have assumed loss of offsite power, reactor trip, but we have assumed an operator action in the analysis, did I misunderstand you? I thought you had said an operator error in the analysis rather than operator action.

A (WITNESS JONES) We had assumed an operator action to be performed in doing that analysis.

Q Can you tell me what the operator action was?

A (WITNESS JONES) The operator action is described on page 5060, starting on line 20 and continuing on onto the next page, that paragraph, and basically it is the -- the

operator action assumed was a manual action to open up some valves in the high pressure injection line such that one high pressure injection pump could feed to all four injection nozzles.

DR. JORDAN: This action was described in your direct testimony, was it not, in the written testimony?

WITNESS JONES: Yes, it is. It is also described on page 3.

BY MR. POLLARD: (Resuming)

Q In your discussion on Friday in general through all of the analyses and all of the tables in your direct testimony, were those analyses based upon the specific, the performance of the specific components in Three Mile Island Unit 1 such as the high pressure injection pumps, or were they performed on a generic basis?

A (WITNESS JONES) The analyses which were performed were performed on a generic basis, and they generally speaking will deliver or assumed less flow to be injected than the actual TMI system as modified will provide.

Q If I could direct your attention now to the paragraph that began on page 5059 and continuing on 5060 of the transcript, you state that subsequent investigation after that concern was raised determined indeed that we had not done a sufficient job in examining all break locations.

Can you explain to me, please, which subsequent

investigations you are referring to?

A (WITNESS JONES) Well, to go over the history a little bit, in the early '70s we performed a set of small break analyses for the TMI -- well, the generic 177 plant, which was applicable to TMI 1. After that analyses, as time evolved, model modifications were made over that time period, and other analyses were done for other B&W type reactors. An internal concern was raised at B&W which -- basically the concern was that the analyses done for the other plants was showing that the pump discharge break was the worst case while the older analyses for TMI 1 had indicated that the suction break was the worst location. This internal concern was evaluated and analysis was performed, and that analysis determined that indeed the pump discharge break was the worst location, and the whole analysis was performed on that basis, of looking at the pump discharge break, and that subsequent investigation phrases specifically relating to the initial evaluation of the concern, which was a single case study which indeed showed that there may be a problem, and then we went on to continue the analysis.

Q This determination that you had not done a sufficient job of examining all break locations, and that in fact the worst break was on the pump discharge, that determination was made after Three Mile Island Unit 1 was

first licensed, is that correct?

A (WITNESS JONES) Yes.

Q Can you describe, please, the substantial model changes that were made over the life of the plant?

A (WITNESS JONES) There were various refinements in the noding scheme utilized in the model, and probably the most significant change that was made was a new model and technique was employed for examining level, swell and bubble rise within the system.

Q And were there other substantial model changes?

A (WITNESS JONES) There may have been other changes made. I just don't remember what I would consider biggest model changes made, but I can't to my mind, no. No other pops up immediately.

Q When you say now that these were model refinements, do you use that phrase to mean the same thing as a substantial model change?

A (WITNESS JONES) Well, the term substantial model change basically was relative to its impact on the analysis, while model modifications or refinements may or may not lead to changes in the results, or significant changes in the results.

Q Did the two model refinements that you mentioned, did those lead to a substantial impact on the results?

A (WITNESS JONES) I am not sure the noting change

really had any big effect. I said the treatment of the level, swell and bubble rise in the system probably had the biggest impact.

Q Being sensitive to the Board's concern that the record be clear, could you give us a general explanation of what you mean by a node in a model?

A (WITNESS JONES) Well, a node or volume is simply that. It is basically a region which is defined for the computer code which is a model of a select portion of the system. It models the volume of that particular region. It models the relative elevations for the inlet flow and exit flow from that model. It models the initial pressures, items such as that, the physical location of a piece or a part of the primary system. For example, one node might be used to represent the hot leg of the system. That node would incorporate the height of the hot leg and the proper volume, total volume of the hot leg.

Q Would it also be within one node that you would calculate the average temperature of all the water within that node?

A (WITNESS JONES) Yes.

Q During your discussion of Table 2 in your direct testimony, you made reference to Exhibit 9, and I am sorry, perhaps you can help me remember which figure in Exhibit 9.

Perhaps I have found it faster.

I would like to direct your attention to Figure 2 in Licensee Exhibit No. 9.

Am I correct that when you performed this analysis you assumed operation of two safety valves for at least a portion of the time? Is that correct?

A (WITNESS JONES) That is correct.

Q Can you give me your opinion of what Figure 2, how Figure 2 would be changed if in fact only one safety valve opened?

A (WITNESS JONES) Basically between roughly 1800 seconds and 2300 seconds, roughly, the system pressure would increase. I have no idea as to where it would go.

Q Did you perform an analysis, perhaps in another document, where you did assume only one safety valve would open?

A (WITNESS JONES) No, we did not.

Q In analyses performed in accordance with 10 CFR 50.46 and Appendix K, isn't it required that you assume a single failure?

A (WITNESS JONES) Yes, but this analysis was not dsone for 50.46 compliance. Neither was -- let me just clear that up. I did not state it Friday, but all the analyses discussed from Table 2 on to the last table of my testimony was not performed for 50.46 compliance. Only the first table is the analysis for 50.46 compliance.

Q If you were doing the analysis depicted in Exhibit 9 for the purpose of demonstrating compliance with Appendix K, would you then assume or would you be required to assume that one safety valve did not operate?

MR. BAXTER: Excuse me. Just a point of clarification. Are we still talking about Table 2 where there is no small break LOCA?

MR. POLLARD: Excuse me. I am referring to Figure 2 in Licensee Exhibit No. 9 which was referenced in Mr. Jones' testimony, dealing with Table 2 in his direct testimony on UCS Contention 8 and ECNP Contention 1E.

MR. BAXTER: Thank you.

WITNESS JONES: I am not sure that I would ever have to do this analysis in the first place to show compliance with 50.46 in that I believe -- it is my understanding that no single failure will wipe out the emergency feedwater system, but even if I take that assumption and do it, if I take the failure of the safety valve, then I would have to high pressure injection systems, both pumps operating, and that would result in a significant change in the required capacity, relief capacity for the system.

BY MR. POLLARD: (Resuming)

Q One more question on Figure 2 of Exhibit 9.

Without changing any other of the sequence of

events or your assumptions that you have given in your testimony, as I understand what you just testified this morning, that if one safety valve failed to open, the pressures shown in Figure 2 would go higher, but you don't know by how much.

A (WITNESS JONES) It would go higher, but only between that specific timeframe, the roughly 1800 to 2300 second period, and I have no -- I have not done any calculations, nor what its final pressure state would be. But once that 2300 second timeframe passed, it would come down to basically the same pressure.

Q May I ask you another question?

Assuming again all of your original assumptions and one safety valve failing to open, but with the additional change of using 1.2 times the ANS decay heat value, would the pressure go even higher between those times?

A (WITNESS JONES) Can I have the question read back? I think I missed something and I want to answer the question properly.

(The reporter read the pending question.)

WITNESS JONES: Yes. Between that specific time pressure would go higher than the case of 1.2 ANS.

DR. JORDAN: May I ask this for further understanding?

The pressure shown on Exhibit 9 that we have been

discussing, that pressure is the pressure which the safety valves have released, is that right?

WITNESS JONES: Yes.

DR. JORDAN: And you are saying now if there was only one safety valve released, the pressure would build up and increase the flow out of that safety valve.

WITNESS JONES: Yes, the system pressure would have to increase to discharge roughly the same amount of volume, but there would be many other changes on the system, because as the pressure goes up, the volume relief necessary goes down. So it is not just a linear function. If you do a boiling calculation, you boil a little more but it takes up less volume at higher pressures, so that it is not a one to one relationship. It does not double or anything like that. It has other feedbacks that are difficult.

DR. JORDAN: I see, but you do say you have no feeling for how high the pressure might go, whether it might exceed the limits, the stress limits of the pressure vessel.

WITNESS JONES: That is correct. I have not done the calculation and I would rather not guess.

DR. JORDAN: All right. I understand.

BY MR. POLLARD: (Resuming)

Q If we can move on now to your testimony on Table 3, you referenced there Licensee Exhibit 5.

I would like to direct your attention first to

Figure 6.2.22 of Licensee Exhibit 5.

CHAIRMAN SMITH: Read the title of the figure.

MR. POLLARD: The title of the figure is Figure 6.2.22, Break Quality versus Time, 0.02 Square Foot Break of Pump Discharge, No Auxiliary Feedwater.

CHAIRMAN SMITH: I think everyone is ready.

BY MR. POLLARD: (Resuming)

Q Can you explain the label on the left hand side of the graph which states "Pipe Quality?"

A (WITNESS JONES) That is the label given to this type of figure by the computer code, and it is basically the inlet quality to the flow path which they call pipe quality. The only exception to that would be where you have heat addition or heat removal within a path, and there you would see, dependent on the assumption used in the code, you would see possibly the effect of the addition of the heat.

Q On Figure 6.2.22, on slightly over 1500 seconds, I assume, the flow quality oscillates there.

Can you explain physically what is happening in the plant that caused that quality to fluctuate?

A (WITNESS JONES) The fluctuation is partly a function of the assumptions we use in the code and the way does its calculations, and what it is is we have placed the break at the exact bottom of the fluid volume that we are representing. However, in order to properly, or in order to

account for the pipe quality of what is leaving the system, we have placed a small, roughly one inch height on the break. In other words, it spans into the fluid volume approximately one inch.

Now, if the injected fluid is enough to cover that one inch pipe, you will see pure water. If all of the water is discharged instantaneously out of the pipe, you would see pure steam, and if it is in between that one inch path, you would see some fractional quality. And what is happening here is we are getting down to very low qualities in this location, on the order of .09, .07 -- I am trying to read the graph. It is on that order. And all that is happening is as you continue to inject, you are at certain times in the calculation covering the pipe path, and because you cover the pipe path, you get larger flows, and that subsequently results in it uncovering a little bit, and you get a little overshoot on the quality part.

DR. JORDAN: Which way does quality go?

WITNESS JONES: Liquid is zero and pure steam is 1.

BY MR. POLLARD: (Resuming)

Q So that am I correct, then, from your explanation that this oscillation of quality at this point is an effect of the computer model and may not necessarily occur in the actual plant?

A (WITNESS JONES) I believe you will probably see,

if you truly had a breakdown at the bottom of a pipe like this, you would see a sloshing over the break during the transient, when the system has reached low inventories, to where it had drained down to near the bottom of the reactor, the inlet nozzles to the reactor vessel, and that you would see water coming by from the HPI. For example, you might see some mixing, and you would probably see some of these oscillations.

Think what you get in both this and what you get in the real world is more or less an average value, even in the early timeframes, and that these oscillations are not really that unreasonable. I would not be surprised to see them.

Q But I am correct that this oscillation, as I understood your oscillation, results from the way you actually modeled the break.

A (WITNESS JONES) Yes.

Q If I compare Figure 6.2.22 with Figure 6.2.23, am I correct that the pressure oscillations observed -- excuse me, that the flow oscillations observed in Figure 6.2.23 at slightly over 1500 seconds are the result of the change in quality depicted in Figure 6.2.22?

A (WITNESS JONES) That is correct.

Q If you refer to Figure 6.2.23, we see a rather large perturbation in break flow around 500 seconds.

Can you explain why we don't see any similar perturbation in the quality at 500 seconds?

A (WITNESS JONES) Well, you do. If you look at the pipe quality chart, you see that at around 500 seconds, the system goes from zero quality to roughly 34 percent quality. This sudden drop is a result of a change in the break flow as a result of the change in the discharge models from a subcooled discharge model, Bernoulli, to a saturated fluid discharge model, which is the Moody model.

BY MS. WEISS:

Q What was the first one?

A (WITNESS JONES) Bernoulli.

BY MR. POLLARD: (Resuming)

Q Would you expect to see such a change in the actual plant, or is this once again a result of the analysis where you are changing from two different types of models.

A (WITNESS JONES) I don't believe the magnitude change you see there will occur because the Bernoulli model itself is a highly conservative discharge model and will tend to overestimate the leak flow. So the step change would be smaller. But I believe that you would see a step change basically because the experimental data done on subcooled discharge and saturated flow discharge indicates that you get very large flows for subcooled discharge, and after you receive roughly a 2 percent pipe quality, you will

have a fairly substantial decrease, and to go from slightly subcool to 2 percent quality is not a very large change and could occur quite rapidly.

Q I would like to move on now to your testimony dealing with Table 4, in which you referenced Licensee Exhibit 5, Figure 6.2.62.

DR. JORDAN: Is that 62?

MR. POLLARD: Yes, 62 is the exhibit in Exhibit 5, and I would also like to direct your attention to transcript page 5088.

I'm sorry, the discussion actually begins on page 5077. Dr. Jordan asked you a question with respect to this analysis, whether it uses the set points of the PORV and the reactor trip prior to or after the accident, and your answer was no, this analysis was done assuming the old set points.

DR. LITTLE: That's 5087.

MR. POLLARD: I'm now on 5088. I just read Witness Jones' answer on lines 3 and 4.

DR. LITTLE: You referred to 5077.

MR. POLLARD: I'm sorry. I was always on page 5087 and 5088.

BY MR. POLLARD: (Resuming)

Q Basically my question deals with your answer which begins at line 10 on page 5088.

Can you explain why the analysis would not be much

different if you had used the PORV and reactor trip points that will be in place prior to the restart of Three Mile Island Unit 1?

A (WITNESS JONES) Basically the reason for that is this accident results in, as analyzed, would result in a PORV actuation in about a four to six second timeframe at 2350 psi, or it might have been -- I don't remember the exact set points. It might have been 2300, but in that 2300 plus psi, and the reactor trip at a higher pressure, which would occur at about eight to ten seconds for this event.

With the inverted set points, you would have your reactor trip occurring in the four to five second timeframe, and the possibility of hitting the PORV would occur in roughly only another two or three seconds for this case. So you are talking about the accident occurring, or changes in the analysis on the order of about a three or four second timeframe. If anything, a new analysis would result in better consequences because you would have an earlier reactor SCRAM and less heat in the system, but the subsequent follow-on actions of the system would be basically the same.

Q Perhaps you have already answered my next question, but could you refer to Figure 6.2.64. As I understand the answer which you gave before I asked the question that if you had the earlier reactor trip set point,

this peak temperature shown in Figure 6.2.64 might actually be somewhat lower.

A (WITNESS JONES) That is correct.

Q Can we refer now to transcript page 5090? Actually, the paragraph of your testimony begins on page 5089 and continues on 5090.

In this particular analysis, it was originally assumed that the reactor coolant pumps keep running, is that correct?

A (WITNESS JONES) That is correct.

Q And it is your testimony that with the reactor coolant pumps running, this gives you forced circulation, which keeps a very good heat transfer to the steam generators. Is that correct?

A (WITNESS JONES) That is correct.

Q So from the standpoint of the effectiveness of ECCS, or the ability to cool the core, let me put it that way, it is better to have the reactor coolant pumps running for this particular analysis.

Is that correct?

A (WITNESS JONES) No, that is not correct.

Q Can you --

A (WITNESS JONES) The reason why basically is this transient does not really -- whether you have power to the pumps or not does not really result in a fairly large loss

of inventory if the high pressure injection pumps are left running. So in both cases the core would remain continually covered with water, and the cladding temperature would remain within a few degrees of the saturated fluid temperature. Running the pumps might give you an incremental few degrees on the temperature of the cladding, but in all cases it would be below 700 degrees, and I don't consider that significant, and in fact we have analyzed the case of a PORV failure without power to the pumps, and it shows that the system would remain quite cool.

DR. JORDAN: But now, refresh my memory. There were now some cases in which the core would uncover unless you took off the power to the pumps. Was that right?

WITNESS JONES: That's correct, but those were break sizes in a range of between .025 to .2 square feet, and the PORV is a .007 square foot break. So it is below that window.

DR. JORDAN: I see.

BY MR. POLLARD: (Resuming)

Q If we can move on to your testimony on Table 5 in your direct testimony, and on Friday you referred to Licensee Exhibits 6 and 7, Figure 2 in both exhibits.

CHAIRMAN SMITH: Mr. Pollard, may I have the Exhibit number table again?

MR. POLLARD: Licensee Exhibits 6 and 7, Figure 2

in both exhibits.

BY MR. POLLARD: (Resuming)

Q We will now be referring to your testimony at page 5094 of the transcript.

Once again, your answer begins at page 5093 and continues on 5094.

Am I correct that your testimony dealing with Exhibit 6, that Figure 2 shows that you are not capable of cooling the core with one high pressure injection pump?

A (WITNESS JONES) Well, the figure shows that given the loss of allk feedwater to the system, and using the assumption of 1.2 ANS, which is not a very realistic value with the decay heat curve, then on a generic basis for all plants, we could not assure that we could cool the core for this specific circumstance.

I am not so sure that that would necessarily apply in the case of TMI 1 in that this analysis was done at 2772 megawatts. TMI 2 has about an 8 percent lower power level. Its HPI system will produce roughly 10 percent more flow than what was assumed in the analysis. So it would tend to counterbalance, even though 1.2 ANS assumption.

DR. JORDAN: I think you misspoke and said TMI 2.

WITNESS JONES: TMI 1, excuse me.

BY MR. POLLARD: (Resuming)

Q As I understand your answer to my question

earlier, you testified that these analyses were done on a generic basis. Is that correct?

A (WITNESS JONES) That is correct.

Q And you have not done this analysis specifically for Three Mile Island Unit 1.

A (WITNESS JONES) That is correct.

Q Now, as I understand your testimony on Friday, when you then moved to Exhibit 7, Figure 2, that this demonstrated that the core could be cooled with one high pressure injection pump, assuming that you used 1.0 times the ANS standard value for decay heat.

My question is on your generic analysis depicted in Exhibit 7, if you had used 1.2 times the ANS standard value for decay heat, would you then get essentially the results similar to what you did get in Exhibit 6?

A (WITNESS JONES) You would get exactly the same results for this generic evaluation because that is the only difference between the two cases.

Q Perhaps I should have asked the question that way. Yes, thank you.

Continuing from transcript page 5094, you then say you are going to move on now to Table 6 of your testimony, and then on page 5095, at line 16, you state, "And we have assumed that the emergency feedwater is delivered to the steam generators."

My question is at that point how many pumps did you assume were available, emergency feedwater pumps?

A (WITNESS JONES) One. And in fact, the way the code works, it is not even a full one pump.

Q I'm sorry, I didn't understand.

A (WITNESS JONES) The way the code works, it is not even a full one pump. The flow rate is modulated to maintain level in the code, and we have input a value which would be more or less representative of one pump, but in fact we are not even using the full capacity of one pump.

Q Not even the full capacity of one motor driven pump as opposed to one turbine driven pump?

A (WITNESS JONES) On the average, that is correct, yes.

Q I'm sorry, I didn't understand the phrase "on the average."

A (WITNESS JONES) There may be a period of time in the analysis where we may be using a larger capacity than, say, one motor driven pump could produce for a short time period, which is the period where we just start to re-establish the boiler condenser mode, possibly. For that, what I am saying, on an average basis, over a timeframe in there, we would be using roughly, say, 300 GPM or 100 GPM at a very specific instant in time. We may be using more over, say, a 5 or 10 second time period.

Q In the analyses that you are discussing in this answer here on page 5095, what assumptions were made in the computer analysis about what level the steam generator was filled to, and how fast did the steam generator water level get to that pump?

A (WITNESS JONES) Well, the analysis is based on a 50 percent level. I have to look up the time.

Q Fifty percent of which?

A (WITNESS JONES) On the operating range.

(Pause)

WITNESS JONES: I don't have the information with me to be able to tell you that. I can say, though, in general what happens is in fact the system basically comes down and settles out at 50 percent of the operate range rather than fills to 50 percent of the operate range.

BY MR. POLLARD: (Resuming)

Q So that normal level in the steam generators is normally about 50 percent in the operating range, regardless of power level of the plant?

A (WITNESS JONES) At full power, yes.

Q At full power. What about at 25 percent power, what is the steam generator level?

A (WITNESS JONES) I don't know. I expect it is less than 50 percent of the operate range.

Q If we go now to your testimony on Table 7 from

your direct testimony, where you referenced -- excuse me. We may not need the exhibits. We can just refer to page 5101 of the transcript.

A (WITNESS JONES) Excuse me. Which page?

Q 5101.

What we are discussing here is the small break loss of coolant accident with a delayed reactor coolant pump trip, is that correct?

A (WITNESS JONES) That is correct.

Q At the bottom of page 5101, the last paragraph, you explained that at this point you have to refill the reactor coolant system, and specifically the reactor vessel in the core, and try to recover the core. And then you testified "but these are high pressure transients, and you do not have a pump which is capable of refilling the system rapidly as you do at low pressures for large breaks."

My question is, is that because of the limited capacity of the high pressure injection pumps?

A (WITNESS JONES) Yes, that is correct.

Q I am correct here that in this analysis you had assumed that both high pressure injection pumps -- excuse me, the two high pressure injection pumps were running, is that correct?

A (WITNESS JONES) The majority of the analyses performed did utilize two high pressure injection pumps for

this scenario, where the pumps ran, the system went to a high void fraction, and then the pumps trip out, the reactor coolant pumps.

Q Can you please refer to Licensee Exhibit 10, Figure 2.5.

I think it is Figure 2.5, but looking at the page it looks like it might be 2-5.

A (WITNESS JONES) Is that on page 28, I believe?

Q 28, page 28.

Can you give me your views on how that figure would change if you had only one high pressure injection pump available?

DR. JORDAN: Before you do, review for me the situation that we are referring to at this figure.

WITNESS JONES: Okay. This is an analysis of the seven breaks in the bottom of the collate pump discharge piping, with the reactor coolant pumps remaining operative, and two HPI pumps on.

Now, as far as how the system pressures were changed for these cases, you would see probably very little impact for the .2 square foot break. The .025 square foot break would tend to just basically just float now out around 1100 psi for a longer period of time, and the other cases would tend to be somewhat higher in pressure. I am not sure how much, but they would tend to be slightly higher in

system pressure.

And to give you an idea as to what it is worth, Figure 2-8 on page 31 shows a comparison of the system pressure trace for the .05 square foot break with one or two HPIs, and there may be a hundred psi differential between the two at 3000 seconds.

BY MR. POLLARD: (Resuming)

Q Then do I understand your testimony correctly that the reason you are tripping the reactor coolant pumps is because you do not have an emergency core cooling system pump that is capable of rapidly refilling the system to cover the core, and that if you did have such a pump, it would not be necessary to trip the reactor coolant pumps.

A (WITNESS JONES) I am not so sure I would want to characterize it in that fashion. Rather, for this specific scenario, the high pressure injection systems have not been designed to handle, but even if you have larger high pressure injection pumps and could somehow tolerate this high system void fraction evolution, there have been various concerns raised about the integrity of the reactor coolant pump to operate in high system voids, and that it still may not be prudent to continue operation of the reactor coolant pumps through a transient as it may induce secondary LOCAs, additional LOCAs or vibrations or whatever, where the integrity of the pump could lead to other problems.

Q Am I correct that the analysis you performed showed that for a spectrum of small breaks between .025 square feet and .2 square feet, that it was necessary to trip the reactor coolant pumps in order to assure that you had adequate core cooling?

A (WITNESS JONES) It was necessary to trip the reactor coolant pumps in order that if you want to postulate the scenario of the loss of the pumps at any time, that you would not have adequate core cooling, that we could not demonstrate adequate core cooling.

If, however, if we could keep the reactor coolant pumps running, we would have maintained adequate core cooling. It was only this delayed pump trip scenario that caused the problem, not whether the pump is on or the pump is off initially being the problem. It is the delayed scenario which caused the specific problem.

Q Well, during an accident such as you are analyzing here, where do the reactor coolant pumps receive their electrical power from?

Could I ask the question first of this witness?

A (WITNESS JONES) My understanding is they get it from offsite source. I thought that perhaps Mr. Broughton could talk specifically for TMI 1.

Q Have you done an analysis of the offsite power system for Three Mile Island Unit 1?

A (WITNESS JONES) I have not personally.

Q Then you don't have any opinion as to whether or not it is likely or unlikely that offsite power could be lost at any time during an accident.

A (WITNESS JONES) As I stated, I have not done such an analysis and I do not know.

DR. JORDAN: Let's make the record clear. Would the other witness respond as to where the power comes from to operate the reactor coolant pumps?

WITNESS BROUGHTON: The power for reactor coolant pumps at TMI 1 does come from offsite power sources.

DR. JORDAN: Thank you.

BY MR. POLLARD: (Resuming)

Q Are you aware of any recommendations that the Advisory Committee on Reactor Safeguards has made over the years with respect to the capacity of emergency core cooling system pumps?

DR. JORDAN: And you are referring to the high pressure injection pumps now?

MR. POLLARD: I asked the question generally, first with respect to any emergency core cooling pumps.

DR. JORDAN: Fine.

WITNESS JONES: I just can't remember off the top of my head any recommendations specifically dealing with increasing or decreasing the size of the emergency core

cooling system pumps. Right now I just cannot recall it.

BY MR. POLLARD: (Resuming)

Q You don't recall anything, whether high pressure pumps or low pressure pumps. You just don't recall any recommendations.

A (WITNESS JONES) No, I just really cannot remember any at this time. There may have been some, but I just don't remember.

Q Thank you.

We will turn now to your testimony on Table 8 of your direct testimony, and on Friday you referenced Licensee Exhibit 13, Figure 1.

If it is possible, I would like you to also have in front of you Licensee Exhibit 5, Figure 6.2-60, and also Licensee Exhibit 8, Figure 1.

CHAIRMAN SMITH: Mr. Pollard, I think we should change our procedure.

Well, that's fine. Every time you relax, you come back with a new chart in a new exhibit. I think when you go on to a course of examination, I think we should take a moment and you just read out the papers we will need, and then we will gather them. What you did this time, in due course, the transcript will not reflect -- well, never mind.

We will stop at the beginning of each section and we will identify at once all the papers we need.

What was the chart in Exhibit 6?

MR. POLLARD: No Exhibit 6. We are looking at Licensee Exhibit 13, Figure 1.

CHAIRMAN SMITH: Got that one.

MR. POLLARD: Licensee Exhibit 5, Figure 6.2-60.

CHAIRMAN SMITH: Got that one.

MR. POLLARD: And Licensee Exhibit 8, Figure 1.

BY MR. POLLARD: (Resuming)

Q Perhaps, Mr. Jones, if you could refresh our memory as to which analysis each of these figures are depicting first.

A (WITNESS JONES) Okay. The one from Exhibit 13, Figure 1 is a small break LOCA, specifically a .01 square foot cold leg break in the pump discharge piping with no feedwater, both main or auxiliary, with two HPIs actuated at 20 minutes, and also assuming that the PORV is either open or sticks open at 20 minutes.

Figure 6.2-60 of Exhibit No. 5 relates to Table No. 3. It is a .01 square foot break in the pump discharge piping without any feedwater, and two high pressure injection pumps actuated, manually actuated at 1200 seconds by the operator.

Figure 1 of Exhibit No. 8 also is utilized in the development of Table 3 of my testimony. It is the same analyses as in Figure 6.2-60 except instead of the operator

actuating two high pressure injection pumps, he actuates the emergency feedwater system, which then depressurizes the primary system and leads to an automatic actuation of the HPI pumps.

Q Recalling Dr. Little's question to you about the number of significant figures shown on the pressure scale, can you offer an explanation as to why the peak pressures in these three figures are different?

A (WITNESS JONES) The peak pressures I do not believe are different between Figure 1 of Exhibit No. 13 or Figure 6.2-60 of Exhibit 5. The Figure No. 1 of Exhibit No. 8 is slightly higher, and the cause of that difference is basically the auxiliary feedwater injection was re-established at 1250 seconds while the other analysis assumed the operator action at 1200 seconds. So there is a slightly longer period of time of repressurization before any action was taken.

MS. WEISS: Mr. Pollard has finished with the questioning on the testimony from Friday. I don't know whether there is any more. Would you like us to go right into the questioning on the prefiled written direct?

DR. JORDAN: The what?

MS. WEISS: We are finished with the questioning on Friday, so we can either go into the questioning on the prefiled written direct testimony of these witnesses, or if

anybody has any other questions on what happened Friday.

MR. BAXTER: We are really talking, in my view, Mr. Chairman, about both. We have been relating these to the direct testimony.

DR. JORDAN: I, of course, asked the questions on Friday that I wanted to kow where the testimony went, so I will have no further questions on Friday's testimony.

Let's find out if either the staff or the state has any questions. I think if they do have questions on Friday's testimony, now is probably the best time.

MR. CUTCHIN: I have none on Friday's, Mr. Chairmn.

MR. ROBERT ADLER: We have none on Friday's.

DR. JORDAN: All right. Then I guess that does complete Friday's testimony and we can then move into your cross examination on the written testimony.

MS. WEISS: If we are going to go straight ahead, I would like a short break.

CHAIRMAN SMITH: Let's take five minutes.

Before we leave the hearing, why don't we just take the noon break now, and we will reconvene at 1:00 o'clock.

(Whereupon, at 11:47 o'clock a.m., the hearing in the above-entitled matter recessed, to reconvene at 1:00 o'clock p.m. the same day.)

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AFTERNOON SESSION

(1:00 p.m.)

MS. WEISS: There are three things I wanted to bring up before we got into the cross examination for this afternoon. The first was cross examination plans, Item 4 on the schedule. You had originally directed that those be submitted by tomorrow. It now appears as if we won't get to those this week.

I have written the cross examination plans. My office is closed today because it is a federal holiday, my secretaries have those days off. They are now typing, or they will be, as of tomorrow, typing part of it. We could have the cross examination plan on the Licensee's witnesses in then by Friday, or by Thursday. Those are sent Federal Express. But have only written drafts of the cross examination plan for the staff witnesses on Item 4.

CHAIRMAN SMITH: Ms. Weiss, I don't think that we have to put you to the effort to get the first phase on Licensee's panel by Federal Express. Why don't you suggest a date that you bring the entire package.

MS. WEISS: We can bring it over the first day next week, Tuesday of next week.

CHAIRMAN SMITH: Are there any objections to that?

MR. BAXTER: I have no objection. I didn't consider it impossible that we would get to that issue by

the end of the week. I was hoping we would, but I have no objection.

CHAIRMAN SMITH: Do you really think we will get to that?

Well, if that is the case, if it looks like that is going to happen, why don't you just submit your dsraft, sincew it is not going to go in the record anyway, if that will be acceptable.

MS. WEISS: Well, they have in my office the written draft because they are typing for the licensees, but I have my written draft for the staff which I could give you, but that would mean they woulds have to rederal Express, but that's fine.

CHAIRMAN SMITH: How about the funny phone? We could send it up on the --

MS. WEISS: The telex? I have no access to telex facilities.

CHAIRMAN SMITH: Federal Express is not reliable. It is just lucky that we got the last batch because the hotel said they would not accept it, but somebody at the desk did in fact accept it.

Well, why don't we worry about it when the time comes, and then in the meantime you are going to proceed producing it and see what happens.

MS. WEISS: The second issue was the transcript

for Friday. Does the Board have one copy of the Intervenors' transcript?

CHAIRMAN SMITH: We received, I believe, four this morning. We provided one for somebody. We provided one --

MR. ROBERT ADLER: We have one copy.

MS. WEISS: Okay, fine.

And the third subject was for the witnesses tomorrow, the staff witnesses who will be on the first thing in the morning, we would like to request that if they are not familiar with that NSAC sequence, which is USC Exhibit 1, that we used to question the Licensee witnesses, that they bring with them whatever accident sequences that they are familiar with so they can be questioned on whatever they prefer to be questioned on.

MR. CUTCHIN: The staff will bring with them what they prefer to be questioned on.

CHAIRMAN SMITH: That seems very accommodating.

MS. WEISS: Can you tell me what that is?

MR. CUTCHIN: My guess it is in the NUREG-0600 and in the Bogovin report.

CHAIRMAN SMITH: Do you have that here with you, Ms. Weiss?

MS. WEISS: No.

CHAIRMAN SMITH: Well, during the break I will loan you my copy.

Whereupon,

ROBERT C. JONES and THOMAS GARY BROUGHTON,

called as witnesses by counsel for Licensee, Metropolitan Edison Company, having been duly sworn by the Chairman, were further examined and testified as follows:

CROSS EXAMINATION -- Resumed

BY MS. WEISS:

Q Mr. Jones, on page 3 of your testimony you are discussing the analyses which were performed prior to the TMI 2 accident in order to show compliance with 10 CFR Part 50, Appendix K for Unit 1, is that correct?

A (WITNESS JONES) Yes, that is correct.

Q Can you tell me when that model was approved?

A (WITNESS JONES) Roughly September of 1978.

Q That is precise enough.

Have any changes been made in the model since September of '78?

A (WITNESS JONES) For the purposes of demonstrating compliance to 50.46, no.

Q I take it from the way you phrased your answer that there were some changes made.

Were those made in connection with a submittal that you made to the NRC after the accident?

A (WITNESS JONES) There were certain noding changes used in some of the analyses submitted after the accident.

Q Noding changes. And when were these analyses submitted?

A (WITNESS JONES) The largest package was the May 7, 1979 report and then the other. I am not sure which model was used in the other supplements, in the other exhibits that we submitted.

Q I just want to make sure the record shows which supplements you are referring to.

Are those supplements represented by any of the Licensee exhibits?

A (WITNESS JONES) Well, as I said, I am not sure. They are represented by the Licensee's exhiits, but I am not sure whether the analyses used to revise noding schemes, but those would have been basically all the exhibits from 6 to 13, with the exception of 12, which are the guidelines.

Q Okay. But the only one that you are sure contains the changes in noding would be May 7, '79, is that correct?

A (WITNESS JONES) That is the only one I am absolutely sure of at this time, and it was not all of those analyses. It was some of the analyses that are in that report utilized a revised noding scheme.

Q Can you tell me which utilized the revised scheme?

A (WITNESS JONES) These are basically the analyses that are referenced as Table 6 in my testimony.

Q Okay. Those are the analyses of a very small LOCA

with loss of main feedwater, is that correct?

A (WITNESS JONES) That is correct.

Q And was the change in noding for the purpose of modeling greater detail?

If there is something wrong in the way I have asked that question, maybe I just should have asked you what was the effect of the change in noding?

A (WITNESS JONES) Well, I am having an equally difficult time with exactly that question.

The reason that we made the model change, or specifically what the model change was was the addition of a node to represent the 180 degree bend in the top of the hot leg, and the node itself was more than just the bend. It included down into the upper plenum of the steam generator. The purpose of that node was to predict the interruption in natural circulation that would occur once voids had, a significant amount of voids had developeds during the accident, and that was the main purpose of the additional node.

Q Is it true, then, that before the accident the hot leg, the entire hot leg had been presented by one node?

A (WITNESS JONES) For each loop, yes, that is correct.

Q And your testimony is that you changed that so that the hot leg was divided into two nodes, is that correct?

A (WITNESS JONES) Well, as I said, the additional node was more than just the hot leg. It included the upper plenum of the steam generator.

It is basically the down side from the elbow in the hot leg, down to the upper tube sheet in the steam generator.

Q And the effect of this change was to predict the creation of system voids which had not been predicted by the model in its previous forms, is that correct?

A (WITNESS JONES) No, that is not correct. The purpose of the node was to -- basically the purpose of the node was to isolate the heat removal from the steam generator from that region in the system which it would not be able to affect, and for these very small LOCAs which would utilize the steam generator for a substantial amount of heat removal, it was necessary to isolate that region in the system in a separate node in order to show the potential for an interruption in natural circulation, while at the same time not having the boiler condenser mode established.

Q Then is what you are saying with the old model, you did not predict an interruption of natural circulation but that the change in noding resulted in predicting an interruption in natural circulation?

A (WITNESS JONES) Well, we didn't specifically run the case with the old noding. Just based on the way the

codes operate, before we ever even started the analysis for these smaller breaks, we added the node. It was our judgment that that node would be necessary to properly predict the transient response.

Q It is true, isn't it, that none of your old model results for any cases predicted an interruption in natural circulation.

A (WITNESS JONES) All the old results predicted interruptions to natural circulation in that we predicted voiding would occur in the system.

Q Let me tell you what is giving me a problem in understanding your answer.

Do you by any chance have NUREG-0565 with you?

A (WITNESS JONES) Yes, I do.

Q Could you turn, please, to page 4-4.

(Pause)

CHAIRMAN SMITH: Okay, Ms. Weiss, we can follow.

BY MS. WEISS: (Resuming)

Q Let me direct you to the bottom paragraph on page 4-4. Now, that paragraph is discussing reference 62, which is that May 7, '79 B&W analysis that we have been discussing.

Is that correct?

A (WITNESS JONES) That is correct.

Q And the last full sentence on that page says, "It was also found that for the small break transients involving

that circulation, that the development of a steam bubble in the upper 180 degree bend of the hot leg leading into the steam generator (candy cane) could not be adequately described because of the core's noding detail for the pipe and steam generator primary side in the existing nodal representation."

And I guess I would like to ask if that describes thew change and the purpose for the change that you have been discussing.

A (WITNESS JONES) For the specific very small breaks that are talked about in Table 6 of my testimony, the .05 and the .01 square foot break, the sentence is generally speaking accurate. You could not get the detail and predict that interruption as well with the coarse noding type. As a matter of fact, I don't believe you will predict it at all.

Q And other than the change in noding, were there any other changes in the model after September of '78?

A (WITNESS JONES) Not to my knowledge.

Q Has the code been applied to different accident scenarios than it was before the Three Mile Island 2 accident?

A (WITNESS JONES) Yes. It has been applied to different scenarios than would normally be considered for 50.46 compliance.

Q Could you describe to me what those are?

A (WITNESS JONES) These are basically all of the analyses that are described in Tables 2 through 8 of my testimony, and the biggest -- well, the two major items are that we have looked at cases with the assumption that for some reason all the main and auxiliary feedwater is lost, and we have looked at smaller sized breaks than would normally be considered, and we have looked at the effect of delayed reactor coolant pump trips.

Q Before we go on -- I'm sorry, I would like to refer you back just for a minute again to page 4-4 of NUREG-0565 and read the sentence above the one I read before, "In performing the TMI 2 transient comparison, B&W found that a four node pressurizer model was needed in place of a single node model to properly compute the effects on PORV flow of incoming cooler water during the initial primary coolant swelling phase of a loss of feedwater accident."

Is that another example of a noding change which you made after the accident?

A (WITNESS JONES) That was a noding change which was incorporated into our analysis of the TMI 2 event specifically. It is not a noding change that we have used in, as far as I can remember, any of the other analyses. It was just used -- its basic effect is only for the first roughly six minutes of the TMI accident, until the time that

the primary system saturated. Up until that point in time you need essentially a nonequilibrium pressurizer to look at that repressurization phase that occurs in the initial system response.

Q Was that noding change, a four node pressurizer model, incorporated into any or the analyses which you have presented in your testimony?

A (WITNESS JONES) Not that I can remember.

Q Back to page 3 of your testimony again, where you are discussing the analyses performed prior to the TMI 2 accident, you testified that they assumed the use of only safety grade equipment for accident mitigation and assumed no mitigating operator actions within ten minutes of the initiating event, except as follows, and then you then give two exceptions. The first exception is that emergency feedwater was assumed to be available.

Now, given that emergency feedwater is a non-safety grade system, how did you justify assuming its availability in your Appendix K analysis?

A (WITNESS JONES) Basically we just used emergency feedwater in the analyses. I don't remember any, off the top of my head, remember any specific justification that was especially done for that.

But as far as I know, Appendix K does not make a statement about the use of non-safety grade equipment.

Q You are not aware that the -- that one is required to assume the failure of all non-safety grade equipment in analyzing the consequences of accidents?

A (WITNESS JONES) On plants of this vintage, of the earlier plants that came into operation in the early '70s, as far as I know, I do not know of any specific dictate that says that you shall only use safety grade equipment for those plants.

Q Okay. So you think that is a requirement that has changed over time but did not apply at the time that TMI 1 was licenced.

MR. BAXTER: I object. I don't believe that we have had a foundation that there has been any requirement established.

MS. WEISS: The witness said on plants of this vintage he was not aware of that, and I was exploring that.

MR. BAXTER: But he hasn't testified there is a requirement now. It is assumed in your question.

CHAIRMAN SMITH: I think the objection, without further explanation on your part, Ms. Weiss, should be sustained.

BY MS. WEISS: (Resuming)

Q You do accident analyses for plants that are currently being licensed, don't you?

A (WITNESS JONES) Yes, I do.

Q And don't you regularly assume the failure of all non-safety grade equipment in making those analyses?

A (WITNESS JONES) In performing those analyses, I don't remember doing a detailed review of failing non-safety grade equipment, but we have relied on the current plan for plants under construction, we have relied basically on only safety grade equipment.

Q With respect to the second exception that you give, that is, operator action to cross connect the HPI system,, was that an operator action which your analysis assumed to be completed within ten minutes of the initiating event?

A (WITNESS JONES) Yes.

Q What is the significance of the ten minutes?

A (WITNESS JONES) Basically what the analyses showed was that that action had to be taken within ten minutes to assure safe consequences as defined by 50.46 for small break LOCAS in the pump discharge pipe, but I would like to note that action no longer exists.

Q All right, that is an action which -- well, let me strike that before we get onto that.

I am just asking with reference to the two exceptions which you set out, you say that your analyses prior to the TMI 2 accident assumed no mitigating operator actions within ten minutes, with that one exception.

I am wondering what is the significance of the ten minutes. Does that have some regulatory meaning? Why is it ten minutes instead of five minutes, twenty minutes?

A (WITNESS JONES) Well, there were other operator actions as we described the other day, such as the switchover to the emergency sump. Generally speaking, a rule of thumb, if you wish, that is kicked around is, say, twenty minutes for operator action, but I know of nothing that says it is a regulation. It has basically been a rule of thumb.

Q A rule of thumb. You mean it is not generally considered a good idea on a plant design to require operator actions to be performed to assure safety within ten minutes?

A (WITNESS JONES) By rule of thumb, what I mean is the NRC generally requires that you do not take credit for operator action within the first twenty minutes, generally speaking.

Q Okay.

And the testimony goes on to say that NRC required you to change that even before the accident.

Is that correct?

A (WITNESS JONES) Specifically what do you mean?

Q Didn't NRC require you to modify the design so that no operator action would be required to make that cross connect within ten minutes? I mean, they required you to do

that before the accident?

A (WITNESS JONES) It is my understanding that they required --

DR. JORDAN: Would you repeat? I missed the question in part.

WITNESS JONES: --the elimination of the operator action at ten minutes to cross connect the HPI. It is my understanding that that request was made to the Licensees to modify their systems. That is my understanding.

BY MS. WEISS: (Resuming)

Q Isn't that the effect of your testimony, Mr. Broughton, in the middle of page 4? Isn't that what you are talking about?

A (WITNESS BROUGHTON) The testimony that discussed kthe high pressure injection lines having been modified does pertain to the elimination of operator action.

Q And was that required by NRC prior to the TMI 2 accident?

A (WITNESS BROUGHTON) Yes. This whole issue came up before the accident. The changes were designed, as I recall. We had scheduled to install them. All of those actions were scheduled before the TMI 2 accident.

Q Do you have a copy of Fiogure 302-661 in Section 9 of the Restart Report before you?

We have a Volume II. It is section --

A (WITNESS BROUGHTON) I don't. I'll try to get one.

(Pause)

Q I am going to ask you, if you are going to have to break and get a copy, I am going to ask you to show us with particular reference to numbers, so the record will be clear, where the flow limiting devices will be installed.

(A brief recess was taken.)

MR. BAXTER: We are ready.

BY MS. WEISS: (Resuming)

Q Could you tell me, please, with reference to the valve numbers, where the flow limiting devices are going to be installed?

A (WITNESS BROUGHTON) Yes. On the figure in the Restart Report, I'll give you an example on one of the four injection lines, and it is a similar installation for all lines.

In Section A-3 is the injection line for Loop A. It is labeled "to Pump A" and in Sectin A-3 is a check valve labeled MUV 107A. The flow limiting device would be installed between that check valve and the next check valve toward the coolant system, which is labeled MUV 95. There would be a similar flow limiting device on each of the four injection lines installed between those two check valves.

There is more to the modification than just the flow limiting device, and that consists of a cross connect

line which, in the example of Injection Line A, would run -- it would be another piece of pipe which would run from the injection line A, between check valve 107A and the flow limiting device over to the injection line for Pump C, which is in Zone 3-D on this diagram.

So the cross connect would be upstream of the flow limiting device in the two injection lines that it cross connects.

DR. LITTLE: Is there a similar cross connection between B and D?

WITNESS BROUGHTON: Yes, there is also a cross connection between injection lines B and D, and it fits into the system in the same relative position, that is, downstream of the check valve and upstream of the flow limiting device.

DR. LITTLE: Okay.

Is the other diagram that you gave a sort of an enlarged diagram of this particular --

WITNESS BROUGHTON: Yes. The other figure contains all of the information relative to the flow limiting devices and the cross connect lines, and it is simpler to use. It does not contain all of the detail that is on the larger diagram.

MR. BAXTER: Excuse me, Dr. Little.

For the record at this point, Mr. Broughton, could

RESPONSE TO SUPPLEMENT 1, PART 1,
QUESTION 36b

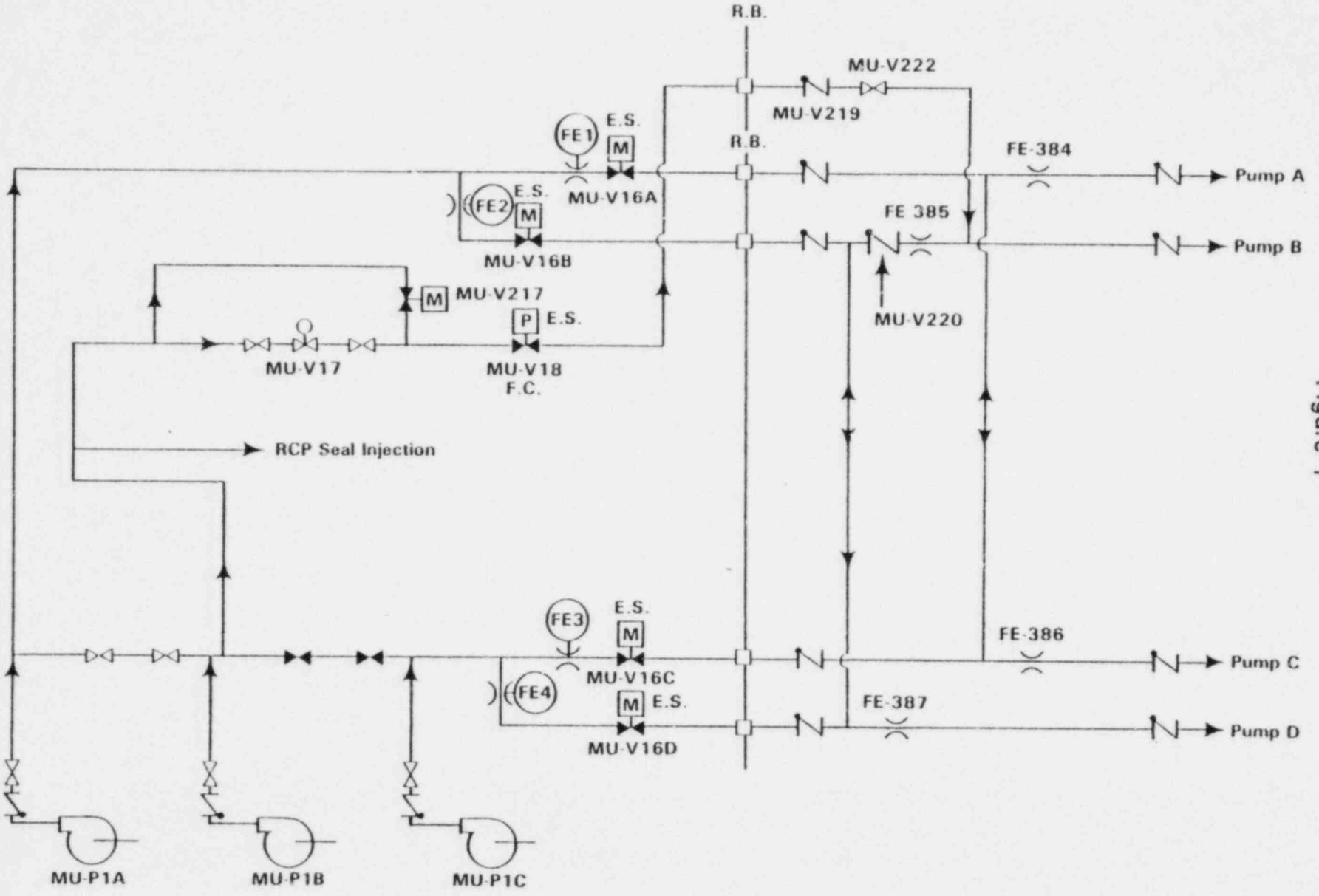

Figure 1

Am. 9

you read the title to that figure that you just referred to, and Dr. Little did?

WITNESS BROUGHTON: This figure is entitled Figure 1, and it also has a label on it "Response to Supplement 1, Part 1, Question 36B," and that is from the Restart Report also.

CHAIRMAN SMITH: Would it be helpful if we were to place Figure 1 into the transcript at this point in the testimony?

MS. WEISS: That's fine.

MR. CUTCHIN: Mr. Chairman, it might also help to note that that is a figure that appeared in Amendment 9, on the copy I have.

CHAIRMAN SMITH: Amendment 9? I'm sorry.

MR. CUTCHIN: The figure that was handed out also had the notation Amendment 9 down in the corner, so that it is clear which amendment it came from.

CHAIRMAN SMITH: Yes.

Do you have an extra copy for the Reporter.

Would you please bind it into the transcript at this point?

(The document, Figure 1, Response to Supplement 1, Part 1, Question 36B, follows:)

DR. LITTLE: Mr. Broughton, I have one question.

What does the designation FE mean? It is an abbreviation for something.

WITNESS BROUGHTON: The abbreviation FE? That pertains to flow element. It is a method of designating what the component is.

BY MS. WEISS: (Resuming)

Q Before we move off of the diagrams, the figure that we asked you about, No. 302-661 from Section 9 of the Restart Report is labeled Revision 18 at the bottom.

So the effect of these changes which we have just discussed is to removed the need for operator action within ten minutes, that's correct?

A (WITNESS BROUGHTON) That's correct.

Q In the analyses which you discussed this morning, at least some of those analyses required operator action to trip the reactor coolant pumps within ten minutes. Is that correct?

A (WITNESS JONES) That is correct.

Q And they also require other operator actions that perhaps go beyond ten minutes, in particular, manually increasing steam generator water level, is that correct?

A (WITNESS JONES) The operator guidelines do call for such actions, yes.

Q Before the accident B&W was able to demonstrate

compliance with 10 CFR Section 60.46 and Appendix K, on the assumptions that, one, the oeprator did what he was supposed to do, and two. the operator did not do what he was not supposed to do, and three, with the exception of a single failure, all equipment functioned as designed.

Is that correct?

A (WITNESS JONES) That is correct.

Q And that is also an accurate description of the assumptions of your post-accident computer analyses, is that correct?

A (WITNESS JONES) That is correct also.

CHAIRMAN SMITH: What was the answer?

WITNESS JONES: Yes.

BY MS. WEISS: (Resuming)

Q Would you agree with me, then, that the accident showed that demonstrating compliance with 10 CFR 50.46, by computer analyses, is not enough in itself to assure safety, but that operator actions are required as well, appropriate operator actions?

A (WITNESS JONES) That is correct.

Q The sentence beginning on just about the middle of page 5 of your testimony is "From these analyses, it was concluded that multiple failures must occur before LOCA scenario can result in a challenge to 10 CFR 50.46 limits."

Would you define multiple failures in the context

of that sentence?

A (WITNESS JONES) More than one failure.

Q And within that definition, multiple failures did occur during the TMI 2 accident, is that correct?

A (WITNESS JONES) Yes.

Q The sequences which you begin to discuss on page 5 and which you also discuss in your tables, for the first sequence, a loss of all feedwater without a small break LOCA, why doesn't high pressure injection automatically initiate?

A (WITNESS JONES) A typical loss of feedwater transient would not result in the depressurization of the primary system to the emergency safeguards system actuation set point with a loss of all feedwater, that is, the auxiliary feedwater not working, you would not expect this system to depressurize any further than that also, so you do not get HPI actuation.

Q So that is one of the examples where operator action is required to initiate high pressure injection and assure cooling, is that correct?

A (WITNESS JONES) Yes.

Q In your second sequence, a small break LOCA with loss of all feedwater, you state that ECCS may not be automatically actuated. This is probably on the record in several other places, but for my benefit, would you tell me

under what cicrcumstances ECCS would not automatically initiate under that scenario?

A (WITNESS JONES) For a small LOCA with all feedwater being lost, with a break size less than approximately .01 square feet.

Q So this is the second example of a scenario requiring operator action to initiate high pressure injection to assure adequate core cooling.

A (WITNESS JONES) Yes.

Q It is your sixth sequence that appears on the second half of page 7, am I correct that that is an example of a scenario where operator action is required to trip the reactor coolant pumps?

A (WITNESS JONES) The sequence, the first sequence, the full sequence starting on page 7, that is where -- that is analysis performed which says that the operator should trip the reactor coolant pumps following a LOCA.

Q And the last sequence, on page 8 of your testimony, a very small LOCA with loss of all feedwater, am I correct that that is an example of a scenario where operator action is required with 20 minutes to actuate either high pressure injection or emergency feedwater in order to assure adequate core cooling?

A (WITNESS JONES) Yes.

Q Let me just go back for a moment to your first

sequence which is discussed in greater detail on Table 2, page 13 of your testimony. If you assume this scenario and make one change, and that is you assume no loss of offsite power and the reactor coolant pumps therefore continue to run, would that add heat to the system?

A (WITNESS JONES) Yes.

Q Do you know if it would add approximately 20 megawatts of heat?

A (WITNESS JONES) When it is passing pure liquid, it would add roughly that much is my understanding.

Q Would that case require the operator to act sooner than 20 minutes to initiate emergency feedwater?

A (WITNESS JONES) No.

Q I'm sorry, to initiate high pressure injection.

A (WITNESS JONES) Probably not.

Q Is that because you don't think that 20 megawatts would be a significant addition of heat?

A (WITNESS JONES) The analysis which was performed did not go and look at the exact latest time the operator could take the action. There was something on the order of 1000 cubic feet or so of water left in the primary system, and I just don't believe that it would change the time significantly at all.

Q You have assumed in the scenario on Table 2, core decay heat rate 1.0 times the ANS scenario, whereas in all

other sequences you have assumed the --

CHAIRMAN SMITH: No.

BY MS. WEISS: (Resuming)

Q Whereas in many of the other sequences you have assumed a standard value 1.2 times, why did you choose to use a lower value for core decay heat in this scenario?

A (WITNESS JONES) Well, the main thing we used -- well, first of all, the objective of the analysis was to develop analytical bases to support operator guideline development. This case which was performed assumed a fairly substantial number of failures in the system. It includes the failure of all the feedwater, and it also includes an additional failure occurring in the high pressure injection system, and under that situation, we assumed 1.0 times the ANS because that is a more realistic value of core decay heat to utilize.

Q Do you know how sensitive the analysis is to that assumption?

That is, if you assumed the 1.2 times the ANS standard value, would that require operator action to initiate high pressure injection within 20 minutes or less than 20 minutes?

A (WITNESS JONES) I don't know. It probably would require it to be actuated earlier, but I am not -- I just don't know what the consequences would exactly be.

Q And insofar as you are aware, B&W has not run this scenario with the assumption of a decay heat rate at 1.2 times the ANS standard value.

A (WITNESS JONES) I believe we had looked at a 1.2 times the ANS value for these other assumptions with this transient, and that an actuation at 20 minutes would not -- was not demonstrated to provide adequate core cooling, but we did not continue that analysis in any great detail. The objective was to keep the core covered, and that analysis did not result in complete coverage of the core.

CHAIRMAN SMITH: Mr. Jones, is that when you decided to go back to 1.0 times, when you saw that it would run -- that it wouldn't work at 1.2?

WITNESS JONES: I don't remember the exact time that analysis was performed. There was an analysis performed many years ago which looked at a scenario very similar to this which showed 1.2 would not work, and we went to 1.0 based on that analysis. When we did this, I do not know whether we looked at -- I cannot remember whether we looked at a 1.2 or not.

CHAIRMAN SMITH: It was because of the 1.2 analysis that you ran it at 1.0.

WITNESS JONES: Yes.

BY MS. WEISS: (Resuming)

Q If you compared the number of failures assumed in

Table 2 with the number of failures that occurred during the TMI 2 accident, would you say that Table 2 assumes significantly more failures than occurred during the TMI 2 accident?

A (WITNESS JONES) I would say they are similar numbers.

Q Okay, Mr. Brouchton, I just have a couple of questions of you.

You discussed the TMI 1 procedures on page 10 of this testimony. In particular, you say that the procedures require that upon automatic initiation of high pressure injection, flow shall not be reduced until and unless, and I won't read those conditions, but you then give the conditions.

Are you saying that for all loss of coolant accidents, HPI should not be reduced except under those conditions as specified in your testimony?

A (WITNESS BROUCHTON) Yes.

Q Do these conditions apply regardless of whether the system is actuated, the high pressure injection system is actuated manually or automatically?

A (WITNESS BROUCHTON) On page 10, the listing of items specifically applied to automatic actuation of the system. On the top of page 11 I discuss requirements which must be met to reduce high pressure injection flow if the

system is initiated manually.

Q Can I just ask you what accounts for the difference?

A (WITNESS BROUGHTON) In the manual case, the requirement to be above 1600 pounds is there because should there be a further reduction below 1600 pounds, the system would automatically actuate, so the intention is not to secure the system in a manual mode below 1600 pounds.

Q But if it had been automatically initiated, and by it I mean high pressure injection, and the pressure dropped below 1600 psi, it would be appropriate for the operator to throttle high pressure injection, assuming the other conditions were present.

A (WITNESS BROUGHTON) He could throttle if he met the condsitions specified on the previous page.

Q We have a problem understanding the answer. Let me explain why.

Suppose that the operator sees pressure dropping, approaching the automatic set point for a high pressure injection, and he manually initiates it before it reaches that set point, why should the conditions for terminating high pressure injection be any different in that case than if he had waiteds for it to actuate automatically?

A (WITNESS BROUGHTON). Well, the basic condition for terminating in all cases -- and it is specifically called

out in terminating it for manual initiation -- is a 50 degree subcooling margin exists, but it can be maintained. If it is possible to maintain the 50 degree subcooling margin with less than full flow and the system has been manually initiated, that is permissible. A goal of the 50 degree subcooling margin is maintained even though the flow is reduced. If the 50 degree subcooling margin cannot be maintained, then it is required to increase flow again to full flow if required to maintain subcooling.

BY MR. POLLARD: (Resuming)

Q I'll just try one more question, and then we will leave this.

I am focusing on your sentence on page 11 where you make a distinction between the instructions to the operator if the high pressure injection system had been initiated manually, and what I am having difficulty understanding is that if the conditions under which you would be allowed to throttle flow had the system been initiated automatically, if those conditions are satisfied, which is what you say in that sentence, why does it make any difference whether the pressure is above or below 1600 pounds?

A (WITNESS BROUGHTON) I think in terms of ensuring that there is adequate subcooling, it makes no difference at all. I think the difference has to do with the condition

that the plant winds up in. In the case where you manually initiate it, you are now back to both at 1600 psig set point, which is the automatic initiation set point.

MS. WEISS: Those are all the questions that I have on the Licensee's testimony in response to UCS Contention 8 and ECNP Contention 1E. It is my personal belief the record would be clearer if we complete this before we go on to the Licensee's responses to the Board questions on 8, but I will, of course, defer to the decision of the Chair. That is a separate piece of written testimony.

DR. LITTLE: Mr. Jones, there are two cases in which the value of 1.0 times the ANS standard value were used, were those shown in Table 2 and 5, and I wonder if you have a copy of the comparable analyses if you assumed a 1.2 times ANS value. I don't necessarily mean right with you, but do these exist?

WITNESS JONES: As I stated, I am not sure about for Table 2 except it is an older analysis that was done in the 1975 vintage. There may be such an analysis around, there may not, that I could have put my hands on very easily.

As far as Table 5 is concerned, there is a direct analysis which has been performed which is Licensee's Exhibit No. 6.

DR. LITTLE: All right, so you don't think -- you don't know whether you have one, a similar one for Table 2

at 1.2 times.

WITNESS JONES: No, I do not.

DR. LITTLE: How long does it take to do one, to run an analysis when you just have one number to change?

WITNESS JONES: With typical LOCA modeling, it would take about 20 computer hours. So as far as, you know, other than trying to get on a computer and getting it off and on during the day when people do work, it is not a difficult analysis to do.

DR. LITTLE: Actually, though, you could come up with a written description similar to this table with similar results here based on your knowledge, you could come up with a similar summary of results assuming you used 1.2 times, couldn't you?

WITNESS JONES: Well, yes, and I could do that now. Basically, as far as the Conclusion No. 1, the operator action within 20 minutes to initiate emergency feedwater, that would still be shown to be acceptable for 1.2 ANS. As far as the operator action within 20 minutes to just actuate an HPI pump, I am not sure that that would be shown to be acceptable, certainly not -- well, I said I believe what we had found out in the past was that that 20 minutes was just too late, and I am not sure how much of an impact an extra five or ten minutes earlier in actuating that system would make on the overall analysis, whether it

would show acceptable results. It is a fairly substantial breakdown of your systems, your normal system functions. That is, you have lost all feedwater, and then you have lost on top of that even the other HPI pump, and that is a fairly substantial set of circumstances to deal with, and I am not so sure that one HPI would show adequate core cooling with a 1.2 ANS.

DR. LITTLE: I have one other question before we leave this whole topic.

Licensee's Exhibit 14, and many other instances which we have discussed in the last several hearing days, showed that part of the water which could be used in cooling would be that recirculated from the reactor building sump, and I questioned the -- whoever can answer -- the characteristics of the pumps that would be used to do this pumping. Are those pumps designed to pump water that may have particulates or other characteristics other than nice clean water which pumps easily?

Are the pumps capable of handling contaminated water with maybe even large particulate material present?

WITNESS JONES: I am reasonably -- what I have been told. I do not design these specific pumps. I set criteria for the pumps. What I have been told is that the pumps are capable of handling particulate matter, at least the low pressure injection pumps very easily because they

have got very large clearances within the pumps themselves. High pressure injection pumps have very tight clearances, and there's always, the particle sizes are adequately screened out or somehow kept down to a minimal size, on the order of a quarter of an inch or less, the pump would be able to pass that material; the wear rings or some mechanical part of the pump would indeed be damaged, but not in a way that would reduce the capability of the pump to continue its function. It would be apt to degrade the head of the pump, its pumping head, by about 5 or 10 percent of the numbers I have been told.

DR. LITTLE: Under a quarter inch size? It will accommodate up to a quarter inch size?

WITNESS JONES: At least up to that size is I know what they have looked at because that was a survey of the screen sizes of the sump protecting the discharge lines. The screen sizes were quarter inch and less.

DR. LITTLE: So the reason there is for screening out larger chunks of material.

WITNESS JONES: That is my understanding, but that is very plant specific in design, but that is at least my understanding of what hey have looked at.

DR. LITTLE: Do you agree with that, Mr. Broughton, or do you have additional information?

WITNESS BROUGHTON: Yes. I cannot add anything

more to the detail, but those are concerns which were considered in the design of those systems.

DR. LITTLE: Okay.

CHAIRMAN SMITH: Dr. Jordan feels that Ms. Weiss' suggestion is a good one, to finish up the testimony here before we go to the testimony on the Board's questions.

Do you have any comment, Mr. Baxter?

MR. BAXTER: No. In this case, Mr. Chairman, I think they are severable. They will be later examined, but I think the Board questions really are intermeshed, and we would prefer to see them together, but in this case I think it is fine.

CHAIRMAN SMITH: Mr. Cutchin?

MR. CUTCHIN: I have only a couple of questions, Mr. Chairman, but if you are asking if I have any objection to this, the answer is no.

CHAIRMAN SMITH: Mr. Adler?

MR. ROBERT ADLER: We have no objection.

CHAIRMAN SMITH: Mr. Cutchin?

BY MR. CUTCHIN:

Q Mr. Jones, in your testimony in response to UCS Contention 8 and BCNP 1E at page 7, ands again at page 8, you are identifying the need to trip reactor coolant pumps to avoid inadequate core cooling.

Can you give me an idea of how rapidly one assumes

those coolant pumps are tripped following the accident initiation?

A (WITNESS JONES) Well, the analysis itself is given an exhibit number 11 --

Q Can you give me just the number, or a range?

A (WITNESS JONES) It is a range of trip times, the earliest time being on the order of three minutes to trip the reactor coolant pumps, and to the best of my knowledge or experience, at plants which have reached the signal, such as Crystal River 3, the action was taken in something on the order of 30 seconds to trip the reactor coolant pumps.

Q And what is the consequence of not tripping that rapidly for the scenarios where tripping has been found to be important in terms of peak clad temperature numbers or other requirements?

A (WITNESS JONES) Well, assuming that the pump trips in the window which has been identified as pump trips could lead to possible core cooling using Appendix K which show peak cladding temreratures possibly in excess of the criteria. However, under realistic examination where you assume that the safety systems function normally, you use a realistic core decay heat and use a typical power profile within the core; the peak cladding temperatures on the worst case was less than 2000 degrees.

Q Thank you.

Mr. Broughton, I have one question for you. On page 10 of that same testimony -- and it runs over to page 11 -- you cite what the procedures require in order to reduce HPI injection flow. That starts on page 10. Over on page 11, one of the requirements is to avoid excessive reactor vessel pressures down from temperature limits.

Can you explain what that concern is?

A (WITNESS BROUGHTON) Yes. The concern would be that for the cold injection water into the vessel, too high a pressure in the system could violate limits established to prevent damage to the vessel due to high pressures under low temperatures.

Q And in that circumstance, the concern for core damage would be greater than the concern for subcooling or maintaining the 50 degree subcooling?

A (WITNESS BROUGHTON) Yes. In addition, at TMI 1, because of the age of the vessel, it is likely that both of those concerns can be met, in other words, that the 50 degree subcooling margin can still be met while reducing the pressure low enough to prevent this concern of overpressure at low temperature.

MR. CUTCHIN: Thank you. No further questions.

CHAIRMAN SMITH: Mr. Adler?

BY MR. ROBERT ADLER:

Q Thank you. We just have a few questions.

I would like to go back for the reason for the reliance on manual actuation of HPI. Just to summarize briefly your testimony, you stated that under certain scenarios, the pressure would not reduce to the 1600 psi set point and therefore the HPI would not be automatically actuated, is that correct?

A (WITNESS JONES) That is correct.

Q Is that the only reason for reliance on manual actuation?

A (WITNESS JONES) I don't really quite understand your question. It has to be manually actuated because you have not reached the applicable set point, the lower pressure set point, and that is the reason we relied on operator action.

In addition, there is sufficient time for the operator to take that action, and then it is 20 minutes after we have lost feedwater. All the scenarios that require the operator action within 20 minutes to either establish the emergency feedwater or actuate the HPI are all cases in which you have an event in which both the main and the emergency feedwater systems are lost.

Q My question is really this: Have you considered the possibility of using some other signal besides pressure to automatically actuate the HPI?

A (WITNESS JONES) To my knowledge, it has not been

considered, no.

Q You haven't done any studies to determine whether some other signal could be used either to replace pressure or to supplement it for those cases where the pressure did not go down below 1600?

A (WITNESS JONES) As far as I know, no such studies have been done.

Q In your opinion, might it be possible to use the subcooling margin, again either to supplement pressure or to replace pressure as a signal?

A It may be possible to use the saturation meter to actuate the HPI, at least as a supplemental. Whether it is necessary or not, I don't know. My own feeling is with 20 minutes to take such an action, that is more than sufficient time to just actuate the pumps from the control.

Q But didn't you state to Ms. Weiss that you felt that automatic actuation was preferable to manual actuation?

A (WITNESS JONES) I don't remember whether I stated that or not.

Q Well, do you?

A (WITNESS JONES) Having automatic actuation of the entire system generally speaking would be preferable. Whether it was required or not or whether there were any other problems in choosing that set point would have to be examined further before I could say right away yes, go do

it, but it is preferable in my mind to rely on automatic actuation. Whether it is necessary or not, again, is questionable.

Q When did it first become apparent to you that delayed reactor coolant pump trip can lead to unacceptable consequences?

A (WITNESS JONES) In roughly the July timeframe of '79.

Q After TMI 2.

A (WITNESS JONES) Yes.

Q Why, in your opinion, was that not discovered previously?

A (WITNESS JONES) The typical Appendix K analyses require utilization or assume a loss of offsite power. Generally speaking, you either assume a loss of offsite power as part of the initiating event, such as on a reactor trip where you have lost the station and possibly perturbed the grid, or you assumed the pumps remained running continuously. Why exactly this scenario was not analyzed previously, I don't really have an answer except for what I just stated. That is the way that the analyses were performed in that timeframe previous to the TMI accident.

Q Since the accident, have you performed any sensitivity studies or scoping studies to determine whether other non-safety systems or components may continue to

operate and that they will lead to unacceptable consequences?

A (WITNESS JONES) I personally have not done such a study. I am not sure whether such studies have been done or not.

Q Is it conceivable to you that there are other systems that should be studied?

A (WITNESS JONES) As I stated, I am not sure whether those other systems have been studied or not. It is beyond my typical scope of responsibilities. I just do not know whether such work has been done or not, so I really cannot answer your question.

Q As I understand your testimony, Tables 2 through 7 do not meet Appendix K. Is that correct, or do not necessarily meet Appendix K.

A (WITNESS JONES) Well, Tables 2 through 7 were not done to demonstrate compliance with Appendix K or 50.46. They were done to develop operator guidelines.

Q I understand that.

A (WITNESS JONES) I do believe they meet Appendix K, however, all the assumptions --

Q That you used in the scenarios meet Appendix K?

A (WITNESS JONES) Some of the scenarios that we have analyzed beyond the bounds of, say, what I would consider an Appendix K analysis. If I made appropriate modifications to make it meet Appendix K, then the specific

events would meet Appendix K.

For a quick example, Table 2, loss of all feedwater without small break LOCA, it has both a failure of the emergency feedwater system, a single failure in the HPI, and the core decay heat is 1.0 times the ANS standard. If I take my single failure to be the loss of emergency feedwater, then I would have two HPI pumps. If I had two HPI pumps, I could meet a core decay heat of 1.2 times the ANS standard value.

Similarly, the other analyses run the same way.

Specifically, no, some of the tables will not meet Appendix K because they are beyond the Appendix K assumptions on failures, for example.

BY MR. ROBERT ADLER: (Resuming)

Q Let me ask you this. Are there any LOCA's that should not have to meet the requirements of Appendix K, notwithstanding the fact that you didn't perform these studies for that purpose?

A (WITNESS JONES) I am having trouble answering questions when you say meet Appendix K. You mean 10 CFR 50.46, the specific criteria? Because Appendix K, the requirement features of a model and analyzed in a LOCA, that is one item, and then there are the specific design criteria, so I am having trouble directly answering your question.

Q Okay. You used .46.

A (WITNESS JONES) In my opinion, the LOCA's by law are required to meet 10 CFR 50.46 using an evaluation model which is in compliance with Appendix K, and I believe that is the circumstance for TMI 1.

Q So none of the analyses that you have performed here should fall without the Appendix K bounce.

A (WITNESS JONES) Well, some of the analyses would never meet Appendix K. I think I -- them directly. Table 2 would never be done for an Appendix K analysis -- it is beyond the design basis -- because of the multiple failure scenario. Table 3, you could state, would be within the Appendix K bounds, and has indeed been shown to meet 50.46.

Table 4 is within Appendix K. That meets 54.46. Table 5 again is beyond the Appendix K failure criteria. Table 6 meets Appendix K assumptions in all respects, and meets 50.46. Table 7 is not really a 50.46 analysis, except what it demonstrates is -- It is a specific scenario. What it demonstrates is that if the operator terminates the reactor coolant pumps upon receipt of the appropriate signal, then the Appendix K analyses remain valid, and Table 8 again, Table 8 would be covered by the Appendix K assumptions, and meets 50.46.

Q Mr. Jones, what I don't understand is this. It seems that whenever you say an analysis is outside the scope of 50.46, you then modify one of your assumptions. For example, in Table 6, you use a 1.0 ANSI instead of a 1.2. I am trying to understand why whenever you perform the LOCA analysis you don't automatically use the assumptions that would be required by 50.46 in Appendix K in light of your statement that no LOCA should be outside of the bounds of 50.46.

A (WITNESS JONES) I said no LOCA analyzed within the Appendix K guidelines should be outside of the bounds of 50.46, and to go and discuss -- well, first off, as I stated, the specific reasons for doing Tables 2 through 8 ultimately were to develop opreator guidelines. They were not done for 50.46 compliance. Those are listed on Table 1.

Now, as far as Table 5 is concerned, Appendix K requires you take a single failure, and in fact, Appendix K is quite specific in what that single failure is. It is a failure of one of the safety systems, but then that becomes the only failure you have to consider.

Now, if I take that as a failure, then I have my feedwater system available to me, my emergency feedwater system, at least one pump, and with that, I can take a PORV failure with 1.02 ANS, emergency feedwater available and a single failure and a high pressure injection system and end up with results that meet the 50.46 compliance.

Now, when I said these other cases also met Appendix K --

Q Before you get to that case -- excuse me -- doesn't that assume that emergency feedwater is completely safety grade?

A (WITNESS JONES) It is my understanding that the emergency feedwater system is safety grade for a LOCA.

Q Okay. Go on.

A (WITNESS JONES) Now, as far as the other analyses that I was referencing, what I did was, I made one minor little twist of the Appendix K assumption, which was, instead of taking a single failure specifically listed in Appendix K, which is in the emergency core cooling system, I am just changing that or had in answering that question just

changed that to the very general design philosophy which was a single failure.

Now, if I do that, then, while I don't know of any single failures that completely eliminate the emergency feedwater system personally, I have assumed that a single failure exists which wipes out the emergency feedwater system. Given that, a LOCA 2 HPI is available, 1.2 ANS, I can meet the 10 CFR 50.46 limits, and I believe that meets the intent of Appendix K, as far as the single failure requirement.

MR. ROBERT ADLER: We have no more questions.

CHAIRMAN SMITH: Dr. Jordan has a few questions that appropriately belong before your redirect.

DR. JORDAN: Mostly clarifying questions. First of all, with respect to small break LOCA's, no matter what size, with no other failures, can you meet the 50.46 criteria without any operator actions?

WITNESS JONES: In the short term, yes. There is a long-term manual action of raising the steam generator level and also the sump switchover.

DR. JORDAN: What was the first one?

WITNESS JONES: Raising the steam generator level.

DR. JORDAN: I see. That is a long-term action?

WITNESS JONES: That is correct. All of the analyses done to demonstrate compliance with 50.46 and all

these other evaluations which have been performed which you feel are based on the normally controlled subpart and the steam generator, 50 percent on the operating range

DR. JORDAN: You are saying that even if he fails to ever go to the 95 percent, he will still meet the temperature criteria? I really don't understand quite.

WITNESS JONES: In the long term. It is break size dependent, again, like many of the small break. In the long term, the breaks which require the steam generator or were to utilize the steam generator for heat rejection, then I believe it is at least prudent to raise the level to 95 percent or thereabouts. I am not sure of whether it is absolutely necessary or when that time has to occur, but it would be on the order of, say, an hour after the accident, and it would be a very long-term action, not a short-term action.

DR. JORDAN: Now, those cases, other than the Table 1, were all assuming a failure beyond a single failure. Is that right? Well, it was a failure -- Table 1 was the failure being a break. The other tables were a break plus another failure or a failure of a major system such as the emergency feedwater system. Is that right?

WITNESS JONES: No, I wish the tables grouped up and simplified that easily. Generally speaking, that is correct, but Table 1 is not all the break, but includes a

single failure which specifically is a -- for the analysis, which it loses its HPI pump.

DR. JORDAN: Yes, but not a failure of any system.

WITNESS JONES: That is correct. As far as the tables that follow, there are -- the scenarios, which are a loss of all feedwater type scenarios with the small break LOCA or without are indeed beyond the design basis type failures.

DR. JORDAN: I see.

WITNESS JONES: But there are cases, for example -- Table 6 specifically, which is basically a typical small break LOCA scenario, except that we have just analyzed smaller sized breaks. That is all Table 6 represents. They are the typical normal small break LOCA assumptions right there in Table 6.

DR. JORDAN: Why wasn't Tabl 6 included in Table 1?

WITNESS JONES: They were not included in Table 1 from the sense that Table 1 was first to present only the pre-TMI 2 LOCA evaluations.

DR. JORDAN: I see. Table 1 goes down to 0. --

WITNESS JONES: .04.

DR. JORDAN: .04. And Table 6 goes to smaller breaks. Is that the main difference?

WITNESS JONES: That is correct. Table 1 is not only -- it is not just the pre-TMI LOCA evaluations. They

are also the 50.46 spectrum analysis.

DR. JORDAN: I see. Does 50.46 say you only have to go to .04?

WITNESS JONES: No, 50.46 and Appendix K coupled up say you must demonstrate that all small break LOCAs which are defined as breaks greater than that which can be handled directly by the make-up system can be controlled within the limits or mitigated within the limits of 50.46, but the typical design analyses are only performed -- are performed on a sufficient spectrum to demonstrate compliance and provide the reasonable assurance. That is what Table 1 represents.

Basically, all that Table 6 does is confirm that judgment.

DR. JORDAN: I see. Table 6 then requires no more operator actions than you had in Table 1, which was zero, excepting for, as you say, the long-term reactions of sump and steam generator height.

WITNESS JONES: That is correct except for, as I stated in the testimony, when we did Table 1, we had assumed an operator action which has since been --

DR. JORDAN: That has been obviated?

WITNESS JONES: That has been obviated by the design changes to the HPI system. Now, there is, of course -- we have added one.

DR. JORDAN: There is what?

WITNESS JONES: We have now, of course, added an operator action, which is the prompt tripping of the reactor coolant pumps upon receipt of an ESFAS signal. And that is by analyses which are discussed in Table 7.

DR. JORDAN: Did you not get an ESFAS signal in Table 6?

WITNESS JONES: You do get an ESFAS signal for Table 6.

DR. JORDAN: You do get it.

WITNESS JONES: Yes. Which means that the operator today, by today's procedures, would be required to trip the reactor coolant pumps for those breaks.

DR. JORDAN: All right. Then, is this not a case where an additional operator action is required, excepting, as you say, if he doesn't do it, it is conceivable as the pumps continue to run you would be all right?

WITNESS JONES: Well, I wouldn't want to call that the only case that fits that circumstance. Basically, all the cases have assumed, with the exception of the analysis in Table 7, I would assume the reactor coolant pumps were lost. Now, all of the analyses done between Table 1 through Table 6 were done without the knowledge of this delayed reactor coolant pump trip are a potential problem.

And then what Table 7, the analyses on the reactor

coolant pump trip aid was to make the procedure such that you would have the pump trip as assumed in the evaluations for all cases now.

DR. JORDAN: Well, you can gather my concern is with the requirement for operator action, particularly short-term operator action, and the signals that the operator gets to trigger him to take the proper action. But first, I want to make sure that we have considered all of those cases where a single failure requires operator action.

Now, then, is pump trip then the only short-term action that must be taken by the operator?

WITNESS JONES: Yes.

DR. JORDAN: All right. Now, how strongly is the operator triggered? What is the basis? What instruments is he looking at? What warning signals is he getting?. How sure is he that he knows what operator action should be required and would he take the right action?

WITNESS JONES: Let me discuss that first from the guidelines, and how we developed the action. Once we found this problem and looked at the short-term nature of the short-term time frame required for the action, we decided that we needed to have an alarmed signal in the control room. It was also felt that the operator should not be required to diagnose the accident. That is, if he has a low pressure ESFAS actuation, he just goes and does it no matter

what the accident is.

There have then been several evaluations which have been performed to demonstrate that for other transients, not LOCA related, and that action does not result in unsafe consequences, and so what we did when we developed it was simply say, ESFAS, trip it, and as a matter of fact, in the guidelines, they state that that is his first priority, basically.

DR. JORDAN: I see. And there is no question that when there is an ESFAS signal, the operator knows it. Is this right?

WITNESS JONES: That is my understanding. There are in the control room.

DR. JORDAN: Can you supplement that answer?

WITNESS BROUGHTON: Yes. If an emergency safety feature actuation signal were generated in addition to the enunciators, the lighted windows above the control panels, there is a specific enunciator that would indicate that the signal had degenerated. In addition to that, there is a complete status panel which indicates the conditions of all of the competence in the safeguard system, and when the safeguards signal is generated, those components begin to move from their standby condition into their functioning condition.

So, there are many indicators that would indicate

that indeed there was an ESFAS signal which would tell the operator to take this prepared action.

DR. JORDAN: Mr. Baxter, we will be getting into the nature of the instruments available to the operator at a later time. Isn't this correct?

MR. BAXTER: We have a contention, several contentions on the control room configuration and the instrumentation available, yes, and the human factor is reviewed.

DR. JORDAN: All right. Then I think I won't ask about that at this time.

Now, we did discuss the case where high pressure injection was initiated either by operator action or automatic action, and you mentioned conditions for terminating high pressure injection. Among them was the 50 degree subcooling margin. Was that correct?

WITNESS JONES That is correct.

DR. JORDAN: Now, I know we do get into this later, but is there a meter which says what the subcooling margin is?

WITNESS JONES: That is my understanding. There will be such a meter in there.

DR. JORDAN: Mr. Broughton?

WITNESS BROUGHTON: Yes, at the time of restart of TMI 1, there will be an indication in degrees Fahrenheit of

subcooling.

DR. JORDAN: I see. And I gather it is the licensee's position that that is adequate. You do not have to have a meter which indicates level of the cooling or inventory.

WITNESS BROUGHTON: That is correct. The current guidelines which were developed to deal with the small break LOCA can be implemented using instrumentation that will be available at TMI at the time of restart. They did not rely at the time on any level indicators within it.

DR. JORDAN: Level indicators what?

WITNESS BROUGHTON: Within the reactor vessel or the primary coolant system.

DR. JORDAN: All right. That is what I am interested in. Just give me a moment to look through this.

(Pause.)

DR. JORDAN: I believe that is all the questions I have for these witnesses.

CHAIRMAN SMITH: Mr. Baxter? I thought we might take our afternoon break a little bit later today, because we are going to be running late, but if you would prefer to have that now --

MR. BAXTER: No, this is fine.

REDIRECT EXAMINATION

BY MR. BAXTER:

Q Mr. Jones, there were several questions about your use in two instances of a core decay heat value of 1.0 times the standard ANS value as opposed to the use in other cases of 1.2 times the standard ANS value. Can you tell me if you are familiar at all with the process of selecting the ANS standard value, whether indeed that itself is a realistic value or whether there are conservatisms built into that value, and if so, what they are.

A (WITNESS JONES) Well, the ANS standard value that I am quoting in the testimony is the -- I guess, the 1971 proposed standard that is referenced in Appendix K, and in Appendix K and in that standard there was an uncertainty listed on the decay heat of plus or minus 20 percent in the first 1,000 seconds after shutdown.

There have been a substantial number of investigations over the last few years looking at the decay heat curve, and performing core decay heat experiments, and those studies have demonstrated that the original ANS standard, the 1971 version, is somewhat conservative, and best estimates in the long-term for core decay heat based on, you know, for a realistic determination, and that a factor of 1.0 is adequate to properly define the core decay heat with time.

MR. BAXTER: I have no other questions.

CHAIRMAN SMITH: Anything further of this witness

over this series? Anything further on this testimony?

MS. WEISS: Yes.

RECROSS EXAMINATION

BY MS. WEISS:

Q It is true so far as you know, Mr. Jones, that Appendix K still stipulates use of the ANS standard value for core decay heat?

A (WITNESS JONES) Well, Appendix K stipulates the use of 1.2 times the 1971 ANS standard, yes, along with a whole host of other assumptions.

Q Dr. Little asked you questions, Mr. Broughton, I think, about the capability of the low and high pressure injection pumps to handle particulate matter, and I believe you said that the HPI pumps can handle anything under a quarter of an inch. Is that correct?

A (WITNESS BROUGHTON) Mr. Jones is the one who provided those.

Q Okay.

A (WITNESS JONES) Yes, that is what I stated, that that is the size that I have some knowledge about the ability of the pump to handle. It does not necessarily mean it could not handle more. I just don't know.

Q Have there been any tests run on this?

A (WITNESS JONES) I really don't know. What I answered Dr. Little with was what I had read in internal

memoranda, not from my own firsthand examinations and calculations.

Q Mr. Broughton, do you know if any tests have been run on this?

A (WITNESS BROUGHTON) No, I do not.

Q With respect to the memos that you saw, Mr. Jones, do you recall how long the pumps were assumed to operate?

A (WITNESS JONES) The analysis that I can remember did not have a specific time in it. What it did was looked at the effect of the circulation of the particulate matter, what it would do mechanically, which was to wear down some of the clearance range of some port in the pump. The wearings, I believe, were called. They were called. And the effect of that would be to decrease the ability of the pump to produce head somewhat, and other than that, that is all I know about the analysis. I do not know any more than that.

Q You discussed with Mr. Adler the fact that typical Appendix K analyses assume a loss of off-site power. Did that appear anywhere in Appendix K?

A (WITNESS JONES) I believe it does.

Q Does it specify at what time off-site power is lost?

A (WITNESS JONES) I do not believe it specifies the time.

Q You testified that the post-accident analyses were done to prepare operator guidelines rather than to demonstrate 50.46. That is generally correct, isn't it?

A (WITNESS JONES) That is correct.

Q Are those operator guidelines for operator scenarios that you do not consider in demonstrating compliance with 50.46 and Appendix K?

A (WITNESS JONES) Those are indeed scenarios which are not normally considered in demonstrating compliance to 50.46.

MS. WEISS: Thank you, gentlemen. I have no further questions on this piece of testimony.

CHAIRMAN SMITH: Anything further?

DR. JORDAN: I guess one thing further on that last question. Is this then -- these scenarios, are they made to answer one of the lessons learned requirements?

WITNESS JONES: I don't really know the exact source. I do know that following the accident there were seven meetings with B&W and NRC along with the licensees, and that the staff wanted operator guidelines developed by the vendors, and to perform additional analyses, and picked some of the specific scenarios.

DR. JORDAN: I see. All right.

MR. BAXTER: Dr. Jordan, I believe the Board could take notice that this requirement is embodied in orders by

the Commission to each of the B&W operator licensees.

DR. JORDAN: This is one of the original orders.

MR. BAXTER: Yes, sir.

DR. JORDAN: Okay.

MR. CUTCHIN: Mr. Chairman?

CHAIRMAN SMITH: Is that incorporated in I&E Bulletin 7905, C&O&C, and by reference incorporated into the notice of hearing of this case?

(Pause.)

MR. BAXTER: Mr. Chairman, I was just reading Item D of the short-term actions complete for the potentials for small breaks and developing and implementing operating instructions to define operator action.

CHAIRMAN SMITH: Then you will have three on I&E Bulletin Numbers 7905C and 7906C requiring analyses and then guidelines for operator action.

DR. JORDAN: What I had in mind was whether this was aimed at Item 2.1.9 of NUREG-0578, namely, the analysis of design and off-normal transients and accidents. But I don't need an answer from these witnesses with regard to that question.

CHAIRMAN SMITH: Shall we proceed, then, with the next item of testimony?

Let's take our afternoon break. We will return at 3:20.

(Whereupon, a brief recess was taken.)

CHAIRMAN SMITH: Mr. Baxter, are you ready to proceed? The testimony is already in. I had overlooked that. Ms. Weiss?

CROSS EXAMINATION

BY MS. WEISS:

Q Gentlemen, the questions I will be asking you now refer to licensee's testimony of Robert C. Jones, Jr., and T. Gary Broughton in response to the Board question on UCS Contention 8. That testimony is dated October 28th, 1980.

Let me refer you first to Page 3, which discusses Recommendation 2.1.2.A of NUREG-0565. That recommendation is that you provide a system which will assure that the block valve protects against a stuck open PORV. Your answer states that the need for such a system has not been determined by appropriate analysis whichis called for by Item 2.K.3.7 of NUREG-0660. What is the publication date of NUREG-0660?

A (WITNESS BROUGHTON) I don't know.

Q Do you have a copy of it in front of you? I would like you to locate Section 2.K.3.7 in that document.

A (WITNESS BROUGHTON) That may be a confusing reference, because the items in 0660 were not specifically enumerated, and this particular reference 2.K.3.7 is one which was attached to an 0660 item by a clarification letter

which was provided, I believe, in September. I think it would help if I explain what I have referred to here.

Q Just one second, before you get to that. It is true, isn't it, that there is no Item 2.K.3.7 in NUREG-0660?

A (WITNESS BROUGHTON) There is nothing there that has that specific label on it. The way that items have been referenced in 0660 is through a system which goes through the recommendations of other documents and has assigned them numbers. I believe the first place that that was published was in a clarification letter on items which were in 0660.

Q You are speaking of the clarification letter dated September 5th, 1980?

A (WITNESS BROUGHTON) That would be the letter from Mr. Eisenhut, I believe. It is the September Eisenhut letter that I was referring to.

Q Now, let me refer you to the section of 0565, and I ask you to read, if you will, to yourself the section entitled, 2.1, Expected Frequency of Small Break Loss of Coolants Accident, 2.1.1, Conclusions, and 2.1.2, Recommendations. Those appear on Pages 2-1 and 2-2 of NUREG-0565.

A (WITNESS BROUGHTON) Yes, I have read that section.

Q Isn't it true that the purpose of recommendation 2.1.2.A is to mitigate occasions when a PORV may stick open?

A (WITNESS BROUGHTON) That would be the purpose of

having a block valve which automatically shut.

Q And in fact the staff considered doing this by permanently closing the block valve but rejected that precisely because that would increase challenges to the safety valve. Is that correct?

A (WITNESS BROUGHTON) Yes.

Q And instead of permanently closing the block valves, they recommended an automatic PORV block valve closure system. Is that correct?

A (WITNESS BROUGHTON) Yes.

Q Then this recommendation takes account of your stated concern of not increasing challenges to the safety valves. Isn't that correct?

A (WITNESS BROUGHTON) It would take care of that concern with regard to the first time in a scenario that the power operated relief valve would be challenged. However, if there were a subsequent challenge to the primary system pressure boundary with a closed block valve on the PORV, then that might in turn challenge the safety valve.

Q Isn't it true that there is a separate concern in 0565 to reduce the challenges to the PORV and that that concern resulted in Recommendation 2.1.2B. That is, that you provide analyses showing that the PORV will open in less than 5 percent of all anticipated overpressure transients.

A (WITNESS BROUGHTON) Yes. The attempt there is to

release challenges to the PORV.

Q So in this section of 0565, there are two concerns expressed. The first, to reduce challenges to the PORV. The second, to mitigate the consequences of a PORV, and Recommendation 2.1.2A goes to the latter. Is that correct?

A (WITNESS BROUGHTON) Yes.

Q And isn't it true that the clarification to which you refer in Section 2.K.3.7 of the staff's September 5th letter goes to the former? In other words, that is what the analyses are for.

A (WITNESS BROUGHTON) Yes, I believe .7 does adopt Recommendation B from 0565.

Q And for purposes of clarity, the licensee has not provided us an automatic block valve closure system for TMI 1.

A (WITNESS BROUGHTON) That is correct.

Q Wouldn't you say, then, that your testimony on Page 3 is really addressing Recommendation 2.1.2B and not 2.1.2A?

A (WITNESS BROUGHTON) Well, I don't believe so, because the work done to identify the frequency of challenges to the valve would identify scenarios in which the valve was challenged and allow us to look at how those challenges might be altered by an automatic block valve, a system that we might instal.

Q Do you know of any accidents that have been identified and analyzed where the pressure would go twice to the PORV set point?

A (WITNESS BROUGHTON) Well, we have looked at scenarios which would cause cycling of the PORV. As a matter of fact, the extended loss of feedwater case that was presented in the loss of coolant accident testimony was such a scenario.

Q Are you referring to bleed and feed?

A (WITNESS BROUGHTON) I am referring to an item which was listed in one of the tables of the last testimony, which looked at the plant response following an extended loss of main and emergency feedwater, which would have then used feed and bleed after a period of some minutes.

Q So you would in fact -- this would be a bleed and feed scenario?

A (WITNESS BROUGHTON) It would be a bleed and feed scenario or a scenario which could be terminated by restoring main or emergency feedwater.

Q Isn't it true that for bleed and feed scenarios, your analysis relies on the safety valves?

A (WITNESS BROUGHTON) The feed and bleed could be accomplished either by safety valves or by power operated relief valves.

Q But power operated relief valve is not safety

grade. Is that correct?

A (WITNESS BROUGHTON) It is not safety grade, but if it functioned, it would be used in place of the safety valves.

Q I understand that, but in those circumstances, the concern of reducing challenges to the safety valves hardly seems a valid one.

A (WITNESS BROUGHTON) I don't agree. For example, in that scenario, if the power operated relief valve is being used, and emergency feedwater can be restored, then by having used the power operated relief valve, should it fail to shut, the block valve would allow that flow path to be isolated. If instead the safety valve had been in operation, there is no isolation path for the safety valve, so it is desirable to use the power operated relief valve when it is available.

Q Is it your understanding, before we go on to the next item, that Recommendation 2.1.2A referring to the automatic block valve closure syster calls for some further analyses to justify installation of that system?

A (WITNESS BROUGHTON) Yes.

DR. JORDAN: Isn't it even further than that? If you read the September 5th letter of Eisenhut, on Page 7, Enclosure 2 refers to automatic PORV isolation and calls for completion of design by 7/1/81.

Now, doesn't that mean the design must be started by now?

WITNESS BROUGHTON: I believe we initiated work on the design, and I don't know the status of that work, but certainly one of the inputs to the design would be this review of other events, and the method in which a design might challenge safety valves instead of reduce challenges to those valves.

BY MS. WEISS:

Q Your testimony is, Mr. Broughton, that on Page 3, that desig and installation of an automatic PORV block valve closure system is not being pursued at this time. Is that correct?

A (WITNESS BROUGHTON) That is correct. I believe we initiated the design work very shortly -- I don't know the time frame we initiated the work, but the work is not actively pursuing now.

DR. JORDAN: Well, I think it probably won't help to pursue this question at this moment. However, this is something that we will be asking the staff about. So, be prepared.

BY MS. WEISS: (Resuming)

Q Your testimony on Page 4 with respect to Recommendation 2.1.2.B of NUREG-0565, this is the recommendation that the clarification in the September 5th

letter refers to. Is that correct?

A (WITNESS BROUGHTON) Yes, that is correct.

Q It is not clear to me from the testimony whether you are saying that there are or are not events which will cause the PORV to open. Could you clarify that for me?

A (WITNESS JONES) There are events that would cause the PORV to open.

Q Those include loss of feedwater, loss of external electrical load, turbine trip, uncontrolled -- control rod withdrawal from startup conditions, inadvertent closure of main steam isolation valves, and inadvertent moderator boron solution. Is that correct?

A (WITNESS JONES) That are not correct. Those are anticipated transients. The recommendation states that you demonstrate that you will not open the PORV in no more than 5 percent of all of those transients. Some of those transients will open the PORV. Not all of them.

Q Okay, which?

A (WITNESS JONES) As was stated in the testimony on Page 5, the inadvertent control rod withdrawal can possibly result in PORV opening. And that is the only one.

Q That is the only event?

A (WITNESS JONES) That is the only anticipated transient that is listed that would possibly cause an opening of the PORV with the revised set points in the

anticipatory trip functions.

Q Are there other transients which are not listed on Page 4 which might result in opening of the PORV?

A (WITNESS JONES) Well, there are total loss of feedwater scenarios, of course, that can result in the PORV being open. A feedwater line break. A rod withdrawal from power could possibly cause it to occur. A rod ejection accident. Those are some of them. But most of those transients are not anticipated transients. A feedwater line break, for example, is a design basis transient.

Q Right. So in the letter you have included both anticipated transients and accidents?

A (WITNESS JONES) In that expanded list, the majority of them were accidents.

Q This recommendation calls for the licensee to document that the PORV will open in less than 5 percent of all anticipated overpressure transients, et cetera. Has that documentation been presented to NRC?

A (WITNESS JONES) Not to my knowledge.

Q Do you know when it will be?

A (WITNESS JONES) No, I do not.

Q Do you know if it is being prepared?

A (WITNESS JONES) I do not believe that such a document is being prepared at this time.

Q With regard to Recommendation 2.2.2.A on Page 9 of

your testimony, and Recommendation 2.2.2.B -- they are presented together -- they first state that the analysis method used for small break LOCA analysis by B&W should be revised, documented, and submitted to NRC for approval. The second recommends or calls for plant specific calculations using the NRC approved model for small breaks to be submitted by all licensees to show compliance with 10 CFR 50.46.

Is it your testimony that NUREG-0565 does not state that approved B&W small break evaluation is deficient for demonstrating compliance of TMI 1 with respect to 1 CFR 50.46 and Appendix K?

A (WITNESS JONES) That is correct.

Q And in making that answer, did you consider the sentence on Page 2-3 of NUREG-0565 that has been read earlier into the record, and I quote, "The analysis methods must be revised and verified before they can be considered for NRC approval under 10 CFR 50.46?"

A (WITNESS JONES) Yes, I have.

Q Let me refer you to Chapter 4 of NUREG-0565.

DR. JORDAN: Where is this, Ms. Weiss?

MS. WEISS: Chapter 4 of NUREG-0565. Basically, Page 4-1, 4-2, and 4-3.

BY MS. WEISS: (Resuming)

Q Did you consider making that statement the eight

listed concerns on Pages 4-1 through 4-3?

A (WITNESS JONES) Yes, I had.

Q Would you agree that for Item Number 1, experimental data for the verification of methods for two-phase natural circulation are currently not available?

A (WITNESS JONES) Not at this time I wouldn't.

Q You would not agree with it?

A (WITNESS JONES) No, I would not.

Q Where is that data?

A (WITNESS JONES) That data has been obtained from the recent LOFT experiments.

Q Which tests are you referring to?

A (WITNESS JONES) It is my understanding that L35 exhibited periods of two-phased natural circulation.

Q Do you know approximately when LOFT Test L35 was performed?

A (WITNESS JONES) It is a relatively recent test. It is in the last month or two, but I do not remember the exact date.

Q Can you describe that test?

A (WITNESS JONES) The experiment is roughly a simulated .1 square foot break in the intact loop of the LOFT facility. They tripped the reactor coolant pumps. Thehsystem underwent a depressurization transient. I believe they had no ECCS model in that transient. And after

some period of time later on, they isolated the break, and I forget when they turned on the HPI. It was some time prior to the isolation of the break. And it was expected that there was reflux boiling or boiler condenser type operation occurring during the experiment, or would occur. The exact date of report has not been released on that experiment yet.

Q Did B&W predict results for that experiment before it was conducted?

A (WITNESS JONES) No.

Q So you have no basis of comparing the experimental data with model predictions for this particular test?

A (WITNESS JONES) Not as a pre-test prediction, no. It is possible we may do some post-tests, but I do not know whether we will or will not.

Q Do you know whether natural circulation was lost at all during the test?

A (WITNESS JONES) Well, liquid natural circulation would have been lost. I expect they lost two-phased natural circulation also at some period of time, and then later on in the transient, after they isolated the break, they did a simulation of the plant depressurization by depressurizing the secondary syste of the plant. That was the counterpart L35A experiment.

(Pause.)

BY MR. POLLARD:

Q Now, if I could direct your attention to the third item beginning on 4-1 and continuing over to 4-2, which is discussing the pressurizer model for analyses of small breaks, and it states that these modeling differences may be significant for various postulated breaks. Do you agree with that concern expressed in Item 3?

A (WITNESS JONES) No, I do not.

Q Have you submitted further analyses to the NRC since the document was published on this item?

A (WITNESS JONES) No, we have not.

(Pause.)

Q On Item 4 on Page 4-2, it states that, "The calculation of core level and core heat transfer are important features of the small break model. Limited experimental data is currently available to justify these models. Although the current capacities have been satisfactory, the experiments are not challenging to the codes. More experimental data must be obtained for code verification."

Do you agree with that concern?

A (WITNESS JONES) Not totally.

Q Have you submitted any further analyses to address this concern since this data was published?

A (WITNESS JONES) No, we have not.

Q Item 5 on Page 4-2 discusses a concern about the

number of nodes used to represent the primary system and concludes by stating that thus the modeling detail could have a significant effect on the calculated times for various events such as ECCS actuation.

Do you agree with that concern?

A (WITNESS JONES) No, we do not.

Q Have you submitted any further analyses to the NRC address this concern since this document was published?

A (WITNESS JONES) No.

DR. JORDAN: Mr. Pollard, I lost that item. Where was it?

MR. POLLARD: That was Item 5 on Page 4-2 of NUREG-0565.

DR. JORDAN: Thank you. I have it now.

BY MR. POLLARD: (Resuming)

Q Item 6 on Page 4-2 states, "During the recovery period from a small break LOCA, the thermodynamic equilibrium assumed and four control volumes could be presumed in error if the predicted system pressure. This could in turn introduce errors in both the break discharge and safety injection flow. The rate at which the water is refilling the system can affect speed condensation. If the condensation efficiency is less than 100 percent, system pressure would be higher than predicted."

Do you agree with that concern?

A (WITNESS JONES) Yes, I do.

Q Have you submitted any analysis to address this concern to the NRC since this document was issued?

A (WITNESS JONES) No.

Q What are your plans for addressing this concern?

A (WITNESS JONES) At this time, B&W has proposed some work to various of our licensees, and we will be examining and developing a program to respond to this document in total. As far as this specific item is concerned, this is an analysis item or this is an item talking about the later stages of the system refill and the analyses do not go that far. The analyses are performed to the time that long-term cooling has been established.

All this item addresses is, how fast can you recover the system, that is, refill it, not whether you have inadequate -- whether you are getting adequate core cooling during this period. Here you are refilling the system following the small break LOCA, and we will probably not address this at all.

Q With respect to the licensee in this case, have you made this proposal to Metropolitan Edison?

A (WITNESS JONES) Yes.

Q Have they responded?

A (WITNESS JONES) They have participated in some of the planning and discussion of the concerns listed in this

document, not just this item specifically. As far as their response to the Commission or whether they are going to participate in the long-term program that may occur with this document, I do not know of any response at this time.

Q When was that proposal originally presented to the licensee?

A (WITNESS JONES) Well, let me try to put a little bit of history on this. This document has been out since January, well, some time -- Really, the end of February was the first time this document was officially published, irrespective of the date on the cover. And when the document -- the Commission did not apply the document to anybody. The first time that positions were asked that I know of on the document were via the September 5th or thereabouts, the September Eisenhut letter, the clarification letter.

At that point in time, B&W had some meetings with the owners of our plants to discuss the various concerns and how to respond. That is still as of the last date that I have been involved in, it was still progressing to develop a response to the NRC officially and as far as I know, I was not until just the end of October that the Commission -- the Commissioners finally approved the clarification letter from Mr. Eisenhut, and I believe that the response on this item is due to the Commission some time around mid-November, but

I am not exactly sure on that, as far as development of a plan to respond to this document.

Q November of which year?

A (WITNESS JONES) I believe this year. November 15th.

Q So you personally are not involved in developing the response. Is that correct?

A (WITNESS JONES) Not since I have started coming to the hearing.

Q We can move to Item 7, which begins on Page 4-2 and continues on Page 4-3. It states that "The discharge rate of two-phased fluid through the PORV and safety valves is an important consideration for some transients. These include postulated stuck open PORV or safety valves and primary system depressurization for very small or zero break LOCA's by opening the PORV if all feedwater is lost. There is a lack of discharge rate data for two-phase fluid at high pressure.

"Most experimental data is for steam at low pressure. If the actual valve flow is lower than assumed in the calculations, primary system depressurization to the high pressure injection set point might not occur within the calculated times."

Do you agree with that concern?

A (WITNESS JONES) Not really.

Q Is it that you disagree with every single sentence in the concern, or is it only a portion of it that you disagree with?

A (WITNESS JONES) Well, that is the problem with all of these. There may be a feature here and there which is characterized correctly, but as a whole, I disagree with the item.

Q Let me just take the last sentence, then. If that sentence is true, would you agree that there might then be some additional accident scenarios which would require manual initiation of high pressure injection?

A (WITNESS JONES) Well, it depends on your concept of lower. In my opinion -- no. I do not totally agree with that thing. I think that a break in a stuck open PORV or safety valve, because of its direct impact, because it is a steam space leak, which is what the valves have been qualified for, would depressurize the primary system down to the high pressure injection subpoint in very rapid fashion, and in fact the TMI 2 incident proves opening a PORV will depressurize you to the high pressure injection subpoint, and the safety valve is a heck of a lot bigger than a PORV.

Q Was there not a case in your testimony on UCS Contention 8 and ECAP Contention 1E where you discussed a transition flow from steam through the PORV and then a two-phased fluid and then pure liquid and back again?

A (WITNESS JONES) That was a response to a total loss of feedwater scenario, and that was the safety valve I was talking about. But if you had the PORV then you would actuate it -- you would see the same conditions.

Q And that is, of course, what this item is addressing, zero break LOCA's, which means no break?

A (WITNESS JONES) That is right, and this item is, in my opinion -- well, if you go back and look at the clarification for this item, which is later in the document, it is basically an item for the Davis Besse plant which does not have the high head, the high head HPI pumps, that is, HPI pumps capable of pressurizing the system at the 2700 psi.

And for them, if they could not restore feedwater, you would have to open the PORV and wait for a system depressurization to allow their HPI's to function belows their shutoff head, which is about 1800 psi. It is a generic item for all other plants such as Westinghouse and combustion engineering plants which generally do not have such high head pumps as the B&W high pressure injection system is.

Q Your last comment moves me to draw your attention to the first sentence on Page 4-1, where it appears that to me, at least, from the first sentence, that this section is being written with respect to B&W. And now you are telling me that here we have an item that may apply to one B&W plant

but it is primarily being addressed to Westinghouse plants. Did I understand you?

A (WITNESS JONES) Well, this is not the only bulletin known as -- this is not the only generic evaluation of small break loss of coolant accidents around. There are three others, for GE, Westinghouse, combustion Engineering plants that are very similar to this, and in fact most of these concerns are interwoven between each of the documents, and in fact in the clarification letter I understand that we were to address the recommendations of every document, because they weren't sure they got them all intertwined properly.

So, it is addressed to -- some of these items may be much more appropriate to a Westinghouse plant than they are for a B&W plant, and this is one of them, with the exception of, again, the Davis Besse plant.

Q If we could move on, please, to Item 8 on Page 4-3, I won't read the whole Item 8. It says, beginning about the fourth sentence, "The amount of steam present at the injection location is the predominant factor which determines the core flood tank master lift. The results of an analysis will be influenced by the model and the modeling assumptions used to calculate the core flood tank flow. Additional studies will be required to obtain the necessary information to perform an Appendix K analysis."

And then it goes on to note, "Additional work to do this is under way."

Do you agree with any part of the concern expressed in Item 8?

A (WITNESS JONES) Well, I agree with it in the sense -- with the equilibrium models that currently exist throughout almost all reactor vendors in the country and throughout most of the country using equilibrium models in the sense that the cold core flood tank water could introduce a rapid depressurization of the system due to the instantaneous condensation of steam as would be calculated by the code.

My specific position relative to this item is addressed on Page 11 of this testimony.

(Pause.)

BY MS. WEISS: (Resuming)

Q Are all of the concerns listed, the eight concerns listed on Pages 4-1 through 4-3 of NUREG-0565 related to performance of the models with the Commission's regulations and Appendix K, 50.46?

A (WITNESS JONES) I missed your question. Could you please repeat it?

Q Now, we have made a distinction some time in the course of the testimony between analyses that were performed under Appendix K to show conformity with 50.46 and analyses

which were done in order to provide operator guidelines. I just wanted to make it clear at this point that all the concerns listed in Chapter 4 of NUREG-0565 that we have discussed relate to conformance of the B&W models for Appendix K and 50.46 purposes.

A (WITNESS JONES) Well, let me try to characterize it this way. First off, the specific document, 0565, relates -- is a review of and relates to the analyses performed to develop operator guidelines. They are not a comprehensive review of the 10 CFR 50.46 analyses and Appendix K analyses.

What the staff has listed that we have just been discussing is based on some of these analyses and what they have seen, if they now have some questions, and they just want further justification.

To the best of my knowledge, the staff has not said that our models are not in conformance with Appendix K to date, and they are not valid for demonstrating conformance to 50.46, but they do want to address some of these -- they want some of these specific concerns addressed over the next year or so.

Q At least they haven't said that anywhere outside the context of NUREG-0565.

MR. BAXTER: I am sorry. I don't understand the question. If the witness understands it, he can answer it.

CHAIRMAN SMITH: I didn't understand it either. But that is hardly determinative.

Could you clarify it?

MS. WEISS: I am not sure it is necessary.

BY MS. WEISS: (Resuming)

Q In other words, beyond this document that we have been discussing, NUREG-0565, you have not received any direction from NRC to tell you that you were not in compliance with 50.46.

A (WITNESS JONES) Well, in fact, I don't believe the Commission has stated to date that we are not in compliance with 50.46, even in this document as modified by the clarification letter. It is my understanding that an acceptable response to this document would be to provide additional information to the staff that demonstrates that the old model, the presently approved model is in compliance with Appendix K, and that these items are not significant actors for demonstrating adequate core cooling as defined by the regulations.

Q Do you see anywhere in this document where it is stated that these are not significant concerns, are not significant efforts in terms of 50.46?

A (WITNESS JONES) No, I do not. But by the same token, there was a letter out which clarifies -- The way I understand it, it is a clarification of this document.

Q You think the September 5th letter clarifies NUREG-0565?

A (WITNESS JONES) It tells you or provides you some guidance on how you could respond to this document if you wish.

Q Well, we may have to talk to staff some more about what they meant. They may be the appropriate people to discuss that.

Let's go to Page 11 of your testimony, Recommendation 2.2.2.C, which states, "The effects of core flood tank injection on small break LOCA's should be further investigated to determine the amount of condensation realistically expected and to determine its effect on heatup and core uncovering. The condensation model and modeling procedures, i.e., injection location used in computer analyses, require further investigations to assure that the effects of CFT injection are biased in a conservative manner.

"Semi-scale and LOFT test data should be used to verify the models."

So, is it accurate that that recommendation calls for a comparison of your model results or your model predictions with semi-scale and LOFT test data?

A (WITNESS JONES) It appears to call for such a comparison, yes.

Q And your testimony does not discuss any such

comparison. It just reiterates the model results. Is that correct?

A (WITNESS JONES) That is correct.

Q Do you know whether some LOFT data indicates less pressurization than predicted by B&W model?

A (WITNESS JONES) Well, one of the large break experiments indicates that effect, that the models will tend to underpredict the system pressure after core flood tank injection. The small break data such as L31, the recent LOFT experiment, does not seem to indicate that the models will tend to grossly overpredict the system depressurization following core flood tank injection.

Q I have stated the question wrong, but you answered it correctly. The LOFT data indicates less depressurization than predicted, rather than less pressurization. Correct?

A (WITNESS JONES) Yes. It should be less depressurization.

Q Do you have any experimental data from semi-scale or LOFT to show that the B&W model is conservative in this respect?

A (WITNESS JONES) Well, the analyses that are generally done for comparison to both the LOFT and semi-scale experiments are generally what you would call best estimate analyses. They are not analyses performed to demonstrate margins of conservatism in your model, but

rather to demonstrate whether your model can reasonably predict the actual phenomena. There are problems in peforming such analysis in the fact that these models are not designed to be totally best estimates, but we have performed analyses of the semi-scale experiment S07-10B.

The model was conservative in that it actually predicted a slower depressurization and a total core uncovery, which is -- the total core uncovery was a direct result of how we treat heat addition in the code, and we do that in a very conservative manner, and it caused us to grossly underpredict or to grossly overpredict the amount of core uncovering.

As far as the LOFT L31 experiment, we have performed an analysis of that. The experimental comparison was in reasonable agreement with the actual data, as best we could determine.

MS. WEISS: Could we have just a moment?

(Pause.)

CHAIRMAN SMITH: Do you need more time, Ms. Weiss?

MS. WEISS: No, we can go. Thank you.

BY MS. WEISS: (Resuming)

Q Can I refer you to your testimony on Recommendation 2.3.A of NUREG-0565, and related conclusions from 0623 with regard to tripping of reactor coolant pumps? The recommendations appear on Pages 12 and 13, and the

testimony appears on Page 13.

I am particularly concerned with the language of NUREG-0565 itself, and you have quoted it. "Tripping of the RCP's in the event of a LOCA is not an ideal solution. The licensees should consider other solutions to the small break problem. For example, an increase in HPI flow rate. In the interim, until a better solution is found, the RCP should be tripped automatically in the case of the small break LOCA."

Now, before I get into that, does that jog your memory at all or refresh your memory at all with respect to the questions Mr. Pollard was asking you earlier about ACRS recommendations, with particular respect to HPI?

MR. BAXTER: This is on HPI pump capacity?

MS. WEISS: Flow rate. Yes.

WITNESS JONES: I still can't remember whether the NRC -- I mean, whether the ACRS has indeed made this recommendation or not. It is possible that they have. But as I said, I just don't remember.

BY MS. WEISS: (Resuming)

Q An increase in HPI flow rate in particular is what I am referring to.

A (WITNESS JONES) Well, this recommendation itself from 0565 makes some statement to that effect, but again, as far as an ACRS specific position, I just don't remember one.

Q Okay. Why isn't tripping of the reactor coolant

pumps in the event of a LOCA not an ideal solution?

MR. BAXTER: You are assuming that the witness agrees with this statement.

BY MS. WEISS: (Resuming)

Q Yes, if you agree with it.

A (WITNESS JONES) Well, I am not so sure that I agree with this statement. The general concerns raised or the -- the very general comments along this vein are relative to forced circulation cooling, not the cooling process itself, but the better coupling to the steam generator for finer plant control.

I mean, as far as the core cooling, as far as core cooling is concerned, I don't believe it matters. I think tripping the pumps is a fine solution.

Q Is it accurate that tripping of the reactor coolant pumps can aggravate non-LOCA transients?

A (WITNESS JONES) It can make them -- well, I don't know what you mean by aggravate, but in a very general sense of, again, plant control, yes.

Q Well, there is a statement on this same page from Conclusion 6.0(5). NUREG-0623 states that "For B&W plants, tripping of the reactor coolant pumps during severe overcooling events increases the potential for interruption of the natural circulation due to steam formation in the coolant loops."

Now, do you agree with that?

A (WITNESS JONES) Well, if you have the reactor coolant pumps, you do not have to worry about an interruption of natural circulation. Certainly, tripping the pumps during a severe overcooling event which could pull or result in a steam void in the reactor coolant loops could possibly interrupt natural circulation.

However, we have performed analyses of overcooling accidents, including a failure, a double-ended failure of the steam lines which results in a very substantial overcooling transient on the primary system, and natural circulation was maintained in at least one loop throughout the transient.

Q What did that assume with respect to the operation of the reactor coolant pump?

A (WITNESS JONES) That they were tripped at the low pressure ESFAS signal.

Q The recommendation speaks of a better solution. It states, "In the interim, until a better solution is found, the RCP should be tripped automatically." Has B&W devised a better solution?

A (WITNESS JONES) We have not been actively pursuing a better solution to the problem. One of the recommendations or what 0623 recognizes is that the analysis models with the reactor coolant pumps running need further

analytical bases or experimental bases, excuse me, and has recommended that a pre-test prediction of LOFT test L36 be performed in order to benchmark the analytical methods with the pumps running in a highly voided primary system during a small break LOCA.

Following that analysis, we possibly may pursue looking at alternate solutions. It may be found out that the analysis methods are indeed highly conservative, and that in fact the pump trip is not required, but there are still mechanical problems relative to the pump operating at high void actions and potential damage to the reactor coolant pumps which would still have to be resolved, and I am not so sure that we will actively pursue development of alternate schemes or other ways of keeping the reactor coolant pumps on indefinitely through a LOCA.

(Pause.)

BY MR. POLLARD: (Resuming)

Q If you did pursue the option of providing a much higher capacity high pressure injection pump, would not such a higher capacity pump also reduce the severity of the problem you talked about of having reactor coolant pumps run with a large number of voids in the system?

That is, if we have a high enough capacity pump, won't the magnitude of that problem also be reduced?

A (WITNESS JONES) Oh, it would be reduced somewhat,

but we have performed studies with one and two HPI pumps, for example, and the change in the management system void fraction on a specific case, on the order of 5 percent.

So, if I added, say, two more high pressure injection pumps and could guarantee that they would run, et cetera, which might solve this problem, because you will have excess refill capacity at that time, it might recover the core fast enough. You still may run to 85 percent void fractions in the primary system.

So, while they have reduced the severity somewhat, it still could be a problem.

BY MS. WEISS: (Resuming)

Q What is the basis for assurance that the reactor coolant pumps will be tripped within three minutes of the onset of a LOCA?

A (WITNESS BROUGHTON) Well, we have implemented procedures which require that, and we have trained operator to those procedures. The evolution is simple. It can be performed from the control room, and there are several different indications in the control room that would indicate that this action was required.

BY MR. POLLARD: (Resuming)

Q When you were answering a question, I believe, from the Board earlier, before the break, as I understood your testimony, you said that the training given to the

reactor operators is that any time that there is an engineered safety feature actuation system signal, that they should trip the pumps. Did I understand you correctly?

A (WITNESS BROUGHTON) That is correct.

Q I thought that he would only trip the pumps if high pressure injection had been actuated by low reactor coolant system pressure. But that he would not trip the pumps if high pressure injection had been actuated, for example, by high reactor building pressure.

A (WITNESS BROUGHTON) You were correct. I was not explicit enough when I made the remark earlier. The requirement for tripping the pumps, the requirement for tripping the reactor coolant pumps is when high pressure injection has been initiated.

Q Initiated automatically by specifically low reactor coolant system pressure?

A (WITNESS BROUGHTON) The 1600 pound low reactor coolant system pressure.

Q And for no other automatic initiation of high pressure injection?

A (WITNESS BROUGHTON) That is correct.

Q Now, again, going back to your earlier answer to us, what indication is available to the operator, I understood you to refer to a status panel for emergency core cooling system components. Wouldn't that status panel look

exactly -- excuse me. Wouldn't it look exactly the same whether high pressure injection was actuated by low pressure or by high building pressure?

A (WITNESS BROUGHTON) That status panel would --

Q So he cannot rely upon that as an indication of whether or not he should trip the reactor coolant pumps?

A (WITNESS BROUGHTON) I should retract what I said just a minute ago. There are several indications on that status panel, and I believe from the status panel that he can tell exactly what signal has caused the engineered safeguards feature to actuate. I am not familiar enough with that panel to say certainly that is the case.

Q Let me ask the direct question that I thought the Board has asked you originally, but before I ask that, let me just clarify it one more time.

It is your testimony that the operators are instructed to trip the reactor coolant pumps only when the high pressure injection system has been automatically initiated by low reactor coolant system pressure.

A (WITNESS BROUGHTON) Yes.

Q What indication or instrumentation does the operator have to make the decision or the determination that high pressure injection has been initiated automatically by low reactor coolant system pressure?

A (WITNESS BROUGHTON) I am aware of seven

indications that he has. I am not sure that I know exactly which one he uses to distinguish high pressure injection initiation from low system pressure from some other reason for initiating high pressure injection.

Q Perhaps we can turn to something else. Let's assume there is some instrument in the control room which directly indicates either reactor coolant system pressure or directly indicates the cause of initiation of high pressure injection.

You feel fairly certain there must be some kind of instrument like that, do you not?

A (WITNESS BROUGHTON) Yes.

Q Is it not true that that instrument would also indicate low pressure in the reactor cooling system if the event was not a LOCA but perhaps an overcooling event on the secondary side?

A (WITNESS BROUGHTON) Yes, that is true.

Q So that given your present instructions to the operator -- by present instructions, I mean at the time of TMI 1 restart -- the operator will trip the pumps without trying to determine whether it is a LOCA or, for example, any other kind of accident like an overcooling event?

A (WITNESS BROUGHTON) That is correct. He would trip the pumps and then try to diagnose the event. He would not try to make the diagnosis before tripping the pumps.

Q Then in your testimony, on Page 13, do I understand it correctly that eventually when you install the proposed reactor coolant pump trip system, that system will have the ability or hopefully will have the ability to distinguish between LOCA's versus overcooling transients, and it would not trip the pump, would not trip the reactor coolant pumps for secondary accidents?

A (WITNESS BROUGHTON) There are still some secondary overcooling events which would cause both a low reactor coolant system pressure and a low subcooling margin. In those cases, the reactor coolant pumps would be tripped automatically by the automatic system.

Q Let me ask it a different way.

After the installation of the automatic pump trip circuit, are there some accident scenarios for which the automatic circuit would not trip the reactor coolant pumps, but that before this is installed the reactor operator would trip the reactor coolant pumps?

A (WITNESS BROUGHTON) Yes.

Q Assume that we have an accident, a small break LOCA, if you will, which is causing the reactor coolant system pressure to decrease. The operator observes this pressure decrease, and determines that it will eventually get low enough to automatically start the high pressure injection system. But he decides to help out and manually

initiates high pressure injection.

Under those circumstances, where the high pressure injection was manually initiated, what are your instructions to the operator with respect to manually tripping the reactor coolant pumps?

A (WITNESS BROUGHTON) I would have to look at the specific procedure to indicate what those requirements are. My recollection is, he would still be tripping coolant pumps, but I am not sure if he would trip them because he started high pressure injectioner on some other signal, and I would have to look at procedures to be sure.

Q Do you have those procedures here with you now?

A (WITNESS BROUGHTON) I do not.

Q Which procedures are they?

A (WITNESS BROUGHTON) I believe it would be the small break, which does not cause an automatic depressurization.

Q Let me back up again. I thought before we got to this question, your testimony was unequivocal, that your instructions to the operator were to only trip the reactor coolant pumps if we had automatic initiation of high pressure injection only on low reactor coolant system pressure.

Are you now saying that you are not sure that that is correct?

A (WITNESS BROUGHTON) It would also be that he would trip for another scenario that doesn't include those indications that I mentioned previously.

MR. BOLLARD: Could we have the answer read back, please? It was too soft.

(Whereupon, the Reporter read back the previous answer.)

MS. WEISS: We think it important at this point in the proceeding that we know exactly what the procedure calls for. We don't have a copy of it. I wonder if you all have a copy that is relatively accessible.

MR. BAXTER: We do. I just wonder if it is worth breaking --

MS. WEISS: We can do it the next time the gentlemen are on the stand or we can do it tomorrow morning.

MR. BAXTER: We will attempt to retrieve it now while you are continuing.

(Pause.)

CHAIRMAN SMITH: I thought you were going to proceed.

MS. WEISS: It is taking us a minute to regroup.

CHAIRMAN SMITH: Well, in the meantime, Dr. Little has some questions.

DR. LITTLE: Mr. Broughton, on Page 13 of your testimony, the last sentence reads, "The NRC staff has

accepted this approach as described in NUREG-0680, and refers to C-21F," and on reading that, it states, "We agree in principle that the proposed method for pump trip in the event of an ESFAS meets the intended long term requirement of I&E Bulletin 79-05C," and it goes on to state that "Further review is going to be required of the proposed designs prior to installation," and so on.

Now, what is the correct status of acceptance by the staff of this approach?

WITNESS BROUGHTON: When we develop a specific design to implemant this trip, we would then submit that design to the NRC for additional review.

DR. LITTLE: So the approach is accepted in principle, but there is a long way from that to acceptance of Yet Ed's proposed way of dealing with it, the actual implementation.

WITNESS BROUGHTON: The actual implemenation would involve things like how many censors were used to detect the various conditions that were part of the trip, what would the logic be within the design to cause the coolant pump breakers to open, and those are additional things which we have not yet been able to define.

When those are defined, that would be presented to the staff for their review of how we actually implemented the approach.

DR. LITTLE: I guess that is a long-term requirement. Do you know right offhand when the schedule for that is? The schedule to have final designs in?

WITNESS BROUGHTON: I do not know when the final design is scheduled, no.

MS. WEISS: Is that it?

DR. LITTLE: Mr. Smith is looking something up. He has a further question on it.

(Pause.)

CHAIRMAN SMITH: It seems that 79-05C is a short-term requirement in the order. How does it happen to slip into long-term in the SER? If I am correct. Let me review it to see if I am correct.

It is the short-term action. Item Number 2 provides for NRC review and approval, so I can see the basis upon which the staff might conclude that it is long-term in their view.

MR. CUTCHIN: Are you referring now to the order itself?

CHAIRMAN SMITH: I am referring to the order and notice of hearing. Short-Term Item Number 2 states that "Licensees shall provide for NRC review and approval of all applicable actions specified in I&E Bulletins 79-05A, 79-05B, and 7905C." And 79-05C being the one which is referred to then as a long-term action, apparently, in the

SEP.

When I began the question, I was not sensitive to the use of the terms "review and approval" rather than "implementation," which I think could be an explanation.

MR. CUTCHIN: I don't have a ready explanation at my fingertips. When we get our turn on the stand, I am sure we can try to clear that up.

CHAIRMAN SMITH: Very good. Thank you.

(Pause.)

CHAIRMAN SMITH: Okay. I am surrounded by people pointing out the error of my ways here. 79-05C in itself is divided into short-term and long-term actions, too, and is referred to elsewhere, too, so disregard it.

MR. JORDAN: In looking at the staff's table on your long-term Item Number 1, the RC trip, they say the licensee has complied.

MR. CUTCHIN: You are looking at which table, now, in the restart, SEP, NUREG-0680?

DR. JORDAN: Yes, on Page B-7.

(Pause.)

CHAIRMAN SMITH: We are not seeking an answer now. Procee with the examination.

DR. LITTLE: We will probably get to this in questioning people later on, but if you look at Page B-3 in NUREG-0630, about two-thirdsof the way down the page, at the

paragraph just before conclusions, it says in the column, Comply Reference. "Comply" means that the licensee is in full compliance with that part of the order, and that is the designation that is given for the automatic RCP trip.

MR. CUTCHIN: I can only say, Dr. Little, that I did not choose these categorizations, and if the Board is interested in hearing an explanation, we will get the right body on the stand to explain it. But I personally cannot explain that.

DR. LITTLE: Yes, because right now, the further we go, the more confused we get as to the current status of it. We have got about three documents that say different things up here.

MR. CUTCHIN: I am not at all surprised, and we will assume a duty to try to clarify that for you.

We are shooting at constantly moving targets, unfortunately, and the dates seem to keep changing. I suspect when this NUREG-0737 that we referred to as superseding the September 5th Eisenhut letter comes about, that has, as I indicated the other day, been submitted to the printer, some more of these dates will change.

Hopefully, we will have the last word when we get that.

BY MS. WEISS: (Resuming)

Q With respect to your testimony on Page 14, on

Recommendation 2.3.2.B of NUREG-0565, first -- Well, the recommendation states that "The B&W small break LOCA analyses rely on equipment which has not previously been characterized as part of the reactor protection system or part of the engineered safety features, the equipment used to provide the necessary RCP trip, pressurizer PORV and PORV block valve and equipment used to actuate the PORV and PORV block valve fall into this category."

As I understand your testimony, you disagree with that. Is that correct? Or are you talking about different analyses?

A (WITNESS JONES) Well, I think the testimony states my position, that, you know, as far as some of the specific items mentioned there, we have not relied upon them, and the only items that we have used are the emergency feedwater system and the equipment used to provide the reactor coolant pump trip are the only items we have utilized in the analyses.

Q But specifically, you state that you have not relied on the power operated relief valve and the PORV block valves. Is that correct?

A (WITNESS JONES) That is correct.

Q What about the concern with limiting challenges to the safety valve that Mr. Broughton talked about earlier?

A (WITNESS JONES) We have not relied upon these

within the LOCA analyses. While the PORV may serve the function to limit challenges to the safety valves, as far as demonstrating that the core can be kept cool, we do not rely on the PORV or the PORV block valve.

Q I am tempted to talk some more about the operator procedures, but I think that is all on the record. I won't belabor it at this point.

A (WITNESS JONES) Well, again, the recommendation is specific to the LOCA analyses, and that is how it has been addressed.

Q Okay.

DR. JORDAN: Excuse me. When you say the equipment used to provide reactor coolant pump trip, are you referring now to automatic pump trip or what equipment?

WITNESS JONES: Basically, I was referring to the present equipment in the plant, the breakers, et cetera, whatever equipment is necessary to trip the pumps. You have to press a button which does something, as they say, something else, and which does something else, and it may be trips the pump. I don't know the actual mechanics.

DR. JORDAN: I see. Okay. I didn't know what you meant by equipment, whether it was eequipment that the operator relied on or whether it was equipment required to actually trip the pump itself. Okay. I understand.

BY MS. WEISS: (Resuming)

Q Your testimony on Page 16 covers Recommendation 2.6.2.A of NUREG-0565. And that recommendation is, "The various modes of two-phase natural circulation which are expected to play a significant role in plant response following a small break LOCA should be demonstrated experimentally. In addition, the staff requires that the licensee provide verification of their analysis models to predict two-phase natural circulation by comparison of the analytical model result to appropriate integral systems tests.

The recommendation calls for experimental demonstration of multiple and two-phase natural circulation, but your testimony does not mention any experiments. Do you plan to do any?

A (WITNESS JONES) Well, the testimony says that further work may be done in this area. At the present time, we have not developed any formal plans to examine the various roles of two-phased natural circulation and the ability of the codes to predict them, at least not formally. Such work may occur, however.

Q Mr. Broughton, do you intend to do any tests on Three Mile Island Unit 1 to demonstrate that two-phased natural circulation will work?

A (WITNESS BROUGHTON) To my knowledge, we will not be doing any such tests.

Q Let me refer you to Page 18 of your testimony, Page 18, Recommendation 2.6.2.B of NUREG-0565, which states that "Appropriate means, including additional instrumentation, if necessary, should be provide in the control room to facilitate checking whether natural circulation has been established."

Is it true that your testimony asserts that verification of natural circulation is derived from instrumentation showing, one, constant differential between system hot and cold leg temperatures, and two, the cold leg temperatures approaching secondary system saturation?

A (WITNESS BROUGHTON) Yes, that would allow you to determine that subcooled natural circulation was taking place.

Q And it is your testimony that if those conditions are present, natural circulation has been established?

A (WITNESS BROUGHTON) That is correct.

Q On Page 19 --

DR. JORDAN: This is one, by the way, that I have particularly marked for the staff's consideration.

BY MS. WEISS: (Resuming)

Q Page 19, Recommendation 2.6.2.C, calls for licensees to provide analysis which shows the plant response to a small break which is isolated and PORV fails to open upon repressurization of the reactor coolant system at the

PORV set point.

Is it you testimony here that under these circumstances, adequate core cooling would be achieved through bleed and feed? Is that what this says?

A (WITNESS JONES) I believe it would be provided by either bleed and feed or via the establishment of -- or with use of the steam generator, and to remove energy in the ECCS to provide the fluid to keep the core covered, irregardless of the scenario that you would be dealing with, I believe. It could be handled in either way.

Q Aren't you describing on Page 19 and 20 -- you say "Adequate core cooling would be continuously maintained for this transient by the fluid provided by HPI." Are you describing bleed and feed there?

A (WITNESS JONES) I am really not trying to describe bleed and feed at all. And in fact, I am describing this more like a transient in which the steam generator is continuously available, except what I have assumed is that you have a very small break LOCA. You get to an interruption in circulation. Then you isolate the break. Now you have lost both the break as a heat sync and the steam generator as a heat sync because the natural circulation path has been interrupted.

The system repressurizes as a result of the continued energy addition. The PORV opens. That will

result in liquid inventory loss and the probable re-establishment of boiler condenser in the steam generator, and that is your heat removal mechanism from the fluid in the primary system, but the actual heat removal from the core is provided by maintaining sufficient inventory via the HPI pumps.

Q But if the PORV is stuck open and the block valve also remains open, isn't it the case that you are losing fluid through that break?

A (WITNESS JONES) Oh, certainly. It is just that when I have been talking feed and bleed in the past, talking about it without the steam generator cooling being available, that is why I did not characterize it that way. But yes, you would have fluid being lost continuously through the PORV.

Q Recommendation 2.6.2.D of 0565, your testimony appears on Pages 21 and 22. That calls for the licensees to provide an analysis which shows the plant response to a small break in the pressurizer spray line with the failure of the spray isolation valve to close.

Have you done that analysis?

A (WITNESS JONES) No, I have not.

Q So your testimony on Pages 21 and 22 is with reference to results of B&W pre-accident LOCA analysis. Is that correct?

A (WITNESS JONES) The testimony is a general description of the plant behavior and what would be expected to occur during such a transient.

Q Do you intend to provide the analysis called for by the recommendation?

A (WITNESS JONES) At this time, again, we have no formal plans to perform this analysis.

Q Is it your testimony that no high pressure injection water will escape through the break in the pressurizer spray line without first entering the vessel?

A (WITNESS JONES) Yes.

Q Do you have Drawing 302-650 of the restart report? That appears in Volume 2, Section 9. I am going to get you with these drawings when you are real tired and can barely read them.

CHAIRMAN SMITH: Would you give us that citation again, please?

MS. WEISS: That is drawing 302-650, Volume 2, Section 9 of the restart report.

(Pause.)

BY MR. POLLARD: (Resuming)

Q By the way, this figure is also noted as Revision 18.

Let me understand your testimony. You are talking first about the analyses for TMI Unit 1 which show that less

than 70 percent of the high pressure injection was calculated to enter the core due to the direct bypass of the injected fluid at the break, which was assumed to be located in the bottom of the cold leg pump discharge piping between the HPI nozzle and the reactor vessel.

Just for illustration purposes, would that break, for example, be located -- If you look on Zone F-3, we have reactor coolant pump 1-B, and we show a pipe running from reactor coolant pump 1-A, entering the reactor vessel at Zone D-4.

So, is that correct, that you are assuming a break somewhere along that line?

A (WITNESS JONES) No, I was specifically assuming the break to be located -- This is not a very good drawing to point it out, but --

Q Metropolitan Edison provided it.

A (WITNESS JONES) This is a schematic. And it would be between the notation two and a half inch connecting up to a makeup pump in Zone F-3, F-4, right at that boundary. It was assumed to be located in the path from that point to the vessel.

Q And looking at this figure, I can visualize how water entering from the makeup pumps in that two and a half inch nozzle and on its way flowing to the vessel might go out the break. What I do not understand is your next

sentence, which says, "For a spray line break, no high pressure injection fluid would bypass out the break."

Now, as I understand from this diagram, the spray line is shown traveling from the discharge of reactor coolant pump 1-A in Zone F-3, it goes up, straight up the diagram through Zones E-3, D-3, and C-3, and then enters the top of the pressurizer.

Is that the spray line?

A (WITNESS JONES) Yes, it is.

Q Can you please explain, using this diagram, how a break on that spray line, that water entering from the high pressure injection pumps would first go through the reactor vessel without going out the break?

A (WITNESS JONES) Well, first off, you have to understand the process that occurs during the small break LOCA during the transient. Number One, you have a relief path from the vessel to the break, from steam being generated in the vessel via the vent valves, so that what actually occurs during the analysis is, the vent valves allow the steam to pass above the water surface while the water runs into the vessel.

That steam velocity is very low and it is insufficient to entrain any water, and that is the basic reason for the statement that none of the HPI water would be expected to go directly out the break without seeing the

reactor vessel.

And prior to that -- let me just -- I forgot to mention one other point. Prior to that point in time, where the valve -- the vent valves are open and passing steam, you would have circulation through the loops, you would have a mixing cap such that the HPI flow would tend to be preferentially swept into the vessel.

Q Am I to understand that what you are saying is that the reactor coolant pumps are either running or they are still coasting down?

A (WITNESS JONES) No, no. Initially, when the system remains in a subcooled state, you would still have a circulation path around the system, and this is even after the reactor coolant pumps are running. I mean, have tripped and coasted down.

Now, I am not sure the exact timing for a two and a half inch break, whether or not it would occur. What I am saying is, as long as the system is subcooled and you have natural circulation flow, you would have a circulation into the vessel through that path.

With the formation of steam in the vessel and the opening of the vent valves, then what you would get is a countercurrent flow of steam back toward the break, with the water preferentially draining into the vessel.

Q When you did this in computer analysis, did you

model this piping system for Three Mile Island Unit 1, or did you do a generic analysis?

A (WITNESS JONES) Well, as I stated, this is not a computer analysis that is in the testimony. It is based on the analyses which have been performed for the plant. It is a description of the expected plant behavior and what would be expected to occur. It is not a specific computer analysis.

Q If you haven't done a computer analysis, what is your basis for expecting that in the event of a break in the spray line, none of the high pressure injection water would go out the break without first going through the vessel?

A (WITNESS JONES) It is based on just the general comments I have just made, which are the results of analyses of other small break LOCA's. That is the specific phenomenon that occurs in that region of the system during a typical small break in the cold leg.

Q Let me just ask you one final question. If I direct your attention to Zone F-3 and 4, or perhaps it is only F-3, can you tell me the distance between where the spray line comes off reactor coolant pump 1-A discharge and where the makeup pump injection line enters the discharge piping of reactor coolant pump 1-A, the physical distance between those two points?

A (WITNESS JONES) I don't remember it. I just

don't know.

Q Thank you.

BY MS. WEISS: (Resuming)

Q Page 25 of your testimony deals with Recommendation 2.6.2.F of NUREG-0565. It calls for licensee to provide an anlysis of the possibility of impact director coolant pump seal damage and leakage due to loss of seal cooling on loss of off-site power. If damage can't be precluded, licensees are called upon to provide an analysis of the limiting small break LOCA with subsequent reactor coolant pump seal failure.

Have you reviewed past occurrences of pump seal failure?

A (WITNESS BROUGHTON) I have not personally done that, but I believe that is something that was done in preparing the analysis, that response to this.

Q So the analysis that you discuss in your testimony was not personally performed by you?

A (WITNESS BROUGHTON) No, that is correct.

Q Did you supervise its performance?

A (WITNESS BROUGHTON) No, I did not.

I have reviewed the results of it.

Q You simply reviewed the results of it?

A (WITNESS BROUGHTON) I have reviewed the results of the analysis, yes.

Q Do you know whether in any of the past occurrences of pump seal failure, what the cause was of the damage to the seal?

A (WITNESS BROUGHTON) For this particular event, where we are looking at the loss of cooling, it is expected that the mechanism would be damaged to a rubber seal in the pump.

Q My question referred to historical instances of pump seal failure and the causes of damage to the seal in past instances.

A (WITNESS BROUGHTON) I am not aware of all causes of seal failure in the past.

Q Are you aware of how long seal water had been lost before damage to the pumps occurred?

A (WITNESS BROUGHTON) If you are speaking about particular events other than loss of cooling water, I am not aware of those events.

Q Where does the seal water come from?

A (WITNESS BROUGHTON) The seal water is injected from the high pressure injection system.

Q And those are the same pumps as used for emergency core cooling systms?

A (WITNESS BROUGHTON) Yes, they are.

Q If engineered safeguards are actuated, is the seal water automatically shut off?

A (WITNESS BROUGHTON) I don't recall.

Q If you assume that loss of seal water occurs on actuation of engineered safeguards, with that loss of off-site power, so that the reactor coolant pumps continue to run, wouldn't damage to the reactor coolant pump seals be essentially instantaneous?

A (WITNESS BROUGHTON) The analysis done shows that it would probably be several minutes before the seal began to heat up and that it would take several minutes beyond that before the seal was damaged enough to increase the leakage.

Q What analysis are you talking about? Are you talking about the analysis contained in Mr. Reed's letter, which you discuss in your testimony?

A (WITNESS BROUGHTON) Yes, I am.

Q By several minutes, do you mean less than five minutes, less than ten minutes?

A (WITNESS BROUGHTON) I believe the analysis showed that within four to five minutes, the temperatures in the cavity around the seal would begin to increase, and it is not until the temperature begins the increase that then there is any possibility of damage. So at least that four to five minute period before any damage could be inflicted.

DR. JORDAN: While you are waiting, I would like a little clarification there. Are you saying that you have

not considered a small break LOCA which is initiated by the loss of a seal from the reactor coolant pump? Is this outside of the design basis?

A (WITNESS JONES) Well, specific analysis of a seal, a pump seal failure has not been directly considered. Generally speaking, they have fairly low leakage rates that can be handled by -- that would keep the system in solid condition continuously upon activation of the HPI.

DR. JORDAN: So you have no reason to believe that such a leak would lead to loss of inventory or something of that nature? Haven't you said essentially that the systems that have been provided will handle such a leak?

WITNESS JONES: Yes, I believe it would. I don't believe the core would ever become uncovered. It would be much less severe than any of the LOCA's we have analyzed specifically. And in fact, I do not believe that the system will ever ever become saturated.

DR. JORDAN: I see. Is this essentially what was said in the letter?

WITNESS JONES: I have not read that letter.

WITNESS BROUGHTON: Yes, the analysis goes on to indicate that it is estimated that at the end of 60 minutes the leakage out of the seal might be as much as 10 gallons per minute, but it is not expected to be any greater than that.

DR. JORDAN: But even if it were greater than that, wouldn't it still be within the capabilities of the system to handle it?

WITNESS BROUGHTON: Yes. The 10 gallon per minute leakage would be well within the capabilities of the normal makeup system to handle.

DR. JORDAN: I see. Thank you.

BY MS. WEISS: (Resuming)

Q Is it possible for a single failure to interrupt seal water flow to the reactor coolant pump? For example, failure of the seal water control valve

A (WITNESS BROUGHTON) I do not know.

Q I want to refer you to a diagram we looked at this morning, 302-661 in volume 2, chapter 9 of the restart report.

A (WITNESS BROUGHTON) Would you repeat the number of the figure again?

Q It is 302-661.

I would like you to review that diagram, please, to yourself, and see if you can, after looking at it, answer the question of whether it is possible for a single failure to interrupt the seal water flow to the reactor coolant pump.

(Witness inspects the document.)

A (WITNESS BROUGHTON) From this diagram, if MUV-20 were shut it would prevent seal water to the pumps.

Q Let me refer now to Recommendation 2.6.2.C of NUREG-0565, which appears on page 26 of your testimony. And you have also included some material from NUREG-0623. These require the licensees to require three test predictions of LOFT test with reactor coolant pumps running and verification of small break models with pumps running against interval systems experimental tests, in particular

the LOFT test.

Has any of this been done?

A (WITNESS BROUGHTON) The LOFT test L36 has not yet been run.

Q Wasn't the LOFT test originally scheduled for March of 1980, the LOFT test referred to?

A (WITNESS BROUGHTON) I'm not sure if it was L36, scheduled for March 1980.

Q Well, you say -- you quoted from 0523, and the last line indicates that that test was scheduled to be performed in March of 1980. You have no reason to disbelieve that, do you?

A (WITNESS BROUGHTON) There are several LOFT tests in the series, and I don't know which specific test had originally been scheduled for March 1980.

Q Well, in any case, they haven't taken place yet. Do you have any idea when they have been scheduled for?

A (WITNESS BROUGHTON) I believe LOFT 3-6 is in December. I believe LOFT 3-6 will be in December, but I don't follow the schedule of those tests closely enough to know for sure.

Q December 1980?

A (WITNESS BROUGHTON) December 1980.

Q And when will you provide your test predictions?

A (WITNESS BROUGHTON) I don't know what schedule

those are on. Those are being done for us by B&W, as I indicated.

Q Do you know, Mr. Jones?

A (WITNESS JONES) Yes. The LOFT test is scheduled to be run some time around the middle of December. The blind prediction will be provided around, I expect, the end of January.

Now, to understand what that means, because it doesn't sound like a pretest prediction, what it is is the models are to be created prior to the actual LOFT test. That is, the computer simulation model and the time zero edit of the model is to be provided to the staff roughly a week to two weeks ahead of the test.

The test will then be run. Engineers at EG&G in Idaho will review the tests, will assure that the test specifications given to each of the vendors were appropriate, that no abnormal occurrences occurred during the test or, if such did, like a system failure or something, they would tell us about it. They would give us the actual initial conditions of the experiment. We would modify the model, submit it to the staff as appropriate, based on this information, showing where the deviations are, the why's and wherefore's, run it, and then provide them with the prediction.

So it is basically a blind -- it is essentially a

pretest prediction except for getting the actual experimental conditions.

Q So you would make no changes in the model except to accommodate any unexpected conditions during that test itself?

A (WITNESS JONES) Unexpected conditions or the actual temperatures, et cetera, around the loop that they will get, that they will have during the experiment. Because we get a wide range that we are asked to set them all up for.

Q We have some confusion about Mr. Broughton's answer that he was not sure that the LOFT tests previously scheduled for March 1980, and particularly referred to in 0623, in the language that you quote on page 26, is the same LOFT test -- that is, test L-36 -- which is mentioned in 0565, and which is also mentioned in your testimony.

And our question is: Didn't you, in preparing the answer, have to determine whether these were all the same tests? And this would be to Mr. Broughton, because he prepared the answer. Didn't you in fact -- didn't you have to determine whether they were talking about the same test in order to make the question -- the answer responsive to the question?

A (WITNESS BROUGHTON) Yes, but the answer is primarily directed toward LOFT D-6.

Q The answer is directed entirely toward LOFT L 3-6; is that correct?

A (WITNESS BROUGHTON) Yes.

Q And it is still your testimony that you are not sure whether the tests referred to in the quoted section of 0623 is LOFT L 3-6?

A (WITNESS BROUGHTON) No, I'm not.

Q On page 27, recommendation 2.6.2.8, on page 27 of your testimony, direct licensees to provide certain information with regard to the effects of noncondensible gases during a small break LOCA. And you describe -- in the last sentence on page 27, you say: "While further examination of the effect of noncondensibles on the condensing heat transfer process within the steam generator may be performed, provisions are available at TMI-1 to assure adequate core cooling."

Is this another -- well, what provisions are you talking about? Are you referring to bleed and feed?

A (WITNESS JONES) Yes, assuming, as they stated above on that page, that the noncondensible gases somehow magically grew to a size which would totally prohibit condensing heat transfer.

Q And you have not provided the information called for specifically in the recommendation at items 1 and 2 of recomendation 2.6.2.8?

A (WITNESS JONES) No, we have not. We have not provided information to the staff on those two, the NRC, on those two items.

MS. WEISS: Thank you, gentlemen. Those are all the questions we have for you at this time.

MR. BAXTER: Mr. Broughton, do you have an answer to the previously posed question about procedures perhaps requiring tripping the reactor coolant pumps?

WITNESS BROUGHTON: Yes, I do. We were discussing the conditions under which an operator would trip reactor coolant pumps, and by some inconsistent use of terminology on my part I confused what the operators had been told to do through their training and their procedures.

I have reviewed the three particular procedures which apply to small break LOCA's that are large enough to cause automatic safeguard system actuation at 1600 pounds, ones which are too small to cause automatic actuation at 1600 pounds, and a more general procedure which covers operator actions following a trip of the reactor, in which it would not be expected to have pressure go below 1600 pounds, but which is possible.

And in those procedures the instructions to the operators are that when reactor coolant system pressure reaches the 1600 pound set point for high pressure injection that he is to manually trip the reactor coolant pump. So,

based on that instruction, if the pressure drops to 1600 pounds and automatically starts high pressure injection, he would manually trip reactor coolant pump or, if he has already restarted high pressure injection but the system pressure continues to drop and goes below 1600 pounds, he would also manually trip reactor coolant pumps.

BY MS. WEISS: (Resuming)

Q Can you tell us the numbers of the procedures you are looking at, and also the revision number?

A (WITNESS BROUGHTON) Yes. The procedure which deals with automatic initiation of high pressure injection is 1202-6(b), Revision 4. The procedure which deals with a leak in the reactor coolant system which does not result in automatic high pressure injection initiation is 1202-6(a), Revision 4. And the procedure which applies following reactor trip is 1202-4, Revision 17.

Q Are these all emergency procedures? Are they all prefixed "EP"?

A (WITNESS BROUGHTON) They are all emergency procedures, with the prefix "EP."

MS. WEISS: We would like to take a look at those, either now or overnight. We have 6(b), but I don't think we have either of the other two.

CHAIRMAN SMITH: Why don't you do it overnight, and we can proceed, then, with the examination.

Mr. Cutchin, do you have any questions?

MR. CUTCHIN: I have no questions, sir.

CHAIRMAN SMITH: Mr. Robert Adler?

MR. ROBERT ADLER: No questions.

CHAIRMAN SMITH: Dr. Jordan has only a very few questions. We could possibly wind up tonight if you could look at those procedures and then Mr. Broughton could be excused, as can Mr. Jones.

MS. WEISS: I also forgot something. I am very sorry. We want to move 0565 into evidence. It is my understanding that nobody else intends to do so, and we would like to get it on the record at this point. I think it is probably a matter of sufficient notice, but I would like to get it in at this point.

MR. BAXTER: I don't think it is a matter of sufficient notice. We don't have a sponsoring witness. And among other things, I would like to review the document with the potential offer in mind. I am not prepared to offer objection at this point as to its relevancy.

CHAIRMAN SMITH: Mr. Cutchin?

MR. CUTCHIN: I'm not even sure that the staff wants to put it in evidence. Mr. Chairman, I believe if there are problems she can mark it as an exhibit for purposes of examination. But I don't even have a sponsoring witness here to support that document as representative of

anything at the moment.

CHAIRMAN SMITH: Well, I am convinced that one way or the other Ms. Weiss can get at least appropriate parts of it into evidence. And the question may be how hard it is going to be for her to do that.

MS. WEISS: Well, the Board did ask specific questions on it, and all the testimony that we went over today was specifically referenced to the references in 0565. I cannot imagine a circumstance where it could be successfully argued that the document is irrelevant.

We've had Mr. Jensen on, who states that he was one of the authors of the document. If they want to get a principal author here, that's fine.

CHAIRMAN SMITH: There's going to have to be an accommodation to the Union of Concerned Scientists on it, and it can be reasonably operated. 0565 so pervades the issue, pervades the testimony and is relied upon, or distinguished, shall we say, so frequently; and then the Board question I believe specifically referenced the document.

However, there is a problem with just receiving it into evidence without in the first place limiting it to the portions upon which you want to rely, and, second, giving the Licensee in particular an opportunity to address the recommendations, the merits of the recommendations, which

you have pretty much done anyway.

MS. WEISS: That was the testimony.

CHAIRMAN SMITH: That was the thrust of the testimony. However --

MR. BAXTER: Mr. Chairman, our testimony does not rely on the document at all. We have addressed it in response to the Board question because we were asked to. But there are chapters in there which discuss bases for some of the recommendations which I cannot attest have or haven't been addressed, and the implications of citing those in proposed findings in the abstract I'm just not prepared to address today.

CHAIRMAN SMITH: When I say they have had an opportunity to address it, they have had an opportunity to address it. Accepting the report as it is, which we may not wish to do, if it's going to be offered into evidence -- well, let's consider that problem overnight, and maybe tomorrow something will occur to us that will satisfy your needs.

MR. CUTCHIN: Mr. Chairman, maybe I can help. I'm not sure whether it will help or hinder. But if Ms. Weiss wishes to put the document in for the purpose of evidencing the truth of the recommendations made by the staff, I have no problem. But if it is for something more, I would have to hear for what purpose she's offering it before I know

whether I would have an objection.

Clearly those are staff recommendations. Our witnesses tomorrow will address the staff's interpretation of those recommendations and give its position as to the acceptability of the Licensee's comments on those recommendations. But I am not sure for what purpose she wishes to move it into evidence.

MS. WEISS: I am not sure I understand the nature of --

CHAIRMAN SMITH: Well, I think that is pretty clear. If you're going to offer it solely as evidence that the staff has made such recommendations, you'll have no problem. But if you're going to try to attach to the recommendations a presumptive quality that those recommendations have merit, which I think is the whole idea of your testimony, then you're going to have some problems.

MS. WEISS: Well, you know, the recommendations have been interwoven throughout the testimony, throughout the questioning that we've had of these two witnesses, and indeed with Mr. Jensen, and the document was specifically the subject for the Board questions. And we are interested in, as our questions indicated, not only what are the recommendations, but is the purpose behind them and what is the basis for them.

Now, I can understand a problem with parties being

afraid that we are going to cite at some later date portions of the document which we have not gone over. It is not our intention to do that.

CHAIRMAN SMITH: That is a separate problem.

MS. WEISS: Yes, that is a separate problem. But we do intend to rely on it as a substantive matter, and I think that --

CHAIRMAN SMITH: And you will probably be filing proposed findings that, to the extent that the Licensee fails to comply with those recommendations, somehow it is not doing what it should do. And this is exactly what Mr. Baxter's point is, and he has a full opportunity to, not only to, as Licensee has done, address the recommendations in this manner, but even address it that it is going to be offered for that purpose, the basis for the recommendation in the first instance.

MS. WEISS: Well, he can. They've got a heck of a better opportunity to address it than we do, through the direct witnesses. And I thought that was the purpose of this testimony. We did not question on any sections of this that were not directly within the scope of the direct testimony today.

So I don't know whether he is anticipating rebuttal or what the problem is. I don't think we have expanded the scope at all with the references that we've

made to 0565 thus far.

CHAIRMAN SMITH: Well, you have the essential problem. Well, Mr. Baxter, you offer your objections to it. I don't want to -- maybe I'm stating objections you don't really have.

MR. BAXTER: Well, we have been asked questions, to address recommendations made in the document. We have taken those we thought applied to Licensee and we have given a status report, if you will, on the way they stand.

The document is a broad one. It's got a lengthy chapter, I think chapter 4, of 60-some pages which discuss a lot of experience at higher loop plants of B&W, at some CE plants. I just don't think, unless we have some further elucidation of what the purpose of the offer is, that I could agree to put that in.

If I were going to say that I was going to have to come back and rebut, I'm not sure what I'd have to rebut at this point.

MS. WEISS: Well, I understand him to have stated the objection that we just discussed, that he is afraid that we are going to cite parts lurking in here that we haven't asked questions about. And we do not intend to so do.

MR. BAXTER: Well, the recommendations are quoted in the testimony.

CHAIRMAN SMITH: Has there been a point of

discussion of the main reason that the document will be cited by the Union of Concerned Scientists, and that is that there is a presumption that the staff recommendations are necessary within the scope of this proceeding, and any failure to comply with them has to be met by a preponderance of the evidence by the Licensee? That would be the effect of it, and the way she's trying to offer it, she's trying to offer it, as I understand it, that whatever the staff is recommending in the document is entitled to a presumption that those actions are necessary for the restart of the unit, which would change the burden to the Licensee to establish either that they have agree and complied or they have disagreed.

MR. BAXTER: And I don't think we have had any witness that testified that those recommendations have that status.

CHAIRMAN SMITH: That is exactly right.

MR. BAXTER: In fact, some of them may have been abandoned by the staff.

CHAIRMAN SMITH: That's right.

DR. JORDAN: My question -- I was the one responsible for getting this in here. And the presumption was indeed that this was a staff document. And my question therefore not only went to the Licensee, but to the staff. And I will be asking the staff particularly now, having

heard the Licensee's testimony, having heard that in some cases the Licensee does not plan to meet the requirements, does the staff really believe the requiremets are necessary requirements, do they plan to enforce them, and if so when?

And so I believe, therefore, since I started it, that was the reason for it.

CHAIRMAN SMITH: After that the problem will go away as far as the exhibit is concerned.

MR. CUTCHIN: We understand that to be Dr. Jordan's concern, and it is our intent tomorrow to have someone on the stand who can address particularly each of the NUREG-0565 recommendations related somehow to either a NUREG-0660 or a recommendation, and then an 0660 implementation requirement or NUREG-0737, which as I said before is the official version of the Eisenhut September 15th letter.

If and only if there is some implementation requirement laid on the Licensee by one of these mechanisms, you will see, does the staff consider it something that is required, and then it may be pre-restart, it may be post-restart, that the implementation is required. And we will be prepared to address that item by item tomorrow.

CHAIRMAN SMITH: Well, the problem still remains about Mrs. Weiss' request that it be marked for identification. I think that you can do whatever you wish,

mark it and offer it and we can rule. But wouldn't it be better for you to come back to this document after the staff has addressed the issues in there, and then argue that you've had your foundation for it?

MS. WEISS: That's fine. I still understand Mr. Cutchin to say that they'll raise the same objection. But we'll argue it in that context. There's no reason to get into it today instead of tomorrow.

I have a question about what he just said. What piece of direct testimony is it that you were referring to that goes to each of these recommendations one by one? If I may direct that question.

MR. CUTCHIN: I'll be handing out a chart tomorrow. I'd be happy to hand it out to the parties tonight so they can study it in advance. But that is a document that we have put together since we had the discussion last Friday, for the purpose of putting on a live witness who can walk the Board and the parties through that document and key it to various recommendations between 0565, 0660, 0737, and whatever is appropriate.

But it is a key, if you will. I'd be happy to pass it out tonight and we can decide tomorrow what we label it as.

DR. JORDAN: Don't you have a key in some of the documents you submitted last week?

MR. CUTCHIN: We have, in response to Board Question 2, a key between 0660 and I believe it is 0694 documents. Now, we also have another character, and that is 0737, which is the Eisenhut letter transformed. It gets more confusing as we go along.

CHAIRMAN SMITH: Well, I think you had better provide the key to the parties.

MR. CUTCHIN: I'll hand it out tonight and we'll decide what we do with it tomorrow, or how we label it and so on.

MS. WEISS: Is that it for the surprise witnesses or do you have anything else up your sleeve?

CHAIRMAN SMITH: Anything further? Now, we were considering the possibility of finishing with these witnesses. Have you had an opportunity to review the operating procedures?

If it is not realistic to conclude with these witnesses, let's not try. It's after 6:00, but Dr. Jordan doesn't have much to go. No one else has examination. And you have a vague, narrow area on these operating procedures.

MS. WEISS: If you give us about five minutes to look this over.

CHAIRMAN SMITH: Is that what you wish? It's up to you. Do you want to try to excuse the witnesses this evening?

MR. BAXTER: Well, frankly, they'd like to be, but they'll both be here tomorrow.

CHAIRMAN SMITH: Anyway? That would really be helpful if they can.

MR. BAXTER: If we could, though, I would like to proceed and finish with them at 9:00 o'clock instead of our previous arrangement, because it looks like we're within close range, and it keeps the record together.

CHAIRMAN SMITH: All right, that's good.

Is there anything further, then? All right, we will adjourn and meet tomorrow at 9:00 a.m.

Let the record show that the chart referred to by Mr. Cutchin has been distributed to the Board, at least, and the parties I can see.

MR. CUTCHIN: And I understand, Mr. Chairman, that it is substantively correct. There may be a couple of minor modifications in the morning, in which case we will hand out the as-corrected copy for use.

(Whereupon, at 6:04 p.m., the hearing was adjourned, to reconvene at 9:00 a.m. on Tuesday, November 12, 1980.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

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in the matter of: METROPOLITAN EDISON COMPANY (TMI UNIT 1)

Date of Proceeding: November 11, 1980

Docket Number: 50-289

Place of Proceeding: Harrisburg, Pa.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Alfred H. Ward

Official Reporter (Typed)

Alfred H. Ward

Official Reporter (Signature)