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UNITED STATES

NUCLEAR REGULATORY COMMISSION

In the matter of:

EVIDENTIARY HEARING AT SACRAMENTO

MUNICIPAL UTILITY DIESTRICT

(RANCHO SECO) DOCKET NO. 50-312

Place: Sacrament, California

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Conference Room W-1140

U.S. Federal Building

2800 Cottage Way

Sacramento, California

Thursday, March 6, 1980

The Atomic Safety and Licensing Board met, pursuant to recess, at 9:00 o'clock a.m., the Honorable Elizabeth Bowers, Chairman of the Board, presiding.

PRESENT:

ELIZABETH BOWERS, Chairman

MR. FREDERICK J. SHON, Member

DR. RICHARD F. COLE, Member

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PROCEEDING

MRS. BOWERS: I think we can proceed with Mr. Ellison.

BY MR. ELLISON:

Q Mr. Karrasch, I would like you to refer again to CEC 15. Do you have CEC 15?

A (Mr. Jones) Yes, we do.

Q Referring again to figures A & B three, and A & B four on Page AB 36 and 37 those figures compare the results of the first two minutes of using the MINI-TRAP model versus the MAXI-TRAP model. Can you tell me whether or not in using the MINI-TRAP model, beyond the two-minute period, you used any correction factors to take into account the apparent difference between the MINI-TRAP model and the MAXI-TRAP model?

A (Mr. Jones) No, we didn't.

Q Can these models either the MINI-TRAP or the MAXI-TRAP predict natural circulation?

A Yes, they can.

Q Can they predict it in voided conditions?

A Yes, they can.

Q Can they predict reflux boiling?

A I am not sure whether the trap models can or cannot.

Q The trap models are different than the CRAFT-2 codes; is that correct?

A Yes.

Q Could you briefly explain the significant differences between those models?

MR. BAXTER: Mrs. Bowers, I believe that question as asked and answered yesterday afternoon. That is my recollection that the question was asked about the significant differences between these two codes.

MR. ELLISON: If my recollection is correct, I asked yesterday about the significant differences between the mini-trap and the maxi-trap models and not about the difference between the trap models of the CRAFT-2 model.

MRS. BOWERS: Maybe the witnesses can recall.

MR. JONES: Well, my understanding of the differences between the two codes are basically built around the transients they are designed to analyze.

The trap code is designed to analyze secondary system breaks, and it has much more detailed representation of the steam generator heat transfer phenomena for events such as steam line breaks.

It has the capability of modeling the turbine and steam system components.

The CAFT code has been designed to meet the requirements of Appendix K which is for LOCA evaluations and it includes effects such as clad rupture and swelling models, items such as that.

It does not have a secondary, a detailed secondary steam side representation.

Those are the kind of differences that exist between the two codes.

BY MR. ELLISON:

Q Does the CRAFT-2 code represent in more detail the primary system than the trap codes?

A I am not sure what you mean by detail. Each code is capable of predicting or modeling the primary side detail within some noting restrictions of the coes, but I am not sure what they are for the trap code. The basic equations that are in the codes are conservation of energy mass and momentum and they will be the same.

The unique differences come about from items such as the cladding swell and rupture models and being specifically oriented towards a LOCA code, bypass models for core flood tank injections, things like that.

The basic equations, the transient flow equations, my understanding is that they are the same.

Q Now, you mentioned that there may be noting restriction differences and if I understood you correctly, you are not sure what the differences between the codes are, in that respect; is that correct?

A That is correct.

Q Can the trap code account for the presence of non-condensibles?

A No, it cannot.

Q Mr. Karrasch, I would like you to refer to Page 14 of your testimony. In the sequence of events that is listed there for a loss of main feedwater transient, is there any operator action assumed?

A (Mr. Karrasch) No, sir, there is not.

Q Would you refer to Page 46. In Line 4 and continues through Line 7 you refer to corrective action being initiated and then you go on to describe what it is.

Both 1 and 2 would be operator actions; is that correct?

A (Mr. Jones) The section that that is in is the overheating transient, where what we are considering in describing these actions which are operator actions is a failure of both auxiliary feedwater trends to come into play.

Q Could you refer to Table 1 of Page 58 and also to Table 2 on the opposite Page. In comparing these tables, for that matter, Tables 3 and Tables 4, I noticed that in some instances you assume a core decay heat rate of 1.0 times the ANS standard value and in other instances you assumed the 1.2.

Could you explain why you assumed different values for the different tables?

A (Mr. Jones) The analyses that are described in Tables 1 through 4 -- let me go through them one at a time, I think that will be easier.

Table 1 discusses a case where we have a loss of all feedwater to the steam generators and there is no small break LOCA present.

This is a highly degraded condition where you do not have any heat removal from the steam generator for some extended period of time.

Which means the automatic function which actuates the aux feed has failed, for both trains.

In doing this analysis, we have also assumed the single failure on top of it within the high pressure injection system, and what we have decided to do when we did the analysis was to take a look at a more realistic approach to the primary system hydrodynamics.

We just took out the normal conservatisms associated with the ANS decay heat curve. So we went to 1.0.

The second Table again assumes -- this one has a small break LOCA and again all feedwater is lost.

We assume both high pressure injection trains function and we use the normal Appendix K assumption as far as the ANS standard value, standard decay heat value.

What we were doing here was basically Appendix K assumptions where we would taking a single failure and in fact a multiple failure for the Rancho Seco design, which would result in all feedwater being lost, rather than a failure in the HPI train. So we used the normal Appendix K assumption of 1.2 ANS.

Table 3 is basically the same type of analysis; loss of mainfeedwater with the PORV failure. Here we have used 1.2 times the ANS and Appendix K evaluation of this transient.

We do have auxiliar feedwater actuated and we have taken a single failure in the HPI system.

Similarly, in Table 4, we have again assumed all feedwaters failed with this PORV failure and on top of that the additional single failure of the HPI system, so we used a more realistic value of decay heat, the 1.0 times the standard.

Q With respect to Table 1, do you know whether it would have changed the results which you summarized there if you had used 1.2 times the ANS value?

A (Mr. Jones) Yes, it would have changed the results.

Q Can you explain how?

A (Mr. Jones) Using 1.2 times the ANS decay heat value, I am not sure that we could demonstrate that we could keep the core cool with one high pressure injection pump and the failures of all the main and auxiliary feedwater systems on top of it.

We would get some partial core uncovery, we have not evaluated whether it would be acceptable or not.

Again 1.2 times the ANS standard of decay heat value is a very conservative assessment.

The present data on decay heat indicates that the 1972 or 71 ANS standard, decay heat curve more -- it had reasonably predicts that today's experiment and that the known multiplier or conservative factor is necessary because the curve fits the data.

Q What data are you referring to?

A There has been a whole series of data and I really can't pick them all out.

There is some data at Los Alamos, they have some data from Shore from England and I am sure there is various other data, I think Oakridge did some, I am not really sure in the literature; and what they show is that 1971 when the ANS decay heat curve was generated, their assessment at that time was that there was an areoband of approximately of 20% on the decay heat curve.

The data that we have obtained that has been obtained since 1971/1972 shows that it is much more accurate and demonstrates that the actual best estimate curve has that the ANS has in the standard fits or is conservative to the decay heat data -- the new decay heat data.

Q Do you know of any systematic studies the data that you are referring to the assumptions in the ANS standard value?

A I believe I have seen one but I cannot tell you where offhand.

Q With respect to Table 2, would it change the results that are summarized there if you were to assume a single failure in the high pressure injection system?

A All of the things in the analysis being equal, it would change the results. Again, you still have the conservative decay heat value being utilized and we have used a very conservative actuation setpoint for the

HPI system relative to the Rancho Seco design. We used an actuation signal of 1350 psi, wherein the Rancho Seco ESFAS signal was 1600 psi and what happens in this transient is we miss, we do not depressurize to the emergency core cooling setpoint of 1350 in the first few minutes prior to the generator dry out.

If you go back to the realistic assumption on the ESFAS actuation, the results may be tolerable with one HPI.

Q You use the words may be tolerable, have you looked at that case?

A No, we have not.

Q With respect to the difference in the high pressure injection system setpoints, could you explain why you did not use a more realistic setpoint?

A When we performed small break loss of coolant accidents, we generally do a generic calculation for all plants; that is we take the highest power level, the lowest ECCS setpoint, the lowest volume HPI pump capacity and items such as this to bound the consequences for any of our 177 lower loop plants.

Some of the lower loop plants which have a lower power rating have a lower ECCS setpoint and we are using that setpoint plus putting 6% conservative factor for instrument error due to degraded environment. The actual, or our estimate today of what that degraded environment would due to the setpoint is only a 2% reduction.

Q In this analysis and for that matter, the other analyses when you assume, let me clarify, when I say the other analyses, I mean the other analyses that assume high pressure injection, automatic actuation in your testimony, do you assume that the high pressure injection system begins injecting at its full capacity immediately upon reaching the setpoint?

A No, we do not. We have got the delay time associated with opening valves and items such as that in starting up the pump.

Q So, the difference in the setpoint you are discussing here was not intended to account for a delay time in actuation?

A No, it was not.

Q With respect to Table 4, can you tell us whether the results that are summarized there would be different had you used 1.2 times the ANS standard value?

A In that case it would have been different in that the high pressure injection, one high pressure injection pump may not have been capable again of handeling the accident by itself. However, the analysis that we have done one with 1.2 times the ANS standard, with one HPI and that analysis shows that the operator would have better than 30 minutes to re-establish feed-water to the steam generator and still have acceptable conseuences.

Q Were the assumptions in the second analysis that you mentioned, the one that uses 1.2 times the ANS value, aside from the change in the ANS value, were the assumptions the same?

A To the best of my knowledge they were.

Q With respect to Table with the results that are summarized there change had you assumed a single failure in the high pressure injection system?

A No, they would not have.

Q With respect to Table 2, did you assume operator action?

A As listed in the summary of results for breaks greater than .01 square feet, no operator action was required and none was assumed in the analysis.

In the second point, starting on Line 17, two operator actions were analyzed, these included, these were either the operator initiating auxiliary feedwater only, or the operator initiating the high pressure injection.

Q With respect to Table 3, did you assume any operator action?

A No, we did not.

Q Did you assume operator action with respect to Table 4?

A No, we did not.

Q Did you assume operator action with respect to Table 5?

A No, we did not.

Q And with respect to Table 6?

A No, no operator action. On Table 6, let me back up on that one, we did examine the cases where the operator manually trips the high pressure injection pumps -- I mean the reactor coolant pumps, promptly upon ESFAS.

Q Could you refer to Page 45 of your testimony, Line 11.

The two sentences that follow Line 11 suggest that analyzing an over coolant transient caused by an overfeed of main feedwater bounds a similar transient assuming an excess auxiliary feedwater addition, could you explain why that is so?

A The main feedwater overfill case was analyzed because that system has the capability of removing full core power. The auxiliary feedwater system is capable of only removing the decay heat power roughly 6%. So there is a lot large mass of cold water being injected into the generator at any given time. Its rate is much larger so you get a faster cooldown out of that case.

Q Would that be the same as saying that main feedwater flow is higher than auxiliary feedwater flow?

A Yes.

MR. SHON: Excuse me, Mr. Ellison, there is one thing that I am a little confused about now. As I understand it, the auxiliary feedwater flow is a good deal colder than the main feedwater flow and although its volume is less, the total enthalpy extract could be more. I don't know how they are proportion.

Further, for a transient non-steady state situation, the auxiliary feedwater is being introduced into a portion of the heat exchanger or steam generator that is much, much hotter and you might expect momentarily much, much larger heat transfer.

How do these two factors influence your answer?

MR. JONES: Well, when I answer the question the first, time, I was accounting for not only the rate but the temperature contribution, there is a difference in temperature.

The main feedwater is only capable, is at approximately 450°, the aux feedwater is at 90°, but there is a much larger difference in the flow rate that easily compensates for that energy difference, temperature difference.

If you want to raise it to saturation, to say a 1000°, there is much more capability of storing energy in the main feedwater versus the aux feedwater system.

As far as the introduction or the place that the feedwater is introduced, that makes very little difference in the over cooling event wh... you are analyzing it with the analysis that we have done which has the reactor coolant pumps on.

You have direct coupling and you are taking out all the heat you can, basically, and where it is coming out is not that significant.

MR. SHON: Thank you.

BY MR. ELLISON:

Q Have you done an analysis with the same assumptions that you used in analyzing excess main feedwater transients that assume an excess auxiliary feedwater transient?

A Could I have that question read back?

(Whereupon the Reporter read back the question)

A We have not analyzed specifically, at least to the best of my knowledge, an excess auxiliary feedwater flow transient but basically for the reasons that I just gave, it is a much less severe transient than an access of main feedwater overfill.

Q Would the TRAP code either of the TRAP codes account for the differences mentioned by Mr. Shon?

A Yes, it would be capable of handeling that difference.

Q Can you tell me whether Babcock & Wilcox performed the small break LOCA analysis that was considered in the licensing of Rancho Seco?

A Yes, we did.

Q Are you familiar with that analysis?

A It depends on how far back you are going.

Q Let me ask you the question and see if you can answer.

Do you recall in that analyses whether reflux boiling was considered?

A Which analyses -- small break LOCA analyses have been performed several times over the year for the

Rancho Seco facilities and I am just not sure which one you are talking about.

There has been additional small break LOCA analyses performed while the plant has been operating.

So, I am really at a great loss to try to understand which one you are looking for.

Q Okay. Do you have a copy of CEC 1 and CEC 10?

A No, I do not.

Q With respect to CEC 1, it request for admission Number 50 as the Licensee to admit that natural condensation cooling which, for the record, was the term we used to the equivalent to reflux boiling. Now whether natural condensation cooling -- that natural condensation cooling was not considered during the licensing of Rancho Seco as a method of core cooling, the Licensee denied that request of admission and in CEC 101 at Page 12, answer Number 30-50, the Licensee explained that natural condensation cooling was assumed in a small break loss of coolant accident analysis performed in support of the licensing of Rancho Seco.

Can you shed any light on how it was assumed in the small break LOCA analysis?

A If we were analyzing a small enough break size, which would have been utilizing the steam generator for heat removal, then during the evolution of the transient you would have gone into natural condensation cooling.

The analyses which have been performed which demonstrate compliance to 10 CFR 50.46 do include such cases.

Q Mr. Jones, I am going to show you a copy of a document that has a cover letter addressed to Mr. R. J. Rodgriguez, and the cover letter is from the Babcock & Wilcox company signed by Mr. Janis. This letter is dated September 5, 1979 and transmits parts one and two of the latest revision of the small break operating guide.

I would like this document identified as CEC 18.

MR. LEWIS: Do the other parties have it, Mr. Ellison?

MR. ELLISON: Yes. I am giving them out now.

(Whereupon CEC 18 was marked for identification.)

Once again, for the record, I will note that the stamp and some of the writing on the upper right hand corner was added by California Energy Commission as was the underlined word, copy, at the top of the page.

BY MR. ELLISON:

Q Mr. Karrasch, do you recognize this document?

A (Mr. Karrasch) Yes, we do.

Q Are these the operating guidelines that we have referred to particularly with respect to raising the secondary side of the steam generator level to 95%?

A (Mr. Jones) 'Yes, they are.

Q And during your cross examination and in your testimony when you referred to Babcock & Wilcox operating guidelines, is this the document that you are referring to?

A For the small break LOCA guidelines, yes, this is one of them.

MR. LEWIS: Mr. Ellison, I would like to bring to your attention, for the record, and I had no idea what the changes are between the September 5, 1979 operating guidelines and the other one but there is a November, 1979 small break operating guidelines which incorporates inadequate core cooling and rather than let you go very far down this road, perhaps we could show that to you and that might assist you in getting a sounder record.

MR. ELLISON: Let me show this to the witness.

(Whereupon the witnesses were shown a copy of the document)

MRS. BOWERS: We have a business matter to take care of that will take a few minutes, perhaps this would be a good time to take a recess, I know it is a little early, but this has to be done now, or I go to jail.

(Whereupon a recess was taken at 9:40)

(Whereupon the proceeding came to order at 10:00)

MRS. BOWERS: Are you ready? Am I correct that there is just one copy of that November analysis?

MR. ELLISON: Well, we have two, apparently I have one that the Staff has given me and I have given the other one to the witness.

BY MR. ELLISON:

Q Mr. Karrasch, you have two sets of B&W operating guidelines before you, one dated September 5, 1979 that we identified as CEC 18 and the other that the Staff has just furnished me dated November. Can you tell us which of these two documents are the operating guidelines that you have been -- assuming one or the other are the operating guidelines that you have been referring to; could you identify which one?

A (Mr. Jones) The general reference we have been making has been to the November set of guidelines the most recent ones. But, the guidelines given in CEC 18 are basically the same guidelines and would not have affected any of my answers.

The big difference between the two, besides some editorial changes of which I am not sure which ones they were is basically the addition of the inadequate core cooling guidelines within the package -- excuse me I would like to make a statement to clarify something yesterday.

Yesterday, I was handed a document, CEC 17, by Mr. Ellison, in reviewing the document last night that document is not totally correct, in the summary of results section, about three or four pages in, there is are results from the 12.2 square foot double ended steam line break which lists there is 12 cubic feet of steam in the primary loops.

That is not the correct value for that analysis, it is something more. My understanding is approximately

100 cubic feet of steam and was a result of an error in the calculations which were originally performed at this time.

The remainder of the graphs and conclusions are the same.

We did not run a new analysis of the matter of converting the results to come out with how many cubic feet of steam we had and my recollection is that the wrong density was utilized in the conversion.

So I did want to clarify that one item in the document. Excuse me, Mr. Ellison for interrupting.

MR. ELLISON: That is fine.

BY MR. ELLISON:

Q Let me ask you just a couple of quick follow up questions with respect to CEC 17.

Is there a subsequent analysis, or is CEC 17 with the amendment you just made to your knowledge, the most up-to-date analysis of this type of transient, or accident, pardon me?

A I am not sure whether there is another document that exists or whether it was just a letter saying change the numbers to 100 cubic feet or not.

I know there was no other analysis that was done for SMUD that invalidates these results that are presented, t- the best of my knowledge.

Q So, to the best of your knowledge, the results with the amendment that you just made and are presented in CEC 17 are correct?

A To the best of my knowledge.

Q Returning to the operating guidelines -- are the November operating guidelines that include the core cooling information the most up-to-date B&W operating guidelines for Rancho Seco facility?

A (Mr. Karrasch) With respect to the issues addressed in these guidelines, they are. These small break guidelines first address the overall actions that the operator should take to recognize and recover from the small break LOCA.

In addition they go on following directions from NUREG 0578 to give the operator the information he needs to detect and recover from a condition of inadequate core cooling.

Following the publication of these guidelines in November, follow on work has also been performed in response to several other points in NUREG 0578 with respect to inadequate core cooling.

Q Mrs. Bowers, I have to apologize for not having the November operating guidelines. Unlike the Staff, SMUD is, to my knowledge, not furnishing us with everything that they send to the NRC, or with the latest additions items like this in order to have an accurate and complete record, I think I agree with Mr. Lewis, I think it probably would be better for us to refer to the latest guidelines.

So, if I may, my cross examination of these witnesses is about to conclude, that if I may I would like to reserve the right to have copies of the November document made and identified for the record; rather than CEC 18.

MRS. BOWERS: That will be fine.

BY MR. ELLISON:

Q With respect to the November document, Mr. Jones, at Page 19, find a pressure temperature limit curve labeled Figure 1.

Have you found that?

A (Mr. Jones) Yes.

Q Yesterday, we were referring to operating guidelines for protection of vessel integrity; do you recall that discussion?

A Yes.

Q Were these the guidelines that you were referring to at that time?

A These are the same generic type of guidelines I was talking about, yes.

Q Are you aware of any subsequent analysis by B&W that would have changed the information that is presented as Figure 1?

A Not to my knowledge.

Q That is all I have with respect to that document.

Mr. Jones, in your statement of qualifications and of course this examination, you mentioned that you were involved in developing Babcock & Wilcox's predictions for the semi-scale test; is that correct?

A (Mr. Jones) The recent ones, yes.

Q Have you had an opportunity to review the actual results of the semi-scale test?

A Before the semi-scale SO-7-10B, which is the one that I participated in, we have yet to receive the data to make the comparison.

Q And that would be the same semi-scale test that is referenced in Page 4-80 of NUREG 0565?

A Yes, that is the same one.

Q So, would I be correct in assuming how accurate your predictions were at this time?

A We have a little bit of an idea as to how well we predicted the experiment. The problem with SO-7-10B experiment is that they stuck open a relief valve on the secondary side of the steam generator which they did not tell us about -- or they told us about that we could not get the information to model that portion of the system.

In examining our prediction and looking at the phenomena that was occurring, it became apparent that the generator was a very important part to the prediction. That it was dragging the primary system pressure down with it and we did not in our prediction initiate high pressure injection at the normal setpoint.

We could not get down to it, and we believe that is partly an effect caused by this stuck open seconary side valve. So, we are really not sure what, or even how valid the comparison is until we see much more information on that valve.

Q Mr. Jones, with respect to NUREG 0565, and particularly Part 4.0 that begins on Page 4-1, it appears a discussion by the NRC Staff of the analytical model to be used at B&W for the analysis of small breaks; are you familiar with that?

A Yes, I am generally familiar with the discussion that is given in that section.

Q Okay, because of the order of witnesses in this proceeding, we have not had an opportunity yet to question the Staff in respect to section, but based upon what is written in NUREG 0565, it appears that the Staff has some reservations with respect to certifying that the B&W analysis satisfies the requirements of 10-CFR-50.46.

Is it your belief that the B&W analysis does satisfy those requirements?

A (Mr. Jones) Yes, it is my belief that they do.

Q Do you have any understanding of the reasons why the Staff has not reached that conclusion?

A Well, I am not so sure that they haven't. I have been trying to discuss the sequences at best I can understand it.

B&W has performed a set of small break calculations for the Rancho Seco facility which had demonstrated conformance to Appendix K, the model is in conformance with Appendix K, is approved, and the results of the analysis show that we meet 10-CFR-50.46.

To date, I have no information in my hand which says that my model is no longer valid or that those analyses are no longer valid.

Now, the analyses that were performed at that time dealt with the break size range of rougly .04 to .3 square feet.

Some would be, the items that were discussed within NUREG 0565 are not appropriate for that break size range.

Those are the design basis small breaks for demonstrating conformance to 5046.

Now, after the May 7th Order, or actually prior to it, we were instructed to perform a set of analyses for the Staff, because of the assumptions that were utilized in the analyses, it is unclear as to how these analyses were ever meant to meet Appendix K or 5045.

Now, there are selected cases where the analyses were done basically with the small break model, but cases where we are looking at a loss of all feedwater, were beyond the scope of Appendix K.

It takes a multiple failure and Appendix K requires a single failure.

So, it is not clear as to how the analyses were ever meant to demonstrate comrliance with Appendix K and 5046.

Rather, these analyses were all performed to look at alternate event sequences and multiple failures in order to develop an improved set of operator guidelines. That was the basic features of that analysis, so, I can't understand myself where the basis is that we are out of compliance with 5046 at this time.

Q Referring to Page 4-10, at the bottom of NUREG 0565, in section 4.1.1.4 conclusions. You find the last sentence states accordingly, integral verification of the methods should be included as part of the request for approval under 10 CFR 50.46.

Has B&W made a request for approval under 10 CFR 50.46 for this analyses?

A Again, my understanding of 0565 is that it is basically a document which is examining the May 7th analyses.

These analyses were never intended to try to meet 5046 and we have not made a request that they be considered in compliance with 5046.

Q Do you know whether B&W intends in the near future to make such a request?

A Again, these analyses are outside the scope of Appendix K already and I just don't see how they ever were intended to meet 5046.

Selected analyses again may be submitted to the Staff with additional information that the Staff deems it necessary in order to get conformance, in order to get those analyses to have acceptance and approval under 5046.

Q Both in the sentence that I just read and also on the next page, 4-11, Section 4.1.1.5, which is entitled recommendations, Part B, both of those statements suggest that additional analysis is recommended by the Staff, presumably by B&W to show compliance with 5046.

Do you know whether B&W intends to provide that additional analysis?

MR. BAXTER: Mrs. Bowers, this seems to me repetitive. I think the witness has given his views abou. the relationship between this analysis and 10 CFR 50.46. He is asking the same thing over and over.

MR. ELLISON: Pardon me. No, the question now I asked earlier whether -- Mr. Baxter is correct, when I asked the relationship between this analysis and 5046 and received an answer from Mr. Jones which I assume expresses B&W's view that that they are somewhat unrelated, but nevertheless, the fact remains with the Staff appears to be asking for additional analysis and even though B&W may feel that it is not necessary to show compliance with 5046, they may plan to do the analysis and submit it to the Staff.

So, my question now is whether or not they do intend to do the analysis regardless of whether they feel it is required.

MRS. BOWERS: Well, we think it is a different matter and we would like to have the witnesses respond.

MR. JONES: I have read this document 0565 and partly understand some of the Staff's technical concerns.

In my opinion, I believe that we already account or the model conservatively accounts for is the effects that are listed in 0565.

B&W at this time has not made a decision relative to performing additional analyses but I am sure we will be discussing this with the Staff in the future.

BY MR. ELLISON:

Q Mr. Jones, in yesterday's discussion of the non-condensible gasses and their possible effect on natural circulation, I recall that you mentioned that with respect to, under pressurization events over cooling events without a small break LOCA, that the core flood tank would be valved out, do you recall saying that?

A No, I did not. What I stated yesterday was that if the operator after a small break LOCA was in the process of performing a cool down of the primary system, then he could valve out the core flood tank during the transition.

Q Could you briefly explain how that woud be done?

A The specifics on how that would be done would have to be addressed to the licensee. I am not sure exactly how it is done.

Q Can you tell me whether in any of the analyses that we have been discussing you assume that the core flood tank had been valved out?

A No, the analyses did not, but the analyses are performed through the major transient until we have demonstrated that the injection flow provided to the system is sufficient to match the core decay heat and the system has stabilized.

The guidelines for valving out the core flood tank, is that after you have a stable system, and you are depressurizing, and there may be another criteria or to that I just can't recollect off hand then he can valve out the core flood tank, it is outside the range of where we would historically analyze these events.

Q Are you aware of any procedure that would direct the operators at Rancho Seco to attempt to valve

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out the core flood tank after it had been actuated to prevent the nitrogen blanket from entering the system?

A I am not aware of any procedure that says that, but I am not aware of their procedures either.

Q Did you assume any such procedure in any of your analysis?

A No, we did not.

MR. ELLISON: At this time, Mrs. Bowers, I would like to move the admission, first of all, CEC 3, that is the Failure Modes and Effects Analysis of the integrated control system.

MRS. BOWERS: Maybe we should take these one at a time.

MR. ELLISON: That is correct.

MRS. BOWERS: Let me check with Mr. Baxter.

MR. BAXTER: Could I have just a minute?

MRS. BOWERS: Yes.

MR. BAXTER: Mrs. Bowers, it is a little unusal I think the Energy Commission offer evidence to our witnesses but in this case, CEC Exhibit 3 is testified to by our witnesses in their testimony so we would have no objection to the admission of that Exhibit.

MRS. BOWERS: Mr. Lewis?

MR. LEWIS: We have no objection.

MRS. BOWERS: CEC Exhibit Number 3 is admitted into evidence.

(Whereupon CEC Exhibit 3 was received into evidence)

MR. ELLISON: Mrs. Bowers, I would also like to move the admission at this time of CEC 15 and 16 and I move them together because they are actually one document.

MRS. BOWERS: And that is the Consumer Power Campany Report; right?

MR. ELLISON: That is correct.

I think it would be proper for the cover letter to be removed from CEC 15 as that represents a product of Consumer's Power Company and not Babcock & Wilcox and has not been identified by these witnesses, so, I would suggest that commencing with Page A & B 1 which is the Babcock & Wilcox analysis identified by these witnesses that that be labeled CEC 15, with the first three pages be struck and I would move the admission of CEC 15 and 16 --

MR. BAXTER: I may have other problems, but I certainly have a problem with that procedure, Mrs. Bowers, I think these are important to an identification of both the time frame and other information about these analyses. They certainly don't prejudice the identification of who the author is. I think it should be complete.

MR. ELLISON: That is fine, we have no problem with it being complete. I was anticipating that people would object because these witnesses are not from Consumers Power, and technically not identified in the first part of this document but we have not objection to its inclusion.

MR. BAXTER: We have no objection to Exhibits 15 and 16 being received.

MR. LEWIS: No objection.

MRS. BOWERS: CEC's Exhibits 15 and 16 are admitted into evidence.

(Whereupon CEC Exhibits 15 and 16 are recieved into evidence.)

MR. ELLISON: I would also like to move the admission of CEC 17, that is the Babcock & Wilcox analysis identified by these witnesses yesterday, RCP Pump Trip, non-LOCA cases.

MR. LEWIS: Mrs. Bowers, of course this document was already amended on the record and I am not quite certain how that should be noted, but it would seem if we are moving the admission of the document and at least one of the figures has already been corrected, that should somehow be noted.

Additionally, and I was going to go into this in my cross, but there is a subsequent document, that does update this report and that was a document which I believed we showed to Mr. Webb earlier today.

As I recall, there was some discussion with the witness about whether or not there was a later document, he could not say whether or not there specifically was a later document, but he was aware of the fact that the analysis had been updated.

That document is dated October 24. It is a letter dated October 24th to the NRC from Mr. Mattimoe, which the cover letter very briefly states, in the letter to you dated August 27, 1979, I don't know if that is a mistake, because I notice that this one says the 29th, but in any event, the district provided an analysis entitled, Analysis Summary in Support of an early RC Pump Trip, the attachment to this letter carries a revision to Section 3 of that analysis, and I believe that the Section 3 analysis was the basis for the witness's amendments that he gave on the record today.

MR. BAXTER: Before we follow this chain of correspondence any further I have an objection -- this particular piece of correspondence with its attachments was obtained by the Energy Commission during discovery and inspection of documents available in the SMUD offices and I certainly had no objection to them using this for the purposes of cross examination.

I think, however, that it became clear that the witnesses had neither prepared this or used it or relied upon it in the direct testimony they offered here.

So, I would object to the admission of this evidence as not having been sponsored by any witness here in this proceeding or relied upon.

DR. COLE: Mr. Lewis, I have a problem, you said the changes were in Section 3, is the Section 3 that you are referring to the summary of results page?

MR. LEWIS: Mr. Capra is going to brief me on what it means.

MRS. BOWERS: Well, another thing, Mr. Lewis, you referred to the cover letter from Mr. Mattimoe to NRC, August 27th, and of course what we have got here on CEC 17 is B&W of Mr. Janis to Mr. Rodriguez.

MR. BAXTER: He is talking about another document that we don't have have, Mrs. Bowers.

MR. ELLISON: I would like an opportunity to respond to Mr. Baxter.

These witnesses have identified CEC 17 as a B&W analysis with which they have some familiarity. They have testified regarding it in cross examination and they have stated to the best of their knowledge this is the most up-to-date and correct analysis for this circumstnce in as much as it is pertinent to this hearing and has been identified by these witnesses as being correct, and it certainly appears in the transcript of this proceeding, and would make for a much more complete record and would be perfectly appropriate to admit it into evidence.

MR. BAXTER: Mrs. Bowers, it is just not the way evidence is introd [illegible]d in evidentiary hearings I am familiar with. I mean we have had a wide range in cross examination here and I am not being wholly critical of that but Mr. Ellison is free to bring in all sorts of documents from a scientific literature and cross examine and test the witnesses knowledge about their direct testimony, but this piece of paper, in spite of the fact that it was prepared by the same company they work for, was not relied upon in their testimony, and so just the fact that we now have some transcripted cross examination about it is no basis for introducing it into the record.

MR. ELLISON: Mrs. Bowers, if I may make one quick response. I agree with Mr. Baxter that simply because there has been discussion of the matter on the record it should not be -- that alone -- should not be grounds for admitting evidence into a proceeding.

However, these witnesses have not said that they are totally unfamiliar with this document or that it is incorrect, just the opposite. In effect, they have adopted it, they have said it is B&W analysis they have familiarity with and that they believe it is correct.

MR. BAXTER: I don't know what the term adoption means, and the fact that they are familiar with it likewise, does not mean it is being sponsored here.

If Mr. Ellison would like to introduce evidence, he is going to have his own witnesses and may introduce evidence through them, but not by just showing our witnesses any piece of paper and they say they are roughly familiar with it and therefore it gets into the record here.

MR. ELLISON: The important statement I believe is that the witnesses believe that it is correct, I would also point out that if the Board desires to have a complete record, there is certainly no way aside from subpoenaing the author of this document that the California Energy Commission can produce a witness who prepared it or who is personally familiar with it.

So, when the Licensee presents Babcock & Wilcox witnesses who identify the document as being product of their company with which they are familiar and believe it is correct and that it would be very appropriate for the Board to consider that evidence.

MR. BAXTER: In which they have stated they relied upon in no way in the preparation of their direct testimony here.

MRS. BOWERS: Mr. Lewis?

MR. LEWIS: My major concern here is that I think that the CEC Exhibit 17 has been revised and the discontinuity between the two documents that I was identifying to you is apparently owing to the fact that B&W sends certain analyses to its licensees, including SMUD and SMUD sends these analyses on to the NRC, apparently, what did happen was that SMUD, that there were revisions going on and that SMUD did send to us on October 24th a revised analysis of early reactor coolant pump trip which is my understanding is a revision to the analysis contained in CEC Exhibit 17.

So, my overriding concern is that I think that CEC Exhibit 17 really is no longer the B&W analysis, the present B&W analysis, of the double ended steam line break.

I may be able to clear this up on some cross examination of the panel. I am not quite sure what else to suggest --

MRS. BOWERS: Mr. Ellison, were you furnished the August 29th Babcock & Wilcox letter by the Licensee?

MR. ELLISON: Yes, ma'am.

MRS. BOWERS: Well, you know, I am a little concerned. There is a provision that under discovery that anything furnished must be updated if a later revision comes out.

MR. BAXTER: Well, let me explain how the documents were made available to the Energy Commission.

I don't know if you followed the

discovery process very well, it was rather extensive.

We had so many documents that were covered by the request that rather than to identify and list them all, which we were requested to to, we set up an office at SMUD's headquarters here and simply arranged the documents by category and had file drawers in there which Energy Commission Personnel then spent days in reviewing and selected those they would like to have copied and we copied them for them.

So, I can't swear to it but, the document should have been in there, the updated version. They were all made available for them to inspect and copy and I have not tried to endeavor to keep a list of very document the Energy Commission reproduced and every time some remotely related subject matter was produced to send it to them.

But, this document given its date of October should have been there as well and they may or may not have copied it, I don't know.

MR. ELLISON: Mrs. Bowers we do have a list of every document that we copied, unfortunately, it is not with me here now, it is back at the En.rgy Commission.

We do not have a list of every document that was provided, I don't think it would be helpful to the to the Board for Mr. Baxter and I to debate whether or not it was in fact produced by the Licensee. To the best of my knowledge, it was not --

MR. BAXTER: Now, what do you base that remark on, Mr. Ellison?

MR. ELLISON: I base that remark on the understanding that I have with the people that went through those files that they were to procure everything that related to, among other things, this topic.

MR. BAXTER: Well, before you impugn our production of documents, it may be that one of your people made a mistake. I think that is inappropriate.

MR. ELLISON: No, I certainly don't mean to impugn your production of documents, and I don't want to engage in this kind of discussion. The point is there is a subsequent document and to the best of my knowledge we do not have it and for whatever reason, I think it would be appropriate for us, at this time I would withdraw our moving the admission of CEC 17 at least temporarily to a wait the NRC Staff and their cross examinations since they are the only ones at this time that I know of, aside from the Licensee who have the updated analysis.

MRS. BOWERS: Mr. Lewis, do you have additional copies?

MR. LEWIS: No, I am sorry, I do not. We can try and get them.

I mean I had not initially intended to offer -- the copy that we have not only is it marked up but it is also titled on some pages and I think there are some pages that don't have complete --

MRS. BOWERS: I don't believe we want it.

MR. LEWIS: I may be able to ask some questions on cross examination based upon it that may help to clarify the record to some extent but I don't know that I really have a clean document to offer.

MRS. BOWERS: Mr. Ellison, let's go back to you.

MR. ELLISON: Well, Mrs. Bowers to summarize this discussion, the only thing that I can say is that these witnesses have identified the analysis that we are talking about and there may be updates to it.

They have testified to with respect to over-cooling events, analysis that bound overcooling events, worst case overcooling events, and specifically I am referring to Pages 44 and 45 of their testimony.

They have stated that the analysis in CEC 17 is to the best of their knowledge, correct and we think it is certainly relevant and admissible

As far as the problem of the updated analysis there is nothing that I can do about that at this point since I do not have the updated analysis report. So, I think it would be more appropriate for us to see if the Staff and myself cannot work out something in terms of getting that document and deal with this matter at a later time.

MRS. BOWERS: I really didn't mean to invite you to talk more about 17, I thought perhaps you had other exhibits.

MR. ELLISON: No, not at this time. I do have a couple of additional questions for these witnesses.

BY MR. ELLISON:

Q Mr. Karrasch, are you aware of any instances where the low pressure safety features actuation setpoint of 1600 has been reached in a Babcock & Wilcox plant like Rancho Seco during an overcooling event without a LOCA?

A (Mr. Karrasch) Yes, I am.

Q Could you describe those events?

A I am familiar with two such events that have occurred at Rancho Seco since the startup of the plant.

One of them occurred in March of 1978 and was an overcooling of the reactor coolant system due to an uncontrolled injection of auxiliary feedwater for some period of time.

I don't exactly recall the details of that transient, except to say that the overcooling by the auxiliary feedwater was of the order of magnitude of about 200 to 250 degrees in a time period of one hour and that the high pressure injection system did come on as designed and maintained reactor coolant system pressure and reactor coolant system pressurizer level -- excuse me -- it maintained pressure and pressurizer level during the entire course of the event.

The second one occurred in January of this year, I believe -- excuse me -- in January of 1979 and I am not familiar with that one even to the extent I was of the first one, except that I do know the high pressure injection system was actuated and reactor coolant system pressure and inventory were maintained.

Q Do you recall the initiating event for the second one?

A No, I do not recall.

Q Your answers were specific to Rancho Seco, has this also occurred in other B&W plants like Rancho Seco?

A Yes, I believe it has.

Q Could you give us an estimate of the number of times?

A I am trying to search back into my memory just from general knowledge of the operating experience of B&W plants to give you an estimate, but it would be very rough in order of magnitude type estimate, but let me say that to the best of my knowledge there have been approximately 15 to 20 overcooling events on B&W plants which have come close to the actuation point for the high pressure injection system, and by close, let me say within 100 to 200 psi.

I don't know if the high pressure injection system was actuated on those events, this is just a very general knowledge from looking at operating experience over the years.

Q That is fine.

In those 15 or 20, that ball park estimate that I recognize, but in those events would that include events where manual initiation of the high pressure injection system avoided reaching the setpoint for automatic initiation of that system?

A I am really unable to answer that because, again, I am using overcooling in a very broad definition here as being just the times when the reactor coolant system temperature fell below 530 to 540 degrees F. I am really not aware of the specifics of the actuations of the high pressure injection system during events like that whether they be automatic or manual.

Q It is possible that you could prevent safety features actuation on automatic safety features actuation by manually initiating high pressure injection; isn't that correct?

A Yes, in some circumstances if you had an overcooling event underway and the reactor coolant pressure was dropping at the starting of second makeup pump for instance would inject more fluid in the reacotor coolant system and help to minimize the pressure reduction that was occurring.

Q In the overcooling events as you define them in the broad definition that you gave them, would you expect that all of those overcooling events would result have resulted in one way or another in having too great a heat sync in the steam generator?

A Yes, really the only way you can overcool the reactor coolant system is to either inject too much feedwater or relieve too much steam, you know, over and above that, which the normal control systems are designed to do on a reactor trip incident.

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MR. ELLISON: That is all I have, thank you Mr. Karrasch and Mr. Jones.

MRS. BOWERS: Well, we took a break very early today, so let's take another ten minutes.

(Whereupon the Board recessed for ten minutes)

(Whereupon the Board proceeded at 11:00)

CROSS EXAMINATION

MRS. BOWERS: Are we ready to proceed, Mr. Lewis?

MR. ELLISON: Mrs. Bowers, before we proceed I would like to clear something on the record and state for the record that we have found out during the break, the Licensee did produce to us the document that the Staff has and that we indeed copy it so perhaps apologies are in order.

I have been told that the reason why, it was our feeling from looking at the document that one of them was a B&W analysis mailed from B&W to SMUD and it is our understanding that the second document was that same analysis mailed from SMUD to the NRC and that and that was the reason for the confusion. So, just to set the record straight.

DR. COLE: Mr. Ellison, that is what you thought it was, but that is not actually what it was.

MR. ELLISON: No, we found out from what the Staff is saying is that there is the correction from Mr. Jones and so the documents are different in that respect and since I have not had a chance to review the documents there may be other differences as well that I am not aware of.

DR. COLE: Thank you.

BY MR. LEWIS:

Q Mr. Karrasch and Mr. Jones, I have a few questions, broadly speaking in the area of natural circulation and --

MRS. BOWERS: Could you pull your microphone closer?

Q -- tripping of reactor coolant pumps. I would like to direct your attention first of all to Page 32 of your testimony, and to the sentence beginning on Line 16 which says, these analyses were performed utilizing conservative assumptions over a wide range of plant conditions, and the analyses referred to are analyses of the ability of natural circulation to maintain adequate core cooling. My question is were all of these wide range of plant conditions described situations where the reactor coolant system was subcooled?

A (Mr. Karrasch) Yes, for this section on natural circulation, the reactor coolant system was subcooled, both as an initial condition prior to the natural circulation and also during the condition of natural circulation.

Q I would like you to turn to Page 37 in your testimony, starting on Line 22, you make the following statement, to determine that natural circulation has been established, the operator must monitor three basic parameters: (1) reactor coolant temperature, (2) reactor coolant pressure, and (3) steam generator level.

My question would be in what way could an operator determine whether or not or distinguish between single phase natural circulation and reflux boiling?

A I think the basic difference between single phase water liquid natural circulation and reflux boiling is the reactor coolant temperatures, during single phase solid water natural circulation, the reactor coolant temperatures are subcooled and during reflux boiling you would be indicating a saturated condition of reactor coolant system temperature.

Q Would the operator be observing a difference in Delta T between cold and hot leg for these two different conditions?

Would he be able to tell whether or not he was in single phase or in reflux boiling based upon Delta T?

A (Mr. Jones) No, not based off of Delta T, he would only be able to tell it by the saturated temperatures in the hot leg.

Q Now, I would like to ask you a question based upon statements made by Mr. Webb in his testimony, give me one moment.

Do you have a copy of Mr. Webb's testimony?

A (Mr. Jones) Yes, I do.

Q Okay, I would like you to look at Page 7, footnote 11 -- I think it is Page 9, footnote 11 -- where there is a discussion regarding a scenario in which it is stated by Mr. Webb that manual initiation of HPI could delay reactor coolant pump trip following a small break LOCA.

Mr. Webb asserts that this would result in earlier emergency core cooling system initiation but more loss of mass out of the break. Could you give us your assessment of the net affect of manual initiation of HPI in this type of situation?

A (Mr. Jones) Well, let me -- there are three types of small breaks we would really want to consider to address this problem.

The first is a very general classification of small breaks which would be small breaks without steam generator feedwater.

For these cases, for that specific case, you again go into another two classes of accidents. Ones that will just generally depressurize the system by itself below 1600 psi and then the very small breaks which would not necessarily depressurize the system prior to the generator drying out.

For the cases where you have a rapid depressurization of the primary system, manual acuation of the high pressure injection pump will only aid you in the process in that it puts a little more water in the system.

But, its effect would be relatively insignificant, both on the delay to ESFAS and certainly there should be no voiding in the primary system when you get down to 1600 psi, and I will go and explain that system in a few minutes.

For the smaller breaks which would normally not depressurize the system, those breaks would end up having to go to essentially a bleed and feed type mode of operation or a very high pressure condition and those breaks are extremely small and are not in the break size range where tripping the reactor coolant pumps are necessary.

That is, the .01 square foot for example is a break that without feedwater does not get to ESFAS initiation and breaks smaller than that.

In the pump trip study, what we found was that the lower end break, the lower break size that causes problems is a .025 square foot break.

So, that the smaller breaks getting the HPI on earlier is just a benefit in that you supply and makeup earlier to the system rather than any delay, waiting for an operator to initiate it and that is the first case.

I will go to the next simple case which is a break which is within the capcity of the high pressure injection system itself.

There are select break sizes relatively small around 005 square feet, whereby actuation of high pressure injection pumps on the automatic ESFAS signal will match the leak rate right at that time.

Now, if you have feedwater available you will maintain a solid system and have natural circulation with a supply of HPI water matching what is going out the break.

Actuating the HPI earlier, at a higher pressure rate would be of no important whatsoever in that when it came on, the system would depressurize or repressurize slightly until it finds this equilibrium point.

The last classes of breaks are the ones that would normally depressurize to 1600 psi and ultimately start voiding in the primary system.

For these breaks, what you have to examine at what system pressures would you flash in the primary system and start creating voids?

If you have natural circulation established in the system which you do after you lose the reactor coolant pumps, the flashing pressure is roughly 1400 psi so that if you actuate the high pressure injection pump above 1600 psi, or waited to 1600 psi makes very little difference in that the system will be solid at that time.

Additionally, in small break LOCA the pressurizer will not drain until you are below 1600 psi.

So you will have, essentially, a solid water condition within the primary system with the exception of normally the pressurizer and the early actuation of the high pressure injection just again provides a little more fluid inventory in the primary system to be used later on in the transient.

So, looking at the scenario that Mr. Webb has in his testimony, which suggests that manual actuation of high pressure injection pumps could lead to a partial core uncovery, I am in disagreement with him for the reasons I just stated.

Q All right, let me turn to this matter of the double ended steam line break accident analysis.

Let me show you this letter of October 24, 1979 which I referred to earlier and ask you if you are familiar with the letter with the analysis contained in it?

Do you have your own copy of that?

A I am not really sure, let me check.

Q If you can't locate it, it is not important, it is just that that is my only copy.

A I believe I just found it.

Q Okay.

A No, I am afraid I have the earlier version of that.

Q Please go ahead then and look at the version

I just gave you.

A I am ready I guess.

Q Are you familiar with that analysis?

A Yes, I am familiar with it, in general terms.

Q Is that the analysis which you were referring earlier when you amended the voiding figure given in CEC Exhibit 17 for identification?

A No, it is not.

Q Is the analysis contained in the October 24th letter which I showed you B&W's analysis of the same type of accident which we are considering here, the double ended steam line break?

A It is the same type of accident, let me try to draw the perspective a little bit.

It is my understanding that the analysis in CEC 17 was done at the request of SMUD for a concern that they had.

That analysis, as I stated previously, the error of 12 cubic feet in that document is incorrect, and it is my understanding that the revised number for this SMUD specific requested analysis is 100 cubic feet.

The analysis in the October 24th letter is a more severe analysis which was performed generically for the B&W plants in response to I & E Bulletin 7905C so they are slightly different analyses and I am not sure exactly what the differences are I would have to read this further to be able to pick it out. It is not obvious from the text, and I would have to refresh my memory before I could say what the exact differences are.

Q Based upon that document, under the more severe analysis assumptions used there, do you know what extent of voiding, what maximum extent of voiding would be associated with the double ended steam line breaks?

There is a Table I can specifically, a graph, a figure which I can specifically point you to. (Indicating to Mr. Jones)

A (Mr. Jones) It is approximately a void of 250 cubic feet in one of the loops, and there is another void in the other hot leg loop which appears to be about 200 cubic feet.

Q Were those with the pumps tripped?

A Yes, they were. Wait, let me check on that before I make the statement.

Excuse me, with the pump trip, the void was 250 cubic feet in one loop and in the other loop it was about 125 cubic feet and that was with a pump trip.

Q What did your figure of 100 that you gave earlier in your testimony, does that assume that the pumps have been tripped or were running?

A This was based on RC pump trip.

Q Can you give me your opinion as to the degree to which this 250 cubic foot voiding space, the degree to which that might inhibit natural circulation?

A Well, the analysis that was performed included the effect of that voiding on natural circulation, I don't have a one-to-one relationship as far as if it hadn't in effect or how significant it was, however, the analyses demonstrated that adequate natural circulation was maintained throughout the transient to keep the core cooled.

MR. LEWIS: Mrs. Bowers, those are my only questions. Unless the Board thinks it is necessary, I believe that the testimony will stand on its own. I am not certain that this letter has to be in the record, Mr. Jones has testified as to the revision to the CEC 17 for identification, and unless the Board believes this document has to be in the record, I am content to rest upon the development of the record as it exists.

DR. COLE: Was CEC 17 admitted into evidence?

MR. LEWIS: No, it has not been admitted, it was withdrawn in fact. It was identified, I mean it has been identified and that is all and as far as I am concerned I think that that is probably sufficient for the development of the record, it was used for cross examination purposes and I believe has been revised through the oral testimony of the witness.

MR. BAXTER: Excuse me, Mrs. Bowers, I don't know that there is an offer pending, may I proceed with redirect at this point?

MRS. BOWERS: Well, one problem we have with this, Mr. Jones had very little time to go through the later version of CEC 17 and perhaps it is better to, over the luncheon break for him to have an opportunity to study it more carefully and then go back to this, because we think he was asked to do an instant analysis that may not have given him sufficient time.

MR. LEWIS: I am perfectly happy to have that done. I am not choosing to sponsor this document, it concerns a double ended steam line break analysis and it is not a matter which I am seeking to have considered but certainly if the Board believes that it would make a clearer record, perhaps there is -- well, I will do what the Board wishes with regard to trying to get this document into the record or not.

MR. SHON: Mr. Ellison, I believe you withdrew your offer of CEC 17 pending the production of this Staff document and a chance for the witnesses to look at it. Has what you heard made you decide that you still want to offer it or still want to leave it withdrawn?

MR. ELLISON: Mr. Shon, as the record now stands, we would like to offer CEC 17 and the analysis that the Staff has presented for precisely the reasons that were voiced by Mrs. Bowers.

We think that these witnesses have been able to identify these documents as the B&W analysis that is applicable here and for that reason we believe that it is both relevant and admissible, but at the same time the oral statements that have been made on the record were made on the basis of a rather quick review and that is the only thing appears on the record. We believe the record ·ld be some what incomplete.

I think that in order to insure that the record actually reflects the B&W Analysis, it would be better to have a contained analysis itself.

So, yes, I would move admission of CEC 17 and the document identified by the Staff.

MR. LEWIS: My only problem with that is that I am not quite sure that will totally clarify the record.

It is my understanding that one of the things that has to be cleared up here is exactly what is the relationship between CEC 17 and this October 24th letter.

The October 24th letter is a revision to an analysis sent to us by SMUD and it is a revision only to Section 3 of that analysis.

Properly speaking, the October 24th letter by itself, it really can't stand by itself, it would have to be as part of an analysis including the original submission of August 27th and also the revised analysis which Mr. Jones spoke of when he indicated the revised steam voiding amount.

It is apparently not specifically this analysis. So --

MR. BAXTER: Mrs. Bowers, excuse me, I have to point out that I think this illustrates my problem with receiving into the record documents that parties come into the hearing room with and use for cross examination.

They haven't been reviewed carefully by counsel or the witnesses in advance and sponsored, and perhaps not researched as to their correctness as of today.

MRS. BOWERS: Why don't we let this matter rest until after the luncheon break and then perhaps we can get a clearer picture of what we have and the witnesses will have an opportunity to also take a look at it.

MR. SHON: I would also like to say that in regards to CEC 17, you know a substantial fraction of that document consists of graphs whose origin and relationship to the rest of the document was never accurately established yesterday even.

No one knew exactly which case they were although some thought it was one and some thought it might be others. I am reluctant to rely in our decision on any document or pair of documents that seems so ill waste together.

I would like to know after lunch just where the graphs came from, what Section 3 means since we can't identify any Section 3 in the previous document, and why the various numbers of cubic feet of voids appear in the two documents, the corrected document and so on.

It is quite confusing as it stands right now, I think.

MR. ELLISON: Mr. Shon, we would agree that there is some what of a confusion on that point and we would hope that both the Licensee and the Staff would do whatever they can to clear it up. We are in a little bit of a poor position to do that since we haven't author` these documents a.d don't access to those.

MR. BAXTER: Yes, you did.

MR. ELLISON: I am willing to state that for the record I didn't author these --

MR. BAXTER: I am sorry, I thought you said offer, excuse me, I didn't hear you.

MR. ELLISON: Since we did not author the documents nor have access to those people who did. The only thing I think really what we could do in that respect would be to take the opportunity to ask Mr. Rodriguez who is the person identified as receiving the document when he appears what his understanding of it is.

MRS. BOWERS: Mr. Lewis, have you concluded your cross examination?

MR. LEWIS: Yes.

MRS. BOWERS: Fine.

Do you want to go ahead, Mr. Baxter?

MR BAXTER: The last comment then, are we withholding the offer of CEC 17 and the yet unidentified, for the record, October 24th letter until Mr. Rodriguez appears; is that the way it is going to be left?

MRS. BOWERS: I thought after following the luncheon break that we might be --

MR. BAXTER: Mr. Rodriguez is not appearing after lunch.

MRS. BOWERS: I know, but getting a clearer picture on this. Of course, be possible to put Mr. Rodriguez on for this limited purpose.

MR. LEWIS: Let me make a suggestion. We do have in our records here the August 27th analysis of which this is an update and perhaps the parties can get together at the luncheon break and try and agree on just what might be submitted to the record in order to clarify this matter, but I think we can try and talk about that at lunch.

REDIRECT EXAMINATION

BY MR. BAXTER:

Q Mr. Jones, you were questioned by Mr. Ellison about the current requirement that the reactor coolant pumps in B&W plants have to be tripped upon reaching the safety features actuation system, low pressure setpoint.

Is this phenomena which requirest the tripping of those pumps because of certain class of small break loss of coolant accidents unique to B&W plants?

A (Mr. Jones) No, it is not.

Q Has the NRC imposed the same reactor coolant pump trip requirement on all licensees with pressurized water reactors?

A It is my understanding that all licensees irregardless of the vendor plant type are required to trip the reactor coolant pumps, the only difference is their choice of signals.

Q Mr. Karrasch, you were asked a lot of questions about what the NRC Staff calls, the sensitivity of the Babcock & Wilcox once through steam generator.

You have addressed the design and operating characteristics of the system at Pages 22 to 25 of the pre-filed testimony, but would you summarize for us your views on the responsiveness of the system and how the so-called sensitivity has been affected by the post Three Mile Island actions?

A (Mr. Karrasch) Yes, I will.

During the cross examination, the real issue of sensitivity and the consequences of it didn't get a chance to get defined and our testimony don't feel the testimony got on the record during the cross examination, and what I would like to do is briefly summarize that testimony and see if I could put this sensitivity issue in perspective, from the B&W standpoint.

As shown on Page 17 of our testimony, the B&W nuclear steam system and the once through steam generator have been safely designed to minimize reactor trips during minor secondary system transients and attempt to keep the reactor online producing power both during and following the secondary system disturbances.

Now, to accomplish this goal, we really utilize the heat transfer characteristics of the once through design to provide this close coupling that we have been talking about this week between the primary and secondary systems to allow a relatively rapid response of the primary system temperature and pressure to changes in the feedwater flow rate.

There are numerous discussions of that in various pieces of testimony in this hearing, and I think the point is that the discussions that describe the rapid response of the primary system to feedwater flow changes is an accurate representation of the responsiveness of our system.

This response of the primary system which has often been called sensitivity is really an inherent feature of the design, which when coupled with the integrated control system, results in a design which can automatically respond to these feedwater flow changes that occur and hopefully maintain the reactor within the limits of the reactor protection system.

So, our approach to this favorable system design feature has really been to deal with the consequences of the responsive system and provide a focus on plant safety and core cooling.

That is, we feel to really enjoy the more desirable features of a responsive system and have taken positive action to focus on plant safety.

In order to do this, we used what is called a defense in depth approach to minimize any risk which might be associated with the responsive system.

This defense in depth approach really begins with the integrated control system itself.

As we have been talking this week, the ICS has been wn to maintain the reactor online within the limits of the reactor protection system for secondary system upsets.

That is the design goal of the integrated control system as to enhance plant availability and keep the reactor producing power.

Now, if the integrated control system and the nuclear steam system cannot accomodate the secondary system upsets or the feedwater transients we have been talking about, the reactor protection system is designed to trip the reactor and preclude the onset of an unsafe condition while the reactor is at power.

The RPS with a high pressure trip or the low pressure trip or a high reactor coolant system temperature trip is designed to stop the power operation, if you will, for an onset of an unsafe condition.

Firing a reactor trip, our plant systems are designed to achieve and maintain a stable, hot shut down condition.

The pressurizer, which we have been talking about and the pressurizer control system is designed with the heaters, the pressurizer spray, and the normal makeup and purification system to maintain a stable pressure control in the reactor coolant system following the reactor trip.

The steam generator heat removal systems, utilizing the main feedwater system, and the auxiliary feedwater system if required, are used to remove the core decay heat following reactor trip.

These systems are designed to be effective in removing decay heat for both natural circulation and forced circulation conditions.

Now, should a failure in these systems occur, which might result in an overcooling or undercooling condition, the high pressure injection system is designed to maintaint the reactor coolant system inventory and re-establish the pressure control during an overcooling event or provide core cooli'g in the very unlikely event that you have an extended .oss of all feedwater or all OTSG cooling.

So, the high pressure injection system is there to restore pressure control in the event of an overcooling event or to remove the core decay heat in the event that you lose the steam generator cooling.

Both of those are the normal control systems for containing hot shut down.

Now, the changes made at Rancho Seco since TMI as a result of a May 7th Order, we feel that really addressed the sensitivity issue my minimizing the potential for RCS temperature and pressure disturbances thereby contributing to the defense in depth that I have been talking about for safe reactor operation.

I would like to just summarize once more the testimony that we gave on the effect of the changes in the May 7th Order with respect to the sensitivity of the reactor coolant system.

The first is that the high pressure coolant -- the high reactor coolant pressure trip setpoint is lower than the setpoint for the PORV increase.

This, of course, was done to minimize challenges to the PORV and the possibility of the valves sticking open causing a small break LOCA and causing a transient situation in the reactor coolant system which may further challenge the safety systems and burden the operator.

To further minimize the reactor coolant system transient response to secondary system upsets, the hard wired trips on loss of main feedwater and turbine trip have been installed.

These of course result in a prompt decrease in core heat generation in the event of loss of feedwater or turbine trip and provide an additional margin for a voiding the reactor coolant pressure increase which might challenge the power operated relief valve.

Again, a feature added to minimize the reactor coolant system temperature and pressure changes as the result of an upset in the secondary system.

The third change is the reliability of the auxiliary feedwater system to provide secondary cooling including the direct indication of auxiliary feedwater flow in the control room has been enhanced.

end 4
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In addition, procedures have been developed and implemented for initiating a controlling auxiliary feedwater independent of the integrated control system.

This change, again, further assures that auxiliary feedwater will be available to minimize over-cooling and undercooling events following a loss of main feedwater and again, minimize challenge to the safety systems and burden on the operator.

The additional small break LOCA analysis that was performed and the operating guidelines which were implemented, provides further defense in depth that an inadquate core cooling condition will not be achieved during a small break LOCA.

And last, a Failure Modes and Effects Analysis of the first line defense in depth that I talked about the integrated control system was completed to confirm the ability of the ICS to maintain stable reactor power operation in the event of secondary system upsets and really confirm the adequacy of that first line.

So, thus, the changes made since TMI really serve to decrease primary system fluctuations as a result of secondary system upsets. Decrease the probability of a transient which may challenge the safety systems and burden the operators.

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MR. BAXTER: Mr. Ellison also asked you some questions --

MR. ELLISON: Pardon me, Mr. Baxter, please. I would like to state something very quickly for the record.

In the course of that answer, Mr. Karrasch you stated the reliability of the auxiliary feedwater system had been enhanced, other than asking you whether or not -- pardon me -- I am going to address this to the Board. Other than asking Mr. Jones and Mr. Karrasch whether or not they were prepared to address that question and the answer received was, no, we did not ask any questions with respect to auxiliary feedwater reliability, we understand that Mr. Dietrich will address that question for the licensee.

So, I would like Mr. Baxter to clarify whether these witnesses were meant to be offered on that issue, and if so, I would like the opportunity to cross examine them on that issue.

MR. BAXTER: Well, I think they have in their testimony simply in one place summarized what some of these Post TMI modifications are.

Indeed the contention issue, I am not sure which it is the question that addresses the auxiliary feedwater reliability is addressed in Mr. Dietrich's testimony and he will be the one who will respond to your questions about it.

I think a cross reference is made in the written testimony to that effect.

MRS. BOWERS: Earlier today, Mr. Baxter, you made some reference to some matter that you felt was outside the scope of the direct prepared testimony, and I think I shoul mention for the record, as you know, under the Administrative Procedure Act, we are here to conduct an Evidentiary Hearing on all matters that are relevant and that are material and in these Administrative Hearings we do not hold to the narrow viewpoint that a matter that is not explicitly covered in the direct is not a matter that it is often a matter that we want to hear about and we have the witnesses here and we are going to proceed.

So, it is broader rather than holding it strictly to the file prepared testimony.

MR. BAXTER: I understand. I was not suggesting a strict construction of that rule either, I think there is a rule of reasonableness that attaches to the subject matter of their testimony though.

May I proceed?

MRS. BOWERS: Yes.

BY MR. BAXTER:

Q Mr. Ellison also asked you several questions about what has been identified for the record as CEC Exhibit 3, the Failure Modes and Effects Analysis of the integrated control system.

And you testified that that analysis does not include multiple failures.

Would you tell us what you consider to be the significance of that fact?

A (Mr. Karrasch) Well, even though I testified that the scope of the Failure Modes and Effects Analysis did not consider mulitiple failures, the overall objective of the analysis was to evaluate the effects on the NSS of single failures was really met.

Maybe to help clear up the concern about multiple failures I could add that the plant safety analysis which was performed for Rancho Seco in the final safety analysis report has really provided an analysis of bounding events which have been shown to envelope the consequences of multiple failures in the integrated control system.

For instance, the double ended steam line break analysis which is documented in the FSAR and which we talked a bit about during the hearing.

It really bounds an overcooling event which might be caused by multiple failures in the integrated control system which might open all the turbine bypass valves; or open all the atmospheric dump valves, just as an example.

To convince yourself that multiple failures are covered, you really have to look at the FSAR, the basis for the FSAR analyses and study the fact that the FSAR results are acceptable and meet the criteria for safety of the plant.

Now, the safety systems which mitigate the consequences of these bounding analyses which are presented in the FSAR, the reactor protection system and the emergency safeguards systems specifically are designed to very rigorous safety standards and have been shown to be independed of the integrated control system.

So, we feel the fact that multiple failures were not considered in the failure modes and effects analysis really has no impact on the conclusion that was drawn from that analysis.

Q You testified that in CEC Exhibit 3, the functional block diagram approach was used as opposed to a component block diagram; how does that affect the capability of the analysis to determine the effect of ICS failures on the operation of the primary system?

A If you will remember during the testimony, we stated that the functional block method was adequate to determine the effect of single failures on the NSS response, and that of course was the original intent and objective of the failure modes and effects analysis.

As you also remember, we stated the analysis assumed the worst case assumptions of high and low failures of each input of each functional block and of each output, and what this ends up doing if you follow your way through the analysis it ends up in high and low failures of all the actuated equipment that the ICS controls.

One of the points that didn't come out in the hearing this week was that the number of pieces of equipment that the ICS controls is really quite small.

If you look at that actuated equipment, it can be summarized as follows:

It is the feedwater control valves themselves;

The feedwater startup valves;

The turbine control valves;

The turbine bypass valves;

The feedwater pumps; and,

The control rods.

The control system takes inputs from many places and manipulates those inputs in such a fashion as to control a relatively limited number of actuated devices.

The analysis we did, the functional block analysis with high and low failures, ended up with the actuated equipment going to a worst position.

That is a valve would go full open or full closed, or the feedwater pump would go to its high speed stop or it would turn off as a result of the single failure that we picked in the analysis.

And therefore, we feel that the resultant effect on the NSS has really been deemed to be a worst case analysis.

Now, the component evaluation would essentially obtain the same information about worst case information with the additional feature that failure probabilities of the components would be included in the fall tree portion of the component analysis.

We feel, that this extra probability analysis, if you will, of failures of the ICS really would not be very meaningful at this time.

The much more meaningful approach has to be to look at your operating data and your field experience and determine how many of the -- to determine first how many failures you have had in the integrated control system and then how many of those have put the plant in a transient condition and as stated in the Failure Modes and Effects Analysis, our examination of this data, shows that the ICS has encountered six failures which cause reactor trip in 35 years or reactor operation.

We feel as a result of this favorable result that additional performance of a probability analysis really would not be useful.

Q I would like to refer you now to the tables on Page 5-14 of CEC Exhibit 3.

You were asked questions about the implications of the operating experience reflected there with reactor trips and ICS malfunctions.

How do the Post-TMI actions affect your views of the significance of this data?

A If you remember in the testimony and during the cross examination, we testified that our conclusion that the ICS indeed prevent more reactor trips than it causes as a result of the comparison of tables 5-8 and 5-7.

Table 5-8 shows that there have been a total of six failures in the integrated control system on all B&W plants in 35 years of operation that have caused a reactor trip.

Table 5-7 states that there have been 47 instances prior to TMI-2 where we have had successful runback actions without anticipatory reactor trips.

In examining Table 5-7, in light of the changes that have been made subsequent to TMI-2, we feel that the anticipatory reactor trips on loss of feedwater and turbine trip will cause the number in that Table to decrease somewhat.

If you look at the turbine trip load rejection, where we had 14 successful occurrences, that quite clearly would go to zero.

On the feed pump trips, which is an overpressure transient, the lowering of the high pressure reactor trip setpoint could cause us to not to be able to have a successful runback on a loss of one feed pump; and that number 14 would reduced somewhat.

A control rod drop is a low pressure transient in the reactor coolant system and the changes made since TMI-2 probably will not impact the successful runback action there.

Similarly, a 10% step load increase is a low pressure transient and should not be affected by the changes made since TMI-2.

A step load decrease is similar to a feed pump trip where you have a storing of energy in the reactor coolant system due to the action of the ICS and the lowering of the high pressure trip setpoint and the increase of the PORV setpoint could cause the Number 11 to be decreased.

I think the important point from all this is that we feel the Number 47 will not be reduced to less than 6 and that the failure rate reflected in the Number 6 will probably not change also as a result of the changes made subsequent to TMI-2.

So, as a result of this comparison, we believe that our conclusion that the ICS prevents more trips than it causes, will continue to be valid even subsequent to the changes made at Rancho Seco since TMI.

Q The Oakridge National Lab Report which reviewed the Failure Modes and Effects Analysis was marked for identification as CEC Exhibit 4 and you were asked a number of questions about statements within that document including some criticisms and recommendations for further study.

What in your view though is the significant conclusion in that report?

A If you remember during the discussion on the Failure Modes and Effects Analysis, it was brought out that the Analysis was performed really to determine the effect of ICS single failures on the response to the nuclear steam system.

In the overall conclusion that we drew was that the majority of single failures that can occur in the integrated control system could indeed cause a reactor trip; but that the reactor core does remain protected for these failures in the ICS.

In addition, the second part of the Analysis had a rather thorough review of all B&W operating experience to study our actual ICS operating performance and look at plant transient events.

The overall conclusion that was drawn in the report from this operating experience review is that the ICS is a reliable control system which has had only 6 failures in the 35 years of reactor experience, causing reactor trip.

We feel, as stated earlier, that the conclusion that the ICS has met its design goal in preventing more trips than it caused has indeed been met.

We then had quite a bit of discussion on the Oakridge Report which was prepared in response to an NRC request to really critique our Failure Modes and Effects Analysis and draw its own conclusions.

Our review of that report shows that it indeed has provided a very favorable critique of the Failure Modes and Effects Analysis.

On Page 14, in the Evaluation and Recommendations, the Oakridge Report states that the Failure Modes and Effects Analysis is a significant asset to plant safety and availability.

We find no evidence that the ICS provides more frequent or more severe challenges to the plant protection system than other control systems of similar scope. Nor do these challenges exceed the plant protection system capability.

The report continues on in addressing the operating experience that examination of failure statistics in the B&W Analysis reveals that only a small number of ICS malfunctions resulted in reactor trips.

These data supported by conversations with plant operators demonstrate that the system is failure tolerant to a significant degree.

This feature is also evidenced by noticing the large number of postulated failures in the analysis that could result in a reactor trip compared with the experience low trip rate and practice.

The positive results of the FMEA, and the operating experience of the ICS show that the control system itself has a low failure rate and that it does not instigate a significant number of plant upsets.

This Analysis further shows that anticipated failures of and within the ICS are adequately mitigated by the plant protection system and many potential failures would be mitigated by the cross checking features of the control system without challenging the plant protection system.

Finally, the report closes with a statement that we are satisified that failures within the ICS do not constitute a significant threat to plant safety and that further analysis of this type may not be economically justifiable.

It is not evident that re-doing the analysis at this point to include this information would be worthwhile and it is our opinion that more detailed analysis would not provide significantly more enlightening information for purposes of the FMEA.

As a result of the conclusions presented in the Oakridge report and the conclusions presented in our Failure Modes and Effects Analysis, we feel that the report has indeed met its intended objectives.

Q Mr. Jones, you have been asked about your analysis of overcooling transients which is presented in the pre-file testimony and in addition about analysis of overcooling accidents which are discussed in CEC Exhibits 15 & 16 and about the Brookhaven Analysis which is at least discussed in CEC Exhibit 6.

Would you compare for us these analyses once more and discuss your views on the significance of each analysis?

A (Mr. Jones) Well, first, what we did in the purposes of creating the testimony was we were looking trying to address an overcooling of the steam generator as a result of a feedwater transient as that was the scope of the hearing.

The testimony has the analysis, or the results of the analyses of a main feedwater overfill, and demonstrates that no void formation occurs even though the pressurizer empties briefly.

Now, CEC 15 & 16 include results of that analysis, but also discussed in CEC 15 was the results of an overcooling accident, which is a steam line break.

In answering cross examination, I was questioned on the results of the overcooling accidents also which were different from my testimony in that void formation does occur and is expected to occur for these accidents; but the final conclusion is that the analyses demonstrated even for the overcooling accident that adequate natural circulation would be maintained to keep the core cooled.

CEC 6, which is the Brookhaven Study has a transient which is quite different from our transient that was analyzed for the testimony in that it not only includes a main feedwater overfill, but a concurrent auxiliary feedwater overfill coupled with a failure in the steam bypass system which causes it to stick open after being actuated.

So, in effect, it is a a double overfill, if you wish, coupled with a small steam line break which is substantially different than our analyses and there is no single failure that we can identify that would give this sort of scenario to the primary system.

But, irrespective of the results of whether it was an overcooling transient or an overcooling accident or whether you look at the Brookhaven analysis and CEC 6, or CEC 17, what all these analyses have demonstrated as is stated in the testimony, or at least the overfill accident is that adequate core cooling is maintained throughout these transients, although, even though void formation occurs for the overcooling accidents in CEC 15 and the Brookhaven analysis, and CEC 6.

This is really not that unexpected in that in an overcooling transient, you don't lose any liquid, it shrinks, you may have some flashing, but supplying high pressure injection ultimately recovers the system pressure which will condense the steam in the primary system and allow natural circulation to be re-established.

Additionally, the B&W reactor has an alternate means of decay heat removal which is unique to most PWR vendors in that we have the capability of going to a bleed and feed mode of operation.

As a result of all these analyses, we feel that the B&W reactor is safe from an overcooling standpoint and again, does not enhance void formation and is capable of handeling any possible void formation that occurs in the reactor coolant system during feedwater transients.

MR. BAXTER: That concludes my re-direct examination.

MRS. BOWERS: We will break for lunch, one hour.

(Whereupon the Board recessed at 12:00 for lunch)

AFTERNOON SESSION

(Whereupon the Board came to order at 1:00)

MRS. BOWERS: We would like to begin.

Mr. Lewis, do you want to report the success of the venture with the CEC 17 and its cousins?

MR. LEWIS: Well, I don't have much to report. As I understand it Mr. Jones was checking with his employer and trying to sort out which analysis is attached to which.

I mean, we put together what we think is the way the fall together, but I think until we have a stipulation -- I guess my suggestion is going to be that -- I think what we are talking mainly about is a matter of having them into the record, the proper and most recent analysis, and I am wondering if that can't be sorted out, it may not be able to be sorted out today. I am wondering if that could be sorted out subsequently if that most recent analysis can be substitued for CEC Exhibit 17.

MR. BAXTER: I still have an objection to the relevancy of the document. It has not been relied upon by these witnesses in the testimony and it has been used in some of the related earlier document was used in cross examination.

We now have discover that it has been supersceded by one or more documents and these witnesses they are familiar with what was put in front of them, it is obvious that in at least in an hour of diligent effort we have not be able to sort out the situation, and I object to the relevancy of the offer, and I would object to leaving it open for these witnesses potentially to be recalled for his matter.

MR. LEWIS: Even if it is simply for the purposes of identification of the document.

As I told you before, I am not really offering the document as what I consider to be our case. But, in response to the Board's concern that the record should reveal what the correct evaluation is, even if is only for purposes of identification of the document, at least it would seem that the proper evaluation should be marked for identification at a minimum.

MR. BAXTER: If we could ascertain "what the proper evaluation" is, I would be happy to submit it for identification purposes when we return in April.

MR. ELLISON: Mrs. Bowers, if I may, I would like to speak to this matter, but I would like to separate my comments with respect to the relevancy of the document and with respect to the confusion over which is the proper document.

First of all, with respect to the relevancy of the document, these witnesses have testified at Page 44 and 45 of their testimony as well as under cross examination, to the response of Rancho Seco to overcooling events.

They have stated that the analysis referenced in their testimony bounds an overcooling transient caused by excess auxiliary feedwater addition in the testimony of that respect.

They have ident. ied that this is a B&W analysis of, whether this is considered to be the bounding case for an overcooling transient or accident in the Rancho Seco facility and on that basis, I think it is relevant. With respect to the confusion of the document, I would first of all like to ask Mr. Jones, it is my understanding that he has been able to sort out what the situation is; if that is not correct, Mr. Jones, please correct me; but if it is, could you explain what the most current analysis -- what document sets forth the most current B&W analysis of this type of event.

MR. JONES: I have attempted to sort them out. It is not totally reconciled, but as far as what is the most recent analysis of the double ended steam line break, that is the October 24th, I believe it is, submittal.

MR. ELLISON: In that case, Mrs. Bowers, based on the discussion that we had, I would move admission of the October 24th submittal.

MRS. BOWERS: Substituted for your CEC 17?

MR. ELLISON: That is correct.

MR. BAXTER: I object for the reasons previously stated.

MRS. BOWERS: The Board thinks that the subject matter here is very relevant to the matters that we are considering in this proceeding.

It can be handeled one of two ways, we can accept the document, the October 24th document, for what its worth under the situation that exists today, or a B&W person who was involved in preparing it could participate in these proceedings.

MR. ELLISON: We would not object to either approach, Mrs. Bowers.

MR. BAXTER: If I am given that choice, I guess I would have it received for what its worth, given the state of the record.

The reason it was not addressed in this testimony is because we understood the scope of the hearing to be feedwater transients and that is what the overcooling events are that are addressed in the testimony.

This, of course, is steam line break analyses.

MRS. BOWERS: Well, somebody is going to have to make some copies of that October --

MR. BAXTER: I don't even think it has been identified, I don't believe, for the record as an Exhibit Number at all, at this point.

MRS. BOWERS: Well, I just spoke a minute ago to Mr. Ellison asking if he were substituting the October 24, 1979 document for the CEC #17 which has been identified earlier and he said yes.

They have been overruled, leave CEC 17 as it is and give us a new number.

MR. ELLISON: All right, I would like to have the October 24th letter be identified as CEC #19.

I think, if I may, a suggestion about the copying that we don't have sufficient copies at the momen . but we do have one that has been furnished by the Staff to us, and we would propose to copy it on Monday and transmit it to all the parties together with the testimony of CEC 5-2, which is to be filed Tuesday.

MR. LEWIS: But, if your copy is the same as ours, there are certain pages which are not completely reproduced. There is a problem with it, one is off the page --

MR. ELLISON: Well, maybe we can work that out off-the-record and find a good copy somewhere, I am sure.

MRS. BOWERS: I understand, Mr. Baxter, prior to the luncheon break, you had concluded the re-direct?

MR. BAXTER: That is correct.

MRS. BOWERS: Well, let's check with the other parties and see if there is some further cross. Mr. Ellison?

MR. ELLISON: Yes, ma'am just briefly.

BY MR. ELLISON:

Q Mr. Karrasch, in response to Mr. Baxter's questions on the sensitivity or responsiveness, if you will, of the Babcock & Wilcox nuclear steam sytem, I understood your testimony to say essentially that while the system was inherently reponsive, that that responsiveness had been considered in the design of Rancho Seco and systems had been designed to control it; is that a fair statement, a fair summary of the answer you gave?

A (Mr. Karrasch) I don't know if I said systems were designed to control the responsiveness. I think that I said was that the consequences of the responsiveness had been included in the plant safety analysis.

Q Would it be fair to say that the plant safety analysis considered the responsivness in the context of systems that exist at the plant to, specifically, such as the PORV setpoints, the integrated control system, the auxiliary feedwater system, the entire plant that

that the ar ysis was integrated to examine the responsiveness and what effect it would have.

A I think that is a fair statement, yes.

Q Is it true that in examining the consequences of the responsiveness once you consider the integrated control system?

A I am not really sure I understand the intent of your question, Mr. Ellison, could you maybe restate it for me, please?

Q Certainly. Would it be true that the integrated control system been designed with the responsiveness in mind and serves in part to control it?

A Yes, I think the integrated control system has been designed in conjunction with the responsiveness of the OTSG to minimize the primary system pressure and temperature changes as a result of feedwater flow upsets.

Specifically during power operation.

Q Yes, it would be true then, it would it not that the consequences of the responsiveness would be quite different if one didn't take into account the reliability of the ICS?

A No, when I talked about consequences of the responsiveness, I was really referring to transients resulting in reactor trip, the primary function of the ICS is to perform its function prior to the reactor trip.

I wouldn't say the ICS is there mitigate the consequences of a response of nuclear steam assessment.

Q Is it your opinion that the responsiveness of the system would be the same whether or not the ICS was functioning properly?

A The inherent coupling of the steam generator will be present irrespective of the control system uses.

The control system we have elected to use is designed to take advantage of that close coupling and maintain the primary system parameter within the limits of the protection system.

Q Isn't it true that the integrated control system was B&W's method essentially of concerning a responseiveness of the system and as the name implies integrating all the various aspects of it under the control of a single system so the the responsiveness could be managed?

A Would you repeat your question?

Q Will the reporter read that back?

(Whereupon the Reporter read back the question)

A I am unable to respond to that, I don't understand the question.

Q Let me rephrase it. The purpose of the ICS, is it not to integrate the primary and secondary system taking into account the fact that the B&W has the once through steam generator?

A Yes, I think that is right, and it really takes advantage of that responsiveness I was talking about to work with it to keep the reactor plant online.

Q So, it is true, is it not that without the once through steam generator you would not need an ICS of the type used in B&W plants?

A I really have no basis to comment on that.

Q Do you know of any plants other than B&W plants or B&W plants that do not have the once through steam generator that have an ICS that compares to the ones used at Rancho Seco?

A To the best of my knowledge, almost all pressurized water reactors would have an automatic control system of sorts, and that automatic control system would be moving the turbine throttle valves, controlling the feedwater flow rate and moving the control rods in the reactor.

The unique feature of the ICS is that it integrates those three signals to achieve an optimum response of the actuated components as I talked about this morning.

The other control system essentially does the same thing.

The difference that I have in my understanding is that the other control systems are a feedback type of control system. Where you wait for one variable to change before taking action to actuate another component and change another variable.

Q It is true, isn't it, that in considering the effect of the responsiveness created by the once through steam generator that you have taken credit for the ICS?

A We have taken credit for the ICS to enhance the plant availability of a reactor with a once-through steam generator.

Q Is it your opinion then that the responsiveness of the once through steam generator and the ICS are unrelated after a reactor trip in all circumstances?

A Well, the ICS does perform a function following a normal reactor trip.

With main feedwater available, the ICS controls steam generator level at a pre-established setpoint.

In addition to that, it controls the steam pressure through the control of the turbine bypass valves.

But, it is doing those two functions separately. It is no longer working as an integrated system. On the one hand it controls steam generator level and on a separate function, it controls the steam pressure and that, of course, is the normal mode of removing decay heat through the steam generators.

Q Isn't it true that both before and after the reactor trip, the responsiveness can effect the ICS and the ICS can effect the responsiveness?

A My definition of responsiveness was changes in reactor coolant system temperature and pressure as a result of changes in feedwater flow rate.

While the reactor is at power, the ICS does effect that responsiveness and through the feedback functions of the pressure and temperature in the reactor coolant system, the responsiveness effects the ICS, if you want to look at it that way.

Following the reactor trip, that really isn't the case, because the ICS performing two independent functions and that inter-relationship between responsiveness and the control system, really no longer exists.

The control system performs its intended function that brings the steam generator level to a certain value and controls steam pressure to a certain value; and our analysis shows that if the control system does that, that we get a normal response following a reactor trip.

Q If the control system does not do that, particularly if it does not control the steam generator level properly, isn't it true that the responsiveness would come into play at that point?

A Yes, if it doesn't perform its function as designed, it couldn't put excess water into the steam generator or not enough and then the responsiveness would come into play.

Q So, it is true, is it not that in considering the ability of Rancho Seco to respond to feedwater transients, and the responsiveness of the system to feedwater transients, that one should take into account the reliability of the ICS?

A (Mr. Jones) The analyses that are performed for the plant do not require any knowledge whatsoever about the reliability of the ICS.

They are performed independently of that systems reliability. We do an overfill calculation, and overcooling calculation to demonstrate that it can be mitigated safely.

While the function that may be causing that may be an ICS failure or an inadvertent block valved failure or something, it is not really tied to a specific function, be it the ICS, or whatever.

On the undercooling, it is basically the same way. If you have an undercooling event, the analyses are performed irregardless of how that plants got this the ICS reliability does not really enter into the safety calculations performed for the plant.

Q It does enter into the probability that you are going to get into those sequences though; does it not?

A (Mr. Jones) Yes, that is correct.

Q In discussing FMEA in response to Mr. Baxter's re-direct, on a couple of occasions, Mr. Karrasch, you referred to FMEA as examining single failures that within the ICS that would cause reactor trips.

Is that your understanding of the focus of the ICS, FMEA?

A (Mr. Karrasch) The focus of the FMEA was to show the effect on the NSS of single failures in the ICS and then show that the action of the ICS would not result in an unsafe condition.

As a result of that analysis, we showed that there were a number of failures in the ICS that would result in a reactor trip.

Q In as much as you testified that the ICS does play a role in the plant's response to feedwater transients, isn't it a somewhat different question whether the ICS can cause a reactor trip or cause an impact on the primary system than asking the question whether if an impact is generated by some other failure, whether there are failures in the ICS that would change the plant's response to that first failure?

A (Mr. Karrasch) I am going to have to ask you to try that one again. I lost the train of thought.

Q I understand your testimony with respect to the Failure Modes and Effects Analysis, it examines single failures within the ICS cabinets; and had asked the question, what would their impact be on the nuclear steam system, in particular, would they cause a reactor trip. Isn't it true that that does not necessarily answer the question of what the effect of failures in the ICS would be in a circumstance where some other failure has occurred which is required involved the response of the ICS?

A (Mr. Karrasch) I followed it pretty well right up to the last phrase of your question. Let me see if this response will answer your question.

If you remember this morning, I discussed the fact that even though we didn't include multiple failures of the ICS, that the safety analysis that was performed bounded the consequences of any multiple failures in the ICS.

So, as a result of that comparison in the safety analysis, we have concluded that mulitiple failures in the ICS cannot cause the plant to go beyond the bounds of the original safety analysis and put the plant in an unsafe condition.

Q Did the ICS assume any other failures in the system, outside the ICS cabinets?

A You mean did the Failure Modes and Effects Analysis?

Q That is correct.

A Not to my knowledge. Well, yes, excuse me a moment.

With respect to ICS failures itself, that statement is correct, it did not include things inside or outside the cabinets; but it did include failures of the inputs to the cabinets high and low, and it did include failures within the cabinets which would cause the actuated devices to fail high and low.

You have to be careful when you try and bound this thing with cabinets because the cabinets don't have any effect on the plant, it is the valves, and the pumps and the control rods that really have the effect on the NSS.

Q Mr. Jones, do you recall the question that Mr. Lewis directed to you with respect to the footnote that appears in Mr. Webb's testimony?

A (Mr. Jones) Yes, I do.

Q Has Babcock & Wilcox, to your knowledge analyzed that specific sequence, or was your answer just based on your judgment at this time?

A It is based on my judgment and an understanding of the physics that occurred during the small break accident.

MR. ELLISON: That is all, thank you.

MRS. BOWERS: Before we go to Mr. Lewis, there was a discussion earlier about the October 24th analysis report being CEC's Number 19, but I believe I did announce that it was being admitted into evidence.

(Whereupon CEC Exhibit 19 was received into evidence)

Mr. Lewis, do you have any questions?

MR. LEWIS: Now, are 17 and 19 in evidence --

MRS. BOWERS: No, 17 has been marked for identification, but 19 was offered and it was admitted.

BY MR. LEWIS:

Q Mr. Jones, I may have misunderstood you earlier today, and I just wanted to clarify one thing.

Did you state at one point that the capability to go into a bleed and feed mode was unique to B&W reactors as opposed to other types of reactors?

A (Mr. Jones) In my definition from there is basically using only safety grade equipment. It is unique. I understand that the other vendors could possibly go into it by opening the PORV at very low pressures and can perform it, but that is a non-safety feature and I was thinking using only safety equipment.

Q Mr. Karrasch, let me explore something with you which Mr. Ellison just went over.

Was it your statement that you know of no multiple failures of the ICS that could produce a situation let us say worse than the bounding overcooling accident?

You were talking about comparing it to the safety analysis report and I think you were probably talking about a bounding overcooling accident.

A Yes, Mr. Lewis, that is correct.

Q Was it your statement that you know of no multiple failures of the ICS that could produce more serious effects than the bounding overcooling accident?

A The statement I meant to make and I hope I did make was that the safety analysis performed in the Rancho Seco FSAR is intended to bound any event or any accident which could be caused by a multiple failure in the integrated control system and that includes overcooling, steam line break, loss of coolant accident.

The complete spectrum of accidents in Chapter 15 of the FSAR.

Q Right, well what I am trying to clear up is that you have also testified that you have not conducted -- that B&W has not conducted an analysis of multiple failures of the ICS; isn't that correct?

A Yes, that is correct.

Q So, your statement is more qualitative in that sense, that is you have not done an actual analysis of multiple failures of the ICS to thoroughly determine that there is no such multiple failure scenario that might not exceed that accident.

A Well, the reason I can make the statement that I did with confidence is that the safety analysis does not really include the effects of the ICS.

If you were to completely unplug it from the plant, if you will, the results presented in Chapter 15 of the FSAR would still be valid and bounding.

Q Let me ask you this --

MRS. BOWERS: Will you please use the microphone?

MR. LEWIS: Okay.

BY MR. LEWIS:

Q I didn't mean to cut into the middle of your answer, but can you think of no way in which any failure of the ICS could put the plant in a worse condition than it could be if there were no ICS at all?

A Yes, that is really the basis for my statement this morning.

MR. LEWIS: Those are my only questions.

MRS. BOWERS: Mr. Baxter, we are back to you, I might also tell you that the Board has some questions.

MR. BAXTER: I have nothing further.

DR. COLE: I will try to be brief, Mr. Karrasch and Mr. Jones, I know you have been on the stand for a couple of days.

On Page 4 of your testimony, on Lines 7 through 10, you state that additionally the frequency of feedwater transients resulting in reactor trip is not greater for B&W systems compared with other pressurized water reactor nuclear units.

Is your basis for that statement the data that is on Page 14 of your testimony; or is it some where else?

MR. KARRASCH: The basis for that is on Page 14 of the testimony plus our understanding of the favorable operational experience with the integrated control system.

It is not specifically just that data, but the statement is being made with respect to looking at that data and just our general knowledge, if you will, of the performance aspects of the ICS.

DR. COLE: How did you happen to pick 1978?

MR. KARRASCH: The reason we picked 1978 was in direct response to the contention that was originally put forth, Dr. Cole.

The question asked us to look at the number of feedwater transients over the past year and we wanted to respond directly to the question, but really what we feel the important consideration here is really not the frequency of these transients, but the important consideration is the capability of the design to safely accomodate the transients.

DR. COLE: On Page 5 of your testimony on Line 11, you refer to void formation and you are considering only steam there, aren't you? Steam voids, condensible fluids, condensible gasses?

MR. JONES: Well, the statement generally refers to steam voids, however, as is stated within the testimony in the void section, we have included, or looked at the effect of non-condensible gasses and if I can provide a reference.

We have looked at, at least specifically, at the effect of non-condensible gasses during a LOCA and that is addressed on Page 47 in the paragraph starting at Line 12.

DR. COLE: Do you expect you might get any considerable amount of non-condensible gasses, and still be able to make the statement that you make on Page 5?

MR. JONES: Well, the only mechanism that we see to generate a significant amount of non-condensible gasses would be either a total emptying of the core flood tanks, which does not occur for almost, as far as I know only for very big size LOCA's which would not rely on the steam generators for heat removal anyway; or, by circ metal water reaction analyses that have been

Performed under Appendix K shows the cladding temperatures are maintained to approximately 1100° and essentially no metal water reaction occurs.

DR. COLE: All right, thank you.

On Page 9 of your testimony, on Line 10 you referred to regulating control rod assemblies on Lines 10 and 11 --

MR. KARRASCH: Yes, sir.

DR. COLE: Are these special rods, or are these just the normal reactor control rods to which you have attached?

MR. KARRASCH: Those are a portion of the normal reactor control rods.

The Rancho Seco plant has 69 movable control rod assemblies mounted near the reactor head. The control rod drive mechanisms and the control rods in 69 of the 177 fuel assemblies.

Twelve of those rods are designated for normal control of the Doppler coefficient to maintain changes in fuel temperature and maintain a power level in the core.

DR. COLE: And these are the ones connected to the ICS system?

MR. KARRASCH: That is correct.

DR. COLE: Okay. Mr. Lewis was asking you some questions about the ICS system and whether you could postulate the situation where having the ICS system installed might aggravate a situation. Now, with respect to say a turbine trip or a loss of feedwater, what would happen to the ICS system, or the ICS?

MR. KARRASCH: Well, on a turbine trip, or a loss of feedwater, the reactor trips now immediately and the control rods, all 69 of them drop into the core.

DR. COLE: Let me tell you what my concern is. If, for example, on a turbine trip, the ICS controls feedwater and flow to the turbine, it interacts with the control rods and if one system diminished like steam flowing to the turbine, would the ICS react to pull out the control rod to provide more power to make more steam which is now lower than it was or what happens?

MR. KARRASCH: Do you want me to try and give you a description of plant response prior to a reactor trip, because once the reactor trips, the control rods unlatch and are totally separate from the ICS.

They drop into the core by gravity, and then the only purpose that the ICS plays following that reactor trip is to control steam pressure with the turbine bypass valves and steam generator level with the startup control valve.

Prior to a reactor trip, the ICS does things to a turbine control valves, the mainfeedwater flow control valves and the control rods to meet the demand of the station for megawatts.

Can you clarify your question a little more as to whether you want to discuss the response prior to a reactor trip?

DR. COLE: Well, several places in your testimony you say that the role of the ICS is minimized during an accident situation.

Could you explain to me what you mean by that statement?

MR. KARRASCH: By that statement, I was referring to following a reactor trip.

DR. COLE: Okay, now what happens following a reactor trip?

MR. KARRASCH: Following a reactor trip, the turbine trips immediately on a signal from the reactor protection system and the integrated control system sends a signal to the main feedwater flow control valves to close very quickly to turn off feedwater.

The reactor power has been reduced very quickly due to the reactor trip and so now the demand for feedwater flow is also reduced very quickly.

So, the ICS tells those main control valves to close within ten seconds or so.

In addition to that with the turbine now tripped, you have closed off your heat sync for the steam. So, the ICS tells the turbine bypass valves to open to relieve that steam directly to the condensor and to bypass the turbine.

You are then, for the first 30 seconds or minute or so, you are in a condition where feedwater flow is being reduced very quickly, steam pressure has increased due to the loss of the heat sync subsequently relieved out the turbine bypass valves and then controlled at a constant pressure of a thousand pounds.

So, say beyond a minute or two after the trip, the ICS is controlling steam pressure at a thousand pounds with turbine bypass valves and the ICS is controlling steam generator level.

DR. COLE: With feedwater flow.

MR. KARRASCH: Yes, with feedwater flow through what is called the startup control valve. Those are really the only two things the ICS does afte the reactor trip.

DR. COLE Okay, thank you.

MR. KARRASCH: I would like to clear the record about my 69 control rods dropping in. I guess I have to be honest here.

There are eight control rods which will not drop in, they are called axio power shaping rods, and are designed to stay in position on a reactor trip.

So 61 of the 69 control rods dropped in on a a reactor trip.

DR. COLE: But 61 is all you need.

MR. KARRASCH: Yes, sir.

MR. SHON: Just out of curiosity, about how much reactivity is that worth at various times in the lifetime; do you know?

MR. KARRASCH: The total wo:) of all the control rods?

MR. SHON: Of the SCRAM rods, in other words, how far down do you shut, when you shut?

The reason I ask is that it has some bearing in that the prompt drop in reactor power is not really to zero or even to decay heat, but is to a number governed almost directly by the number of dollars worth of reactivity plopped in and I am just curious as to that number.

MR. JONES: Well, the safety analyses generally are performed using a Delta K over K -- I forgot how to convert to dollars of about three, but the rods themselves, there are stuck rod assumptions worse time in life and stuff like that, typically, it is about five Delta K over K. It is a 5% shutdown margin.

MR. SHON: 5%, not a factor of five. Thank you.

DR. COLE: I am looking at Board Question H-C-16 on Page 7 and the contention states that SMUD, the Licensee, has done insufficient analysis of the failure mode and effects analysis of the integrated control system.

Would it be your position, sir, that the document entitled CEC Exhibit Number 3 would be evidence to indicate that a sufficient analysis was done; or is there insufficient and/or are there others?

MR. KARRASCH: Our testimony is intended to state that the Exhibit that you have before you was the sufficient failure modes and effects analysis that was performed.

DR. COLE: Even considering the fact that it took only a single failure direction? One failure at a time, no multiple failures?

MR. KARRASCH: Yes, sir, that is correct.

DR. COLE: Is that your position?

MR. KARRASCH: Yes.

DR. COLE: On Page 15, Line 12, the sentence reads--Important considerations in assessing the the expected impact of the transient are-- and then you go on to describe certain things.

The considerations are really assumptions, aren't they? Would assumptions be a better word in there? Isn't this an assumed condition and then you describe the things that are your assumptions?

MR. KARRASCH: No, these are more results, I would say Dr. Cole than assumptions. Possibly the word, considerations isn't the accurate one here.

As a result of the loss of main feedwater sequence that I describe on Page 14 and at the top of Page 15, the following are statements of fact as a result of that feedwater transient. That was the intent of those five items in Lines 14 through 22.

DR. COLE: This is what you said will occur?

MR. KARRASCH: Yes, that was meant to explain first the normal sequence of events on a loss of feedwater and then to try to highlight what we felt were the important results of that transient.

DR. COLE: Okay. On Page 16, Lines 9½ and 10 is a part a of a contention of a question which asks -- what changes in the system and procedures have been made to ameliorate this situation?

Your response to that is really on Page 22 to 24; is that correct, sir? Starting at Line 17, Page 22 and continuing to Line, maybe the rest of the Page on Line 24.

MR. KARRASCH: Yes, Line 15 on Page 24, that is our intent. That is our response to that question, yes, sir.

DR. COLE: On Page 22, where you talk about some of these consequences of TMI ᴊu indicate that the high reactor coolant pressure trip setpoint was lowered and the setpoint for the pressurizer Pilot Operated Relief valves was increased.

Do you know what those values are at Rancho Seco, sir?

MR. KARRASCH: Yes, sir, the high reactor coolant system pressure was changed from 2355 to 2300 psig and the opening set pressure for the PORV was changed from 2255 to 2450 psig.

DR. COLE: So, when would the pressurizer relief valve ever be used?

MR. KARRASCH: Hopefully, it won't. The only time it would be used would be on a pressure excursion which went very close to the safety valve setpoints of the pressurizer which are set at 2500 pounds.

As we described during the cross examination this week, such an event might be an extended loss of all feedwater where the reactor coolant system heats up and pressurizes to that h:gh setpoint.

DR. COLE: Okay, thank you.

On Page 23, Lines 10 to 13 you indicate that procedures have been developed and implemented for initiating and controlling auxiliary feedwater independent of the integrated control system. Could you just briefly tell me what they do there, how that is done?

MR. KARRASCH: I think I would really prefer to address that to the Licensee, if it is okay. I have brief general knowledge of what was done, but it is really not specific.

DR. COLE: That would be fine, sir. I will take that up with either Mr. Rodriguez or Mr. Dietrich.

MR. BAXTER: I have a comment for you, Dr. Cole. This issue of the independence of the auxiliary feedwater control from the ICS was the subject of a motion for summary disposition and we had an affidavit which was granted by the Board on February 6th, and we had an affidavit in there by Mr. Rodriguez which attach the procedured in fact.

DR. COLE: Well, if I recall the contention, I think the contention was that it was not independent and then the information you provided indicated that yes, it was. So, that was done and we had no contradictory evidence, we had to grant that, but if the information that I need is in there, I will look at it.

MR. BAXTER: I understand, I think they said we did not have the procedures and I think we attached them, that is all. For your information, they are there.

DR. COLE: On Page 24 at the top of the page, sir, you indicate that additional small break LOCA analyses have been performed. Are you referring to NUREG 0565, and is it 0623, or what are you referring to there?

MR. JONES: Well, those are the Staff's conclusions of the analyses we have performed, they are in several places, but the biggest one is a May 7th report which was issued -- I don't even want to try and remember the title, it was Small Break LOCAs and Transient Behavior in B&W's 177 fuel assembly plant, that is the best I can give you for the title.

There were additional letter reports which were in response to 7905C which is the reactor coolant pump trip studies, the delay trip in the reactor coolant pumps.

Those are the analyses.

DR. COLE: All right, sir.

On that same page you refer to a pre-TMI scenario, and that scenario is described on pages 19 and 20, is that correct, sir?

MR. KARRASCH: Yes, sir, that is correct. The sequence on 19 and 20 was meant to describe a scenario prior to the modifications made subsequent to TMI 2 and the scenario described that you just referred to was meant to for comparative purposes with the modifications that have been made.

DR. COLE: I understand, sir.

On page 26 with respect to part A of additional Board question 1 on line 7, I read in your testimony that Mr. Dietrick will be discussing installation of these and operating experience, I assume at Rancho Seco.

Do you have any first hand information on the performance of control grade trips?

MR. KARRASCH: Yes, sir, I do. It is current to about a month ago. I suspect. But, to the best of my knowledge since the control trips were implemented we have had approximately 12 challenges to that control grade trip system and we have had one failure of the reactor trip. That one failure occurred at our Arkansas nuclear station last summer.

MR. COLE: So, the first 4 you had one then your 8 in a row, huh?

The first 4 times it was challenged you had one failure, is that correct, sir?

MR. KARRASCH: Oh, yes. That is right.

DR. COLE: So, you say you have had 8 subsequent events and no failure in that period?

MR. KARRASCH: To the best of my knowledge, yes, sir.

DR. COLE: Okay.

Now, page 29, sir, with respect to the design of the pressurizer, the basic design criteria that I believe you set out briefly here, does not consider any loss of fluid in any one of the criteria that I see here, is that correct, sir?

MR. KARRASCH: No, I don't think that is correct, Dr. Cole.

If you look at line 9, that portion of the size and calculation is performed to define the maximum expected contraction in reactor coolant line following a reactor trip. During a normal reactor trip, there is a loss of fluid inventory from the pressurizer due to the fact that you have shut down your heat source very quickly.

DR. COLE: I realize, and it flow back into the reactor. But, I am talking about out of the system.

MR. SHON: An actual loss of material from the reactor coolant pressure boundary is what Dr. Cole is talking about.

MR. KARRASCH: No, sir, that is not included in the design basis of the size for the pressurizer.

Are you referring to a loss of coolant accident?

DR. COLE: Yes.

MR. KARRASCH: The pressurizer design basis does not account for fluid inventory loss due to a break in the system.

DR. COLE: Can you think of any scenario wherein you would lose fluid through a pilot operated relief valve or a safety valve and then, by that when the system contracts to then drop it entirely out of the pressurizer lose all the fluid in the pressurizer?

MR. JONES: If you have an overheating transient, or an undercooling transient, and exercised the PORV and go into a feed and bleed mode of operation to keep core cooling, if you are in that operation for an extended period of time, and then get the auxiliary feed water system on line and start spraying into the generator, you could easily pull the volume out of the pressurizer.

You could, depending on how long you want to withstand such a transient, if you don't get along within about I think it is 10 minutes or so, then you would have a potential in the subsequent cool down of the system of draining the pressurizer. But, in that draining process, you will depressurize the system below the high pressure injection setpoint, well, actually you have got it on, it would just refill itself, is what will happen to the system ultimately.

DR. COLE: Wouldn't you then have voids in the upper head of the vessel, though?

MR. JONES: You may have voids in the upper head of the vessel for that type of a transient, but we have the vent valves up in the upper head of the vessels so that bubble cannot really migrate down into the reactor core unless the entire system is draine. of liquid.

DR. COLE: You say you have vent valves?

MR. JONES: Between the upper head and the down comer you have vent valves. If you create a steam bubble in the reactor vessel in the upper head due to boiling in the core, for example, which is something that happens every time in any small break LOCA analysis, that regions becomes a high -- attempts to pressurize. When that region pressurizes or attempts to, the valves will open and the steam will be displaced into the downcomer region and ultimately the steam will migrate back up and bubble up through the system or if there is a break in the system, out to the break.

So, that the bubble cannot migrate down into the reactor vessel because it keeps hitting these paths to escape, it has got the vent valves, and it has got the correlate nozzles, the hot leg nozzles, so, it has got many opportunities to bleed out of the reactor vessel without uncovering the core.

DR. COLE: Is there any plan to provide direct release of any collected gases near the top of the reactor vessel? Can that be done?

MR. JONES: At this time, I believe it is an NRC recommendation in one of the documents and I believe it is still being evaluated. I don't know its current status.

DR. COLE: But, there is a valve at the top that can be opened, is that correct, sir?

MR. JONES: At the top of the upper head? I am not really sure.

DR. COLE: How do you fill the reactor vessel initially?

MR. KARRASCH: There is no remotely operated valve at the top of the reactor vessel head.

DR. COLE: I didn't say remotoly operated.

MR. KARRASCH: Okay, yes. There are event on a number of the control rod drive mechanisms to vent gases out during the initial fill of the system, but that is a local manual operation right now.

DR. COLE: You can't get people to do that, and we wouldn't let them do that that way, right? Okay.

MR. JONES: Not me.

MR. KARRASCH: That is a low pressure operation right now during start up operations.

DR. COLE: On page 32, sir, line 13, you again refer to analyses that had been performed and could you provide me some guidance as to what references you used there, sir? What is your basis?

MR. KARRASCH: Yes, B & W has performed analysis of the nuclear steam system under natural circulation conditions for times when all the reactor coolant pumps are inoperative and natural circulation is required to remove decay heat.

The results of those analysis are reported in the safety analysis reports.

DR. COLE: What safety analysis reports, sir?

MR. KARRASCH: The safety analysis report for the original operating license of Rancho Seco.

DR. COLE: Thank you.

On page 43, line 4, you referred to operating guidelines, could you provide me with a reference there, sir?

MR. JONES: Those are basically the guidelines that were in CEC #18, I believe. The small break operating guidelines contain such information. It is really a very simple statement which is, if the system is saturated, actuate the high pressure injection.

DR. COLE: Page 55, sir. The beginning of page 4 you describe how the system would perform, and then in lines 1 through 18 you have statements concerning your characterization of the system.

The last item you have mentioned is injection flow by one high pressure injection train matches core boil off at approximately 1000 seconds.

You continue with the high pressure injection, but then what?

MR. JONES: You mean after that time?

DR. COLE: Yes, sir.

MR. JONES: You would get a subsequent refilling of the primary system, it would be a slow process as the decay heat boils off.

The operator can aid in that process by either establishing a second train of high pressure injection or after the system is stabilized, he can depressurize the primary system which will give him more high pressure injection flow and, again, allow the system to refill quicker. That is the scenario.

At this time what you have is a stable situation where all that will happen is you will just start to refill the system with inventory.

DR. COLE: The high pressure injection system, the of water for that system is the borated water storage tank?

MR. JONES: That is correct.

DR. COLE: Say you had a leak in the system through a pilot operated relief valve or a safety valve, how long would you be able to pump into that system with the finite storage of the borated water storage system, which is about what, 60,000 gallons?

MR. JONES: No, it is on the order of 300,000 gallons, 350,000 gallons at maximum design HPI flows with two pumps it would be 1000 gallons gallons a minute. So, you would have 300 minutes or roughly 5 hours.

Once you have depleted, I am sure you do not go and empty the tank all the way, you usually leave some inventory left in it. When you get to a lower level in the tank, you switch over to the sump and you use the low pressure injection pumps to provide suction -- well, low pressure injection pumps pulls their suction from the reactor coolant sump and then, provides suction to the HPI pumps and then into the primary system.

DR. COLE: So, then you have got to continue a system going, and you could go for whatever time period you need?

MR. JONES: Right.

DR. COLE: All right, sir, thank you.

On page 58, on line 15, you refer to the ANS standard value, and I believe you characterize that as a -- let me not say how you characterize that, sir.

Let me ask you how you characterize that?

MR. JONES: What is it? Well, Appendix K gives the exact definition to the standard that I am talking about. It is the ANS 5.1 standard and it's date is,at least a draft standard, came out in about 1971. That is the standard we are using without the normally employed 1.2 times that value for conservatism in Appendix K.

DR. COLE: So, could you then say that using a core decay heat of 1.0 times the ANS standard value is not conservative but accurate?

MR. JONES: It is, well, remember the event we are looking at here has essentially 3 failures in it. It is a multiple failure scenario. It is both auxiliary feedwater pumps are lost, actually as a 4th failure because you have loss of offsite power which loses your main feedwater system, you have lost your aux feedwater system both trains, and you have a single failure in the high pressure injection system.

It is a multiple failure scenario and we did not use the normally conservative 1.2 ANS standard.

However, as I stated previously, I believe, that the actual ANS standard that the 1971 vintage standard in comparison to today's standard shows that it is conservative or best estimate in general.

MR. SHON: Excuse me a moment, Dr. Cole.

There was, I believe, in about 1974 or 1975 a very considerable flat when a fellow whose name, I think, was Englander, suggested that the ANS decay standard was an error for not having failed to consider certain activation products.

Do you recall how that was resolved?

MR. JONES: No, I can't. I expect it was Dr. Shore. He is the only Englander I know of who usually works with decay heat standard but the new standard that has been developed, the way I understand it, has accounted for all major products.

MR. SHON: Thank you.

DR. COLE: Sir, in table 2 , and table 3 and table 5 you used a value of 1.2 times the decay heat or 1.2 times the ANS standard value to estimate the decay heat. Was that because of the 20% aeroband estimate?

MR. JONES: Yes, that is the typical Appendix K assumption which is 120 % of the ANS standard.

DR. COLE: You indicated that recent information indicates that the ANS standard value is, the aeroband is less than 20% or at least you intimated that.

MR. JONES: Yes, my understanding of the new curve, number one, the new curve generally is lower than the old ANS standard and then, the aeroband is down to, I think, 2 or 3%. It is a very small aeorband at this time.

DR. COLE: So an acceptable value now, in your opinion, is 1.0 times the old value?

MR. JONES: That's correct.

DR. COLE: All right, sir, thank you.

I have no further questions at this time. Thank you.

MRS. BOWERS: We will take a ten-minute break before Mr. Shon begins.

(Whereupon the Board had a ten-minute.)

MR. SHON: I just went back on the record now. I had a few questions I wanted to ask. I think not many actually most of the questions I noted down were asked by other people already. It's been rather thorough cross examination.

The first one is something that Mr. Ellison touched briefly on but I wanted to fix it more exactly in my mind. It has to do with CEC #18.

The pages there are not well numbered but it is figure 1 of Appendix A. A plot of pressure versus temperature for the pressure temperature limit curve.

All I wanted to be certain of was that in one of the instances in which you have said a reactor operator could repressurize the system using the high pressure injection pumps which pump in cooler water, there is no situation in which the action of the pumps would be such considering the temperature that they are causing to exist within the system and the pressure that they can generate that you would move from the region marked acceptable to the region marked unacceptable on this curve. Is that correct, that you could envision no such situation?

MR. JONES: I don't believe I quite said that, Mr. Shon.

You can and will as time progresses possibly move to the unacceptable region. It would take several hours to get there. Even under the action of all high pressure injection pumps. These instructions are guidelines for the operator to prevent him from moving from the acceptable to the unacceptable region, but if they didn't take any action, then, of course, you could cross that boundary eventually.

MR. SHON: I see. So, it could happen but it wouldn't happen soon, is what you are saying?

MR. JONES: Yes, it wouldn't happen soon and the operator has been instructed to take this action.

MR. SHON: I would like to now direct your attention to CEC Exhibit #6, specifically graphs 8 and 9, I think they do not have page numbers.

In particular the line on the graph labeled 1 in each case. As I understand graph 8 which shows the upper head steam quality as a function of time after about a 150 seconds or so, the upper head steam quality is 1, is that right?

MR. JONES: That is what the figure indicates, yes, sir.

MR. SHON: On number 9, at about 150 seconds the pressurizer level is just about coming up and the pressurizer is starting to refill, is that right?

MR. JONES: That is what the curve indicates.

MR. SHON: This suggests that there are some region, and possibly even a long region in time, since the curves are not continued past 170 seconds, in which one could have a bubble in the upper head, and still have a positive indication of pressurizer level, is that right?

MR. JONES: That is what the curves indicate. I am not familiar with the details of the Brookhaven model to be able to state whether it categorically represents the mixing phenomena that would normally occur up in that region.

I find it difficult to accept this analysis on face value.

MR. SHON: Did any of your analyses show a similar thing, that is a steam quality of one in some portion of the core pressure vessel at a time when the pressurizer was filling?

MR. JONES: The only analyses that would have such a -- that would exhibit that sort of effect would be a small break where the very small breaks, these are one square foot type breaks, when you interrupt the circulation path, well, in a small break you get down the saturated conditions, you do boil vigorously in the core, and when you interrupt the circulation path you create a bubble in the upper head.

But, for these breaks when you interrupt the circulation path, you also start a system repressurization which does result in the pressurizer refilling. When you get to the reflux boiling mode of operation, you do have a bubble in the upper head and as you cool down the system with time, the pressurizer is empty. But, you can have a bubble there during a small break accident.

MR. SHON: With regard to the notion of how the operator determines subcooling, we said several times or you said several times, that he does this or that the Psat/Tsat meter does this by looking, I think you said, at hot leg temperature and pressure. Is that right?

MR. JONES: That's correct.

MR. SHON: The hot leg temperature is actually a mixture or an average over the upper core face exit temperatures, is it not?

MR. JONES: No, the hot leg temperatures are measured up near the candy cane region, up around the 180 degree U-bend.

MR. SHON: Well, that makes it even more in the direction that I was going to ask you about. Since the water there has actually mixed from when it left the core and traveled some distance from the core, is it not likely that that temperature would be lower than temperatures further on down in the system?

MR. JONES: It could be slightly lower, yes. If the pumps are running, it would be basically the same temperature, if the reactor coolant pumps are off there would be some transport delay time before you would see the heated core temperature.

MR. SHON: But, it is also true that the core is not uniformly hot, exactly?

MR. JONES: No, it will not be uniformly hot, no. It would be difficult to say that it was uniformly hot.

MR. SHON: Since this then represents water that comes from various places in the core and has mixed before it gets to the measurement, there will be temperatures in the core that are higher than and that are lower than this particular thing, or so one would expect thermodynamically from an intuitive standpoint. Is that right?

MR. JONES: Yes.

MR. SHON: What assurance do you have that even though you see subcooling when you look at the panel board everything is subcooled at the hottest point in the core, or do you have such assurance?

MR. JONES: No, you may not be able to assure that but the consequences are you may have some local boiling within the core region. Those bubbles will be capable of mixing and condensing with the surrounding fluid.

If you have a void in the core, it will only be local bubbling and the core would stay cool at that time frame.

MR. SHON: Thank you.

I would like to have you take a look at what has been labeled CEC #17, I realize that that is not the document that ultimately got into evidence but I trust the one that did has something similar in it.

On almost every case, case 1, 2, certainly and -- well, certainly in cases 1 and 2 that are listed in that document, the decay heat was assumed to be at beginning of life after 31 equivalent full powered days, and was taken, well, in at least one of them, to be half of the ANS standard as I recall it at 10 minutes.

What justification do you have for doing this calculation so early on in life when the decay heat is less than it would be later on in life? There doesn't seem to be any reason why such an accident should happen early rather than late, is there?

MR. JONES: No, there isn't. The main reason for the choice of beginning the life assumption, was to minimize any heating of the fluid by the core which would minimize the delta row from a natural circulation standpoint.

Also, it was chosen to maximize the shrinkage effects, this is the same type of concept minimize the delta T between the hot and the cold regions so that you will get more shrinkage.

MR. SHON: But, it would seem that at the same time that you minimized the delta T, you also minimized the amount of heat which it was necessary to transfer, and hence it is not obvious to me that these two things are not in some measure compensating for one another that the lesson heat output you have assumed does not give your estimate an optimistic bias any way. Has any sort of sensitivity study been performed to tell whether later decay heats represent less desirable situations?

MR. JONES: I am not really sure whether any other -- any different times in life have been examined or not.

It is my belief that you are correct in the extent that they probably do somewhat self compensate for each other.

The judgment that was made was that this would provide a worse case by maximizing the overcooling. But, as stated previously, since this is a non-LOCA event, it is a break of the secondary side, it almost means that you are going to recover from it with no problems any way.

You could shrink the primary system down to cold temperatures and not uncover the core and keep it cool.

MR. SHON: Well, the measures that you had sort of used for the severity of the thing, I thought, was the total amount of void formed in each case, wasn't it?

MR. JONES: No, the measure that was being used to bench moth these was not the amount of voids, but rather we would maintain circulation through the core and keep the core cooled. Not necessarily how many cubic feet.

By minimizing that delta row, you minimize the -- excuse me, my mind just flipped in.

On sensitivity, we have examined the sensitivity to zero decay heat which is in the other direction and my understanding of that analysis is that you do get a larger void formation for the known decay heat case.

Where we shut off the decay heat within 100 seconds, we ended up with a bigger void for this type of an event, than using a, I am not sure what the comparison was, but anoth:r case with decay heat in the model.

So, use of the lower decay heat such as is 31 effective full power days is conservative from the standpoint of void information and natural circulation.

It just slipped my mind.

MR. KARRASCH: Mr. Shon, you have another piece of information there that tends to lead you to the conclusion that the heat imput is not as important as the cool down rate here. That is the fact that with the reactor coolant pumps on, you have a greater cool down rate.

The reactor coolant pumps are putting in about the same power into the coolant that the decay heat of the core is.

MR. SHON: That's correct, I realize that to be true. Yes.

There is one matter that seems to me fundamental to all the treatments that I have seen and that you have mentioned in your codes, and in your modeling of these transients and accidents.

If I am not mistaken, and correct me if this impression is wrong, you assume thermodynamic equalibrium in each node, do you not?

MR. JONES: Yes, that is correct.

MR. SHON: Now, the Staff has pointed out some consequences that might ensue because of this assumption in particular the business of steam bubbles in a subcooled liquid that could collapse or slugging that could occur, both of these are water hammer matters, and you have given us some assurance that these water hammers are not serious, is that correct?

MR. JONES: I believe so, yes.

MR. SHON: What about the more general question? The question that says, you know there must be cases, for example, in the emptying of the pressurizer and the blowing of pressurizer steam into the hot leg in which the node, at least the node nearest the pressurizer entry is not an equilibrium. Therefore, you code is not calculating the right thing.

How much of an effect can this have, and in what direction?

MR. JONES: I don't think that the magnitude of the steam flow is coming through the surge line.

My guess is that for a LOCA event, it probably is not significant because you go to saturated conditions as a matter of cost in the event. So, the steam bubbles coming in if it does create a momentary void will not really have any significant effect on the LOCA calculations.

On the non-LOCA events as I stated previously, it is just my opinion that in all likelihood that that steam bubble will collapse as it rises through the 30 cubic, well, the 30 feet of cold water that exists in the hot leg.

It is possible that they may momentarily exist, and if it does, it should tend to slow down the depressurization somewhat which would minimize the effect of the overcooling transient.

MR. SHON: Is there any chance that the operator guidelines that had been developed following the kind of calculations you done could be in error or could be misleading in any way because of non-equilibrium effect?

MR. JONES: No, I don't really believe so, and the reason for that is that the guidelines not only state that the system must be 50° subcooled which is a fairly large subcooling margin and should be capable of easily condensing any of these type of voids that were considered, but they also require that the pressurizer level will be back on scale at a minimum of a 100 inches.

The effect that we were just discussing is when the pressurizer is empty and we are not telling the operator to shut the pumps off until he has his pressurizier level back.

MR. SHON: When we mention an empty pressurizer, often it happens on a depressurization or cooling transient, is that right?

MR. JONES: That is correct.

MR. SHON: I don't know exactly what the Rancho Seco system looks like, but when pressure tends to be lost in a system like this, as a rule, the pressurizer heaters automatically come on. Do they there, too?

MR. JONES: Yes, that is correct.

MR. SHON: If the pressurizer empties, while the heaters are operating, is there any chance to damage to the heaters that might later make it difficult for the operator to maintain proper temperature and pressure control?

MR. KARRASCH: Yes, there is.

We have installed an automatic cut out of the pressurizer heaters at Rancho Seco, or the licensee has installed it , to assure that if the level falls below the point of the pressurizer heaters,that the power will not be supplied to the heaters.

MR. SHON: Okay, I just wanted to make sure it was there. I don't think that device had been mentioned previously. If it had, I don't recall it.

CEC #15 and 16, at figures A and B-3 and A and B-4, yesterday, under Mr. Ellison's cross examination, you were asked to compare the results given here for the MINI-TRAP and MAXI-TRAP codes.

It was noted that only MINI-TRAP was carried on in time beyond this region, and some interesting things happened in times beyond this region.

That MINI-TRAP and MAXI-TRAP didn't track one another exactly, the MAXI-TRAP was a more detailed noding with more flow paths. You were asked whether because in each case figures 3, 4, and 5, the MAXI-TRAP values lay below the MINI-TRAP. Whether this did not indicate that MAXI-TRAP was more conservative, that is, less optimistic than the code you actually used. And, you immediately said, yes, it was that way.

It is not obvious to me when you have both a falling temperature and a falling pressure that the thing that you are looking for which is saturation, is necessarily aggravated by the lower of the two curves.

Could you clear this up for me? How do you know without steam tables?

MR. JONES: What I did not do yesterday, and I should have done, was do another comparison which is figure AB-5. What you find when you compare figure AB-4 and AB-5, is that the system pressure is basically being controlled by th pressurizer level as you would expect. About all the difference you are going to get between these two codes because they are tracking each other fairly reasonable in time is basically a time shift.

MAXI-TRAP would get you to 1600 Psi earlier which would get your ESFAS on earlier, but you would probably get the same effect as far as you will probably empty the pressurizer and it will recover in a 25 or 30 second time frame. I believe that is the times in here.

Within the time frames listed in here. It was a 50 second it might be 55 or 45 on either side, but I don't think it would be significantly different in either model.

MR. SHON: AS I understand you now, you are essentially telling me that it is not true that the more detailed code is necessarily more conservative after all?

MR. JONES: I am not really sure as to if I had to choose which one was conservative now that I have looked at that other curve last night, I believe that they would basically be equivalent and I am not sure I would want to make a judgment on which one of these two are conservative.

MR. SHON: Lastly, you got into a discussion this morning with Mr. Ellison, about whether or not the figures presented in 0565 were figures or were a review of a report to show compliance with 5046 in Appendix K and you felt they were not. Is that right?

MR. JONES: That is correct.

MR. SHON: Is there an analysis currently accepted for small break LOCA's that does comply for Rancho Seco with 5046 in Appendix K?

MR. JONES: To the best of my knowledge, yes, sir.

MR. SHON: I just wanted to be sure of that. Thank you. That is all I have.

MRS. BOWERS: Let's go to the parties to determine if there are questions based on Board questions.

Mr. Baxter?

MR. BAXTER: I have no further questions.

MRS. BOWERS: Mr. Ellison?

MR. ELLISON: No, ma'am.

MRS. BOWERS: Mr. Lewis?

MR. LEWIS: I would like to -- did you want to take a break?

MRS. BOWERS: We will take a ten-minute break.

(Whereupon a ten-minute recess was taken at 3:00 p.m.)

(Wherupon the Board resumed at 3:05 p.m.)

MRS. BOWERS: Mr. Lewis, Mr. Novak was previously sworn but I don't believe anyone else has.

MR. LEWIS: Yes, let me introduce them quickly and then you can do so.

Nearest to the Reporter is Mr. Thomas Novak, who has been previously sworn. Next to him is Mr. Mark Rubin, Mr. Robert Capra, Mr. Dale Thatcher, Mr. Phil Matthews, and Mr. Paul Norian.

This panel will be testifying on all of the so-called Category 1 issues, in addition, we will be sponsoring through Mr. Capra the June 27, 1979 evaluation of licensees compliance w h the May 7th Order.

Also, we are going to sponsor at this time, Mr. Capra's testimony in response to FOE contention 3-C.

The thought there is that many of those items would entail testimony, backup testimony from some of the same people who are here sponsoring their own testimony.

Madam Chairman, may we swear the remainder of the panel?

Whereupon,

MARK RUBIN

ROBERT CAPRA

DALE THATCHER

PHIL MATTHEWS

PAUL NORIAN

were called as witnesses, having been duly sworn, was examined and testified as follows:

DIRECT EXAMINATION

BY MR. LEWIS:

Q Let me begin with Mr. Capra. If you are having any trouble hearing me, tell me, because I am going to have to be moving back and forth here.

Mr. Capra, let me show you a document entitled, "Robert A. Capra, Professional Qualifications". Would you please look at that and tell me whether or not that is a true and correct statement of your professional qualifications?

A (MR. CAPRA) This is a true and correct statement.

Q Thank you.

Mr. Capra, let me show you a document entitled, "Evaluation of Licensees Compliance with NRC Order", dated May 7, 1979,"Sacramento Municipal Utilities District, Rancho Seco Nuclear Generating Station" dated June 7, 1979.

Are you familiar with this document?

A Yes, I am.

Q Could you please state the nature of your responsibilities with respect to that document?

A This document was prepared by the Bulletin and Orders Task Force the Office of Nuclear Reactor Regulation. During that period of time, on June 27, I was the Babcock & Wilcox project manager for the Bulletins and Orders Task Force.

I assisted in the final preparation of that document.

Q Are the statements contained in that document true and correct to the best of your knowledge and belief?

A To the best of my knowledge, they are.

MR. LEWIS: Mrs. Bowers, I would like to offer for inclusion in the record -- let me say, this has already been included in the record as if read. It was included on the transcript of February 27th following transcript 362. I believe it was the 27th. It was Thursday.

MRS. BOWERS: The 27th?

MR. LEWIS: The 27th, yes.

It appears following transcript 362 and also following the written statement of Mr. Mattimoe, but at this time I would like to have it admitted into evidence in this proceeding.

DR. COLE: Do you have extra copies of that, with you now?

MR. LEWIS: I am afraid I do not.

MR. CAPRA: I have a copy with me, if you want to use this one.

MRS. BOWERS: Let me check with Mr. Baxter.

Mr. Baxter?

MR. BAXTER: I have no objection.

MRS. BOWERS: Mr. Ellison?

MR. ELLISON: No objection.

MRS. BOWERS: Well, the document which you have identified will be physically inserted into the transcript as if read and it should mention it is already in the February transcript.

(Whereupon said document was inserted in the record.)

BY MR. LEWIS:

Q Mr. Capra, do you have in front of you a document entitled, "Testimony of Robert A. Capra on Implementation of Long Term Modifications Established in the Commission Order of May 7, 1979"?

A (MR. CAPRA) Yes, I do.

Q Did you prepare that testimony?

A Yes, I did.

MR. BAXTER: Excuse me, Mr. Lewis and Mrs. Bowers.

We have a large number of witnesses up there and I would propose on the licensee's part, at least, to stipulate subject to any corrections the witnesses wish to make the receipt into evidence of all the testimony prefiled by these witnesses as well as their statements of qualifications.

MR. ELLISON: We would also so stipulate.

MR. LEWIS: Fine, well then, let me go down the line and ask the witnesses whether or not they have any corrections to their testimony and if they do to please identify those corrections.

Mr. Norian, do you have any corrections?

MR. NORIAN: I do not.

MR. LEWIS: Mr. Matthews, do you have any corrections?

DR. COLE: Mr. Lewis, I want to get the pieces of testimony out of the documents. Would you identify them by the number, anyway?

MR. LEWIS: Yes. The testimony that is involved is the testimony of Mr. Matthews on the adequacy of pressurizer and pressurizer relief tank size, Board question 21.

The testimony of Mr. Matthews on the reliability and timeliness of the emergency feedwater system, Board question CEC 1-6. If you had my list that was attached to my testimony, I am going down there.

DR. COLE: I have your list.

Okay, then I have no problem.

MR. LEWIS: Well, I am going down, it is Mr. Matthews, through item 4. Mr. Green, that is a later item, Mr. Wing, is a later item.

Then, there are 5 pieces from Mr. Norian and then Mr. Rubin and Mr. Novak. Then, Mr. Wilson,.is a later item. Then, you have item 16 from Mr. Novak, which is being sponsored.

The only item from the second side of the page would be Mr. Capra, is the very last,item 23.

Is that a satisfactory identification?

Mr. Matthews, do you have any corrections to your testimony?

MR. MATTHEWS: Yes, I do.

MR. LEWIS: Would you please identify which piece of testimony you are referring to and the page?

MR. MATTHEWS: The first piece of testimony is my testimony on the pressurizer relief -- pressurizer and pressurizer relief tank size, Board question 21, I believe.

On page 12, item 3, the last sentence of that item states that"a modification to provide positive indication of the PORV position would be accomplished during the present Rancho Seco refueling outtage".

That is now incorrect. That modification will be completed June of this year.

The other piece of testimony I would like to correct is my testimony on the reliability and timeliness of the emergency feedwater system on page 17, item 2.

MR. LEWIS: Item 2 at the bottom of the page?

MR. MATTHEWS: Correct, going on to page 18, indicates that modification was planned to be accomplished at the end of this present Rancho Seco refueling outtage.

The completion of this is "will be delayed." The Staff has requested the licensee for further information relative to this change before we want them to proceed with it.

DR. CCLE: So, what do you want to do? Do you want to delete that sentence or insert something else?

The first sentence on page 18.

MR. LEWIS: I believe, Dr. Cole, the cleanest thing to do would be to -- well, he is amending his testimony to state that as a subsequent development, the Staff has asked the licensee to do some further studies prior to the time the Staff wants to approve this modification.

I guess I don't really have an exact deletion that is proposed to be made. But, we would like that modification to be on record.

MR. BAXTER: At least as it is introduced, though, as a licensee commitment to implement, it is not.

Is that my understanding of the amendment?

MR. LEWIS: I believe that should stand modified.

Does that complete your corrections, Mr. Matthews?

MR. MATTHEWS: That completes it.

MR. LEWIS: Mr. Thatcher, do you have any corrections?

MR. THATCHER: No, I do not.

MR. LEWIS: Mr. Capra?

MR. CAPRA: Since my testimony deals mainly with the implementation of long term modifications, my testimony is a continuous moving target.

There are some very minor changes, basically, updates, no real changes to the rhythm testimony. So, I think my answer is no.

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MR. LEWIS: Mr. Rubin?

MR. RUBIN: Yes. For the testimony regarding additional Board question number 3.

DR. COLE: Which number is that on the list?

MR. RUBIN: If you will turn to page 8 of that testimony. In question 16, in the response, the last line in that response, it has section Roman numeral III.E.5 of NUREG-0660, the correction should be section Roman numeral II.E.5.

You turn to page 9 of the same testimony, line 7 reads, "Since immediate required automatic manual actions are the same" the correction to that is automatic and manual actions are the same.

Also attached to my piece of testimony on this issue,is my professional qualifications. There is one change on that where it states that I am currently on temporary detail to the Bulletins and Orders Task Force.

That Task Force has been dissolved and I have returned to my original duties in the reactors sytems branch division of system safety.

For the testimony, regarding the design basis for the rancho Seco Safety systems, California Energy Commission contention 1-1 and 1-12, on page 4, --

MR. LEWIS: That would be number 13 on our list.

MR. RUBIN: It is three lines up from the bottom, and it reads, 'Is considered since the safety systems are subjected to periodic testing" the correction to that line is, considered unjustified since the safety systems are subjected to periodic testing.

On that same page, in the response to question 8, the second paragraph the third line has the statement, "At present, none of the thermal cycle limits have been approached" this is in relationship to the high pressure injection system specifically, it is my understanding that one of the high pressure injection nozzles, has been subjected to 31 thermal cycles to this date. The limit, as I understand it, is 40 cycles, whether it is being approached as a matter of conjecture but it is approaching somewhere in that ballpark. I would like to correct that.

That concludes the corrections to mine and Mr. Novak's testimony.

Q Mr. Novak, do you have any corrections to your individual responsive testimony on 53A?

A No, I haven't for the record just so the Board and other parties would know that the Bulletins and Order Task Force completed the bulk of its work shortly before the end of last year, at which time all members of that task force were reassigned to their previous duties, so its is just a point that we are carrying forward with this hearing as a residual effort of that Task Force.

Q I imagine you would be carrying forth with this residual effort for quite some time?

Mr. Norian, let me show you a document entitled NUREG-0565, Generic Evaluation of Small Break Loss of Coolant Accident Behavior that Babcock & Wilcox designed 177 FA operating plants.

Are you familiar with that. document?

A Yes, I am.

Q Could you please describe what your responsibilities were with respect to that document?

Q This document was the work of a group of 4 to 5 men. My responsibility was to take a draft report of that group and fold them into the form that is shown here.

Q Are you familiar with the contents of this document?

A Yes, I am.

Q ARe the analyses and documents in there true and correct to the best of your knowledge and belief?

A Yes.

MR. LEWIS: Mrs. Bowers, we would like to have this identified as Staff Exhibit 2. Since this does form the background to -- since this document does contain the Staff's analysis of the adequacy of small break loss of coolant accident analyses done by B & W, we are asking at this time that it be introduced into the record of this proceeding and received as evidence.

It will be marked as an exhibit simply because of its size but we are offering it for the truth of the statements there in.

MRS. BOWERS: I am sorry I have been scrambling with all these papers. Is that listed on this list? No. Okay. Do we have a copy of it?

MR. LEWIS: Yes, ma'am.

MRS. BOWERS: Well, we have a couple of things I think, pending. First, NUREG -0565 will be marked as Staff Exhibit 2, and you are asking that it be admitted at this time, Mr. Lewis, is that correct?

MR. LEWIS: Yes.

MRS. BOWERS: Mr. Baxter?

MR. BAXTER: We have no objection to its receipt I was not advised that it was going to be offered and I am prepared to cross examine the witness today on this exhibit, so, I will have to reserve cross examination for the next session.

MRS. BOWERS: Mr. Ellison?

MR. ELLISON: No objection.

MRS. BOWERS: Well, Staff Exhibit 2 which is NUREG -0565 is admitted into evidence and Mr. Baxter has reserved the right to cross examine on it at a subsequent session.

(Whereupon Exhibit #2 was marked and received into evidence.)

MR. LEWIS: That is perfectly fine with me. I thought I had made it clear to Mr. Baxter that it was going to be introduced, in any event, these witnesses are certainly going to be back. I see no problem.

Mrs. Bowers, I don't know if the record is clear now as to whether or not the testimony of these witnesses has been received into evidence. There was a stipulation, I believe.

MRS. BOWERS: That's right. Well, the document, the testimony that you identified will be physically inserted in the transcript as if read.

Of course, the transcript will show that there have been several corrections or editions.

Mr. Lewis, how do you intend to proceed? We have got these different Board questions, and the different contentions.

MR. LEWIS: I will tell you how I had planned to proceed and if that causes anybody a problem, please let them say so.

My intention in bringing the panel up as a group was that there is a high degree of interdependence among the issues in this case. Even though identified members of the Staff are sponsoring particular pieces of testimony. I thought that the easiest way to do it was to have sitting at one table all those people who I thought could contribute to a full answer to the questions.

My intention was to proceed much in the way that the licencees panel on category 1 issues did. That is to say to make this entire panel available first of all, to the licensee for cross examination, then to California Energy Commission for cross examination, and then the Board questioning.

Of course, as we go along I am sure we will be addressing questions to particular people who have sponsored particular pieces of testimony. But, certainly to the extent that there is someone else who is not the author of that piece to testimony who can assist in a full answer, I had encouraged them to offer their views.

That was the reason for organizing it this way.

MRS. BOWERS: Mr. Baxter, do you have any problem with the suggested procedure?

MR. BAXTER: No.

MRS. BOWERS: Mr. Ellison?

MR. ELLISON: No, ma'am, I don't have any problem right now. I would like to suggest when we are questioning a witness about the testimony that they offered, I would like to be able to ask that witness by name, the question, and if that witness can't answer it, have them say they can't answer it and perhaps, refer to the rest of the panel. Other than that, we have no problem.

MRS. BOWERS: Why don't you proceed, Mr. Lewis.

MR. LEWIS: I am available for cross examination.

MRS. BOWERS: Mr. Baxter?

MR. BAXTER: I have very few questions, so there will be some pauses while I dig through the paper to get from one to the other.

MRS. BOWERS: Well, what are you going to start out with?

MR. BAXTER: Well, the first question, I believe, will be addressed to Mr. Mathews, although if it is not his area,please, the proper person can speak up, this is with respect to -- well, let me just ask the question.

CROSS EXAMINATION

BY MR. BAXTER

Q Is it your view, Mr. Mathews, that the loop seal in the pressurizer surge line was responsible at Three Mile Island for the filling of the pressurizer?

A (MR. MATHEWS) It is my opinion that the loop seal played a very minor role in giving the erroneous pressurizer level indication for the reason that the conditions that existed there once that system had reached saturation pressure, the steam forming in the primary system acted like a piston to burst water into the pressurizer and since the PORV was opened at that time, that was delayed pressure in the region of the system.

So, that the water level lowers in the pressurizer giving a indication of level in the pressurizer, when, in fact, the primary system was not full. That phenomena would happen with or without a loop seal.

Q Let me refer you, Mr. Mathews, to your second piece of testimony on the reliability on timeliness of the emergency feedwater system.

On page 18, you are listing there a series of commitments that reportedly the licencee has agreed to implement additional AFW system design and procedural modifications. Item 4, on page 18, I wonder if you could identify for me the basis that you have for that commitment on the part of the licensee?

A Yes, in attachment 2, to a letter dated September 17 from SMUD to NRC, which transmitted the Rancho Seco auxiliary feedwater reliability study, attachment 2 listed changes which SMUD intended to make as a result of that reliability study.

I cannot recall exactly which item in attachment 2, but it, and I can dig it out if we need to, it --

Q Date of the letter again, please?

A The date of the transmittal letter was September 17, 1979 which transmitted a Rancho Seco auxiliary feedwater system reliability analysis. In the attachment 2, it indicated that SMUD would incorporate a preceaure, adopt a procedure to provide a means for verifying the flow path from the condensate storage tank to the steam generator.

In the attachment 2, it speaksto the fact that this procedure will be part of the textback which SMUD had previously transmitted which in turn is related to in service inspection and testing that is done under the Section 11 of the ASME code.

So, that was the basis for this item 4 in my testimony.

Q I am not able to locate the document righ now, with Mr. Lewis' indulgence, we will return to that later if time permits.

I am now going to refer to testimony number 13 by Mr. Rubin and Novak on Safety Systems, page 3.

The last sentence in the answer to question number 6, since there has been a increase in reactor trips, and increase in HPI actuation is also likely. Is the reference there to automatic actuation of HPI or manual?

A (MR. NOVAK) The reference is specifically to automatic actuation, however, whenever a depressurization would follow reactor trip as has occurred from the operating experience in some cases, there is always the potential for the operators of bringing up HPI manually, but the specific reference there was for automatic actuation.

Q Why would an increase in reactor trips result in an increase likelihood of automatic HPI actuation?

A This gets into the area of overcooling tranients which were discusses previously.

A review by the Staff following TMI, following the modifications that were performed after TMI rose some possible concerns regarding overcooling transients in B & W plants. A review of the operating history show there have been occurrances where reactor trips have been followed by overcooling transients.

Strictly from a probabalistic sense, if there is some potential for overcooling following the reactor trip and the frequency of reactor trips does increase, there would likewise be some probabalistic chance of an increase in HPI actuation following reactor trip.

Q Do you know whether there has been any actual operating experience in which there has been automatic actuation of HPI in the situation you have described?

A Yes, I believe there have been some cases where overcooling has occurred following a reactor trip, there was a case at Crystal River but in that case there was depressurization from loss of reactor inventory in addion. But, I do recall seeing operating experience where system depressurization following trip was low enough to actuate the HPI system

Q Could you identify those events for us?

A I don't believe I have that material here, though a number of the overcooling transients prior to TMI which was also reviewed was outlined in NUREG-0560 and I recalled occasion where the transients reviewed in that document did lead to automatic HPI actuation.

In addition, following the TMI modifications, I recalled I believe one or two cases where depressurization occurred but I do not have the transient information here.

Q Could you identify those for us please when we return for the next session?

A Certainly.

Q I would like to refer now to CEC Exhibit # 2, which is the Staff's response to response to California Energy Commission request for admissions.

Request #51 --

MRS. BOWERS: Just a minute, please.

Did you say Request 51?

MR. BAXTER: Yes. Page 15 of that document.

BY MR. BAXTER:

Q The Request is that high pressure injection system cannot maintain reactor coolant inventory if certain size small breaks occur and the reactor coolant pumps are operating. The response is yes and goes on to amplify that the Staff's reviewed analytical studies which predicted for a certain range of small breaks, the reactor coolant inventory continues to decrease even with full HPI flow.

My question is would the inventory continue to decrease to a point whe e inadequate core cooling would occur?

A (MR. NOVAK) No. The reference that I was making of course, since it was no specific size break range given as a definition of small breaks in being responsive to the request,I considered any break approaching -- up to something on the order of even square foot, but certainly a large small break for which previous analysis have shown that the plant would depressurize and that you rely on additional systems to provide emergency core cooling.

So, in the responses there is no analysis that I was referring to which would suggest that inadequate core cooling would be a consequence of HPI failing to maintain the core cover.

Q Thank you.

That is all the cross examination I have.

MRS. BOWERS: Mr. Ellison?

CROSS EXAMINATION

BY MR. ELLISON:

Q Mr. Novak, let me pick up right where you left off with CEC #2 and question #51 and see if I can clarify your answer.

A (MR. NOVAK) CEC #2, I just only arrived, so I am not sure on your terminology.

Q That is the document you were just referring to, it is right on the order of the last question.

The question is whether the high pressure injection system cannot maintain reacotr coolant inventory of certain sized small breaks occurring and reactor coolant pipes are operating.

Is there a size of small break which can with the reactor coolant pumps operating discharge enough coolant inventory to exceed the capacity of a high pressure injection system but for which you cannot depressurize sufficiently to reach other portions of the emergency core coolant system?

A I am going to answer the question first, and then I will invite Mr. Norian to comment because his testimony is very related.

At first I thought your question was related to concern over a small band of breaks which are possible within the small break region for which if HPI is the only portion of the emergency core coolant system that is actuated and if during the course of this accident to the reactor coolant pumps for one reason or another are tripped off, could there be a possibility with the conditions present in the system at that time HPI would not be sufficient to maintaim core cooling?

For that postulate, I would answer the question, yes. But, your question does not suggest whether or not the reactor coolant pumps were tripped. So, assuming that they were on if I had just a small loss of coolant accident, and there is no break size that I am aware of, if the reactor coolant pumps continue and the high pressure system is available, that the course of that accident is not acceptable.

A (MR. NORIAN) I believe that is correct. I have no further comment.

Q I don't know if you were here through all the cross examination of Mr. Karrasch and Mr. Jones, but we engaged in a discussion of whether you could continue to operate the reactor coolant pumps during the event which provided the basis for the reactor coolant pump trip without causing serious damage to the pumps.

Do you believe that you could operate the pumps through that kind of event without damaging them?

A (MR. NOVAK) Well, we have just very limited data on the performance of the reactor coolant pump. The one piece of data, of course, is the Three Mile Island 2 experience where a certain amount of pump vibration is noted and suggested to the operating staff, at that time, to trip the pump.

I don't recall at this time whether we have tried to back calculate. I think estimates have been made to the void formation in the reactor coolant system and in the proximity of the pump at that time.

My recollection is that there was a school of thought that there was no reason to suspect that the pumps would fail for that small range breaks if indeed the pumps were not tripped. i

I would also point out, though, that in my mind part of the answer is that the solution that the Staff is looking towards as more of a final solution, is to trip the pumps automatically based on a condition in the region of the pump which suggests sufficient voids from which you can conclude that, indeed, you have a small break LOCA.

Now, I have tried to give you an answer which has a couple of ingredients. One, you would have to go back to the analysis, and perhaps, Mr. Norian can recollect some of the conditions in the reactor coolant system that are provided if you do no trip the pumps.

Those analyses were done and that consequence is acceptable. Now, one has to go back and examine the fluid conditions that are present in the pump and try to then decide, one, do I have data for which I could conclude for those conditions the pump would perform acceptably?

We know at both extremes it is probably alright, given you have single phase liquid or single phase steam, the pumps could perform without damage.

Concern is one of operation through a region where you have two-phase flow for which there would be a certain amount of question as to the possible flow regime that exists. In other words, the structure of the steam and water mixture as it flows through the pipe.

This would be one question that I think would have to be resolved.

Q Is it your opinion then that the effect of running reactor pumps with void fractions from, I will just pick some arbitrary figures,from let's say 30 to 90% for long periods of time is still somewhat uncertain by the fact of having the effect on the pump?

A Yes, I think there is a question as I pointed out there is no specific data that I could refer to to show that the voids in the high range which you quoted up to 90% where the consequences have been acceptable.

There was a question in the minds of the Staff that the pump perhaps would have questionable performance. Rather than leave that as a question, the action then to trip the pumps immediately following a reactor trip followed by an automatic ESFAC would be the solution that would treat this uncertainty, recognizing, of course, then that there may be times when you are tripping that pump when you, indeed,did not have a loss of coolant accident.

Q As long as we started out with this topic, I think I will stick with it a little bit and ask and address this to you, Mr. Novak, because we have been engaged in discussion, but if we need anybody that can contribute to it -- you suggested that the Staff has concluded that automatic reactor coolant pump trip, at least in the perceivable future, is the best solution, or the best procedure. Is that correct?

A Well, I don't know that it is the optimum, I am familiar with some of the recommendations in NUREG-0565 which is just, perhaps, increased ECCS injection.

I also envision in my own mind that as this problem is better studied, that being a better evaluation of this loss of coolant accident over this small range, my judgment is that it may very well be demonstrated that even that small window is also an acceptable consequence.

I am not at all convinced that the analytical capabilities that have been used to date clearly demonstrate that this is an area for which the current emergency core coolant systems could not amply cool the core.

It is just my -- I am reasonably familiar with some of the models that are used. There are many assumptions that go into those models and I think the specific fact that we are going after that being that there is a potential for separation of liquid and steam which at one time is not separated, for example, in homogeneous models but then following a reactor trip you have phase separation, you have the potential for not refilling the core in the period of time for which you would, perhaps, leave it uncovered long enough to overheat a portion of it at least.

My judgment of the probable behavior is that you are going to get separation even while the pumps are operating, and that, indeed, the actual behavior of the system is going to be something different than either extremes.

I feel that there is more work that can be done both in experiments as well as analytical developments that suggest that, perhaps, this region does not result in unacceptable consequence.

On the other hand, I think the more experience we have with operating reactors, I think it is a common feeling that tripping of the reactor coolant pumps is not the most desirable solution and certainly, we are going to be looking and I am sure the industry is going to be looking for better solutions to the problem.

The fact being that, in fact, the industry I would assume is looking for better solutions to the problem since they would be expecting other transients other than loss of coolant accidents to occur with a higher frequence for which they would prefer to have the reactor coolant pumps operated.

Q Do you share that view, you would prefer to have the coolant pumps operated?

A If I had another solution at this time, yes.

Q Do you understand the Staff's position, then, to be that the matter is still somewhat unsettled and that the reactor coolant pump trip is sort of an interim precaution until the effect of that particular size of small breaks is better understood and other solutions are examined?

A I don't know if interim precaution is the right term.

What we have to do in order to conclude that the continued operation of a plant does not represent an unreasonable risk to health, safety of the public is to examine the full spectrum of accidents and transients.

What we have here is the situation where there is a small window of breaks that the analysis suggests that for a certain combination of scenarios the pumps tripping sometime during the accident, there is a potential for continued uncovering of the core in overheating.

I can accept tripping of the pumps early as a adequate solution to the problem at this time, recognizing that there is going to be continued work in this area to find a better solution to the problem.

Q Do you believe that it is possible in an operating plant like Rancho Seco to go back and make changes that would increase the capacity of the high pressure injection system sufficiently to enable you to operate the reactor coolant pumps?

A Your question is hypothetical. It is certainly is possible. I don't know what would be involved but I couldn't conclude at this time that no, there is no way you could deliver enough water. Obviously, it is possible to identify in theory, what would be necessary to provide a delivery of water to the system for which there would be no need to trip the pumps. I don't know what that would involve but my judgment is when you compound it with other assumptions, for example, loss of offsite power, well, that would not, it is a quirk, if you have a loss of offsite power you don't have to worry about tripping the pumps, they come down all by themselves.

But, recognizing the piping systems and the other things that would go on, it isn't just a question of adding a pump to the existing system.

I want to also make one point that is also a rush to design, when I was sitting around earlier and talking, and that is to go back to the Crystal River event again. Probably part of the behavior of the event was the fact that with 3HPI pumps on, we deliver in excess of 1000 gallons per minute in a rather short period of time, so what I am suggesting is that when one does in looking at one aspect of a pump, and that being, can we deliver more water to this system for specific range of small break, I think we should be careful to examine all aspects of the problem, for example, events for which you get the injection ans then the operator decides to permit that water to continue on, and you have very high flow rate trhough the safety valves, etcetera, so on to the surface, that may not be a particular problem of concern, but I would point it out to say, I would be careful to look at the design modification in ECCS sytems.

The ACRS has incurred and industry for the last several years t make improvements in emergency core coolant systems. and just as a matter of fact, there have been improvements to one class of Westinghouse plants, that being the upper head injection system. I would just point out it was a very laborius, tedious task to actually define the merits of that system.

It took several years of Staff review. Indeed, it was a different system then just adding an additional pump to the emergency core cooling system. It was intended to be a specific system intermeate at between high pressure injection system and core flood tanks.

My only point, though, is that I'd be careful in evaluating additional systems, but certainly that should be done. I am only saying it is not a quick solution. I think there is a lot of work to be done in evaluating the ECCS improvement.

Q Just to clarify, the thrust of my question really is whether the idea is so impractical, or so complicated, that it does not merit study?

A I would also ask Mr. Norian since he may have some comments on that particular recommendation in 0565.

A (MR. NORIAN) I would just note that the LOFT test facility this year is planning to run some tests where the pumps are kept on for a series of small breaks. This will help to Staff and the vendors to develop perhaps a better set of codes to do this calculation. One output of this effort may be a series of calculations that perhaps show that tripping the pumps at a certain time is not quite so bad.

This is the kind of work that it takes time to run the test and once the tests are run, it takes a long time to look at the test results and fix up the codes to match those test results and do the calculations and go through all of that.

It is a job that will take, I am sure, at least of couple of years. So, the answer, I don't believe, is just around the corner and the interim I feel the best thing to do is to trip the pump. As they are now being tripped.

A (MR. NOVAK) Lastly, just for the record, I am referring again to NUREG-0565, it is my understanding that every recommendation that is identified in NUREG-0565 has been included in the latest draft of our action plans tat are being considered by the Commission. What they are doing is trying to order them in terms of priority.

That recommendation to look at ways of improving ECCS so as to not have to trip the pumps, has gone into our action plan.

Q You invited an unfair question, so here it comes. If something is given a low priority in the NRC action plans, what are the chances that it is going to get done in my lifetime?

A In discussing my appearance here I postulated that unfair question to all the members of management. The answer was the following. The Staff, and this is the office of Nuclear Reactor Regulation through Mr. Denton, is making decisions as to if the Staff puts these recommendations in then, in their judgment, they should be following through.

Now, what you will see when you look at the totality of the action plan if it exceeds all reasonable resources that the Commission can provide and that industry can provide, but it doesn't mean that we drop out certain things. Unless the Commission decides in their own review to delete these actions, we should assume that they would be followed through.

Now, it also follows, and this is something that I think is reasonable, that given an action such as this is not the first priority. I seems reasonable to reassess second and third priority items after a year or so. Our knowledge of how these systems behave, is growing each year. I would expect that we will be able to move with more wisdom towards the second and third priority items. It will be reshuffled and reevaluated.

The most direct answer to your question is at this time there is an expectation that this work would be followed through and only after the Commission has agreed to the priorities can we see, or begin to expect on what time frame.

Q One last question, would it be in your opinion reasonable for this Board in its present inquiry to assume that Rancho Seco or that the Staff position, anyway, would be that Rancho Seco should maintain the reactor coolant pump trip for the forseeable future?

A Well, I think that is a fair question and I would expect that the Board would have to make a judgment just like everyone of us whether they can see the priority in their minds such as to say that a first action should be to raise the ethicacy of the emergency core coolant system. Or, if in their judgment, they could find a reason to say that the information provided suggests that the present mode of operation is acceptable for the interim recognizing that additional work is being done.

That speaks to what our discussions reveal.

Q Mr. Rubin, I would like to address this question to you. I understand from Mr. Lewis that you are co-author of the document that we have identified as CEC Exhibit #5, that is the -- are you familiar with CEC #5?

A (MR. RUBIN) Yes, I am familiar with that document.

Q Do you have a copy?

A I have a copy which I believe includes what is incorporated in your exhibit but I don't have a copy of your exhibit.

MR. LEWIS: The exhibit begins with the cover letter for Mr. Gammel, do have that?

MR. RUBIN: No, I do not.

MR. LEWIS: Well, let me give you the exhibit, then.

MR. CAPRA: Mrs. Bower, can I interrupt for jsut a second?

I would like if I could to be excused from the panel for about 5 minutes. I have to get the Staff documents to the mailroom, and I am afraid if I don't leave now, I am going to wind up hand carrying these things myself on the plane.

If I can just have a few minutes, then, I will rejoin the panel, seeing how they are questioning Mr. Rubin.

MRS. BOWERS: Any objection?

MR. ELLISON: I hve no problem with that. You don't have any involvement in this letter do you, in this CEC #5?

MR. CAPRA: No, I do not.

MR. ELLISON: Fine.

MR. LEWIS: Even though our documents are going we are still here and prepared to fully answer.

MR. RUBIN: Alright, I had a chance to look at the document.

BY MR. ELLISON:

Q With respect to CEC #5, could you identify those parts of that exhibit that you participated in writing.

A I participated in writing the insert which is entitled ,"Primary System Perturbations Induced By Once Through Steam Generators".

Q Your co-author of that insert was Dr. Ross, is that correct?

A I don't know if co-author is the correct word. Dr. Ross was the director of the Bulletins and Orders Task Force. I worked under him. We both participated in the production of that document. He, of course, had final control over it.

In addition to Dr. Ross, other people in our management chain did review it, including, I believe, Dr. Denton since it went out under -- excuse me, Mr. Gammel and I believe Dr. Denton also in some of the letters that went out to applicants.

MR. BOWERS: I need some clarification. You mentioned insert, now, are you talking about the entire enclosure to CEC #5? There is the cover letter of W.P.Gammel, and then there is the next letter to Mr. Paris, Tennessee Valley Authority from Mr. Denton.

MR. RUBIN: Right, that is the one I have on my document.

MRS. BOWERS: Yes, and then are you talking about the enclosure, "Primary System Pertubations Induced Once Through Steam Generator", that you participated in the drafting of this or the writing of this?

MR. RUBIN: That's correct along with Dr. Ross and also involvement of other members of the NRC management.

The primary responsibility was Dr. Ross' and mine but other review functions were performed by NRC management.

BY MR. ELLISON:

Q As between you, Mr. Rubin, and Dr. Ross, was this strictly a joint effort or was one of you the principle author of this document?

A (MR. RUBIN) That is a difficult question to answer.

When this topic was undertaken, there were a number of discussions among Dr. Ross, myself, Mr. Novak, other members of the Bulletins Orders Task Force,following those discussions Dr. Ross put together, I guess we could call it a draft thoughts on the subject that was turned over to me for review editing, additional supporting information, and this is a review of his original draft.

I made a number of additions, changes to his early draft that was then turned over to Mr. Novak, who I believe reviewed it and was passed on to Dr. Ross.

From there, it was, I believe, reviewed and modified some more and then went out in the form you seen on these documents.

After I turned it over to Mr. Novak and was given to Dr. Ross, I did not have any further impact on that document.

MR. LEWIS: You now know how the NRC Staff works so well.

But, Mr. Ellison, I am just wondering, I think that if there are questions that you have on the document, my understanding is that Mr. Rubin would be in a position to go a far way to answering those questions. If we were to establish certain questions he couldn't answer, we could worry about that if we come to it.

BY MR. ELLISON:

Q When was this document drafted, Mr. Rubin?

A To the best of my recollection, the majority of the work was done in September of last year. Perhaps, a little bit was done in the beginning of October. After that point, it was in the hands of Dr. Ross.

Q Have there been additional anaysis or subsequent events that would cause you to change any of the statements that you have made in this document as of now?

A Well, you have to put this document into the proper perspective. Following the incident at Three Mile Island and the actions taken by the Staff and by Babcock & Wilcox and by the applicants, a number of actions were taken to mitigate the type of instant that occurred at Three MIle Island. Which was loss of feedwater, undercooling of the core.

The first priority, I guess you could call it, of the Staff was to deal with those eventualities. However, Dr. Ross thought there might be a potential concern for over-cooling transients which are exactly opposite of the incident coming out of Three Mile Island which was undercooling.

To try to gather information and to provide assurance that the actions following TMI hadn't created undesirable situation, a study in this area was initiated by him.

This document was produced and was sent out to the construction permit holders of B & W plants. As you can see from the document, it raises some concerns of overcooling transients.

These concerns were in a early stage to some extent they were preliminary in nature. It was thought that these concerns might exist. More information certainly should be obtained in this area. It had been observed from some of the operating experience in B & W plants that overcooling transients were occurring and there was some thought that perhaps they were undesirable.

You will notice looking through the document that a lot of the suggestions as to possible concerns are prefaced by the adjective "may" cause certain undesirable results. The intent of this was to raise the consciousness of the applicants and of the NRC to this possible concern and to allow investigations and further research to proceed on subject.

Now, to return to your original question, if anything has occurred which might change my views, to some extent I would say yes. One of the concerns that was developed in this document was that falling overcooling event, an overcooling transient, there might be the potential for avoiding an inability to cool the reactor core.

Since this insert was produced, additional analyses have been carried out, I believe, by B & W and also some under the Staff's direction, which show that core coolant can be maintained.

To that extent, we are now obtaining assurance that overcooling events appear to not include the potentiality to the extent they have been analyzed at this point of undercooling the core.

Q Could you identify for the record the analyses that you are referring to and specifically I am wondering if it's -- can you identify for the record, the analyses that you referred to?

A I am aware of analyses submitted by the Midland Construction permit holder, and I believe that that was done by the B & W Company. The Staff also had analyses done by the Brookhaven National Laboratory on overcooling transients and I believe they have been discussed previously.

Q Mr. Rubin, I am going to show you copies of documents, it is really one document, that has been identified as CEC #15 and some tables that also belong in it, identified as CEC #16.

Is that the B & W analysis that you are referring to?

A CEC #15 appears to be the analysis provided in the Midland plant response.

CEC #16I believe was also included in that response, though I should include the fact that reviewing these analyses has not been my responsibility.

Q Was that the responsibility of any other member of this panel?

A (MR. NOVAK) I assigned the responsibility to another engineer in the branch.

I wonder if I might add a comment to some of the dialogue.

I think it is fair to say that the substance of the concern that we discussed with regard to overcooling really came to focus when we met with the licensees of B & W reactors durin the end of August. I think there is an 823 meeting and I think the Board and all parties have a copy of the minutes of that meeting. At which time the description of the events for which overcooling occurred was discussed by each of the licensees.

I think that the concern that Mr. Ross had was that there were events for which, at least he indicated, pressurizer water level went off scale.

His judgment was that there might not be a direct or an obvious safety concern. The ability of the operator to recover from events such as that, certainly, there was a suggestion, at least, that they were being hampered by the loss of pressurizer level, of course, they all came back.

His concern and then I think what prompted him to go in this area, in terms of getting additional analysis,was to investigate some actions or behavior of the plant to better understand what could be done to perhaps maintain the pressurizer water level within the readable range for a certain rang of overcooling transients.

Obviously, when you go, there has been a certain number of actual operating experiences where the indicated range has gone off scale low, and I am not sure whether the argument is that subsequent analysis still suggested that the pressurizer did not empty.

So, you have theoretical analysis which could be the worst case transient which might show the worse shrink effect in effect on the reactor coolant system show the largest overcooling then you could go to the worst events from operating experience.

Now, all of this is directed, then, towards better understanding of the behavior of B & W design to see what system modifications could tend to dampen the effect.

Dampen from the view of perhaps responsive actions of other systems that could respond in a fashion timely enough so that, for example, one, one primary such as pressurizer water level would still remain with in the readable range to the operator and to maintain his assurance that he understands the behavior of his plant.

I think this something that we will continue to look at at all plants. I think this is one of the attitude changes that the Commission has recognized. So, I don't want to single out that the only plant that is getting scrutiny in the B & W design. All plants are being scrutinized as reflected in the action plants.

Here was an area that we saw enough indication, and perhaps, in all honesty, I think there was a question whether the Staff in dictating certain actions to be -- I don't know if dictated is the right term, but certainly the changes in the setpoints on the SCRAM setpoint raising the PORV may have overemphasized the other side of the problem. So, I think there was a concern that we go back and look to see if, indeed, the actions taken to undercooling were too severe in terms of trying to see whether there is a balance on how the plant should behave.

I think through the 5054 F request for information plus our own analysis, we are getting a better understanding of the plant performance, its sensitivity, and the kind of modifications that are practical in the sense of showing a return in terms of sensitivity.

So, I am not surprised that there is some dialogue on this point, I think it was healthy that we got started on in it because even the Crystal River event suggests to me a sensitivity in the sense of showing that when you go and leave procedures in place, the operators responded prudently but indeed some 33 thousand gallons of water ended up on the containment floor.

So, I think it is this attitude of going back and seeing the actions that are taken both procedurally and through system response, to identify where additional improvements can be made. I would expect that this kind of work is going to be an ongoing effort for the next several years. Certainly, in terms of priorities I can't see us completing all of those actions without any sooner than the next several years.

Q Mr. Rubin, you stated that the memo expresses some possible concerns, that are perhaps somewhat tentative, but in my reading it also seems to express some conclusions, so I would like to ask you about those.

Unfortunately the document is not-- the pages aren't numbered, but the third page into the section which you wrote above the Roman numeral II, appeared two paragraphs. For the record I am going to read them. "The Staff is concerned by the inherent responsiveness of the B & W OTSG design. On some specific instances represented in the next section of this paper, the Staff concerns are also of a general nature.

It is felt that good design practice and maintenance of the defense and death concept requires a stable, well-behaved system. To a large part meticulous operator attention and prompt manual action, is used on these plants to compensate for the systems sensitivity rather than any inherent design features.

The Staff believes that the general stability of the B & W plant control systems should be improved and that the plant response to OTSG feedwater perturbations be dampened."

With respect to that statement, I would like to ask you, Mr. Rubin, whether you still agree with it?

A (MR. RUBIN) We have to differentiate here between what may be unacceptable plant performance and undesirable plant performance.

It was the belief of Dr. Ross and I also concurred that the B & W once through steam generator of performance that we had observed in the operating history, indicated some sensitivity, perhaps, overly sensitive to feedwater transients.

At the same time, it must be noted that this was verified by analyses that have been performed to this point, that the plant response is not unacceptable that it meets the requirements found in Commission regulations and that the plant can safely sustain the transients there under consideration.

At the same time, the Staff, through its reviews and actions strives to increase as it stated here, the defense indepth concept, to increase levels of safety where possible, above what is currently obtained. If there is a definite benefit that is available.

I believe that some actions are probably possible to reduce the B & W plant sensitivity, perhaps, you call it plant response that would make the plants somewhat more well behaved. If the response is dampened somewhat, we believe that would probably be a desirable result.

To that extent, I still believe, still am in concurrence with this paragraph, that to reduce the sensitivity of the plant or perhaps the consequences of that sensitivity is a desirable goal.

It is a subject which is currently under Staff review.

Q Do you still believe that according again to a large part of meticulous operator attention and prompt manual action is used on these plants to compensate for the systems sensitivity rather than any inherent design features?

A It is my belief and understanding that the requirements for operator response is somewhat quicker on the B & W plants for some transients to prevent such events as loss of pressurizer level.

The inference here is not that there are no inherent design features in the B & W plants which provide acceptable plant safety. That is not the case.

But, it is my understanding that operator action at some B & W plants requires, perhaps, quicker response and perhaps a little closer attention of following feedwater transients to prevent what we viewed as undesirable consequences such as loss of pressurizer level.

Though, details on operator performance and training in these areas will be provided by another witness who is not on this panel.

Q How does the Staff determine whether this kind of inherent responsiveness is in your words acceptable or unacceptable as opposed to desirable and undesirable?

Are there specific Commission regulations that govern this kind o' responsiveness or is it more judgmental?

A (MR. NOVAK) It is judgment. I think it involves a number of ingredients, certainly the basic characteristics of the plant have to be well understood by the designers, by the operators. So, I think the training of the operators, the simulated training, forms the ingredients for which you decide, indeed, there is sufficient protection so that for transients, which have a certain level rate, we can expect that the systems are there to protect against fuel damage and that the operators are trained adequately to respond to those events.

We don't have rigorous criteria in this area, it is a judgment. You have to be licensed to operate any facility.

I think you might, I don't know I would suspect that that would be a very important ingredient to whether or not the system is too sensitive.

I think the actions that are required of the operators are viewed by our operating licensed personnal, and they make judgments as to whether or not these personnal are sufficiently trained to respond accordingly.

So, I guess the best answer I can give you is that it is culmination of the design review and the operator training.

But, then, again we continue to review the operating history, certainly since Three MIle Island 2 our interest has been raised to a larger degree than has been suggested in the past.

So, I think, the test is an ongoing test. I think this discussion of overcooling transients is an example where we go back and are reexamining whether indeed there is sufficient equipment already designed in the plant, so that the response of the operators is not taxed by the system itself.

One last point, and I don't mean to distract from the discussion, but I think in balance one has to also look at some of the acceptable features of a design such as this.

Certainly, overcooling transients are to be reduced if possible. But, on the other side of the coin, I can recall that I don't know of any steam generated tube ruptures on B & W plants.

Now, you have to strike the balance. You have to decide, if I want a very sluggish system, I may have a system that is not very sensitive. But, in the overall operation of that plant, doesn't, indeed, represent no greater risk to the health and safety of the public or the plant.

Steam generated tube failures, if there is some characteristic of a U-tube design, then it would suggest this sensitivity as undesirable. Now, I think what we were looking for is to see ways in which that level of undesirableness could be reduced without, indeed, losing any of its attractive features.

Now, whether or not we can achieve that is the question. I think that was the Staff's intent. To improve on it without giving up anything.

MR. SHON: Mr. Ellison, if I could just interrupt for a moment, there is one question I would like to ask Mr. Rubin.

Several times, sir, you mentioned that there are things you can do about this sensitivity of the primary to the secondary to the steam generator, things you can do to lessen that and cushion it and decrease it. What sort of things are you talking about? I don' think you actually mentioned any specific changes. They might be big ones or small ones, I would like to know what they are.

MR. RUBIN: Well, this topic is under active review by the NRC Staff at this time. The review is rather preliminary in nature. We have got responses from the CP holders at B & W plants. Those responses include various suggestions of the applicants on ways to mitigate or reduce the sensitivity, and improve plant response.

Some of the items which have been included are changes in the control system, such that the likelihood of excess feedwater, main, or auxiliary feedwater is reduced.

There are others in the documents, but the decision on which ones, indeed if any will actually be implemented is currently undergoing review and there are no firm decisions.

As far as the physical or theoretical changes which could be made, obviously they run the gamut of everything from change in the steam generator at the most extreme to minor control system changes depending on what type of modifications or reduction plants is deemed necessary and desirable.

The review is still preliminary in those issues.

MR. SHON: Thank you. Thank you, Mr. Ellison.

BY MR. ELLISON:

Q Mr. Novak, in the middle of your last answer you said that if the once through steam generator design and precisely that inherently was less prone to tube failure and that kind of thing, that that also ought to be a factor in evaluating the overall design.

To clarify the record, I would like to ask you whether it is your opinion that inherent in the steam generator once-through steam generator design, is the fact that it is less prone to failures in those types of small leaks?

A (MR. NOVAK) Well, let me amplify. What I was recalling was my recollection that in observing operating reactor plant performance, I did not recall steam generator tube failures occurring on the B & W design nuclear steam supply system.

There have been failures on U-tube designs, therefore, I am not sure whether Westinghouse has combustion engineering.

My only point was I felt that is an ingredient in looking at the design that one has to look at a number of features, the complexity of the design, performance, its reliability and I can only think that that was more or less an observation than I thought at least should go on the record, not to distract from the sensitivity issue because I think it is an issue we are concerned about and will continue to be concerned about, but I just felt that for the record we ought to recognize that you just can't give up sensitivity, you just can't get away with sensitivity and go somewhere else and say you have lost nothing else. You may, indeed, have lost some other feature of the design which, indeed, also has some safety instrumentations and I just wanted that to be on the record.

Q So, I gather you are speaking generally and hypothetically rather than --

A Well, no, I think the operating experience shows let's say taht you have less steam generated tube failures, in operating the reactor for the B & W plant compared to one that has a U-tube steam generator.

Q What is the basis for that statement?

A That is my, I could be proved wrong, that is my recollection of the operating data.

Q Are you familiar with the problems with tube failures that were experienced in the Oconee 1 site?

A I am vaguely familiar, yes, I remember those.

I am talking about large failures. Tube ruptures from which there is a substantial release of primary system water to the secondary side for which the plant, I am thinking, for example, of the recent Perry Island incident, I am using that kind of an example.

A small rupture, a leak, that requires the plant to shut down, becomes a succeeding text back leakage, is an operational problem as opposed to severe steam generator tube leakage requires a quick shutdown and is taken to cool down the plant immediately, to reduce or site risks.

Q When you refer to tube failures, you're distinguishing two large failures you might get in U-tube failures from a smaller size tube failure that you might experience in Babcock & Wilcox, is that correct?

A I would say in general, yes.

Q And you also mentioned in evaluating and judging the adequacy and sensitivity that you considered a large number of factors, and one of the ones you pointed out was the operator capability.

Would I be correct in assuming you have less confidence in the operator at Rancho Seco, you would be more concerned about the facilities?

A I would not say I have less confidence. I don't think that is a correct statement.

Q I said, if you had less confidence in the facilities then you do in the operators at Rancho Seco, you would be more conserned about the sensitivities as to it, your judgment of the adequacy of the acceptability, is somewhat depending on the assessment of t he procedures, and the operations of the plant, and that sort of thing?

A First, le me point out, as I said in my earlier remarks, the decision on the acceptability of the plant is a judgment by a number of people on the Staff.

I don't myself feel that I am an expert to evaluating an operator's performance and if I have not been trained, my training does not provide me with a basis for making recommendations.

If I have an expert opinion, I think it would be on plant behavior. I think I understand the system dynamics, and my training would be in that area, so in terms of sensitivity, I think I am attuned to the way plants are behaving, and what I would expect certain systems, changes in system variables to produce.

I do not think I can give you a direct answer to your question except that I have, I support today the operation of the Rancho Seco plant and any B&W plant based on the experience that I have gained through the orders permiting these plants to start back up again.

We did a detailed review of the operating procedures, the operated training, the plant dynamics, including how the plant behaves in terms of loss of coolant accident and feedwater transients.

We continue to look at these plants, to see if there is not a basis for suggesting that these plants, that there is an additional safety margin that can be achieved through some plant modification.

The plant I think is persuing this through these studies on plant sensitivity.

That was the purpose to that review to see if indeed there was a basis for going forward with changes.

First, on plants still under construction, where ideally the impacted changes are less than the plant that is operating, but indeed after that work is done, if a judgment is made by the staff that certain changes merit

Babcock to operate reactors, they will be done, but there has been no decision as of this time.

In fact, we have not finished the work yet on plants still under construction, but it is certainly going to be a two stage process, and the impacts are expected to be different, just as they will be different even in the spectrum of plants still under construction.

Of those very early and those very late in construction, and that is the thrust of the work we are doing today, to see if we can learn enough about these plants, in terms of system sensitivity, and possible design changes, proposed by these licensees to in effect reduce the sensitivity, and the proposal that Mr. Rubin pointed out were those advanced by the licensees themselves.

They recognized the problems to a certain degree, and they are pursuing it.

Q Is there anyone among you on the panel that participated directly in making a judgment at Rancho Saco, given the sensitivities that are identified here, that these sensitivities were acceptable and that Ranch Saco should be allowed to continue operating?

A I certainly did for one.

Q Anyone else?

A (Mr. Capra) Are you speaking just strictly of the sensitivity?

Q Yes.

Q I am speaking of the sensitivity issue.

I would like to address this question to anyone on the panel who could answer it.

A little bit of a background to it, I am referring first of all to the statement in suggesting operator action, and manual actions would be used in Bobcocks and Wilcox plants to condensate for the sensitivity, having read that statement here and having heard your statement, you are not an expert on operator capability and operator action, but you did participate in a judgment, that sensitivities were acceptable, my question to the entire panel is whether or not the decision that the sensitivities were acceptable, included consideration of operation of the plant, the procedures, whether they were acceptable or whether the sensitivities were determined to be acceptable in the absolute, without considering the operations and procedures of the plant.

A (Mr. Novak) Mr. Ellison, I think the problem is perhaps atleast a two part question, and if you could restate it, it would be helpful.

Q Sure.

In determining that sensitivities were acceptable, did you consider and evaluate the ability of the operators at Rancho Saco to control or manage sensitivity, if you will?

A Let me answer first.

The answer is yes, basically, because it is a two part answer.

One, in responding to the sensitivities, one has to look to see if the procedures one uses to respond to these events are adequate, and those procedures were reviewed by the staff, specifically, how one responds to feedwater transient and small loss coolant accidents, and then in terms of licensing, and this was the specific thing that I did not feel qualified to answer, was to perform an examinatio: whereas the right people went through a very thorough walk-thru, and in the examination of the operators to convince themselves that these people had the knowledge of the expected plant performance to these transients, and it was only after this examination and perhaps some additional training when weaknesses were identified that a conclusion was reached that the operating staff and the sytem proposed for that plant was acceptable to permit restart of the plant.

That was the criteria we were using.

A (Mr. Rubin) Let me add if I could, provide again a little perspective on the entire study on B&W sensitivity to overcooling transients.

When the study went forward, it was really going forward in two parts, as has been brought out in the document, there were general concerns of the B&W sensitivity, but at the same time there was an actempt to discover that there were any unacceptable consequences from this overcooling sensitivity.

This was pursued among the bulletin and the task force and also pursued with members of the lessons learned task force to try to identify possible situations, or consequences, which would lead the staff to I think the conclusion that there was an unacceptable situation occurring from the overcooling transient, B&W plant response to them.

If that had been the case, other action would have been required.

In an attempt to determine that there were unacceptable consequences if the analyses requested in the letter went out, the announcement had been performed at B&W, on the staff's behalf, are we continued to investigate this topic to determine if unacceptable consequences would result to this point.

To this point we have not uncovered any unacceptable consequences from these overcooling transients, and we continue to study this matter and ofcourse we continue to pursue it from a perspective of the defense in depth concept to moderate the sensitivity point in general.

Perhaps that is a little insight.

MRS. BOWERS: Mr. Ellison, I think we will have to adjourn when the air is shut off in this building when it is a bad situation.

We will then reconvene at nine o'clock on Tuesday, the eighth of April.

MR. LEWIS: With this panel I assume?

MRS. BOWERS: I would assume so.

MR. ELLISION: Fine.

MRS. BOWERS: We stand in recess.

(Whereupon, the Board was recessed at five o'clock p.m.)

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that there is one negative aspect to that design.

All I am saying is, I think this discussion of the sensitivity of the once-through steam generator is healthy but it certainly isn't complete in terms of deciding whether or not that plant design is adequate. Because, I think you have to look at the whole picture of that plant operation.

What has been the amount of leakage from the design? If a once-through steam generator design mechanically is simpler and offers less potential for leakage and steam generated tube rupture than that should go on the score board as well as the fact that the operator has to be more attentive to responses and overcooling events and so forth.

The contention is good. I think it is healthy to get this discussion. I just feel that it is important, though, that that whole problem be kept in context. I wouldn't want to leave out the fact that other aspects of the design of steam generators, nozzles are of interest to the Staff. We continue to look at all aspects.

I guess, what I am really saying is there is no perfect design in this world that I know of. So, I don't know where we go to. I think any design that we come up with is going to have attractive features and it is going to have some unattractive features.

I think as we have tried to focus here, we categorize