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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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In the Matter of:

SACRAMENTO MUNICIPAL UTILITY DISTRICT : Docket No.

(RANCHO SECO) : 50-312

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Conference Room W-1140
United States Federal Building
2800 Cottage Way
Sacramento, California

Monday, May 12, 1980

The above-entitled matter came on for hearing, pursuant to recess at 9:00 a.m.

BEFORE:

ELIZABETH S. BOWERS, CHAIRMAN
DR. RICHARD F. COLE, MEMBER
MR. FREDERICK J. SHON, MEMBER

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P R O C E E D I N G S

MRS. BOWERS: I believe we were with Mr. Ellison on his cross examination of Mr. Capra.

Whereupon,

ROBERT A CAPRA

the witness on the stand at the time of recess, was resumed was a witness and, having been previously duly sworn, was examined and testified as follows:

MR. ELLISON: Before I begin, I would like to introduce on the record, Ms. Mary McDermid. She will be assisting me for this week in place of Mr. Lanpher.

CROSS EXAMINATION (RESUMED)

BY MR. ELLISON:

Q Mr. Capra, would it be fair to characterize NUREG-0667 as the staff's lastest most-current opinion about the transient response of B & W reacotrs?

A I thought I made this clear in our session on Saturday that this is not a staff document. It is still a task force document.

Q Let me rephrase --

A It is the latest opinion of the members on the task force.

Q All right. Could you refer to page 5-64 of NUREG-0667? Actually, it might be more instructive to refer to page 5-63 first. The second conclusion is stated there. I

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am not going to read it, but essentially it says that it is necessary to provide the operator with certain information to accurately assess plant conditions.

Referring to 5-64 -- at 5-64 the task force recommends a number of different types of indications that it believes is a minimum set.

I am particularly interested in item C, the wide-range reactor coolant system temperatures in a hot leg, the cold leg and the core outlet. Could you explain why the task force chose to recommend that paricular indication?

A I think you have to take all of the parameters as a set. The purpose of the recommendation as it is stated is such that upon the loss of normal power supply, such as was experienced at Crystal River, the operator will have working knowledge, or have all the parameters necessary to make an assessment of the status of the reactor coolant system.

Item C, with respect to temperature, it is very important that the operator understand what the reactor coolant system temperature is. The purpose of a hot leg and cold leg, of course, is that in the event you lose reactor coolant pumps, it is necessary to use those indications to determine status of natural circulation flow.

Q Could you explain a little more precisely how wide a range the task force is recommending?

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A I do not remember the exact number the t-h went down to in the past, but when you are cooling down, t-h indication fairly soon would go offscale low. So, in order to determine natural circulation using a t-h method, you would need to have it go below what it normally -- it originally indicated.

I do not remember what that number is.

Q Was there any concern with the upper bound of the t-h indication, and whether the operator had sufficient readings on the high end of the scale?

A I do not think the upper end has been changed at all. We have been mainly concerned with the lower end. You do have core outlet thermocouple temperatures where the range is high enough, we feel now.

Q This remains a recommendation of the task force?

A Yes, it does.

Q Are you familiar with the range of the hot leg temperature indication at Rancho Seco?

A No, I am not.

Q Next, I would like to refer you to table 7.1 which occurs at page 7-14. This is, of course, part of chapter 7, which is not a part of the draft that was originally stipulated in this proceeding. As I understand it, it was essentially written by the probabilistic analysis group, is that correct?

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A That is correct.

Q I have a couple of questions about the footnotes. First of all, could you explain what -- or amplify what is meant by the first footnote?

A One of the plants looked at in WASH-1400 for the risk assessment of the reactor safety study was Surry units one and two, in order to make a comparison of the B & W --

Q I'm sorry. You are referring to th asterisk here?

A I am sorry.

Q I am interested in the one designated number one.

A Footnote one refers to the characteristic of the short dry-out time of the once through steam generator. We are talking about when the steam generator dries out faster than it would on the comparison plant. In this particular case, Surry.

It would be more of a probability that you would lose steam pressure sooner to try the turbine driven feedwater pump.

Q Do you know whether the -- either the task force or the probabilistic analysis group did any analysis of how long you could maintain enough steam pressure to drive the turbine driven pump following a loss of feedwater to the OTSG?

A I do not think we have done any analysis. I know

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the question has come up before in meetings with the licensees, particularly on Davis-Besse, since they were a plant that does not have a motor driven auxiliary feedwater pump.

They have demonstrated that they can start that pump with very low steam generator pressure, somewhere around the range of, I believe it is 75 to 100 pounds in the generator, but how long it takes to get there, I think it is very dependent -- very plant-specific, depending on how tight the system is and what is happening, where the steam is going.

For instance, it is going to make a significant difference if you have steam being supplied to the main feedwater pumps at the time, maybe the discharge valves are closed or the AFW control valves are shut. You are still supplying steam to the turbine.

It depends upon what the status of the turbine bypass valves are. So, the time varies. Again, I can't give a specific number how long it would take you to get there.

Q When you stated that licensees have demonstrated they could start the pump at low steam pressures, were you referring to the Davis-Besse pump or to all of the plants, including Rancho Seco?

A That is the only one that I know that we have talked about at meetings. I do not know about the other facilities or Rancho Seco in particular. I do not think --

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it has not been as much of a concern at Rancho Seco as it has been at Davis-Besse. Mainly because of the fact that they have the motor driven pump, plus the tandem pump.

MR. BAXTER: They, meaning Rancho Seco?

THE WITNESS: Yes.

BY MR. ELLISON: (Resuming)

Q One last question on this. Do you know whether the -- with respect to the Davis-Besse pump, they demonstrated they could start the pump and maintain it running long enough to generate enough steam to keep it running to boot-strap the operation, if you will?

A Yes. If you are talking about sufficient steam pressure, not only to keep the pump going but to supply the steam generator.

Q If you could, I would also like you to explain footnote number three.

A I am sorry. I cannot explain that one.

Q Do you have any idea what the pronounced effect of frequency of core damage that they are referring to is?

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MR. SHON: It does not seem all that obscure to me. It seems quite clear.

MR. ELLISON: Do you want to try your hand at explaining it?

(General laughter.)

MR. SHON: I would be perfectly glad to.

THE WITNESS: Then you can correct me if I am wrong.

The first sentence is self-evident, I think. It says the direct effect on the frequency of dominant sequences is negligible, so what you are seeing here is,the direct effect of undercooling transients with respect to severe accidents is negligible.

MRS. BOWERS: Mr. Capra, my copy only has one sentence. You just referred to the first clause.

MR. SHON: He means the first clause, I think.

THE WITNESS: I am sorry.

(Pause.)

THE WITNESS: My interpretation of the second clause there is,the combination of the effect on the frequency of core damage coincident with the failure of the containment structure.could rival the dominant accident characteristics or the dominant accident sequences

I do not know if that -- maybe that is just a paraphrasing of the footnote, but I think what it is trying

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to say is, this effect combined with the failure of the containment could give significant effects.

BY MR. ELLISON: (Resuming)

Q Okay. I guess my question is, when I read Footnote 3, I got the impression that the probabilistic analysis group had determined that frequent undercooling transients had a pronounced effect on the frequency of core damage. Is that a fair reading of that footnote?

A Yes. That is what -- although it does not refer you to that footnote, if you look at accidents or small releases, you see the frequent undercooling transients, and that has a large input to the frequency or severity of accidents.

MR. SHON: I just assumed it meant that of the dominant sequences identified in WASH 1400, which are the ones we have been talking about, this particular thing did not change their frequency, but that since it did greatly increase the frequency of small core damage, it might by an independent probability bring some other sequence that involved containment failure into the dominant sequence category.

Isn't that essentially what it is?

THE WITNESS: Yes, the -- if you look at the definition in 0667 of severe accidents, it is essentially Release Categories 1, 2, and 3 of WASH 1400, which implies containment failure in one form or the other. So that in order to have a severe accident, you have to have severe core damage coupled with early containment failure.

BY MR. ELLISON: (Resuming)

Q That is consistent with my understanding of the breakdown of accidents here, but just to clarify,a severe core accident damage without containment failure would be considered an accident and not a severe accident. Is that correct?

A Yes, that's right. Even a core melt that went through the base mat of the containment building would still be classified as an accident, provided there was no containment failure.

Q Isn't the melting of the core entirely through the core mat containment failure?

A Yes, but I mean releasing to the atmosphere.

Q Okay.

Lastly, could you try your hand at Footnote Number 4? Let me be a little more specific. I am particularly interested in whether Footnote Number 4 suggests any common mode failures for the high pressure injection system and the auxiliary feedwater system?

(Pause.)

A I do not think that is necessarily implying common mode failure, or saying it is increases the frequency of common mode failure. What they are saying is that the delayed start of auxiliary feedwater upon loss of main feedwater is going to increase the probability of transient induced loss of coolant accidents, lifting the PORV, lifting the safety valves, and that if you couple that with a failure of the high pressure injection system, and the auxiliary feedwater system with -- since the auxiliary feedwater system did not come on, may have been the contributor to begin with, that your chances of turning this undercooling transient into an accident with the consequences under the accident definition -- those are greater.

Q Referring to the table itself and the division of incidents from accidents, from severe accidents, would it be a fair statement that an accident would be a degraded incident and a severe accident would be a degraded accident? Do you understand the question?

A Yes. There certainly is a degree of increased number of incidents or a large number of incidents that would give you a higher probability of one of those incidents turning into an accident. A high number of accidents would certainly give you more of a probability of turning an accident into a severe accident.

So, I think that is what you are asking: Is there a relationship going from 1 to 3 there, incidents, accidents, and severe accidents?

Q That is the thrust of my question, and therefore wouldn't it be fair to say that if a particular parameter 1 through 7 had a large impact on the frequency of incidents, that it would simply by probabilities increase the likelihood of a severe accident?

A Well, I think you have to go through the middle one first. You have to go through accidents first, and the fact that these particular characteristics in the probabilistic analysis staff's assessment means that there is a larger probability of incidents in B&W plants than there is more of a probability of having an accident in a B&W plant.

However, they feel that that is not -- in overall characteristics it is not a large contributor. They say essentially a small increase in probability. Now, when that goes on to severe accidents, the chance of a severe accident is essentially on par with other PWR's, and the reason that they give for that is that if you look at the Release Categories 1 through 3 in WASH 1400 which this severe accident definition relates to, it implies that you have containment failure.

Containment failure is not dependent upon NSS. It is a balance of plant system. So, no matter what type of PWR you put in the containment, you are still dependent on the balance of plant to maintain that integrity. That is why for severe accidents there is a very minimal effect for a B&W plant versus another PWR design.

Q That was also my understanding when I read Chapter 7, that the probabilistic analysis staff in evaluating the impact on severe accidents was looking for whether the effect being considered, 1 through 7, would cause common mode type failures or have ancillary effects that would lead to a more severe accident.

With that understanding in the background -- Well, first of all, do you believe that is a fair understanding of what they did?

A Would you mind rephrasing that again, please?

Q Certainly. I am distinguishing two different things. One is a kind of a probabilistic analysis that one would do, let's say, in WASH 1400, where you took independent events, and by taking the frequency of those events and multiplying, you received the overall frequency for the combination of independent events.

I am distinguishing that from sort of common mode failure analysis where you look for, in effect, that in and of itself would lead to an ultimate outcome, and of course my reading of Chapter 7 that their probabilistic analysis staff in evaluating the effect of these parameters on the left here, on severe accidents, was doing the latter. They were really looking for whether the effect in terms of its common mode failure type effects would lead to an increase in the frequency of severe accidents. Is that correct?

A They considered that. Yes.

Q That leads to my question which I asked before, but I am not sure that I understood your answer, so I am going to ask it again.

You mentioned the fact that the NSS system is independent from the containment building that houses it and to have a severe accident you essentially have to have a failure of both, and because they are independent, there is no reason to suspect that the manufacturer of the type of NSS is going to have an outcome or effect on the frequency of severe accidents.

My question, though, isn't the frequency of severe accidents perhaps oversimplified? The frequency of NSS failure times the expected frequency of containment failure?

A I think it has some impact on it. I am not saying it is a completely negligible effect, but the effect is small when you consider that in order to have the containment failure for Release Categories 1 through 3, you are going to have to have a failure of several systems which essentially are independent and not subject to common mode failure.

For instance, you have to have a complete failure of auxiliary feedwater. You have to have a complete failure of high pressure injection. You have to have a complete failure of the containment spray system, and you also have to have a complete failure of the containment cooling system.

Q I understand that, but my question is, wouldn't it be true that if something had a large effect on the potential failure of the NSS system, and no effect whatsoever on the potential of the containment building, that it would still have a substantial effect on the overall frequency of severe accidents, and the reason why I ask that question is because it seems to me that for example, just for the purposes of example, if you doubled the expected frequency of NSS failure, that you would double the frequency of severe accidents?

A The answer I gave you is the best one I can. I do not think it is that simple, that you can double the frequency of accidents and have that consequently double the probability of a severe accident. You call upon additional systems to function, to mitigate the consequences of this severe accident that you may not call upon for an incident or accident, unless it progresses through.

Q Do you disagree with my assumption that you reach the probability of a severe accident by multiplying the expected frequencies of the multiple failures that would be involved in that accident?

A Yes.

Q You do disagree with that?

A No. I agree with what you said.

Q Okay.

MR. SHON: Mr. Ellison, isn't what you are driving at essentially the core of this Footnote 3? That Footnote 3 says there will be an increase in certain large accidents due to the increase in probability of a nuclear steam supply system failure, multiplied by the coincident failure-- it increases that only kind of accident that involves that kind of sequence. It might increase it a little, or it might increase it a lot, but I think the gist of what they are saying there is that it does not have a truly direct effect on accident sequences. It simply varies one of the factors in the probability.

MR. ELLISON: That is correct.

MR. SHON: It is not necessarily true. I think that this would be a large effect. It could be large, small, or indifferent, depending on how much those sequences contribute to the entire sum of Release Categories 1, 2, and 3.

MR. ELLISON: It was Footnote 3 that spawned my question, and the bottom line of the whole line of questioning for me is whether or not the expected frequency on severe accidents that is given in Table 7.1 took that into account or did not take it into account.

It was not clear to me from Footnote 3 whether the authors of this document were saying -- referring to Footnote

3 -- take a look at Parameter Number 2. It was not clear to me.

BY MR. ELLISON: (Resuming)

Q My question to you, Mr. Capra, is whether you can clarify this for me. It is not clear to me whether the authors are saying the effect on severe accidents is small, and that effect results from this increased frequency of incidents and the impact that it has on the total probability, or whether they are saying the impact is small and by Footnote 3 also noting that in addition to that small impact is the inherent increase of probability that results from just increasing any one of the parameters involved.

A Was that a question?

MR. SHON: I think in a sense what you are asking is whether the phrase "might rival dominant sequences in probability" means there might be an effect here for which we have not accounted that would be substantial compared with the known effects, that we have said it is small, but we recognize there is an effect for which we have not accounted that might rival the already known large effects.

Is that the way they mean it, or do they mean, we know it is small?

THE WITNESS: I really cannot answer that.

MR. ELLISON: That was the gist of my question.

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BY MR. ELLISON: (Resuming)

Q Mr. Capra, I gave you, this morning about 30 seconds before the hearing began, a copy of the NRR study that has been identified as CEC-26.

MRS. BOWERS: Wait a minute.

BY MR. ELLISON: (Resuming)

Q Referring to page 1-2 of NUREG-0667, about a third of the way down the page, there is a reference to this report. For the record, CEC-26 is the same document, is it not, that is referred to at page 1-2 as reference one?

A Yes, that is correct.

Q You are familiar with that document?

A Yes, I am.

Q Did you have a role in preparing CEC-26?

A No, I did not.

Q Would it be fair to say that -- it said in 0667 that the NRR status report provides the basis for the shutdown orders?

A That is correct. The CEC-26 -- it is my understanding that CEC-26 was prepared by the staff as a result of the preliminary findings which are documented in NUREG-0560, which has been referred to here as the Tedesco Report.

It is the generic assessment of feedwater transients from B & W designed reactors. Based on the work of that

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task force, they had come up with some preliminary findings that they felt were of a concern and generated this particular document, CEC-26.

Q It is true, isn't it, that Rancho Seco was shut down because of the concerns that are set forth in CEC-26?

A CEC-26 was a working document. It was essentially a status report briefing to the Commission. There was a great deal of discussion that went on a Commission briefings and between Mr. Denton the director of NRR and the individual licensees during those three periods -- during those three days or so in question that I was talking about.

The basis of the concerns and the things that were eventually agreed upon are contained in that NRR status report. That did serve as the crux of the reason for the shutdown, yes.

Q The Tedesco Report was not prepared until after the shutdown, is that correct?

A That is correct. The Tedesco Report came out in mid-May.

Q Could you refer to page 1-3 of CEC-26 on the copy you have, Mr. Capra. The page numbers are at the top.

Under the paragraph that is headed, "Defense In-depth." The very last sentence states "If HPI is initiated, this system could operate in the inventory road (since there is no LOCA) and balance losses through the release and

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safety valves.

"The mode of core cooling needs to be confirmed by further analysis."

Then it refers to section three. You may want to read the preceding sentences, but they essentially are referring to the feed and bleed mode, here.

A That is correct.

Q What further analysis has been done in the feed and bleed mode since this was written?

(Pause.)

A I do not believe there has been any substantial analysis to verify that feed and bleed will work. You cannot really do that until you get a qualified PORV that will allow you to feed and bleed through that.

Also, the same with safety valves. The safety vale or relief valve -- not the relief valve, the primary safety valves are safety grade. They still have not been qualified to pass either two-phase or solid flow.

So, it is not until you have some confidence in the fact that those valves will perform under that design condition, additional analysis is not really necessary, I do not think, at this time.

I believe that the concept of feed and bleed certainly is a viable concept, but you cannot really take credit for that type of core cooling until it is a proven

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fact.

None of the plants have been designed for that. We have seen it demonstrated, of course. It has happened a few times on various plants.

Q What is your understanding for the basis of the statement in CEC-26 that feed and bleed needs to be confirmed by further analysis?

A It is not a proven concept. It needs to be -- it needs to be developed further.

Q Referring to the next page of CEC-26, page 1-4, which is the conclusion section, the third item which appears at the top of that page refers to system design changes based upon the results of the first two items.

The first two items are further analysis and test on transient performance. The second is failure modes and effects analysis of the ICS. Of course, we have been discussing the FMEA in this proceeding.

It has been admitted into evidence. With respect to the FMEA, can you summarize what changes SMUD has made in the ICS based upon the FMEA?

MR. LEWIS: I guess the question calls for an objection. We did have -- we did have Mr. Thatcher, here, in earlier sessions. I think it was abundantly clear that he was the staff's expert on the ICS, and the failure modes, in effect, anaylsis of the ICS.

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It may well be that Mr. Capra also has some information on that, but I really feel like the question is looking for a degree of detail that we tried to develop during the time we went into this with Mr. Thatcher.

I cannot recall whether that specific item was asked and addressed or not, but I think the record that was developed there on the steps taken, with respect to the ICS is really the record that we should have on that subject, rather than trying to elicit it from Mr. Capra.

MR. BAXTER: It is my impression, too, Mrs. Bowers, that we were here to do with Mr. Capra on this round was to address changes from the draft copy of 0667 that was put into evidence earlier in this hearing.

Mr. Capra has testified as to the changes in the chapters. Of course, the additional chapters are seven and eight. I do not see what this line of questioning has to do with that material.

MRS. BOWERS: Mr. Ellison?

MR. ELLISON: Certainly. I do not agree with Mr. Baxter. This is our opportunity to, as I understand it, to examine Mr. Capra on 0667. We did not do that previously because, as everyone recognized, it was a draft.

Even had we done it -- strike that.

More importantly, with respect to Mr. Lewis's objection, I did review the transcripts over the weekend.

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Unfortunately, I do not have the cites, but I can get them. At the time that we examined Mr. Thatcher, two items were stated. First of all, Mr. Thatcher had not completed his review of the FMEA, apart from the Oak Ridge review, which was going to be submitted later by the staff in this proceeing.

It has not been submitted thus far. More importanly, Mr. Capra testified earlier in this proceeding that SMUD had sumbitted on January 21st of this year, their response to the staff's request that they analyze the FMEA and propose what measures they were going to take in response to it.

When Mr. Capra testified previously, he stated that the staff had not reviewed that response at that time. So, I think it is both relevant and perfectly in order with the ocurse of events that have taken place to ask Mr. Capra if they have reviewed that document and what responses SMUD has made since he did testify earlier on that subject.

Lastly, I would point out for the record that off the record this morning, I told Mr. Capra that I would ask him this question, and asked him to review the January 21st letter. So, I do not believe that he would be unprepared. If he is he can just say so.

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MRS. BOWERS: Mr. Lewis, do you have a response?

MR. LEWIS: Well, I think that that has helped to clarify for me what he is looking for. I will withdraw my objection.

MR. ELLISON: Would you like me to repeat the question?

MRS. BOWERS: The objection has been withdrawn.

THE WITNESS: I understand what your question is. You said -- you asked me what action SMUD has taken as a result of the FMEA. As you brought out, they have submitted, and I testified earlier, on January 21st, 1980, they did respond to our request of November 7, 1979. In that request, the staff had asked all the B&W licensees to identify what actions they had taken as a result of the recommendations contained in the ICS reliability analysis.

I mentioned at that time -- I didn't have the letter with me -- that they had made that response, and it has not -- staff evaluation of that particular document had not been made. The status of that has not changed. We do not have an evaluation of this particular document available at this time.

Subsequent to Mr. Thatcher finishing his testimony here, he has gone back and has been working to try to keep up with his commitment that he made to the Board to complete that analysis within about 30 days. I have talked with him on several occasions since, and he is working on that. Until that evaluation is complete, and the staff criteria is identified, we cannot take the individual letters of the licensees and say whether they are acceptable responses until we know what the staff position is.

However, if you -- to answer your original question, what actions have the licensees taken, they are identified here. I can go through them. There are essentially six items that were identified in the reliability analysis which we had asked the licensees to give us the status of action on.

The first one was the non-nuclear instrumentation/ ICS power supply reliability. The licensee has made substantial changes in there on nuclear instrumentation power supplies, some of which are identified in this particular letter. I don't know the exact status of that at this time. I know that the items have been completed. The acceptability of design changes, as I said, I am not prepared to address, but the power supply reliability has been increased.

The second item dealt with reliability of the input signal from the non-nuclear instrumentation/reactor protection system to the ICS, specifically, reactor coolant flow signal. In the District's response, the District said that they are considering changes, two changes in this particular area. The first is, they are considering changes in the jack, or hard wiring the flow signal to the ICS. Secondly, they are also considering the use of auctioneered reactor coolant system flow input into the ICS. However, this work has not been complete yet.

They state in this letter that current engineering workloads make near-term analysis of these potential improvements unlikely. So, they have considered these, but unless something has changed since the submission of this letter, these actions have not been completed.

The third item was ICS/balance of plant system tuning, particularly feedwater condensate system and ICS controls. However, we had asked specifically in our November 7th letter for three subparts of that. Essentially we had asked them to identify any previous problems thatthey had experienced with respect to start-up and shut-down since we mentioned on the record last time that some of the problems that we had seen in the past seemed to be not necessarily with the ICS but in the transfer of feedwater control from manual to automatic or vice versa, on a shutdown.

We asked them for the bases of operator intervention in ICS, and also asked them for the procedures that were used to perform the operation, essentially transferring back and forth. There is a very extensive response on this particular item. The majority of this document, the January 21st letter, supplies the details, essentially three plant procedures to control ICS. There is System Operating Procedure A71, Procedure B2, plant heat-up and start-up, and B4, plant shutdown and control.

The answers to our three subparts are contained in those procedures. In addition, they go through what training operators are given in the ICS, about a two or three page response to that, both the hot license and requalification phases.

The other three areas were classified in the B&W report, 1564, as mainly balance of plant areas,but related. The first one dealt with main feedwater pump turbine drive minimum speed control to prevent loss of main feedwater or indication of loss of main feedwater. The District's response in the January 21st letter states they currently are considering the purchase of a new main feed pump control system, whereas the system would have dual control oil systems; either of the control oil systems would be able to control the main feed pumps at minimum speed. If the system is purchased, it may be installed during the 1981 refueling outage.

As far as I know, there hasn't been any change to that.

The next item dealt with a means to prevent or mitigate the consequence of a stuck-open main feedwater start-up valve. Essentially they feel, the District feels that there is no action required on this particular item. They go through a qualitative assessment of the effects of a stuck-open start-up feed valve, and essentially there is no effect during power operation. During power operation, of course, the start-up valves are fully open.

However, there could be effect less than 15 percent power. However, they feel that the response would be slow enough where the operator action would certainly be able to catch it in time, before any undesirable consequences took effect.

And the last one is a means to prevent or mitigate the consequences of a stuck-open turbine bypass valve. The District had experienced, or Rancho Seco had experienced a stuck-open turbine bypass valve early on in the operation of the plant. I believe it was during start-up testing. However, they feel that seeing how upstream of these valves there are manual valves which can isolate the turbine bypass valves, no additional action would be necessary on this particular recommendation.

I guess in summary I would say that out of the six items, they have definitely taken action on the Number One item which was of concern, which was the NNI/ICS power supply reliability. The -- there are a couple of recommendations which they are still considering action on, but it does deserve further analysis on their part, and a couple of the recommendations which they feel have sufficient justification not to take any further action.

As I mentioned, the staff evaluation of this response is not complete, but I would expect it to be complete shortly, as soon as the overall generic assessment of reliability analysis is complete.

MR. LEWIS: Mr. Ellison, let me interrupt for one second, because I am becoming concerned about what I think is a non-continuity of understanding as to what it is I understand the staff owes to the Board and parties on the subject. It is my understanding, and I so instructed Mr. Thatcher, that the Board had requested to see the staff's analysis of whether or not it was going to adopt at this time and require of licensees at this time that they take actions with respect to the recommendations of the Oak Ridge National Laboratory Report, which analyzed the B&W failure modes and effects analysis, and that is what Mr. Thatcher is preparing and what we hope to supply to the Board and parties.

I have no recollection of having been asked to or undertaken to provide you with the staff's specific response to the January 21st, 1980, letter from the District, and if I am wrong about that, I suppose I should know it, but that is my understanding of what I was asked to do. That is my understanding of what is being --

MR. BAXTER: That is mine as well, and I think, given time, we can find transcript to support that, that the Board said -- Mr. Thatcher testified he had not completed any written analysis of the ORNL conclusions, and the Board stated that they would hold the record open for that written analysis.

DR. COLE: That was my recollection.

MR. ELLISON: I think it would be a useful addition to the record to have the staff's evaluation of SMUD's specific response to the FMEA, but it is my recollection that the Board has not, at least until now, requested that.

MR. LEWIS: Well, it may from your point of view be a useful addition to the record, but I don't have it, and it is not my understanding that the Board had asked that such an item be included in the record, so I am preparing what I was directed to do, and that will be available, but --

MRS. BOWERS: I personally don't recall that we said we would keep the record open for this, but I sometimes have a bad memory. I know we said we would keep the record open until the final 0667 came out.

MR. LEWIS: In fact, what I was really planning to do was send to the Board and parties for their information Mr. Thatcher's status report on where things stand with respect to the Oak Ridge National Laboratory report. It was not -- based on my assessment of where things were left, it was not my view that that necessarily had to be put into the record, but my recollection is that what was asked is that people wanted to see where that stood, and is how I intended to proceed with that.

I am through.

MRS. BOWERS: After you do that, Mr. Lewis, of course, any party or the Board could then raise questions, and ask for something further.

MR. LEWIS: That is always within their prerogative.

MR. ELLISON: Mrs. Bowers, I expect that we are not going to receive the staff's -- correct me if I am wrong, but I expect that we are not going to receive the staff's review of the FMEA until after the hearings on this matter are concluded. So, I think our opportunity to ask questions on it may be now.

MRS. BOWERS: Well, if the record is kept open, then a motion could be filed, whatever would seem appropriate at the time.

BY MR. LEWIS: (Resuming)

Q Mr. Capra, returning to your summary of the District's January 21st response, do you know whether the changes that were made in the NNI power supply, Item 1, were made in response to the FMEA or were made in response

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to the FMEA or were made in response to perhaps the lightbulb incident?

A I think it was a combination of both. I know they had several -- they had made -- originally made, following the lightbulb incident, some changes; however, additional long-term modifications were considered, and some detailed analysis was done by the District, and I think that absent the FMEA, that these particular changes may have been made anyway, but I think the important thing here is that the changes were made regardless of what the source was.

end SRB 1
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5/12/80
S-2 folw S-1
SRB/srb 1

Q And it is correct that those changes were represented in the January 21st letter as have been completed, is that correct?

A No. Some changes were made prior to the submission of the January 21st letter. However, additional changes were made during the current -- the previous refueling outage which has just been completed.

Q So it's your understanding that all the changes that are referenced in that letter have been made at this time?

A With respect to NNI/ICS, I believe that's correct.

MRS. BOWERS: I need a little help to find out about where the January 21 letter is. I just went through CEC's exhibits and didn't see it. Did I miss it?

MR. ELLISON: No. It's not been identified in this proceeding.

BY MR. ELLISON (Resuming):

Q Referring you to page 1-5 of CEC Exhibit 26, Item 1 -- and this is still part of the Conclusion section -- states or poses the question: "Do challenging events arrive at a frequency high enough to be of concern." And answers it by saying, "Yes." In light of the staff's review which is set forth in 0667, would it -- is it your understanding that the answer to that question would still be yes?

A Yes.

Q Further down the page, the next item, number 2, poses the question, "Does the ICS perform satisfactorily?" And the first item listed underneath that question is B&W has stated and we agree that, "...we are not satisfied with the reliability of the integrated control system."

This raises a number of questions in my mind that I'd like to pose to you. First of all, do you know where and when B&W made that statement?

A No, I don't. I have not seen that written anywhere.

Q Secondly, it seems to me that the ICS reliability study and also the Oak Ridge review of that study, as you recall, that study was divided into two parts. There was the FMEA and then there was the summary of operating experience, and a great deal of reliance, at least in my mind, was placed upon the summary of operating experience section, particularly by Oak Ridge.

Inasmuch as the staff felt, or at least the authors of the NNR report felt on April 25th, that the operating experience with respect to the ICS had not been satisfactory, can you tell me whether in light of 0667 the staff has changed its conclusions with that respect?

MR. BAXTER: Excuse me, can I have clarification about the staff or the authors' statement that operating experience had not been satisfactory?

MR. ELLISON: I'm referring to the statement that appears on page 1-5 of CEC 26 under Item 2(a) that the staff agrees that they are not satisfied with the reliability of the ICS.

MR. BAXTER: But it doesn't indicate in any way that they have examined any operating experience. I think that mischaracterizes the statement.

MR. ELLISON: Let the record reflect your statement, but I think the statement at 1-5 certainly raises that inference.

THE WITNESS: I think that at the point where this statement was made, as Mr. Baxter pointed out, we probably had not reviewed any operating experience. A lot of our concerns about the integrated control system were based on myth and folklore I think a little bit. We had not done any review of the integrated control system; we were concerned that it was possibly a contributor to the transients experienced in B&W plants, and it was logical that we wanted to investigate that.

BY MR. LEWIS (Resuming):

Q Referring to Item (e) further down the page, 2(e) where it's stated that "Even when the ICS works well, there may be a response to a feedwater transient; wide swings in reactor pressure, pressurizer level and average reactor coolant temperature." What's your understanding of the basis for that statement, at the time it was made?

A Well, there have been transients that have occurred at B&W plants in a post-trip situation. The control of feedwater or miscontrol of feedwater has led to, as it says here,wide swings in reactor coolant pressure, pressurizer level and reactor coolant temperature. Now, whether that is a problem associated with the integrated control system or whether feedwater control is shifted to manual and it's operator error, I'm not sure of the basis for the statement but it's a fact that we haven't seen that before.

Q This statement suggests that there wouldn't be a problem in the ICS. It begins with, "Even when the ICS works well..." and then goes on. So is it your understanding that the basis for this statement, given that introduction, was simply that there had been transients with wide swings as are described, or do you know whether there have been any reason for the first part of the statement, that swings resulted even when the ICS was working properly?

A Not taking part in the generation of this document, I don't know the bases for the statement, but it is a fact, and I have seen transient responses, which have resulted in this type of behavior. And specifically, we've mentioned a couple of times in this proceeding the April -- excuse me, the August 23rd, 1979 meeting with the B&W licensees to discuss post-TMI feedwater transients in B&W plants.

Specifically, I can recall the Crystal River transients which I mentioned before where they tripped the reactor four different times within a 24-hour period while trying to operate on three reactor coolant pumps. At that particular time, there were feedwater control problems. However, it was not attributed to faults in the ICS; they were attribuced to trying make a transition from manual operation of feed control into automatic.

There were other transients identified during that meeting in which similar plant response was experienced, but I don't beleive that any of the transients that were discussed during that meeting were associated with any ICS malfunctions.

Q Referring to NUREG 0667, Section on the ICS and the NNI which is 5.3.1 et cetera, the staff concludes with a number of recommendations for changes in the ICS power supplies and whatnot. I'm referring to page 5-61 where the conclusions and recommendations begin.

MR. BAXTER: Excuse me, Mr.Ellison, I'm having trouble hearing you. What was the page number?

MR. ELLISON: I'm referring to page 5-61 and subsequent pages where the conclusions and recommendations for that section appear.

BY MR. ELLISON (Resuming):

Q The first conclusions, which appears on 5-61 at the very last sentence states, "Third, the normal control systems should be improved to reduce the number of challenges to the safety systems." The normal control system would be the ICS, isn't that correct?

A As used here, I believe that's correct.

Q So would it be fair to say that as of the writing of 0667, the staff is still concerned about the performance of the ICS?

A Let me modify my answer a little bit when I said the ICS. I think we may have a definition problem again. I think we're really talking about not just the ICS cabinets, but the entire control system itself which, of course, talks about the input signals to the ICS. And throughout the rest of the previous discussion in this particular chapter I think it's pointed out that that does appear to be the problem. I don't think anywhere in here in this particular report you'll find any identified problems with the ICS itself.

Q With respect to the ICS as broadly defined, including its inputs and whatnot, would it be fair to say that there are identified concerns here?

A Yes. An example, for instance, is the February 26th event at Crystal River.

Q The Task Force goes on at the bottom of 5-61 and 5-62 to make a number of recommendations with respect to the ICS and its power supplies and whatnot. Do you know whether the District's response to Item 1 in your January 1st letter suggests that they've taken the actions that are recommended here?

A No, not all of the actions. There are some.
Q Could you identify which ones remain to be done?
A I think they are too interrelated; I really can't do that.

Perhaps if I can go back to that answer, I can give you a couple of examples, for instance. I believe Subpart (a) of the recommendation, "The power buses and signal paths for non-nuclear instrumentation and associated control systems should be separated and channelized to reduce the impact of the failure of one bus." I'm pretty sure that the work that has been accomplished there meets the intent of that particular subpart of the recommendation. I'm not saying that it meets it 100%; that's basically the intent of what the District was trying to accomplish during modfications that they made.

Now, certain other subparts, I feel that no action has been taken because they're newly identified by the staff. We're not even sure if they're practical to accomplish. For instance, Subpart (d) talks about "The control system failure as a response to failed input signals can cause substantial plant upsets requiring action by engineered safety features or safety valves in addition to reactor trip. The control system should have provisions for detecting gross failures and taking appropriate defensive action automatically, such as reverting to manual control or some safe state." Now, the

District has not taken any action on that; however, the Task Force is not really sure that that's feasible or even desirable. It's one of these types of recommendations that needs analysis, it needs work to see if it's feasible. So I would not expect that they would have taken any action on this. Even if they would have been directed to, which they have not been at this point.

Q You're familiar, aren't you, with the testimony of Mr. Rodriguez in this proceeding that Rancho Seco -- and I'm sure I'm going to mis-state this technically. But that Rancho Seco has indication that is powered by one NNI bus and transmission of indication essentially is powered by another. Do you recall that testimony?

A Yes.

Q Does that meet the intent of Item (a)?

A That's one of the things I said that there are possible exceptions to that, but I think the general intent of the work that they have done is to accomplish items such as Item(a).

MRS. BOWERS: Excuse me a minute, I'd like to speak to the people who just came in the room. Are you acquainted with the procedure that the Commission has set up for these hearings? You cannot use special lighting; you have to use the lighting that's in the room, and then you cannot roam the room; you have to be in one stationary spot for your

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photography, while we're in session.

(Short pause.)

Do you have much more, Mr. Ellison?

MR. ELLISON: I have several more questions. It might be a good time to take a break, if that's what you're thinking.

MRS. BOWERS: We'll take a break, a 10-minute break.

(Short recess.)

S-3 flws.

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Tape S3
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MRS. BOWERS: We are on the record.

BY MR. ELLISON (Resuming):

Q Mr. Capra, I understand that one of the principal concerns of the staff which led to the shutdown and is stated in CEC 26 was that the ICS might simultaneously cause a feedwater transient and inhibit the AFW system. Is that also your understanding?

A That may have been an initial concern. As I said, not having been on the ground floor of the developing of this, I think that perception may have existed, yes.

Q Could you refer to 0667, page 5-58, and specifically to the second full paragraph beginning, "Simultaneously..." near the bottom of the page.

DR. COLE: I'm having trouble hearing you, Mr. Ellison, could you move the microphone a little closer to you?

MR. ELLISON: Could you hear the references I just gave you?

DR. COLE: Yes.

BY MR. ELLISON (Resuming):

Q The first question is with respect to the requirement from NUREG 0578 that's referenced here to install a control grade automatic initiation. Would this be an automatic initiation of the AFW system on safety feature signal?

A Not just necessarily a safety feature signal. It was one of the other recommendations that the actual auto start signals do need to be evaluated to be sure that we're actually using the appropriate signals. There's a table in NUREG 0667, one that I made a correction to on Saturday, which shows the various signals which are used for all 9 plants, but they all vary. Whatever the initiated signals are however, they need to be safety grade.

This particular item in NUREG 0667 -- correction. In NUREG 0578, really just talks about initiation; safety grade initiation. What we're talking about here in 0667 is a little more than that; safety grade control and initiation.

Q So both of those requirements would address the initiation signals as they presently are. Is that correct?

A I don't understand what you mean.

Q For example, Rancho Secc has a safety grade initiation of AFW on SFAS.

A That's correct.

Q Would I be correct in stating that the NUREG 0578 requirement would also require in the long term safety grade signals in addition to SFAS if they're already there, which they are?

A Yes.

Q Let me clarify. When I said "which they are," I didn't mean as safety grade, but there are additional signals which would initiate AFW to SFAS.

A Yes. We've also identified in -- I don't know the specific date of the letter, but in our letter in which we sent to the District our review of the AFW reliability analyses. One of the items in there was for them to consider automatic feedwater initiation on low steam generator level, which is not a signal that they presently have.

Q The last sentence in that paragraph on 5-58 suggests that the implementation of this requirement should effectively remove initiation of the auxiliary feedwater system from the ICS. Is that a correct reading of that statement?

A That's correct.

Q First of all, if the ICS is considered to be a reliable system, why is the staff interested in removing the initiation of the auxiliary feedwater system from it?

A We want a fully and safety grade auxiliary feedwater system which includes initiation and control. I think it was previously brought up in staff testimony when we had the panel here in the second session that this has already been committed to by the District, and they intend to implement a fully safety grade auto initiation and control system for auxiliary feedwater by -- I believe they've committed to the refueling outage of 1981. Now, there still is -- we have not necessarily accepted that particular date, as of now. Our requirements still are by January 1981.

DR. COLE: Excuse me, Mr. Ellison. I thought that

that was a system that was independent of the ICS, but I didn't know they were going to disconnect the integrated control system from that.

THE WITNESS: From the auxiliary feedwater system, yes. Right now, the ICS is tied in with the auxiliary feedwater system for normal control of the auxiliary feedwater system. For instance, on a loss of feed, the ICS will control steam generator level at the low level limits. Or, if you lose reactor coolant pumps, then it will automatically feed the system up to the 50% level in the operating range.

However, during an SFAS signal, initiation of the auxiliary feedwater is initiated completely independently of the ICS and the actual flow path goes through the SFAS or the AFW bypass valves, vice the ICS flow control valves.

DR. COLE: Fine, I think we're talking about the same thing. Thank you.

MR. SHON: Under those circumstances, the SFAS initiation, what controls auxiliary feedwater?

THE WITNESS: I don't believe there is control. Those valves go wide open and the pumps come on and it takes manual operator action to throttle down AFW flow.

MR. SHON: Thank you, I just wanted to establish that. Please proceed, Mr. Ellison.

BY MR. ELLISON (Resuming):

Q I'm particularly interested in the initiation of AFW on loss of main feed rather than on SFAS. It's my understanding that it would be typical for a loss of main feed transient to result in high pressure in the RCS, and that you wouldn't reach SFAS for some time.

A That's correct.

The initiating signals are not dependent on ICS for loss of feedwater. They don't go through ICS. The control, the level control for the steam generator, goes through ICS.

Q Is it still true today at Rancho Seco that on a loss of main feed, the ICS would be controlling the auxiliary feedwater flow?

A Provided it functioned properly, yes.

Q Recognizing that SMUD has developed procedures for the operator to take manual control of auxiliary feedwater in the case of an ICS malfunction, can you tell me whether any other action has been taken with respect to the concern that on loss of main feed, particularly from an ICS failure, that the ICS might fail in such a way as to also improperly control the AFW system?

A I'm not sure if I understand your question.

Q Okay, let me repeat it. Recognizing that procedures have been developed at SMUD for the operator to take control of the AFW system, apart from that, has anything been done since Three Mile Island to insure that on a loss of main feedwater, the ICS would properly control auxiliary feedwater flow?

MR. LEWIS: Mrs. Bowers, I'm going to interpose an objection. I think that we are offering Mr. Capra to testify on 0667. Admittedly, 0667 is a comprehensive document. We have had numerous witnesses earlier in this proceeding, I'm thinking particularly of the first staff panel, who were available to be cross examined on such items as how the ICS functions and the particulars of that. And I think we're getting into questions now that are very specific questions about ICS functioning, AFW functioning. We had an ICS witness, Mr. Thatcher. We had an AFW witness, Mr. Matthews. And we did have 0667 although in draft form, available at an earlier point in this proceeding.

I just -- once again, Mr. Capra may be able to provide some answers to some of the questions being answered, but I really think that we're developing a record which I really thought we had already developed, and we're developing it not through the cognizant staff person. So I think that we should confine the cross examination to the 0667 document and to the recommendations of the 0667 document.

MRS. BOWERS: Mr. Ellison?

MR. ELLISON: Mrs. Bowers, very briefly, it's my opinion that's exactly what I'm doing. I'm interested in this discussion in this third paragraph on page 5-58 of 0667, and admittedly it is a comprehensive document. And this paragraph raises questions in my mind about where we stand today with respect to the ICS and some of the recommendations that are made in here address that problem as well. So I think it's perfectly appropriate for me to address these questions to Mr. Capra at this time.

If Mr. Capra doesn't have the answers, he can certainly say so. I'm not suggesting that Mr. Capra should give answers in areas that he's not knowledgeable of. I agree with Mr. Lewis that that would not create a good record. But if he has the answers, I think it would create -- there's nothing inappropriate or nothing that would detract from this record if he provides the answers.

MR. BAXTER: I agree essentially with Mr. Lewis. I think that we did go over much of this material with the staff witnesses who were offered earlier on. And I think if you've had occasion to review 0667, they did a very conscientious job of trying to rehearse and summarize all of the attendant requirements and changes that the Commission has done. So to the extent that the document does make reference to other things that have been required, we could reopen the entire record of the proceeding and go through it all again, but I don't think that's the guts of what Mr. Capra is here to testify about. It does make reference back to those things but I don't think anything has changed here,

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as Mr. Capra's testimony earlier today already indicated.

And not only was there opportunity to cross examine on this particular section of 0667 last time, but in fact, there was cross examination by me and the Energy Commission last month.

P4 flws.

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tS-3

MRS. BOWERS: Mr. Lewis, do you have a further response before the board considers?

MR. LEWIS: Yes. I think that the question of the status of the ICS and the -- the time that Mr. Thatcher testified earlier, the procedures to operate the AFW system independently of the ICS were already in place.

Although, there may be for the recommendation now in 1667, I do not believe that the factual situation is altered.

I would have to look back at the previous record to see exactly what cross examination did take place of Mr. Thatcher on these points. I don't have the recall of exactly what it was, but it seems to me that the testimony that we had in the proceeding at that time, although it was in advance of the issuance of 0667, was based upon the same factual setting.

I think that that was the place for it. I don't know whether it was explored or not, but even if it was not, that was the place to explore these rather precise questions about the ICS system.

(Board conferring.)

MRS. BOWERS: Mr. Lewis, the board is going to overrule your objection. We recognize the reality of the situation here. We heard the testimony from Mr. Thatcher and Mr. Matthers at least a month ago, some of it six weeks

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or more ago.

Some of it was early March. Here we are in the middle of May. Also, this was -- the draft was a very large document to be thoroughly reviewed at that time. Now, because of Crystal River, we recognize there was essentially a week's delay before we proceeded with the panel.

Anyway, we think that Mr. Capra may well be able to respond to your questions. If he cannot, it is certainly understandable because there were people here who were far more familiar with the details of these systems.

MR. ELLISON: I would like to make two points for the record. First of all, one, to clarify the questions that I am going to ask and, secondly, Mrs. Bowers, you mentioned a moment ago that with respect to the distribution of 0667, there was a week because of the Crystal River evant to examine that document.

I believe you are thinking of 0565 which did appear at about the same time as the Crystal River event. There was, I believe, a two or three day delay in the proceeding at that time; 0667 came out later. There was no delay in the proceeding as a result of that.

With respect to the questions I am going to ask, I am most concerned, Mr. Capra, with your conclusion or the task force's conclusion at 5-61, which I referred to earlier but I will refer you back to it.

The first conclusion that appears there states that the auxiliary feedwater system must be highly reliable and independent of the normal control system.

BY MR. ELLISON: (Resuming)

Q My questions really are addressed to the basis for that recommendation. Referring back again to the preceding page, 5058, the question that I posed earlier was: Apart from the procedure changes, do you know whether there has been any action at Rancho Seco to assure that the ICs does not cause a feedwater transient and simultaneously fail in such a way as to improperly control AFW?

A There have been no changes that I am aware of in the ICS cabinets themselves that we talked about. However, the actions that they have taken with respect to the reliability of the power supplies and the input signals to the ICS by taking those actions makeing the system more reliable, it has had the net effect of increasing the realiability of the integrated control system, itself.

If you look at it as an entire system, including non-nuclear instrumentation inputs. Given a failure in the ICS itself, it could still have, of course, the net effect of not maintaining auxiliary feedwater at the desired level.

However, the procedures have been developed. We have audited those procedures and checked the operator's understanding of those procedures to take manual control, to

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take corrective action.

Q Is the task force conclusion that I read to you earlier, or refered to you earlier, about taking the AFW system out of the ICS based upon a dissatifaction in the long-term of relying upon the procedures to independently control AFW?

end tP-4

A Yes.

tP-5 flws

Q I am going to apologize for this question in advance, but I do not have a specific reference. I recall however that one of the recommendations of the task force was that the licensees or B & W's study possible design changes to reduce or remove the OTSG senstivity. Is that correct?

A Yes.

Q What specific types of design chagnes do you have in mind?

A It is recommendation ten.

MR. LEWIS: What page does it appear on, Mr. Capra? What page is it discussed at? Is there a particular section of a document that we can be looking at?

THE WITNESS: Yes. It should be page 5-19. This is an area where we have made the recommendation very broad because we are not sure what the best fixes are, or if the possibility exists that we can reduse the sensitivity of the once-through steam generator.

However, there are certain means by which we think it may be possible. Until the analysis is done, we are not really sure what the actual benefits will come out of it -- examples will be that we considered within the task force.

Our -- I think, for instance, to have the facility operate with less superhear, operate at a different level, or a level control in the once-through steam generator which would be a higher level.

It is not operating at a specific level now, but based on steam pressure and the amount of superheat, one passive method that was discussed that we are not sure of the feasibility is possibly providing a surge tank effect, or a surge tank on the feedwater lines themselves, such that if you had a loss of feedwater, you would have a surge volume similar to a core flood tank which would provide passively feedwater for a certain period of time which would give you a longer time to get on the auxiliary feedwater system to prevent the steam generator from drying out.

It is possible to change set points on the secondary side, either on the turbine bypass valves -- maybe I said steam generator bypass, turbine bypass valves, or steam generator safety valves.

There are a lot of possibilities. Until sensitivity studies are done to see if they are feasible and what net effects they would have, it is not possible to be definitive

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on what the best way to go would be.

Q Do you think it is a fair statement that it is going to be a long time before -- even if design chagnes are found to -- that will reduce the sensitivity before those changes can be identified, reviewed, and implemented at Rancho Seco?

MR. LEWIS: Mr. Ellison, could you tell us what a long time means to you? It is a very amorphous term.

MR. ELLISON: It is a fair statement.

BY MR. ELLISON: (Resuming)

Q Do you think it could be done in within two years?

A It is possible. It depends on what the analysis comes up with, specifically what needs to be done, and what the best course of action is. This would not be treated, most likely, as a separate item.

For instance, if you look at recommendation number nine, which is system response modification to prevent pressurizer level loss and ECCS actuation, and look at recommendation 19, which talks about performance characteristics for response to anticipated operational transients, I think those three will probably be taken as a whole, and see if a solution to all three can be found at once.

There has been action -- I am not sure of the exact status of it, but B & W and B & W licensees have discussed taking all three of those recommendations for action now.

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B & W is pursuing ways to see if it is feasible to accomplish some of those studies.

As I have said, I have not heard anything formal about it, other than it was discussed in one of our meeting with B & W and the licensees that they were looking at ways to do those three together.

Q So, you believe it is possible that the sensitivity could actually be minimized, or maybe even removed within two years?

A That would just a guess on my part. .. really do not know. We had envisioned, for instance, recommendation 19, the development of performance criteria to actually be applied to all light water reactors, or all PWRs.

Possible different performance criteria for response for BWRs. If that was the case, that would involve rulemaking and changing the recommendations and all. So, in that particular case for it to be adopted Commission-wide, as part of the regulations, it would take a long time.

However, that is not to say that B & W or the licensees could not develop their own criteria and apply that to their plants.

Q Lastly, I have a couple of general questions about the way this document was prepared. Mr. Baxter asked you some questions pertaining to the thought that had gone into the recommendations and whether or not you had considered interactions between the recommendations and the incorporation of them into the system. Did I correctly understand your testimony on Saturday that you had not considered what the impact of these recommendations taken together would be upon the operation of a given facility?

A No, I did not say that we had not considered them. Now, these recommendations are recommendations that appear on the surface to the task force to be good solid recommendations that should be pursued.

As pointed out in section seven, it is quite possible that one, two, or more of these recommendations may have some detrimental effects. The reason that the task force has still -- still feels that all of these recommendations should be purused is until it is actually determined whether these things are feasible to accomplish, whether the good points would outweight the bad points, it is not clear.

What I had said to Mr. Baxter was, I believe he was concerned that whether a separate task force was put together. We went and we reviewed various things without taking a look at other reauirements that had been already levied on B & W plants.

His question to me was: Have we considered the requirements that have already been imposed on B & W plants, or actions they have taken on their own? We any of these --

bfm9

was it possible that some of these recommendations that we made be in direct conflict with those of other requirements?

My answer to him was that being fairly familiar with the requirements that had been imposed on B & W plants, I see in none of the 22 recommendations or the recommendations that are actually require licensee action that are in conflict with any of the previous requirements for the B & W plants.

Q Have you completed your answer?

A Yes.

Q Do you see any of them that are in conflict with one another?

A Do you mean out of the 22, are any in conflict with one another?

Q Yes.

A No, I do not see it that way. It may be possible, depending upon what -- I think I mentioned in Chapter 8 or section 8 there, that it is quite possible that by doing certain recommendations that may negate the necessity to do certain other recommendations.

There may be alternatives which are proposed by licensees to meet certain goals intended by our recommendations that, again, may negate having to follow through on certain other recommendations.

An alternative, for instance, that I can think of

bfm10

is one of the recommendations in the report deals with having a high radiation signal which would isolate containment in purge.

Now, if the licensees had committed to not -- to only purging during cold shutdown, there would be no necessity really to have that signal.

So, I am not saying that is necessarily an acceptable alternative, but it certainly seems reasonable on the face of it that that would be an acceptable way to go.

MR. ELLISON: Mrs. Bowers, at this point, I would move the admission of CEC-26.

MRS. BOWERS: Mr. Baxter?

MR. BAXTER: I would oppose the offer, Mrs. Bowers. We have not had a witness here to sponsor this document. I believe to the extent that the matters in this document are discussed, they have been updated quite extensively by other staff witnesses who did appear here in person to sponsor thier views.

I understand Mr. Ellison's interest goes towards that basis of the May 7 order. I would submit this was a report by a group of the staff. In my view, the Commission's basis for its May 7 order is adequately set forth in the order itself, which discusses the phenomenon which the Commissioners would be concerned with.

MRS. BOWERS: Mr. Lewis?

bfm11

MR. LEWIS: The problem with its admission, it seems to me, is that there are so many other documents that speak to the same question.

For example, 0560, NUREG-0560, well, that is not in evidence. My concern is that this document which represents a very early document in the development of the staff's views with respect to the sensitivity of B & W reactors standing in the record alone could create a misimpression of the totality of what there is that has been investigated with respect to this subject.

I am not proposing, by the way, that all those other things come in, because there are a lot of them. I do not think that is the way to develop the record at this point.

I think it has been identified. I really do not think that the statements inthis document should carry evidentiary weight. At to what the thinking is now -- well they certainly could not carry evidentiary weight as to what the thinking is now, with respect to sensitivity of the B & W reactors.

I suppose they could arguably carry evidentiary weight as to what the thinking was at the time the order was issued. I just have problems with this one document coming into the record and standing there in the record without all these other things.

I am taking the position that the testimony offered by the staff including the May 7th -- including the June 27th review of compliance with a short-term modification really cover the territory of what we felt had to be on the record.

So, I would object to its admission.

MRS. BOWERS: Mr. Ellison?

MR. ELLISON: Mrs. Bowers, I believe this is a very simple problem. As Mr. Lewis has stated, this document does have valid evidentiary weight with respect to the thinking of the staff at the time that the shutdown was conceived.

There are subsequent documents, but they were prepared after the shutdown. This document was referenced in 0667 for precisely that purpose, for being the basis, at least in part, of the shutdown order that we are considering in this hearing.

I think it is, on its face, obviously relevant to this proceeding. In addition, Mr. Baxter pointed out in raising his position, that there had been subsequent changes, that there had been subsequent analyses. We would not offer this document as replacing that analysis, but as a basis for comparison of where we were at the time that the plants were shut down, and compared to where we are today, which I think is well summarized in 0667.

bfml3

I think it would be very instructive for the board to have this document before it to provide that basis for comparison.

MR. BAXTER: To repeat my very fundamental problem, Mrs. Bowers, the staff produces a lot of documents. We, in this hearing, build a record on the testimony of witnesses. I think it is clear we have not had a witness here who has testified to the truth of the matters asserted.

We could all walk in with lots of staff documents and offer them into evidence. I do not think that is the way you build a reliable and soulnd record.

MRS. BOWERS: I have one more question before the board considers this. Mr. Lewis, did I understand your position correctly? You consider this somewhat of a historical background document?

MR. LEWIS: Yes, ma'am.

MR. BAXTER: We do not know whose thinking this represents, however. The offerers have not been identified, to my knowledge.

MR. ELLISON: Mrs. Bowers, we do know whose thinking this represents. It represents the Office of Nuclear Reactor Regulations, which was responsible for the shutdown of these facilities.

Mr. Baxter's point basically -- assuming for argument's sake that this is correct, that this is hearsay,

bfm14

that there has not been a witness offered in this -- I have two responses on this.

First, hearsay that is realiable, is admissible in administrative proceedings. So, the question is that is not dispositive of the fact that it may be hearsay evidence that would go more to the weight that the board might give it.

For the purpose of examining what the NRR thinking was at the time the shutdown was made, I think this document has a great deal of credibility, and is recognizing 0667.

The second point with respect to the hearsay is that this document -- I think that the importance of this document has been recognized by all the parties in this proceeding. It has been available to them throughout the cross examination of the various witnesses.

There have been witnesses from NRR who have appeared, who could have been cross examined on it.

MR. BAXTER: Hearsay is a very interesting argument. It is not the one I made, however, Mrs. Bowers.

MRS. BOWERS: A very minor logistics problem, Mr. Ellison. As you know, the copies that were furnished, somebody with a yellow wax pencil did some marking out.

MR. ELLISON: We have additional copies that don't have that problem. We will distribute them.

MRS. BOWERS: That reproduced black.

bfm15

DR. COLE: They were all the parts he felt were significant.

MR. ELLISON: That is right.

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(Board conferring.)

MRS. BOWERS: We are going to admit the document into evidence, which is CEC Number 26. It is dated precisely April 25, 1979, and we will give it the weight that we think it is entitled to.

(The document referred to was marked for identification as CEC Exhibit Number 26 and was received in evidence.)

MRS. BOWERS: We do want better copies.

MR. ELLISON: Do you want them now, or do you want them at the break?

MRS. BOWERS: I think the break would be sufficient.

MR. ELLISON: Okay.

That is all the questions I have for Mr. Capra.

MRS. BOWERS: Mr. Lewis, do you want the Board to proceed?

MR. LEWIS: Yes, ma'am.

BOARD EXAMINATION

BY DR. COLE:

Q I will try to be reasonably brief, Mr. Capra.

On Page 1-3 of NUREG 0667, in the middle of the second paragraph, you refer to an overall integrated NRC action plan. Could you tell me the current status of that, sir?

A To the best of my knowledge, there have been five versions of the action plan, five drafts from its original inception. I believe that Draft 5 is actually the final version which will be presented to the Commission for approval. It is my understanding that that should have been back or should have been completed this week. I think either -- correction, last week, Monday or Tuesday. I don't happen to have a copy of it with me, but it is the final version that is going before the Commission for approval.

Q All right, sir. Is this also referred to as the TMI 2 action plan?

A Yes.

Q On Page 8-2 of NUREG 0667, in the first paragraph that begins on that page -- it is Page 8-2 -- you refer to existing requirements contained in the TMI 2 action plan. What is the current status of the recommendations or comments or requirements contained in the action plan?

Are certain of them now existing requirements? I wonder why you chose those words, sir.

A Maybe that was a little bit of a misnomer. No, they are not requirements yet. However, the vast majority of them will become requirements as soon as the Commission approves the action plan. I do not know what types of revisions they will be based on Commission comment, but that is the reason there have been so many drafts to begin with. It is based on comments from the ACRS, based on comments from the Commission. I am not sure if there has been any public input to it. Also, the original version did not include the recommendations from the Rogovin report. That has now been updated to incorporate those.

Q All right, sir. So they are not requirements yet?

A That is correct.

Q As indicated on 8-2?

A That is correct.

Q All right, sir.

On Page 1-6 of NUREG 0667, referring to long-term solutions, you state that the task force believes that acceptance criteria for plant performance during anticipated transients applicable to all plant designs should be developed. I want to make sure I know what you mean there, sir. Could you give me an example of one acceptance criterion that might be considered here?

A This is the same as Recommendation 19. We are just emphasizing it here. I will give you an example of some performance criteria that we as a task force have proposed.

If you turn to Page 5-27, the first full paragraph starts out, "Although the development of performance criteria must be the product of extensive evaluation and review, the task force offers the following preliminary example that should be considered in order to focus attention on the overall goal to be achieved. This example is not to be considered a specific recommendation of the task force."

Then we go on with an example. A, for instance, heats incapacity shall be esatblished such that the availability is assured for X minutes following loss of all feedwater with no other failures. B, no failure of a control function should lead to the actuation of an engineered safety feature.

Q All right, sir. That is very helpful.

A That is the type of example.

Q Fine. Thank you.

On Page 2-1, on the eighth line from the bottom, the sentence that begins with, "This sensitivity," you state, "This sensitivity is further compounded by the lack of sufficient functional and design interface requirements between the nuclear steam supply system and balance of plant systems."

I am not sure I know what you mean there, sir. Could you elaborate on that?

A For instance, the fact that adequate design interface that we gave an example of in here is the auxiliary feedwater system. As a matter of fact, it is readily apparent during a loss of feedwater that you need initiation and control of auxiliary feedwater in a very rapid fashion. However, it is possible for certain plants to sustain a loss of feedwater and have the steam generator boil dry before you actually get flow from the auxiliary feedwater system into the steam generator.

Another example is that auxiliary feedwater is very important to a B&W plant if you want to assure the adequacy of a secondary feed synch and the fact that the signals used to initiate auxiliary feedwater vary from plant to plant, as shown on Table 5-1, I believe, in the report. That is the type of thing we are talking about.

Q The feedwater system then is considered under the balance of plant.

A Yes. What we are really saying here about the interface is that we feel we should -- that the nuclear steam supply should take a more active role in determining the requirements for auxiliary feedwater and actually take a look at what is being supplied rather than just saying you need auxiliary feedwater. You need it in X amount of time, and you need X amount of gallons per minute flow. You should actually look at the initiating signals and what signals should be utilized.

Q All right, sir. I understand your position.

On Page 2-3, the Item Number 2 at the top of the page, you state -- the report states, "The once-through steam generator design is technically sound; however, it requires a highly interactive and responsive control system," and then in parentheses you have, "i.e., the integrated control system."

Does that mean that -- that those B&W plants which have and use an integrated control system are then satisfactory with respect to that problem? I mean, do they have a highly interactive and responsive control system which you say they need?

A The statement in parentheses there,"i.e., the integrated control system," is just defining what we mean by control system. We are not saying that the integrated control system meets the requirements of highly interactive and responsive.

Q Do all B&W plants have a system similar to the system at Rancho Seco, an integrated control system?

A Yes.

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P7 folll.

Q On Page 2-5, just a general question. The top portion of the page you refer to the possibility of over-filling taking place. What are the consequences of over-filling the steam generator?

A Just overfeed by itself, of course, has the consequences of reducing primary system pressure, and possibly exceeding the cooldown rate limits associated with the tech specs for each plant, but the overfilling concern that we are talking about here is the possibility of feeding the steam generator up to such a height -- as you may recall, the steam piping comes out the side on a once-through steam generator such that you come up and actually fill the steam lines, and there is the possibility of either water hammer taking place, the actual weight of the water taking place or having an effect on the steam piping itself that possibly these supports were not designed to hold that weight.

Also, the possibility of filling the lines going to the turbine driven auxiliary feedwater pump. But the main concern here is failure of the main steam lines.

Q All right. Thank you.

On Page 2-6, Item B, with reference to the power supply logic arrangement, the concern here is the elimination of mid-scale failures, and I do not understand why the power supply logic arrangement might be involved in that, and not just the type of readout meter that you have. With a loss of power, I would think that the kind of meter that you would want would be one that would demonstrate a loss of power by going to zero or going off-scale rather than failing at mid-scale, but I do not understand how the power supply logic arrangement would be involved in that, and not just the readout meter.

A It depends on the --

Q Can you alleviate my confusion here?

A It depends on the electrical input that is actually driving the meter. For instance, if a full-scale deflection one way is X millivolts, and the downscale reading is exactly the opposite, say, plus ten millivolts, is full-scale deflection on the high side. Minus 10 volts is essentially the zero reading or bottom scale, and an absence of power or essentially zero volts would drive the meter to mid-scale, or the meter would fail at mid-scale, so the actual power supply input or the signal input is the thing that says where the meter is going to fail on loss of power.

Q Can't meters be modified so as not to behave that way?

A I suppose it is possible. What we were really concerned about here is, look at B and C together, although we have them separated. What we are concerned about is having these indications unambiguously indicated to the operator. Now, if a mid-scale failure on the meter is close to the normal operating parameter, which in a lot of cases they are, then it may not be easily recognized by the operator that that meter has in fact failed, but you know the electrical input to the meter is the thing that tells the meter on absence of power where it is going to go, whether it is going to go high, whether it is going to go low.

By and large, most of the meters that are associated with the integrated control system do fail on mid-scale. We are not saying necessarily to eliminate that. We say, consider the elimination of it, because it is possible that that could have some negative effects by doing that. If you are not just failing a meter there, but you are failing the control device, the thing that is being controlled by that signal, you may want that particular valve, let's say, to fail mid-position, whether it is fully opened or fully closed.

Q All right, sir. Thank you.

On 2-10, Item 16, sir, does this mean that the committee that worked on NUREG 0667 has some serious reservations about the wisdom of the current criteria for tripping and restarting pumps.

A I do not think that there is a big concern that the criteria are wrong. I think there is a concern on the part of the task force that the NRC has not completed the review of the criteria yet. They are in fact in place and being utilized, but the NRC has not taken a formal position, and we feel the staff should do that.

Q On 5-2, Item 2 on the page, reference is made to high pressure injection pumps and the last part of that item you referred to Davis Besse -- Davis Besse 1 as the only B&W plant without a certain capability. Do you know-- Do you know the rationale behind Davis Besse being set up the way it was set up?

A I have discussed it before with B&W. I do not really remember the rationale. At Davis Besse they have separate high pressure injection pumps from the make-up pumps, whereas the other B&W plants, they utilize the same pump or one of the three HBI pumps. I believe that economics played a point in it. It was not felt at the time that you needed a feed and bleed capability. I do not know if that was even thought of at the time, and the only thing that was necessary was to provide make-up capability at whatever the required make-up flow was, which was not really the capacity that the high pressure injection pumps were, so you could provide separate pumps for that capability, and then for the ECCS considerations you could provide the lower head high pressure injection pumps.

Q On Page 5-37, Item 7 on that page, the second sentence in that item, the report states,"Challenges to the AFW system of operating B&W plants have been frequent because of the unreliability of the mean feedwater systems and their associated control and support systems."

Could you tell me, sir, the basis for the statement that the main feedwater systems are unreliable?

A If you will turn to Page 4-15, which is Table 4.2 under Reactor Trips, it says -- the column is divided into -- there are two columns associated with this table, Pre-TMI 2 and Post-TMI 2. You note that the total number of reactor trips Pre-TMI 2 was 232 of them. Feedwater transients were the cause of 88 of those 232 trips. If my map is right, that is about 40 percent or so. If you look at the Post-TMI 2, there were a total of 38 trips at the time of this writing, and 15 of those were associated with feedwater transients. Again, it is a little bit less, but not much, about 38 percent or so.

Feedwater transients are a significant contributor to trips in B&W plants, as well as other PWR's. The main feedwater system on B&W plants is not that significantly different than other PWR's.

Q Are you saying, sir, that all PWR's have this kind of a problem, or does B&W have more of a problem with the main feedwater system than other kinds of plants?

A I have not gone back and reviewed the entire operating history, but for instance, I went back and looked at feedwater transients post-TMI for all operating plants. I had to do that as part of the response to an interrogatory, and comparing the three PWR vendors in the United States, B&W, CE, and Westinghouse. B&W fell right in the middle with respect to sheer number of feedwater transients. CE had the most feedwater transients per pound.

Now, during this short period of time -- it was about an eight-month or nine-month period of time, B&W was second and Westinghouse had the best record. However, a substantial number of t ps that occurred in all PW's are associated with feedwater transients.

Q All right, sir. Thank you.

Mr. Capra, is there anywhere in this report -- I am referring to Page 5-41, with respect to this question -- where the task force encourages efforts to strengthen the reliability of the main feedwater system. Are there any specifics or suggestions as to how that might be accomplished in here with respect to the main feedwater system?

A No, there are not.

Q Is there any reason -- was that not in the charge of the task force? Is that the reason why it might not have been included or was not included?

A It could have been. I am not saying it was eliminated from the charter of the task force. I think we recognize that an upgrade of the main feedwater systems on all the plants would be a highly desirable thing. However, we get back to the problem of trying to enforce action in that area which the feedwater system is not a safety-related system, and it is hard for us to require licensees to make modifications in non-safety related systems.

Thatis one of the reasons we have identified the auxiliary feedwater system as a safety-related system.

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Q All right, sir. On Page 5-50, the middle of the page, there is a sentence that begins on the fourth line of the second full paragraph on the page, "Should loss of all four reactor coolant pumps occur, the level is controlled at a higher level in the steam generator (i.e., 50 percent on the operating range indication)."

I thought I recalled hearing testimony here that they would normally operate at about the 50 percent level, and then under these conditions if the recirculating -- the reactor coolant pumps go out, they would then move the operating range up to 95 percent. Is that recollection correct, sir?

A Somewhat.

Q Well, straighten me out, will you?

(General laughter.)

A As we have discussed before, there is no set operating level in the once-through steam generator. The ICS does not control feedwater to X percent in the operating range, when you are operating above 15 percent power. It depends upon the amount of superheat you have in steam pressure, and reactor power, but the level does vary. You can read the level during operation, and it is around 50 percent at 100 percent power. It may be a little more. It may be 60 percent. I am not sure. It is not really important.

However, when you experience a reactor trip, the ICS will control -- assuming that the reactor coolant pumps are still operating.

Q I am sorry. The reactor coolant pumps?

A Assume they are still operating following a trip. You are operating along at 100 percent power, and you have a reactor trip, so you have a turbine trip that leads to a reactor trip. The level in the steam generator was somewhere around 50 percent on the operating range. If the reactor coolant pumps are still running, the ICS will tell the feed reg valve, the valves regulating feedwater, to shut until the level in the steam generator comes down to approximately 30 or 36 inches. It varies from plant to plant on the start-up range indication, so even though the feedwater pumps are still operating, you are not actually feeding the steam generator. It is boiling down. You do not need that much feedwater in there.

Now, at some point in time, you have to trip the reactor coolant pumps either because you have reached an SFAS limit and you have to trip them manually or you experience a loss of off-site power. You no longer have forced flow, so the ICS now has a different set point to Control Steam Generator Level 2. It will no longer control it at the 30 inches on the start-up range. It will tell the ICS to maintain level at 50 percent of the operating range, to promote natural circulation. If you get to 50 percent at the operating range and you still do not have natural circulation, you can take manual control and raise it above 50 percent, up to the 95 percent level.

So, what we are talking about here is two different subpoints for the ICS. When the reactors trip, it will either control on the start-up range if the reactor coolant pumps are running or at the 50 percent level with the reactor coolant pumps tripped.

Q All right. Thank you.

Page 7-15, Item 9, the last sentence of Item 9, "It may also provide a later point of no return for saving the core during primary coolant boiloff." I do not understand the use of the term "point of no return" there.

A What they are saying here is, if you experience a problem with the plant such that you have no core cooling for a period of time, say, auxiliary feedwater does not come on, you cannot get main feedwater back, and for some reason high pressure injection fails, you are going to experience boiloff of the primary coolant until you get down to a point where you eventually do core damage.

What this means is that even if there is a delayed initiation of feed and bleed, there is no set time, as long as you get it initiated before you do core damage that will essentially mitigate -- mitigate the event. That didn't

Q No, sir.

(General laughter.)

Q I still do not understand the "later point of no return."

A Okay.

Q It might be the last resort until such time as you can get some other system on, but"point of no return" means to me something possibly other than the way you used it here.

A The "point of no return," meaning the onset of core damage --

Q Well --

A What they are saying here is that B&W plants have this capability to feed and bleed. Having that capability is an added benefit. If the only thing you had to rely on was auxiliary feedwater and you could not get auxiliary feed-water on, you would eventually have core damage, providing pressure stayed up above the pressure of the shutoff head of the high pressure injection pumps for plants that do not have high head injection, whereas this capability here that the B&W plants have may actually give you an extended period of time to mitigate.

Q All right, sir. I want to make sure I understand.

This sentence reads, "It may also provide a later point of no return for saving the core during primary coolant boiloff." Could that sentence be changed to read, "It may provide an additional safety system or method for saving the core during primary coolant boiloff?" Does that change any of the meaning you wanted to impart in that sentence?

A I think that there is something a little more subtle in that particular footnote, but I do not know what it is. What you are saying is certainly correct, but if that -- I cannot say that the two are equal. I think there is something that the probabilistic analysis staff means it is a little bit different than that. I have heard it before but I cannot recall it.

Q All right. Page 7-24, Item 1A under Single Failure Criterion, the first part of that section says, "We believe almost all B&W plants have an auxiliary feedwater system already meeting the single failure criterion for its mechanical aspects."

What about Rancho Seco? Do you know, sir?

A The mechanical system for Rancho Seco's auxiliary feedwater system is safety grade and thus does meet the single failure criterion.

Q Thank you. On 7-27, under Item H, other requirements, the third sentence in that item, the report states, "With two train AFWS designs, even ones of comparatively high reliability, loss of all feedwater is a rare but distinctly credible event."

The next sentence states, "We judge that a return interval of once in a thousand reactor years is about the best one might confidently expect for a loss of all feedwater in PWR's having two train auxiliary feedwater system designs. When you use the word "rare" are you referring to something with a probability of 10^{-3}?

A Yes, sir.

Q So that means with the number of reactors that we have, we can expect that kind of an event with X number of reactors once every how many years -- some frequency that seems to me to be fairly low, or fairly great frequency.

The thing that bothers me about this, sir, is, have we experienced any event where all feedwater was lost?

A I do not know.

MR. SHON: Excuse me. Now, I am a little confused. At Three Mile Island 2, essentially that is what happened, isn't it?

THE WITNESS: Yes.

MR. SHON: So we have experienced one in 500 years, and that looks like about 10^{-3}.

THE WITNESS: I was not really counting Three Mile Island, but I suppose I should have. This is talking about not necessarily a sustained loss of all feedwater, but a situation similar to Three Mile Island where auxiliary feedwater fails to initiate a loss of main feedwater. It does not mean chat it will develop and do core damage.

BY MR. COLE: (Resuming)

Q I was looking at that as a loss of feedwater, not a temporary loss of feedwater. I was looking at it as a total loss of feedwater. That is not what you considered when you said loss of feedwater -- loss of all feedwater here, sir?

A That is correct.

Q All right. On Table 8.1 on Page 8.1-2 and following, you have listed Priority 1 and Priority 2 items, also categorized by action group. Sir, have you looked at the 22 items listed in Table 8.1, your 22 recommendations or requirements with respect to Rancho Seco?

A The task force did not. I have taken a quick look at it to see, you know, which ones I know that they have begun some work on, and which ones are not applicable to them.

Q That is what I was going to ask you about, sir.

A Do you want me to run through that quickly?

Q Would you do that?

A Now, when I go through these, if I say that work has begun or whatever, it does not necessarily mean that the exact requirements, if they turn into requirements, are being complied with at the present time, but that there is work in this area, is what I am talking about, specifically --

MR. BAXTER: Excuse me, Mrs. Bowers. Mr. Capra was asked this question when he testified last time, starting at Page 1241. He went through each of the 22 recommendations and identified those that had been committed to Rancho Seco and started. Has there been any change, to your knowledge, since April 8.

THE WITNESS: No, but I think I can do it much quicker. I am not sure, without going back and reviewing the transcript. Recommendations 1 and 2 should be taken as a whole, I feel. They both deal with auxiliary feedwater system upgrade, with respect to Number 1 making it an engineered safety feature system, and Number 2, the automatic initiation and control.

The automatic initiation and control has been committed to being upgraded to safety grade by the licensee. As I said, there may be a problem with their interpretation of the date and hours. Essentially Recommendation Number 1, the system upgrade, that has been identified by the staff. Most of the requirements have been identified by the staff in their February 26th letter to the licensee with respect to all the requirements necessary to upgrade the auxiliary feedwater system.

BY DR. COLE: (Resuming)

Q That is now a requirement?

A Item Number 1 now as a result of the licensee's submission of the reliability analysis, the auxiliary feedwater system reliability analysis back in December, the staff reviewed that document, and generated certain requirements which were transmitted to the licensee on February 26th.

A lot of those requirements, when implemented, will go to meet most of Recommendation Number 1 and Number 2.

Q All right, sir.

A Recommendation Number 3 is not applicable to Rancho Seco. Recommendation Number 4 may or may not be applicable to Rancho Seco. They have a steam line failure system which -- at least it is my understanding right now -- does not interact with auxiliary feedwater, and that is one of the things we were concerned about.

Item 5, improvements in plant control systems for NNI and ICS, we have addressed that. They have taken actions based on the lightbulb incident.

Item 6, I do not believe any action has been taken on.

Item 7, I do not know the status of. It is applicable, but I do not know that they have taken any action on that.

Item 8, the high radiation signal for vent and purge isolation, that is applicable to them. However, they may fall in the category where it is acceptable for their commitment not to purge during operation. That may be an acceptable alternative, and that commitment, I believe, is in effect now,not with respect to this recommendation, but in a separate review.

Items 9 and 10 and 19, as I mentioned earlier, B&W and the licensees have taken some action on these to see what they can do on their own before it becomes a requirement.

Item 11, modifications to eliminate immediate manual actions, I do not believe any work has been done on that yet, but it is applicable.

Item 12, the qualified I&C technician on duty, that is applicable. I don't believe that is in place right now.

Item 13, operator training on the Crystal River event, they have conducted operator training on Crystal River 3. I am not sure of the status of that.

Item 14, emergency procedures for loss of NNI/ICS, Rancho Seco had developed those prior to this task force.

Recommendations based on the lightbulb incident. Item 15, mandatory simulator training for requalification. We heard from Mr. Rodriguez. In practice, even though it is not a requirement, their requalification program, or an NRC requirement, for all practical purposes, they do do that now.

Recommendations 16 through 22 -- okay, Item 16 is a staff action; Item 17 is a staff action; Item 18 is a staff action; Item 19, mentioned earlier, that is really a joint NRC -- joint NRC and licensee requirement, if it becomes implemented.

Item 20 is a joint action, the continued evaluation of the need to trip reactor coolant pumps during small break focus. That needs to be done really by all PWR's and vendors and NRC staff combined.

Item 21, re-evaluation of the location of AFW injection into the OTSG, as I mentioned Saturday, GPU says they have done an analysis which we asked them to submit if it appears to be a generic analysis. That may be acceptable for the staff to review with no further licensee action until we make an evaluation on it.

Recommendation 22 is an NRC staff action item.

Q I just have one more question, Mr. Capra.

Before you were talking about B&W and the feed and bleed capability, and that you could not really count it as a system because it really has not been adequately tested. I do not know whether you used those exact words or not, but what in your opinion needs to be done with the equipment associated with the feed and bleed concept in order to qualify as an additional safety feature?

A This is just my own personal opinion from what I have heard around the NRC staff with respect to those, but certainly there was no problem with the feed portion of feed and bleed. That is already a safety system. What we are talking about now is the discharge of the water from the reactor coolant system. If there happens to be a break of sufficient size to handle all the water, then it is no problem, but we are talking about a case where there is no break, and the only exit for the water is either through the PORV and or the safety valves. Either of these valves have been qualified for either two-phase flow or solid water flow. They have been qualified essentially for steam.

There is an EPRI program under way right now to do some testing on these valves to get at least some performance characteristics. I doubt from what I have heard that PORV's as they are presently configured are actually going to pass the test for solid water and two-phase flow for a sustained period of time. It is possible the safety valves will do that, and if the safety valves can provdie sufficient relief capacity then that may take care of the problem.

But essentially you need qualified relief paths, and right now there are not qualified relief paths, but it has worked in all situations where it has been used.

DR. COLE: Thank you. I have no further questions.

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

(Whereupon, at 12:05 p.m., the hearing was recessed for lunch, to reconvene at 1:05 p.m. of the same day.)

Suzy foll.

A F T E R N O O N   S E S S I O N

MRS. BOWERS: On the record. Mr. Capra?

Whereupon,

ROBERT A. CAPRA

the witness on the stand at the time of recess, resumed the stand and having been previously duly sworn, was examined and testified further as follows:

MR. LEWIS: Mr. Shon, let me just say preliminarily that the staff's evaluation of the Oak Ridge analysis of the varied modes and effects analysis is apparently being telecopied to us today, so we should have something available tomorrow.

MR. SHON: That's good.

FURTHER BOARD EXAMINATION

BY MR. SHON:

Q Mr. Capra, what I have, I think, is really fundamentally only one question but I may have to ask a few preliminary and clarifying questions. I don't think it will take too long.

Just as a sort of an aside, have you ever heard of the butter-keeper paradox or the butter-keeper syndrome?

A No, I haven't.

Q You see, I think it applies to some extent to this case. A long time ago, mankind was in the cold cruel world, you know, and built a big box called a house and he heated it up so he could be warm inside that. But then he discovered his food started going bad, so he built a refrigerator, a cold box inside the warm box inside the cold, cruel world. Then he discovered that the butter got too hard, you see, so he invented the butter-keeper, which is a warm box inside the cold box inside the warm box inside the cold, cruel world.

Now, a lot of the things we've been talking about with the B&W system, the integrated control system and the pilot operated relief valve were put there originally because the designers thought it was a pretty sensitive system and they were meant to help it override transients, isn't this true? I mean, there was relief a little, a reactor burp, and then things would settle back down.

A Yes, sir, that's correct.

Q And then you discovered that the thing we'd done to help it override transients could sometimes aggravate a transient or even cause one, and that's the situation we're in now. Isn't that right?

A Yes, sir.

Q Now you have 22 more things we ought to do, on top of the things we've already done to stop the things that we've already done to stop the things that we've done from doing bad things, right?

(General laughter.)

A I understand what you're saying, but I think some of them are more fundamental things that probably should have been done in the first place and they're fundamental to design. They aren't, let's say, bandaid approaches. For instance, upgrading the auxiliary feedwater system to an engineered safety feature. In my estimation and in the estimation of the Task Force, it probably should have been done from the start. The diversity and redundancy of the power supplies and all, that should have been done from the start, I would think.

So there's a lot of them that are going back to the basic design of the plant. Performance criteria -- that probably should have been done from the start. It's not that we're adding systems upon systems or boxes in boxes.

Q Well, the difficulty that strikes me is that the fundamental trouble seems, in part at least, the thing that got the chain started, is the sensitivity of the B&W system. And yet, when I look at, for example, page 7-18 of the document 0667, I notice that the one of these 22 things that seems to be a direct approach to this, which is number 10, is the only one for which the entire table here that is supposed to tell whether you do good things or bad things by doing that, for which the entire table is nothing but a series of question marks, so it's the total unknown of the bunch, isn't it?

A Yes, sir. But you understand why. You can't make an evaluation of what effect it's going to have unless you know what the thing is that you're fixing.

Q So it seems as if the fundamental approach, the approach that says, well let's try to make the system itself somehow inherently less sensitive, is at least up to here the one thing that nobody has any really good suggestions on. Is that right?

A I didn't say we didn't have some suggestions. As a matter of fact, I think I mentioned those suggestions in response to Dr. Cole's question -- things that the staff has kicked around. But the impact on those things is hard to quantify until analysis is done and the sensitivity studies are done to see what the best thing or groups of things is to do. So I would not expect that the Probabilistic Analysis staff can take a requirement that says, perform an analysis to see if you can reduce the sensitivity, and actually assign any risk reduction potential associated with that, since there may be nothing that could be done, let's say. Or there could be a great number of things which would make their estimate run the whole gamut from negligible to high.

Q By the same token, I should imagine that no one has any real idea of how long it might take to do something that would represent, in a sense, a fundamental change of this sort.

A Not unless we identify what those changes are. Some could be done rather quickly. For instance, if it involved a set point change in the secondary side safety valves or the turbine bypass valve setpoint for operating with less super heat or whatever, those changes themselves would not take long to accomplish. However, the supporting analysis to insure that while you may be improving one thing you're degrading something else; that may take a little bit longer. But I would think that those types of fixes are a little easier than, for instance, upgrading an auxiliary feedwater system to safety grade, or adding on a third train. This would take an extremely long time.

MR. SHON: I see. Thank you, I have no further questions.

THE WITNESS: Mrs. Bowers, I'd like to see if I could clear up one thing that I had mentioned earlier in response to CEC question concerning footnote 3 on Table 7.1. Hopefully, I'm not opening Pandora's Box here, but I did call back to the Probabilistic Analysis staff and got one of the three gentlemen who wrote this section who was in. He didn't write that particular footnote, but tried to interpret it for me and I think it was essentially what Mr. Shon had said. Footnote 3 means that while they do see no big impact or direct impact of that particular -- here we're talking about the frequency of under-cooling transients. While they feel that there's no direct effect of that particular characteristic leading to a severe accident, it is possible that due to uninvestigated common mode failure, such as the water coming out of the reactor coolant system into the quench tank and a rupture of the quench tank, ruptured disk and possibly an accumulated amount of water possibly flooding out engineered safety features and things like that, leading to problems with containment overpressure and possibly rupture the containment.

So they say it's not a very significant footnote; it's almost an out to say that we haven't investigated all possibilities yet. And I asked if it was possible that that footnote could almost equally apply to all of them, and the answer was yes.

BY MR. SHON (Resuming):

Q And the fact that the effect might, as they say, rival dominant sequences in probability didn't mean that by ignoring it you're ignoring a substantial effect or anything like that

A No. I meant that due to factors that they may not have considered or may not have investigated, there may be some hidden common mode failure that could bring this particular scenario up to a more significance than it appears to have here.

MR. SHON: Thank you.

MRS. BOWERS: Mr. Lewis?

MR. LEWIS: I have no questions on Redirect.

MRS. BOWERS: Mr. Baxter?

MR. BAXTER: I have no further questions.

MRS. BOWERS: Mr. Ellison?

MR. ELLISON: I have just two.

CROSS ON BOARD EXAMINATION

BY MR. ELLISON:

Q The first one is with respect to Table 7.1. Would it be fair to say that the difference between the severe accidents and the accidents would be the integrit f the containment?

A Yes.

Q And the second question I have is just to clarify a response that you gave to Dr. Cole. He asked you with respect to the 22 recommendations where we stood, where SMUD stood, with respect to each of them, and I recall your answer with respect to recommendation number 5 which are the improvements to the ICS and the NNI as being that SMUD had done a lot of work in this area and had substantially complied with it.

But I also recall asking you whether the recommendations in this report that are the number 5 recommendations were the things that were identified in the January 21st letter as SMUD's improvements to the NNI/ICS system. And I recall that answer as being basically no, that there were some differences. So could clarify for me whether or not -- just where SMUD stands with respect to the improvements of the ICS and NNI that are recommended here?

A On Table 8-1, which is the implementation section, that was the table that I was reading from before, the short identification of that particular item is "Improvements in Plant Control Systems, NNI/ICS." Over in the righthand column you'll see that there are similar requirements which should be considered before implementing this recommendation, some of which overlap.

The first document is BAW-1564, which is the ICS reliability analysis. The January 21st letter which we were talking about was the response by SMUD of actions that they were taking to comply with the recommendations identified in BAW-1564.

So since our recommendation 5, meaning the Task Force recommendation 5, is closely coupled with that item, BAW-1564, the compliance, full compliance, if that were the case, with BAW-1564 would be in partial fulfillment of the Task Force recommendation 5.

The same with the other two items that are identified here, the NSAC-3/NPO-1 report which is the Crystal River evaluation. There were recommendations in that particular document which may have already been completed by SMUD. The same with I&E Bulletin 79-27; they have responded to Bulletin

79-27.

If full compliance with all three of these particular items has already been accomplished, that would go a long way into completing the items identified in recommendation 5, but not fully. There may be some that have not been identified. So they're all interrelated.

MR. ELLISON: That's all.

MRS. BOWERS: We would like to hear from the parties and we have questions concerning this document. Now, it's Staff Exhibit No. 4 and NUREG-0667, and we have been told that this is the final submittal of the Task Force, but it has not yet received the blessing of the powers to be that it would actually be issued in exactly this form with this language.

THE WITNESS: This report has gone to the printer in this version, so it's going to come out in the blue cover NUREG version, just like you see it with the exception of the changes that I made Saturday to the document when we first started talking about this. But when it comes out as an official NUREG document, that still does not mean that its recommendations need to be implemented, or are going to be implemented. That has to be directed by the Director of Nuclear Reactor Regulation.

MRS. BOWERS: I thought you told us that you were uncertain as to what changes Mr. Denton might make.

THE WITNESS: No. He will not make any changes to this document. What he does with the recommendations contained in this document is what I'm not sure of at this point.

MR. SHON: In other words, he might opt to follow all of them or none of them or some of them or some modified version of some of them, is this right?

THE WITNESS: That's right.

MR. SHON: And ultimately, the Commission would have to give its approval, also, or just Mr. Denton?

THE WITNESS: I'm not sure of the politics involved in that; whether Mr. Denton can do that on his own or whether he does need Commission approval. I know we briefed the Commission.

MR. SHON: This represents, in effect, the final staff report but not necessarily an official staff position, as far as the recommendations and priorities are concerned.

THE WITNESS: It's a final Task Force report which does not represent a staff position.

MRS. BOWERS: So where do we go from here, Mr. Lewis, if we follow correctly, it's not an official staff position.

MR. LEWIS: Right.

MRS. BOWERS: So what do you expect the Board to do with it?

MR. LEWIS: I don't know. It's a recommendation. It's a set of recommendations from this Task Force.

MRS. BOWERS: But do you expect this Board to decide which recommendations to accept or reject or to modify?

MR. LEWIS: I think that it bears upon the whole question of adequacy of the short-term and long-term actions. Once again, that has to come back and be the focus of this hearing. I don't want to get into the question now of whether or not this Board could or should consider requiring anything of this type. I think that's quite a thorny question.

But I think it's here as a document represented by and prepared by a Task Force which certainly has a lot of information that bears upon your decision as to whether or not the short and long-term actions required by the May 7th order were adequate. That's how I see it. And, of course, in that sense I'm not sure that it has bearing upon that question. I'm not sure that its status as not yet adopted would really matter. It still has a lot of useful information, I guess, as to the determination of the question of adequacy of the May 7 requirements.

MR. BAXTER: We didn't object to the offer of the document because it seemed to me that the text in the document, the discussion of the operational characteristics of the B&W plants was relevant to the testimony that a number of the Category 1 staff witnesses were offering, and indeed, there were at least three members of the Task Force who were

srb 12

here to testify in person. I don't think it is -- well, the record can be briefed on whether it's adequate for the Board to address the specific recommendations as they might apply to Rancho Seco. But I agree with Mr. Lewis that ultimately it bears perhaps on the background of the adequacy of the May 7th order, but that's in the final analysis what the Board is here to decide. I don't think the weight of the evidence is affected greatly by the fact that the Commission or Mr. Denton has not blessed everything that's said here. I doubt that he read any of the other staff testimony that was filed here.

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MRS. BOWERS: Mr. Ellison?

MR. ELLISON: Very briefly, I think this constitutes part of the evidence before the board. Witnesses have been here to testify to this, particularly Mr. Capra, today.

So, we would treat it like any other evidence before the board, that it is something that the board can, and we hope will, consider in carrying out the mandate that it has received from the Commission.

That mandate, I think is quite clear, does empower the board to if it finds that remedial actions are necessary, to order those actions is they are supported in the record.

So, I think we can reserve this matter for the briefs. But I think our position should be made quite clear at this point that, yes, the board believes that based upon the evidence in this proceeding, including this document that some of these recommendations should be implemented at Rancho Seco. That it has the power to do that and that it would be supported by evidence in the record, particularly this document.

MRS. BOWERS: Do you have anything further, Mr. Lewis, on this?

MR. LEWIS: No, I do not.

(Board conferring.)

MRS. BOWERS: We have nothing further on 0667.

bfm2

MR. LEWIS: May Mr. Capra be excused?

MRS. BOWERS: Any objection?

MR. ELLISON: No objection.

MRS. BOWERS: Is Mr. Capra going to be around the rest of the week?

MR. LEWIS: I understand that he is.

MRS. BOWERS: Then he is excused, then.

(The witness was excused.)

MRS. BOWERS: Is the next witness Mr. Wilson?

MR. BLACK: Staff at this time would like to call Bruce A. Wilson to the stand.

Whereupon,

BRUCE A. WILSON

was called as a witness by Staff counsel and, having been duly sworn, was examined and testified as follows:

DIRECT EXAMINATION

BY MR. BLACK:

Q Mr. Wilson, could you state and position with the NRC for the record, please?

A My name is Bruce Wilson. I am an examiner with the Operator Licensing Branch of the NRC. I have been with the NRC since October of 1973. During the past year or so I have been involved with the Bulletins and Orders Task Force, and just recently with the B & W Sensitivity Study.

Q For this proceeding, have you prepared three

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separate pieces of testimony?

A Yes, I have.

Q Could you identify those please?

(Pause.)

A Yes. The three pieces are NRC Staff Testimony of Bruce A. Wilson on Control Room Design, on Instrumentation and Diagnosis and Control of Off-normal Conditions, and on Operator Training and Competence.

Q Do you have any corrections or additions to the testimony on Instrumentation for Diagnosis and Control of Off-normal Conditions?

A No, I do not.

Q Do you have any additions or corrections to your testimony on Operator Training and Competence?

A I wish to make two clarifications to this testimony. On page 17, there is a question:" Does the licensee conduct interviews with its operating personnel to discuss their performance on tests administered?"

This is in regard to the requalification program. The answer I've given is "True."

However, the requirement of the requalification program of Rancho Seco is that if the operator gets a less than satisfactory grade, then his written examination will be discussed with him. I have seen the practice at Rancho Seco during one examination they administered.

bfm4

They routinely discuss with the licensees their performance on the examination, although each one is not required. Also, on the submittal -- on the enclosure part to small break phenomena description of plant behavior, this was a November submittal that we received from B & W. It is such, it includes the reactor pump trip criteria.

This was not the description we used when auditing the Rancho Seco operators back in June of last year. BAsically, it was the same description, however, it was up through figure six.

Anything after that was in addition to this & W submittal that we did not use.

MR. BLACK: Mrs. Bowers, I might point out and indicate to the board and parties that this enclosure entitled "Part II, Small Break Phenomena - Description of Plant Behavior," was attached to the testimony for informational purposes only.

The copies that we have given the reporter do not include this attachment; however, Mr. Wilson certainly can be cross examined on that enclosure.

We did not intend it to be a part of the staff pre-trial testimony.

MRS. BOWERS: I want to make sure that I am looking at the right portion of this. Would you identify it again?

MR. BLACK: Yes. It is entitled "Part II, Small

Break Phenomena - Description of Plant Behavior." It follows Mr. Wilson's professional qualifications statement in his testimony on operator training and competence.

There is a May 14, 1979 letter also that is in this attachment that is not intended to be a part of our pretrial testimony. That follows in this attachment.

A letter from Rancho Seco Nuclear Generating Station to Mr. Harold R. Denton, dated May 14, 1979.

MRS. BOWERS: That did not reproduce very well.

MR. BLACK: No, it did not. As I indicated, we did not intend it as part of our pretrial testimony, but only attached for informational purposes.

MRS. BOWERS: What if you can't read it?

DR. COLE: It's less informative then, right?

MR. BLACK: Right.

MRS. BOWERS: Look at page 8, the top of page 8 and the middle of page 8. You copy may be fine, Mr. Black.

MR. BLACK: My copy is a little light as well.

MR.SHON: The material that follows that letter, which is the requried training prior to restart or some wuch thing, has reproduced even less legibly.

MR. BLACK: Yes, I see that.

MRS. BOWERS: That is what I'm looking at. When I said page 8, it must be a part of that.

MR. BLACK: If anybody has a problem with the bad

bfm6

copy on this, I think that we can probably, somewhere around the room, find a good copy.

Hopefully, we can -- if we squint a little bit, maybe we can see what it says and do our examination accordingly.

MRS. BOWERS: The enclosure to the May 14th letter, if you look on page 8, the top of mine, I know something was there.

MR. BLACK: Well, if we get to that section and people cannot read it, my copy is legible enough so I can read it. So, we can fill in the missing blanks if need be.

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BY MR. BLACK: (Resuming)

Q Now, Mr. Wilson, turning to your testimony on control room design, do you have any additions or corrections to that testimony?

A No, I do not.

Q Now, as corrected, does all of this testimony -- Is alloof this testimony true and correct to the best of your knowledge?

A Yes, it is.

Q And do you adopt it as your testimony in this proceeding?

A Yes.

MR. BLACK: Mrs. Bowers, at this time we would like three pieces of testimony from Mr. Wilson, the first one entitled NRC Staff Testimony of Bruce A. Wilson on Instrumentation for Diagnosis and Control of Off-Normal Conditions, the second one entitled NRC Staff Testimony of Bruce A. Wilson on Operator Training and Competence, and the third testimony entitled NRC Staff Testimony of Bruce A. Wilson on Control Room Design, be incorporated into the record as if read and constitute evidence on behalf of the Regulatory Staff.

MRS. BOWERS: Mr. Baxter?

MR. BAXTER: No objection.

MRS. BOWERS: Mr. Ellison?

MR. ELLISON: No objection.

MRS. BOWERS: The documents you have just identified will be physically inserted in the transcript as if read and admitted as evidence.

(The material referred to follows:)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of SACRAMENTO MUNICIPAL UTILITY DISTRICT (Rancho Seco Nuclear Generating Station)	)))))))	Docket No. 50-312 (SP)

NRC STAFF TESTIMONY OF BRUCE A. WILSON ON OPERATOR TRAINING AND COMPETENCE

(Board - CEC Question 1-7, CEC Issue 3-1, CEC Issue 3-2, CEC Issue 3-3, Board Question 32, and FOE Contention III(e))

Q. Please state your name and your position with the NRC.

A. My name is Bruce A. Wilson. I am an employee of the U.S. Nuclear Regulatory Commission assigned to the Operator Licensing Branch. From May 1979 until December 1979 I was with the Systems Group of the Bulletins and Orders Task Force.

Q. Have you prepared a statement of professional qualifications?

A. Yes. A copy of this statement is attached to this testimony.

Q. Please state the nature of the responsibilities that you have had with respect to the Rancho Seco Nuclear Generating Station.

A. I was responsible for reviewing part of SMUD's responses to the Commission Order of May 7, 1979. Specifically, I reviewed their procedures to ensure that their revised procedures were in accordance with the requirements of the Order and complied with the Small Break Loss-of-Coolant Accident Guidelines that were developed by B&W. I also conducted an audit of some of Rancho Seco's operators and senior operators to evaluate the training they had received concerning the TMI-2 accident and the resulting impact at Rancho Seco.

Q. What issues are addressed in this portion of your testimony?

A. I am addressing Board - CEC Question 1-7 and CEC Issue 3-1, which state:

Board Question CEC-1-7

> Do the operator training actions responding to Subparagraph (d) of Subparagraphs a-e for Rancho Seco fail to give sufficient attention to providing appropriate analytical bases for operator actions?

CEC Issue 3-1.

> Whether personnel adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of its operation?

Q. Prior to the TMI-2 accident of March 28, 1979 what type of training did Rancho Seco licensed operators receive to assure their understanding of the mechanics of the facility, basic reactor physics, and other fundamental aspects of its operation?

A. The procedures and criteria for issuing licenses to operators and senior operators are set forth by Commission regulations; 10 C.F.R. Part 55. NUREG-0094, "NRC Operator Licensing Guide," is a guide that expands and explains the regulations for obtaining a license. The specifics of the training program established by the Licensee to prepare candidates are contained in the Final Safety Analysis Report (FSAR); Section 12.3. This program was for the initial plant staff or "cold" license applicants. After the plant achieved criticality, the initial "hot" license applicants received the same training, while replacement operators recieved the training that is specified in Rancho Seco Topical Report T1-76, "Operator Training Program for Hot License Candidates." In order to maintain a license all personnel must participate in the requalification program that is outlined in Rancho Seco procedure AP-25, "Licensed NRC Operator Retraining." The cold, hot, and requalification training programs were reviewed and approved by the NRC.

The training of the Rancho Seco licensed personnel began in 1966 and continued through the licensing of the initial group of operators in 1974. More than one-half of the presently licensed personnel received all or most of the following training; several months observation at an operating nuclear plant, a twenty week course in basic reactor physics and engineering, a two month course in PWR technology taught by B&W in Lynchburg, Va., and a six week simulator course also taught by B&W. In addition, these personnel participated in the startup activities of the unit which included testing components and systems and writing routine and emergency procedures.

The replacement operators participated in the hot license training program, which contains all the essentials of the cold program (described above) with several exceptions. Since the plant was operational, they were able to gain a great deal more practical training and therefore the observation training at another plant was deleted and the simulator course was shoretened.

Since December 1973 the Commission has required SMUD (and all other utilities) to have in effect a Requalification Program in which each licensed person must successfully participate in order to obtain a renewal of his license. The key aspects of the Rancho Seco program are the following: an annual written examination of comparable scope to the NRC test, an oral exam administered by facility management, a lecture series, assigned inidividual study, and a one week simulator course. Although attendance at the simulator course is not required by the Requalification Program, it has been SMUD's practice to send nearly all of their personnel every year. The few exceptions have been members of the management staff whose duties sometimes conflict with the simulator training. The Requalification Program is regularly audited by the NRC's Office of Inspection and Enforcement (I&E) and Operator Licensing Branch (OLB). In the future, the requalification exams will be administered by OLB.

Q. What additional training has been provided to Rancho Seco licensed operators pursuant to Subparagraph (d) of the short-term actions required by the Commission's May 7, 1979 Order?

A. To ensure that post-TMI information was adequately understood by Rancho Seco licensed operators, the following training and evaluations were performed:

1. Each licensee has completed the TMI-2 sequence training on the simulator.

2. Each licensee has successfully passed a SMUD administered TMI related written examination, in which 90% was the passing grade.

3. The above exams were audited for content and grading by the NRC.

4. SMUD conducted special training sessions on the concepts and use of the small break LOCA procedure.

5. Seven of the fourteen licensed personnel assigned to shift duty were audited by NRC.

6. Several deficiencies revealed by the audit resulted in SMUD contracting with General Physics Corp. for additional training.

7. An additional audit was conducted by General Physics (not by the individual who had administered the training).

8. A followup audit of 8 operators was conducted by an NRC inspector, with no deficiencies uncovered.

Q. What steps has the NRC taken to determine the Rancho Seco operators' level of understanding of the training.

A. Initial interveiws of Rancho Seco licensed personnel were conducted on June 1, 1979 (3 licensed personnel) and on June 2, 1979 (4 licensed personnel). These interviews were conducted by myself and Philip Johnson, an inspector from I&E Region V.

Q. Did your interviews explore the operators' understanding of the analytical bases of actions which they may be required to take?

A. Yes. The subjects covered were: TMI-2 Sequence of events, small break LOCA phenomenon, and the bases for changes to the licensee's LOCA procedures and other design and procedure changes made at Rancho Seco as a result of the TMI-2 accident. As a reference to discuss the analytical bases for the actions required in the small break procedure, Mr. Johnson and I used B&W's "Part II; Small Break Phenomenon - Description of Plant Behavior," a copy of which is attached hereto. In particular, we used Figures 1 through 5 of the above document to determine if the licensed personnel were aware of the behavior of the plant as a function of break size and equipment availability.

Q. What were the test results, particularly on those portions related to operator's analytical understanding.

A. We found that the operators could satisfactorily explain the analytical basis for the small break phenomenon. We found, however, that there were some deficiencies in [illegible] knowledge of thermodynamics, natural circulation, and the TMI-2 sequence. These deficiencies could partly be attributed to the fact that some of the operators we interviewed had not yet attended the TMI-2 training session at the simulator. In view of these deficiencies, the Licensee contracted with General Physics Corp. of Columbia, Md. to conduct additional training in these areas. This training was audited separately by another employee of General Physics and re-audited by Mr. Johnson, who found no deficiencies in the analytical understanding of these phenomena among the eight licensed operators he audited.

Q. On the basis of the tests that the NRC has conducted, do you believe that Rancho Seco licensed operators adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of its operation?

A. Yes. I conclude that Rancho Seco operators adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of its operation.

Q. On the basis of the tests the NRC has conducted, do you believe the Rancho Seco licensed operators adequately understand the analytical bases of the actions they may be required to take pursuant to Subparagraph (d) of the Commission's short-term required actions?

A. Yes.

Q. What issue are you addressing in this portion of your testimony?

A. I am addressing CEC Issue 3-2 which states:

CEC Issue 3-2

> Whether personnel are properly apprised of new information pertinent to the facility's safe operation and ability to respond to transients, particularly information on operating experience of other reactors?

Q. Does the licensee, SMUD, have a program for apprising its personnel of new information pertinent to the facility's safe operation and ability to respond to transients, particularly information on operating experience of other reactors?

A. Yes. The licensee has stated that through the Requalification lecture series significant operating events at Rancho Seco and other facilities may be discussed. Additionally, "Standing Orders," which shift supervisors are directed to discuss with their shift crews, may contain such information. Finally, when the licensed personnel participate in the annual simulator course at B&W in Lynchburg, Va. they are often exposed to events that have occurred on other B&W plants. <u>See</u> "Licensee's Answers (Set No. 2) To the California Energy Commission's First Set of Interrogatories Dated November 15, 1979," Answer to Interrogatory 22 (December 4, 1979).

Q. Does the NRC have a program for disseminating to reactor licensees, permittees, and applicants operational information from other licensed reactors?

A. Yes. The NRC's Office of Management and Program Analysis (OMPA) has several means for dissiminating operational information. The first is a Licensee Event Report (LER) monthly listing. This is a computerized listing of LER's at each operating plant. Each LER is catorgorized as to cause (mechanical failure, human error, etc.) and there is a brief description of the event.

Secondly, OMPA publishes a document called "Power Reactor Events" in which signficant events which could have generic implications are described. Upon a licensee's request, it can receive copies of these documents. Special printouts of LER's may also be requested by the individual licensees.

OMPA also distributes the Gray Book, "Operating Units Status Report," which is sent to all licensees that have submitted input for it.

Q. Has the NRC undertaken any efforts to improve the dissemination of operational information?

A. The Commission has established an agency-wide Operational Data Analysis and Evaluation Office to provide coordination and an overview of all operational data analysis - related activities performed within the NRC. The individual program offices have also been directed to establish operational data analysis capability.

Q. Has the nuclear industry undertaken a program for the review of plant event reports and data?

A. Yes. The Electric Power Research Institute (EPRI) has founded a Nuclear Safety Analysis Center to systematically review available plant event reports and data. Also, the industry has established the Institute for Nuclear Power Operations (INPO). One of the functions of INPO is to review and analyze nuclear power plant operating experience and feed this information back to the utilities. The utilities can then incorporate this information into the training programs.

Q. Is the NRC considering further requirements for imposition on licensees regarding dissemination of operating experience to their personnel.

A. Yes. The Commission is considering imposition of a requirement that licensees review their administrative procedures to assure that operating experience from within and outside their organizations is continually provided to operators and other operations personnel and is incorporated into training programs. Draft NUREG-0660, Action Plans For Implementing Recommendations of the President's Commission and Other Studies of TMI-2 Accident (12/10/79), Task I.E.2. Operating plant licensees would be required to have completed this task by September 1980.

Q. Based on the above programs, do you believe SMUD's personnel are now being properly apprised of pertinent new information?

A. I believe the Licensee has a program through which its personnel can be apprised of pertinent new information. Additional requirements may be imposed by the NRC on licensees with regard to dissemination of operating experience. The NRC Staff believes that substantial improvement can be made in the process of dissemination of operating experience. However, based on my audits of licensed personnel at Rancho Seco, I conclude that they have an adequate understanding of the implications of the TMI-2 accident. The licensee's program of disseminating information on the TMI-2 accident has, I therefore conclude, been successful in enabling its operators to understand the implications of that accident.

Q. What issue is addressed in this portion of your testimony?

A. I am addressing CEC Issue 3-3, which states

<u>CEC Issue 3-3</u>

> Whether NRC and SMUD adequately ensure that emergency instructions are understood by and are available to plant personnel in a manner that allows quick and effective implementation during an emergency?

Q. Please describe the organization of the Licensee's Emergency Procedures.

A. The Licensee's Emergency Procedures (EP's) are generally divided into six sections: Purpose, Description, Symptoms, Automatic Actions, Immediate

Operator Actions, and Subsequent Operator Actions. During an emergency situation, the licensed operators must diagnose the event by matching the plant parameters with the Symptoms as listed in the EP's. They must then ensure that the Automatic Actions have occurred and take the required Immediate Operator Actions. These three steps must be done by memory. The operator should get out the appropriate procedure, ensure that the above three steps have been accomplished correctly and then follow the instructions listed under Subsequent Actions.

Q. Are the EPs available in a manner that allows quick and effective implementation during an emergency?

A. Yes. The Licensee's emergency procedures are contained in a red book in a desk drawer immediately behind the control console in the control room.

Q. Does the Licensee have procedures to ensure that procedures are kept up-to-date?

A. Yes. Administrative procedures exist that are intended to ensure that these procedures are kept up-to-date. The Requalification Program also covers the latest procedure revisions.

Q. How does the NRC determine whether licensed personnel have an adequate understanding of EPs?

A. Through the examination process, the NRC determines whether EP's are understood by licensed personnel. Applicants are asked on the written examination to write down those portions of selected emergency procedures that must be committed to memory. On the oral examination, the applicants are asked to simulate or "walk through" these procedures and demonstrate to the examiner their familiarity with and understanding of these procedures.

Questions concerning every EP are not asked of each applicant. It is an audit process, as is the remainder of the oral and written examination. Typically, two of the EP's will be on the written examination, three or four will be discussed in the control room during the oral examination and several more during the walk-through in the plant. The examiner will cover different EP's in the oral examination of other applicants. In this way, the examiner can cover all or most of the EP's.

The knowledge and use of emergency procedures is always included as a topic on the exit interview that is conducted between the examiner(s) and the licensee's management. On the basis of the examinations conducted to date at Rancho Seco, the NRC is satisfied that licensed personnel understand the emergency procedures.

Q. Does the Licensee have a program for determining that licensed personnel have a continuing understanding of EPs?

A. Yes. Through the Requalification Program, the licensed personnel must demonstrate continuing understanding of EPs. Section 3.2.1 of the Requalification Program requires the following:

> . . . each licensed Senior Operator or Operator shall participate in an oral examination with the plant superintendent or his designated representative. This examination and evaluation shall contain the following:
>
> 1. A discussion of required actions during abnormal or emergency conditions.
>
> 2. A simulation of abnormal and emergency conditions while in the Control Room showing each action and controlling device to be operated.
>
> 3. Should the performance of the licensed Senior Operator or Operator be deemed unsatisfactory, the Senior Operator or Operator will participate in an accelerated review program tailored to place emphasis where there is clear indication of need.
>
> 4. Upon completion of the accelerated review program, the individual shall be subject to re-examination.

SMUD has made this oral examination an annual requirement. This exceeds the requirements of Appendix A, 10 C.F.R. 55.

Q. How were Rancho Seco emergency procedures changed as a result of TMI-2 and the May 7, 1979 Commission Order.

A. The Commission Order required all B&W licensees to develop and implement operating instructions to define operator actions for potential small break loss-of-coolant accidents (SBLOCA). B&W then developed guidelines to be used in the rewrite of the LOCA emergency procedures. With the use of these guidelines, the Licensee rewrote EP D.5 "Loss of Reactor Coolant/ Reactor Coolant Pressure." The NRC staff reviewed this revised procedure to ensure that it conformed to the guidelines. We also "walked-through" the procedure in the Rancho Seco control room to ensure that the steps were in a logical order and that the instruments and controls were readily available for the operators to perform the required tasks. On the basis of this review we were satisfied that the revised procedure met the requirements of the Commission Order.

Q. Were any other emergency procedures changed?

A. Yes. Nearly all of the emergency procedures have had some revisions in the last few months. Most notably, EP D.14 "Loss of Steam Generator Feed" was revised to include actions to be taken in the event all feedwater was lost for an extended period of time. This procedure, and several others, incorporate the 50^{O}F subcooling criteria. Emergency Procedure D.1, "Load Rejection," D.2, "Turbine Trip," and D.3, "Reactor Trip" were revised to include the new turbine trip - reactor trip circuitry. Finally, all of the EP's were revised to include a reminder to the operators to check alternate instrument channels of key parameters.

On the basis of our review, we believe the Licensee has made significant improvements to the emergency procedures.

Q. What contentions does this portion of your testimony address?

A. I am addressing FOE Contention III(e), which states:

FOE Contention III(e)

> The NRC orders in issue do not reasonably assure adequate safety because no procedures exist or have been taken for the determination of the adequacy of operator competence.

I am also addressing Board Question 32 insofar as it relates to the competence of licensed personnel at Rancho Seco. Board Question 32 states:

Board Question 32

> Rancho Seco, being a Babcock and Wilcox designed reactor, is operated by personnel and management whose competence has not been adequately tested and evaluated, namely testing has not been conducted as to whether such employees can act responsibly and appropriately to make judgment decisions during a loss of feedwater transient, personnel interviews have not been conducted to properly evaluate the test results with such employees and some employees have never been tested because of grandfathering, and therefore is unsafe and endangers the health and safety of Petitioners, constituents of Petitioners and the public.

Q. Does the Licensee, SMUD, have a program and procedures for testing the competence of its operating personnel?

A. The response to CEC 1-7 and CEC 3-1 contains information regarding the initial training, retraining and evaluation of Rancho Seco licensed personnel. Included was an outline of the Licensee's approved Requalification Program in which the operators receive an annual facility administered written and oral examination. These programs satisfy the present NRC requirements for testing the competency of operating personnel.

Q. In the period since the TMI-2 accident on March 28, 1979 what steps has the Licensee taken to test and evaluate the competence of its operating personnel to act responsibly and appropriately to make judgment decisions during a loss of feedwater transient?

A. The response to CEC 1-7 and CEC 3-1 listed the additional training and evaluations conducted since March 28, 1979. This training included diagnosing and responding to a loss of main feedwater transient. In addition, emergency procedure D.14 "Loss of Steam Generator Feed" was revised to contain guidance for the operators to respond to a complete loss of feedwater. By virtue of their participation in simulator training, all of the licensed operators have received additional training and lectures on loss-of-feedwater transients.

Q. Does the Licensee conduct interviews with its operating personnel to discuss their performance on the tests administered?

A. Yes. The Requalification Program has provisions for discussing test performances with the licensed operating personnel.

Q. Has the Licensee exempted any of its licensed personnel from being tested to determine their ability to make proper judgment decisions during a loss of feedwater transient (i.e., "grandfathered" those personnel)?

A. Literally, the answer to this question is yes, in that the licensee's training coordinator, who wrote and graded the licensee administered examinations concerning TMI-2, was not required to take that examination. Pursuant to the licensee's approved Requalification Program, the person writing the annual written requalification examination plus a maximum of two others, who may assist in its preparation and grading, are exempt from that examination.

Q. Briefly summarize the history and results of licensed operator testing by the NRC at the Rancho Seco facility.

A. Since May 1974, a total of twenty six applicants have been examined and subsequently licensed at Rancho Seco. Eighteen originally applied for a complete senior operator examination, i.e. operator written, senior operator written, and an oral examination. All of the eighteen passed the examinations on their initial attempt. Eight other applicants have applied for reactor operator licenses since January 1975. Two of these applicants failed initially, but passed a subsequent examination within one year and were issued reactor operator licenses. Four of these eight licensed operators subsequently applied for upgrade to senior operator. One of these applicants failed the initial examination, but passed a subsequent one.

Q. As a result of the Commission's May 7, 1979 Order, what additional testing of licensed personnel at Rancho Seco has been conducted by the NRC?

A. The additional testing of licensed personnel at Rancho Seco that has been conducted by the NRC includes the following:

1. Oral interviews by an OLB examiner and an I&E inspector of 7 licensed personnel on June 1 and 2, 1979.

2. Re-audit of 8 licensed personnel by the same I&E inspector on June 7 and 8, 1979.

3. Written and oral senior operator examinations administered to a licensed Rancho Seco operator by an OLB consultant examiner on November 29, 1979.

Q. Would you say that these NRC tests have covered whether the licensed operators can act responsible and appropriately to make judgment decisions during a loss of feedwater transient?

A. Yes. The attached letter from J. J. Mattimoe, SMUD, to Harold R. Denton, NRC was used as an aide by NRC personnel conducting the audits in June, 1979. The following subjects were covered in the control room with the Rancho Seco licensed operators:

1. Verifying AFW flow on loss of 4 RCP's (pg. 1).

2. How to power AFW pumps from essential Nuclear Services buses 4A/4B (pg. 1).

3. Reason for stationing an operator at FW Valve-055 during surveillance tests (pg. 2).

4. AFW values that have been added to the locked valve list (pg. 2).

5. Control of AFW flow independent of ICS (pg. 3).

6. Changes to emergency procedure D.14, "Loss of Steam Generator Feed" (pg. 3).

7. Modifications to AFW flow indications (pg. 4).

8. Procedure for transferring AFW pump suction to alternate supply (pg. 5).

9. Changes to emergency procedure D.10, "Loss of Reactor Coolant Flow/RCP Trip" (pg. 5).

10. Changes to control room annunciators for all auto start conditions of the AFW system (pg. 5).

Also attached to Mr. Mattimoe's letter is the lesson plan for instruction of licensed personnel.

During the audits conducted in June 1979 by the NRC, no deficiencies were found in the licensed operators' ability to respond responsibly and appropriately to a loss of feedwater transient.

Q. Did the interviews conducted on June 1-2, 1979 reveal any other areas of weakness?

A. Yes. Certain operators displayed insufficient comprehension of thermodynamics, natural circulation, and the TMI-2 sequence.

Q. Has the NRC conducted follow-up interviews with these individual licensed personnel to discuss these areas of weakness?

A. Yes. On June 17 and 18, 1979 Mr. Philip Johnson of Region V conducted eight follow-up interviews. The follow-up interviews demonstrated substantially improved knowledge in these areas. On the basis of the follow-up interviews, Mr. Johnson found the operators' comprehension in these areas to be adequate.

Q. On the basis of your review of the Licensee's training and testing program, do you believe the Licensee has effective procedures for determining the competence of its operating personnel?

A. Yes. I believe the licensee's present procedures are effective for determining the competence of the operating personnel.

BRUCE A. WILSON

PROFESSIONAL QUALIFICATIONS

I am a Reactor Engineer in the Operator Licensing Branch, Division of Project Management, Office of Nuclear Reactor Regulation. I am responsible for developing, preparing and administering examinations for applicants for reactor operator and senior reactor operator licenses. I am assigned to the Power and Research Reactor Group, which is primarily responsible for administering examinations on Combustion Engineering and Babcock & Wilcox designed reactors in addition to research reactors.

I received a Bachelor of Science Degree in Mechanical Engineering in 1966 from Syracuse University and a Master of Science in Nuclear Engineering in 1967 from the University of Washington.

In 1967, I entered active duty with the United States Air Force and was assigned to the 10 Megawatt Nuclear Engineering Test Facility (NETF), Wright Patterson AFB, Dayton, Ohio. From 1967 to 1968, I was a Project Engineer in the Experimental Branch where my primary function was to design and perform safety analyses of in-core irradiation test experiments.

From 1968 to early 1970, I was Chief, Reactor Engineering Section, where I performed safety analyses for reactor modifications and safety limit bases for technical specifications. During this period, I was certified as a Reactor Operator and Shift Supervisor at the NETF by the Air Force Directorate of Nuclear Safety.

From 1970 to 1971, I was assistant to the Chief, Operations and Maintenance Division during the final decommissioning and entombment of the facility.

In 1971, I was transferred to the Armed Forces Radiological Research Institute in Bethesda, Maryland. For eight months, I was Project Manager in the Accelerator Division and then transferred to the Reactor Division, where I was Assistant Physicist-in-Charge of a TRIGA Mark F reactor. I received a Senior Reactor Operator's License for this facility from the U.S. Atomic Energy Commission (AEC) and was primarily responsible for experiment safety review, technical specification revision and training.

In October 1973, I resigned my commission with the Air Force and joined the Operator Licensing Branch of the AEC. From May to December 1979, I was assigned to the Systems Group of the Bulletins & Orders Task Force.

My functions on this Task Force were to review and approve the Small Break Loss-of-Coolant Accident (SBLOCA) Guidelines developed by Westinghouse and B&W, and to insure that the applicable facilities have developed emergency procedures incorporating these Guidelines. Finally, I audited the operators and training records to determine that sufficient training had been conducted regarding the SBLOCA phenomenon and the revised emergency procedures.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
SACRAMENTO MUNICIPAL UTILITY DISTRICT	)	Docket No. 50-312 (SP)
	)	
(Rancho Seco Nuclear Generating Station)	)	

NRC STAFF TESTIMONY OF BRUCE A. WILSON
ON CONTROL ROOM DESIGN

(Board Question 31)

Q. Please state your name and your position with the NRC.

A. My Name is Bruce A. Wilson. I am an employee of the U. S. Nuclear Regulatory Commission assigned to the Operator Licensing Branch. From May 1979 until December 1979 I was with the Systems Group of the Bulletins and Orders Task Force.

Q. Have you prepared a statement of professional qualifications?

A. Yes. A copy of this statement is attached to this testimony.

Q. Please state the nature of the responsibilities that you have had with respect to the Rancho Seco Nuclear Generating Station.

A. I was responsible for reviewing part of SMUD's responses to the Commission Order of May 7, 1979. Specifically, I reviewed their procedures to ensure that their revised procedures were in accordance with the requirements of the Order and complied with the Small Break Loss-of-Coolant Accident Guidelines that were developed by B&W. I also conducted an audit of some of Rancho Seco's operators and senior operators to evaluate the training they had received concerning the TMI-2 accident and the resulting impact at Rancho Seco.

Q. What issue are you addressing in this testimony?

A. I am addressing Board Question 31, which states:

> Board Question 31
>
> Rancho Seco, being a Babcock and Wilcox designed reactor, has a control room configuration which is poorly and inadequately designed for plant operators to avoid a loss-of-feedwater transient, and therefore is unsafe and endangers the health and safety of Petitioners, constituents of Petitions and the public.

Q. How could a control room, and Rancho Seco's in particular, be configured for plant operators to avoid a loss-of-feedwater (LOFW) transient?

A. The configuration of the control room has very little effect on whether or not a LOFW transient will occur. One unlikely means by which the configuration may have an effect, however, is if controls for valves and pumps are located in areas where accidental actuation of them is possible.

In my opinion, the Rancho Seco control room is configured such that this accidental actuation of feedwater controls is very remote.

Q. Can the control room configuration have any effect on the operators' ability to diagnose and respond to a LOFW?

A. Yes. A study undertaken in connection with the "TMI Special Inquiry" has shown that the TMI-2 control room design was a highly probable contributor to the accident. NUREG/CR-1270, "Human Factors, Control Room Design and Operator Performance at Three Mile Island 2." A Human Factors Engineering Test and Evaluation (HFE T&E) was performed on the TMI-2 control room and was compared with studies of two other similar vintage control room designs. These evaluations included labels, markings, controls, displays and measures, and work space. In all evaluations and comparisons the TMI-2 control room was judged very inferior.

Q. Specifically, what factors in control room design would affect the operators' ability to respond to a LOFW transient?

A. One of the significant factors that was identified was color coding. The color red was found to have 14 different meanings, while green and amber had 11 each. Panel layout was also identified as being very important. Controls and indications for system components should be logical and consistent. A significant number of violations of this principle were found at TMI-2, in particular, the arrangement of the emergency feedwater controls and displays (<u>see</u> Figure 5, NUREG/CR-1270, Vol. 1).

Q. How does the Rancho Seco control room compare with TMI-2?

A. A formal HFE T&E would have to be performed at Rancho Seco for an accurate comparison. I believe such a study would show Rancho Seco to be far superior. The Rancho Seco Station Manual specifies control room criteria. Several of the criteria are the following:

1. Arrange controls, indicators, recorders and alarm indicators in functional groups and in a functional sequence wherever practicable.
2. Use uniform types and arrangements of control devices for similar functions wherever practicable.
3. Arrange the safety features devices on the panel in such a manner that the operator will have all necessary controls for a given system in a functional grouping.

The NRC presently has no regulations or criteria pertaining to the concept of Human Factors Engineering in control room design[1/] and, therefore, we do not know the degree of planning and effort that went into the Rancho Seco control room. However, on the basis of a comparison with other control rooms it appears the Licensee devoted considerable attention to its design.

Q. Do you think the Rancho Seco control room is designed to provide sufficient information and controls for the operators to safely respond to a loss-of-feedwater transient?

1/ NUREG-0660 contains a draft Task Action for Control Room Design (Action I.D. 1), including proposed development of standards.

A. I have spent a limited amount of time in the Rancho Seco control room during the site visit of June 1 and 2, 1979 in response to the Commission Order. However, I have spent a good deal of time at the B&W simulator, which is fashioned after the Rancho Seco control Room. On the basis of this experience and having been in or conducted operator examinations in 35 different nuclear power plant control rooms, I would rate the Rancho Seco control room design among the best. During the week of February 10, 1980 I will be conducting operator examinations at Rancho Seco and will evaluate the control room configuration and the ability of the reactor operator applicants to respond to a loss-of-feedwater transient. This evaluation may be included in supplemental testimony.

BRUCE A. WILSON

PROFESSIONAL QUALIFICATIONS

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In October 1973, I resigned my commission with the Air Force and joined the Operator Licensing Branch of the AEC. From May to December 1979, I was assigned to the Systems Group of the Bulletins & Orders Task Force.

My functions on this Task Force were to review and approve the Small Break Loss-of-Coolant Accident (SBLOCA) Guidelines developed by Westinghouse and B&W, and to insure that the applicable facilities have developed emergency procedures incorporating these Guidelines. Finally, I audited the operators and training records to determine that sufficient training had been conducted regarding the SBLOCA phenomenon and the revised emergency procedures.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of SACRAMENTO MUNICIPAL UTILITY DISTRICT (Rancho Seco Nuclear Generating Station)	)))))))	Docket No. 50-312 (SP)

NRC STAFF TESTIMONY OF BRUCE A. WILSON ON INSTRUMENTATION FOR DIAGNOSIS AND CONTROL OF OFF-NORMAL CONDITIONS

(CEC Issue 5-3a)

Q. Please state your name and your position with the NRC.

A. My name is Bruce A. Wilson. I am an employee of the U.S. Nuclear Regualtory Commission assigned to the Operator Licensing Branch. From May 1979 until December 1979 I was with the Systems Group of the Bulletins and Orders Task Force.

Q. Have you prepared a statement of professional qualifications:

A. Yes. A copy of this statement is attached to this testimony.

Q. Please state the nature of the responsibilities that you have had with respect to the Rancho Seco Nuclear Generating Station.

A. I was responsible for reviewing part of SMUD's responses to the Commission Order of May 7, 1979. Specifically, I reviewed their procedures to ensure that their revised procedures were in accordance with the requirements of the Order and complied with the Small Break Loss-of-Coolant Accident Guidelines that were developed by B&W. I also conducted an audit of some of Rancho Seco's operators and senior operators to evaluate the training they had received concerning the TMI-2 accident and the resulting impact at Rancho Seco.

Q. What issues are you addressing in this testimony?

A. I am addressing CEC Issue 5-3a, which states:

<u>CEC Issue 5-3a</u>

> Are the special features and instruments installed at Rancho Seco adequate to aid in diagnosis and control after an off-normal condition engendered by a loss-of-feedwater transient?

Q. What is generally meant by a loss-of-feedwater transient?

A. A loss-of-feedwater (LOFW) transient is usually regarded as a partial or total loss of main feedwater flow to one or both steam generators.

Q. What conditions could cause or initiate a LOFW transient?

A. A wide variety of conditions could cause a LOFW transient. One of the more common causes is tripping of one or both main feedwater pumps as an equipment protective measure for the pumps. Usually the motive force for the pumps are steam turbines which have a number of devices to initiate shutdown of the turbines. Some of the automatic trips for Turbine protection are loss of lubricating oil, loss of condenser vacuum, thrust bearing wear, and overspeed. The pumps also are protected against abnormal conditions, such as the case at TMI-2 where inadequate suction pressure was sensed by the pumps causing them to trip.

Instrumentation malfunctions can also cause LOFW transients. For example, the pressure transmitter that senses inadequate suction pressure may fail causing a pump trip when one is in fact not needed. A failure of the main feedwater flow transmitter may cause the Integrated Control System (ICS) to close the feedwater control values, thus initiating a partial LOFW even though the pumps are still running.

Q. How is the plant designed to handle safely a LOFW transient?

A. The plant is designed to handle safely a LOFW basically by means of three systems: the I.C.S., the Reactor Protective System (RPS) and the Auxiliary Feedwater (AFW) System. The ICS is designed to initiate a runback (i.e., a reduction in power) of the reactor and turbine to within the capacity of the remaining feedwater in the event of a partial LOFW.

The RPS will shut down the reactor in the event of a loss of both feedwater pumps or a partial LOFW with which the ICS, for some reason, is unable to cope.

The auxiliary feedwater system is designed to automatically start and deliver water to the steam genertors for decay heat removal following the loss of main feedwater and reactor shutdown.

Q. What information or data is necessary for the operators to diagnose and respond to a LOFW transient?

A. The operators need information with respect to the following:

a. The magnitude of the loss of feedwater, i.e. whether one or both pumps have been lost or whether control of feedwater flow has been lost;

b. whether the ICS is responding as required;

c. whether the RPS has been called upon to shut the plant down, and

d. whether the auxiliary feedwater syste , if required, is functioning as desigged.

Q. As a result of the NRC review of the Licensee's response to the May 7, 1979 Commission Order, have you identified any areas where there was insufficient instrumentation and capability to immediately retrieve necessary information or data during a LOFW transient at Rancho Seco?

A. Yes. In order to verify or perform the actions described in the previous answer, we found that Rancho Seco, as all B&W operating reactors, did not have suitable indication of auxiliary feedwater flow. Therefore, flow detectors were installed on each of the auxiliary feedwater headers and flow indicators were installed in the control room. When required to verify feedwater flow, as, for example, in emergency procedure D.14, "Loss of Steam Generator Feed," the operator must:

1. Verify auto start of both auxiliary pumps and operation of auxiliary feedwater valves.
2. Verify auxiliary feedwater flow indicated on FI-31801 and FI-31901 located on H2PSA to both once-through steam generators (OTSG's) and levels maintained at ≥ 24 inches on the Startup Range ($\sim 10\%$ on the Operate Range).

We found the instrumentation and capability for information retrieval sufficient for the operators to perform all of the other actions as described in the previous answer.

Q. In your opinion, are the instruments described in this testimony adequate to aid in diagnosis and control after an off-normal condition engendered by a LOFW transient?

A. Yes.

BRUCE A. WILSON

PROFESSIONAL QUALIFICATIONS

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MR. BLACK: I would further note for the record that Mr. Wilson's professional qualifications statement is attached to each of these three pieces of testimony, and also will be incorporated into the record as well.

I have no further supplemental direct of Mr. Wilson, and he is available for cross examination.

MRS. BOWERS: Mr. Baxter?

MR. BAXTER: We have no questions.

MRS. BOWERS: Mr. Ellison?

CROSS EXAMINATION

BY MR. ELLISON:

Q First of all, a preliminary matter. Off the record this morning, I asked you if you could review SMUD Exhibit 20. Have you had an opportunity to do that?

A Yes, I have.

Q SMUD Exhibit 20 is the interrogatory responses in the Three Mile Island 1 inquiry that were provided by the licensee. Would you refer to the table, which, unfortunately, does not have a number -- Let me back up. Perhaps it does.

Would you refer to Attachment 1, which is designated Tabulation of Reportable Occurrences at Operating Nuclear Power Plants for the Period January 1, 1969, through December 31, 1979? And refer to the page that -- which addresses Rancho Seco, which I guess is four or five pages back? The far righthand column on that page, under Total Reports, I find the figures 102 total reports from all causes of which 34 were caused by personnel error.

Has everybody had an opportunity to catch up with where we are?

MR. SHON: What page?

MR. ELLISON: This is the reportable occurrence tabulation, Attachment 1, which is several pages long, and I am on a page that unfortunately is not numbered, but has Rancho Seco. The first plant listed -- the first plant listed is Quad Cities 1. These are in alphabetical order if you are having trouble finding them.

MR. BAXTER: It is not the computer print-out but the previous tables.

BY MR. ELLISON: (Resuming)

Q In the far righthand corner under Total Reports, appears the figures, All Causes, 102 reports for Rancho Seco, of which 34 were caused by personnel error. This morning I asked you, Mr. Wilson, if you could review that proportion of personnel related LER's to all LER's for Rancho Seco, and compare that to all of the other plants that are listed here.

Did you have a chance to do that?

A Yes.

Q Where does Rancho Seco fall with respect to the other 69 facilities in terms of its proportion of personnel errors to total LER's?

A It would appear on a quick look basis that it is first.

Q Which would be to say that it had the highest proportion of personnel errors to total LER's. Is that correct?

A That is true.

I also took a look at it on the basis of total reports as compared with time of operation, and it would seem to rank very favorably that way also.

Q Where are you referring now?

A Well, if you look at the total reports, they submitted 102 LER's in the 5.29 years that they have been operating, which on a proportion basis is fairly low for a plant that has been operating that period of time. Also in terms of clarification I did a fairly basic study of the LER's attributed to licensed personnel error for the NUREG-0667 study, and the basic conclusion that I drew from this, one, is, I did not look at Rancho Seco in particular, but there was a slightly higher proportion of LER's attributed to licensed personnel error on B&W plants.

I have since seen in Section 7 that number was -- the difference as compared with other PWR's was compared to be insignificant from a statistical standpoint.

Secondly, when you attribute them caused by personnel error, it does not identify whether or not it was licensed personnel. They have only been categorized by licensed personnel since January, 1978. And thirdly, categorizing LER's by personnel error is a very inexact science at this point. As I pointed out in 0667, we found instances where there was a plant in theperiod since January, 1978, that had something like 25 LER's attributed to licensed personnel error, and the same design plant -- this was a Westinghouse plant -- had none, so on the basis of the judgment of the person who is writing the LER, it is whether or not -- what the initiating cause, personnel error, equipment malfunction, whatever.

Q A couple of clarifying questions. You stated that you had compared Rancho Seco's total reports to its time of operation and found it favorable. We have already had some testimony in this proceeding about that. That is covered in the table which is listed Category 5 at the back. Is that correct?

A I am sorry. Was that a question?

Q Yes, I am just trying to determine whether the analysis which you make, which is essentially comparing the total number of reports to the length of operation of the facility would compare to the data which is presented in Category 5.

A Yes. What I basically said was this -- I did about the same breakdown as I did with trying to determine on a proportional basis, to take a quick look at how they went compared to the other plants in total number of LER's submitted, as a function of its operating history, and as you can see, they are in about the upper third or so, the upper fourth.

Q Okay. I have a couple of questions about this. I really intended this just to be a preliminary matter, but with respect to Category 5 and the total number of incidents over a period of time, the ranking Number 1 would be good. Is that not correct? That would mean you had fewer incidents of LER's. Isn't that right?

A I am not -- I am not sure. This is the first time I have seen this table.

Q When you said that you thought that the number 102 LER's over 5.29 years of operation was favorable, that is what you meant, isn't it?

A Yes.

Q You also -- You compared that to the figure 34 personnel errors in 102 total LER's which you also, I believe, referred to as favorable, and I would --

A No, I apologize. That is a misuse of the term, sir. It was not favorable.

Q Okay. Just to clarify, is it your testimony, looking at this -- assuming this data is correct -- that of the LER's submitted by Rancho Seco, it had the highest percentage of them being attributed to personnel error of all the plants that are listed here?

A I said the proportion appears to be highest.

Q Those are my questions with respect to that.

You participated in the evaluation of the Rancho Seco operators after the facility was shut down. Is that correct?

A When you use the term "evaluation", I would like to clarify it. It was an audit of the training that the operators received, yes.

Q This would be an audit of the special Three Mile Island training. Is that correct?

A Yes.

Q You have testified that you interviewed seven of the Rancho Seco operators at that time. Is that correct?

A No, it is not. I interviewed three, Phil Johnson, Region 5 inspector, interviewed the other four initially -- excuse me. I had four; he had three. It was seven.

Q So there were seven between the two of you?

A Right.

Q And you testified that some of the operators you interviewed displayed an inadequate knowledge of natural circulation phenomenon and fluid dynamics and that sort of thing. Is that a fair statement?

A That is true.

Q How many of the seven that were interviewed did not respond to your satisfaction?

A I believe I answered that in one of the interrogatories. I cannot remember the answer at this time. I think it was either three or four.

Q As a result of that, the staff required additional instruction. Is that correct?

A That is true.

Q Would that be instruction for all of the operators at Rancho Seco?

A Yes.

Q Would you describe the additional instruction that was given?

A Only in general terms. I was not here to observe it. They contracted with General Physics Corporation of Columbia, Maryland, to provide additional instruction. When I returned to Washington, we met with two members of General Physics in Mr. Collins' office, Paul Collins, my branch chief.

We discussed the findings with him, and the deficiencies we found, and one instructor from General Physics came up, and to my knowledge, performed additional training of the operators, and a second General Physics employee came out and conducted re-evaluations of the operators, and then Mr. Johnson of Region 5 did the follow-up NRC audit of those operators.

Q Mr. Wilson, Ms. McDermid is going to give you a document which I would like identified as CEC 48.

(The document referred to was marked for identification as CEC Exhibit Number 48.)

MR. ELLISON: This is a document that was furnished to us on discovery, and it appears to be an exam which I believe may be the exam that you gave to operators as part of your audit. I would like you to look at it and identify whether it is the exam that you gave.

THE WITNESS: No, it is not.

BY MR. ELLISON: (Resuming)

Q Could you identify that document?

A It appears to be the exam that the Rancho Seco training staff administers to its operators.

Q Do you know when the exam was administered?

A Some time in May of last year.

Q Do you know whether this exam was given before or after your audit?

A It was given before.

Q Do you know how the operators performed on this exam?

A I have it written down some place. I do not know the results offhand. I know most of them passed it with a passing grade, which was 90 percent.

Q Would you say that this exam is representative of the types of exams that are given to Rancho Seco operators as part of their requalification program?

A No, it is not.

Q Could you explain the difference?

A As a result of the Commission orders in May, we were directed to -- the facilities were directed to conduct training of their operators. As part of the audit -- I was on the audit team -- we requested that they administer an examination, a written examination to all of the licensed operators who participated in the TMI 2 training, that the facility administered and graded, and we would audit the results.

The precedent was more or less set at Oconee. We only -- when I say "we" from the Operator Licensing Branch -- only requested they administer and grade the examination, and Mr. Denton made a site visit to Oconee in which he established the passing grade of 90 percent. This is a specialized exam, and to my knowledge it is not representative of the type they give for requalification.

Q I recognize that this exam only covers the TMI accident, and was special in that sense. My question about representative, however, is whether you believe this is the format and the types and general difficulty of the questions that are given in the requalification program.

A I have not audited an annual examination that the Rancho Seco operators have received as part of the requalification program, but in accordance with their program their examination is divided into a number of categories. It generally follows the guidelines in Appendix A of 10 CFR Part 55.

So, the format is very much different from this examination here.

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Q Could you describe the types of questions that you posed that you did not feel you got satisfactory answers to in the original audit?

A Do you want the specific question or the general area we were talking about?

Q If you can recall the specific question, that would be the best.

A More or less, to the best of my memory, we asked the operators what indications they expected to receive that would indicate to them that they had sufficient natural circulation flow.

In some cases, their response was they did not know -- they knew it was not a proper delta t, but they did not know the proper range that it should be in.

We further posed the question that if they higher the delta t, then the better the natural circulation flow in which we had three operators respond to, saying, "Yes, that would indicate better flow," which is not true.

The second area was the TMI-2 sequence of events. We used one of the figures from the 79-05(a) bulletin, I believe it was, in which it showed the response of the pressure and level in the TMI-2 pressurizer in the first several minutes of the accident.

We found that some of the operators were unable to explain adequately why the pressurizer level was increasing while system pressure was decreasing.

In the third area of deficiency, we found -- posed the situation in which a small break had occurred. The primary system depressurized to saturated conditions. We asked them what they would expect to see primary temperature do if they depressurized a saturated system.

Several of the licensees indicated that the temperature would go into the superheat range, which in the absence of any external factors it would not. It would follow the saturation level.

Q Am I correct that this audit came after the licensee had conducted the special post-TMI training program?

A Yes.

Q So, it would have come after the time that was spent on the B & W simulator, is that correct?

A No, not necessarily. Initially, the Commission order said that they would assign one shift supervisor who had received the TMI-2 training on each shift.

When we went out there, I believe, they had met that commitment. They had further committed to providing the TMI-2 simulator training for all licensed personnel. We were out there in the end of May and the first several days of June.

They did not complete the simulator training until somewhere around June 22nd, I believe it was.

Q So, do you know whether the people, particularly those who gave you unsatisfactory answers, had had the simulator training at the point that you were auditing them?

A Some of them we talked to did not.

Q Some had?

A Some had.

Q It was sort of a mixed bag?

A (Nods in the affirmative.)

Q Could you refer to the last question on CEC-48, the question on the second page, question number 6? 6(a) asks the operator to briefly discuss how the operator can ensure that natural circulation is occurring.

Would you expect a proper response to that answer to include discussion of the indication and proper temperature ranges for verifying natural circulation?

A No, that is not what the questions asks. I audited a numer of -- I forget how many -- several of the examinations that the operators took.

Their written answers to this test. By and large, as I recall, they answered the question basically correct, but they would look for a stable delta t across the core between t-hot and t-cold. They would be looking at t-h and t-c indications.

So, from a basic standpoint, you would have to say they answered the question correctly. What we did were the oral exams. The oral exams lasted about an hour each, was to probe a little deeper, to ask a follow-up question to see what their level of understanding was.

If they said that delta t was supposed to be 50 degrees, we said, "Suppose delta t were 100 degrees, would that indicate better or worse natural circulation?"

So, what we did with the oral exam was to probe deeper than you can with a written question.

Q So, would it be fair to say that you are asking questions that went beyond what the licensee administered in this exam?

A Well, I think that is stretching the point a little. It could have clarified the question in 6(a). It says "Briefly discuss." Now, briefly discuss, they could answer the question correctly by saying, "Well, we expect to see a delta t on 40 degrees."

If I was writing the question, I would write it much more pointedly. That is to say, exactly what indications or list four indications you would expect to see, proper natural circulation, which would include a delta t -- having auxiliary feed to the steam generators, that bypass valves will be opening periodically to remove steam, and that there was steam indication in the steam generators, and that it was subcooled.

So, I would ask the question more pointedly.

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Q Following the responses you received, did you inform the operators or SMUD management of what questions had been -- had not been satisfactorily responded to before they initiated their current retraining program?

A Yes, I did. I sat down with their training coordinator, Jack Mau -- first we had a management meeting. I believe it was on a Saturday between the Rancho Seco staff and the NRC staff.

We basically discussed that we did find the training needed some improvements. We did not get into the specifics in that meeting, but I did sit down with their training coordinator afterwards and tell him the specifics of what we found.

Q Some time later, you returned and reaudited. Is that true?

A I did not. No. Mr. Johnson of Region V did.

Q Are you familiar with that second audit?

A Only from talking with Mr. Johnson.

Q Mr. Johnson is going to be a witness in this proceeding.

A No, a different one.

Q I'll address my questions to you.

A Phillip Johnson conducted the first audit with me, then he conducted the second follow-up audit.

Q Let me address my questions to you about the second audit. I'm glad you clarified that. I could have reserved a lot of questions for the qrong guy.

Do you know how many operators were audited the second time around?

A Eight, I was told.

Q Do you know whether they were -- whether they included all of the original seven?

A As far as I can remember, I was told that five of the original seven plus three others.

Q Do you know whether they included all of the original operators who had essentially failed the first audit?

A No, I do not.

Q Do you know what the questions that were posed to the eight operators on the second audit were?

A No, I do not.

Q Do you know whether they were audited on -- as a completely new audit on all of the TMI problems, or whether they were audited on those areas where they had proved deficient in the first audit?

A I assume there were pretty much the same areas that they proved deficient on the first time. However, they probably posed the same type of questions using different wording, which is a typical examining tool that we use.

Q Would it be fair to say then, that your first audit posed some questions that you got unsatisfactory answers to that you told SMUD management what questions had not been properly responded to, that you then came back and posed substantially the same question, perhaps with different wording, and got satisfactory answers?

A I guess an example would be best to explain that. The concept -- we were looking whether or not they understood concepts. By concepts, I mean termodynamics and heat transfer and fluid flow.

Initially, we found, as I said before, that the first time we posed a hypothetical situation, if you depressurainze a saturated system, what happens to temperature in which they responded unsatisfactorily. At least some of them did.

If I today follow-up, I would not ask the same question that way. I would pose it a different way. I would ask them, say, at no load, when they are first bringing the plant up from shutdown, how they control primary system temperature with a secondary header pressure controller in which they would have to relate back to saturated conditions, and what happens to you in a saturated system when you pressurize it.

So, it is essentially looking for the same concept, but it is asking it a different way.

Q Do you know whether in the second audit the questions

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were posed as differently as you have described here?

A No, I do not. I only got to talk to the inspector briefly after his follow-up of it. I would assume he did not ask the same specific questions, but slightly different in order to ascertain whether or not the concepts we had gotten across to the operators --

Q Why would you assume that?

A Well, I think that Mr. Johnson is an intelligent person. He -- no. He observed the first evaluation I did of an operator at Rancho Seco, because I had been involved at Oconee.

I have been examining for about six years now, so there are certain techniques in examining -- how to pose questions. He more or less observed the first one, then conducted the rest of the audit examinations by himself.

So, he needed techniques that were involved.

Q Were you involved in the post-TMI audits of any other B & W facilities?

A Yes, I was involved in Oconee and in the follow-up of Crystal River. I say "follow-up" because it was another examiner plus I and the instpector who did the initial audit at Crystal River.

Q Did those operators exhibit the same deficiencies that the Rancho Seco operators did?

A Yes, they did.

Q So?

A In fact, there was another examiner who performed the audits at Arkansas and Davis-Besse. Of the operating B & W utilities, we found the only one who did not require follow-up evaluations was Davis-Besse.

Subsequent to the guidelines being approved by the NRC for the remaining NSSS vendors, Westinghouse, CE, and General Electric, we also performed audits and found the same deficiencies in the knowledge of thermodynamics, heat transfer, and fluid flow.

So, I would say the situation was not only for the B & W reactors.

Q This general area, is this something that was part of the NRC operating license exam prior to TMI?

A I am sorry. I did not hear the question. Did you say was it part of our exam?

Q Yes.

A It was not a separate category as it is as of May 1 of this year, but we did ask questions concerning it, particularly in category C which is general operating characteristics on the written, and category J which is specific operating characteristics of a senior examination.

Q Would you have expected an operator to understand the concepts that we are discussing?

A Yes.

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Q Let me give you an example of a fluid flow phenomena and ask whether you would expect an operator to understand this, that when fluid, let's say, in the pressurizer is discharged to the operated relief valve and it loses pressure in the tail pipe following its exit from the PORV, that its temperature would drop.

Would you expect an operator to understand that concept?

A Post-TMI? Yes, we would.

Q Would you expect it prior to TMI?

A No, because I probably would have answered it the same way they did.

Q But you would now, is that correct?

A Yes. As a matter of fact, I came back to the exam severalweeks ago and found the same thing. This was on a Westinghouse reactor. Operators still do not understand how you -- now, you said when fluid is discharged from the PORV. I assume you are referencing the depositions of one of the Rancho Seco operators. You have to be more specific in giving the initial conditions.

For instance, if the steam space in the pressurizer was released, due, for example, to stuck open PORV, yes, you have a throttling process. If you are in the feed and bleed mode in which you are discharging to the vales and looking at the downstream temperatures, it is a different situation.

Q You have had an opportunity to review the answer in the deposition. Is that correct?

A Yes, I have.

Q Do you believe it was correct?

A No, I do not believe it was correct, but again, I emphasize that the initial conditions must be specified up to that point of questioning.

The individual involved was led to believe, I assume, that he had a solid system. The questions up to that point postulated the feed and bleed mode of cooling in which the pressurizer is full of water.

So, he would assume that -- he may have been carrying on from the previous line of questioning, assuming that the pressurizer was discharging the water.

Now, I am not trying to make excuses for his answer because, as I said, even several weeks ago I found licensed operators when I specifically said it was a stuck open PORV from the steam space, they answered incorrectly.

Q If the operator involved here -- for the record, Mr. Morisawa -- had assumed that we were in the feed and bleed mode. He was discharging either two phase or solid water. Was his answer correct with that assumption?

A I do not recall his specific answer. How it was worded. I think it was postulated that the pressurizer --

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the coolant in the pressurizer. It was not specified whether it was solid water, two phase or steam -- was less than 600 degrees, I believe it was.

I think the question was: Do you think that the downstream temperature would indicate approximately that value?

As I recall, he said yes. So, I assume his answer depending upon conditions, how long has it been discharging, has it reached thermal equilibrium, what is the pressure, has a quanch tank rupture disc blown?

There is a lot of postulated questions you can attach to it.

Q Do you believe that the conditions would be such that the temperature in the tail pipe would be virtually identical to the temperature in the pressurizer?

A No.

Q That was his answer. Isn't that so?

A I would think so, yes.

end tP-11

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Q Are you familiar with the -- Well, are you familiar with the training hot license and requalification programs of other utilities?

A Yes.

Q Would you say that Rancho Seco's program is substantially different than industry practice?

A No.

Q Are you --

A It depends on what you mean -- again, what you mean by substantially. There are differences, but I would not regard them as substantial, no.

Q Generally --

A Generally.

Q -- patterned the same?

A (Nods in the affirmative.)

Q Are you familiar with the training of the TMI 2 operators?

A I am more familiar with the training of the TMI 1 operators, and I assume the training of TMI 2 was pretty much the same.

Q Apart from the special TMI training which was undertaken in response to the May 7th order, what would you feel are the major differences between the SMUD program today and the TMI 2 or TMI 1 program as it existed prior to the accident?

A I have not made a detailed comparison of them, but I assume they are fairly similar.

Q Could you refer to Page 2 of your testimony on operator training, the answer that begins at the top of the page. In the second sentence, you state that you reviewed their procedures to ensure that their revised procedures were in accordance with the requirements of the May 7th order.

Could you describe in a little more detail what you mean by that, and particularly what you saw as the requirements of the order?

A I think that says it right there. I say specifically, "I reviewed their procedures to ensure that their revised procedures were in accordance with the requirements of the order and complied with the small break loss of coolant accident guidelines that were developed by B&W."

I don't recall the exact words of the order right now, but the order says they were to develop procedures and train the operators to respond to small break loss of coolant accidents.

Q Did the order actually set forth any requirements for the procedures themselves other than they be developed?

A I do not recall. Not that I know of.

Q My question is, you state that you reviewed the revised procedures to see that they complied with the requirements of the order.

The thrust of my question is, what criteria did you use for your review?

A The B&W guidelines.

Q At Page 4 of your testimony, unless I say otherwise, I am referring to the same testimony, operator training. The very last sentence of the answer that concludes there states, "The requalification program was regularly audited by the NRC's office of Inspection and Enforcement and the Operator Licensing Branch."

Could you describe in more detail how the NRC audits the requalification program?

A The Operator Licensing Branch, of which I am a member, is responsible for auditing the written examination, and any other quizzes that are given as part of the requalification program. We reviewed the examination in terms of quality to ensure that it is essentially the same as our standards, and in terms of the grading criteria, to be sure that the grading was also in accordance with our standards and uniform for the exams that we audit.

We have a procedure for doing this. We are supposed to look at three operator and three senior examinations and sit down and grade a category ourselves and compare our grades with those that the facility gives.

I&E, on the other hand, is more responsible for assuring that the rest of the commitments made in the requalification program are in fact performed, such as attendance at lectures, reactivity manipulations, and so forth.

Q Do you review the course materials that are presented?

A No.

Q Do you attend the lectures?

A No.

Q But you do take a section of the requalification exam, grading yourselves, and compare your grades on that section to those of the licensees. Is that correct?

A Yes.

Q What would be the procedure if your grades came out substantially different than the licensees?

A Well, I brought the procedure with me. It will take a while to find it, but basically, if a number of -- not a number -- we usually grade two different categories at a minimum, and if our grades come out more than five points lower on several of the comparative categories, and we grade further categories, usually one, then the pattern has developed that their grade is significantly higher than ours -- and I mean by significant, five points higher than we would have given on pretty much of a pattern basis -- then we ask them to reproduce all of the requalification examinations and send it in to headquarters, Washington, for our review.

So, we review all of the examinations.

Q Has that ever occurred with SMUD?

A No.

Q Do you review at least one section of every requalification exam?

A Initially, and how the program was set up was that the first two years a member of the Operator Licensing Branch was to perform this audit of the written examinations once per year. If no deficiencies were found, then we would go to an every two year basis. If we did find deficiencies that were not signigifant enough to return to headquarters, then we would go to a one-year basis.

If they were significant, then we do follow-up action that is at the discretion of the branch chief.

Q So what has been the practice with SMUD? Every two years? Is that the --

A I believe they had to be audited two years in a row and possibly every two years after that. This was the guidelines. We have not always been able to follow them in every case because of resource limitations.

Q Can you describe to the best of your knowledge what the pattern has been with the licensing in this case?

(Pause.)

A This is the requalification file that we maintain for Rancho Seco. At this time I am sure an audit was performed of the written examination at least once. I know the unit was down for quite a while, but -- when they had the turbine problems, I think, back in 1975 or 1976 -- and I know an audit was performed, yes. Okay, one was performed in 1975, and to the best of my knowledge another one was performed -- however, I cannot find the evaluation sheet in here.

Q Did you participate in the other one?

A No, I did not.

Q Did you --

A When you say "the other one," I did not participate in any of them.

Q Did you -- Do you have any recollection of when the other one that you recall but cannot substantiate was?

A I think it was around 1977 or so, because I remember the examiner who was assigned to come out here and perform the examinations.

Q Did you expect to find a record of that in the file you are looking through now?

A Yes.

Q If you have an opportunity -- Don't worry about it now -- but if you have an opportunity during the break, I would appreciate it if you would go through the file and you would come back and complete your answer.

A Okay.

Q Does the staff take any -- play any part in the selection of what items will be taught, what subjects will be taught, and what subjects will not be taught in the requalification program, or did they leave that to the utility?

A Guidance is given in Appendix A to Part 55 as to the subject matter that should be covered, and aside from that, no, we do not give particular guidance as to what subjects must be taught.

Q Has the staff made any effort since the Three Mile Island training to verify that the lessons learned from Three Mile Island had been incorporated in the licensee's requalification program?

A We have not to the best of my knowledge required any changes to be made in the requalification program as a result of -- not just lessons learned. I am thinking particularly of the task force -- but I am thinking about the master action plan, but we have required changes to be made in the overall training and qualification of licensed operators.

Q Would you briefly describe the changes that you have required?

A No, not briefly. I would have to -- I have the list here. There are 16 recommendations that Paul Collins made to the Commissioners that were subsequently adopted.

Q I have a letter here that I would like identified as CEC 49.

(The document referred to was marked for identification as CEC Exhibit Number 49.)

end P12
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flws P12
srb 1

Q I'd like you to look at it briefly and tell me whether this letter, which is to all power reactor applicants and licensees from the Commission and signed by Harold Denton, and the date on mine I believe is March 29, 1980, the subject is Qualifications of Reactor Operators -- whether this letter describes the changes that you're referring to.

A Yes, I think it refers to most of them.

Q Are these now requirements that licensees have to adopt these criteria?

A Depending on the effective date,yes. Some of them have already been made into requirements as of May 1st of this year, and some, for instance on 1B, the effective date isn't until December 1st of this year.

MRS. BOWERS: Mr. Ellison, the date that's stamped at the top here is March and there's a 2 and there apparently was another number. Maybe Mr. Wilson has a copy that shows --

THE WITNESS: Mine has the same omission but it is a March 29th letter, yes.

MRS. BOWERS: March 29.

MR. BAXTER: I do note that's a Saturday. You still think that's the right date?

THE WITNESS: We have been known to work on Saturday's occasionally.

(General laughter.)

MR. ELLISON: We'll stipulate that it was on or about March 29th that this letter was mailed out.

Mrs. Bowers, rather than having the witness describe these recommendations, these have been identified as the upcoming requirements, I would just move the admission of CEC Exhibit 49 as a substitute for having the witness go through those requirements.

MR. BAXTER: No objection.

MR. LEWIS: No objection.

MRS. BOWERS: CEC Exhibit 49 is admitted into evidence.

(The document referred to, heretofore marked for identification as CEC Exhibit No. 49, was received in evidence.)

BY MR. ELLISON (Resuming):

Q I'd like to refer you back to the exam just for one more question. The exam, of course, is CEC Exhibit 48 in this proceeding. My question is whether you think that this exam covers the same basic areas as your audit.

MR. BAXTER: Which audit are we referring to now?

BY MR. ELLISON (Resuming):

Q I'm referring to the audit that was conducted in the end of May, early June, in response to the May 7th order, but I'm referring to both the first and the second parts of that audit.

A I'm sorry, the question was?

Q Whether this exam is roughly representative of the subject matter of your audit.

A Yes, it is. I believe that Rancho Seco personnel were in contact with some of the other utilities that had already administered this type of examination and asked them what subjects to cover, what type of questions.

Q So just to clarify, if I was interested in the subject of your audit, would it be fair for me to look to this exam?

A Yes.

Q In your audit, did you learn whether or not each of the operators at Rancho Seco had read the B&W small break analysis?

A No, we did not. I don't know.

Q Do you know today whether they have?

A No. Whether they read it, I don't know. Whether they were instructed in it, yes, as part of the requalification training they received at the B&W simulator. And this was not the TMI-2 special training; it was a one-week requalification training. I attended two days of that training session -- one training session, excuse me -- at B&W.

Q When did you attend the training session?

A February or March. I think it was February. Yes. It was right when Crystal River happened.

Q And did you observe the training of Rancho Seco operators as opposed to other utilities?

A Yes.

Q It would probably be easier for the reporter if you wait for me to finish, even if you know where I'm going.

A I'm sorry.

Q At the time you conducted your audit, there was no requirement for tripping the reactor coolant pumps, is that true?

A That's true.

Q Did the staff re-audit when that requirement took effect?

A When you say staff -- well, there are two parts to that. I&E may have done it, and when you say audit as in the same context as the other audits, I doubt it. Now, they may have performed an inspection to see if they did include the reactor coolant pump trip criteria and training in it, but I&E may have, but OLB, Operating Licensing, did not.

Q When you say they may have audited to see if they included the criteria in it, is the "it" you're referring to the requalification course?

A No, I'm sorry, I meant in their training and in their procedures.

Q In both the training and the procedures? Is that something the I&E department would routinely do?

Let me clarify my question. I asked you earlier whether you reviewed t e course materials, the lectures, that sort of thing, and you responded no. Does I&E review the course materials, sit in on the lectures and monitor, if you will, the progress of the training itself?

A I think they have on occasion sat in on lectures. As to how much they monitor the whole training program, I really don't know.

MR. LEWIS: Mrs. Bowers, may I suggest a break now?

MRS. BOWERS: All right, we'll have a 10-minute break.

(A short recess was taken.)

MRS. BOWERS: On the record. Mr. Ellison?

THE WITNESS: Could I clarify the second audit of the requal exam?

BY MR. ELLISON (Resuming):

Q Please do.

A Apparently, this is -- well, this is the file that we maintained for the Rancho Seco Requal Program, and they did have a lengthy shutdown in 1975 and 1976, and I have a letter from Mr. Oubrey to our branch about the annual written exam, and it was reviewed apparently by one of our headquarters examiners who I knew was out here in that period of time but his report is not contained in the exam. It does reference his audit in that he said that the exam that they had given he says, "The content and depth of the exams were satisfactory, but the exams themselves were unnecessarily long and difficult to grade." And he then went through and made some recommendations on how to restructure the requalification exam. But I don't have his report of the exam in here, so apparently, that was the last time an audit was performed of the Rancho Seco Requal exam.

Essentially, the audit states that the exam was in excess of our requirements.

Q When was that?

A This letter was dated November 18, 1976.

Q When you say he stated the exam was overly long and difficult to grade, is that what you were referring to as being in excess of the requirements? Or does he say something else?

A Oh, yes. "This examination was unique in that it regrouped the traditional RO-SRO sections." Mr. Buzy is the examiner and he further discussed the advisability of "arranging the questions within a section of the examination and in ascending order of difficulty and depth of knowledge." And then he went through to discuss what a typical exam should be. He had an RO and SRO level.

Q Is it your understanding from reading that letter that this was the same kind of audit as the first one? A formal audit?

A Yes, it is my understanding.

Q Would you explain why there is no report of it in the file?

A No, I can't.

Q When you conducted the audit in connection with the May 7th order and examined 7 of the operators and found that, I believe you testified, 3 or 4 of them responded unsatisfactorily. When you came back the second time, why didn't you audit all the operators rather than 8?

A I didn't come back the second time.

Q That's a generic "you."

A Okay, the NRC?

Q The NRC, yes.

A Well, it was essentially the same type of audit as we have done in the conduct of giving any examinations. It is an audit process; therefore, there's quality control type of things. When you look at one widget out of 100 if it looks good then you have sampled it adequately and pass on. When you find a bad one, then you sample some more. In this particular instance, he audited 8 out of the 14 people who were assigned to shift duty, which is more than 50% of them.

Q And they all passed?

A They all responded satisfactorily, yes.

Q How do you go about selecting the operators that you choose to audit?

A On the first audit in which I participated, we selected them because they were assigned to shift duty that particular day we showed up. We came in there, tried to arrive onsite about 6:00 or 6:30 in the morning and talked to three of the members of the shift that were due to go off at 8:00 o'clock, and then when the new shift reported on, we talked to four members. I believe there are only three required; however, there were four licensed personnel that we could talk to that particular afternoon, so it was the people who were available onsite at that time.

Q Was the procedure any different than the second audit?

A I'm not aware of how we did the second one. How we selected the 8 people.

Q Would that be typical for the NRC to show up at the site and audit whoever happens to be present?

A No, this was not a typical thing that we do. This was only in response to TMI. It was typical in terms of the audit processing that we used for determining in the compliance with the training requirements of the order, but we had never done this before.

Q So when you refer to an audit, with the exception of the one done in compliance with the order, you're referring to the review of the examination that you described earlier but not to an oral examination of the operators. Is that right?

A Yes, that's correct.

Q Referring to page 7 of your testimony, the question that appears in the middle of the page which begins, "On the basis of the tests that the NRC has conducted,..." then it goes on, what tests were you considering in your answer?

A These were the oral examinations that we conducted as part of the two audits, post-TMI, the oral examinations.

Q And that would be the same for the next question?

A I would amplify that -- I wrote this testimony before I went down to B&W to watch the requalification training, so I'd say on the basis of the tests we have conducted and the requalification training that I have witnesses at B&W, I do believe that the operators adequately understand the analytical actions.

Q But that wasn't the test, is that correct?

A No.

Q That's just observation.

A Yes.

Q So with respect to the other question on page 7, the one that appears at the very bottom, the word "tests" refers to the same audit tests that the preceding question does?

A Yes.

Q Referring to the next page, page 8, where you respond to CEC Issue 3-2, the question that follows that is, "Does the licensee, SMUD,have a program for apprising its personnel of new information pertinent to the facility's safe operation ..." et cetera. And you respond, "Yes, the licensee has stated ..." and you go on to describe the requalification program, et cetera. And you refer to the licensee's answers to California Energy Commission interrogatories. Do you have any other basis for this response other than what you've given here?

A No, I don't.

Q I'd like to refer you once again to CEC 49. That's the new operator training requirements. Are you familiar enough with those requirements that you can answer some general questions without having to read CEC 49?

A I'll try.

Q If you need to read it, stop me. Will these requirements increase the amount of training that is given to operators at Rancho Seco, or will they change the subject matter without increasing the amount of time that operators are trained?

A A subjective answer on my part would be that it will increase the training, yes.

Q And what's the basis for that?

A One, they will have to have two things that presently are not required, or three things actually. One is the new subject material that will be covered on the examinations.

We have a new category on both the operator and senior operator dealing with heat transfer fluid dynamics and fluid flow.

Secondly, there's a requirement that they spend three months on shift as an extra man. That in itself will decrease the proportion of time spent on shift in the control room learning how to operate the plant versus in the classroom. And thirdly, the grade criteria for the examination has been increased, which will require more training in order to pass the written examination.

Q Is there anything in these requirements that specifically mandates increased training as opposed to changing the nature of the exam or changing the subject matter that would be tested?

A Not that I'm aware of. I'd have to look through this to be positive of that. You're saying do we specifically say you must have two years of this type of training, plus simulator training and so forth? No, not that I'm aware of.

Q To clarify your answers, you're referring to the licensing exam, so I assume you're referring to licensing training. Is that correct?

A Training of licensed operator applicants, yes.

Q In that case, let me ask you the same question with respect to the requalification program. Are you aware of requirements in this document that will result in the increased amount of training as part of the requalification program?

MR. BAXTER: Excuse me. When we say increased amount of training, Mr. Ellison, are we referring to time?

MR. ELLISON: Yes. Essentially, referring to the amount of time spent in training.

THE WITNESS: Under Paragraph C., the requalification program, it does say that the program should be modified to require ce tain control manipulations.

BY MR. ELLISON (Resuming):

Q I'm sorry, you're at paragraph C. on what page?

A On page 5. This does not, on a time basis, increase the amount of requalification training that they must receive.

Q Would it be fair to say as a general matter that the requirements that are set out in -- well, I would presume that the requirements that are set out in CEC 49 have not been required until the issuance of this document. Is that correct? These are new requirements?

A I'm sorry. This document has been issued.

Q I know, but prior to its issuance, these were not requirements, correct?

A Correct.

Q Would it be fair then to say that at least in the NRC's mind, it is felt that each of these areas was an area that was not being sufficiently addressed by licensees?

A When you say NRC I'd have to say yes, also including the Commissioners because these requirements were made as a result of the recommendations that Mr. Collins made to the Commissioners as modified by their requirements.

Q May I refer you to the last couple of pages which set out the control manipulations that operators are to participate in. You say that you've had an opportunity to observe some of the B&W simulator training. Can you tell me whether all of these control manipulations are typically a part of the simulation training?

A As part of the requalification training? Hot license training or cold licensing, or all?

Q Requalification training first of all.

A All of them, no. In general, on requalification training they don't spend as much time as compared with their initial training on normal plant evolutions. They tend to concentrate more on transients and abnormal conditions.

Q With the exception of the normal operations then, would you say that all of the abnormal situations described on this list are already a part of the requalification program?

A I'm sorry, I think we're getting sidetracked here. What we're looking for essentially is that the requalification program presently requires and has required in the past each licensed operator manipulate their controls through 10 reactivity manipulations.

Now, in normal circumstances, the licensed operators receive these control manipulations as part of their every day duties. So I would say it's fairly uncommon to have an operator assigned to shift duty or a senior operator not fulfill the 10 reactivity manipulations over a 10-year period.

Now, the requalification program allows that if they don't fulfill the 10 manipulations on the plant, then they can be performed on the simulator.

Q My question, however, relates to not the reactivity manipulations but the abnormal events that are described here. It seems to me that the ones that are most pertinent are the ones that are in the first half of the second page. And having observed the simulator training at least for a couple of days, can you tell me whether it would be typical for a trainee undergoing requalification simulator training to experience all of these?

A All of them, no.

Q The majority of them?

A For example, I observed two days out of the five that they participated in requal training, and the abnormal or accident situations start essentially with number 7. So in my observations, they did experience number 7, loss of coolant; they did not have 8 or 9; they did have 10; they did not have 11, 12, 13, 14; they did have 15, they did have 16, they did have 20, they had 22, they had 23, they had 25 and they had 26.

Q Did they have 17?

A Not in the two days I observed them.

Q Do you have any knowledge of what might have been presented to them in the remaining days?

A The Rancho Seco operators -- at the time I did have the knowledge. Right now I can't remember. I did read the schedule of what they were supposed to perform, and I can't recall the other three days that I was not there.

Q Would it be generally true that during the simulator training in a given evolution, or given abnormal event, that the operator begins with the plant in stable operation under normal conditions and then the trainer essentially fails something and requires the operator to respond to that failure? Is that a fair characterization of the way it's typically done?

A Yes. Many of them, during requalification.

Q In the --

A I think this may help. This is requalification training conducted at the B&W simulator. This is not for Rancho Seco but is for another B&W unit, and typically, they receive -- it's one-week training. They receive four hours per day on the simulator and four hours in the classroom to discuss what they observed or what they expect to observe. So what they had in this particular schedule was a normal operation, reactor startup from all rods into 100% power, and a reactor trip. The second day was power operations with unannounced casualties. As was the third day, the fourth day and the fifth day.

Now, that was in general. And the specifics -- they keep track, for instance, this is a simulator training sheet of -- it will list the licensee's name, he has completed a one-week training program, and it tells what positions he was assigned on shift, whether he was the shift foreman, shift supervisor, reactor operator and so forth, auxiliary operator. And it tells the number of evolutions performed during that week.

For instance, they had a dropped rod, reactor trip, reactor coolant pump trip, turbine trip, a failed steam generator level instrument, a reactor startup to 10^8 amp, somewhat intermediate range, a startup to 5% power and a startup to 15%, power escalation. This goes on for a way. Do you want me to read the whole thing? It's a significant number of different reactivity manipulations that they do perform, both normal and abnormal.

Q Let me ask you some questions based on that. Any multiple-failure events?

A They're not listed here specifically as multiple failure events, but in the two days I did observe, yes, they did have multiple failures.

Q But there are none listed there? Is that what you're saying?

A Well, they don't list them specifically by the title multiple failures. They will give them an initiating event, such as a pump trip or reactor coolant pump or feedpump trip or some depressurization, whether it's a loss of coolant or whatever, that causes the SFAS actuation and then one of the ES functions fails to perform as required. Yes, these are multiple-failure events. They have done that.

Q In the two days that you were there, how many multiple-failure events did you observe?

A Oh, three or four I suppose.

Q Any that went beyond two failures?

A Not that I recall.

Q Mr. Rodriguez testified that -- I asked him some questions about whether operators had observed various kinds of degraded conditions on the simulator and he responded, as I recall, that in some cases they had and in some cases they hadn't. But that often they hadn't because they were presented with a problem and if they solved it correctly, they never saw the degraded conditions that would result if they hadn't resolved it correctly. Do you agree with that? Is that generally the way it goes? They present people with a problem and if they solve it, they don't see the results of failing to solve it?

A That's true in some cases, yes.

Q Is that a typical event?

A Well, what do you mean typical? It depends on what the initiated event was. They do normal evolutions in which there are no malfunctions; they do abnormal evolutions in which there are no secondary or second or third malfunctions; and they do a combination of all of the above. Yes, they can recover from abnormal transients and never see the degraded situation, and sometimes it's beyond their control and they do get to a degraded situation.

Q Let me ask you this. Do they ever start with the reactor in an extremely degraded condition and ask the operator to diagnose what's wrong with it and recover normal operation?

A Not that I'm aware of, no. We have been discussing that as shift technical adviser training.

Tape 6

Q Referring to page 11 of your testimony, in the first answer that appears there, halfway through, appears the sentence, "The NRC Staff believes that substantial improvement can be made in the process of dissemination of operating experience." What are some of the substantial improvements that the staff is aware of that could be made?

A Well, as I discussed previously in the testimony, we have set up a new division-level function with the NRC to evaluate operating experience. They have formed the Institute for Nuclear Power Operations, INPO; the Nuclear Safety Analysis Center, NSAC. So the organizations are there. We now need the mechanism of getting the information sorted out, getting the pertinent information down to the operators where it is supposed to do and will do the most good. And right now, I haven't been able to keep track of it as much as I'd like to, but right now I think we have the organizations there; we still need the mechanisms to get the proper information to the operator.

Q Those mechanisms don't exist at this time?

A Apparently not, from what I've seen so far.

Q Do you have any recommendations for licensees as opposed to industry as a whole or the staff?

A Do you mean do I or does the NRC?

Q Either.

A The staff does recommend that they include it as part of the requalification program; that they discuss operating events at other power plants,yes.

You see right now, it's very difficult because our present system, we do disseminate the LER's to the facilities and typically this is a computer printout that runs anywhere from 70 to 80 pages and takes hours and hours to read, and much of it is unnecessary information. So what we need is a mechanism to get rid of the riff-raff and get to the heart of the matter of what's pertinent to the operators.

Right now I guess that's subjective judgment on the part of the person disseminating the information to the operators of what is relevant to their knowledge and what isn't.

Q You mentioned a moment ago that operating experience at other reactors could be incorporated into the requalification programs, and that was one of the staff's recommendations. Do you know whether that's presently done at SMUD?

A In addition to what I have written as far as my testimony, no, I know of no other means for doing it.

Q This answer that you give on page 11, is that based upon the same interrogatory response that the answer on page 8 is based upon?

A Yes, it is. It only addresses the TMI-2 event.

Q Is it based on anything else?

A No.

Q Does the staff play any part in the writing of procedures at SMUD?

A Normally, no. The procedures we review from the standpoint that -- our Branch, Operating Licensing, reviews Sections 13.5 of the PSA and FSAR to see that the procedures will be developed in accordance with the applicable REG GUIDE which is 1.33. Once they commit to following the REG GUIDE, our branch's involvement in it is essentially through. Then it is up to some I&E inspection function, I believe, to make sure that the licensee develops their procedures that way. They review them.

We have set up a new branch within the revised NRR reorganization under the Human Factors Safety Division that is going to look at procedures. Now, whether they are going to look at all procedures or just emergency procedures or what, at this point I don't know. That's a relatively new function. It's only been set up a few weeks ago.

The only procedures that we have really had a direct bearing on how they were written are the ones that they had to rewrite in accordance with the Commission orders of last year.

Q You described the new functions of the Human Factors group, and I wonder if you could distinguish for me how their role is going to be different than what I&E does today with respect to reviewing procedures.

A No, I can't answer that. We have undergone the reorganization; there is a functional description out, but I haven't -- this has only been out in the last two weeks and in the last two weeks I was one week on the road and one week preparing for the hearings here, so I haven't had a chance to read it. I don't know what their exact function will be.

Q As part of the requalification audit, does --

A Excuse me, I do have -- if you would like to see it, it's a functional description of the NRR reorganization. This is what I said I haven't had a chance to read yet.

Q No. Maybe off the record I'll take a look at it. I don't think it would be very productive to go into it since neither you nor I have read it.

Do you know whether I&E at the present time, or any branch of the staff, reviews procedures with the operators to test the operators understanding of what the procedures say, and their format and when you use one procedure and when you use another and that sort of thing?

A I think an inspector can answer that much better than I can, but I am aware that they periodically check the technical content of procedures with the operators. For instance, they may do a valve lineup with a procedure to make sure every valve is identified properly. But I think that's more or less just normal procedures and not, say, abnormal or emergency procedures.

Q On page 13 of your testimony you described to some extent how the staff reviews emergency procedures to determine that the licensed personnel understand them. And you begin your answer by saying, "Through the examination process..." Is this the requalification examination that you're referring to?

A No, it isn't. That's the -- the question was how does the NRC determine, so the answer is not in the requalification exams, but basically in the initial licensing exams of the operators. Through the requalification exam, there is a requirement in Rancho Seco's program that the plant superintendent or his designated alternate walk through the emergency procedures with the operators on a periodic basis.

Q But the staff doesn't review that? Is that correct?

A The requalification? I believe I&E does inspect against it, that they have possibly a checklist within each licensee's folder that they have reviewed the procedures.

Q So I&E would look in the folders to see if the --

A I believe so.

Q Let me fini:n my question before you believe so.

A Sorry.

Q And give me a license to ask anything. The I&E would look in the folder to see whether the procedure review had been checked off? Is that essentially what they would do?

A Yes, I would imagine so. Under Rancho Seco's requalification program for records and documentation they have an individual training file and an individual training manual that they must maintain this documentation which is subject to I&E audit.

Q On the licensing exam, when you walk through the procedures, how is that done, and let me give you a description and tell me if this is accurate. Would you take a procedure, one procedure, and ask the operator o follow it and walk through what he would do?

A Are you referring to normal procedures or emergency?

Q Emergency procedure.

A No. We usually would hypothesize an event and ask the operator -- say, for instance, a loss of coolant accident. What he would expect to see and point out the relevant indications in the control room if he had a loss of coolant event. Then what automatic actions would take place, and to simulate a walk-through is required immediate actions.

Q Does the walk-through go beyond the immediate memorized actions into the written procedures which are not required to be memorized?

A Yes.

Q It does? It goes beyond that into the --

A Usually, once the operator has performed or told us what his immediate actions -- not told us, but showed us, what his immediate actions would be, we say well, what would you do next. And we would hope his response would be to get out the procedure to make sure he performed all of the immediate actions and then to find out what his subsequent actions would be.

Q And if that is his response, do you continue on with him having the procedure in hand, and walking through the remainder of it?

A Yes, mainly because it enables us to -- well, it's an examining trick, I'm not sure I should publicize it.

What it does is it simply leaves us the capability to have to memorize a procedure, so when the operator gets it out he'll look down at his immediate actions and say well, I figured out that one and that one, so he performs a self-critique for us. And then, we go over his subsequent actions to determine whether he knows them and what the reasons for them are.

P13 flws.

Q You mentioned that you start the sequence by postulating an accident and telling the operator you have this kind of accident. Do you ever start the walk-through by saying -- by giving him indications, by giving him parameters but not telling him what the type of event is, and asking him to identify for you what sort of event he is experiencing?

A Yes, we have -- I have. I found it is not as effective as the other method, because generally once you give them the symptoms -- if you list the symptoms for him, it is perfectly clear to him what the accident is, while almost always we find it is a better test of knowledge if you ask him all the symptoms he expects to see and possibly what happens if he does not expect or does not get one of the expected symptoms.

By the first technique, you are in essence providing some knowledge for him, so it is better to ask him what his knowledge is.

Q In my experience in looking at Rancho Seco's procedure, one emergency procedure often refers to another, and it often does that based upon sort of an indication logic. If the indication says this, you go to that procedure. If it says something else, you go to another procedure.

In the course of doing your walk-through, do you pose to him situations that would require him to have several procedures added at the same time?

A Yes.

Q Do you ever conduct these examinations of an entire crew together, or do you conduct them as an individual?

A Always as an individual. When we conduct the examinations on a plant, when we conduct them on the simulator, they are generally as a crew.

Q Would the difference between conducting the test on a simulator versus conducting them at the plant correspond to the distinction between cold licensing and hot licensing, or does that correspond to something else?

A No, it corresponds to something else. Generally, the examinations we conduct on simulators in the recent past -- when I say recent past, for the last three years or so -- has been on instructors who are wishing to get a license to enhance their credibility as instructors. I am trying to remember your original question. It was why do we do it on simulators as a crew and individually on a plant?

Q No. I was just curious as to whether -- under what conditions, what kind of testing you conducted on the simulator versus what kind of testing you conducted in the plant. You have answered my question.

A Okay.

Q At the bottom of Page 1?, where you state, "On the basis of the examinations conducted to date at Rancho Seco, the NRC is satisfied that licensed personnel understand the emergency procedures." Once again, am I correct in assuming that you are referring to the licensing examinations?

A Yes, the NRC licensing examination.

Q On Page 14, you describe the licensee's requalification program. In preparing this answer, where did you obtain your information about the requalification program?

A From this file I was referring to previously (indicating). This is the question and answer on Page 14. Is that correct?

Q That is correct.

Is there any particular document in that file that you referred to to prepare this answer?

A Yes, this is Rancho Seco's Administrative Procedure AP-25, licensed NRC operator retraining.

Q Other than that, do you have any personal knowledge with respect to how these things were done?

A Excuse me. How what things are done? How they are tested on the knowledge of emergency procedures?

Q The various parts of the requalification procedure you are describing here.

A No, I have not observed them being performed at the plant. I am sure you are aware of one of the new licensing requirements or requalification programs will be that the NRC conduct the requalification exams.

Q When do you expect that to begin?

A The last projected date I saw, which is not an official estimate, as far as I know, is in about two years, and in order to do that, we will have to essentially double or triple our present staff.

Q On Page 15, the second set of question and answers, at the bottom, you describe -- you state that "Nearly all the emergency procedures have had some revisions in the last few months."

Are you familiar with the number of changes to emergency procedures at Rancho Seco in the recent past?

A In the recent past? This answer was based on -- we had received a revised set of emergency procedures from them. I can't remember when. It was either January or -- December or January. I believe it was January, and it was almost a total revision -- as I recall, every one of these 17 emergency procedures was revised.

Q Revised from when?

A I do not know from when. See, when we conduct the examinations at a facility, we ask them for their latest set of procedures, and they submit this to us. The previous revision dates of those procedures, I do not know. They can be all over the spectrum of dates.

Q When SMUD changes their emergency procedures, do they -- they are not required to inform the NRC as a general rule, are they?

A No.

Q How would you become aware of the changes in them -- SMUD's emergency procedures?

A The -- Well, for instance, to give you an example, I would not become aware of any changes they had made from the revised set that I received, let's say it was in January, unless I was going to give examinations out there again, in which case they would have to send me their latest revisions to the procedures.

Other than that, I would not know.

Q Prior to the set you received in January, what was the next previous time that you received a set of SMUD's procedures?

A That was back when we did the -- back in May or June of last year.

Q You received those in connection with your audit of SMUD's compliance with the May 7th order?

A Yes, but it was not all of their emergency procedures. It was only the ones required by the order.

Q The last sentence on Page 15, you state, on the basis of our review, we believe that the licensee has made significant improvements to the emergency procedures. In light of what you have just testified to, what is the nature of the review that you are describing here?

A Well, the changes that were described in the above paragraph, that their LOCA procedure was rewritten in accordance with B&W guidelines, that the loss of steam generator feed procedure was revised to include what actions to take for loss of all feedwater. The other procedures, emergency procedures, were included -- did include the new circuitry for the reactor trip on turbine trip. I guess what you are getting at is, how do we know these were not subsequently changed? How do we know these are the latest revised procedures?

Q That is not my question, but it is a good one. Why don't you go ahead and answer your own question?

(General laughter.)

A From their latest revised set as the one I received in January, no, I do not know.

Q Here is my question.

(General laughter.)

A I do not stop asking questions.

Q I am interested more in whether you -- you testified earlier that the staff does not really formally review SMUD's procedures, and yet here you said on the basis of your review you think that the licensee has made significant improvements. What that tells me is that you have received a copy of the procedures in January, and you looked at them, and in your mind they have made some improvements, but that that would be distinguished from a formal review of SMUD's procedures as part of your regulation of the licensee's activities in which you apply certain set criteria to determine whether the procedures are adequate.

That is kind of a long question, but with that preface, could you put your review in one of those two categories?

A Okay. Let's back up a second. This piece of testimony was written back in last year, and I was addressing primarily the procedures that were affected by the Commission order. They did include, like I say, all the procedures -- all the procedures were revised to include this reminder to check all the channels and so forth, which is a push to the operator to make sure things are going as he imagines his instruments are telling him.

Subsequent to these revisions, we have not formally reviewed the changes to SMUD's emergency procedures This testimony is based on the procedures we reviewed as a result of the Commission order. We would expect that they would not change in substance the information contained in the procedures that we required by the order.

If they did, we would expect to be notified of that, say, for example, by an I&E order.

Q When you say "change in substance" are you referring to changing the actions that the operator is directed to take as opposed to the format, the way the procedures are written, the amount of information conveyed, that type of thing?

A No, by "change in format" I would -- "change in substance" I would mean that if they went back and revised, for example, Procedure D.5 to perform some action that was contrary to the B&W guidelines, that would be what I would mean by a "change in substance." All facilities make routine changes in emergency procedures. If they required a different action to be taken by the operator that was not in conflict with the B&W guidelines, then we would not expect to see the change or required to be made aware of it.

Q So that the guidelines are what govern your participation in reviewing those procedures?

A Yes, sir.

end P13
Bob fol.

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tP-13

bfml

Q Directing your attention to page 19 of your testimony, the answer that begins at the bottom of that page and continues on to page 20 describes a number of subjects that were covered on the control room for Rancho Seco licensed operators.

First of all, these were covered in the, what I refer to as the May 7th order audit.

A Yes.

Q Which of these activities that are set forth here, these ten activities, would you have not expected Rancho Seco's operators to be able to do prior to the Three Mile Island Accident?

A The first verifying auxiliary feedwater flow, they would be able to do, however, they could not do it with as much confidence as they can do it now, because prior to TMI they did not have auxiliary flow indicators in the control room. Now, they do.

How to power the AFW pumps from the essential nuclear sercies buses. I cannot recall that. I know there was a requirement that they had before they could load one of the -- either of the motor driven pumps on the nuclear services buses.

So, I am unsure of number two. Number three was a result of TMI. They did not do that as far as I know in the past. Number four, of course, was not done before.

bfm2

number five was not pre-TMI. Number six --

Q Let me clarify my question before you go further. For example, number five, you are stating, I believe, whether it was required that they do this.

A Yes.

Q Before Three Mile Island?

A Yes.

Q My question is whether you would have expected that they would have been able to do this even though it was not required prior to TMI. Could you go through these items with that question in mind?

A Let me see. Okay. Yes, I would have expected them to be able to do it before Three Mile Island on number five, for example. They did not have a procedure that we requried, but I am sure the operators were aware of motor operated bypass valves that they could use.

Six and seven, of course, are self-explanatory. Those came as a result of Three Mile Island. Number eight, I assume they would be able to do that prior to TMI. Number nine was as a result of TMI. Number ten was a change -- design change or facility change that was post-TMI also.

Q I would like to ask you a couple of questions about the simulator. You stated that you observed the simulator training. Are you familiar with the capabilities of the B & W silulator?

A Basically, yes.

Q It is my understanding that the B & W simulator had to be modified in order to reproduce the Three Mile Island accident. Is that correct?

A That is correct.

Q It also had to be modified in order to simulate the Crystal River accident. Is that correct?

A That is correct.

Q Mr. Rodriguez testified that the B & W simulator was a physical model of the reactor and that it therefore could respond to most of the actions that might take place in a facility. Could you explain why, first of all if you agree with that statement, that it is that kind of physical modelling; then if you do, explain why the simulator had to be modified in order to reproduce those two accidents?

A Okay. Going to the first part of that, you said the simulator was a physical model of the reactor. It is not strictly true.

What I would say is the simulator is a model or a close representation of the Rancho Seco control room. The reactor and cooling systems are modifed by computer programs.

Typically in simulation, the equations used in modelling both the -- say for example, the kinetic behavior of the reactor and the hydraulic thermodynamic behavior of the reactor system and the steam generators they simplify to

bfm4

use the least amount of comupter time and computer memory. So, they have had fairly simple computers that are used in solving the equations for the given set of parameters in which the simulator is put into.

The Rancho -- excuse me. The B & W simulator was one of the first. The modelling techniques used in representing the primary and secondary system, the reactor core and so forth are fairly basic.

They did not include a computerized simulation of something, for example, like two phase conditions in any place in the primary system, except the pressurizer, the nodalization of the whole system, primary and secondary systems is very basic compared to most detailed calculations.

This is true for most simulators. I think in the later development of simulators in the last couple of years they are expanding the computer abilities -- capabilities and modelling techniques in them.

So, these are training tools. They are not essentially engineering or diagnostic tools for accident situations. So they do not simulate accident situations in all cases.

Most of them are models. Now, initially, I think, most of the computerized -- the computer modelling of the accidents was based on chapter 15 analysis in the FSARs which applies a great deal of conservatisms to begin with.

bfm5

It places most limiting conditions to present power or certain degraded flow situations to calculate worst-case conditions. So, the initial modelling of accidents was based on FSAR calculations rather than best estimate calculations.

Q Do you have a copy of NRC exhibit number 4, which is NUREG-0667?

A Yes. Now, I do.

Q I would like to refer you to page 569. At the bottom paragraph, it says "The disadvantages of the B & W simulator training are: one, age and fidelity of the simulator."

Is that what you were referring to, essentially?

A Yes. I say "age" because like I say, it was developed back in about 1969 or 1970, I believe. It was not the first. It was the second; I think Dresden was the first.

Q Do you think if you set out to build a simulator today that you could build one that had a substantially higher fidelity than the one that is presently at Lynchburg?

A Definitely. It just becomes the case of what is cost effective. You can get a CDC 7600 computer behind it and put all the fidelity you want into it, but nobody is willing to pay $30 or $40 million for a simulator.

Q In reading 0667, I got the impression that some of the later simulators were actually being constructed, or

bfm6

being thought about by utilities, might have a greater fidelity than B & W's. Do you think that is true?

A Yes, I think it is.

Q Are you familier with some of the more recent simulators that have been built in this country?

A More recent -- the latest ones I have been to, I think, are Sequoyah and Browns Ferry.

Q When were they instructed?

A Somewhere around 1976 or 1977.

Q Do you know what they cost?

A It depends on what you include in the cost, but basically -- roughly I think it was about $5 to $6 million.

Q Do you think that is a good ballpark figure for what it would cost for a similar simulator today?

A No.

Q What do you think would be a better figure?

A Some time last year, I cannot recall exactly when, I was over to the Singer-Link devision in Silver Spring, Maryland. They said that now it is just for the simulator alone, now.

It is about $8 million. From a utilities standpoint, they have to include the cost of instructors, maintenance, overhead, building to put it in, and so forth. So, it comes to quite a bit more than that.

Q What do you think the whole package would cost?

bfm7

A Salem -- excuse me. Public Service Gas of New Jersey just announced their plans to build a training center with a simulator. They were -- this is just hearsay between another person and myself when I was up there. They were talking about $20 million.

Q Did that include the building?

A I think so.

Q That included the trainers and personnel involved as well?

A Well, you have to -- I mean -- let's say from a utility standpoint, you have to put on the payroll people who will maintain the simulator. You have to constantly be debugging it, or troubleshooting problems.

Q But the $20 million figure includes those costs as well?

A I think so. I have not -- we don't get into cost effective studies of simulation.

(Pause.)

I would like to , just for the record -- I don't know if this is the forum for it but I wrote this particular section. Somehow, some words got misplaced. This is not for the sake of Rancho Seco. It is for the sake of Davis-Besse.

I did want to say in that particular paragraph --

Q Before you go on, you are referring to the paragraph in 0667?

bfm8

A Yes. On page 5-69.

Q Go on.

A Originally, that was structured to say: One, the age and fidelity of the simulator and, two, it may be counter productive for Davis-Besse operators.

I think any simulator training, even if it is not on a replica of a plant is productive. Although, in all cases, for instance, TMI they derived a great deal of benefit from simulator training.

Crystal River operators stated that their simulator training helped them very greatly during the event of February 26th.

Somebody in the translation dropped out the words "may be."

Q Would it be your opinion that simulator training is the most effective tool available to a utility today in teaching its operators how to respond to transient conditions?

A Definitely. It is much easier than the putting the plant through them.

Q I am also thinking of lectures and these types of things.

A No. It is far better to be able to demonstrated it or to -- either way. Talk about it first, then demonstrate it; or have them respond to it on the simulator, or have it the other way, depending upon time allocations. Have

bfm9

them respond to it on the simulator, then have them go back and talk about what they saw, what alternate sequences they could envision, what they could to about it.

Q That leaves me one more question on the simulator. In your understanding of the B & W program, does the trainer in the section -- the lecture section describe to the operators what they are about to simulate before they go into the simulator?

Do they talk about it beforehand and then go in and simulate it; or do the go in and simulate the situation and then come back and talk aboutit afterwards?

A Both. It depends on how the simulator time is allocated. This is not just true of B & W. It is true of most simulator training centers that I have been in contact with.

They will have two groups of operators or applicants there simultaneously. so, they might have the Rancho Seco people in so they get the simulator from 8:00 a.m. to 12:00 noon. They may have some other B & W plant, say Davis-Besse in and they get the simulator from noon to 4:00.

Then they shift to other utilities. So, you get four hours in the simulator and four hours in the classroom. The other utility is switching the other way.

Q If you had the afternoon session on the simulator, you would be discussing the events before you actually saw

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them simulated.

A It is possible. If they did not want to prepare the operators beforehand, then they may show it to them in the afternoon. When they come back the next day, talk about them.

Q I would like to ask you some questions on the testimony with respect to instrumentation for diagnosis and control of off-normal conditions.

My first question relates to the entire testimony. CEC Issue 5-3a discusses the ability -- discusses essentially -- I will read it. "Are the special features and instruments installed at Rancho Seco adequate to aid in diagnosis and control after an off-normal condition engendered by a loss of feedwater transient?"

Is it your understanding that this issue -- that it only goes to the instruments that are involved in responding to a loss of feedwater transient without complicating circumstances?

A No, I would include other circumstances, yes, but essentially it was a loss of feedwater transient that initiated it.

Q Could we refer to Page 3 of your testimony After discussing what might cause and what conditions might appear from a loss of feedwater transient, at the bottom of the page you respond to the question, "How is the plant designed to handle safely a loss of feedwater transient?"

You describe three systems, the ICS, the RPS, and the AFW system. Isn't it true that there are a number of instruments and controls that you have not discussed in your testimony that would be involved in the response to a loss of feedwater transient that degenerated to something more complex than that?

A Well, yes. If in fact there was a loss of feedwater transient and the auxiliary feedwater system did not respond, yes, but if part of the system responded and performed its function, it would not involve, for example, a high pressure injection system.

Q At the bottom of Page 4 appears the question, "As a result of the NRC review," et cetera. Have you identified any areas where there is insufficient instrumentation and capability to immediately retrieve necessary information during the loss of feedwater transient?"

You identified the auxiliary feedwater flow meter. I would like to ask you to address the same question, but to assume that not just a simple loss of feedwater, but some of the more likely degraded conditions that might result from that, loss of natural circulation, saturated conditions in the core, that sort of thing.

A What do you mean, more likely?

Q Let's begin with just those -- Let me give you three specific situations. The feed and bleed mode.

A This is assuming the loss of all feedwater.

Q This answer that you gave is assuming the loss of all feedwater. Is that what you are saying?

A That is what I think you are trying to go to.

Q Yes, feed and bleed would assume the loss of all feedwater.

A Yes.

Q Okay. But that is my point. What I am asking you is, a loss of feedwater transient can degrade into other off-normal conditions, and I believe that is within the scope of the issue, CEC 5-3a, and so I am asking you if you have identified any areas where in your mind there is insufficient instrumentation at Rancho Seco to respond to some of the off-normal conditions that might be engendered by a loss of feedwater transient, and to help you, I will give you two or three areas that I am interested in: feed and bleed mode; core cooling; saturated conditions in the core; and loss of natural circulation.

A All right. Those are the conditions. What is the question? What instrumentation is necessary?

Q Have you -- No. Have you identified any instrumentation that is not at Rancho Seco right now that you believe would be helpful or necessary in responding to any of those situations?

A No, I think they have sufficient instrumentation to respond to them, assuming the condition does not degrade further. For instance, you postulate loss of NNI.

Q In the last part of your answer, you say, if you had postulated loss of NNI. Are you saying that if there was a loss of NNI, that there would not be sufficient instrumentation?

A Depending upon what was the source of -- whether there was total loss of NNI, partial X or Y bus, or whatever, yes, it is possible we would not have sufficient instrumentation.

Q Do you believe it would be possible that the failure of one of the power supplies -- one of the power supplied buses to the NNI could lead to a situation for which there could not be adequate instrumentation?

A Well, you can postulate a number of things. So far we have gotten into loss of main feedwater, we got into loss of auxiliary feedwater. They must have had a LOCA some place. Otherwise, they would not get to two-phase conditions in the primary.

Q I am not assuming all of these conditions simultaneously. I am looking at the three of them individually. I am treating now -- Let's assume something similar to the "lightbulb incident" or Crystal River, where you have a loss of one of the power supply buses to the NNI, and not any of the other things unless they would result from that failure alone.

Do you feel that that event alone might create situations in which there would not be sufficient instrumentation?

A The problem they had at Crystal River was, they were not able to terminate HPI because they did not have the pressurizer level indication. All three of them came off into the same bus that was lost. I don't know what the scheme is at Rancho Seco, but if in fact all three pressurizer level instruments had the same bus, and this is one they lost, yes, there is insufficient instrumentation and they will end up doing the same thing Crystal River did, but it is not unsafe.

Q Why didn't you mention this saturation in your testimony?

A Why didn't I?

Q Yes.

A Why did I?

Q Why did you not mention it? Strike that.

My question -- I was -- You are familiar with the saturation meter, are you not?

A Yes. Not Rancho Seco's. It was not there when I was there last. I understand it was being installed during the present shutdown.

Q Do you believe it could be useful in responding to situations that might result from a loss of feedwater transient?

A Well, I have just said I am not familiar with what Rancho Seco has.

Q This is a generic matter. Saturation meters generically. Do you think that they are unnecessary instruments for responding to situations that might result from feedwater transients?

A Necessary, no. Helpful, yes. They are not safety graded presently, as far as I know.

Q Have you been through the Rancho Seco control room for the purpose of examining what the instrumentation is there for preparation of your testimony?

A I was through the Rancho Seco control room for the purpose of examining operators, and I was looking at the instrumentation as kind of an ancillary function to my examining process, in the back of my mind, trying to evaluate what instrumentation was available.

Q When was that?

A February, I believe.

Q Do you know where the temperature sensors for the saturation meter are?

A Where the sensors are?

Q Yes, where does it read t-hot?

A If it is reading the t-hot RTD, it comes off of the hot leg, from the vertical portion of the hot leg.

Q Is that your understanding of what it does read for the hottest RCS temperature?

A That is my understanding of what a t-hot sensor is. If you want the hottest point, you would take the incore thermocouples.

Q Do you know whether the saturation meter takes the incore thermocouples or whether it takes t-hot or --

A I don't believe so. Our recommendation in 0667, they have the capability to use the thermocouples in the saturation meter. Most of the plants I am aware of have been using the t-hot indication off the RTD's.

Q Could you refer to Page 5-64 of NUREG-0667?

There you will find the task force recommendations with respect to minimum set of parameters to enable an operator to assess plant status. If these recommendations were to be adopted, what is your understanding of what changes would be necessary at Rancho Seco?

A Shall I take them individually? I am not sure I can remember exactly the details of the Rancho Seco system, because I have been to two other power plants since then. After a while, they kind of all mesh together. But I know for one thing they do not have wide range hot-leg temperatures, or they did not when I was there, and secondly, all of these -- if these are going to meet safety grade criteria -- say, pressurizer level, the pressurizer level instruments at B&W are not safety grade, or make-up tank level.

I am not sure what they mean by wide range steam generator level. They have three different systems for monitoring that.

Source range and intermediate range should be safety grade, and BWST level is in the control room. It is a tech spec required instrument, so it must be. Core outlet temperatures of course, are not. They are thermocouples.

Q Have you completed your answer?

A Yes.

Q Aside from what is safety grade and what is not safety grade, you mentioned only the wide-range RCS temperature and t-hot, I believe. Are there others that would not satisfy these requirements even if these requirements only required that the indication be present in either control grade or safety grade?

A Could you repeat the question? I am not sure I follow you.

Q It seemed to me in your answer you mixed two things, whether the indication was there as described here at all, and then secondly whether it was safety grade, and you responded to a number of them with respect to whether they were safety grade. It was not clear to me whether they were there at all.

A I am sorry. What I was doing was going back to the -- the intention of this was to have this set of parameters completely independent, supposedly, from the present control -- NNI control system such that these will be available. I think that was the intention of this. I did not write this section, so I cannot speak for the author.

If your question is, what does Rancho Seco have now in the control room, assuming no failures, the only one I can identify offhand that would be wide range hot leg temperature.

MR. LEWIS: That is the only one you can identify that does not have, is that correct?

THE WITNESS: That I do not believe Rancho Seco has in the control room now.

BY MR. ELLISON: (Resuming)

Q In your other answer, you described which ones were not safety grade. Is that correct?

A The intent, I believe, of this particular recommendation is to have a safety grade set of minimum parameters, and this is the list they supposedly came up with. In order to make them safety grade, there are a lot of these that would have to be changed.

end 15

Q I would next like to refer you to your testimony on control room design.

Are you aware of the EPRI study that has been identified as CEC 33 in this proceeding, the study that was done in 1976 on several reactor control rooms, one of which was Rancho Seco?

A Yes, I have read it. When you say, am I familiar with it, I have not read it for the last couple of years.

Q In preparing your testimony, did you go through and review that document?

A No, I did not.

Q Are you aware of the -- First of all, were you aware when you read it that one of those plants was Rancho Seco?

A No.

Q There has been testimony in this proceeding that the Rancho Seco control room is quite small. You have been in a variety of control rooms, I presume. Are you aware of any that are smaller than Rancho Seco's?

A As an absolute answer, I would have to say I do not know, when you take total square foot area. Now, the Rancho Seco control room shift supervisor's office is considered at the control so it is normally considered part of the control room, but for the purpose of my answer, I will not assume that is part of it nor anything to the right of the door where you go through to the access to the RPS and radiation monitoring cabinets. Offhand, I would say it is -- the only one I can compare with it -- I -- Like I said in my testimony, out of the 35 I have been in, I think it is the smallest.

Q Referring to Page 4 of your testimony at the top of that page, in your first answer, you say, "A formal human factors engineering test and evaluation" --Is that correct?

A Yes. That was defined in the previous page, on Page 3. Human factors engineering test and evaluation.

Q "Would have to be performed at Rancho Seco for an accurate comparison." Do you think it would be a good idea to do such a study at Rancho Seco?

A I think it is already a requirement, as part of the master action plan. We had subcontracted -- contracted out to the same firm that did this study of the TMI control room to develop criteria and guidelines for human factors judgment of control rooms.

Q Is it your understanding that SMUD is under some regulatory requirement to do such a study at the present time?

A As far as I know, it is not a regulatory requirement. We were discussing before the status of the master action plan. It is Revision 5 or whatever that supposedly is final, and the Commissioners have not adopted it yet. I know that is one of the recommendations.

Q In the next sentence, "I believe such a study would show Rancho Seco to be far superior." Do you believe there should be such a study?

A Yes, because the question was, how does Rancho Seco control room compare with TMI 2, and I think it is far superior to TMI 2. I have been in TMI 2 a number of times. Rancho Seco does have the control room design -- the control room design does violate certain human factors engineering test and evaluation principles. I say that from the basis -- Dr. Alan Swain, I think, was the one who did that EPRI study, and I took a course in human factors engineering from him in 1976, and on the basis of that experience, and my involvement in the control rooms, I would say that Rancho Seco is a superior control room, but it does violate some human factors principles.

Q Which principles?

A One would be the height of some of the cages is above the normal -- I guess above the fifth percentile person. They are above his normal level of vision, and therefore would tend to provide an erroneous reading, possibly provide an erroneous reading. They do not use minutes except, as I remember, just on the electrical board.

Mimics is a basic principle for providing an aid to the operator. All of their controls, indicators, and so forth are not laid out in a functional sequence. Many of them are, but not all of them.

Then, when you get into human capabilities in terms of reach, it would be hard for me to make a judgment on that particular aspect. There are certain other human factors principles, ability to see, like the height of the operator: if he was standing behind the normal console, can he see all of the indications on the back panel? And I doubt if the normal person could.

Whether they were all necessary and relevant, I do not know.

Q One of the human factors engineering principles, as I understand it, is that controls should be located in functional groups. How near -- How broad an area would you consider to be an effective grouping for controls that have to be operated simultaneously, or for controls of related indication?

A Well, with the requirement we have two operators in the control room, I would say, within shouting distance.

Q As long as they are in shouting distance, it does not matter--

A As I said before, I already said I recognized some of the controls and instruments at Rancho Seco are not arranged in functional groups, but as compared with some other facilities, they are much better.

Q Are you aware of the location of the auxiliary feedwater flow meter at Rancho Seco?

A Yes.

Q And are you aware of the location of the auxiliary feedwater Bailey control valves, and the remaining AFW controls?

A I know where the motor operated SFAS valves -- the bypass valves on the auxiliary feedwater system are located. That is on the back safety panel. The Bailey controllers, I think, are located on the front panel, as I remember, and yes, you cannot read flow from the bench board, from the auxiliary flow indicators accurately. They can see if it is in a specific band, but if you ask them to read out what is the flow rate, they have to go to the back panel to read it.

Q Inasmuch as Rancho Seco has one of the smallest control rooms, is it your feeling that there is sufficient space in the control room to add some of the indiciation that we have discussed in NUREG-0667? And do that consistently with human factors principles?

A Again, you are asking for an opinion.

Q That is correct.

A Yes, in my opinion, I can.

(Pause.)

Q I guess I will not ask you about the color and decor of the control room, which has been discussed in this proceeding.

That is all I have.

MRS. BOWERS: Do you have some questions?

BOARD EXAMINATION

BY DR. COLE:

Q Mr. Wilson, the control room design testimony, Board Question 31, the Board question was framed a little differently than as is stated on Page 2. Are you aware of that, sir?

A No, I am not.

Q I will read to you Board Question 31, and I think you have responded to that, but I want to give you the opportunity to add anything if you so desire.

The Board question HC-31, "Are there features of Rancho Seco's control room design and configuration which make it difficult for operators to avoid a loss of feedwater transient?"

Now, what we mean there is not necessarily to avoid a loss of feedwater transient, but to respond to a loss of feedwater transient is what we meant, though nobody wrote it that way. As I indicated, I think you have responded to that, but in view of that question phrased that way, would you like to add anything at this time?

A I think one of the outstanding issues on that, in addition to what I said in the testimony, I think the control room design and the instrumentation is adequate to respond to a loss of feedwater transient. One of the outstanding issues is whether a vessel level indication would be required.

I can only offer two things: one, my opinion that I do not think it is required, and secondly, in just reading the issue -- the latest issue of Nuclear Engineering International, the Swedish regulatory agency or whoever regulates atomic power in Sweden feels the same way. All of the changes they made,apparently they did not think it was necessary to have a vessel level indicator.

That is all I have to add to what I had in the testimony.

Q All right. On Page 5 of the same testimony, in the last sentence, you allude to thepossibility of supplemental testimony. I assume that that is not planned by you, sir.

A No, I did not have any more supplemental testimony. What I have added so far is that I acknowledge that I have more familiarity with the Rancho Seco control room now as a result of the examinations than most of my testimony -- it was based on my knowledge of the B&W simulator, which essentially is a mock-up of the Rancho Seco control room, and I do acknowledge that, as I have said previously.

There are human factors principles that are violated more than I first thought of in the control room, but I think once we do make these evaluations that are required by the master action plan, evaluate the control rooms in accordance with these developed criteria and guidelines, we will find that Rancho Seco would be superior to most if not all of the control rooms.

Q On Page 18 of your testimony, on operator training and competence, the question at the bottom half of the page and the answer at the bottom half of the page, you summarize the history and results of licensed operator testing by the NRC at the Rancho Seco facility.

Other than the experience that you describe in testing operators as related to TMI 2 incidents, have you observed other Rancho Seco operators?

A Observed or examined?

Q Under examination, or observed them in the control room. And what has been your experience with the operators that might assist you in making an evaluation of them?

A I examined three Rancho Seco employees -- applicants in February, and one instructor who was a General Physics Corporation employee. Two of the three Rancho Seco operators passed as reactor operators, which they applied for, and the General Physics employee passed as a senior operator. The one operator who failed, failed the written examination under the new grading criteria. He would not have failed under our old grading criteria.

I also have observed two -- we were talking about before, there were two licensed senior operators, and one applicant about two weeks later on the B&W simulator.

end16
Bob fol

tP-17
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tP-16

Q You were not involved at all in examining the 18 that originally applied for senior operator license?

A No, sir. I believe that was prior to when I joined the NRC or AEC, then. Let me rephrase that. I think there were two sets of examinations given.

First they were examined on the simulator only as a test of the simulator's ability in the cold licensing program. They were not -- no records were kept of them on a pass/fail basis individually.

As I said in the testimony, the result -- maybe it was in interrogatory -- the results of those examinations -- on the basis of those examinations, AEC then approved the B & W simulator as a training facility.

The cold examinations, I cannot remember when they were given, '73 or '74.

Q I am trying to determine whether you have had enough experience with enough Rancho Seco operators and with operators of other plants to permit your professional evaluation as to how the Rancho Seco operators stack up against other operators of other plants.

Can you do that, sir? If you cannot that is fine.

A On a statistical basis, I have examined by now three -- on a currently licensed basis, they have 16 people assigned to shift duty. I have licensed two of them.

I have given examinations to every other operator

bfm2

at most B & W facilities and CE plants that I can think of. Literally hundres of examinations. I've been doing it for six and a half years.

I think, from that I have seen of the Rancho Seco operators, on the basis of the four that I examined, including the general physics employee, of serving the operators on the simulator and looking at the results of their examination process, the cold and hot license training programs and the requalification programs, they stack pu very favorably with other operators in training programs that I have experience with.

DR. COLE: I have no further questions.

BY MRS. BOWERS:

Q I am still having a problem understanding just exactly what the simulator does. Now, its computer makes certain lights flas, or lights go out that should be on to give a message to the operators. Is that right?

A The simulator essentially mimics -- reproduces the indications in the Rancho Seco control room. They will inciate to the operator on the basis of what is performed at the instructor's console or what is performed on the machine itself, the evolution what they are trying to perform or abnormal situation.

The simulator is basically designed to train the operators in normal plant operation. It does respond well

bfm3

to many abnormal situations or to emergency conditions.

Q In a problem area, an emergency condition, there is a time frame here when certain things have to be done or you are in trouble. It is also keyed to that?

A The simulator?

Q Yes.

A Yes, that is one of the reasons the programming of the simulator is fairly basic, so it can use a fairly simple computer and perform the calculations in real time.

If they wanted to do very detailed engineering calculations, for instance, as they do for the loss of coolant accident analyses, these calculations are so involved they cannot be performed on a small computer with real time.

It takes a significatn amount of time, so you would not get the feedback. The operator would see the response of the machine to tweek a knob or change a set point and so forth.

So, the calculations that are performed by the computer are fed back in the time frame in which they would expect to have them happen at the plant.

Q I believe you mentioned that you observed a crew at the B & W simulator at Rancho Seco. You talked about one person. Was that in the oral or written exam, or was it in the control room?

A There are two different instances. I said earlier

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that we normally examine operators on the simulator in a crew type situation with a shift supervisor, senior operator, and reactor operator, or just two people essentially which is the minimum required now.

I did not examine the Rancho Seco operators at B & W. I observed what they were taught and shown and performed as part of the requal. It was not an examination of those operators.

When we examine an operator at a plant, when the plant is either cold shutdown, hot shutdown, or operating, it is always on a one on one basis. We never examine a crew on a plant at the same time.

Q But does any part of that examination take place in the control room?

A The examination -- the normal licensing examination?

Q The one on one.

A Yes, normally the oral examination -- there are two parts to the exam: the written -- assuming he is going for a reactor operator's license. There is a written exam he must take and the oral examination.

The oral examination will generally take about four hours and at least two hours -- usually about two hours are spent in the control room.

Q I have asked earlier witnesses about the problem of operators keeping their cool and being inflappable in a stressful situation. Is there anything presently required

bfm5

suggested or coming up to kind of focus on this problem in trying to evaluate operators?

A No. Well, as part of that I can say we do have -- they must be medically qualified, but we do not accept on a simulator test time in terms of how they respond under a stress situation.

Even that is not a true stress situation because they know it is a simulator and not a real plant. I can only go on the basis of the TMI event, which was a true accident; and the Browns Ferry incident, which was very close to one.

The response of the operators in the control room at the time, I think in looking at the TMI events, the one thing that contincually amazes me was how the operators did respond responsively.

They made mistakes, granted. They contributed to the severity of the accident, but I think most operators at a power plant would agree that once they saw indication -- I recall the testimony -- the depositions of one of the operators in the control room.

He said he looked up at one point and the radiation monitoring panel at TMI-2 is directly in front of the main board, main control board.

He looked up and saw every monitor, alarm high, red light on. That just would have scared the hell out of me. The second thing they saw was the source range nuclear

bfm6

instrumentation increasing despite their efforts to add boron. That similarly would have scared the heck out of me.

Those operators -- there is not one mentioned that they even considered bailing out in that situation. Browns Ferry was a similar situation. They lost just about everything.

They had juryrigged a situation to keep the core covered. They responded under a very stressful situation and brought the plant to a safe shutdown condition.

MR. BAXTER: Excuse me, Mrs. Bowers. We have been discussing operator depositions in this proceeding. Mr. Wilson, are you referring to TMI-2 operator depositions before the Kemeny Commission?

THE WITNESS: Yes.

BY MRS. BOWERS: (Resuming)

Q Has there been any that you know of, any planning or programming for utilities to try to share simulator time. Now, for instance, Diablo Canyon is Weshinghouse. We learned recently that they apparently -- well, somebody said -- testified that they now have a simulator. Would there be any benefit at all for Rancho Seco operators to spend any time on that?

A Well, again, this is an opinion. Since it is a Westinghouse plant, I would say no.

bfm7

Q Well, we also heard testimony from Mr. Rodriguez that the distance and the time from here to Lynchburg, Virginia is a bit of a problem. Also there are some people who do not care to fly. So, I was just trying to think in terms of -- for instance, around the Chicago area, you know, there are a number of plants other than Davis-Besse which, of course, has been talked about a lot here being B & W. I am not sure of the vendor on some of the others.

The close proximity should lend itself to some sort of in-time working out agreements.

A I do not think it owuld benefit the operators. Again, in my opinion to train on a simulator with a different NSSS design.

The operator licenseing branch has allowed start-up certifications of Westinghouse and CE plants to be performed interchangeably.

In other words, I can think of one just recently I was at. It was a Westinghouse plant. They performed the start=up certification on a combustion engineering simulator and talking to the applicants.

They almost unanimously voiced sissatisfaction. They said they really did not receive that much benefit from the training.

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P18 Lupton
foll Bob

Q When you go to a facility like Rancho Seco, and you mentioned being there in February to give tests to several people, are they aware that you are coming at that particular time?

A Oh, yes. The schedule is set up a month or two ahead of time.

BY MR. SHON:

Q I have a couple of questions. Your discipline is close to my heart. I was the first chief of what is now the Operator Licensing Branch when it came into existence 20 years ago or so. Incidentally, we used to ask every pressurized water reactor operator, how do you know you are not growing a bubble in the core. He would trot you over to the temperature and the pressure and that sort of thing --

A I do not ask the same particular question, but one question I have asked recently is to take a look at t-cold and t-hot, and I will say -- B&W, for instance, they will say it is supposed to be about 555; t-hot is supposed to be 603. I will say, the surge line comes off the hot leg, so that should be 603 or a little less. It shows a fundamental misunderstanding of the facts.

Q Yes. He doesn't know what is supposed to be pushing on what.

One thing I would like to do is just briefly read to you the latest version of Board Question HC-32, which you had answered in your testimony at Page 16. In your testimony, entitled Testimony on Operator Training and Competence, we had a little different form of the question. I think you have certainly answered our question, but I want to give you an opportunity to add anything to it you would like to.

As we had stated it in our order, it reads, "What procedures have been used to test and evaluate the competence of Rancho Seco's operating personnel and management?" Do you have anything you would like to add to it, stated in that form?

A I am not responsible for testing the management so I can only talk about the licensed personnel. We do have procedures, our branch, how to test and evaluate operators and senior operators. We have a set of guidelines. Essentially that is what -- how we are supposed to conduct the examinations, written and oral.

Q They are essentially the things you have been telling us about, the kind of examinations you have been giving, the questions you have been using, the reject questions you have been using, and that sort of thing.

A That is a very brief summary of what we ask, yes.

Q In your testimony on control room design at Page 5 you mention during the week of February 10 you would be conducting reactor operator examinations at Rancho Seco, and you will evaluate the control room configuration. You said a little bit about that, but I do not think you gave us a really complete yes or no on what conclusion you reached at that point.

A Well, in terms of the conclusion, do you mean, are they able to respond to a loss of feedwater transient?

Q Yes, on the ability of the reactor operator applicant to respond to a loss of feedwater transient.

A Yes. I can remember -- like I say -- there were four applicants and at least two -- we posed different transients to different applicants to try and get a broad -- touch all our bases with as many different transients as we can. I know at least two I posed a loss of feedwater event to them. We did talk about loss of all feedwater, sustained loss of all feedwater. The basis for the reactor coolant pump trip requirements, and I think two or three of them we even -- we did go through the complete loss of NNI procedure which was prior to Crystal River.

Q This is directed towards the other side of the coin, the other aspect of your testimony, the control room design. What did you find out when you asked these people to respond or pretend to respond to a feedwater -- loss of feedwater transient from the standpoint of how to control them is designed?

A There were two -- Basically my conclusion was the control room, as I said previously, does violate some human factors principles, but I think it would compare very favorably on an evaluated basis with all other control rooms.

There were two things I realized in the Rancho Seco control room -- was that the feedwater systems were not -- auxiliary feedwater systems and controllers were not really functionally grouped like you would expect the two pumps to be side by side and the valves and so forth, in a mimic type arrangement.

Secondly, the auxiliary feedwater flow indications, as I said earlier,cannot be read accurately from the main board. You have to go -- There are small gauges as compared with the normal Bailey gauges that are difficult to read from a distance. They can see -- at least the operators I have had can see where the indicator is supposed to point on a normal -- if the system is responding normally, but they cannot read it accurately.

Q I see. That is the auxiliary feedwater flow indicators, you say?

A Yes.

Q I understand that they are not extremely accurate indicators anyway, are they?

A I do not have any experience with that type of detector. It is an ultrasonic detector. Most of the B&W plants have gone to that design because otherwise they would have to cut the pipes to put in a restricting orifice or some other -- in order to measure DP, so they put this type of gauge on it. I don't know about the accuracy of it.

Q At any rate, is it that parameter upon which the operator would be controlling, or is it --

A No.

Q Go ahead.

A That is only used as a verification. You would not control that parameter by any means. First of all, they would allow the ICS to respond automatically if it is capable of doing it, and as previously testified, it would respond to maintain levels of flood level limits, about two feet, two and a half feet on the start-up range, on loss of main feed pumps, loss of the four reactor coolant pumps.

If the ICS did not respond properly and the operator had to take manual control and control it in accordance with the procedures that they developed in response to the order, it would be controlling on steam generator level.

Q So that is what he would really be watching. Can he see that gauge clearly from the control point?

A I believe the operating range is on a recorder, and I can't remember if that is on the front panel or back panel, but I think they can see that fairly accurately, yes.

Q There were a few other pieces of instrumentation that we have heard recommended as valuable to respond to LOCA's feedwater transients, and so on, that I would like to ask you specifically about. You have already answered on one. That is primary level indication. You said you felt it unnecessary. Is that right?

A Yes.

Q We have also heard it would be good to have something of the nature of a void meter that might not actually register level, but it would show the amount or the location of voids in the primary. Do you think such a thing is necessary?

A I heard that suggested. I do not know the mechanics of what would be involved in trying to come up with such a meter because of the primary system design, where they would try to locate where the voids were, whether they were in the piping, the hot and cold legs, the upper part of the vessel, or whatever.

I think the present criteria we have is sufficient in that if the operator knows he is at least 50 degrees subcooled and he has a level indication and a pressurizer and he knows where his level is, he knows where all the water is and where the voids are. If he does not have the 50 degrees subcooling and his incore thermocouples are not abnormally high and he knows -- or he can assume he may have a void in the primary system, but that the core is still covered -- so he does have to use a variety of instrumentation and make an assumption from that instrumentation.

Q We have also heard that the TSAT meter should really be upgraded to safety grade. How important do you think that is?

A One of the recommendations we have in 0667 is to have the capability of using the thermocouples as an input to the TSAT meter, and if that is the case, I cannot see how it would be made safety grade. If they only use the t-hot indication of the hot legs and the pressurizer pressure -- I am sorry. B&W does not measure pressurizer pressure -- primary system pressure. They are both safety grade, and they can be made to a safety grade TSAT meter.

Q You would have to make a little compensating and computing portion of it, safety grade, I suppose, too.

A Yes. You cannot rely on the normal computer.

Q All right.

Lastly, we have heard that they should have positive indication of flow under natural circulation conditions. That is some sort of flow meter that reads with reasonable accuracy down to the tiny fraction of the normal flow. Do you think that is enough of a valuable thing so that they really should have that?

A I do not know. I have not thought too much about flow indication. I really could not answer that.

Q You noted yourself that a couple of the operators at one point seemed not to realize that if it were flowing fast enough it would have a smaller delta-t than if it were flowing slow, and faced with that sort of thing, where the interpretation looks a bit ambiguous to many people, would that not be a nice thing to have?

A Yes, it would. There are a lot of things that would be nice to have, too, but we still want to maintain the control room simple. The natural circulation flow is dependent on a lot of variables: time after shut-down; how much decayed heat is left in the core; what the levels are in the steam generators; whether or not they have blocked flow in one steam generator.

So, it is conceivable if you have the natural circulation flow meters in one loop you will only get indication in one loop. If flow is blocked in one, and it would be hard to surmise from that situation what your present status is, whether you assume a failed meter or -- I am not sure what kind of -- I am not that familiar with that type of instrumentation, what type of instrument you would use to measure flow.

Without enormous quantities of water, it is such a slow flow rate.

MR. SHON: Thank you. I have no further questions.

MR. LEWIS: Mrs. Bowers, may I suggest we consider that we try to finish with this witness this evening? It may not be that much longer.

MRS. BOWERS: Let's check. How many, Mr. Lewis, additional questions do you have -- I am sorry. Mr. Black?

MR. BLACK: I only have several. I do not expect it would take me more than several minutes.

MRS. BOWERS: Mr. Baxter?

MR. BAXTER: Two or three.

MRS. BOWERS: Mr. Ellison?

MR. ELLISON: Two or three.

MRS. BOWERS: I don't trust lawyers. They all say two or three. Do you want to go ahead then -- Do you want to take a few minute break here?

REDIRECT EXAMINATION

BY MR. BLACK:

Q Mr. Wilson, I want to refer you to your testimony that deals with emergency procedures, and my question is simply, is it your opinion or is it the opinion of the NRC staff that when emergency procedures require an operator to refer to ore than one procedure, do you or does the staff feel that that is too demanding on an operator in an emergency situation?

A No. Typically the case would be, if there were an emergency situation -- let's take the minimum number of required licensed persons in the control room, right now being two, so you have two operators,or possibly a senior operator and an operator in the control room, and an event happens. Let's assume the event is normally -- You have to respond to it quickly. It would be a reactor trip. That is usually the first thing they will see. The operators are going to respond to that situation and attempt to handle the plant as best they can. At least this is my experience. And bring it to a stable situation before they would even refer to a procedure, if they have enough man-power in the control room at the time, or they have gotten to a relatively stable condition with the plant, they will get out procedures as quickly as they can to try and follow it based on what they assume the situation to be.

Now, as -- at Crystal River, using an example, I do not know how many procedures they had out at the time of the loss of NNI incident. They did not have a procedure that particularly addressed that situation. The INPO/NSAC report has identified 13 procedures that during the cou. of this incident they should have referred to, and I imagine they probably were looking back and forth between 5 and 6 or 7 procedures at the same time.

The bottom line, I assume, is, if they do not find a procedure that addresses the situation, they are going to resort to their training, which would be typical in any emergency situation for any man-machine problem. In the case of Crystal River, what they did was, they would go to the most conservative situation, which was a loss of coolant. This was the worst situation they could possibly get into.

The procedures will reference one another, and they say if this happens, go to this, or if this, go to that, and once you get into that procedure, there may be steps that are just not applicable, and you just have to skip them. So the procedure is not absolutely binding. They have to follow every step in it. They will follow the procedures to the best of their ability, but by and large they will respond to their training more than a given set of procedures.

Q When you say respond to their training, what do you feel is the most important aspect of that training, or do you feel it is the combination of all of them?

A I think the most important aspect of the training is that they are able to put the plant in a safe shutdown condition, regardless of the situation.

Q Well, I meant -- Excuse me. I meant, is it the simulator training, is it the classroom training, is it the real-life experience training, or -- I believe Mr. Ellison asked you a question, if you felt that the simulation training was the most important, and I do not remember what your answer was, so I am rephrasing that question and asking you again.

A I don't remember if it was asked -- if that was the most important thing -- I certainly feel it is extremely beneficial because you cannot perform those evolutions on a plant that you can on a simulator, but real-life situation training, I think, is the best teacher. I would trust an operator with five years' experience with a shift supervisor more than I would a person who just came out of the simulator training program. Not trust, but I assume he would have more knowledge of the plant and how to control it.

Q Is it your testimony that the requalification program has been modified as a result of the TMI experience and studies? Is that one of the things that you indicated, that it was modified as a result of TMI?

A Rancho Seco's?

Q All licensees' requalification programs.

A All of them will have to be modified, yes. Whether -- the specific recommendations at this time, I do not know.

Q I might have gotten this wrong, so someone correct me if I am wrong, but I think one of the questions Mr. Ellison asked you is that he knows areas that the requalification program was modified -- Now, that would be the fluid flow and thermodynamics changes and whatever that they will be required to add to the requalification program -- He asked you a question, is it the NRC staff's opinion that prior to these modifications, whether the -- whether the operators were adequately trained in these areas that were subsequently modified, and I did not hear your response to that.

MR. BLACK: Is that the correct characterization of your question?

MR. ELLISON: That is close enough.

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Bob fol.

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BY MR. BLACK (Resuming)

Q In those areas that have been modified in the requalification program, does the NRC staff feel that the operators were adequately train prior to those modifications in those areas?

A First of all, as far as I know, there have been no modifications to the requalification programs. We have published -- not published, but the March 29th letter identified what changes will have to be made. These changes have not been made, yet.

Q Well -- do change

A To change the requalification program, I would imagine the individual utilities will wait until either, one, there is a specific deadline that they have to meet or, two, that we have got all out changes that we want to be made and then they will submit them all at one time because it involves licensing fees and so forth.

It just does not make any sense to change a couple of words in a program; then the NRC is going to charge them a licensing fee for that. They will wait.

Q Now, can you try to respond to my question? I realize now that the program has not been changed, but those areas that have been recommended for change.

Is it your opinion that the operators were inadequately trained in those areas that are being suggested for

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change?

A We have identified -- when you say "the staff's opinion" I guess I would say it is operator licensing branch's opinion that one of the recommendations we made that they were not adequately trained on in the past is specific accident sequences that they should see or experience in the simularors.

This was a deficiency in the requal training, but we had made recommended changes on that in terms of the total requalification program -- I say the program, not the training -- I guess the biggest significant change will be the NRC conducting the examinations.

Q Does that response mean that your branch feels that operators were inadequately trained prior to these suggested modifications?

A No. If we felt they were inadequately trained, we would have brought this up as a safety issue and required the plants to be shut down.

Q One further line of questioning. You were asked to refer to SMUD exhibit 20. On that exhibit, there is a category five, which indicated a -- well -- let me back track a little bit.

You were asked to compare a ratio of personnel errors to total LERs. When you made that comparison, I believe you indicated that Rancho Seco had probably had the

bfm3

highest ratio when you were asked to compare those two numbers.

My question is, when you're looking at a given facility to determine whether their operating personnel are inadequately trained or not doing their procedures right, or what have you, would you look at this specific statistic that you were asked to compare?

Would that be a valid indication of personnel incompetence, let's say?

A No. As I said before, that covers a ten year period. The LERs attributed to licensed personnel have only been identified since January 1978. So, while this goes to December 31st, '79; so that is essentially two out of ten years that were identified as licensed personnel as compared with all other plant personnel.

Q Would it be more meaningful to look at just a raw figure of total personnel errors rather than looking at the ratio of personnel errors to total LERs, if one were looking at personnel competency?

A I would not think so. What I found in the section on operator qualifications in NUREG-0667 was essentially that you can do whatever you want with numbers. You can manipulate them any way you want to to prove your facts.

You can prove your fact and I can prove mine. It is not necessarily true to make a general conclusion from

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those statements.

MR. BLACK: I have no further questions -- I take that back.

(Pause.)

BY MR. BLACK: (Resuming)

Q One further question. Mr. Wilson, are you aware whether Rancho Seco or SMUD has committed to change its requalification program?

A Not that I am aware of.

Q Let me just show you this letter. It is a letter dated September 21, 1979. I ask you if you have seen that letter. If you have, whether that would refresh your memory?

(Counsel handing document to witness.)

(Witness reviewing document.)

A Yes.

Q What is that? Can youidentify the letter please?

A Yes. This had slipped my mind. This is part of the long-term actions. One of them was to upgrade the training programs.

It included the requalification program. It says the District's administrative procedure AP-25, licensed NRC operator retraining has been upgraded and now requires TMI-2 incident and lessons in from that incident to be subject of regulato operator training lecture series.

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Q Who is the letter from and who is it addressed to?

A This is from John J. Mattimoe, Assistant General Manager and Chief Engineer, and addressed to Mr. D. F. Ross, Jr., Director of the Bulletins and Orders Task Force of the NRC.

(Pause.)

MR. BLACK: No further questions.

MRS. BOWERS: Mr. Baxter?

RECROSS EXAMINATION

BY MR. BAXTER:

Q Forgive me for returning to the numbers game for just a second. You testified earlier that there may be some uncertainty or some degree of arbitrariness about the classifying a licensee or vendor report as personnel caused.

Given that and the other accumulation of such data, if a given facility had a smaller overall number of licensee event reports on an average annual basis or a comparatively small number, would there be any concern in your mind as to the safe operation of that plant because of the ratio of personnel caused error -- that it was somewhat higher than for other plants?

A No, not in my mind.

Q Mr. Ellison asked you about some of the new requirements in CEC Exhibit 49, the March 29, 1980 letter, the letter Denton, that are going to be imposed to upgrade requalification training in terms of whether it increases the time of training.

As a general matter, does the NRC express its requirements for the requalification training programs in terms of hours or more in terms of the depth and scop of the subjects covered and the result you desire to see achieved in terms of operator knowledge?

A Only in depth and scope; we make no requirements in terms of time. The only numbers I can recall are the grades on the different examinations.

Q Mr. Ellison also asked you whether in simulator training the operators get the opportunity or are asked to take over the plant in a degraded condition without being told how it got there and work from there.

You indicated you did not believe so; would you expect that in real life a control room watch would be completely turned over to a new set of operators in a degraded condition?

A No.

MR. BAXTER: Those are all my questions.

MRS. BOWERS: Mr. Ellison?

BY MR. ELLISON:

Q In response to Mr. Shon's question about a vessel level indication, you said it was your opinion it was unnecessary. Recognizing that in saturated conditions we

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have had testimony that there is no reliable source of vessel level indication, could you explain the basis for your opinion then, that operators would not be benefited or it would be unnecessary for them to have that indication?

A Yes. My basis is that I can envision a vessel level indication system that would, for instance, be reading out as a meter or a recorder in the control room, and assuming the plant is operating along hopefully as a utility does for quite a bit of time, like a year.

The operators would be coming on shift and typically looking at an offscale indication vessel level. It would be more than likely pegged high.

And if there were such an incident in which the level came back onscale, I think they would be more prone to disbelieve the instrument rather than -- what I am saying is it is a conditioned response. You can look at a particular indication for months and months at a time and it is always reading the same thing or it goes offscale. And once it does become useful -- maybe once in every hundred reactor years or thousand reactor years of operation -- how much validity would the operators attach to its reading?

I think the criteria that they have subcooling and a pressurizer level indication, particularly when we have -- if they do upgrade it to safety grade Tsat meters -- was more than sufficient to prove that he has sufficient water inventory in the power system.

Q Is the Tsat meter going to be of any assistance to an operator if the primary system -- I mean -- is it going to be of assistance to the operator in determining the primary system level, if there are saturated conditions in the core?

A No. His indications at that point, that he has the core covered, would be his thermocouple readings.

Q Thermocouples are not safety grade, are they?

A True.

Q The thermcouple reading would tell you whether the core was covered or not, but it would not tell you what the level was, isn't that true?

A I'm sorry; can you repeat that?

Q Other than determining whether a thermocouple itself was covered or not, could you determine level from reading the thermocouples?

A No.

MRS. BOWERS: You were shaking your head no, is that correct?

THE WITNESS: No, no, you could not determine level. You would know that your core is covered.

But I think it is the feeling of most people in the industry and most people I have talked to that we are not going to have another incident of this type with the amount

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training and emphasis we have been putting on this particular aspect of saturated conditions, loss of coolant accidents.

I cannot envision operators not responding to a situation where they had a loss of inventory in the primary system and throttling back on the safeguards systems causing this particular incident.

end 19

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BY MR. ELLISON:

Q With respect to the question of LER's and the numbers game, would you expect that the older plants would have fewer LER's? Would you expect that?

A Generally speaking, yes.

Q Would that apply to personnel related LER's or only to mechanical failure type LER's?

A Generally speaking, again, it would apply more to mechanical failures, because depending on the turnover rate of the people at the facility, as the people get more and more experience, they will be less and less contributive to personnel error.

Q I am sorry. I got confused in your answer. Are you saying it would apply, but that principle, that the plant as the plant gets older -- Pardon me. Let me finish. You should always know what you are responding to. That that principle would apply to personnel related LER's?

A I said the principle that there would be fewer personnel error LER's as the plant got older, depending on the turnover rate of the personnel. As I said, we found that most of the personnel errors are attributed to unlicensed personnel, so roughly 20 percent or so are to licensed personnel.

Q If I were to postulate to you two plants who have roughly equivalent numbers of LER's and one of them had a very high percent of personnel-related LER's, rather than mechanical LER's, if you were concerned about the operations of the facility as opposed to its mechanics or its design, do you feel it would be valid based on those numbers to pay more attention to the plant with a higher percentage of personnel-related LER's?

A There is one conclusion you can draw from that. I guess as an illustration there was an organization that investigated LER's and they said that -- some of the conclusions were, roughly 20 percent or so of the LER's that are attributed overall to personnel error, and the rest of them to instrument malfunctions or instrument drift, and so forth, if they apply their criteria, they could reclassify a lot of LER's as personnel error.

For instance, one of their criteria was instrument drift, which is currently classified as an instrumentation LER. That should be a personnel LER, because if you know the instrument is going to drift, then you should increase the surveillance such that we catch it outside of its normal parameters on a more frequent basis before it reaches the set point, not the set point, but the point which is outside the limits, one must report it as an LER, so they are saying they raised the number of personnel error or the ratio of personnel LER's from 20 to 50 percent under their criteria.

That is theirs, not necessarily everyone else's. It is a very inexact science in how you attribute an LER to either personnel error or mechanical or instrument or whatever.

MR. SHON: Mr. Wilson, I guess you could say if something fell apart it was a personnel error on the part of the designer.

THE WITNESS: That is true. I think I remember a fish kill case,there was a plant on the eastern seaboard, and they exceeded their environmental tech spec limits on the number of fish they killed due to a thermal grading in the water, and somehow they attributed it to personnel error.

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Q Do you know whether the annunciators in the Rancho Seco control room distinguish between significant failures and insignificant failures or whether they're all indication for annunciation basically the same regardless of what the initiating event would be?

A I'd say some of both. You may have an annunciators that would alarm if the temperature on the component cooling water, for example, went outside the limits, it's hot, which is not an unusual situation. Or you may have an annunciator that says "reactor trip" and the unit just shuts down.

Q Would the signals in the control room appear the same rather than the label on the annunciator? I mean, would the sound be the same, would the light be the same color, the same size?

A I think the lights are the same; I don't recall the sounds. But that's only for Rancho Seco; it's specific for each plant.

MR. ELLISON: That's all.

MRS. BOWERS: The Board has no further questions. Mr. Black?

MR. LEWIS: We have no further questions.

Can Mr. Wilson be excused?

MRS. BOWERS: Any objections?

MR. BAXTER: No objection.

MR. ELLISON: No objection.

MRS. BOWERS: Mr. Wilson is excused, thank you.

(The witness was excused.)

MR. LEWIS: Mrs. Bowers, there is one scheduling consideration we have with respect to one of our witnesses and that is Mr.Morrill from Region V of I&E. He has testimony that's somewhat unto itself, and he would be able to sponsor that testimony without being part of a panel. He's presently on active duty for two weeks with the Navy,so what he needed to do was he needed to have a time specific when he could start. I told him I couldn't give him time specific when he would finish. And if it's agreeable to the parties, that would be Wednesday morning, and if it involved interrupting something, I would propose that we do that.

MR. BAXTER: That's fine with us.

MR. ELLISON: That's fine.

MRS. BOWERS: Fine. You mentioned this Saturday morning, his scheduling problems. We'll convene at 9:00 tomorrow morning.

(Whereupon, at 5:30 p.m. the hearing in the above-entitled matter recessed, to reconvene at 9:00 a.m. the following day.)

This is to certify that the attached proceedings before the

NUCLEAR REGULATORY COMMISSION

in the matter of: SMUD (Rancho Seco)

Date of Proceeding: 5/12/80

Docket Number: 50-312

Place of Proceeding: Sacramento, CA

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

David S. Parker

Official Reporter (Typed)

Official Reporter (Signature)

This is to certify that the attached proceedings before the

NUCLEAR REGULATORY COMMISSION

in the matter of: RANCHO SECO

Date of Proceeding: Monday, May 12, 1980

Docket Number: 50-312

Place of Proceeding: SACRAMENTO, CA.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

SUZANNE R. BABINEAU

Official Reporter (Typed)

Suzanne R Babineau

Official Reporter (Signature)