UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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In the Matter of: : Docket No. 50-312

SACRAMENTO MUNICIPAL UTILITY DISTRICT

(RANCHO SECO)

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

Tuesday, May 6, 1980

The above-entitled matter came on for hearing, pursuant to notice, at 9:35 a.m.

BEFORE:

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

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INDEX

WITNESS	DIRECT	CROSS	REDIRECT	RECROSS	BOARD DIRECT	BOARD CROSS
J. F. Meyer		2799			2854	2888
T. A. Greene		2799			2854	2888
B. J. Mann	2924	2926	2942		2933	
R. Rodriguez	2947	2949				

EXHIBIT	DESCRIPTION	IDENTIFIED	RECEIVED
CEC 33	The Human Factors Review of Nuclear Power Plant Control and Design	2967	

Afternoon Session: Page 2888

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P R O C E E D I N G S

CHAIRMAN BOWERS: On the record.

We will continue from April the 17th with Dr. Meyer and Mr. Greene as witnesses, and they, of course, have been previously sworn.

MR. LEWIS: My recollection is that we broke off in the midst of cross examination by Mr. Ellison, so I presume we resume there.

Whereupon,

JAMES F. MEYER and THOMAS A. GREENE,

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

CONTINUED CROSS EXAMINATION

BY MR. ELLISON:

Q Can you hear me, Mr. Greene?

A (Witness Greene) Yes, sir. Can you hear me?

Q Yes, I can hear you just fine. If you have trouble understanding me, just let me know.

I would like you, if you would, to refer to Page 4 of your testimony, Mr. Greene.

MR. LEWIS: Which item of testimony would this be? Mr. Greene has two pieces of testimony.

MR. ELLISON: This is on the CEC Issue 5-2.

BY MR. ELLISON: (Resuming)

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Q In response to Question 6, you described the design basis accident for the containment building, and you distinguished it from the design basis accident that is used in the design of the emergency core cooling system. At the bottom of the second full paragraph of your response, you state that in the design basis accident for the containment building, the reactor core fuel temperature remains very low and core degradation is unlikely.

Do I understand your statement to say that assuming the conditions in the design basis accident for the containment building, core degradation is unlikely as distinguished, from, say, that were you to get core degradation from some other sequence of events, that it would not exceed the pressures on the design basis accident?

A What I tried to do in my response was to emphasize that in the containment design basis accident, that there is really a different accident scenario, where you try to release large amounts of energy to the containment atmosphere to get a maximum temperature and pressure within the containment building for the bas c design. In the ECCS analysis, you really have a different accident scenario.

That is, different assumptions are made, and the assumptions retain the energy in the core for ECCS analysis. And hence in the containment analysis you really don't talk about core melt or high flow temperatures.

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Q Isn't it true, however, that neither of the design basis accidents that you are referring to assume a significant amount of fuel failure or core melt?

A Right, yes.

Q Isn't it true that there are sequences of events that are conceivable that do lead to a significant amount of fuel failure and core melt?

A There are sequences of events that can lead to core melt, but these are not considered in the licensing process.

Q And would it not also be possible that those sequences that lead to significant fuel failure or core melt could generate pressures and temperatures beyond those of the design basis accidents that you are referring to?

A Yes.

Q You also state in that sentence that in the design basis accident for the containment, that core degradation is unlikely. Is it impossible?

A As I stated previously, the accident scenario is such that we are attempting to remove energy from the core to design the container building, and hence the assumptions in everything we make is such that the core is cool -- to accept that the -- temperatures remain very low -- it is just hard to talk about core melt in the containment analysis.

Q In addition to core melt type accidents, isn't it true that there are also other possible sequences that lead to pressures and temperatures beyond those of the containment building design basis accident?

A There is an accident -- There are other accident scenarios. If you go further than -- one was assuming the analysis, like a single failure, if you assume loss of all your heating capability, you could get pressures higher than the containment design.

Q Can you think of any other sequences other than the loss of the heat removal capability and core melt that might lead to pressures and temperatures beyond the design basis?

A Are you talking about -- I thought your question was, without considering core melt.

Q I am. We have identified core melt as one possible sequence. Now you have just identified another one, which is the design basis accident plus a failure of heat removal systems. My question is, are there any others in addition to those two that you are aware of that might lead to pressures and temperatures beyond the design basis?

A Not that I am aware of.

Q Is it possible that a failure of the steam generator inside containment could result in all the energy of the steam generators being released to the containment building. It could overpressurize the containment.

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A Are you talking about a steam generator, a main steam -- steam line pipe?

Q Perhaps.

A Okay. That accident scenario is considered during the containment design, and that accident results in lower pressure than the loss of coolant accident.

Q Is it your testimony that there are no accidents involving the release of energy from the steam generators inside containment that either have not been analyzed or have been analyzed and result in higher pressures and temperatures than the design basis accident?

A We have considered both the spectrum of loss of coolant accidents and the main steam line break and feedwater break accidents, and the design basis accident which results in the highest pressure is the double line rupture of the hot leg.

Q I understand that you have considered the main steam li e rupture. My question is whether you have consider d the -- done an analysis similar to that for the core melt -- excuse me, the release of energy from the core, and by that, I mean, have you considered the scenario where the maximum release of energy from the steam generators is released into the containment building?

A Yes, but it is limited to a single failure. When we do our accident scenario, we do not go further than a

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single failure.

Q So you don't know what the possible pressures and temperatures would be from accidents involving the steam generator if they involve more than a single failure. Is that correct?

A Yes.

Q Turning to Page 5, the second paragraph, you state that although the containment building design basis accident does not include considerations for core degradation or core melt, two of the engineered safety feature systems do, and you go on to describe how the containment building spray injects sodium hydroxide to accelerate removal of aerosol fission products, and some of the assumptions that go into the design of the spray system.

Isn't it true, however, that neither of these systems are designed to ensure that the energy that would be released from a core melt or Class 9 type accident would not result in overpressurization?

A I think what you are trying to say is that the system is qualified to the maximum temperature and pressure inside the container, and hence we have no insurance that they will operate beyond that, and you are right.

Q So it would be fair to say that although you have assumed a certain amount of fuel failure or core melt in designing these systems, that you have not -- I shouldn't say you, but the systems have not been designed to completely mitigate a core melt. Is that correct?

A They haven't been designed to function above the -- or you have no assurance that they will function above the maximum temperature and pressure that they are designed to. They could mitigate the core melt.

Q When you use the word "mitigate," first of all, you are referring to reducing the pressures in the containment building rather than in the pressure vessel. Is that correct?

A Well,we are talking about the core sprays, I believe, and the core sprays are only used to reduce pressure.

Q Reduce pressure in the containment building?

A Containment building. Yes.

Q And when you use that word "mitigate," you also mean to -- am I correct in my understanding that you mean to lessen the pressures that would result from that accident, but not to mean to ensure that a core melt would not overpressurize the containment? Is that correct?

A Yes, but the original question was that, would the spray help mitigate a core melt, and I said it would help by reducing the pressure.

Q My question is not whether it would help mitigate. My question is whether these systems would ensure containment

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integrity in the presence of a core melt, and I understand your answer to that question to be no. Is that correct?

A Yes. We have no assurance that given a core melt and the container pressure and temperature exceeds the system design, they would function.

Q And system design at Rancho Seco is 59 psig? Is that corr t?

A The containment is designed to 59 psig.

Q At the latter half of Page 5, the last full paragraph, you describe how the combustible gas control system assumes that the emergency core cooling system is in a degraded but not totally failed condition, and that there has been a certain amount of metal water reaction in the core.

First of all, could you describe for me more precisely what you mean by the combustible gas control system?

A The combustible gas control system is an engineering and safety feature system which is required by our regulations to control the hydrogen concentration inside containment.

Q And how does it do that?

A The system for Rancho Seco consists of a purge system -- it is called a hydrogen purge system -- in which the containment atmosphere is taken from the

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containment and filtered through filters and released to the atmosphere.

Q Am I correct in my understanding that the purpose of this system is not to relieve pressure or temperature within containment but to remove hydrogen?

A Yes.

Q What percentage of fuel failure is assumed in the design of that system?

A I stated in my testimony that we assume five times the amount calculated by the ECCS analysis.

Q My question is -- I am sorry, did you complete your answer?

A Yes.

Q My question is in terms of a percentage of the fuel in the core, what would that translate to?

MR. SHON: Mr. Ellison, you asked in terms of the percentage of the fuel in the core. I think you meant in terms of the percentage of the zirconium in the core which reacted, didn't you?

MR. ELLISON: That is correct.

WITNESS GREENE: The numbers are based upon 5 percent, basically 5 percent of the zirconium in the core.

BY MR. ELLISON: (Resuming)

Q The numbers you are referring to are the --

A The numbers used in the design of the combustible gas control system for Rancho Seco was based upon 5 percent of the zirconium in the core.

Q So the numbers used in the design of the ECCS system would be 1 percent? Is that approximately correct?

A I do not know what was used by ECCS.

I will qualify that. The applicant can use various numbers in the design, and what he has chosen to do for Rancho Seco is just take five times the amount of zirconium in the core.

DR. COLE: I didn't understand your answer there, Mr. Greene. You said he took five times the amount of the zirconium in the core?

WITNESS GREENE: Well, I guess I am wrong there. In the FSAR, the applicant presented two analyses. One was used on a certain percentage of the core reactor. The ECCS analysis. And another was using Reg Guide 1.7. And we looked at both -- We looked at the analysis using 1.7, and in that analysis they used 5 percent of the zirconium in the core.

BY MR. ELLISON: (Resuming)

Q Referring to Page 7 of your testimony, the response to Question 8, you state that the present range of Seco containment design is adequate. Could you define for me what your criteria for adequacy are?

A What I was referring to when I said adequate was adequate to withstand the design basis accident for Rancho Seco, which is a loss of coolant accident that does not result in core melt.

Q Further down on the page, at the end of that first paragraph, in your response to Question 8, you state, "It should be pointed out that the containment is capable of withstanding pressure in excess of 59 psig before containment integrity is lost."

Are you aware of any analysis either by yourself or someone else at NRC or someone at SMUD, for that matter, how far beyond psig the containment integrity would be maintained?

A Yes. First of all, the containment, after it is built, they perform a structural test in which the containment is pressurized to 115 percent of design, and then there are two studies that I am aware of that were done. One was done by the structural branch of NRC, which showed -- or the result was that the containment could stand approximately twice the design.

And then there is a study that was done by the structural branch consultants, Ames Laboratory, that also showed that the containment could stand approximately twice the design number.

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Q Apart from the 115 percent over-pressurization test that you mentioned, were either one of the other two performed on the Rancho Seco containment building?

A No. They were done for McGuire and Sequoyah.

Q Do you recall what those two studies -- whether there were any loadings on the building aside from the pressure from within?

A No, I am not familiar with the details of the study.

Q So, you do not know, for example, whether they assume any wind loadings?

A No.

Q Would it be fair to say that although it is likely that the Rancho Seco containment building can withstand pressures beyond 50 psig, that there is a possibility of failure as one goes beyond that figure?

A As the pressure increases, the probability that the containment will fail increases.

Q So, would it be your testimony then that there is a spectrum, if you will, of increasing probabilities of containment failure that begins at the design basis of 59 and extends up to a point where you would be certain that containment would fail?

A I don't know if I would be certain. All I am fm2

saying is that the containment is designed and built to 59 psig. As you exceed that number, the probability of containment failure increases. As you get higher and higher pressure you approach the u-point.

Eventually, the containment will fail. I have no idea what that number is, though.

(Pause.)

Q Mr. Greene, throughout this examination and I believe in your testimony, we have been discussing containment failure from over-pressurization.

Is it not also true that the containment could fail because of seal failures resulting from high temperatures?

A Yes.

Q Do you know what the design temperature limits of the Rancho Seco containment building are?

A I believe it is 286 degrees f.

(Pause.)

Q Dr. Meyer, I would like to address the subsequent questions to your testimony. At the bottom of page 2, you describe the capabilities of the controlled filter venting system. You state that whatever the final choice of systems, the filter vented containment system will result in considerable reduction in societal risk relative to an uncontrolled unfiltered containment failure.

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Above that, you describe some of the attenuation factors for radionuclides, such as iodine might result from a filter gap and release from containment. Is it not also true that the control filter venting system will provide a substantial additional amount of evacuation time?

A (Witness Meyer) There will be a delay in the release for certain of the accident sequences being considered.

For some of the filter vented containment system schemes that are proposed, there would be considerable increase in the times for evacuation. That is correct, but they are dependent upon a specific system. They are also dependent upon the particular accident sequence that you are analyzing.

Q Assuming that one were analyzing the PWR-3 and BWR-3 sequences that were studied in the underground siting study -- pardon me. You are familiar with that study, I assume?

A (Nods in the affirmative).

MR. STEPHENS: For the record, please speak your answer.

WITNESS MEYER: I am familiar with the study.

BY MR. ELLISON: (Resuming)

Q Assuming one were considering the accidents that were considered there, would it be fair to say that one

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could expect a substantial additional amount of evacuation time?

A The two you are referring to are over-pressurization type of containment failures. In those cases, for example some of the analyses being conducted on other reactor plants, there is a substantial benefit in terms of evacuation time.

I do not recall offhand how many hours this buys you, but it is factored into consequence analyses that are normally performed in considering these various filtered vented containment system schemes.

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bgn tP-2

Q Turning to page 3 of your testimony, in response to question five, you describe how one would set the release point for a controlled filtered venting system. Further on in your testimony at the top of page 6, you say that basically the technologies are in place to do the job required to design such a system.

Putting those two statements together, is it your opinion that it is technologically feasible to design a release point as you described on page 3 of your testimony?

A Yes, if you would provide me, for example, with the design bases and the design criteria. I have not seen any evidence that if you are willing to spend the money, you could not build a filter vented containment system, but the important point is the design bases.

The design bases depend upon dominant accident

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accident sequences for a particular reactor plant.

Q Is it your feeling -- you stated in a particular reactor plant -- it is your feeling that those accidents might vary from one plant to another?

A Yes, definitely. From one PWR to another, the major contributors to the overall risk, the major acciden sequences would probably vary.

Q In designing a system such as we are discussing, there are a lot of site-specific factors involved. Is that correct?

A Site-specific, but also characteristics peculiar to the reactor facility itself.

Q Both in the design of the facility and the site where it is located?

A Yes, I was thinking specifically of reactor characteristics themselves, but there are also site characteristics.

(Pause.)

Q What are some of the reactor design characteristics that you would look at?

A For example, in the WASH-1400 PWR analysis, the dominant sequence was the feedwater transient with loss of all AC power, both on-site and off-site. If you would apply the same type of risk analysis to another facility that has a much lower probability for loss of emergency AC, then that particular sequence may not be dominating and a major contributor to the overall risk.

Then, perhaps another sequence that was better protected in the WASH-1400 PWR, but not as well protected in the reactor under investigation may turn out to dominate the total risk.

Q Are you aware of any analysis at Rancho Seco along the lines you are describing?

A Not at Rancho Seco.

(Pause.)

Q You stated, I believe, you could design a system to do the job depending on if you were willing to spend the money. Onpage 6 of your testimony, you note that -- you say it should be pointed out that some of the sophisticated systems are very expensive.

Have you analyzed the cost of designing a system that would do the job at Rancho Seco?

A I have not analyzed the cost of a system for Rancho Seco. NRC is presently involved in doing that type of analysis. However, in conjunction with the design in the Indian Point study that I believe you are familiar with --

Q I gather that you are also familiar with that study, is that correct?

A That is correct.

Q What were the cost figures involved in that study?

A The costs are a function, basically, of two items. One is the length you want to go to to remove the harmful fission products. Of course, being more expensive, the more you require the removal of such radioisotopes as the noble gasses and organic iodine.

It is also very much a function of the design bases that I referred to earlier. For example, you may want to have a system that can be controlled automatically, manually, and have certain passive features.

It may be required to vent large quantities of gasses. Other systems may be required to vent much smaller volumes of gasses. The costs, of course, are proportionate to the volume of gasses required.

The costs presently range anywhere from $15 million to $50 million, bu this is a preliminary estimate where, like I mentioned, the cost may change depending how the requirements of the specific system.

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Q I would like you to assume a system such as that described in the underground siting study, that is, a passive rupture disc type system without a system to remove noble gases and without a system to remove organic iodine. The underground siting study gives a rough cost figure of about $14 million to apply that to a new facility. You mentioned a range of $15 to $50 million for a variety of types of systems. Do you have an idea what that type of system would cost?

A I would have to have a clarification of that. The underground study had no -- in my recollection of the underground study, there was no way to retrofit the design to an as-built plan. Are you referring to the underground study system per se, or to some adaptation of that system to Rancho Seco?

Q Well, let me ask the question both ways. First of all, the figures that you gave of $15 to $50 million, were you referring to a retrofit to an as-built plan?

A That's correct, yes.

Q It is my understanding that the underground siting study figures are for application to a new facility, so -- inasmuch as we are speaking here today of possible application of such a system to an as-built facility, my question would be, assuming you were to apply the system I described earlier from the underground siting study to an as-built

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facility, would you believe that it would fall towards the $15 million range or towards the $50 million?

A Certainly lifting the requirments on hold-up or attenuation of the noble gases and the organic iodine would reduce the costs, but I couldn't speculate beyond that until I had a good understanding of what the dominant accident sequences were, so that estimates of the volume, for example, of gases that have to be vented were better known. It is very difficult to make any more definitive statement than that.

Q Were you assuming in the range of figures you gave earlier, however, that the more expensive systems did have noble gas removal capability, organic iodine removal capability, that sort of thing?

A That is correct, but those studies also take the Indian Point, for example, site into consideration, and I have no knowledge of the Rancho Seco site. Perhaps there would be complications there that would be major in terms of, for example, installing a very large volume suppression pool or a gravel pit, so you are correct, but again, it was specific to the Indian Point site.

Q Am I correct in assuming that you have not compared the Indian Point site layout? I presume, first of all -- strike that. When you refer to the site layout, you are talking about the location of the various parts of the

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facility itself. Is that correct?

A That's correct. Just the space available.

Q Okay. Am I correct in assuming that you have not compared the layout of the Indian Point site to the layout of the Rancho Seco site?

A That's correct.

Q Have you visited the Rancho Seco site?

A No, I haven't.

Q Assuming that one did not design the system to remove noble gases, and that they were released, is it your opinion that they would pass through the filter in such a way as to be released in a dispersed fashion, or do you think they would be released in a concentrated fashion?

A Well, again, it would depend on the specific design that you are talking about. If you had a high stack as the release point, you would have different characteristics about -- you would have different characteristics regarding the spreading out of the noble gases off-site than you would if you had a different venting scheme. That would be a function of the holdup capability that you would get as a result of having a large gravel pit, for example, and a number of other factors.

So again, it is design specific, and I can't comment much further than that.

Q Okay. Just for clarification, unless I state

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otherwise, in all my subsequent questions, I am envisioning a control filter venting system basically like that described in the underground siting study.

A That would be a stack release.

Q Okay. Is it your opinion that that type of system would provide some hold up capability?

A The hold-up capability, certainly there is some hold-up capability. I can't say much beyond that. It would be something that would have to be carefully looked at. But it is very difficult to estimate until one knows what the actual gravel pit looks like and the specific design.

Q The underground siting study looked at that, did it not?

A I am not aware of their doing a quantitative analysis of the noble gas hold-up time in their particular design. Perhaps they did.

Q In your cost figures for retrofitting Indian Point, were you assuming the use of an existing penetration?

A Yes, we are assuming the existence of existing penetrations.

Q Are you aware of which penetrations you are assuming?

A I believe -- sometimes I get the Indian Point mixed up with the Zion facilities, but I think for Indian Point it is a three-foot diameter penetration.

Q Do you recall what that penetration was there for before it was used for this system? I mean, it has obviously not been applied yet, but what its intended design was?

A No, I am not aware of that.

Q How about the penetration at Zion?

A I believe that there is a similar penetration at Zion.

Q Do you have reason to believe that you couldn't use an existing penetration at Rancho Seco?

A Certainly one that would -- that one would want to take a very close look at would be the, I believe, 66-inch purge penetration, perhaps accommodating the filter vented containment system to that penetration. It ce-cainly would be something to look at, but I couldn't say anything beyond that.

(Pause.)

Q Okay. Referring again to Page 6 of your testimony, Dr. Meyer, at the close of your answer to Question Number 7, you describe certain open questions with regard to control filter venting systems. The first one you described is interference with other engineering safety features. Could you describe in more detail what you are referring to here?

A Yes. There are several engineered safety features whose operation might be compromised by a situation

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in which you would have a drop in the containment pressure due, for example, to a filtered vented release as compared to the situation in which the containment without a filtered vent system would have a very high back pressure.

One example is the ECCS system during the part of the double ended pipe rupture accident sequence when you are having a reflooding after the core, after the core has been essentially dried out. The heat transfer coefficient in the core is a function of the pressure, the back pressure in the containment, and it increases as the containment pressure increases. This aids in heat transfer of core heat to the coolant during the reflood.

Also, the steam binding in the remaining portions of the primary loop is less severe the higher the containment back pressure. These two things combined make it more attractive to have a high containment back pressure during reflood than not, so there is this possible situation that you would have a vented filtered system, say, a ruptured disc that would drop the pressure in the containment, thus not allowing your ECCS system to work as efficiently as it might otherwise.

There is another aspect to it.

MR. SHON: Dr. Meyer?

WITNESS MEYER: Yes?

MR. SHON: Before you leave that particular aspect

-- the desirability of a high containment pressure during the reflood phase -- would you under any circumstances be likely to design the filtered containment venting system to operate and release pressure at this point? In most of the accident sequences that you study, is this the point where this thing would operate, or would it operate far further down the line after the reflood phase had either succeeded or failed?

WITNESS MEYER: Well, again, if it depends on the accident sequence that we are talking about. There are situations where containment failure or in this particular case the use of a filtered vented system that would take you down to atmospheric pressure, that that event would in fact cause a core melt, and it would cause a core melt because it would compromise the effectiveness of either the ECCS system or other engineered safety features to operate properly.

So, you could conceive of a situation that there would be relatively high containment pressures and have no core melt up to that point, and --

MR. SHON: But during the reflood phase? It just seems to me that the time scale is out of line here, that reflood would -- the one that you specifically mentioned would surely occur in any sequence you could conceive of before the design release point of the filtered venting

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system. Isn't this true?

WITNESS MEYER: Yes. Well, it depends on what you are assuming for the release point. For a double ended pipe rupture, the release of the energy in the primary system does raise the containment pressure substantially.

MR. SHON: Surely, but it does not raise it to the point where the containment is in danger of failing. In fact, that is exactly the way it is designed, isn't it?

WITNESS MEYER: That's correct. You could have two situations, however, one being that your pressure relief point for your system is below that design pressure, and the second situation could be, since you put in a new system, that it could fail prematurely, but you are quite correct that if your set point is very high and the systems work is designed, that situation would not arise.

MR. SHON: Thank you, Mr. Ellison. Sorry to have interrupted. I believe you were going to go on, Dr. Meyer, and explain another possible sequence in which an interference might occur.

WITNESS MEYER: Another problem situation is pump cavitation that might result by a depressurization of the containment, where in the recirculation mode you may again damage pumps used for the emergency core cooling system, or you may damage your containment spray pumps.

Another problem that is being considered is that

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you could, for example, have a high pressure situation vent most of the non-condensibles out of the containment. Your containment would vent down to a lower pressure, and your engineered safety feature spray system could come on then and pull a vacuum in your system, and this has to be considered also.

Those are just some areas that we are addressing that are of concern in regard to interference with engineered safety features.

BY MR. ELLISON: (Resuming)

Q Each of the problems that you mention, Dr. Meyer, are the result of the depressurization of containment. Is that correct?

A Yes, they are all related to the depressurization.

Q Isn't it true that if containment were to fail without a control filtered venting system, that a containment would depressurize and cause the same problems?

A That is correct, but there are a family of accident scenarios where you would be required to vent but that would not otherwise have failed the containment, and it is that family of accident scenarios that are of concern.

Another aspect to that question is, some of the analyses that are being conducted on the Zion and Indian Point containments indicate that the failure may be partially

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self-sealing, where the concrete would crack open, vent, and then partially self-seal as the pressure dropped in the containment. This may not be as bad a situation as if you had venting down to atmospheric pressure from, say, a rupture disc type arrangement.

Q Are you confident that if containment were over-pressurized, that it would fail in the way you describe it?

A As I mentioned, we are conducting studies in the area, and as the studies proceed, our competence in any particular failure mode, of course, increases. Right now, the preliminary analysis seems to indicate this type of failure for some of the sequences under consideration.

Q For some of the sequences? Is that correct?

A That's correct. There are --

Q But not for all the sequences?

A No. We consider a full spectrum of sequences, including a rather aggressive hydrogen burn sequence that may have a different failure mode than some of the slow pressurization from steam sequences.

Q A moment ago, you mentioned that there were some family of accidents for which you would design the control filtered venting system to actuate even though there was no possibility of containment failure. Is that correct? Did I understand that answer?

A I didn't say it in quite that way. No probability of containment failure. We certainly do not know enough detail on any reactor to accurately present all the accident sequences, their probabilities, and the consequences either with or without any particular filtered venting containment system scheme.

There do exist, however, a family of accidents that with the present containment system would not fail the containment, but would require a venting in case of a system like the one described in the underground study. Now, how big that family is and how significant in terms of the overall risk is an open question.

Q Assuming that you have -- First of all, in the answer, are you assuming that you have set the set point for the filter vented release system above the design basis of containment?

A It would be appropriate for that assumption, yes.

Q So what you are saying is, there is a family of accidents where pressures are generated, pressures and temperatures are generated beyond the design of the containment building, but for which the containment building will not fail. Is that correct?

A May not fail, again depending on the conclusions drawn in the final analyses being conducted on these two containment buildings. But yes, basically that is a correct statement.

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Q Then do I understand that you are testifying that there are certain types of accidents where you are certain that the containment building can withstand pressures and temperatures beyond its design basis?

A It would depend on your understanding and my understanding of the word "certainty." I would agree with Mr. Greene's comment earlier that as you go up in pressure above the design basis pressure, you increase the probability of containment failure. The evidence that I have seen, however, is that that probability remains quite low until you start getting into the pressure areas of about 100 psig. And then depending again on the loading history that you are assuming, the probability for failure increases rather dramatically.

Q Wouldn't those considerations be involved in setting the -- in the design of the control filtered venting system and particularly the choosing of the set point?

A One of the problems, at least with the Zion and Indian Point study, that is complicating considerably that question is that for some of the accident sequences that we are considering, there is a large pressure spike that comes along with the molten core coming in contact with the accumulative water. This pressure spike has been estimated to rise up to about 120 psig.

It is a considerable complication because the

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spike rises so rapidly that the penetrations to the containment would have to be very, very large in order to accommodate that if you have a high pressure point, high pressure set point for activation of the system. Therefore, one of the considerations is to lower that set point considerably in anticipation of that pressure spike occuring later in the accident sequence.

If, for example, you would have that type of accident sequence with a very high pressure set point, let's say, 85, 90 psi, then it probably -- that system probably would not be able to accommodate that particular accident.

end2A&B

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Q Assuming for the moment that you have chosen a set point as you have described well below the spike point on the order of 60, 70 psig, let's say, and returning again to your testimony with respect to the family of accidents that would reach that set point without presenting any possibility of containment failure, I believe you testified that depending on how you define "certain," that you are certain the containment would not fail in those situations. Is that correct?

A I would have an awful lot of confidence that it wouldn't fail, yes.

Q Have you performed any analysis of the Rancho Seco containment building to determine whether those accidents would present no possibility of containment failure?

A I have not performed any analyses. My comments were in reference to the analyses presently being conducted for the Indian Point containment and the Zion containment.

Q Isn't it true that -- Well, are you assuming in this confidence that the conservatisms in the design basis of the containment allow you to exceed that design basis before it fails?

A Yes.

Q Isn't it true that none of those conservatisms have assumed the accident sequences that you are describing?

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A Could you rephrase that question?

Q Those conservatisms were not placed in the calculations for the design of the facility in order to accommodate the accident sequence you are discussing. Isn't that true?

A That's correct.

Q Is it your belief that those conservatisms -- Strike that.

If one were to assume that those conservatisms which were included in the calculations to account for other things are necessary in order to account for those other things, wouldn't it be fair to say that additional loadings beyond those assumed have not been considered in the design of the containment building?

A The approach to the design basis accident -- well, I was going to say, is different from the approach, for example, to the type of accidents that we are considering now, where there is core degradation and core melt. However, there has been no established approach to how to handle the core melt and core degradation accidents and their impact on the reactor facility.

For the design basis accidents, the conservative approach is the approach that has been adapted by NRC and the industry for years, and it is a fortunate fall-out of that approach that there is margin built into containments that

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allows one to make the statement that the realistic failure pressures are considerably above those in the design basis.

Q Is it your testimony, then, that the assumptions that are used in the design,that the NRC has required to be used in the design of containment buildings such as Rancho Seco's are unrealistic?

A My position is that they are conservative. The design basis accidents are analyzed in a conservative fashion. In that sense, you might say they are not best estimate, but they are appropriately conservative as part of the whole philosophy of defense in depth and appropriate conservativisms that are part of structures in general.

A (Witness Greene) Could I just qualify that a little bit? In the design basis accident which determined the design pressure in the containment, there is conservatism in that number, in that we take a conservative approach, but once the pressure inside containment is determined -- for example, in Rancho Seco, it was 52 -- then they build the containment according to ASME codes, all right, and that code also has, I believe, conservatism in it. For example, bridges and buildings, they all are built according to codes and some design number.

So, when you see a load on a bridge, maximum allowable load, so many pounds, you know that trucks go over it that are higher than that, and that is because the codes allow a little conservatism.

Q Isn't it true, though, that those conservatisms are intended t account for uncertainties in the calculations or uncertainties in the actual as-built materials used in consti tion or uncertainties in the actual methods of construction and for loadings that are postulated to be possible at the time of the accident sequences that we are talking about?

A Is that addressed to me?

Q That is addressed to either of you.

A (Witness Meyer) I am not very familiar with how the codes are established and the reasons behind their conservative approaches. Certainly a portion of that would be to take into account unexpected events and a certain amount of unknown, but perhaps Tom has a better --

A (Witness Greene) I would say that some of that is to the -- for example, when you are mixing concrete, for example, and adding water and sand and gravel, you cannot make every batch identical. You are going to have little variations. And as it dries and stuff, you have another variable. So the codes do allow for certain variation in materials. I am aware of that.

Q My question, however, is simply this. Are there not good reasons for those conservatisms? And by "good reasons," I simply mean in order to guarantee that a building

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will withstand a certain design pressure, which is what we are talking about here, isn't it necessary, considering all the uncertainties involved and all the possible loadings that might be postulated for the building, to use the conservatisms you have been describing?

A Yes.

(Pause.)

Q Dr. Meyer, returning for a moment to the costs of the C&P system that you mentioned, the Union Point study, the Zion Point study, you are also familiar, are you not, with the Sandia study for possible control filter venting, retrofit at Three Mile Island?

A (Witness Meyer) Only very peripherally. I am aware of it, yes.

Q Do you have any -- Are you aware of any cost estimates for that operation?

A No, I'm not.

Q Are you aware of any estimates for the time necessary to make that retrofitting?

A You are referring to the Three Mile Island?

Q That's correct.

A All I know, it was done on a crash basis for immediate implementation they felt necessary, so I would imagine that it would have to be implemented in a period of months, but I don't know -- I haven't heard anything

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specific.

Q You discussed earlier some of the open questions with respect to the operation of engineered safety features systems after depressurization from the-- from containment. Isn't it true that if the accident remains within the design basis and those systems function as they should, that the control filter venting system would not operate and these problems would not be presented?

A Basically that's correct. There is always the possibility of an inadvertent operation of any system that penetrates the containment, but basically if the engineered safety features operate as designed, there would be no reason to activate the filtered vent.

Q With respect to the problem of depressurization resulting from a failure of the control filter venting system exacerbating a mild accident into a more serious accident, isn't it true that this is a problem that might arise with failure of any containment penetration?

A Yes, the -- the -- It doesn't matter how you depressurize the containment. If you are in the same point in your accident sequence, it is going to have the same effect.

A (Witness Greene) Let me just add that the containment isolation is such that it is supposed to function to prevent failure, so you have double barriers, double valves in a lot of systems to prevent the containment from

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failing, whereas maybe in a filtered vent system, you would want it to operate. It seems to me that one system that you are requiring not to operate and one that is required to operate, and the one that would operate or that you want to operate would have a higher probability of functioning or coming on.

Q Have either of you done any studies of the reliability of controlled filter venting systems?

A I haven't.

A (Witness Meyer) The reliability of specific systems will be addressed as part of the design Indian Point study. I am not aware of any conclusions that have been drawn regarding those studies. They are in process.

Q At this time, based on the information available now, do you have reason to believe that a controlled filter venting system penetration cannot be made as reliable as any other containment penetration up to the design pressure?

A Certainly for a ruptured disc concept the success of that system in terms of reliability is, I think, quite high for the more complicated your systems become in terms of, for example, automatic or manual venting control, the more problems you have with reliability of the system. Human error, for example, enters in.

(Pause.)

CHAIRMAN BOWERS: Mr. Ellison, would this be a

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good time to take a break?

MR. ELLISON: I am nearly through with these witnesses, so perhaps we could take another five minutes.

BY MR. ELLISON: (Resuming)

Q Dr. Meyer, on Page 7 of your testimony, in response to Question 9, you state that it is the NRC staff's position that a nuclear power plant which conforms to all the licensing requirements, criteria, and regulations presently in place is sufficiently safe to operate.

First of all, with respect to the phrase "presently in place," is it your testimony that -- are you referring to those regulations that exist today or those regulations that existed at the time Rancho Seco was licensed?

A I am referring to the regulations that were in place when Rancho Seco was licensed, plus those additional requirements that have been placed on Rancho Seco since then, and in particular as a result of the various post TMI-2 actions.

Q In this answer, you give that as the NRC staff's position. Have you yourself done an analysis of the safety of the overall Rancho Seco facility and with respect to other matters, the controlled filter vent?

A No, I am only associated with it through the question of core melt and degraded accident mitigation features.

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Q So would I be correct in stating that here you are giving the NRC staff's position, but you have not yourself examined the merit of all of the requirements and criteria and regulations that are applicable to Rancho Seco?

A That's correct.

Q Assuming for the moment that, accepting the staffs position that Rancho Seco is sufficiently safe to operate without a control filtered venting system, is it your opinion that a control filtered venting system would provide some substantial additional protection to the public health and safety?

A I stated earlier in my testimony that it is quite clear that relative to an accident which would result in containment failure, a filter vented containment system would provide a large benefit to the health and safety of the public. There are a whole host of questions, however, regarding, as I have mentioned previously, the dominant accident sequences for Rancho Seco, the containment -- specific containment characteristics, as well as others that would have to be understood before a general statement could be made that the risk would be substantially reduced, the total societal risk would be substantially reduced if a given filtered vent was required to be installed at Rancho Seco.

Q Is it your opinion that the additional matters

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that you mentioned in the latter part of your answer can be resolved with reasonable further study in a relatively short period of time?

A Are you referring to the activities in regard to the rulemaking? Is that what you are referring to?

Q Not necessarily. Since you mentioned some of these things are site specific, it is my understanding that the rulemaking is not examining each site. Is that correct?

A The rulemaking certainly will have to take into consideration the site and reactor peculiarities. In particular, they turn out to be important considerations for what kind of a system if any system at all is required.

The NRC has initiated what I feel is a rather impressive and large program to address these several areas. The utilities it is anticipated will perform a similar complementary program in these several areas over the next year or two, and I think that most of the areas will be sufficiently resolved that firm decisions can be made regarding these mitigating features.

I don't know if that answers your question.

Q Well, yes, it does, but I just wanted to clarify that it was my understanding when you said you believe the questions will be resolved, were you referring to the year or two time frame that you mentioned in your exhibit?

A The schedule for Indian Point and Zion and more importantly for matters considered here the schedule for the rulemaking are such that these issues will be resolved in a one, two, three-year time frame in that range, as opposed to a three-month or a ten-year time range.

Q Is it your understanding that the rulemaking will consider the specific site and design characteristics of Rancho Seco?

A One of the areas that is being investigated is -- well, one program that is directly applicable is referred to as the IREP. It is the Interim Reliability Evaluation Program being conducted by NRC. And it is their intention to do a probabilistic analysis along the lines of WASH-1400 on all PWR's and BWR's, and in that sense -- and that type of information will be folded into the rulemaking deliberations, so certainly in that sense Rancho Seco's site-specific and reactor specific characteristics will be factored in.

I have not seen other ways in which site-specific characteristics will be factored in, though.

What is anticipated, in talking to the people that do the WASH-1400 type analysis, is that the dominant accident sequences will turn out to be few in number and relatively insensitive to the reactor type, assuming that we are talking about PWR's, but that is just an expectation at this time.

Q Is it your belief that at the end of the rulemaking a firm decision could be made on the application of control filtered venting to Rancho Seco, including such matters as what it would cost, what its impact on reliability of the facility would be, exactly how you would design it, that sort of thing?

A The rulemaking would give guidance to how core melt and core degradation accidents are to be folded into the licensing process. What will result from that rule-making will be specific requirements and orders to any and all utilities. What that will be is certainly not known at this time, but requirements possibly, for example, to go ahead and design and have that design approved for a filtered vented system.

MR. ELLISON: Mrs. Bowers, this is a good time for a break. Following the break, Mr. Lanpher will address the Persian Castro, I believe it is Contention Number 20.

CHAIRMAN BOWERS: Fine. We will take ten minutes.

end S-3

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CHAIRMAN BOWERS: Are you ready to resume?

MR. LANPHER: Yes, ma'am.

BY MR. LANPHER: (Resuming)

Q Mr. Greene, I would like to ask you some questions regarding your testimony on board question 20 relating to the hydrogen recombiner issue. At page 4 of your prepared testimony, you indicate that for Rancho Seco, the hydrogen purge system probably would not be used for approximately 13 days after an accident had commenced, but that if a hydrogen recombiner were available, it would be activated or probably would be activated at an earlier time.

Can you please explain why the hydrogen recombiner would be activated earlier?

A (Witness Greene) For the combustible gas control system that has a hydrogen recombiner, it usually means that the containment atmosphere has to be processed through the recombiner, then pumped back into the containment. So, you are not really worried about doses to the public.

Hence, you could essentially, at the time of the accident -- when the accident starts to activate the recombiner and starts reducing any hydrogen that may form in the -- inside the containment. That gas would be just pumped from the containment building to the recombiner then back into the containment building.

Whereas, a purge system would release the

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containment atmosphere to the environment and there would be a dose to the public.

So, on one system you are really not concerned about where, in the other system, you are.

Q Is it fair to say, then, that you would not want to commence the purge operation until certain of the noble gasses have decayed enough so that when you do run the purge system, that there will not be an excessive dose to the public?

A Essentially, yes. The reason I am saying essentially, is because that is what the design is based upon. You would not be allowed to have a purge system if a dose to the public were excessive.

Q So, there is nothing in the design of the purge system which would not allow it to run right after an accident. It is the fact that you want to avoid those doses to the public?

A Yes, yes.

Q Was this difference between a recombiner which vents back into the containment and a purge system the reason for the change in regulations to require recombiners for more recently constructed nuclear power plants than Rancho Seco?

A Yes, I think it was the Commission policy, as low as practical. The Commission felt it would be in the best

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interest to have the system that did not release radioactivity to the public.

Q At what hydrogen concentrations in the containment can a recombiner begin effective operation?

A Any concentration.

Q So, immediately after an accident, it could start work if there were any build-up in hydrogen concentrations?

A Yes.

Q It is my understanding from your testimony that no hydrogen recombiner presently available would have the capacity to handle the rapid build-up of hydrogen which occurred at TMI. Is that correct?

A Yes. The zirconium steam reaction takes place very quickly. If you release huge amounts of hydrogen, the recombiner cannot process that much.

I think typically it processes around 50 cubic feet per minute. The containment is approximately 2 million feet. It would take approximately 27 days to process all of the containment atmosphere.

Q It is correct that the purpose of either a recombiner or a purge system is to keep the hydrogen concentrations below the combustible level of approximately 4 percent concentration?

A Yes.

Q If you assume an accident less severe than TMI,

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where there is, however, a build-up of hydrogen concentration, would the availability of a recombiner as opposed to a purge system -- in other words a system that could be started right away -- possibly help in keeping the hydrogen concentrations below the flammable level?

A Would you repeat that question?

Q Sure. It is my understanding that a purge system cannot be started into operation for at least several days after an accident occurs, while a recombiner could start right away. Would it not be true that having a recombiner starting right away would assist in assuring that you do not reach the flammable concentration level for hydrogen, whereas a purge system would not be able to help you in that situation?

A Once the combustible gas control system is activated, whether it be purge or hydrogen recombiners, you start reducing the hydrogen concentration inside the containment. All right?

Q By having a recombiner, you can start reducing that concentration earlier. Is that not true?

A Yes.

Q What size penetration of the containment building or penetrations are required for a hydrogen recombiner?

A There is no requirement for the size of the penetration. What design basis is assuming is a certain

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amount of zirconium steam reaction. The radiolitic decomposition of water. From that, you size your recombiner.

You can put on various -- they have a blower that essentially processes the containment atmosphere through the recombiners.

Q Are you familiar with what size recombiner would be required for Rancho Seco in order to accommodate the the design basis accident?

A No, but I believe the 4 percent limit is reached in approximately 21 days -- that requires about 16 cubic feet per minute. Recombiners are typically in the 50 to 100 cubic feet per minute range.

Q When you stated that to reach the 4 percent flammable limit, you are assuming the design basis accident. Is that not correct?

A Yes.

Q Is NRC currently analyzing the question of whether the design basis accident for hydrogen build-up should be revised?

A There is a proposed rulemaking onthe whole subject of hydrogen management.

CHAIRMAN BOWERS: Mr. Greene, could you please pull your microphones a little closer, both of them, and tilt the black one up? Thank you.

WITNESS GREENE: You're welcome.

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BY MR. LANPHER: (Resuming)

Q Would one of the subjects of the proposed rulemaking consider whether there should be a change to the design basis accident for hydrogen concentrations?

A Yes, I believe the whole hydrogen question is under consideration.

end tP-3

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Q Is the reason for this rulemaking the large amount of hydrogen concentrations experienced at TMI?

A Yes.

Q Those concentrations go far beyond the design basis accident which had been considered in the licensing of TMI.

A Yes.

(Pause.)

Q At page 3 of your testimony, towards the top of the page, you state that SMUD has made arrangements to borrow a hydrogen recombiner from another utility should the need arise.

Are you familiar with what penetration of the containment would be utilized for that hydrogen recombiner?

A No, the penetration -- no, no, I am not.

Q Are you familiar with any procedures for implementing or hooking up that recombiner if it were needed?

A No, I am not.

Q Is it fair to say that with respect to that

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borrowed recombiner, your only knowledge is that you understand that they have made some arrangements, but you do not know any of the details of those arrangements or, in fact, what the capacity of that recombiner might be?

A I believe it is an Atomic International recombiner. Atomic International recombiners are typically 50 cubic feet per minute. I think it is a relatively simple procedure to tap off some of the containment vent lines to make arrangements to hook that up.

I am not aware of the exact penetration number of what procedures have been made.

Q Are you familiar with whether, once that recombiner was hooked up, whether the containment then would be subject to a single failure which could breach the containment?

A I believe -- no, I have not seen any detailed isolation arrangement on that. Let me qualify that.

One of the problems with this is SMUD has gone beyond our requirements. They have a purge system which is acceptable that meets the single failure criteria. They have done something additional which we do not require.

We really haven't -- I have not seen any details on it.

Q It is my understanding that one of the concerns at TMI with use of the hydrogen recombiner was the fear that when -- if it had been hooked up and put into operation, that

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then the containment would have been vulnerable to a single failure which could have released the radioactivity to the environment. Is that correct?

A Yes. My understanding is that they tapped the recombiner process line between two isolation valves. So when the recombiners had to operate one isolation valve upstream of the recombiner had to be opened. Hence, the one downstream that prevented the atmosphere from -- the containment atmosphere from being released to the environment was subject to a single failure.

Q It is also my understanding that one of the short-terms Lessons Learned recommendations was to have dedicated penetrations for hydrogen recombiners, if they are being used to ensure that they do not have that kind of a situation where a single failure could defeat containment isolation?

A Yes. That was one of the requirements of Lessons Learned -- short-term Lessons Learned.

Q You are not familiar with the situation at Rancho Seco if they borrowed this recombiner, what penetration would be used and whether it would be vulnerable to a single failure. Is that correct?

A You are talking about two different things here. One is the borrowed hydrogen recombiners versus the purge system. On the hydrogen purge system, I believe, they are in the process of making the dedicated penetration require-

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ment.

Q I understand that. I believe that was one of the things they addressed in their January 7th letter. It is being evaluated.

They are upgrading it as a category B item, the purge system. With respect to the recombiner system only, which they have made arrangements to borrow, you are not familiar with whether that system uses a dedicated penetration with double isolation?

A I am not aware of anything associated with that additional back-up system that they have. I am not sure whether the piping is seismic or anything. I do not know anything about that.

Q Is the Rancho Seco purge system operated from the control room?

A I believe that they are making arrangements to incorporate that.

Q Is it presently operable from the control room?

A I do not know what the implementation schedule is on that.

Q Was it your understanding that prior to some changes which either had been done or are being studied right now, that the purge system at Rancho Seco was not remotely operable?

A From the control room, it was not operable from

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the control room.

Q Under those circumstances, how would it be operated?

A I believe it was in the auxiliary building. It could be operated from the auxiliary building.

Q Is one of the reasons that you would have to wait a certain number of days before operating, a certain number of days before operating the purge system the concern about exposure to operators who might have to go down to the auxiliary building to operate the purge system?

A The concern about delaying the operation of the purge system was not in regard to doses to the personnel -- the doses that personnel would receive when they went to open the valves.

It had to do with when you approach the 4 percent limit forthe hydrogen concentration. What I am saying is when that system is needed, it was based upon when the hydrogen concentration in the containment approached the 3 1/2 percent limit.

Q Needed in termsof ensuring that you do not have combustion in containment.

A Right. Keeping it lower than the 4 percent limit.

Q If there were a hydrogen burn in the containment, in other words, you got to the 4 percent limit or something above, is it possible that that combustion, either a slow

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burn or perhaps a detonation of the hydrogen could damage equipment inside the containment building?

A There is always that possibility, but in the main steam-line break for a short period of time, the temperatures in the containment can become higher than the design temperature, but you do not have the heat transfer from the higher tempertaure to the component.

Hence, the component itself may not see the high temperature, just a surface temperature of the component.

Q If there were a detonation, would you not have both a problem of high temperature and just the force of the detonation itself, which would also possibly damage equipment?

A Yes.

Q Are these some of the things that we are trying to guard against by having the combustible gas sytem?

A Yes. The point I was trying to make is just because you have high temperatures inside the containment does not mean that a component would fail. You might have a delay time to allow the component to see the temperature.

You have a thermal inertia associated with the component.

Q Is it your testimony then that equipment would not necessarily be damaged just because you have combustion or detonation, but it is possible that you would have?

A Right.

Q If both the hydrogen recombiner and a purge system were available at Rancho Seco, would this give the facility greater capability to manage hydrogen concentrations?

A Obviously the more systems you have, the more capability you have to handle to hydrogen combustible gas problem. If you had five systems, you would have more capability.

Q So, if you had one hydrogen recombiner in addition to the existing purge system, that would give you additional capability to handle hydrogen concentrations. Can you say yes --

A It would give you added capability to handle not necessarily higher concentrations.

Q When you say higher concentrations, you mean a more rapid build-up of concentrations, or concentrations above 4 percent, or what are you referring to?

A I am referring to -- it would give you more capability to handle a faster rate of build-up of hydrogen concentration. Also, it would give you the capability to reduce the concentration faster, but not necessarily, once you exceeded 4 percent to handle, for example, a hydrogen burn.

Q Once you get above 4 percent, you are in trouble, right?

A Yes.

Q You are subject to having the combustion at that point?

A Yes.

Q But by having an additional system, like a hydrogen recombiner, you would have additional capability of hopefully not reaching the 4 percent level?

A Yes. But, even with the three systems, the two purge systems and the combustible gas -- the hydrogen recombiner system, you still could not handle the type of releases that come with a core melt, or a large percent of the zirconium reaction.

Q An accident like TMI?

A Yes.

Q But something less severe than TMI, but perhaps somewhat more severe than the design basis accident could be handled somewhere in that continuum?

A Someplace in there, yes.

MR. LANPHER: Mrs. Bowers, I have no further questions.

CHAIRMAN BOWERS: Mr. Lewis, do you want us to go ahead?

MR. LEWIS: Fine.

BOARD EXAMINATION

BY MR. COLE:

bfm14

Q Mr. Greene, you are probably aware the first Castro-Mursh contention 20 was restated by the board. Have you seen the form as it was restated by the board in the document we sent out on February 14, 1980?

A I cannot recall the changes made.

Q All right, sir. I will read that. What I want you to do is to then make any statement you want with respect to your testimony, whether you would want to add something to your testimony.

Board question HC-20: Does Rancho Seco's present system for coping with hydrogen release incontainment provide for (a) recombiner availability early enough to respond to a situation like that at TMI-2 ; and (b) proper radiological protection of the surroundings if purging is depended upon.

Now, I think you have already answered part A. Part B is the one that concerns me, sir. Proper radiological protection of the surroundings if purging is depended upon.

A Part of the short-term Lessons Learned, I believe, was to consider shielding for the operation of systems that may be needed after an accident. I believe this is being done now at Rancho Seco.

Q Shielding?

A Shielding to protect the person from exposure.

Q Okay. That would be for occupational dose?

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bfm15

A Yes.

Q What about the people that don't work for the company?

A Non-occupational dose where you've got population exposure -- maybe you should repeat part B of that question again. Maybe I misunderstood.

Q Does Rancho Seco's present system for coping with hydrogen release in containment provide for proper radiological protection of the surroundings, if purging is dependent upon --

Now, I know what we mean when we wrote that. Was that surrounding population, not equipment?

A Okay. They system is not required to operate until approximately 20 days after the accident where the doses are quite small.

I believe the thyroid dose is, as I stated in my question, five rem, whole bodies less than one.

Q You have that on page 6 of your testimony, is that right?

A Yes.

Q Yes, page 6. Where does that information come from, the five rem to the thyroid and less than one rem to the whole body? What is that, sir?

A Okay. The way we evaluate these systems during the review process is that we do a verification of the appli-

bfm16

cant's analysis. We run, for example, the hydrogen -- we do analyses that determine the hydrogen concentration inside the containment following a LOCA.

Then we determine when the 4 percent hydrogen limit is approached, and what type of purge rate would be necessary to keep it lower than 4 percent.

All right. Then we, in the containment system branch, ship this number over to another branch that does the dose analysis. They are the ones that came up with the 5.4 and less than one rem to the whole body.

end tP-4

bgn tP-5

Q Is that --

A This is also started in the SSAR, I believe.

Q That is in 10 CFR Part 100 guidelines. That is all right?

A Not only that, but it is also in 50.44(g). That is one of the rules for a combustible gas control.

Q Are there any of the other regulations that might apply to a discharge of this type? What I am thinking of, should as reasonably achievable be applied to a discharge of this type, and has it been?

A I believe that as low as practical was in existence -- I am not sure when that came into the rule, but the 50.44(g) was adopted -- I want to say two years ago or in that time period.

At that time, we though that backfit on the older

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pfm17

plans was then necessary to require them to have systems that did not relay on purging.

Q So, it is the staff position that with respect to populations doses associated with this purging operation, that they need comply only with 10 CFR Part 100 guidelines and whatever is contained in Part 50 paragraph 50.44(g)?

A At this time, right. As you know, the whole hydrogen question, as I stated before, is being reconsidered.

Q Do you know anything about the status of that consideration of hydrogen generation?

A Only for the short-term, that we are requiring Mark I and Mark II containments to be inerted. The larger containment such as Rancho Seco, we are not requiring any changes.

Q Mark I and Mark II are boiling water reactors?

A Yes, but the uniqueness of that is that they are smaller in volume, containment volume. This is based on the fact that if you have a large percent of zirconium-water reaction, that you could have large concentrations of hydrogen inside the containment, and hence generate large pressures.

Whereas, with the larger containment, even though you have a larger amount of zirconium fuel and steam reaction, the hydrogen concentration still remains quite low.

Q Now, the basis for the radiological dose, you state

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on page 6, is the design basis accident situation for which the purging system was designed?

A Yes.

Q There is a significant difference between that desgin basis accident and the kinds of situations, scenarios, and accidents that is presently being considered as regards hydrogen generation. Is that correct?

A Yes. Okay. For the doses that were considered for the combustible gas control, it uses the -- I do want to say TID releases, but I'm not sure if that is valid anymore.

Q TID, what is that?

MR. SHON: Are you referring to TID 148.44?

THE WITNESS: I forget what TID stands for.

MR. SHON: It is referenced as a footnote in Part 100, is that right?

THE WITNESS: Yes.

MR.SHON: I'm familiar with it.

THE WITNESS: What I am trying to emphasize is the fact that in the combustible gas control for a design basis accident, defidicent products that were released to the containment were fission products that were associated with a degraded core, like 100 percent of noble gasses, I believe, and 50 percent of the halogens and 1 percent of the solids.

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bfm19

So, these were fission products released to the containment when they purged it at final LOCA.

BY MR. COLE: (Resuming)

Q Which design basis is that, sir?

A That is the --

Q Is that the one where you get the five rem?

A Yes.

Q What thyroid dose?

A Considering these releases, after 20 days you purge the containment at approximately somewhere in the neighborhood on 16 to 20 cubic feet per minute. You get this kind of a dose.

Q All right, sir. That is not the scenario that they are presently considering now in a possible future rulemaking hearing.

It is not the issue that is before the Commission via a January 4, 1980 referral from the TMI-1 licensing board. Is that correct, sir?

A Right, yes.

Q Do you know what the status of that situation is right now, sir?

A (Witness Meyer) Are you referring to the rulemaking status?

Q I am referring to the issue that is before the Commission. I am not aware if it formally in rulemaking or

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bfm20

not, or whether it is likely to go in. Could you shed some light on that, Dr. Meyer?

A (Witness Meyer) The rulemaking proceeding that is being planned for safety reviews is being divided into two parts. One, an interim rulemaking, and a more broad and extensive rulemaking.

The interim rule is presently being distributed for comment among the staff within the NRC offices. I would assume that the next step, then, is to go to the Commission with the proposed rulemaking.

Q All right, sir.

A (Witness Greene) I am looking for a paper. There is a proposed interim hydrogen control requirement for small containment. It is SECY-80-107. It is a letter to the Commission from Mr. Denton. It is dated February 22, 1980.

There is another memorandum, also, that either is in draft form or was issued this month. I have that if you want the number.

BY MR. SHON:

Q Mr. Greene, if I didn't misunderstand you when you read the title of that memo, it included the words "Small containment." Is that right?

A Right.

Q That would not be Rancho Seco, then, would it?

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fm21

A These proposed requirements for small containments -- they do discuss large containments in there and what the staff position -- why we do not require -- not -- why were are not -- have any additional requirements for large containments.

Q It seems as if the memo chiefly addresses the matter you discussed a while ago, the interting of Mark I and Mark II, BWR containments. Is that correct?

A Yes. It also discussed the basis why the staff believes that the continued operations of reactors with large containments can continue.

MR. SHON: Thank you.

WITNESS MEYER: It is within the question of major rulemaking that the question of hydrogen control will be addressed for all reactor containments.

MR. SHON: All right. Thank you.

BY MR. COLE: (Resuming)

Q Mr. Greene, still on your testimony, on board question 20, page 2 inthe bottom section of that page in response to a question: Does the Rancho Seco facility have a hydrogen recombiner?

In the second sentence of that, you state: "However, it does have a combustible gas control system which includes a hydrogen purge system."

My question is, sir, what else is there other than

bfm22

the hydrogen purge system? Is there something else?

A Filter, valves, piping, instrumentation, combustible gas control is not a hydrogen recombiner or a purge -- a purge system.

It consists of many things.

Q Could you describe that system to me, sir, briefly?

A I am saying you need a mixing system, you need valves --

Q Excuse me, what, sir?

A Valves, you know, in the piping. The sampling system.

Q So you have a mixing system, valves, and piping, and a samplling system. Did you mention filters before?

A Filters.

Q What kind of filters, sir?

A Charcoal filters that take out the iodine. I kind of do not know whether that is considered part of the purge system or not.

Q Where are they specifically located in the flow diagram?

A They are in -- downstream of the blowers, before it is vented to the atmosphere, they are in the piping.

Q But they are not used exclusively for the purge system?

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bfm23

A Yes.

Q They are?

A Yes.

Q All right, sir. Anything else in the system?

A I do not have that kind of complete description of the system. It is in Section 62 of the Rancho Seco FSAR.

Q Okay. That would be fine.

A I think the whole point of my response was that when you say hydrogen recombiner, you are not talking about a complete --

Q I said, the purge system.

A I believe when you say purge system, you are not talking about a complete combustible gas control system. There are other things.

Like I say, there is a mixing system and other things to accommodate the hydrogen.

Q What is a mixing system?

A When hydrogen is formed, there is a potential for what they call "pockets." That is, high localized concentrations inside the containments.

end tP-5

jl flws

So, they have a system that would mix the containment atmosphere to get a uniform concentration of hydrogen inside the containment. Rancho Seco relies on the fan coolers to do this, and the sprays, also, are part of the mixing systems.

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Q Dr. Myer, your testimony in CEC Issue 5-2 at page 3, in the last sentence in your response to question 5 you indicate or state, "A pressure presently being considered for a passive vent system is about 60 psia." Who's considering that, sir, is that the Commission or is that whoever is investigating those systems? Who's doing that?

A NRC presently has a number of contractor activities underway, the principal one taking place at Sandia Laboratories. And I included this as an example of one of the passive vent system pressure actuation points that has come out of that study.

Q This was a recommendation of Sandia as a possible consideration?

A It was and it is one of the options among many that Sandia is presenting to us as part of their study.

Q They picked 60 because most containment structures in pressurized water reactors are designed in the range of just below 60 psi?

A This particular study is for Indian Point 3 where I believe the design pressure is 47 psig. So it is 13 psi above that.

Q Thank you. On page 4, the table you have on that page, it's not identified by number but the column you have under "Actual", I assume it's pressure accommodation or failure. Tell me, what is that when you say "Actual" on page 4?

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A We discussed earlier this morning the conservatisms that are part of determining the 59 psig dba design basis, design value. The actual is -- if I were to ask a structural engineer for his best estimate of what the containment failure pressure would be, he would give me the actual his best estimate value. That's what I mean by actual. And for a first pass at these systems, that estimation has been twice the design basis.

Q Did you read the testimony of Mr. Daniel Nix in this proceeding, which was in April?

A I was here during that period and I heard the testimony.

Q All right, sir. Do you recall a question being asked him as to what is his best estimate of what the failure pressure, actual failure pressure, might be in a containment structure?

A I do not recall his response. I don't recall the question, either.

Q Ithought I asked him that question and that's why I'm interested in your basis for 118 psig as the actual ultimate strength of the containment structure. Ultimate failure pressure.

A Since I put together this testimony, Sandia has done a more detailed analysis and give now a family of containment failure pressures based on the particular loading progression

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in the containment that range anywhere from 90 psig up to 150 psig.

Q Do you recall how failure was described or specified in that study, sir?

A Anywhere from initial cracking of the concrete all the way to catastrophic failure of the containment where you would have permanent large openings in the containment structure.

Q All right, sir. So they considered structural cracking of the concrete to be failure.

A That's correct. They proposed several definitions of failure in order to allow the flexibility of ascertaining the effect of those kind of failures in terms of release of radioactive fission products. A cracked release with a small leak would have a considerably different consequence analysis than, of course, a more large-scale failure.

Q Sir, you just stated that the range went from 90 psi up to what, sir?

A About 150.

Q 150. And do you recall what happened at 90, as they described it?

A If my recollection is correct, it was the initiation of concrete cracking with the possibility of fissures then working their way through to the outside for leaking of containment atmosphere.

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Q In what document or report was this described, sir?

A There are three advance copy documents that have been produced in conjunction with designing an endpoint study. One is called "Summary of the Zion-Indian Point Study" and the other two are Volumes I and II of NUREG CR1409, with both volumes entitled, "Report of the Zion-Indian Point Study." These have been just released as of last week. At least, the Volume I was released this past week.

Q All right, sir, thank you.

WITNESS GREENE: I have one comment. I have a memorandum before me again from Harold Denton to the Commission that as Enclosure 1 has a copy of the Ames Report that is titled, "Strength Characteristics of the Sequoyah and McGuire Containment", and this also has all the calculations that were done to determine the best estimate of the strength of the containment.

BY DR. COLE (Resuming):

Q Does it contain the original basis of design and then their estimate of the failure point?

A (Witness Greene) I haven't really gone through the document. It's his calculations with all his assumptions.

Q But it doesn't come up to any conclusion that the structure will -- it's estimated the structure will fail at a certain pressure.

A No, I think it says it will withstand certain

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pressure, not fail that certain pressure.

Q All right. How does that pressure compare with the so-called design basis?

A Well, for example, Sequoyah and McGuire's are ice condenser plants and they're designed for low pressures in the neighborhood of 12 to 15 psig and they're talking about I remember the number is 15, designed to 15. They're talking about withstanding 48 psig, so it's a factor of 3.

Q All right, sir. I'm wondering how that would relate to the situation we have at Rancho Seco or any other particular plant. If the design basis is, say, 50, does that mean that the failure pressure could then be translated upwards in accordance with what happened here, from 15 to 48? A ratio there? Is there any correlation between Rancho Seco containment structure strength?

A Yes, that's what the staff did. They basically -- based on this study of McGuire and Sequoyah, they concluded that containments -- plus the other studies that were done I think by the Structure Engineering Branch -- that containments could withstand factors of 2.

Q Factors of 2.

A Higher than design.

Q But are they restricting that to containment structures of the type that were designed to withstand 15 psi?

A What they said is based on the Ames studies for

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McGuire and Sequoyah and the similarity of containment buildings, that they believed that the larger containments -- the containments that are designed to a higher number, up around 50 or 60, could withstand pressures in the range of 100 to 120.

MR. BAXTER: Excuse me, Dr. Cole, you did examine Mr. Nix on this subject matter. It begins at the bottom of transcript page 2700.

CHAIRMAN BOWERS: But it goes on for many pages.

DR. COLE: Thank you, Mr. Baxter.

MR. SHON: Yes, if I remember correctly, he gave the impression that setting an exact pressure for failure would be a difficult thing to do and would involve a good deal of calculation -- more than a simple ratio at any rate.

DR. COLE: Yes. As I recall his answer, he indicated that he could not give me an answer as to his estimate of what would be a likely point of failure. And he declined to estimate that number.

MR. SHON: As Iunderstand your estimate,you're not doing something real simple in your head to get an exact answer, either; you're relying on someone else's rather complex calculations. Is this not correct?

A (Witness Myer) Yes, that's correct. The Indian Point-Zion study, as I understand it, is using the state-of-the-art structural analysis codes in order to determine the failure features and modes --

WITNESS GREENE: I think we're making the point that Mr. Nix did, too, that it was one thing to calculate what a structure can withstand. By doing conservative and making assumptions always in the right directions you can determine that it can withstand a certain pressure, versus when will it fail. That's a very complex -- and I'm not sure it can be done.

BY DR. COLE (Resuming):

Q It might very well be that I asked him the wrong question.

(General laughter.)

Okay. Considering that this information about the pressures that containment structures are able to withstand, their estimate being something of the order of twice the design pressure, is that correct, sir?

A (Witness Greene): Yes.

Q At what point, then, do you think might be a point at which you would want some sort of vented containment system or filtered vented containment system to take over, keeping in mind a premature operation would release at least some radioactivity out to the environment, and to later release might result in a catastrophic failure of the containment structure.

A (Witness Myer): If we knew what that ultimate failure pressure was, and assume for argument 120 psig, then the -- well, a number of questions would come to mind, but certainly you'd like to have the point where you would vent the containment to be as close to that number as practical. But as I mentioned earlier this morning, another major consideration is that once you open this system up, you may not be able to -- if you open it up too late; that is, at too high a pressure, you may not have the capability to handle the large volumes of gases required to reduce the pressure.

So, if you would allow for an unlimited penetration opening to containment, then a set point close to this ultimate strength point would be appropriate.

There are other considerations. For example, how do you factor in other loadings like seismic loadings that may change that failure pressure? Dynamic loadings versus quasi-static loadings may affect that pressure, and of course, different sequences give you different loadings. So it's a very complicated question to answer.

Q You're suggesting that it has to be looked at the same way we looked at the original design basis.

A In many ways that's true, yes.

WITNESS GREENE: When you mention failure of the containment, I believe you were assuming that the containment would kind of fail like a balloon that justs pops, but there's a line of thought where people think that it wouldn't fail

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that way; it would develop cracks in the concrete and as the pressure is relieved, it would just kind of seal back up to a certain extent.

Q Actually, what I meant by failure is some breach in the containment structure of sufficient size to release some of the radioactivity inside, that will be measurable outside. I'd consider that to be failure.

MR. SHON: I think you can probably say that if the break is such as to release a bigger fraction of the radioactive material inside than would get through the filtered venting system, then you'd wa the filtered venting system to work. Isn't this about the size of it? Could you actually define such a point? You know, if it started cracking in the line or tore a little bit, you'd get some out and it would seal back in and you'd say, but no, you don't want it to start there because it only lets out a thousandth of one percent in filters that do that badly? Do you have any kind of approach that would give you an answer like that?

WITNESS GREENE: No, but I think you're beginning to appreciate the dilemma of having a vented filter system.

WITNESS MYER: That's a matter that's being considered.

BY MR. COLE (Resuming):

Q Among those things that are being considered, Dr. Myer, you indicated that more than one aspect of this is

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being studied by Sandia. Do you know if they are considering modes of operation that would include activation of a system at a certain pressure to operate at a certain flow rate, and then as pressure levels off and the system could be deactivated and then at some higher pressure, another level of operation can be initiated; are they considering any possible modes of operation similar to that with the accelerated rates of operation as pressure increases? Step operation?

A (Witness Myers): You're referring to something like a remote control throttling capability to regulate --

Q Yes, sir.

A I'm sure they have considered that. I don't recall that specifically being called out as an option that has been written up in that report.

Q And at 100 psi or 120 psi, you've got it wide open?

A Yes. That type of thing I'm sure has been considered by them.

Q Is it being considered or has it been considered? Have you seen it any of their documents?

A I can't right now think of the report where that is explicitly addressed.

Q Also on that table on Page 4 talking about the actual capability as compared to the design capability, the containment steam energy accommodation, sprays and coolers, this is just a 50 percent redundancy that is described in Mr. Greene's paper. Is that --

A That's correct. Yes.

Q All right. Thank you.

Mr. Greene, in your testimony related to testimony on ECCS issue 5-2, Page 4, in the second paragraph of your response to Question 6, in the first sentence there, you refer to the probability of core degradation under one situation as compared to another. To your knowledge, has this been quantified anywhere, sir? Or is it just your knowledge of the scenario and the core conditions under one scenario as compared to the other, that you intuitively know that the core melt would be more likely under one than the other?

A (Witness Greene) It's the scenario. For example, during an ECCS analysis you low down the reactor vessel and then you have to refill starting with more water in the reactor vessel, and hence you have,until the vessel water fills up to the bottom of the core, you would have relatively little heat transfer from the reactor to the coolant, steam coolant through there, and you get -- temperatures shoot up quite rapidly, whereas in the containment

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analysis, we assume that after the vessel blows down, that it is partially filled with water, and you start refilling with the water at the bottom of the core, and you have these various assumptions in the accident scenario which tend to keep the temperatures quite low.

Q So this was a qualitative assessment of the difference?

A Yes. Yes.

Q Do you know or have you seen any quantitative assessment of the probability of that happening under one scenario as compared to the other?

A No, but it is really hard to compare because in both accident scenarios you do not have core melt.

Q All right, sir. Thank you.

I guess both of you, Dr. Meyer and Mr. Greene, were talking about dominant accident series, and I believe, Dr. Meyer, you mentioned the IREP study, the Interim Reliability Evaluation Study, and indicating that a hopeful outcome of the IREP study would be the identification of the dominant accident series or sequence for different types of reactors. Is that correct, sir?

A (Witness Meyer) Yes, that's basically correct.

Q Do you know what the status of that is?

A They are presently, as I understand it, completing a study for Crystal River. The next two plants scheduled for their study are the Zion Units 1 and 2 and the Indian Point, Units 2 and 3, which is coinciding with other aspects of the Zion-Indian Point study to give us a better understanding of the dominant accident sequences for those two reactors.

I don't know what the schedule is after Zion and Indian Point.

Q I got the impression from you, sir, and correct me if I am wrong, that this -- the kind of information that might come from that with respect to dominant accident sequences or series is information of prime importance in the determination of whether filtered vented containment systems might be necessary or desirable in certain kinds of plants. Is that correct, sir?

A That's correct. It is an important ingredient in answering the question of how much you reduce the risk if you install a filtered vent.

Q Do we currently have that information on Rancho Seco, sir?

A You mean in terms of results of the IREP study --

Q Do we know what --

A -- or in terms of the raw data that goes into such a study?

Q In terms of our knowledge on the dominant accident sequence or series for this particular plant.

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A We do not have that information for Rancho Seco.

Q All right, sir.

How might you describe the state of the art as regards filtered vented containment systems?

A Well, in terms of answering the question, if you would give me the design bases and criteria, could I go out and build such a system with a lot of flexibility in terms of cost, I would say that the state of the art is such that a system could be built.

Q Then you indicated that what we have to do is make decisions as to the conditions under which it would have to be and the basis for design of the system. You need the design information, the end number, some pressure, you need.

A You need -- We need more information about how much volume will be required to vent from the containment, what the decontamination figures are that would be appropriate for the reduction in risk that we are looking for, the pressure set points such that most of the accidents will be accommodated, but yet not so low as to cause problems on the other end. These types of questions have to be addressed.

Q Do you think the state of the art is sufficient so that they would be able to design a system that would operate, say -- say you were going to decide to activate something at some pressure like 100 psi, and okay, the

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system reaches 100 psi, then you are going to activate the system, and you say the state of the art with respect to the design of that system is such that as you have described a filtered vented containment system in your testimony, that one could be designed for that with no difficulty?

A As long as in addition to you giving me the set point, you give me the volume of gases that would have to be relieved from the containment. For example, some of the accident sequences require very large volumes of gases that would mean a 20-foot diameter penetration of the containment. Other seequences require a two or three-foot diameter penetration.

Q You need the accident series.

A You need the accident series. Yes, sir.

Q All right, sir.

(Pause.)

Q I have a question here about noble gas removal, and I didn't write down what page it was on.

How do you visualize this filtered vented containment system for moving any of the noble gases?

A Perhaps a better word is to hold up the noble gases, and there are various schemes for doing that, one being, for example, at Indian Point 3, the availability of the Indian Point 1 containment, which is now a shut-down facility, as a building that could -- you could vent to and hold up the noble gases for as long as you chose. And there are charcoal filter systems that are capable of holding up rather effectively a major portion of the radioactive noble gases. So there are techniques available. They can become quite expensive.

Q All right, sir. Thank you.

I have no further questions at this time.

CHAIRMAN BOWERS: We will recess for lunch.

MR. LEWIS: Mrs. Bowers -- Well, as you wish. Dr. Meyers is scheduled to go on a flight at 5:00 to Chicago, where he has a meeting he has to chair tomorrow morning. He has asked me if possible to accommodate the fact that he would like to try and get on a 2:00 o'clock flight, and in order to do this, I was going to request that we consider going to completion of this panel, which -- I don't know how much more is involved. Well, maybe that is the question. How much more is involved?

CHAIRMAN BOWERS: Well, we will postpone lunch. Let's not cancel it.

(General laughter.)

BY MR. SHON:

Q I just had a very few questions. Dr. Cole has in fact very nicely covered most of the things that I thought about, but while we are on it, the matter of decontamination factor, hold-up, and so on, you were here when Mr.

Nix testified last month, were you not?

A That's correct. Yes, sir.

Q If I recall correctly, at least his dose figures showed very, very substantial decontaminations from the central passage to an aggregate filter, a filter that had graded particulate matter in it.

I take it you do not entirely agree with his view or with the view of the group that did his calculations for him as to the effectiveness of such a filter. Is that correct?

A That is basically correct. There are a number of questions that come to mind and unknowns regarding that type of an approach. As an example, you may be aware that in Sweden they are very interested in filtered vented systems and are in the process of performing experiments to see how good gravel and sand filters are in attenuating certain size particles, and they have -- their initial tests indicate rather discouraging decontamination factors for the type of aggregate-- I believe they are in the range of one-inch size pieces of gravel-- Rather discouraging decontamination factors.

Q That is for particulates. Such a filter would have virtually no decontamination factor for iodine or for noble gases, would it?

A The people that are experts in these areas say that there are -- you can take credit for attenuation of elemental iodine for certain kinds of gravel systems. In particular one that is being looked at at Sandia is a wet system. It is sort of a combination of a suppression pool and gravel filter where they feel you can attain rather large attenuations of iodine, but in terms of organic iodine or noble gases, the only effect you would get is the hold-up time as you push out the air and gases that were there originally and wait then for the release of the noble gases following that.

Q Yes, I understand that. Some flow rated stuff, some capacity, and it takes that long for the material to pass through in any case.

Also with regard to Mr. Nix's testimony, he had seemed quite convinced that there were only two dominant risk sequences as named in WASH-1400, and that this particular feature, filter venting, would substantially improve both of those. There were two very important ones, as you will recall. Is that not correct?

A I believe he was referring to Release Categories 2 and 3. Is that what you are referring to?

Q That's right. Yes.

A Well, in Release Categories 2 and 3 are a whole set of release -- of accident sequences. I would basically agree with them that Release Categories 2 and 3 are

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dominant in terms of risk contributors, but I would have to know more about what he considered the components of those release categories in terms of the actual accident sequences that he considered.

Q In other words, what you are saying is that the accident sequences that form the subset, so to speak, of which Release Categories 2 and 3 are comprised, are the things that you would need a lot more detail on before you could design such a system?

A That's correct, and those are the -- those are the accidents for which you will get a rather large variety of containment pressure and temperature loadings, so even though you may be in the same release category, you may have two accident sequences in that category that give you a rather different signature in terms of containment loadings.

Q I see, and then they would want, say, different pressures at which this thing would activate.

A That is correct.

Q Mr. Greene, there was one little bit of detail in Dr. Cole's questioning concerning our rewording of Question Hirsch Castro 20. I would like to read you a portion of the order that contains that to,so to speak,set the enfironment in which, the background against which we were posing the question, and ask you to elaborate perhaps a little more than you did with Dr. Cole.

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A We said that we note that at pmi 2 what was apparently a hydrogen burn took place only a few hours after the feedwater transient, and then give a number of references in the literature to that, and then you said, "In the face of such published reports, we cannot accept without question the notion that following a feedwater transient no serious accumulation of hydrogen could occur before a recombiner could be installed." And then we said, does Ranchc Seco's present system for coping with hydrogen release in containment provide for the proper radiological protection of the surroundings if purging is depended upon, and taken in context, what we meant was, if you had a TMI-like situation, could you still purge without overdosing the surroundings?

Your calculation, the 5 rem to the thyroid and one rem whole body, made the assumption that you didn't have to purge until the radiolytic hydrogen released after many, many days, made it necessary. We were thinking more in terms of the situation in which you felt you had to purge to prevent a hydrogen burn, a matter of a few hours after the transient and the accident.

Could you discuss that at some length?

A (Witness Greene) Considering the background that you stated, you implied that if you have a hydrogen recombiner, that could handle the release, and --

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Q The other thing we had asked implicitly was, could it, but I think you had already answered that question, that there was none available, at least in the market now, that could handle that. Is that right?

A Right. Recombiners are really used in what we consider the long term, but when you have a lower rate of buildup of hydrogen, it can combine more than is being produced.

With regard to TMI, you know that people talked in the range of 30 percent zirconium clouding reaction, and the pressure that was associated with that was 28 pounds. In the proposed rulemaking, this whole question of hydrogen is going to be considered, and in one of these staff memorandum papers which I mentioned previously, because of the margin and the staff's belief that the containment can withstand twice the design pressure or more, we believe -- the staff believes that even if you do have these hydrogen burns, that the containment can withstand it.

Q I trust, then, ultimately your answer to the question,"Does Rancho Seco's present system for coping with hydrogen release in containment provide for the proper radiological protection of the surroundings if purging is depended upon," against the context of, in a TMI-like situation, your answer would be no, you would not protect them adequately if you had to purge a few hours after such an

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accident. Is that right?

A If you had a TMI release, right, or -- I am not sure whether you would purge. That is the question. Even if you had recombiners, for example, I mean, that would not protect --

Q The recombiners would not protect, and the purging would not protect, and probably would not be relied upon. Is that what you are saying?

A Yes. Now, in the proposed rulemaking, they do talk about other systems that -- you can do things, for example, like inerting, such as they propose for the Mark 1 and Mark 2 containments. That is, put nitrogen in there, and that would prevent the -- an inflammability limit. Another thing they talk about is some haldon systems, put in -- I believe it is bromine or fluoride, but that would also prevent flammability limits, but my understanding is, at high temperatures they break down and become very toxic, so all these are being considered.

Q Thank you. I have no further questions.

CHAIRMAN BOWERS: Mr. Lewis?

MR. LEWIS: No, I have no redirect.

CHAIRMAN BOWERS: And Mr. Baxter? Mr. Diaz?

MR. DIAZ: I have only a couple of questions for Dr. Meyer.

MR. ELLISON: Mrs. Bowers, before we commence,

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we have discussed with the staff a slight problem we have in continuing this, in that Mr. Meyer has referred to some Sandia studies that came out last week that we have not yet seen. What we plan to do is to review them during the lunch hour so that we wouldn't have to face the possibility of calling Mr. Meyer, and apparently the staff agrees with us that for that reason it would be appropriate to take a lunch break at this time.

CHAIRMAN BOWERS: Fine.

WITNESS MEYER: There are hundreds and hundreds of pages. You are more than welcome to take a look at them over lunch, but they are considerable.

CHAIRMAN BOWERS: In lieu of lunch, it sounds like.

MR. ELLISON: Well, we will do the best we can over the lunch hour, and see where we stand.

CHAIRMAN BOWERS: It is almost 25 after.

We will recess now for lunch.

(Whereupon, at 12:25 p.m., the hearing was recessed, to reconvene at 1:25 p.m. of the same day.)

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

(1:30 p.m.)

CHAIRMAN BOWERS: Let me check first with the California Energy Commission. You were checking documents over the lunch hour. Did you complete your review? Fine. Mr. Diaz, do you want to begin cross examination? All my questions are for Dr. Meyer.

CROSS ON BOARD EXAMINATION

BY MR. DIAZ:

Q Dr. Meyer, you indicated that the NRC is at this time investigating the feasibility of implementing control filter venting systems at Indian Point and Zion. Is that correct?

A (Witness Meyer) Yes, that is correct.

Q Why were those two plants chosen for this type of study?

A They were selected for this study because these plants are already located in what is considered very high population density areas near New York City for Indian Point, and near Chicago for Zion.

Q Was the possibility of an evacuation delay one of the main factors that led to choosing these two plants for study?

A The question of the role of evacuation and delay is under consideration at NRC. One thought has been that because they are very large urban areas that outside a

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certain radius, it would be very difficult to evacuate the large numbers of people that would be required.

Q Does that complete your answer?

A Yes.

Q Would the benefits from evacuation delay depend on the population density around the plant?

A Yes, that is correct.

Q Are you familiar with the population density around Rancho Seco?

A I have the 1970 census data for Rancho Seco, yes.

Q What?

A I am not sure how much detail you are interested in.

Q How would you characterize the population density, say 15 miles around the Rancho Seco plant?

A Off hand, well, it is a relatively low population density. I could give you the exact numbers, if you are interested.

Q Would you expect any significant benefits arising from the late evacuation to be available at a site such as Rancho Seco?

A As I mentioned before, this matter is being studied. I cannot say at this time. It is outside of an area that I have been responsible for.

Q You also testified that the NRC is investigating

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aggressively the feasibility of implementing controlled filters, venting systems at different reactors. You mentioned the Indian Point and Zion study, the IREP study, the Sandia contract, impending rulemaking proceeding. Is that correct?

A That is correct, yes.

Q Sir, in view of all these NRC studies, do you know of any reason why this licensee should be required to go beyond or duplicate a current NRC effort and undertake individual feasibility studies of controlled filter venting sytems for Rancho Seco?

A My written testimony states that I feel the appropriate arena for that consideration is through the rulemaking proceedings, not -- that Rancho Seco, as well as most of the PWRs should not be singled out for considerations at this time.

MR. DIAZ: Thank you very much. I have no more questions.

CHAIRMAN BOWERS: Mr. Lanpher?

MR. ELLISON: Mrs. Bowers, with your permission, I will go first, then Mr. Lanpher on his issue.

BY MR. LANPHER:

Q Dr. Meyer, I would like to follow up on your recent responses to Mr. Diaz, particularly with respect to the rulemaking study. Am I correct in my understanding that

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the rulemaking study will consider the application of controlled filter venting to all operating plants in the United States?

A That is correct, yes.

Q Will it not also consider the application of this system to all future applications for construction permits and operating licenses?

A That is correct, yes.

Q You testified earlier that there were a number of things that you would need to know before you could design a system for a specific plant.

I recall among those things were the volume of the gas that you would need to vent, a determination about the effectiveness of the filtration system, the risk reduction that you are seeking, the appropriate set-point for the system.

Is it your belief -- well, first of all, isn't it true that each one of these things has to be examined for a particular reactor?

A In terms of application to a specific plant, that is correct.

I did indicate this morning that it is the feeling of a number of people working in this area that there will be only a small number of dominant accident sequences. There will be a surprising, I guess, similarity going from

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one reactor to another regarding these accident sequences, but we do not know at this time that each one of these sequences is appropriate, for example at Rancho Seco.

Q Okay, I would like to separate out the subject of accident sequences from the items that I mentioned, the set-point, the filtration effectiveness, the volume of gas that needs to be released. These are the things you mentioned would be important considerations in designing such a system.

With respect to those three items, isn't it true that you have to examine each reactor individually in order to determine each of them?

A Yes, there would have to be at some point in the study, a consideration for the peculiarities and differences of one plant versus another.

Q Now, turning to the accident sequences, you mentioned that the IREP study was intended to identifiy the dominant risk accident sequences for individual power plants. Is that correct?

A Yes. From a probability standpoint, the probability of the important accident sequences.

Q Did I correctly understand your testimony earlier that the IREP study was also being done on a plant specific basis?

A Yes. They are completing presently the Crystal

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River analysis and will move to the Zion and Indian Point analyses hopefully starting in June, I believe.

Q Isn't it true that the rulemaking proceeding you are discussing is not going to answer specifically for Rancho Seco the questions of dominant risk, accident sequences, set-point, volume of gas necessary to be vented, filtration effectiveness, those sorts of things?

A The rulemaking will cover a very broad spectrum of questions regarding core melts and core degradation. It is starting from the assumption that the field is open, more or less, and in terms of the -- considering the degradation features, considering reducing the probability of what are considered dominant sequences in mitigation, the question of hydrogen control will be coming up as another mitigating feature.

Another is core retention. Another is referred to as core catchers or core ladles. In that sense, it is taking on the whole question of how do we take into account core melt and core degradation in the licensing process.

A result of the rulemaking will be guidelines, design bases, requirements that will be imposed on operating reactors and reactors under construction related to these several items.

It would be the responsibility then of the individual utilities to act on those new requirements. The specific

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analysis of plants will be factored in through the IREP program. We should keep in mind that although they are doing very specific plant analyses, it becomes clear, after a while, that you can start grouping the various PWRs into various groups, for example, various PWRs.

In ice condenser containments, the utilities will also have a major responsibility to do studies in the area of the filter vented containment, conceptual designs, and assessments as part of the rulemaking proceedings.

Q Do I understand the last part of your last swer correctly that the utilities, as part of their participation in the generic rulemaking proceedings, will be required to submit conceptual design studies for their individual plants?

A I could refer you to the TMI action plan, which will be guiding our operation at NRC for the next two years. The action plan has a task referred to as 2-B, which addresses core melt and core degradation.

In that section 2-B, is a subsection 8, which refers to the rulemaking. The utilities will have certain responsibilities in this area. If I can find the page, I can read a brief paragraph to describe those responsibilities.

"Under description, selected licensees or owners groups will be required to address the feasibility of mitigating features arising from severe accident considerations, including the conduct of conceptual designs for filter vented containment, core retention, and hydrogen control systems."

This is the licensee action that is part of the action plan, which I emphasize is in draft form, but it is part of the thinking now of NRC inthis area.

Q Dr. Meyer, the phrase that you just read begins by saying selected licensees will do that. How will those licensees be selected?

A An attempt will be made to put similar NSSS and containment and balance plant systems into categories. You might have three or four different PWRs that would fall into a category that would have a very similar containment.

NSSS and balance of plant for that particular type of reactor then, there will be a selected licensee to conduct the study.

Q Do you know whether SMUD will be such a licensee?

A No, I do not. I would remind you that this is a draft task action plan and has not been made official, yet.

Q Referring to page 8 of your testimony, where you discuss the rulemaking, you conclude your testimony by saying the Commission has not yet acted on the staff's proposal.

Throughout your answers in the rulemaking, you have been assuming, have you not, that the Commission is

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going to adopt the staff's proposal without change?

(Pause.)

A The two rulemakings that I referred to earlier, the interim rule and the one that is more germane to the our discussions here, the major rulemaking is presently in draft form.

By the end of May, we are intending to issue a proposed rulemeking for comment. I have no way of judging how the Commission will act on this recommended proposed rulemaking. My anticipation would be that they would concur in major elements of it, if not all of it.

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Q Dr. Meyer, with respect to the Indian Point and the Zion studies, could you estimate when a firm decision might be made on whether to implement a system such as we have been discussing at those plants?

A The present schedule is to issue design criteria guidelines in June, and issue about December of this year a staff final report that will give staff recommendations for the direction that the staff feels Indian Point and Zion should go in the area of installing mitigating features.

Q So am I correct in stating that with respect to those facilities, the staff is not awaiting the rulemaking proceeding that we have been discussing?

A That is correct.

Q And am I correct in stating that the reason for that is the staff's perception that those facilities present a uniquely high risk to the public?

A In the sense that they are located in uniquely high population zones, that is correct, yes.

Q You mentioned your familiarity with the Rancho Seco surrounding population is based on the 1970 Census. Is that correct?

A That is correct. Yes.

Q Have you reviewed any more recent figures than that?

A I inquired about more recent figures and was

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unable to come up with more recent data. Obviously, both areas -- both regions, that is, the region around Zion as well as the region around Rancho Seco, have grown in population, but the firm data that I have is for 1970.

Q Mr. Nix stated that the area around Rancho Seco had grown quite dramatically since the last census. Do you have reason to disagree with that?

A No, I do not.

Q Although we have been discussing population density, would it not be true that a plant that was more susceptible to accident sequences would also pose a higher than normal or higher than average risk to the public?

A By definition, yes. If a plant was more susceptible to accidents, it would have to pose a higher risk on the average.

Q And it was the higher risk that led the staff to proceed more expeditiously with respect to Zion and Indian Point. Is that correct?

A It was the higher population density in the vicinity of the site that motivated the direction to study those two plants -- those two sites, I should say.

Q My question is, however, isn't the higher population density important in that it creates a higher public risk?

A Yes, all other things held constant, you double the population, say, within a 30-mile radius and you double the societal risk.

Q So wouldn't it also be consistent with that rationale to give expeditious treatment to a facility that, although it may have a somewhat lower surrounding population, had a somewhat higher probability of accident?

A If that could be demonstrated in fact and in the same way that the population question can be demonstrated, in fact, yes, I agree with you.

Q You stated that you did not know when the IREP study would examine Rancho Seco. Is that correct?

A That is correct, yes.

Q Do you know of any reason other than the present schedule why such a study could not be undertaken at Rancho Seco today?

A By whom are you assuming when you ask the question, by SMUD or by NRC?

Q By anyone. By either. Is there a physical technological reason why you could not do that kind of study at Rancho Seco?

A There was no physical or technological reason. The data that is needed to perform a study is available at every nuclear power plant.

Q I would like you to turn to Page 4 of your testimony. I would like to follow up on Dr. Cole's questions. On the graph that appears there, if I recall your testimony correctly, Dr. Meyer, you gave a couple of different figures for actual containment failures. One is the 118 psig depicted here. The other is the range of 90 to 150 psig that you described as being based upon the Sandia report. Is that correct?

A That is correct, yes.

Q So, would I be correct in stating that the 118 psig figure here is not based upon the second figure, 90 to 150 psig?

A No, it is based on an earlier estimate by structural analysts. The first pass analysis being that containment failure would occur at about twice the design pressure based on an understanding of the conservatisms incorporated in the codes that are used and that type of thing. So, it is a first pass estimate which certainly needs refining, and in fact that is what has been going on in the studies at Sandia.

Q So is it fair to say that you took the first pass estimate that you had heard of, that containments fail at twice their design pressure, and simply applied that to the 59 psig figure given for Rancho Seco?

A At the time when I wrote this testimony, that was the -- I felt that was the best and most appropriate value to use.

Q Do you feel differently now?

A Well, it should be kept in mind that the 90 to 150 is for -- is analysis for very specific containments, and my answer was more directed in a generic sense to what one might expect in general as a first pass for any containment. So, if you were to name a containment for me, I would as my first -- if I was forced to make a comment as to the actual failure pressure, I would still use a factor of two as the initial best estimate.

Q Just to clarify it for me, when you say containment pressure accommodation on Page 4, are you referring to the containment building itself?

A I am referring to the capability of the containment building to withstand that pressure. That would include the liner.

Q Just so I can clarify what I mean, I am distinguishing the ability of the building itself to withstand a given pressure from the ability of systems within containment to reduce pressures or maintain pressures below that.

A Yes.

Q You are referring to the former here?

A Just the former. That is correct.

Q In the paragraph below this chart that explains it, you state that the nature of the conservatisms is the

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redundant systems involved and the single failure criterion. The containment building is not a redundant system. Is that correct?

A No, it is one single building. In that sense it is not a redundant system.

Q So this paragraph that follows the chart does not apply to the containment pressure accommodation portion of it. Is that correct?

A Well, it applies in the sense that you in reality have twice the capability in your containment that you define in your design basis accident pressure.

Q But not for the reasons described in this paragraph. Is that correct?

A In the sense that there are not two containment systems, that is correct. It refers mainly then to the containment steam energy accommodation in terms of redundancy.

Q Does it refer to anything other than that?

A I cited these as examples. There is some -- The problem is to separate out what you strictly mean by a redundant system versus what you mean by a system that is present, that has twice or three times the capability that it was designed for, to meet certain design basis accident criteria. So in that sense, in the latter sense, there are other examples of conservatisms that might not strictly be

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interpreted as redundancies.

Q Then with respect to the containment pressure accommodation, when you say actual versus design, we are talking about a difference in assumptions, are we not?

A I am not sure what you mean by assumptions.

Q Let me rephrase my question for you. If you make the assumptions that one makes in the design and licensing of nuclear power plants for the containment building, the pressure that the building can withstand is 59 psig. Is that correct?

A That is correct, yes. So you do use different assumptions when considering the -- what I refer to as the actual.

Q So what you mean by actual is that you assume that loadings will not occur in the same pattern than we do in licensing or that calculations are more accurate than we give them credit for in licensing. Isn't that correct?

A It is more the latter. It is a matter of relaxing some of the very stringent conservatisms that are associated with the design basis accident. For example, in the design basis accident analysis, you cannot, as I understand it, go beyond yield stress in materials like reinforcing rods, where in a realistic analysis you may take some credit for some plastic deformation and still maintain the integrity of the particular structure that you are analyzing.

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So, it is a matter of the assumptions that you use regarding the integrity of whatever system you are analyzing. If you do not allow for plastic deformation, then that is a definite conservatism.

Q Using your figure of 118 psig for the actual failure point of a containment building, or in reference to that, rather, I understood you to testify earlier that the containment -- the probability of containment failure was rather small, but increasing until you reached 100 psig, and then at that point it increased more dramatically, and the containment figure became a more realistic possibility at 100 psig. Were you making different assumptions for that answer than for your testimony here?

A Well, I had the benefit of the analyses performed for Zion and Indian Point to base that statement on as well as other studies that have been going on since that time in making the statement that you referred to. I only presented this table as an illustration to establish some points that I wanted to make about the conservatisms that are in the present DBA designs, that if you have an accident beyond the design basis, it would be present to accommodate accidents that had loadings, you know, considerably beyond the design basis accident.

So, my point here was to present an illustration to make that point.

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Q Would it be fair then to say that your point is that the Rancho Seco containment building could probably withstand pressures beyond the design basis and considerably beyond the design basis perhaps, depending upon the assumptions, but that you cannot testify exactly to where the containment building would fail?

A That is a fair summary, yes.

Q In response to Mr. Shon's question, you stated that the particulate filter proposed in the underground siting study had been examined in Sweden, and that the results were, I believe, rather discouraging with respect to the attenuation factors. Do you recall that testimony?

A Yes.

Q I would like you to refer to Page 2 of your testimony. In response to Question 4 about two-thirds of the way down the page after describing various different kinds of systems, you state, "For all designs the attenuation factors for particulates and molecular iodine are better than 98 percent. Whatever the final choice, the filtered vented containment system will result in a considerable reduction in societal risk relative to an uncontrolled, unfiltered containment failure."

When you are referring to all designs, were you including the underground siting study type design?

A I was referring to the designs that had been

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proposed as part of the Sandia study program.

Q And did those designs include the type of system that is discussed in the underground siting study?

A It considered a variety of different filtering systems, most of which contained as an important element a suppression pool or a gravel volume submerged in water. I am trying to recall now. They have had so many options, and they have been changing their options. I do not recall exactly, but I do not think they considered one that you could say had a one to one relationship with the California Energy Commission filtered vent.

Q I am less concerned about the exact relationship, but just the general type of system that we are talking about here.

A Yes. And the study that has been conducted in Sweden, I only heard about two days ago in a meeting with some of the engineers from Sweden, and it is this type of thing that we have to understand much better and perhaps incorporate in a reconsideration of the total effectiveness of some of these systems for attenuating particulates and elemental iodine.

Q Have you actually -- Other than talking to the engineers, have you actually seen the study that was done in Sweden?

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A We have been negotiating a bilateral agreement with Sweden on this matter, and we intend to share reports where we send them our studies and they send us their studies, and they did not have available reports to give us at that time.

Q So would it be fair to say that you have not yourself performed nor actually seen any analysis of the attenuation factors of a controlled filtered venting system such as proposed in the underground siting study?

A You mean over and above that actually conducted as part of the study?

Q That is correct.

A My answer is, yes, that is correct. I am not aware of anybody that has critiqued in detail and analyzed the CEC study. We have staff that has read it, as we mentioned in previous testimony, but we have not done a detailed study of it.

Q Did the Sandia study consider the attenuation factors from a variety of different filtering media?

A Yes, they did, from the simplest designs from which I got these original numbers in a previous rough draft report, the simplest designs up to the most sophisticated, where, as I indicate here, you can pretty much attenuate anything you want to or hold up as much as you want to.

Q So would it be fair, then, to say that

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notwithstanding what you have heard from Sweden, that based upon the knowledge available to you today, you believe that a filtering system can be designed that would be extremely effective in attenuating the release of radionuclides from containment?

A My opinion is that such a system can be designed but it is incumbent upon us to factor into these assessments all of the experimental data and analysis that is being conducted throughout the world, and the study in Sweden is just one example of what we are trying to do to make sure there are not some studies which we have not taken into account, but I think the assumption has been up to the time of the study, that sand-gravel filters were a very effective way of attenuating the particulates and the iodine.

MR. ELLISON: That is all I have on that issue. Mr. Lanpher has some additional questions on hydrogen recombining.

BY MR. LANPHER:

Q Mr. Greene, in response to a question from Dr. Cole, I believe you stated that when your regulations were changed to require a hydrogen recombiner on your plant, you decided that it was not necessary to change the combustible gas control system on existing plants or those that were pretty far along in the licensing process.

What was the basis for that decision?

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A (Witness Greene) Well, when the regulations were changed, they also included in that change the cutoff date when certain systems were required or were not required, so the change includes the requirement that the purge system for plants of the Rancho Seco vintage are acceptable.

Q I understand that. What was the basis for the decision, though, that it was acceptable to continue with a purge system for plants such as Rancho Seco instead of requiring them to install a dedicated hydrogen recombiner?

A I believe the basis for that wa, the existing purge system presents no safety problem.

Q When you say it presents no safety problem, you mean it would function adequately to achieve its goals in terms of control of combustible gas?

A Yes.

Q Did you consider in making that conclusion the fact that a hydrogen recombiner can be utilized earlier in an accident sequence than a purge system?

A I do not think so. It makes no difference when a system is activating if it can perform its function.

Q The function of these systems is to ensure that you do not reach a combustible level. Is that correct?

A Right.

Q And I believe you testified earlier that the hydrogen recombiner can be activated earlier in an accident

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sequence. And would it not be true, given that fact, that a hydrogen recombiner has more capability to keep you from reaching a combustible level?

A Not necessarily. Some recombiners are activated on time, and some are on hydrogen concentration, if a system will limit the hydrogen concentration below a certain percent, whether you activate it in one day or in one hour. I cannot see what the concern is.

Q Is the basis for your previous response the design basis accident which was selected for hydrogen gas generation -- Let me rephrase that.

Given the -- If you had chosen a different design basis accident, one which results in greater hydrogen concentrations, would you still be satisfied with hydrogen purge systems which cannot be activated for several days after an accident?

A Hydrogen purge systems can be activated after an accident, depending on the doses you receive.

Q Given your dose restrictions, I believe your testimony before was that you would not allow them to be activated because there would be excessive doses to persons off-site.

A Your original question is, given a different design basis, an accident -- I really have to know what kind of different accidents you are talking about, because the existing recombiners may not be able to accommodate it.

Q The different design basis accident would be one which generates hydrogen -- more hydrogen than the existing design basis accident, which can be handled, I understand, by a hydrogen purge system and it activates somewhere between 13 and 20 days after the accident begins. If you have an accident which produces more hydrogen, would it not be helpful in controlling that hydrogen to be able to have a hydrogen recombiner which you can activate early in the accident to attempt to keep the levels below 4 percent?

A In the design of the combustible gas control, we looked at whether or not the system can do its intended function, and some plants, for example, have purge systems that are activated in ten days, maybe earlier than that, eight or nine days, and we look at the capability of the systems to perform their functions, which is to limit the hydrogen concentration inside the containment below the 4 percent limit. All right? And usually plants activate the system when you approach the three and a half percent limit.

That allows a little margin for error in instrumentation, but we do not look at it in terms of activating sooner or later. If, for example -- If you look at an ECCS system and you see the accumulators come on when the pressure, internal pressure -- the containment system pressure falls below 600, wouldn't it be better to come out with 700 or 800? A lot of them can do an intended function and meet the regulations. They do not present a safety problem. That is all we look for, and we do.

Q If a hydrogen recombiner were installed at Rancho Seco, would it need a containment penetration of the same size as the purge system? I believe that is a 66-inch -- two 66-inch penetrations.

A I believe that is what we call the normal containment purge system and not the hydrogen purge system. The 66-inch line is used during normal operations. Well, it is used to purge a containment when you go into refueling.

A hydrogen purge system has no normal use. It is an engineering safety feature system.

Q What is the size of the penetration for the hydrogen purge system?

A I believe three inches. I am not too sure of that.

(Pause.)

I am not too sure of the number, and like I say, I believe it is three inches.

Q At Page 5 of your testimony, you indicate that there is reconsideration of the design basis for the combustible gas control system, and that any decision on that would probably be deferred pending a rulemaking. Has

300 7TH STREET, S.W. REPORTERS BUILDING, WASHINGTON, D.C. 20024 (202) 554-2345

that rulemaking been commenced yet, or is this again part of the post-TMI action plan which is being proposed by the staff?

A I don't know what you mean by that. We are preparing memos to the Commission on it, but the -- they have taken the staff position on it, so it has commenced in that light.

Q I understand that the staff has taken a position, but the NRC has not decided yet to go forward with the rulemaking on these issues. Is that correct?

A The Commissioners?

Q Yes, the Commissioners.

A I do not think so.

Q You testified in response to one of the Board's questions that with respect to the Mark I and Mark II containments that given their size and perhaps other factors, you are requiring them to be inerted. Is that correct?

A Well, we always -- Most of the Mark I's and Mark II's were inerted anyhow.

Q Has any consideration been give~ to imposing that requirment on, for instance, the PWR's, including Rancho Seco?

A Do you mean, when you say consideratio~ -- We thought about it and looked at it after TMI, and we came to

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the conclusion, for example, like the ice condensers which had a slightly smaller volume than Rancho Seco, for example, and decided not to inert them.

Q If they were inerted, would the combustible gas problem which the -- strike that.

I understand from your responses, I believe, to Mr. Shon's testimony, that the existing purge systems and the existing recombiners cannot handle the quantity of hydrogen --

MR. SHON: My testimony?

MR. LANPHER: Your question.

BY MR. LANPHER: (Resuming)

Q Cannot handle the quantity of hydrogen produced in the short time from a TMI type accident. If you have an inerted containment, do you have the same problem of combustion from hydrogen?

A You have the same amount of hydrogen released, but you do not have a problem of combustion.

Q So would this be -- I am sorry. Did you finish your answer?

A No. I was just going to qualify why.

Q Go ahead.

A Because we removed all the oxygen from the containment.

Q Then one of the questions that Mr. Shon was asking

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was whether there was a way to respond to the Board question or the rephrased Board question to handle the quantity of waste -- of hydrogen produced at TMI would be to inert the containment. Is that true?

A One of the ways to limit the flammability limit of hydrogen is to put in nitrogen into the containment, which is called inerting, yes.

MR. LANPHER: I have no further questions.

CHAIRMAN BOWERS: Mr. Lewis, do you have any further questions?

MR. LEWIS: I have a question or two.

BY MR. LEWIS:

Q Dr. Meyer, do you know whether or not the consideration of inerting of Mark I and Mark II containments is a matter that has been imposed as of this time or is simply a proposal before the Commission?

A (Witness Meyer) As I understand it, it is a proposal before the Commission, and has been incorporated in the draft of the interim rule that I referred to previously.

MR. LEWIS: That is all I wanted to ask.

CHAIRMAN BOWERS: Dr. Cole mentioned the TMI 1 Board referring the hydrogen question to the Commission, and now there is oral argument, about five or six weeks ago, and GE came in in amicus. Do you know anything about the status

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of that?

MR. LEWIS: I do not. It was to go up to Shoreham.

CHAIRMAN BOWERS: But the appeal board did not let it go up. They said they would hold it at their level. But you see, with TMI 1 they do not go through the appeal board on questions like this.

MR. LEWIS: I do not know the status of that.

CHAIRMAN BOWERS: We understand it will be a long time. All right. Do you want to ask that the witnesses --

MR. LEWIS: I would like to have them excused.

CHAIRMAN BOWERS: Any objection?

(No response.)

CHAIRMAN BOWERS: The witnesses are excused.

(Witnesses excused.)

MR. LEWIS: Mrs. Bowers, the staff wanted to mention two preliminary matters which we deferred on this morning in order to get this panel on, but if I may impose for just a few minutes before we bring Mr. Mann on, one matter is that the staff distributed to the board and parties on Thursday and Friday of this last week a copy of the final version of NUREG - 0667, and now I have discussed with Mr. Ellison and Mr. Baxter the fact that I thought it would be appropriate to have that document as an NRC Staff Exhibit in this proceeding, particularly since its draft version was also an exhibit, and obviously, this is the

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final document, and it also contains two sections that are new to it. They both indicated to me that they would like some more time to consider whether or not they wanted to undertake some cross examination on that document, so I guess at the moment I just wanted to make clear to the Board and parties that some time during this two-week period Mr. Capra, who is available here, would be available.

I will sponsor the document through him, and he will be available for cross examination. I see no point in bringing him on at this time to sponsor the document, but perhaps we can fit that in at some later point during the two-week period. That is one item.

The other item, Mr. Black, I believe, would like to address one other matter. We would like to bring it to the attention of the Board and parties.

end 7&8

Suzy foll.

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MR. BLACK: One ongoing program that the Commission has been considering for some time now but has been expedited because of TMI and its aftermath has been a program to re-evaluate the NRC Inspection and Enforcement Program.

One aspect of this program was to evaluate the performance of NRC licensees from a national perspective rather than a regional perspective. And one principal means of performing this evaluation was to establish a management appraisal/inspection program and to evaluate NRC licensees by means of an Inspection and Enforcement Review Team. This Performance Appraisal Team, or as it is known generically, the PAT, Performance Evaluation Team, is comprised of certain chosen I&E inspectors from throughout the various five regions of I&E, and its charter was to examine selected licensee management co.trol systems.

My understanding is that this Performance Evaluation Team has gone out and evaluated certain NRC licensees. During the course of this year and oncoming years I think they'll get around to all the NRC licensees, and the reason that we're bringing it up now is that they are in the process right now of reviewing Rancho Seco.

And if I might just indicate what they are looking at right now -- as I indicated, they're looking at Rancho Seco but they're looking at Rancho Seco's management control systems in the following areas. Whether they licensee has written procedures or policy documents to provide guidance in the management of a given subject area; and whether the procedure and policy documents are adequate for controlling the applicable activities in the subject areas to assure compliance with regulatory requirements; whether licensee personnel with responsibilities in the subject area are qualified to perform their activities and have been trained and retrained to maintain their qualification level; whether the individuals who have been assigned responsibilities in the subject area understand their responsibilities; and finally, whether the requirements for the subject area have been implemented to achieve compliance, and all activities are appropriately documented.

So basically, the orientation of the PAT inspections is to determine how the licensee manages license activities to assure continued compliance with the regulatory requirements and guidance. And this differs from the regional-based inspections which are oriented toward the verification that the licensee is compliance with the regulatory requirements and guidance.

As I indicated previously, the PAT inspection review team is currently going through Rancho Seco management. It has, I believe, already checked out certain aspects of this program by interviewing people at SMUD's corporate headquarters. This week I believe it is out at the site

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and reviewing personnel out there. And as you can basically get from the gist of what I just indicated, they are looking at these licensees to determine whether they do, in fact, have procedures to implement the requirements that the NRC has in effect at the present time. Not only the requirements we have in effect, but also, that which is considered a suggestion through the regulatory guides.

And then when they look at these procedures, they also make sure that the licensee is implementing these procedures; that it has personnel there to fulfill these responsibilities, to make sure that these responsibilities are taken up and down through the chain of command.

As I indicated, this differs significantly, or somewhat significantly, from what the I&E regional inspection team does, or the on-site regional inspector. They are mainly to assure that the licensee has, in fact, complied with the requirements and the guidance given by the NRC, not whether it has a program to get to that compliance.

I'm mentioning this right now because of the -- we are in this Rancho Seco PAT review now, and the preliminary plannings from this inspection have resulted in a number of concerns which may be relevant and material to the issues being considered by this licensing board; namely, whether Rancho Seco or SMUD is -- the management is competent to own and operate the Rancho Seco facility.

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But I must emphasize that these are preliminary findings. The final exit interview will be conducted May 5th and a report will be finalized around July 1st, 1980.

DR. COLE: May 5th?

MR. BLACK: The exit interview will be conducted either May 8th or 9th of this week, and the final report will be issued July 1st, 1980.

So realizing that we have this ongoing proceeding, we are suggesting that we make certain members of the Performance Appraisal Team available as witnesses in this proceeding to offer testimony as to their preliminary findings regarding the management of Rancho Seco. But there again, we would do so with the understanding that these are just preliminary findings but indeed, they can be cross examined as to these preliminary findings.

So what we are proposing and will propose is that we will offer pre-filed testimony with regard to the performance appraisal. We intend to make this available at the end of this week, and hope to have the witnesses available at the end of next week. But there again, we realize that we are offering this testimony at the Eleventh Hour, and that all parties and the Board members certainly have not had sufficient time to look over those preliminary findings and testimony. And therefore, it may be necessary to defer that examination of the Performance Appraisal Team until some later date. But we would suggest right now that we do file the testimony this week and that if all people look at it, and hopefully we can get on those witnesses at the end of next week. So we're throwing that out for the comment of the Board and parties at this time.

CHAIRMAN BOWERS: Mr. Baxter?

MR. BAXTER: I'm not sure that it's necessary to decide anything at this point. I would have to not let go by Mr. Black's description of what the Performance Appraisal Team is actually doing and its difference from what the regional Inspection & Enforcement people normally do.

I've had occasion to read a couple of their reports on other plants, as well as talked to district people who are now going through the process, and I don't see a difference from the verification effort except in its extent and level of detail.

On the other hand, I certainly don't have any objection to the staff adding this additional testimony if it can be accommodated reasonably well within the schedule we have right now. We will make every effort, if given some opportunity, to get our cross examination and any rebuttal prepared soon after we get this additional testimony.

I would like to make clear, though, that I would reserve the opportunity to argue the merits and materiality of such testimony if it becomes apparent that it's going to

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cause a substantial schedule problem for concluding this record. But as of right now, I'd be happy to try to accommodate the schedule Mr. Black has outlined.

CHAIRMAN BOWERS: Mr. Ellison?

MR. ELLISON: Mrs. Bowers, like the Board this is the first that I have heard of the Performance Appraisal Team, and like Mr. Baxter, we have no general objection to what Mr. Black has proposed, but we are in the position of not having seen the testimony and not being able to discern how it would affect both the schedule of this proceeding and also the substance of our cross examination on the issues that it addresses.

So I guess all I can say at this time is that we have no objection to planning on that, but if complications arise, we will inform the Board of that as soon as we can.

CHAIRMAN BOWERS: Hopefully, we'll see it before the weekend.

MR. BLACK: Yes. We certainly intend to give it to you before or on or before Friday. Hopefully, we'll be able to finish it up by Wednesday or Thursday.

CHAIRMAN BOWERS: Another separate matter, Mr. Ellison, the Washington Post a week ago Saturday had quite a headline and the story about Judge Manuel Real's decision. Does that, in any way, affect your participation in this proceeding?

MR. ELLISON: Aside from its impact on the political atmosphere in California, no, there's no direct impact on our participation in this proceeding.

CHAIRMAN BOWERS: We can take a 10-minute break now before Mr. Mann.

(A short recess was taken.)

Tape S-7

CHAIRMAN BOWERS: Mr. Mann.

Whereupon,

BRUCE J. MANN

was called as a witness by counsel for the state of California and, after being first duly sworn, was examined and testified as follows:

XXX

DIRECT EXAMINATION

BY MR. LANPHER:

Q Please state your full name.

A My name is Bruce J. Mann.

Q Do you have in front of you, Mr. Mann, a document entitled, "Prepared Direct Testimony of Bruce J. Mann Concerning a Release of Radioactivity from Containment (CEC Issue 5-1)?

A Yes, sir.

Q Is there attached to that document a "Summary of Professional Qualifications"?

A Well, I can't speak for others, but my copy is not attached but I have a copy of it.

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Q It is attached. Did you prepare this testimony and the Summary of Professional Qualifications?

A Yes, I did.

MR. LANPHER: Mrs. Bowers, we have attached to the copies made available to the reporter a correction sheet which I distributed to everyone this morning.

BY MR. LANPHER (Resuming):

Q Mr. Mann, are those corrections which you prepared for this testimony?

A Yes.

Q And as corrected, is this testimony and your statement of professional qualifications true and correct?

A I would make one slight correction to the statement of professional qualifications, if I may.

Q Okay.

A On that page, the first line of the second paragraph, I made a mistake in my addition and "professional employment experience includes..." correct "twelve" to "eleven years."

Q Except for that correction, is this prepared direct testimony and the statement of professional qualifications true and correct?

A To the best of my knowledge.

MR. LANPHER: Mrs. Bowers, we would like this inserted into the transcript as if read.

MRS. BOWERS: And admitted into evidence?

MR. LANPHER: And admitted into evidence.

MR. BAXTER: No objection.

MR. LEWIS: No objection.

CHAIRMAN BOWERS: The document you've identified will be physically inserted in the transcript as if read and admitted into evidence.

(The above-mentioned document was admitted into evidence.)

MR. LANPHER: The witness is available for cross.

CROSS EXAMINATION

BY MR. BAXTER:

Q Mr. Mann, when did you assume your duties here in Sacramento with the California Energy Commission?

A I first entered into employment with the Energy Commission in, I believe, August of 1978 on a temporary basis, after which I assumed employment on a full-time basis I believe in October of 1978.

Q But you were on leave of absence for some time in 1979 with the Kemeny Commission staff. Is that correct?

A Yes, that's correct.

Q And following that tour of duty with the Kemeny Commission, when did you return to Sacramento to resume full-time your duties with the Energy Commission?

A Oh, I believe approximately mid-November 1979.

Q Did you advise the Energy Commission with respect

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)

Prepared Direct Testimony of
Bruce J. Mann Concerning Release of
Radioactivity from Containment (CEC Issue 5-1)

February 11, 1980

Sponsored by the
California Energy Commission

Prepared Direct Testimony of
Bruce J. Mann Concerning Release of
Radioactivity from Containment (CEC Issue 5-1)

My name is Bruce Mann. I am employed by the California Energy Commission as an Energy Project Specialist in the Engineering and Safety Office. My testimony relates to CEC Issue 5-1 which states:

> Whether systems identified as contributing to releases of radioactivity during the Three Mile Island accident which are outside containment should be changed to vent into the containment building.

Introduction

As a result of the March 28, 1979, accident at Three Mile Island Unit 2 ("TMI"), large amounts of radioactive fission products were released from the reactor fuel. Significant amounts of some of these fission products were transported from the primary coolant system and out of the reactor containment building, and some were subsequently released to the environment.

The release of fission products from the TMI containment building was characterized by diverse paths involving several plant systems. This was partly the result of conditions unanticipated in the design of the facility and partly by circumstances not understood by the operators during the accident itself. The diverse and unanticipated release paths from containment has serious implications for other nuclear power facilities in the U.S. The TMI facility was not a unique design, and it appears likely that similar release paths may exist at other facilities.

connects in various ways (pipes, pumps, valves and other components) to systems outside the containment building.[2]

Additional systems played a lesser role in the transport of radioactivity from the TMI containment. These include the secondary side of the B steam generator and miscellaneous sampling lines which penetrate the containment structure.[3]

A large number of systems located outside the containment building acted either as conduits or receptors for contaminated liquids and gases emanating from the primary coolant system as a result of the accident. These include systems located in the auxiliary and fuel handling buildings adjacent to the containment building.[4]

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2. A good description of these systems at TMI is provided in NSAC-1.

3. See, e.g., NUREG-0600, pp. II-3-1 through II-3-15. While the various investigations of the accident reported to date are in general agreement as to the relative importance of the various pathways from containment, there is uncertainty as to many aspects of the releases through the systems. Thus, amounts, forms, and time sequences of releases are not well known. The General Public Utilities Corporation (GPUC) currently has a study underway to quantify fission product release paths and amounts through the various plant systems. Results are expected to be made public within the next two months. Additional information should be obtained when direct examination of systems inside containment at TMI is possible.

4. The identification and description of process systems which communicate with systems which transported radioactivity from the containment is contained in the report by Lawroski to the President's Commission on the Accident at Three Mile Island (hereafter "President's Commission"). This report describes the extensive interconnections between these process systems. It also describes the potential routes from these systems whereby the materials became available for release to the environment. The GPUC report "Assessment of Off-Site Radiation Doses from the Three Mile Island Unit-2 Accident," TDR-TMI-16 Rev. 0), July 31, 1979, identified systems and routes involved in releases from the facility to the environment.

information. For example, attempts to obtain samples of primary coolant and to determine the identity and concentration of fission products on the morning of March 28 were compromised by the spread of radioactive contamination and high radiation levels in the facility radiochemistry lab.[7] Information from in-plant radiation monitoring systems which would have been of value was rendered suspect by the extensive spread of radioactivity to areas and systems outside the containment. This included information from process monitors, area monitors, and effluent monitors.[8]

Severe in-plant radiation control problems were encountered at TMI. These have been extensively discussed in the NRC I&E investigation.[9] The accident revealed deficiencies in facility design, staffing and operation with respect to worker protection and radiation control (health physics).[10] For example, systems which process primary coolant were not designed to handle the large volumes of highly contaminated fluids generated by the accident. The NRC has identified several deficiencies in the Metropolitan Edison organizational structure, individual staff

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7. See NUREG-0600, pp. II-3-79 through II-3-41, where additional problems in diagnosis of plant systems status through radiological and chemical sampling are described. Contamination of the TMI on-site radiochemical facility also precluded timely and accurate assessment of potential radioiodine releases through analysis of off-site air samples on the morning of the 28th. Id., p. II-3-41.

8. NUREG-0600, p. II-D-1; TDR-TMI-116, pp. 4-1 through 4-5.

9. NUREG-0600, Section II, "Radiological Aspects".

10. Ibid. See also Auxier et al., pp. 31-34.

details of primary coolant process systems between TMI and Rancho Seco.[13]

A. Rancho Seco Makeup and Purification System

At Rancho Seco there are two egress routes from the containment building involving the let-down system. These are the main let-down line and the line for main reactor coolant pump seal water return.[14] Outside of the containment building, these lines feed to the purification system for processing. The purification system connects to liquid radioactive waste (radwaste) treatment systems located in the Rancho Seco Auxiliary Building.[15] The liquid radwaste systems connect to the waste gas system which in turn is connected to the facility exhaust duct.[16]

With respect to the main features, the let-down/makeup and purification systems at TMI-2 and Rancho Seco are quite similar. Both facilities have containment building penetrations for the main let-down and reactor coolant pump seal return lines.[17] The let-down/makeup and purification systems at both Rancho Seco and

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13. The Rancho Seco facility balance of plant design was performed by Bechtel Corp., whereas at TMI Burns and Roe Inc. performed this function.

14. FSAR, p. 9.2-7. The main coolant pump seal water return line is also called the reactor coolant pump controlled bleed-off line.

15. Those systems are described in Section 11 of the FSAR. Figure 9.2-1 shows in schematic form the interconnections between the makeup and purification system and radwaste systems.

16. E.g., FSAR, Figure 11.1-3.

17. For a description of the routes from containment involving the let-down system at TMI-2, see Lawroski, note 4, supra, Section 4.

The reactor building drain header connects to systems outside containment via a 6-inch line which penetrates the containment building. This line connects to the reactor coolant system drain tank. The reactor coolant system drain tank is connected to the reactor coolant radwaste system.[22]

The pressurizer relief tank rupture disk is designed to protect the relief tank from over-pressurization.[23] In the event that the disk is breached, the tank would drain into the reactor building sump.[24]

The reactor building vent header system is designed to vent the gas spaces from systems inside containment. The relief tank is connected to the reactor building vent header system through both a normally closed gate valve and a pressure relief valve.[25] The Rancho Seco reactor building vent header penetrates the containment building through a line which connects to the flash tank of the coolant radwaste system.[26] The flash tank is interconnected to both the liquid and ~~gaseous~~ radwaste treatment systems.[27]

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22. See Bechtel Piping and Instrumentation Drawing (P&ID) #M-560 Sheet 3 and Figure 11.1-1 of the FSAR (amendment 29).

23. FSAR, p. 4.2-28 (amendment 20).

24. Bechtel P&ID Sheet M-520. At TMI, the rupture diaphragm on the reactor coolant drain tank failed at about 15 minutes into the accident. See NSAC-1 Appendix: "System Thermal Hydraulic Behavior" for example. This appeared to be the main route for primary coolant flow to the reactor building sump at TMI.

25. FSAR, Figure 4.2-1 (amendment 29).

26. FSAR, p. 4.2-28 (amendment 20).

27. See FSAR, Figures 11.1-1 and 11.1-3 (amendment 29) for example. The flash tank vents to the waste gas system through the waste gas collection header. Liquids are transferred to the coolant waste receiver tanks in the coolant radwaste system.

C. Reactor Building Sump

The purpose of the sump system at Rancho Seco is to collect all fluids which accumulate inside the reactor building. Various reactor building drains feed into sumps, and the sumps drain by gravity through a containment penetration line to an accumulator tank located outside containment.[32] This tank dumps when full to a sump in the decay heat pump room from which two pumps take suction for transfer of accumulated liquids to the miscellaneous waste tank. This tank is one of the receiving tanks for the miscellaneous liquid radwaste system.[33] The decay heat pump room sump atmosphere vents to the auxiliary building ventilation exhaust system at Rancho Seco.[34]

The reactor building sumps at both TMI and Rancho Seco drain to tanks located outside the containment building. The transfer of liquids at TMI from the reactor building sump is accomplished by sump pumps located inside the reactor building, whereas at Rancho Seco, gravity flow is used.[35] The receiving tank at TMI for liquid pumped from the reactor building sump is the auxiliary building sump tank. TMI also appears to differ from Rancho Seco

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32. FSAR, p. 4.2-26 (amendment 9). For identification of systems which drain into the reactor building sumps and the connection between the sumps and systems outside containment, see Bechtel P&ID DWG #M-592.

33. FSAR, Figure 11.1-2 (amendment 29).

34. See Bechtel P&ID DWG #M-561.

35. For details of TMI, see Lawroski, p. 4-4.

reactor containment building isolation.[40] The reactor building isolation would thus prevent the transport of fluids from containment via the sump drain route. Even if the pressurizer relief tank rupture disk failed and drained into the sump, egress would have been prevented. Since fluids drained from the pressurizer relief tank through the failed rupture diaphragm would be retained in the containment building.

At Rancho Seco the other two potential routes from the pressurizer relief tank, i.e., via the reactor building drain header and the reactor building vent header, are equipped with isolation valves which are designed to close upon reactor building isolation.[41] Successful reactor building isolation prior to the pressurization of the Rancho Seco pressurizer relief tank would thus preclude the passage of fluids from containment via these routes. Similarly, the let-down system at Rancho Seco is designed to isolate upon reactor containment building isolation.[42]

Thus, the Rancho Seco facility appears to be less vulnerable than TMI in terms of experiencing diverse release routes from containment by virtue of the containment isolation on ECCS-HPI actuation. However, there are a number of relevant factors which need to be examined in order to adequately assess the Rancho Seco vulnerability.

An important one is the reliability of containment isolation. This includes the reliability of isolation upon demand (safety

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40. See FSAR, Table 5.2-2, p. 5.2-49 (amendment 17) for example.

41. FSAR, Table 5.2-2.

42. Ibid.

possibility of damage to isolation components.[46] For example, at TMI the pressurization of the reactor coolant drain tank apparently produced pressure surges into the vent line and the vent header which penetrate containment. The possibility of damage from water (including water slugs) and two phase flow accompanied by pressure pulses through systems designed to vent gases and vapors should be considered.

Conclusion

Based upon my review of the Rancho Seco facility, I believe that it presents a better defense system to containment releases than the system at TMI. Notwithstanding this conclusion, I also believe certain actions should be taken to ensure that TMI-type releases do not occur. Mainly, SMUD should perform an analysis to identify additional potential release paths from containment and to evaluate potential failures of containment isolation. This analysis should include the identification of accident sequences and of operator actions which could affect containment integrity and isolation effectiveness. This should utilize a systematic approach such as the use of event-trees and failure-modes-and-effects analyses which include effects of systems interactions and operation actions. The analysis would yield an identification of and a ranking of potential release paths from containment.

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46. Potential pressure transients in systems which could have transported radioactive materials from containment at TMI are discussed by Lawroski, Section 7.

BRUCE J. MANN: SUMMARY OF PROFESSIONAL QUALIFICATIONS

My name is Bruce J. Mann. I am an Energy Project Specialist with the California Energy Commission. My formal education includes a Bachelors Degree in Mathematics (Ashland College, 1960) and Masters Degrees in Bioradiology (University of California, 1964) and Nuclear Engineering (University of California, 1971).

My professional employment experience includes ~~twelve~~ eleven years of service as a health physicist with Federal Government agencies (Public Health Service and Environmental Protection Agency). I have performed as a health physicist at a weapons laboratory (Sandia), a research reactor facility (UCLA), and a high energy physics laboratory (Lawrence Berkeley Laboratory). I have held AEC operator licenses for research reactors at UCLA and UC Berkeley and have performed as reactor operator at these facilities. I have served as a consultant to an environmental engineering firm (Teknekron) and to a U.S. Senate Committee (Environment and Public Works). During May through November 1979 I served as a technical staff member for the President's Commission on the Accident at Three Mile Island.

My major area of professional experience has been in the assessment of public health impacts from nuclear energy programs and facilities. This includes both operational radiation monitoring and field studies as well as conceptual studies for both normal operations and accident or emergency situations. I have participated in several studies of nuclear fuel cycles and was project manager for a major contract study which assessed nuclear fuel cycle radioactive waste management options while employed by EPA.

I am certified in health physics by the American Board of Health Physics and am a registered professional engineer (Nuclear Engineering - California). I am a member of the Health Physics Society, the American Nuclear Society, and the American Association for the Advancement of Science.

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)	)	

AFFIDAVIT OF BRUCE J. MANN

Bruce J. Mann, being duly sworn according to law, deposes and says as follows:

I have prepared and am familiar with the attached document entitled "Prepared Direct Testimony of Bruce J. Mann". The opinions set forth therein are my own and, to the best of knowledge, the facts set forth therein are true and correct.

Dated: February 11, 1980

B. J. Mann
Bruce J. Mann

Sworn and subscribed before me this 11th day of February 1980.

Mary McDearmid
Notary Public

OFFICIAL SEAL
MARY McDEARMID
NOTARY PUBLIC - CALIFORNIA
Principal Office in SACRAMENTO County
My Commission Expires Feb. 20, 1980

BRUCE J. MANN TESTIMONY

Corrections

Page	Line	Correction
1	4 from bottom	"have" instead of "has"
2	3	delete "dis-"
3	line 3 of footnote 4	"are" instead of "is"
12	10	insert "primary coolant" between "reactor" and "system"
12	16	delete "high pressure injection (HPI)" and insert "safety features actuation"
13	5	delete "Since" and capitalize "Fluids"
13	19	delete "ECCS-HPI" and insert "SFAS"
15	21	"operator" instead of "operation"
16	6	"radioactivity" instead of "radioactive"

to or play any other role in the decision to raise this issue, CEC 5-1, with the licensing board when the Energy Commission filed its pleadings last August?

A No, I was not involved at that time.

Q To your knowledge, were your colleagues at the Energy Commission, when you started working on this case, aware of the differences you've testified to between the containment isolation procedures at Rancho Seco and those at Three Mile Island Unit 2?

A I have no idea, I have no opinion. I am unaware if they were aware of it or not.

Q I refer you to page 6 of your testimony. In the fifth line of the first full paragraph under III, you state that Rancho Seco's containment isolates differently than did Three Mile Island's. Then if I could turn you back to page 2 of your testimony, you state, beginning on line 6, that Rancho Seco apparently has a different containment isolation procedure than Three Mile Island's. Is there any doubt in your mind about the fact that there is a different containment isolation procedure?

A No. I think it's correct to state that as far as I'm aware, the Rancho Seco containment system is designed to isolate differently than the case at TMI bearing the TMI accident.

MR. BAXTER: Those are all the questions I have.

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BY . LEWIS:

Q Mr.Mann, given the Rancho Seco containment isolation system, what lines outside containment do you contend could c__ry radioactivity in the event of a feedwater transient?

A Well, in order to answer that question it requires some assumptions. The concern I have with this issue is that in reviewing the experience at Three Mile Island, I pointed out in my testimony but perhaps I didn't emphasize it strongly enough -- the effect of containment isolation at TMI was not a large factor in preventing releases from containment during the accident. Even though the containment isolation system was designed differently at Rancho Seco -- let me correct that. The important thing in my mind here is that at TMI, even after ESFA or safety features isolation did occur, the operators found it necessary in their view to override containment isolation, and the effect of that was that for a long time after isolation did occur there were continuing releases through the letdown system into several systems outside the containment.

So it's not such a straightforward question in my mind.

Q Let me ask you this. Isn't it true that the Rancho Seco containment is designed to isolate either on high reactor containment pressure or low reactor coolant system pressure and that the low reactor coolant system pressure was an isolatior signal that was not available at TMI, too?

A That's correct, that's my understanding.

Q Would it be your understanding that in the scenario in which you had some type of a loss of coolant which tended to depressurize the reactor coolant system, you would likely reach a containment isolation quite quickly from low reactor coolant system pressure?

A Well, you'll have to help me, give me some reference. What do you mean by quickly?

Q Do you know what the times would be to containment isolation in the event of, say, for example, a stuck-open PORV?

A No, not in general. I think it depends, of course, on the detailed sequence of events that actually occurred in a given situation. In reviewing preliminary data on the Crystal River transient, it's my understanding that containment isolation occurred somewhere in the time frame from two to three minutes into the transient. And there's some uncertainty on that time, when it actually occurred, for several reasons.

Q I see. This is on page 12 where you state, "At TMI, ECCS-HPI initiation occurred at two minutes and two seconds into the accident." Is that what you're referring to?

A Would you refer me to the specific line you're

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reading from?

Q The lines aren't numbered but it's on page 12, just before where you say footnote 39.

A Oh, yes, okay.

Q That's the reference to the approximately two minute time frame for --

A No, I think there's some confusion here. I was referring in my previous answer to my understanding of the Crystal River event.

Q Would it be your understanding that one of the principal possible release paths in the event of a stuck-open PORV or stuck-open safety valve would be from overflowing from the pressurizer relief tank, the PRT, and into the various relief lines that come from that?

A Release paths from containment?

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Q Mr. Mann, do I correctly understand that you testified at Crystal River it took between two and three minutes to achieve containment isolation?

A Yes, that is my understanding.

Q If containment isolation were achieved at Rancho Seco under a somewhat similar sequence of events in approximately the same time frame, would you accept the fact that a containment isolation would likely be achieved at Rancho Seco in the same time frame for a similar sequence of events?

A As far as I know, yes.

Q Would it be your belief that if containment isolation were achieved within this two to three minute time period, that would substantially reduce the possibility of releases outside containment as compared to what was experienced at TMI 2?

A Yes, if it were successfully achieved and not defeated either through operator action or through some failure, yes.

Q Would it be your understanding that in the wake of TMI 2, operators of nuclear power plants had been -- have had their awareness of the concerns about making certain that containment isolation is achieved and is not overridden, have they had training in that respect? Do you have any knowledge about that?

A I have no knowledge of this state of affairs. I have not looked into this.

Q Under the possibilities of venting that system into containment, would this involve in your opinion the need to add additional containment penetrations?

A I would like to state at the outset that the way this issue is worded -- this is not particularly the way I would have worded it, so I have not looked into the design modifications that would be required to effect such a proposed solution.

Q Now, in that regard, let me just follow on that point. The issue was originally worded in terms of whether systems identified as contributing to releases at TMI 2 should be considered for vent back into containment. Now, I note at the end of your testimony on Page 15, in your concluding paragraph you suggest, among other things, namely, SMUD should perform an analysis to identify additional potential release paths from containment and to evaluate potential failures of containment isolation.

Am I correct that you are suggesting that SMUD should perform an analysis that goes beyond simply those systems that were identified as contributing to TMI 2 releases?

A I suggest it would be a good idea for SMUD or somone -- I think it is appropriate that the licensee do this kind of analysis, do some analysis beyond. That was apparently --

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that was relied on in the licensee's testimony at the time it was presented, and that is the original FSAR analysis of the containment isolation system.

So, I do not mean to tie it specifically to the sequences -- release paths like TMI, necessarily.

MR. LEWIS: I have no further questions.

MR. COLE: Just one or two questions, Mr. Mann.

BOARD DIRECT EXAMINATION

BY DR. COLE:

Q The contention which you are addressing has to do with venting material that is released from the reactor system that leaves the containment structure to vent that back into the system. Is that correct, sir?

A Well, that is my understanding of the concept that this contention addresses, yes, without being very specific beyond that.

Q Then you make specific reference to a TMI 2 action and in your testimony you describe some of the similarities and some of the differences between TMI and Rancho Seco, and there is a significant difference in the way containment isolation might be achieved at Rancho Seco, and I believe you already testified to that, sir, didn't you?

A Yes.

Q If venting back into containment system involves other openings into the containment system, is it

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possible that these additional openings might create the problems -- some of the problems that they are designed to solve, and have you looked at that, sir?

A Well, let me take the first part of your question first. Yes, theoretically, it is possible that if additional penetrations are required, the very fact of creating additional penetration provides an additional however slight opportunity for another pathway under different circumstances, and the second part is, no, I have not analyzed that.

Q How many different venting systems do you think would be required if they went along with your recommendations on this?

A Well, my recommendations are not necessarily to design and implement such a capability. My recommendation is only to consider it as a possible capability among other actions that might be taken to manage the radioactive materials that might be accumulated in systems outside containment under certain accident sequences.

Q Are you holding yourself to accident sequences, or are you also recommending that on those occasions when these auxiliary systems outside of the containment structure are handling--does the routine you do to waste materials, is that also be vented back into the containment system?

A Well, I had only considered this for those cases that involved some sort of accident or abnormal, if you will,

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accumulation of radioactive materials in those systems, and this was developed in the context of what I learned about Three Mile Island, that event in particular, noting that as far as I know, most of the rad wastes and primary coolant process systems in commercial light water reactors are designed only for -- well, the design basis for those systems, as far as I know, is not for conditions like TMI, for example, but rather for some nominal percentage of failed fuel, on the order of, say, 1 percent, which you could expect under normal operating experience, as it were.

So, what I am really addressing my concerns to are these conditions where you have significantly greater fuel failures, the possibility of degraded cores, and the accumulation of larger volumes of liquids, fluids, from the primary system to deal with. It is a matter of both the amounts of material and the concentrations and types of radioactive materials that you might have to consider dealing with.

Q I would like to get back to the first point that I tried to raise about containment isolation being a line of defense that we might be breaching by introducing more cavities into the system. I have -- Let me start over again on that.

Do you not think that time and effort spent on assuring isolation, adequate isolation, might tend to be more profitable than the system that you are proposing and why do you think it should go by your route, sir?

A Well, if I can, again, I would like to try to clarify what my position is on this issue. It is that various things be considered in view of the potential need to manage amounts and quantities of radioactive materials that could get into these systems outside containment, not necessarily that they automatically be required to be vented back into containment. Somehow this proposed solution of venting back into containment has achieved a status that I personally am not willing at this time to give it just across the board. I think it deserves-- It should be analyzed for each particular facility, taking into account the conditions and the systems, the design and capacities and so forth of those systems at each facility.

Q All right, sir.

A I am not sure I am answering directly your question, but I would like to clear up this apparent --

Q I understand your position, sir. You think we certainly ought to look at it.

A Yes.

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

BY MR. SHON:

Q The major releases at TMI which you said occurred

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later in the incident, and which we are all aware of, long after the four hours and some odd minutes' time when the containment sealed, could you describe what these were and whether or not you think similar releases would be occasioned in Rancho Seco if Rancho Seco had a similar sort of difficulty of any time, that is, a large amount of radioactivity released to the primary system and some perhaps to the pressurizer release tank?

A It is my understanding that the primary pathway from containment at Three Mile Island was through the letdown system, and this was because the operators deliberately chose to operate the letdown system for long periods of time. I am not sure as to the exact length of time, but I believe it was upwards of 10 to 15 hours at least after the accident, and perhaps even longer into the several day time frame, and if that circumstance were to occur at, say, Rancho Seco, I have no reason to doubt that they would experience similar releases under the conditions as stated.

Q In your testimony at Page 15, you urge, "Mainly, SMUD should perform an analysis to identify additional potential release paths from containment, and to evaluate potential failures of containment isolation."

We heard a considerable amount of testimony from the staff to the effect that there is an ongoing analysis of at least release paths and leakage paths and that sort of

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thing that has been demanded of virtually all licensees, and that SMUD is well into their analysis and has identified pumps that leak and valves that need repacking and things on that order.

Is it the kind of analysis you meant, and is it sufficient?

A Well, the last part first. It is difficult for me to know whether it is sufficient, of course, until I would have a chance to review it on a case by case basis. God forbid that I would have to do that, but the first part, I think the wording here is a little bit unfortunate, upon reflection. It would be appropriate to strike the word "additional" from this testimony. I think perhaps that would make the sense of that more clear, because no release paths at Rancho Seco per se have been identified. We are only talking about potential release paths, really, and in view of the Three Mile Island experience, it is suggested that there are several sets of circumstances which could lead to essentially uncontrollable or uncontrolled releases from containment.

That is essentially the thought I would like to make.

Q Well, for example, now, the licensee has identified what these essential and non-essential systems are and made sure that all non-essential systems do

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indeed seal, for example. They have also gone to considerable trouble to identify leakage points.

A Yes.

Q I guess the most important portion of my question is, is what they have done to the extent that you know of what it is or what it is aimed at, sufficient?

A Well, I think those things are steps in the right direction. Certainly the thing that bothers me about this problem is that I think it requires perhaps more analysis of the need or potential need, for example, to operate systems that penetrate containment under certain accident sequences, and this poses a dilemma in my view, the requirement for isolation vis-a-vis the potential requirement to operate certain systems which penetrate containment, and to do this may require the deliberate defeat of containment isolation, and it is a matter that I do not think has been given sufficient analysis.

Q I see. In the matter of discharging pressure relief and process vent systems back into containment, would you envision that there would be a need, for example, to raise the pressure of those fluids sufficiently to discharge them back into containment in the circumstances that would pertain after an accident? Would you need pumps, for example, or blowers of some sort?

A I have not analyzed this situation, but I can relate it to the TMI experience, whereby the pressures that they were experiencing in such systems as the waste gas decay tanks were causing uncontrolled lifting of relief valves and thereby uncontrolled releases into the auxiliary building, the ventillation system, and through the station vent out into the atmosphere.

The pressures at which those relief valves lift, I believe, are in the range of between 50 and 150 psi, and those pressures were certainly greater than the average pressure experienced in the containment building at that time.

In fact, it is my understanding that the staff at TMI attempted to route through let's say an ad hoc procedure, not necessarily using existing systems, but a manually fabricated system. They attempted to route the effluent from some of these release valves back into the containment at TMI, in fact, some time during the days immediately following the accident, but they were unsuccessful.

So, to make a long story short, I believe they felt that they had a pressure head that was sufficient to cause those gases to flow back into the containment under those conditions.

Q That is true under the very special circumstances that existed at TMI 2. However, as you heard this morning and many other times during this hearing, the containment building is designed to take pressures from accident sequences that drive the internal pressure up to 59 pounds or perhaps even 118 pounds. Under those circumstances, would the valves you are talking about, would you not then -- would this not substantially complicate matters and substantially tend to introduce the difficulties that Dr. Cole pointed out, that additional equipment might leak also?

A Yes, I agree. The point I would make in this regard is that I would view such a capability as a discretionary matter requiring a detailed understanding of the circumstances with which the operators were faced, and my position on this matter of venting would be -- even though I am not prepared to suggest that it be required of anyone at this time -- but even if such an analysis were performed which found it to be potentially beneficial, I would recommend that only the capability be available, and whether or not systems would be vented back into the containment would require the deliberation of the senior staff at the facility.

The time frame at TMI, for example, over which this was being considered was in the range of days, for example, and there was plenty of time -- there would have been plenty of time if the information were available to them to consider the alternatives, but the point is that they ran

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out of options to deal with this accumulated material, and they ended up venting it to the atmosphere under conditions that were less than desirable.

Q But you would still require operator consideration and operator decision-making of some sort, thus placing perhaps an additional burden on the operators in a ticklish situation. Is this not true?

A Well, yes, it would require the judgment of the operators, but I would rather have the operators or the facility staff have additional options that they may not have at the present time to manage radioactive materials in a post-accident environment.

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

CHAIRMAN BOWERS: Does CEC have any redirect?

MR. LANPHER: Just one or two.

REDIRECT EXAMINATION

BY MR. LANPHER:

Q In response to one of Mr. Shon's questions, Mr. Mann, you stated you had a concern regarding the understanding of the consequences of operating certain systems after containment isolation, particularly the letdown system. Is one of the analyses that you would like to see performed something along the lines of a failure modes and effect analysis regarding the containment isolation?

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A My feeling is that a system out of analysis -- Let me back up a minute. I do not think that the role of of a containment building under a variety of accident sequences has been adequately analyzed. Now, I would start with a very broad statement of that concern, and by that I mean, in addition to the containment isolation system itself, which is sort of a subsystem of the containment isolation building, looking at what happened at Three Mile Island and possibly the Crystal River sequence, for example. I do not think any analysis has been focused on the role and the potential requirements for performance of the containment building itself and a variety of transients which I would put into a class of severity somewhat less, let's say, than the more severe core melt accident or sequences not necessarily leading to core melt.

WASH 1400, the so-called reactor safety study, did indeed look at the performance of containments for severe accidents, that is, those generally believed to lead to core melt, and they constructed event trees dealing with a variety of situations that would result in containment failures, but I am more concerned about a class of events which challenge various aspects of the containment building but not necessarily of such severe consequences as a core melt.

I think there is a class of events here that

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have not really been thoroughly analyzed in the detail that I think would be appropriate to look at the requirements, the different requirements on maintaining effective containment. Let me put it that way.

Q Would the purpose of these analyses be to identify the possible release paths?

A It would be to do that and it would also be to determine if there are, let's say, weaknesses or potential vulnerabilities or problems in maintaining effective containment of the materials which could be released to the primary coolant system. There are some systems which are part of the containment building that are not under the control of the safety features isolation system, for example, which control or could control the paths of radioactive materials from certain accidents.

For example, events which involve the leakage -- let's say leakage of primary coolant into the secondary system, for example. This was a pathway at TMI which is not generally appreciated, and it was not large in comparison to the primary pathways that have already been discussed, but once you get radioactive material, for example, into the secondary system, there are certain pathways that are potentially available that are not under the control of SFAS, Safety Features Actuation System, control of containment isolation system.

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So, at Crystal River, for example, apparently the operators initially thought, due to the signals they were receiving, that they had a potential or a real steam line break accident, so the operator diagnosis of the situation as required and different actions would be necessary to preclude the release of materials into the containment than would be the case if it were the kind of accident that we are talking about at Three Mile Island, for example, where the release route is somewhat different, and it is normally under the control of the safety features operated containment isolation system.

So, I am suggesting this system -- the containment building ought to be looked at in terms of an event based analysis as opposed to the traditional method of analysis where you look at penetration by penetration and determine if indeed you have isolation redundancy in terms of individual systems or individual components, and you would want to look at the possibility in my view of common load failures which could compromise the ability of containment to isolate on demand.

I do not think this has been done for the Rancho Seco facility, for example.

MR. LANPHER: No further questions.

MR. BAXTER: No further questions.

MR. LEWIS: No further questions.

CHAIRMAN BOWERS: We have no further questions.

MR. LANPHER: May the witness be excused?

CHAIRMAN BOWERS: Any objection?

(No response.)

CHAIRMAN BOWERS: The witness is excused.

(Witness excused.)

MR. BAXTER: At this time, Mrs. Bowers, licensee would call Mr. Rodriguez to the stand.

end 9
Bob follows

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bfml

Whereupon,

RONALD A. RODRIGUEZ

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

DIRECT EXAMINATION

BY MR. BAXTER:

Q Mr. Rodriguez, I call your attention to a document bearing the caption of this proceeding, dated February 11, 1980, entitled in part "Licensee's testimony of Ronald J. Rodriguez," and consisting of 54 pages and three appendices labelled I, II, and III.

Is this document I described testimony which has been prepared by you or under your direct supervision for presentation at proceeding?

A Yes.

Q Do you have any changes or corrections to the tesimony?

A No.

Q Is the testimony true and accurate to the best of your knowledge and ability?

A Yes, it is.

MR. BAXTER: Mrs. Bowers, I move the admission of Mr. Rodriguez's testimony and ask that it be physically incorporated into the transcript as if read.

CHAIRMAN BOWERS: Mr. Ellison?

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MR. ELLISON: No objection.

CHAIRMAN BOWERS: Mr. Lewis?

MR. LEWIS: No objection.

CHAIRMAN BOWERS: The document which you have identified will be physically incorporated into the transcript as if read and admitted into evidence.

(The document referred to follows.)

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February 11, 1980

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
	)	
(Rancho Seco Nuclear Generating Station)	)	

LICENSEE'S TESTIMONY OF

RONALD J. RODRIGUEZ

IN RESPONSE TO

LICENSING BOARD QUESTIONS CEC 1-2, 1-6, 1-7, 5-3a,

CALIFORNIA ENERGY COMMISSION ISSUES 1-1, 1-12, 3-1, 3-2, 3-3,

LICENSING BOARD QUESTIONS E-C 22, 31, 32, 34,

FRIENDS OF THE EARTH CONTENTIONS III(d), III(e), AND

ADDITIONAL BOARD QUESTIONS 2 AND 3

TABLE OF CONTENTS

		Page
I.	PROFESSIONAL QUALIFICATIONS AND INTRODUCTION....	3
II.	TRAINING..	7
III.	CONTENTIONS AND BOARD QUESTIONS.................	19
	A. Operator and Facility Management Competence (CEC 3-1, H-C 32, FOE III(d), FOE III(e))...	19
	B. Small Break LOCAs (CEC 1-7, ABQ 2, ABQ 3)...	25
	C. Emergency Procedures (CEC 3-3)..............	32
	D. Feedback on Operating Experience (CEC 3-2)..	34
	E. Training Unlicensed Operators (H-C 34)......	36
	F. Control Room Configuration (H-C 31).........	39
	G. Instrumentation (CEC 5-3a)..................	41
	H. Pressurizer and Reactor Vessel Level Indication (H-C 22).........................	46
	I. Auxiliary Feedwater System Reliability (CEC 1-6).................................	49
	J. Safety System Challenges (CEC 1-1, CEC 1-12)	50
	K. Natural Circulation (CEC 1-2)...............	52
IV.	CONCLUSION......................................	54
	Appendices I, II and III	

I. PROFESSIONAL QUALIFICATIONS AND INTRODUCTION

Q. What is your name and business address?

A. My name is Ronald J. Rodriguez. My business address is Sacramento Municipal Utility District, Post Office Box 15830, Sacramento, California 95813.

Q. What is your position, educational background and work experience?

A. I am Manager, Nuclear Operations, for the Sacramento Municipal Utility District.

I graduated from the United States Naval Academy with distinction in 1959 with a Bachelor of Science degree in Naval Science. I was a commissioned naval officer for eight years, serving previously in the Navy Submarine Program. I completed the Navy Nuclear Power Graduate-Level Technology Course and served for 41 months in engineering positions on-board an operating nuclear-powered submarine. My areas of responsibility varied among reactor plant systems, main propulsion systems, electrical and reactor instrumentation systems, chemistry, and health physics control for maintenance and personnel protection. For two years I was the training officer responsible for training Navy Nuclear Power Officer and Enlisted Candidates at the Navy's training facility in Windsor, Connecticut. This assignment gave me responsibility for administration of all classroom training, in-plant training progress, and final examination for qualification as Nuclear Power Plant Operators and Watch Officers. My final assignment in the Navy was Chief Engineer of a nuclear-powered submarine. In this capacity I had responsibility for the administration of the Engineering Department, including the qualification and training of Engineering Watch Officers, Reactor Operators and power plant watch standers.

I joined the staff of the Sacramento Municipal Utility District in 1968 and served as Assistant Superintendent for Nuclear Operations until February, 1970. In this position I was responsible for establishing the initial phases of the Rancho Seco Operating Training Program and the selection and hiring of plant operating personnel.

From February, 1970, until January, 1978, I was Plant Superintendent, with direct responsibility for the testing and startup program for Rancho Seco, including the staffing for operations, technical support and maintenance. I was also responsible for the overall direction of vendor personnel assisting in the startup program, and served as chairman of the group established to provide final approval of the functional test program.

Since January, 1978, I have been Manager of Nuclear Operations, with department-level responsibility for the safe and proper operation of Rancho Seco Nuclear Generating Station.

I have completed the six-week Babcock & Wilcox reactor technology course and a ten-week Nuclear Steam Supply System

Simulator Training Program presented by Babcock & Wilcox. I have participated in the entire Licensing Training Program at Rancho Seco, and currently hold a Senior Reactor Operator License issued by the NRC.

I am a member of the American Nuclear Society's Reactor Operations and Support System Management Committee.

Q. What is the purpose of your testimony?

A. The purpose of my testimony is to respond to those contentions raised by the California Energy Commission and Friends of the Earth and questions posed by the Licensing Board with respect to the competence of Rancho Seco facility management and operators, the emergency and other operating procedures employed at the plant, control room configuration and instrumentation at Rancho Seco, and the actual performance of plant systems in response to feedwater transients. My testimony will show that, contrary to the contentions asserted by Intervenors and in answer to questions raised by the Board, there is reasonable assurance that the plant and its personnel will respond safely to feedwater transients.

My testimony is divided into two major sections. Because a number of the issues raised in this proceeding concern facility management and operator competence, I will first present a comprehensive description of the training provided to Rancho Seco personnel. This testimony will address the training provided to the initial licensed personnel in preparation for fuel loading at Rancho Seco in 1974, the current licensing training program, the requalification program, special training following the Three Mile Island accident, and training provided to unlicensed operators on the operation of the auxiliary feedwater system. The second section of my testimony responds specifically to contentions raised by the parties and questions posed by the Licensing Board.

II. TRAINING

Q. Please describe the training which was provided from 1970 through 1974 to the personnel initially licensed to operate Rancho Seco and to the current site management personnel.

A. Appendix I to this testimony describes the very extensive training program undertaken during this period to ensure that the initial operational staff and the facility management were fully prepared to start up, test and operate the plant during the normally complex initial phases of operation.

Q. Please summarize the training program which Sacramento Municipal Utility District now uses to prepare operator candidates for licensing.

A. Appendix II to this testimony summarizes the content of the training program currently used by the District to prepare operator candidates for licensing by the Nuclear Regulatory Commission. Candidates eligible for this training program normally have been employed in the Operating Division at Rancho Seco for two or more years. Individuals eligible for this training program are selected for participation on the basis of a math and science written examination, an interview and an evaluation of previous work performance.

As summarized in Appendix II, the licensing training program for prospective reactor operators is a comprehensive academic and practical program which is divided into four major parts. The first part, the academic phases, is aimed at assuring that the candidate has basic skills in mathematics, and an understanding of classical physics, atomic and nuclear physics, and physics directly related to the reactor core. It includes reactor theory and reactor operations reviews. The Related Technologies Course provides instruction in instrument and controls fundamentals for reactor coolant system non-nuclear instrumentation, balance-of-plant non-nuclear instrumentation, nuclear instrumentation, reactor protective system fundamentals, safety features actuation system fundamentals, the integrated control system, and control rod drive control system. This course also includes instruction in chemistry, health physics, and radiation protection.

The second part of the training program, the in-plant phases, involves actual in-plant operations training. This includes systems and operations training in the Rancho Seco control room, the application of procedures to systems during Rancho Seco control room operating experience, and fuel handling training. This portion of the program provides the candidate with the opportunity to use Rancho Seco systems first hand by utilizing those systems under operating conditions while standing control room watches under the instruction of licensed personnel.

The third part of the program, the simulator training phases, consists of a pre-simulator review course and the simulator operations course. The pre-simulator review course, normally conducted following on-shift instruction, reviews topics covered during the academic phase, with primary emphasis on reactor theory, nuclear instrumentation, major non-nuclear instrumentation systems, the integrated control system, the control rod drive system, and start-up procedures. The simulator operations course is conducted at the Babcock & Wilcox simulator in Lynchburg, Virginia. The B&W simulator is very similar in design and layout to the Rancho Seco control room. The arrangements of controls, the types of controls in the areas that deal directly with feedwater control and reactor coolant system control, are essentially identical to those at Rancho Seco. The course is comprised of 60 hours of classroom presentations and 60 hours of actual simulator operation. The simulator operations course begins with an initial introduction to and familiarization with the simulator control room, reactor startups, and power operations up to 100 percent power. In the second week, the course is expanded to plant operations with malfunctions, including feedwater pump trips, reactor coolant pump trips, load rejections, and instrument malfunctions. The third week continues with power operations in both the manual and automatic control modes, and additional malfunctions for which the operator must take mitigating actions are introduced. The final portion of the simulator operations course involves an operating examination.

The fourth part of the program, the license preparation phases, includes an additional period of control room operating experience and a pre-license review course. While examinations are given throughout the various phases of the roughly one-year training program to test the candidate's retention and progress, a comprehensive oral and written examination is administered by the District to the prospective licensed operator after the pre-license review course. The candidate's performance on these audit exams is reviewed by the training supervisor and facility management. If the candidate passes the audit examination, the District then certifies to the NRC that the licensing candidate is prepared to take the license examination. Prior to the NRC examination, the candidate receives a final in-plant briefing on recent changes within the facility and its current operating status.

Requirements for approval of the operator license application are set forth in the NRC's regulations at 10 C.F.R. § 55.11. The scope and content of the NRC's written examinations and operator tests are set forth at 10 C.F.R. §§ 55.20 through 55.23. Requirements for the renewal of licenses are set forth at 10 C.F.R. § 55.33 and include successful completion of the Rancho Seco requalification program.

Q. What is the Rancho Seco requalification program?

A. The requalification program for licensed personnel is conducted continuously and on a two-year cycle, and follows the requirements of Appendix A to 10 C.F.R. Part 55. This program consists of annual written examinations, regularly scheduled lectures, assigned individual study, on-the-job training including reactor control manipulation, and observation and evaluation during annual simulator training which includes drills on emergency and abnormal conditions.

During the course of the two-year cycle an average of 60 hours of lectures are scheduled to accommodate all licensed operating personnel. The following general subjects are included in the lecture series:

1. Theory and Principles of Plant Operation.
2. General and Specific Plant Operating Characteristics, including operational limitations, precautions and set points.
3. Plant Instrumentation and Control Systems.
4. Plant Protection Systems including the Emergency Plan and Security Plan.
5. Engineered Safety Systems.
6. Normal, Casualty and Emergency Operating Procedures.
7. Applicable portions of the Quality Assurance for Nuclear Operations Manual.
8. General Safety, First Aid, and Radiation Control and Safety.

9. Technical Specifications.

10. Special Plant Evolutions such as Major Maintenance, Refueling, Special Tests, etc.

11. Changes in Equipment and Operating Procedures.

12. Applicable portions of Title 10, Chapter 1, Code of Federal Regulations,

13. TMI-2 Incident and Lessons Learned.

Subjects typically covered in the individual study assignments include:

1 Facility Design Changes.

2. Procedure Changes.

3. Facility License Changes.

4. Operating Procedures.

5. Emergency Plan.

6. Radiation Protection Procedures.

On-the-job training as part of the requalification program includes plant control manipulations involving reactivity changes to demonstrate the operator's skill and familiarity with reactivity control systems. These manipulations may include:

1. Plant startup or shutdown with any ICS station in manual.

2. Adjustments of control rods to compensate for transient conditions (power changes greater than 10%).

3. Startup or shutdown of a reactor coolant pump with the reactor critical.

4. Turbine stop valve exercising or testing.

5. Reactor operations involving emergency or special procedures where reactivity is changing.

6. Changing boron concentration to compensate for shutdown margin, transients or core age.

7. Refueling operations.

8. Reactor Physics Testing.

This training also requires that each licensed operator manipulate the controls a minimum of ten times during the term of the license. Each licensed senior operator is required to manipulate the controls or direct the activities of operators during control evolutions a minimum of ten times during the term of the license.

An annual one-week simulator course at the B&W facility is also a part of the requalification program for licensed operators. The simulator course consists of 20 hours of classroom lectures and 20 hours of simulator operations training. The simulator training provides the opportunity for participation as a control room operator and as a supervisor of control room operators. Consequently, at various times the operator participates in the details of manipulating controls, observes the overall transient, performs evaluations based on instrumentation information, and provides directions to those doing control manipulations.

During these courses multiple failure accidents have been imposed and the operator has been given the opportunity to exercise his diagnostic skills and training in mitigating the consequences of those multiple failure accidents. The simulator has the capability of introducing over sixty individual casualties in the various reactor plant systems. The specific systems which are covered in these casualties include the coolant makeup system, the reactor and its instrumentation, the reactor coolant system, the steam and turbine system, the condensate and feedwater system and various auxiliary systems. The individual casualties can be combined to create multiple failure scenarios and to present the operator with a complex problem in which to practice his training and diagnostic skills. The programming available at the simulator also permits the instructor to fail equipment sequentially and thereby allows full exercise of the operator's training. This tests the operator's skill and abilities to make initial diagnosis of a failure, begin corrective action, discover another failure, and then exercise some alternative corrective method to keep the unit in a safe condition. Credit is given for manipulation at the simulator for the purpose of meeting the minimum training requirement, referred to above, for reactivity control manipulation.

Written examinations are administered by the District at 11 to 13 month intervals as a part of the requalification training program. The examinations are similar to those administered by the NRC and are used to determine the operator's knowledge of the subjects covered during requalification training, operating and emergency procedures, and to determine areas in which retraining is needed.

Q. Was any special training provided subsequent to the accident at Three Mile Island?

A. Yes. One of the short-term actions which the District agreed to perform promptly after the accident at Three Mile Island and prior to the resumption of operation at Rancho Seco was item (e) of the Commission's Order of May 7, 1979: "Provide for one Senior Licensed Operator assigned to the control room who has had Three Mile Island Unit No. 2 (TMI-2) training on the B&W simulator." In addition, one of the long-term modifications proposed by the District, and which the Commission directed, in its Order of May 7, 1979, be accomplished as promptly as practicable, is stated as follows in the Order:

> "The licensee will continue operator training and have a minimum of two licensed operators per shift with TMI-2 simulator training at B&W by June 1, 1979. Thereafter, at least one licensed operator with TMI-2 simulator training at B&W will be assigned to the control room. All training of licensed personnel will be completed by June 28, 1979."

Both of these modifications have been accomplished.

Special B&W simulator training was conducted for Rancho Seco licensed operators between April 20 and June 22, 1979. The purpose of this training was to thoroughly acquaint them with the indications expected during an accident similar to the multiple failure accident that occurred at Three Mile Island. This training included classroom discussions of the basic underlying causes of the accident, a description of how the plant's parameters changed during the course of the accident, and, finally, how the accident was terminated. Emphasis was placed on the seriousness of failure to maintain subcooling and on the verification of subcooling and natural circulation. In the simulator the accident initially was demonstrated to allow operators to observe the course of the various plant parameters. Then the accident was again simulated allowing the operators to exercise control to mitigate and stop the accident before it reached conditions in which core damage occurred.

In addition to the simulator training, group discussions were conducted by the Rancho Seco Operations Supervisor with each operating crew during the period between March 28 and approximately May 30, 1979. These discussions addressed the sequence of events at Three Mile Island, reviews and procedure changes required by the NRC IE Bulletins, saturated and subcooling operations curves, safety features actuation system operation, auxiliary feedwater system operation, control of the reactor trip relay which provides for reactor trip on turbine trip or loss of both feedwater pumps, clarification of technical specifications, and requirements for notification of the NRC.

Training was conducted between April 10 and April 30, 1979, by the Rancho Seco training supervisor for all operators. This training was conducted to upgrade the understanding of the TMI accident and its cause, the voiding phenomenon, procedure changes made to reflect the lessons learned from the TMI accident, natural circulation phenomenon and changes to the plant that were contemplated or actually being made. It specifically emphasized the subject matter of NRC IE Bulletins 79-05A and 79-05B.

Informal discussions were given to operating crews by NRC inspectors between April 10 and April 30, 1979, to cover the same general areas that had been addressed by the Rancho Seco training supervisor. The purpose of this program was to assure that the operating personnel understood the instructions relating to the TMI accident.

Informal training was given by each Shift Supervisor to his crew on plant modification and procedure changes. This training included a plant walk-through to assure familiarity with the location of active components in the auxiliary feedwater system. This training was conducted between April 14 and April 17, 1979.

Formal training was conducted by General Physics Corporation, a consultant to the District, with respect to the TMI accident scenario, small break LOCAs, plant modifications made as a result of TMI, procedural changes to mitigate the consequences of a small break LOCA, void formation theory, and initiation and recognition of natural circulation. This training was conducted between June 8 and June 15, 1979.

Documents distributed to operators to acquaint them with the training and instructions were included with Standing Orders 5-79 through 15-79. In addition, a post-TMI 2 training supplement was issued by the training supervisor to all licensed operators.

The District administered written examinations to the licensed operators on the Three Mile Island training provided. The examination addressed the TMI-2 accident in the following areas:

1. Identification of human, design, and equipment failures that resulted in core damage.
2. The concept of subcooling and the effect on vessel integrity.
3. Procedure changes resulting from lessons learned from the TMI accident.
4. Natural circulation detection and operation.
5. Specific plant modifications.

Appendix III to this testimony contains a summary of the special post-TMI training provided to Rancho Seco operators.

III. CONTENTIONS AND BOARD QUESTIONS

A. Operator and Facility Management Competence

Issue CEC 3-1:	Whether personnel adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of its operation?
Board Question H-C 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.
Contention FOE III(d):	The NRC orders in issue do not reasonably assure adequate safety because no procedures have been taken to assure facility management competence.
Contention FOE 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.

Q. One of the contentions raised in this proceeding questions the competence of facility management. Please describe the training provided to, and the qualifications of the Rancho Seco management personnel.

A. The Manager of Nuclear Operations, Plant Superintendent, Engineering and Quality Control Supervisor, Chairman of the Plant Review Committee, and Operations Supervisor each have a senior reactor operator license issued by the NRC. Prior to the initial startup of Rancho Seco these management personnel all participated in the extensive licensing training program described in Appendix I to my testimony, including the examinations I described above in my testimony on licensing training. Since initial licensing by the NRC in 1974, these management personnel have also participated in the Rancho Seco requalification training program and in the special Post-TMI training, both of which are described above in my general testimony on training for Rancho Seco operators. Consequently, as licensed senior reactor operators the facility management personnel at Rancho Seco maintain a high level of competence and participate directly in the safe operation of the plant.

The Plant Superintendent and I also have been active in industrial organizations dealing with plant activities at facilities across the country. This participation increases knowledge of experience with and improvements in plant management at other units.

Most recently, management and supervisory personnel have begun participation in a command and control training program being presented by a consultant to the District. The purpose of this program, which will be completed in 1980, is to provide management and supervisory personnel with additional training in the command and control aspects of mitigating various accidents.

Q. In your view is the Rancho Seco facility management competent, such that there is reasonable assurance the plant will respond safely to feedwater transients?

A. Yes. The training, testing and experience of Rancho Seco facility management as senior licensed reactor operators refute the statements in Board Question H-C 32 that management competence has not been tested and evaluated and in Friends of the Earth Contention III(d) with respect to facility management competence.

Q. You have mentioned licensed reactor operators and senior licensed reactor operators. What is the difference between these licenses?

A. The Nuclear Regulatory Commission regulations governing operators' licenses define an "operator" as any individual who manipulates a control of a facility. An individual is deemed to manipulate a control if he directs another to manipulate a control. The NRC defines a "senior operator" as any individual designated by a facility licensee under 10 C.F.R. Part 50 to direct the licensed activities of licensed operators. These definitions may be found at 10 C.F.R. § 55.4. As indicated in 10 C.F.R. Part 55, the NRC administers different examinations for these two classes of operator licenses.

There are currently 18 senior licensed operators and 4 licensed operators employed at Rancho Seco. Eight of the senior licensed operators do not normally stand control room watch, but serve in various supervisory and facility management positions. Each crew assigned to an 8-hour control room watch includes three licensed personnel. The control room operator holds an operator's license, while the shift supervisor and the senior control room operator each hold senior operator licenses. The NRC currently requires that two licensed personnel be in the control room at all times during plant operation.

Q. Licensing Board Question H-C 32 states, among other things, that some employees operating Rancho Seco have never been tested because of grandfathering. Is this true?

A. Absolutely not. I have already described the examinations administered to Rancho Seco operators by the District and the NRC. There is no provision of NRC regulations or Rancho Seco administrative procedures which provide for "grandfathering" licensees in lieu of testing.

Q. That question also states that employees have not been tested for responsible and appropriate actions and judgment decisions during a loss of feedwater transient. Do you agree?

A. No. Operators have been tested by the District, its contractors and the NRC on responses to loss of feedwater transients. This testing, which I have described earlier in my testimony, occurred during the licensing training program, the requalification program and the special post-TMI training. The resumption of operation at Rancho Seco following the Commission's Order of May 7, 1979, was based in part on an audit by the NRC Staff to determine that the operators adequately understood the TMI-2 incident, result.'t design and procedure changes at Rancho Seco, diagnosis of and response to small break accidents. Consequently, the operators have been tested to assure that they have an adequate understanding of the consequences of a feedwater transient and the procedures to be followed to mitigate those consequences.

Q. Do operating personnel at Rancho Seco adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of its operation?

A. Yes. The academic, or first part, of the licensing training program described earlier in my testimony includes mathematics, nuclear reactor p.ysics, and other fundamentals of nuclear technology. This course lasts approximately 15 weeks. The ongoing requalification program also includes instruction in the theory and principles of plant operation. Operators are examined in both of these programs to assess their understanding of nuclear technology fundamentals. In addition, the District has emphasized, in the special training provided after the Three Mile Island accident and in subsequent training and communications to operators, the underlying bases for design and procedural changes to enhance the operator's understanding of plant operations.

For all of the foregoing reasons presented in my testimony, it is my opinion that, contrary to Friends of the Earth Contention III(e), the operators at Rancho Seco are competent to respond safely to feedwater transients. I should also observe that this competence has been demonstrated in five years of commercial operation at Rancho Seco, during which operators have been called upon to respond to equipment failures and successfully exercised their sound understanding of plant design, operation and procedure.

B. Small Break LOCAs

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?

Additional Board Question 2: We note (letter D. Ross to J. J. Mattimoe, December 14, 1979) that there is still some dispute as to the fundamental logic for Reactor Cooling Pump (RCP) trip in a small break LOCA.

a. What current instructions to reactor operators govern tripping of the pumps in small break LOCAs and upon what theory of system behavior are those instructions based?

b. What are the implications for safety of operating Rancho Seco until the exact behavior of the system in a small-break LOCA is well understood?

Additional Board Question 3: It appears from a Board Notification issued by R. H. Vollmer on December 5, 1979, that the basic design of the Once Through Steam Generator (OTSG) may so closely couple primary system behavior to secondary system disturbances that gross disturbance of the primary system is inevitable for feedwater transients. Further, it seems there are situations in which an operator may not be able to tell exactly what is wrong or what response is appropriate (e.g. overcooling *vis-a-vis* a small-break LOCA).

a. What changes in the system and procedures have been made to ameliorate this situation?

b. What are the implications for safety of operating Rancho Seco before any uncertainties are resolved?

Q. Item (d) of the short-term actions to be implemented pursuant to the Commission's Order of May 7, 1979, required completion of analyses for potential small breaks and the development and implementation of operating instructions to define operator action. Have such operating instructions been developed and implemented?

A. Yes. Babcock & Wilcox conducted an analysis of small break loss of coolant accidents and developed operating guidelines on the basis of this analysis. The District then applied these guidelines to Rancho Seco, developed and implemented procedure changes to define operator action. These instructions were reviewed and approved by the NRC Staff prior to the resumption of operation at Rancho Seco on July 5, 1979.

The B&W small break analysis, and subsequent analyses B&W conducted at the request of the NRC Staff, are described in Licensee's Testimony of Bruce A. Karrasch and Robert C. Jones.

Q. Briefly, hat procedural changes have been instituted as a result of these new small break analyses?

A. Rancho Seco Nuclear Generating Station Procedure D-5, "Loss of Reactor Coolant, Reactor Coolant Pressure," has been changed to identify specific additional operator actions to be performed in the event of a loss of c'olant accident.

The overall procedure identifies the possibility of a stuck open electromatic operated relief valve as a potential leak source. In the immediate action requirements of this procedure, strong emphasis is placed on maintaining reactor coolant system pressure-temperature relationships to assure that a subcooling condition of at least 50 degrees F. exists. Specifically, the procedure requires that upon automatic initiation of high pressure injection all reactor coolant pumps are tripped and high pressure injection shall not be terminated unless: (1) low pressure injection pumps are in operation and flowing at a rate of not less than one thousand gallons per minute each and the situation has been stable for twenty minutes; or (2) all hot and cold leg temperatures are at least 50 degrees F. below the saturation temperature for the existing reactor coolant system pressure and the hot leg temperatures are not more than 50 degrees F. greater than the secondary side saturation temperature. If 50 degrees F. subcooling cannot be maintained, the high pressure injection system shall be reactivated.

Operating Procedure B-6, "Plant Shutdown and Cooldown," has been modified to specifically address additional operator action to be taken in a small break accident condition with a loss of forced circulation. This procedure directs the operator to verify that auxiliary feedwater is supplying steam generators with feedwater and that steam generator levels are being maintained at 50 percent on the operating range. The procedure further specifies reactor coolant system differential temperatures which confirm natural circulation and identifies differential temperatures at which the operator must take additional action to improve natural circulation flow. The procedure also provides specific direction to the operator in the event that natural circulation cannot be confirmed. This action provides for core cooling by utilizing high pressure injection into the reactor coolant system and releasing energy via the electromatic operated relief valve until either a reactor coolant pump can be restarted or natural convection flow can be re-established.

Q. Board Question CEC 1-7 is concerned with whether training actions for small break LOCAs give sufficient attention to providing appropriate analytical bases for operator actions. What attention is devoted to this subject?

A. In my testimony above at pages 15-18, I described the special training which was provided to Rancho Seco operators following the accident at Three Mile Island. This training gave a great deal of attention to recognition and understanding of the symptoms unique to small break conditions and the reasons for immediate operator actions to mitigate the consequences of small break accidents. The subsequent written examination administered by the District and the oral audit examinations by General Physics Corporation and the NRC Staff confirmed that operators understood the bases for the actions to be taken in response to a small break loss of coolant accident.

Q. Additional Board Question 3 in part asks whether there are situations in which an operator may not be able to tell exactly what the nature of a disturbance is or what response is appropriate. Particular reference is made to distinguishing between an overcooling condition and a small break LOCA. How are the operators able to diagnose such situations?

A. Auxiliary feedwater flow measuring instrumentation has been installed at Rancho Seco to provide the operator with an additional means of confirming auxiliary feedwater flow and a capability of diagnosing the rate at which the auxiliary feedwater is flowing. Main feedwater flow instrumentation in the control room was part of the initial design of the unit.

Procedure D-5, "Loss of Reactor Coolant/Reactor Coolant System Pressure," and Procedure D-14, "Loss of Steam Generator Feedwater," address the actions licensed operators should take to assure that adequate core cooling is maintained. In the symptomatic description of Procedure D-5, dealing with loss of reactor coolant (small break LOCAs), the operator is provided guidance with the statement "system pressurizer level and/or reactor coolant system pressure decreasing without associated decrease in coolant average temperature." This symptom is indicative of loss of reactor coolant transients and distinguishes them from overcooling transients.

Oversupply of either main feed or auxiliary feedwater to the steam generators, for the corresponding reactor power or available decay heat, will cause the average reactor coolant temperature to decrease and the resulting density increase of the reactor coolant will result in a decreasing pressure. The operator's actions to a decreasing pressure condition is dependent upon whether or not that pressure condition is associated with a decreasing or essentially stable average reactor system temperature. Since the loss of inventory will adversely affect core cooling to a greater extent than will the overcooling condition, the operator is directed to assume that a loss of coolant accident is in progress until he can establish the exact cause. This direction is provided in Procedure D-5. Steam generator level alarms and feedwater flow indications in the control room are the diagnostic tools the operator can use to determine quickly whether or not an overfeed condition does exist. The action to stop an overfeed or overcooling transient is simply to close off the appropriate valve or valves.

The instrumentation available to the operator in the control room, as well as the availability of valve and pump controls in the control room, provides assurance that overcooling conditions can be recognized and readily controlled. Training provided to operating personnel in the use of these controls and their knowledge of feedwater flows for full power and steam generator levels for decay heat removal situations further provides them with the diagnostic tools to evaluate the overfeed or overcooling conditions. The training, available instrumentation and control design provide an adequate margin of safety to operate Rancho Seco and safely mitigate the consequences of either a loss of coolant or an overcooling condition.

C. Emergency Procedures

Issue CEC 3-3: 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. How are emergency instructions communicated to plant personnel?

A. Emergency Procedures at Rancho Seco are kept in a single volume red binder, distinct from other plant procedures. Changes to the emergency procedures are issued to operating personnel through the Rancho Seco Special Order program. The Special Order procedure requires that each shift supervisor discuss with his operating crew the content of each special order issued. The shift supervisor must document that this discussion was accomplished. Each licensed operator must review the emergency procedure change and attest by his initials completion of that review. The emergency procedures are always available to the operators in the control room for easy and quick reference.

Q. How does the District ensure that the emergency procedures are understood?

A. Any questions which may arise during an operator's review of emergency procedure changes may be directed to the shift supervisor or the Operations Supervisor, who will

'~ve already discussed the implications of the change with the shift supervisor. The purpose of these review discussions is to assure that the operators understand the implications of the procedures and the reasons for changes and additions.

The emergency procedures are also the subject of training in the requalification program, where operators will practice procedures during simulator training and be selectively tested in the written examinations.

D. Feedback on Operating Experience

Issue CEC 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. Are operating personnel made aware of significant new information pertinent to plant operation and, in particular, of significant experience at other reactors?

A. Yes. Operating personnel are made aware of significant events occurring at Rancho Seco and other reactors through a variety of means. Information related to the plant's safe operation and ability to respond to transients is normally supplied to the Rancho Seco facility staff by recommendations from vendors, NRC Information Bulletins and Circulars, or as a result of Rancho Seco's operating experience.

For those significant events which occur at Rancho Seco, the report prepared for submittal to the NRC is reviewed, and if this report pertains to plant operating conditions, a copy is provided to the operating crews through the Special Order program I described earlier. Information is screened by management personnel, and those items determined to be pertinent may be reflected in revisions to operating procedures, communicated through the Special Order program, or distributed in memorandum form for reading and information.

Events which occur at other units and come to the attention of facility management, and which are deemed to be directly pertinent to Rancho Seco operation can also be communicated to operating crews through the Special Order program. The periodic issue of licensee event reports (LERs) by the NRC is distributed to the Plant Superintendent and the Operations Supervisor. The Electric Power Research Institute's Nuclear Safety Analysis Center is investigating the establishment of a program to provide experienced reviewers of LERs for the purpose of categorizing those reports applicable to a particular facility or class of facilities. These screened LERs would then be forwarded to the appropriate operating licensees. In addition, Babcock & Wilcox independently produces a weekly summary of occurrences at B&W reactors. These summaries are provided to the operating crews.

An additional means of making operating crews aware of significant events is through the requalification program conducted on site by the training organization. The annual simulator training program provides the opportunity for B&W simulator instructors to discuss and for operators to practice transients which have occurred at other plants.

Finally, a more informal but nevertheless effective means employed to communicate important new information is through short lectures by the Operations Supervisor just before or during an operating crew's shift.

E. Training Unlicensed Operators

Board Question H-C 34: Rancho Seco, being a Babcock and Wilcox designed reactor, has not adequately trained unlicensed operators to respond to orders necessary for action which would be required in the event of loss of feedwater transient, and therefore is unsafe and endangers the health and safety of Petitioners, constituents of Petitioners and the public.

Q. What action, important to safe plant operation, might be required of unlicensed operators in the event of a loss of feedwater transient?

A. The design of the auxiliary feedwater system at Rancho Seco includes the concept of operating that system from the control room. In the event of a loss of feedwater transient which requires the operation of the auxiliary feedwater system, trained licensed operating personnel available in the control room would be called upon to operate the pumps and valves necessary to assure that adequate flow of feedwater is available to each steam generator. The multiplicity of pumps and valves operable from the control room allows a licensed operator to alter the mode of operation in the unlikely event that an individual component fails to respond properly. These licensed operating personnel can assume manual control in the control room if automatic control systems fail.

The condensate storage tank supplies water for the auxiliary feedwater system. Under normal conditions, the supply water to the condensate storage tank would be replenished as the reactor coolant system cooled down. In the event that the condensate storage tank level reached a minimum of three feet, then additional valving would be undertaken to provide off-site water supply to the auxiliary feedwater system. The minimum water level in the condensate storage tank is 250,000 gallons, which will assure more than a 24-hour supply to the auxiliary feedwater system before manual valving would be necessary to align the off-site water supply to the auxiliary feedwater pumps. These manual valves are located outside the control room and adjacent to the auxiliary feedwater pumps and may be manipulated by unlicensed operators.

Q. Have unlicensed operators been trained to perform this manual valving to align the alternative off-site water supply to the auxiliary feedwater system?

A. Yes. Since the Three Mile Island accident each shift supervisor has conducted specific training for the unlicensed operators on his crew to assure that they can locate and reposition the valves in the unlikely event that they are directed to do so to assure an adequate supply of auxiliary feedwater. This training included a "walk through" by the shift supervisor to affirm the location and operation of the valves.

Unlicensed operators have also been instructed in the proper procedure for taking local control of the auxiliary feedwater system control valve to each steam generator. This training was to assure that, in the unlikely event of a loss of control to all four of the available auxiliary feedwater level control valves, unlicensed operators would be knowledgeable in the location and operation of those valves in the event they were required to operate valving to assure continued flow of auxiliary feedwater to the steam generators.

I should note that the requirement to use unlicensed operators to operate the auxiliary feedwater system is extremely remote and would require a multiplicity of component failures. Even under these conditions, the extent to which unlicensed operators would be required is limited to the manipulation of a small number of manual valves.

F. Control Room Configuration

Board Question H-C 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 Petitioners and the public.

Q. Does the design or configuration of a nuclear power plant control room have any bearing upon the avoidance of a loss of feedwater transient?

A. No. In the overall design of any power station, the actual equipment necessary to provide feedwater to steam generators is located outside the control room. The Rancho Seco control room design provides for operator control of key portions of equipment, but does not allow for immediate control room operator access to the major equipment itself. I am not aware of any control room configuration which will enable the operator to avoid a loss of feedwater transient.

Q. Is the control room, however, designed adequately to enable the operator to respond to, and to mitigate the consequences of, a loss of feedwater transient?

A. Yes. The control room design provides the instrumentation, immediately available, which the operator needs to diagnose a loss of feedwater transient. The design incorporates automatic features to assure adequate cooling of the reactor core. The instrumentation and equipment controls are adequate to allow the operator to control both the normal and auxiliary feedwater systems.

The Rancho Seco control room configuration includes a compact set of control consoles which allow operating personnel quick access to controllers for a wide variety of equipment. The overall control room layout minimizes the amount of movement the operator must make in taking actions involving multiple pumps and valves.

The instrumentation configuration in the Rancho Seco control room also provides for automatic starting of the auxiliary feedwater system under the following conditions:

1. Loss of both normal main feedwater pumps.
2. Loss of all four reactor coolant pumps.
3. Initiation of safety features actuation system.

These automatic initiation features reduce the dependence on immediate operator action for the conditions described. Operator familiarity with the location and operational characteristics of controls of various components associated with the auxiliary feedwater system is enhanced by the monthly and quarterly surveillance testing of auxiliary feedwater system components.

G. Instrumentation

<u>Board Question CEC 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 instrumentation is available to aid operators at Rancho Seco in diagnosing, and controlling the effects of, an off-normal condition engendered by a loss of feedwater transient?

A. Rancho Seco has installed instrumentation and control capability to diagnose and control the effects of off-normal conditions engendered by a loss of feedwater transient. Specifically, the following feedwater transient diagnostic instrumentation is available in the control room for operator assistance:

1. Auxiliary feedwater flow instrumentation.
2. Reactor coolant system hot leg, cold leg and average temperature indication.
3. Steam generator level indication comprised of five channels of instrumentation, two startup range channels, two operate range channels, and one wide range channel.
4. Steam generator outlet pressure.
5. Pressurizer level comprised of three separate temperature compensated level indication channels.

6. Reactor coolant system makeup flow instrumentation.

7. Reactor coolant pressure instrument channel comprised of four narrow range pressure channels and three wide range pressure channels.

8. Main feedwater flow indication available to each OTSG.

9. High pressure injection system flow indication available to each of four high pressure injection lines to the reactor coolant system.

10. Reactor coolant system loop flow indication for each reactor coolant system loop.

Feedwater transient control equipment for which operational capability exists within the control room includes the following:

1. Startup and shutdown of both auxiliary feedwater pumps.

2. Control of the normal level control feedwater valves in the auxiliary feedwater system and control of the safety features actuated valves associated with the auxiliary feedwater system.

3. Control of the normal main feedwater pump turbines.

4. Control of the normal steam generator feedwater flow control valves.

5. Control of the pressurizer heaters.

6. Control of all three high pressure injection pumps.

7. Control of the individual high pressure injection flow control valves and the makeup control valve.

8. Control of the main feedwater isolation valves.

The sum of this instrumentation and control capability provides the operator with the information necessary to diagnose the onset of a feedwater transient, to determine whether it is a loss of feedwater or an overfeed transient condition, and to take the necessary operator action to mitigate the consequences of the feedwater transient.

In the event of a total loss of feedwater flow, the auxiliary feedwater pumps will be initiated automatically within seconds, and instrumentation available to the operator in the form of recently installed flow meters will allow him to verify that auxiliary feedwater flow is being provided to each steam generator. If primary system pressure and temperature, in concert with feedwater flow instrumentation and steam generator level, indicate that an overcooling condition is in progress, the operator has the capability to reduce feedwater flow to the steam generators through either the normal feed flow control valves, or, if auxiliary feedwater is initiated, through the auxiliary feedwater level control valves.

Both steam generator pressure and steam generator level instrumentation provide additional backup to the installed feedwater flow instrumentation to assist the operator in diagnosing either the loss of feed or overfeed condition.

Pressurizer level, reactor coolant system temperature, and reactor coolant system pressure indications enable the operator to diagnose whether adequate core cooling is being maintained and whether the reactor coolant system is in a subcooled condition. High pressure injection control from the control room also allows the operator to add inventory as necessary to maintain the reactor coolant system pressure and to promote adequate subcooling.

In the event of a malfunction of the electromatic operated relief valve, the control room operator has available in the control room, to assess the position of that valve, pressure and level indication of the pressurizer relief tank and temperature indication of the discharge piping from the electromatic operated relief valve. An additional operator aid in the form of a saturation meter is planned for installation during the current refueling outage. This meter will provide the operator with a continuous and direct display of the amount of subcooling present in the reactor coolant system, and will relieve him from determining the extent of subcooling through a comparison of pressure and temperature to a saturation curve.

All of the above described information is available to the operator in either a meter indication, computer readout, or chart record format. Controls are back lighted, indicating red when energized or open, and green when de-energized or shut. The adequacy of these special features within the Rancho Seco control room has been demonstrated in 34 cases when actual loss of feedwater capacity, to varying degrees, has been experienced. In each instance the operator was able to diagnose the situation adequately with available instrumentation and the controlled response was successful.

H. Pressurizer and Reactor Vessel Level Indication

Board Question H-C 22: Rancho Seco, being a Babcock and Wilcox designed reactor, does not provide control room operators with sufficient data on the water level in the pressurizer and vessel because the operators must interpret information on temperature and pressure in the primary loop and extrapolate water level, and therefore is unsafe and endangers the health and safety of Petitioners, constituents of Petitioners and the public.

Q. Is pressurizer level indication available to the Rancho Seco operators?

A. Yes. The Rancho Seco design provides for three separate water level indications of the pressurizer. The water level indications, which are temperature compensated to ensure their accuracy, have a range of zero to 320 inches. This covers the normal operating level range of the pressurizer and provides sufficient margin above and below that operating range to allow the operators additional time to take action and to restore a proper level within the pressurizer in the event of an off-normal condition. This level indication also provides the operator with low and high level alarms to alert him that an off-normal condition has occurred.

Q. Does the operator have access to sufficient data on the water level in the reactor vessel?

A. Yes. By maintaining the reactor coolant system pressure and temperature within the allowable operating range, the operator is assured that the reactor vessel is in a solid water condition without any significant vapor. Emergency Procedure D-5, "Loss of Reactor Coolant/Reactor Coolant Pressure," provides specific guidance to the operator to enable him to maintain the reactor coolant system in a subcooled condition in the event of a loss of coolant accident. By maintaining a minimum of 50 degrees F. subcooling in the reactor coolant system and operating high pressure injection pumps to provide an indicated level in the pressurizer, the operator will prevent the formation of vapor in the reactor coolant system.

Installation of a saturation meter during the current refueling outage will further enhance the operator's ability to determine adequate core cooling. This instrumentation will compare pressure over a range of zero to 2500 psig and reactor coolant system hot leg temperature over a range of 120 to 920 degrees F., and will calculate the degree of subcooling.

The operator's ability to monitor and measure the subcooled condition is provided by reactor coolant/pressure temperature instrumentation. Reactor coolant system hot leg and incore thermocouple temperature readout provide multiple temperature information. In the event conditions degrade to the point where a steam bubble does occur, the operator can recognize adequate core cooling by observing installed incore temperature thermocouples which are located at the top of the reactor core. Emergency Procedure D-5 requires that the operator run the high pressure injection system until subcooling conditions are established or until the low pressure injection system is operating at a minimum of one thousand gallons per minute per loop. Continuous running of high pressure injection will supply sufficient inventory to keep the core covered under small break conditions.

In short, the available instrumentation and procedural guidance assures that the operator can determine that conditions are degrading to the point where the water level might be established within the reactor vessel, and the action necessary to assure adequate core cooling.

I. Auxiliary Feedwater System Reliability

Board Question CEC 1-6: Will the modifications of Subparagraphs a-e still leave the Rancho Seco emergency feedwater system in a condition of low reliability?

Q. What has been the operating experience with the auxiliary feedwater system at Rancho Seco?

A. The Rancho Seco auxiliary feedwater system has been and continues to be a highly reliable system. Under actual transient and test conditions the auxiliary feedwater system has been called upon a total of 101 times. In each instance the system has provided feedwater, resulting in a 100 percent reliability record in actual practice. The contention that the auxiliary feedwater system has a low reliability is therefore refuted by actual operating experience.

In addition, the steps taken to assess the reliability of the auxiliary feedwater system and to upgrade its timeliness and reliability in response to the Commission's Order of May 7, 1979, are described in Licensee's testimony of Robert A. Dieterich.

J. Safety System Challenges

Issue CEC 1-1: Despite the modifications and actions of Subparagraphs (a) through (e) of Section IV of the Commission's Order, will reliance upon the High Pressure Injection System to mitigate pressure and volume control sensitivities in the Rancho Seco primary system result in increased challenges to safety systems beyond the original design and licensing basis of the facility?

Issue CEC 1-12: Despite or because of the modifications and actions of Subparagraphs (a) through (e) of Section IV of the Commission's Order of May 7, will Rancho Seco experience an increase in reactor trips resulting from feedwater transients that will increase challenges to safety systems beyond the original design and licensing basis of the facility?

Q. How do you assure that safety systems at Rancho Seco do not exceed the facility design and licensing basis?

A. Licensee's testimony of Bruce A. Karrasch and Robert C. Jones describes the relationship between modifications implemented at Rancho Seco in response to the Commission's Order of May 7, 1979, and the frequency of challenges to plant safety systems.

To assure that challenges to the safety systems do not result in operations beyond the design and licensing basis of Rancho Seco, administrative procedures have been established to monitor those transients considered in the licensing basis. Among those transients for which specific data is recorded and monitored are reactor trips caused by loss of feedwater, loss of feedwater to one steam generator resulting in a dry once through steam generator, cooldowns from hot conditions to 140 degrees, and high pressure injection into the reactor coolant system. When these events occur descriptions are recorded in the control room log, various computer logs, recorder traces, or other pertinent records of plant operation. Semi-annually the Engineering and Quality Control Supervisor is required to review these logs to ensure that the number of design cycles is not being approached or exceeded and, based upon this review, determine whether any corrective actions are required. The transient descriptions make use of B&W's specification for reactor coolant system components to ensure that the transient, as described and monitored, is properly categorized.

Consequently, safety systems at Rancho Seco will not exceed the original design and licensing basis of the facility.

K. Natural Circulation

Board Question CEC 1-2: Can poor understanding of natural convection in the Rancho Seco system result in a situation that will lead to inadequate cooling despite the modifications and actions of Subparagraphs a-e?

Q. Licensee's testimony of Bruce A. Karrasch and Robert C. Jones addresses the capability of the B&W nuclear steam system to provide adequate natural circulation for core cooling. Do the operators at Rancho Seco understand natural circulation sufficiently to avoid inadequate core cooling?

A. Yes. Beyond the extensive licensing and requalification training provided to operators at Rancho Seco and described in Part II of my testimony, additional operating procedures and training were implemented in response to the Commission's Order of May 7, 1979. The purpose of these operating procedure changes and training was to assure proper operator response in off-normal conditions to provide adequate cooling to the reactor core.

The procedure revisions and additional training provide guidance for operators in establishing natural circulation cooling of the reactor core in the event forced circulation is lost. These procedures describe specific parameters which operators can monitor and provide specific direction on controlling these parameters in the event of a loss of forced circulation. These parameters are steam generator level, reactor coolant temperature, reactor coolant system pressure and pressurizer level.

The additional training, supplemented by the changes to operating procedures and the control room instrumentation available, assure that the operator has the proper plant system information and guidance to initiate and confirm natural circulation if necessary to protect the core under off-normal conditions.

IV. CONCLUSION

Q. Are the short- and long-term actions and modifications directed by the Commission in its Order of May 7, 1979, adequate to provide reasonable assurance that Rancho Seco will respond safely to feedwater transients?

A. Yes.

APPENDIX I.A

A 13-week observation training program at the Babcock & Wilcox designed Indian Point No. 1 reactor consisted of integration into the operating crews on shift work. The on-shift guidelines were to spend at least 4 hours in the control room and the remaining 4 hours tracing systems and studying. This flexibility maximized the observation training and supplied training over and above that which was expected due to several plant shutdowns as a result of equipment malfunctions.

During this training period, the Indian Point Unit One was shut down for refueling. Some of the typical activities observed were:

1. Removal of reactor vessel head.
2. Preparation of refueling equipment.
3. Removal of spacer rods.
4. Repair of main coolant pump thermal barrier.
5. Repair of main coolant pump electrical connector.
6. Removal and transfer of fuel elements to storage.
7. New fuel insertion into the core.
8. Extensive maintenance on contaminated equipment.
9. Completion of refueling.
10. Installation of main coolant pump.
11. Plugging of steam generator leaking tubes.

APPENDIX I.B

A 520 hour basic reactor and engineering course at Rancho Seco covered the following topics:

Mathematics	89.0 hours
Classical Physics	41.5 hours
Atomic Physics	21.5 hours
Nuclear Physics	64.0 hours
Reactor Core Physics	80.0 hours
Reactor Operation	20.0 hours
Health Physics	42.0 hours
Instrumentation	98.0 hours
Thermodynamics	18.0 hours
Fluid Mechanics	18.0 hours
Heat Transfer	20.0 hours
Comprehensive Examination	8.0 hours
Total	520.0 hours

APPENDIX I.C

An 8-week pressurized water reactor technology course at Lynchburg, Virginia was presented by Babcock and Wilcox Company. The instruction covered in detail the design and characteristics of the major components of the nuclear steam supply system. All facets of the instrumentation and control systems were studied to ensure complete knowledge of the various features of these systems. Instruction was provided in water chemistry, radiochemistry and health physics as it applies to the operation and maintenance of the nuclear plant. Total classroom time was 240 hours. A detailed breakdown of the topics covered is shown below:

1. Once Through Steam Generator
2. Reactor Coolant System
3. Control Rod Drive System
4. Nuclear Physics
5. Auxiliary Systems
6. Heat Transfer and Fluid Flow
7. Fuel Assembly
8. Reactor Vessel
9. Water Chemistry
10. Nuclear Instrumentation
11. Plant Computer
12. Non-Nuclear Instrumentation
13. Electrical Power Requirements
14. Reactor Protection System
15. Integrated Control System
16. Plant Operations
17. Health Physics
18. Safety Analysis
19. Fuel Handling
20. Incore Instrumentation
21. Safety Features Actuation System
22. Test Program

APPENDIX I.D

A 10-week simulator course was presented by Babcock and Wilcox Company at Lynchburg, Virginia.

The 392-hour simulator training program consisted of the following:

1.	Control Room Operation	126 hours
2.	Lectures	148 hours
3.	Study and Counseling	100 hours
4.	Examinations	18 hours

The Control Room operation included all normal plant operations, examples of which are listed below:

1. Heatup from 70^{0}F
2. Heatup from 400^{0}F
3. Reactor Startup (Each crew made a minimum of ten startups)
4. Turbine Generator Startup
5. Turbine Generator Loading
6. Boration/Deboration
7. Power Operation with Load Changes
8. Plant Shutdown
9. Plant Cooldown

In addition to the normal plant operation, each crew experienced approximately 110 malfunctions over a wide spectrum of failures which included feedwater transients and loss of coolant accidents.

The lecture subjects and hours are as follows:

1.	Course Introduction and Plant Familiarization	16 hours
2.	Plant Operations	58 hours
3.	Fuel Handling	6 hours
4.	Soluble Poison Concentration Control	7 hours
5.	Reactor Protective System Review	4 hours

(continued)

6.	Reactor Physics Review	4 hours
7.	Reactivity Balance and Practical Exercises	4 hours
8.	Control Rod Drive Logic Review	6 hours
9.	Nuclear Instrumentation Review	4 hours
10.	Non-Nuclear Instrumentation Review	4 hours
11.	Safety Analysis Review	12 hours
12.	Axial Power Shaping Rods and Power Peaking	4 hours
13.	Operations with Less Than Four Reactor Coolant Pumps	3 hours
14.	Startup Physics Testing	8 hours
15.	Hydrogen Addition and Degassification	2 hours
16.	Safety Features Actuation System Review	2 hours
17.	Integrated Control System Review	2 hours
18.	RC System Leak Detection and Practical Exercises	2 hours
19.	Heat Balance and Practical Exercises	2 hours

At the conclusion of the course, a Senior Reactor Operator level examination was administered by the NRC Operator Licensing Branch. The purpose of the exam was for NRC evaluation of the simulator as a training tool.

APPENDIX I.E

The major startup activities in which the on-site management personnel participated included the following:

A. Preparation/or review of test procedures.

B. Preparation/or review of operating casualty, refueling, and emergency procedures.

C. Participation in initial startup and testing of plant systems during the site activities listed below were:

July 1972	- Commenced 24-hour per day operation.
August 1972	- Initial energizing of 220Kv Switchyard.
August 1972	- Initial operation of fire protection water system.
October 1972	- Initial operation of service water system.
November 1972	- Initial operation of plant cooling water system.
November 1972	- Reservoir system in service.
December 1972	- Initial operation of canal pumping plant.
January 1973	- Plant drainage system in service.
February 1973	- Initial operation of auxiliary steam system.
March 1973	- Initial operation of main lube oil system.
	- Initial operation of makeup demineralizer system.
	- Initial operation of spent fuel cooling system.
	- Initial operation of decay heat removal system.
	- Initial operation of fuel handling equipment in spent fuel building.
April 1973	- Initial operation of component cooling water system.
	- Initial operation of nuclear service cooling water system.

(continued)

May 1973	- Initial operation of the nuclear service raw water system.
	- Initial operation of the circulating water system.
June 1973	- Initial operation of the condensate system.
	- Initial operation of reactor building spray pumps.
July 1973	- Initial operation of diesel driven fire pump.
August 1973	- Initial operation of reactor coolant drain tank system.
September 1973	- Initial operation of high pressure injection pumps and concentrated boric acid system.
October 1973	- Initial operation of reactor coolant pump motors, condenser vacuum equipment.
November 1973	- Initial operation of th: radwaste ion exchanger resin transfer systems.
	- Initial fill of fuel transfer canal and operation of containment building fuel handling equipment and fuel transfer carriage equipment.
December 1973	- Pressurizer operation with steam bubble.
	- Operation of reactor coolant pumps for system heat-up.
	- Cold hydro test of primary coolant system.
	- Initial operation of miscellaneous waste system.

APPENDIX I.F

A review training program during the period from January 2, 1973 to May 13, 1974 included three parts:

Part 1 - On Shift Review of Systems
Part 2 - Classroom Systems Review & General Review
Part 3 - Simulator Training

Part 1

The "On Shift Review of Systems" consisted of the following:

A. A reading assignment was issued with assigned reading in the Final Safety Analysis Report, Operations Manual and the Technical Specifications.

B. A one to two-hour quiz was given ten days later on the reading assignment material.

C. Each quiz was graded and returned with an answer sheet.

D. Following A, B & C, specific questions (7/week) were issued, graded, and returned with an answer sheet. These questions covered a wide spectrum of information from each category of material found in senior license examinations.

Listed below are the reading assignment subjects:

Assignment	Subject Studied
1	Reactor Coolant System, Pressurizer, and Pressurizer Relief Tank System.
	Core Flooding System
2	Reactor Coolant Drain Tank System
	Control Rod Cooling System
	Miscellaneous Waste & Boric Acid Concentrator Systems.
	Borated Water Storage System
3	Spent Fuel Cooling System
	Nuclear Service Cooling Water System
	Miscellaneous Water System
	Miscellaneous Drains & Sumps System

(continued)

Assignment	Subject Studied
4	Waste Water Disposal System
	Plant Cooling Water & Reservoir System
	Demin. Reactor Coolant Storage System
	Reactor Coolant Chemical Hydrogen Addition System.
5	Reactor Building Spray System & Reactor Building Emergency Cooling System.
	Decay Heat Removal System
	Miscellaneous Liquid Radwaste System
	Chilled Water System
6	Steam Generator Secondary Side
	Coolant Radwaste System
	Generator Hydrogen System
	Turbine Plant Chemical Addition System
7	Nuclear Service Raw Water System
	Component Cooling & Turbine Plant Cooling Systems
	Fire Protection Water System
	Auxiliary Gas System
8	Condensate & Feedwater System
	Plant Air System
	Auxiliary Steam System
	Generator Seal Oil System
9	Circulating Cooling Water System
	Reactor Sampling System
	New & Spent Resin System

(continued)

Assignment	Subject Studied
9 (continued)	
	Diesel Fuel Oil System
	Review of All Above
10	Letdown & Purification Makeup System
	High Pressure Injection System
	Diesel Generator System
	Carbon Dioxide System
	Electro-Hydraulic Oil System
11	Electrical Distribution Systems 220KV to 120V.
	Auxiliary Feedwater Pump System
	Turbine Plant Sampling System
	Turbine Lube Oil Transfer System
12	Main Turbine
	Main Feedwater Pump System
	Air Ejector & Gland Steam System
	Extraction Steam, Reheater/Feedwater Heater Drain System
13	Radiation Detection Liquid Systems
	Radiation Detection Gaseous System
	Radiation Detection Area Systems
	Rancho Seco Technical Specifications
14	Reactor Protection System

(continued)

Assignment	Subject Studies
15	Nuclear Instrumentation
	Plant Instrumentation
	Plant Annunciator System
	Full Flow Polishing Demineralizer Resin Transfer and Regeneration.
	Generator and Exciter System
16	Tritium Management for Normal Plant Operation
	Waste Gas System
17	Integrated Control System
18	Safety Features Actuation System
	Reactor Non-Nuclear Instrumentation
19	Control Rod Drive System
	Computer System
20	Fuel Handling System
	Plant Communications System
21	Classroom Lecture - Integrated Control System
22	Classroom Lecture - Non-Nuclear Instrumentation
23	10-Week Review Quiz No. 2
	Review Quiz Grading

Part 2

The "Classroom General Review" was a full time, six days per week, two-week intensive study review course, which covers in detail each of the categories in the AEC licensing examination. Total classroom time was 96 hours.

(continued)

Part 3

The simulator refresher course was preceded by a reading assignment and a two-day full time classroom lecture series where detailed explanations were presented for the integrated control system, non-nuclear instrumentation and the control rod drive system. The simulator training was concentrated in the area of normal plant startup/shutdown, casulty and emergency procedures. The simulator was the final operating review which was intended to place all previously learned and presented material firmly in the operator's mind.

APPENDIX II

Part I. Academic Phases

A.	Mathematics Course	160 Hours
B.	Physics Course	240 Hours
C.	Related Technologies Course	200 Hours

Part II. In-Plant Phases

A.	Systems and Operations Training	240 Hours
B.	Procedures and Operations Training	320 Hours
C.	Fuel Handling Training	24 Hours

Part III. Simulator Training Phases

A.	Pre-Simulator Review Course	40 Hours
B.	Simulator Operations	120 Hours

Part IV. License Preparation Phases

A.	Control Room Operations	80 Hours
B.	Pre-License Review Course	48 Hours
C.	Pre-License Audit Exams	40 Hours
D.	In-Plant Briefing	40 Hours

APPENDIX III

1.	Informal group discussion of TMI-2 by Operations Supervisor.	2 hours
2.	Formal training by Training Supervisor - Sequence of events and immediate directives.	3 hours
3.	Informal discussion on TMI-2 incidents and directives by NRC I&E.	1-1/2 hours
4.	Formal training - Training Supervisor NRC Bulletins 79-05A and 79-05B requirements to include review of failures, review circumstances and and chronology, procedure review, emphasis on seriousness of void formation, plant modification changes.	6 hours
5.	Formal classroom lectures - B&W Lynchburg, Va. TMI-2 incident, emphasis again on seriousness of failures and recognition of failures.	4 hours
6.	Simulator Training - B&W Lynchburg, VA. TMI-2 incident transient, recovery techniques.	4 hours
7.	Audit Quiz on TMI-2 training prepared by Training Supervisor.	1-1/2 hours
8.	Oral Audit Quiz by NRC	1 hour
9.	Additional Training and Audit Consultant (General Physics). Additional training - TMI-2 incident, small breaks, plant and procedure changes.	4 hours
10.	Informal shift training - by Shift Supervisor Plant modifications, procedure changes to include walk thru plant to locate new instruments and equipment location.	

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MR. BAXTER: Mr. Rodriguez is available for cross examination.

CROSS EXAMINATION

BY MR. ELLISON:

Q Mr. Rodriguez, referring to page 3 of your testimony where you describe your professional qualifications, could you first of all briefly describe what your present duties are at SMUD?

A I am presently the Manager of Nuclear Operations for the Sacramento Municipal Utility District. My primary function is the over-all technical operation and administrative management of the staff that operates the Rancho Seco nuclear generating station.

Q Would your duties presently include authorizing procedure changes?

A I do not normally alter procedure changes.

Q Do you have a role in the changing of procedures at Rancho Seco?

A Yes, I do.

Q Could you please describe it?

A The technical specifications require that the manager of Nuclear Operations approve changes to those procedures dealing with security and emergency planning.

Q If a change does not concern the emergency or the security plan, would you be involved in that process

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at all?

A Not normally.

Q Who is principally responsible at SMUD for making procedure changes?

A That encompasses a number of individuals. Are you looking for one in particular?

Q At this point, I am most interested in the person or persons that would -- let me take it step by step. First of all, I am interested in the person or persons who would determine that a procedure change is necessary.

A That determination would be made by a group supervisor.

Q A group supervisor, is that the same as a shift supervisor?

A No, it is not.

Q Could you briefly describe for me what a group supervisor is and where, in the management of SMUD, he would be found.

A A group supervisor is typically an individual who is supervising some members of the nuclear operations department staff. That could be a foreman, it could be a division supervisor, it could be an area head.

Q So, any one of these people that you just described would determine that a procedure change was necessary?

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A He could determine that it was necessary.

Q Do these people -- are they subordinate to you?

A Not all of them.

Q Is there any one of the people that you mentioned that would, in the typical circumstances, be responsible for determining that a procedure change was necessary?

A Yes, any one of them might be responsible for determining that a procedure change was necessary.

Q Once a procedure change has been determined by these people to be necessary, who would actually write the new procedure?

A The individual's supervisor would designate someone, typically in his organization, to write the procedure change.

Q So, depending upon the person making the determination that a procedure change was necessary, am I correct in my understanding that it might be any one of a number of people that would write the procedure change, itself?

A That is correct.

Q Once the procedure change has been written, would it then be transmitted -- who would it be transmitted to?

A It would be transmitted initially to the group supervisor.

Q From there?

A From there is would be transmitted then to the

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plant review committee.

Q Who sits on the plant review committee?

A Five members of the Nuclear Operations Department staff.

Q Would I be correct in taking it from their title that they review the procedure?

A Yes, that is correct.

Q What do they review it for?

A The plant review committee reviews it for its applicability to whatever system is involved. It reviews it to ensure that it meets the design requirements of the system involved.

It would review it to determine if there are any nuclear safety related hazards that are greater then those hazards analyzed in the FSAR.

They will review it with regard to its applicability to this technical specification to determine whether or not it requires a change to the technical specifications. They will also review it in the context, if that particular change is approved, how it might reflect in changes in other procedures.

Q If they were to determine that it did require a change in other procedures, would I be correct then in stating that they would be responsible for determining that additional procedures were necessary?

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A If the -- if the review committee determines that other changes are necessary to be made, then it is the chairman of plant review's responsibility to identify the appropriate group supervisor and establish a time for him when those changes will be brought before the committee for their review.

Q So, would it be fair to say that in addition to the group supervisors, the plant review committee, at times, also determines the need for procedure changes?

A The five members on the plant review committee are all group supervisors.

Q I see. Once the plant review committee has reviewed and, let's assume, approved a major change, where would it go from there?

A I'm sorry, I didn't hear you.

Q Once the plant review committee has reviewed and, let's assume approved of a procedure change, where would it go from there?

A If the procedure is deemed not to involve a change in the technical specifications, or not to involve an accident, different from or greater than the types of accidents analyzed in the final safety analysis report, it will then go to the plant superintendant.

Q Could you clarify what you mean when you say the procedure that would involve an accident not considered in

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the FSAR?

A If the procedure change or the character of the procedure change in the opinion of the plant review committee might introduce an accident that was greater than the types of accidents analyzed in the FSAR, that procedure change would not be approved by the plant review committee.

Q So, when you say involved such an accident, you mean create the possibility for one.

A Yes.

Q Is that correct?

A That's what I mean.

Q You are not saying that the procedure simply might be used in response to an accident that was analyzed in the FSAR.

A What I am saying is if by an operator taking the action that the procedure would prescribe might cause an accident more serious than what was analyzed in the FSAR.

The plant review committee's charter is not to approve that procedure.

Q Once the plant superintendant receives the procedure change, what is his role in reviewing it?

A His role is very similar to the plant review committee's role. It is primarily a back-up check and a management check.

Q So, he would be reviewing it for the same types of

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things that you described earlier with respect to the review committee?

A   That is correct.

Q   Okay. Where would he transmit it?

A   If he approves it, then the procedure would be returned to the group supervisor for implementation.

Q   How would it be implemented?

A   The procedure would be reproduced and placed in the manuals that are applicable. The personnel then utilizing that procedure would have access to the corporate chain.

Q   Once the procedure has been inserted in the appropriate manual in the control room, it would be in effect?

A   That is correct.

Q   In describing your responsibilities, you described generally what I would term "management responsibilities." Yet, I note that you currently hold a senior reactor operator license. That appears at page 5, line two of your testimony.

Is it part of your responsibility to operate the reactor?

A   Not directly, no.

Q   Apart from whatever operation -- direct operation of the reactor is necessary to requalify you for your

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license, do you normally -- have you either now or in the past regularly stood shifts operating the reactor?

A Not at Rancho Seco.

Q Have you, at another power plant?

A Yes.

Q Where?

A In Connecticut.

Q Which reactor is that?

A That was one of three separate reactors that was the Combustion Engineering's S-1C reactor.

Q The position you held there was reactor operator?

A No, I was essentially a shift supervisor.

Q How long did you hold that position?

A As I recall, it was about two weeks after I completed the qualifications.

CHAIRMAN BOWERS: I am not sure I understood that answer. You only worked two weeks at this job, or you started two weeks after you qualified?

THE WITNESS: This was a Navy submarine prototype training facility. I was there as a student for six months, qualified to stand that watch, stood that watch for two weeks, then was ordered to a submarine.

CHAIRMAN BOWERS: All right.

BY MR. ELLISON: (Resuming)

Q In the time that is available this afternoon, I

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bfm11

am going to take some of the testimony out of order, and begin with your testimony on control room design. Could you briefly describe the changes that have been made to the Rancho Seco control room prior to Three Mile Island, but after the initial operation of the facility?

A If you asking about physical changes to the control room, we have not made any change to the control room. If you are asking changes that we have made to instrumentation and controls, I can address that.

MR. COLE: I'm sorry. I did not hear your last response, Mr. Rodriguez.

THE WITNESS: His question was changes that we have made to the control room. We have not made detailed changes to the room. We have made changes to the instrumentation and controls in that room.

He just want to make sure we have not added ventilation or change the size of it or anything like that.

BY MR. ELLISON: (Resuming)

Q When I say the control room, I am including all of the instrumentation and controls that are contained within it. So, a change to an instrument that would be made in the control room would be included in my question, but a change to, let's say, the sensor that that instrument reads outside the control room would not be.

A Okay.

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Q Okay. With that definition in mind, then my question is whether or not you made it -- could you briefly describe changes in the control room that were made after the construction of the control room but before the Three Mile Island accident?

(Pause.)

A I guess I really have trouble about that. What I heard you say was, "Describe the changes that we made to the control room after construction but before the Three Mile Island incident."

I guess I do not know when to -- you know -- when do you want me to say construction was finished?

Q Perhaps it would be easier if we defined this period as from the day the reactor began commercial operation to the Three Mile Island accident.

end t-P-10

jl flws t-P-11

A Well, I will tell you what I can recall just immediately. I am sure I cannot recall all the changes that we made over that five-year period. First of all, we added switches that allow control of the cross connect valves between the decayheat removal systems. We added a switch to control a motor operated valve to provide a leak path off of the decay heat drop line.

We added flow indication for that decay heat drop line, and put that flow indication in the control room. We changed the power supply to the non-nuclear instrumentation cabinets to remove the Z-power. We added the automatic switching of 118-volt power to the NNI cabinets to provide a redundancy and backup. Right now those are the only significant ones I can think of.

We changed the wording on numerous enunciator lights. Specifically what that wording was before and what we went to, I cannot address now. We changed labels on switches to more clearly identify what the switch did, and put breaker numbers on it so the operator could more quickly direct an outside operator where to go if he was having problems with that switch. That is about the extent of what I can think of.

CHAIRMAN BOWERS: Mr. Ellison, I apologize for interrupting. You have been using the term "what have you done this way, that way, in the control room." Well, we did a site visit at the end of February, and my memory is that there was certain equipment just outside a door outside the control room proper. That was sort of part of the control room operation. Is that correct, Mr. Rodriguez?

THE WITNESS: That is true, and that is the general areas where those NNI cabinets are. That is why I spoke to those changes.

CHAIRMAN BOWERS: So you were including that physical area as part of the control room?

THE WITNESS: Yes.

BY MR. ELLISON: (Resuming)

Q Perhaps this would be a good place to ask you, Mr. Rodriguez, could you define for us what the boundaries of the control room are, and when I say control room, I am referring to that area that NRC has requirements that a certain number of operators must be within the control room and so, using that definition of what the control room is, could you describe for us what the control room includes and what it does not include?

A The control room which identifies the areas in which a licensed operator must be available at all times includes the shift supervisor's office, the area which contains the operating consoles which we would call H1EE and H1RC, et cetera, the three consoles that control the turbine and its auxiliaries, the reactor and feedwater

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system, and the reactor system auxiliaries, and the computer. That area also includes what we refer to as upright panels, which includes the panel for controlling the main switching to the major buses and off-site power, the panel that contains the boilerfeed pump and main turbine supervisory equipment, a panel which contains the control rod drive position indicators, safety features panel, and the auxiliary panel.

Immediately adjacent to that general area is a small area which contains the area radiation monitoring panel, the liquid radiation monitoring panel, and the gaseous and particulate monitoring -- radioactive monitoring panel, and the boron panel.

Q I believe the Board has toured the facility, and I believe all the parties are familiar with it and have been in the control room. I understand your last answer. You included all of the central control room, if you will, that main room involving the central consoles, the shift supervisor's office, and the radiation monitoring panels which are just outside the main control room in the area I believe Mrs. Bowers was referring to.

Is that correct?

A That is correct, and in that area also is the logger typer.

Q Okay. Just for the purposes of clarity from this time forward when I refer to the control room, that is the

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area I am referring to.

Could you briefly describe the major changes that have been made in the control room since Three Mile Island?

A We have added in the control room feedwater flow indication for both trains of the auxiliary feedwater system. We have added two TSAT meters. We have added four additional backlighted pushbutton switches for additional control of the DC motor operated high pressure injection valves. We have added a keylock switch for control of the electromatic operated valve. We have added two switches for control of parts of the pressurizer heaters which are powered by the 480 volt power supplies.

We have added a keylock switch to allow the operator to bypass the feedwater pump reactor trip capability. We have added T-hot temperature for both the A and B loop with selector switches to allow selection of either one of two temperature elements. We have added T-cold instrumentation, selector switches to allow selection of T-cold from each of the cool pump discharges.

We have added pressurizer level indication. We have added wide range pressure indication with a selector switch to select from either the A or B side. We have added a makeup tank level indication. We have added steam generator pressure indication for each steam generator. We have added steam generator wide range indication for

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each steam generator. We have added source range indication. We have rescaled the high pressure injection flow meters. We have added enunciation to indicate when auxiliary feedwater is required. That is all I can think of right now. There is probably something else in there.

Q I noticed in your answer that you mentioned, for example, that you had added pressurizer level indication, that you added steam generator pressure indication. Did you mean by that -- I also understand that that indication was present in the control room prior to Three Mile Island. So, my question to you is, do you mean by that that you added additional instrumentation or that you changed the instrumentation in some way?

A Yes. We added additional instrumentation, and we changed it.

Q With regard to changes since Three Mile Island, which if any of those changes that you just described were not in response to NRC requirements?

A The indication for T-hot, the additional indication and metering for T-cold, the additional indication for steam generator pressure, the additional indication for pressurizer level, the additional indication for makeup tank level, the additional source range indication, the auxiliary feedwater flow indication. Those are the ones right now I am sure are not NRC requirements.

MR. SHON: Did you include the auxiliary feedwater flow indication?

THE WITNESS: Yes, sir. In Mr. Mattimo's letter of April 27, 1978, he said we should shut down and install that. The NRC agreed to that. We never received an order that it had to be installed.

MR. SHON: Thank you.

THE WITNESS: The order followed up after that, but we initially said we would shut down and install it.

MR. SHON: It was my understanding that that was required by NRC at least at some time or another.

THE WITNESS: The order, if you recall -- the order that came out listing all the things we said we would do was a verification order, and that is how I interpreted his question. That is the context in which I answered it.

MR. SHON: Thank you.

I am sorry, Mr. Ellison.

MR. ELLISON: You anticipated my question.

MR. BAXTER: Excuse me. You said Mr. Mattimo's letter of 1978.

THE WITNESS: 1979.

BY MR. ELLISON: (Resuming)

Q Is it your testimony, then, Mr. Rodriguez, that the auxiliary feedwater flow indication was a voluntary action on the part of SMUD?

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A I think in the context that we wrote a letter and said we would install it prior to the NRC ordering us to put it in, yes.

Q That letter also stated that you had shut down until it was installed. Is that correct?

A That is correct.

Q Is it your opinion that the shutdown was also a voluntary action?

A Yes.

Q In light of the changes that you have described both before and after Three Mile Island, would you characterize the Rancho Seco control room as being substantially different now than it was when the facility began commercial operation?

A I would characterize it as, we have continued to make improvements in the control room, but that we have not substantially changed its character.

Q Are you aware of any human factor studies of the Rancho Seco control room that SMUD or anyone else has undertaken?

A Yes, I am.

Q Could you identify those studies?

A The Electric Power Research Institute made a study two years ago, if I am guessing right, if I remember correctly, in which Rancho Seco was one of a

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8

number of units that they looked at.

Q Are you aware of any other studies?

A Specifically human factor study of the Rancho Seco control room?

Q That is correct?

A No, I am not.

end 11

300 7TH STREET, S.W. REPORTERS BUILDING, WASHINGTON, D.C. 20024 (202) 554-2345

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Q Mr. Rodriguez, Mr. Lanpher is going to provide to you as well as the Board and the parties a copy of a document entitled The Human Factors Review of Nuclear Power Plant Control Room Design.

First of all, I would like this document identified as CEC Exhibit Number 33.

(The document referred to was marked for identification as CEC Exhibit Number 33.)

BY MR. ELLISON: (Resuming)

Q I would like you to take a moment to examine this document, Mr. Rodriguez.

A I don't think I can do it in a moment. If you want me to examine it, I need some time.

Q First of all, is this the document that you identified a moment ago?

A I am not sure. This is not necessarily the specific one. The Electric Power Research Institute has come out with a number of reports, and I think that was really what I was referring to rather than some specific document.

Q Do you recognize this document?

A I recognize that the Lockheed Missile and Space Company had been contracted by EPRI to do that, and that the project manager, Randall Pack, was the individual we dealt with when he initially asked that we open the facility to his contractor.

Q Have you seen this particular document before?

A I probably have.

Q Are you familiar with its contents?

A Not unless I take some time now and review it again.

Q Okay. Perhaps the best thing would be, then, that we will withhold this line of inquiry and give you an opportunity to review it. I do not think it will be completed today. So --

MR. BAXTER: Could I ask that counsel for the Energy Commission provide some guidance as to the line of interrogation? I think even for overnight study this is a little bit much.

THE WITNESS: I can guarantee you it will not be studied overnight. Not tonight.

(General laughter.)

MR. ELLISON: I would point out this is on the order of some of the materials that other parties in this proceeding have distributed in terms of its bulk. However, I certainly do not think there is any need to identify beforehand my questions. However, what I am interested in here is, of course, Rancho Seco, and there is a great deal of discussion in here of other facilities, and we are not particularly nterested in that.

MR. BAXTER: That does not help me much, Mrs. Bowers. He can ask the witness any questions he would like tomorrow, but I am certainly not going to sit here and represent in any way that he is going to study this thing and sort out everything that relates to Rancho Seco and be prepared to be examined on it. He has not cited the study or relied on it in his testimony.

CHAIRMAN BOWERS: Mr. Ellison, in looking at the page of contents quickly, I do not see any reference there to specific facilities. Now, are you saying in one place in this report it focuses on Rancho Seco, or is it scattered throughout the report?

MR. ELLISON: It is scattered throughout the report, Mrs. Bowers. It is my understanding that although the plant is not named, one of the plants in here is Rancho Seco, and we would provide that nexus. If Mr. Baxter prefers, we could go ahead with the cross examination now. If Mr. Rodriguez is not going to familiarize himself with the document, I can go ahead and ask him the questions I was going to ask him anyway.

I recognize that Mr. Rodriguez has not authored this document. We do not offer it in that context, but we offer it in the context like many other documents have been offered here, to facilitate the Board and the parties'

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understanding of the proceeding. I do not believe it is absolutely necessary that Mr. Rodriguez be thoroughly familiar with the document to answer the questions that I am going to ask him. However, if he prefers that, we are willing to withhold this line of questioning until tomorrow.

MR. BAXTER: Why don't we try it and see how it goes?

(Pause.)

BY MR. ELLISON: (Resuming)

Q Mr. Rodriguez, if you would, I would like you to turn to Page 4-5 of CEC 33. Have you found that page?

A Yes, I have.

On that page is a diagram of a nuclear power plant control room, together with a picture of a control room. This plant is identified as Plant C.

Q Is this the Rancho Seco control room?

A Yes, it is.

Q I would like you to refer to Page 4-9. At the bottom of Page 4-9, under the title Control Board Configuration, appears the statement Plant C,which is the plant you identified a moment ago as Rancho Seco, "The smallest control board separates functionally related interprimary consoles from rear wall mounted consoles."

Do you believe that is a true statement?

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A I am not really sure. They say Plant C, and then they talk about the smallest control board. I do not know what particular control board they are referring to. It sounds to me like they are saying that Plant C has some inner primary consoles and some rear wall mounted consoles, and I will agree that is what we have at Rancho Seco.

Q Do you agree with the statement that the Rancho Seco control room separates functionally related consoles?

A The rear mounted panels are functionally related to the normal operating control consoles, yes.

Q Let me return a moment to the study that you identified, which was performed by EPRI, you stated, by the same people that have authored this document. Would it be fair for me to assume that this is the same study you are referring to?

A I think so. As I said earlier, EPRI has put out a number of documents, and I was referring to them from a generic standpoint when you asked about reports related to human factors engineering, and this is one of those.

Q Okay. It would be my understanding of what you are saying that although there may have been a number of documents there was only one underlying study involved. Is that correct?

A I am not sure that is correct, because EPRI moves

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from one contractor to the next, and whether or not the last two or three reports were from Lockheed or not, I cannot say.

Q You are familiar with the EPRI study as it was -- as it involved Rancho Seco. Is that not true?

A I was familiar with EPRI asking if Rancho Seco could be made available for a study that they had under contract -- that they had a contractor for, and we made it available to them.

Q Do you know whether there were any other contractors who came out and inspected the Rancho Seco facility apart from the Lockheed Corporation?

A I do not know.

Q Can you tell me whether at the conclusion of that study a copy of its conclusions was provided to SMUD?

A Yes, it was.

Q Was this separation of functionally related consoles among the conclusions that was provided?

A I assume so. That is what it says in here.

Q Do you know whether SMUD made any attempt to address this problem?

A Quite honestly, it was not a revelation to me when they said they are functionally separated. That was part of the design of the control room, in keeping it as small as we could.

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Q Further up the page, the second sentence of this paragraph, reading from the beginning of the paragraph, this document states, as indicated above, the configuration--

A I am sorry, I am not with you.

Q I am on Page 4-9. At the bottom of the page, where it says Control Board Configuration, I am reading the first sentence there. It says, "As indicated above, the configuration of the control boards is more important than the absolute size of the control room. Plants A and C vie for the distinction of being the least effective in terms of operator interaction." That is the context for the statement I read earlier, Plant C separates the functionally related consoles.

You responded a moment ago that you were not surprised by this conclusion. My question is whether SMUD has made any attempt to address this problem.

A I guess SMUD does not consider it a problem of such a magnitude that it needs to be addressed.

Q Has SMUD done, to your knowledge, a human factors study such as this of its control room?

A Not to my knowledge.

(Pause.)

Q Are there any such studies being performed at this time?

A In what context? In regard to anywhere, or the

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Rancho Seco control room?

Q Well, with regard to the Rancho Seco control room.

A No, there are no studies at this time being conducted on the human factors aspect of the Rancho Seco control room.

Q Has SMUD contracted for such a study?

A Yes, they have.

Q When do you expect that to commence?

A Probably the end of the spring or early summer.

Q When do you expect it to conclude?

A I have not seen the schedule, so I cannot comment on that.

(Pause.)

Q Could you refer to Page 4-13? The bottom paragraph begins with the sentence, "The two most unwieldy control boards (Plants A and C) have the worst manpower unit ratio, two operators per unit," et cetera.

Now, it is my understanding that there are now more operators than two per shift at Rancho Seco.

A There were at the time of the study also.

Q Do you know whether the conclusion that the Rancho Seco control room was among the two most unwieldy ones identified by EPRI was part of the -- was communicated to SMUD?

A I do not know of it being communicated in any

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special format other than the report has stated here.

Q I am not asking for a special format. I am simply wondering whether this conclusion was something you were aware of or tnat you believe was communicated to SMUD management.

A It was communicated via this report, which I feel confident we received a copy of.

CHAIRMAN BOWERS: Mr. Ellison, when you identified this document, I do not recall that you gave the date, November, 1976. You may have. I just do not recall. But I think it is important for our record to show the date on this document.

MR. ELLISON: The record should reflect that.

BY MR. ELLISON: (Resuming)

Q Do you know whether there has been any attempt to make the control boards at Rancho Seco less unwieldy since this document was produced?

A There has been no attempt to substantially change the control board configurations since this document was produced.

Q I would lik e you to refer to Page 4-14, particularly the section Back Panels. There you find the statement, "At each of the plants visited, controls and instrumentation were placed in areas outside the primary control room area or outside the operator's line of

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sight." Then it goes on to discuss the way those panels are arranged in different plants and concludes the paragraph by saying, "Without exception, operators at all plants complained that display and control access was required during off-normal and normal operations."

Do you know whether there has been any attempt to rectify this situation at SMUD since this document was produced?

A No, there has not been any.

(Pause.)

Q I would like you to refer to Page 4-15 and Figure 4-13, which appears on the next page. This first full paragraph on Page 4-15 begins by describing the need for speedy and accurate diagnosis and response to off-normal events. Further down the page it states that any delay caused by the need for an operator to leave the primary control area are inefficient.

It then states, "Figure 4-13 depicts the traffic flow path of an operator responding to the analyzed task at Plant C. The contrast between the well-grouped functions in the primary area and the placement of the radiation monitoring panels is obvious. This represents time lost and time away from the crucial area of operation."

Referring to Figure 4-13, the figure purports to describe the flow paths at Rancho Seco for a

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single operator responding to a steam generator tube rupture prior to shutdown initiation."

Is it your understanding that this is an accurate depiction of the travels of a single operator responding to that event?

A If there was only one operator available in the control room throughout the course of this particular type of incident, yes, this probably accurately describes his movement.

Q Would that accurately describe his movement as of today?

A Again, if he was the only operator available throughout the incident, it would describe his movement.

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Q I would like you to refer to page 5-36 and also to figure 5-46 which appears on the opposite page. With respect to figure 5-46, it is not clear to me. Is this Rancho Seco, or can you tell from this picture?

A Yes, it is.

Q Okay. This figure, as well as the accompanying text on page 5-36, which appears right in the middle of the page describes how the reactor control panel is on the back wall and located remotely from the reactor control panel.

Underneath the figure, it describes -- it states the rod monitor display shown in figure 5-45 is poorly placed with respect the reactor control panel in the foreground.

Do you know whether SMUD has changed this arrangement since this report was authored?

A Yes, I do know. They have not changed that.

Q I would like you to refer to page 5-42 and figure 5-53 which appears on the preceding page -- two pages preceding that, pardon me -- the preceding page.

CHAIRMAN BOWERS: Which figure?

MR. ELLISON: 5-42. The figure appears on 5-43.

CHAIRMAN BOWERS: I did not recognize that as a separate figure.

BY MR. ELLISON: (Resuming)

Q Does figure 5-43 -- is that a picture of the Rancho Seco control room?

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bfm2

A Yes, it is.

Q Here is described a situation where feedwater controls that were additionally grouped in one location at the primary console in the foreground are now spread out illogically. Is it your -- is it true that the feedwater controls at Rancho Seco were intially on the primary console in the foreground and had been moved?

A No, it is not.

Q Do you believe that this picture is accurate of the Rancho Seco control room as it is today? Let me clarify that question.

I am, of course, referring to with respect to the feedwater controls that are the subject of this comment.

A It is a poor xerox copy, but the general layout appears to be what is at Rancho Seco, the shape of the console, the number of controllers on the individual consoles. The reference to moving the feedwater controls, I cannot comment on unless the author would specifically tell me what he is referring to.

Q Could you take a look at page 5-42 at the very top where the author describes or references that figure? There, he states that modifications to panel design should not violate human factors principles. In some cases --

A I am not with you, again.

Q I am at the top of page 5-42.

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A Mine starts out "our second point."

Q I started in the middle of the sentence. I was trying to abbreviate, but I will start from there. "Our second point is that modifications to panel designshould not violate human factors principles." In some cases, the initial logic of the panel lay-out was violated as modificatons became necessary.

For example, in the lay-out of the primary panels shwon in the foreground in figure 5-43, the feedwater controls were located on the right hand console segment. Later, the panels were backfitted with some feedwater controls on the console segment left of the center panel.

This illogically separates functionally related controls for feedwater pumps and feedwater valves.

The operators complained about this awkward arrangement. Is it your understanding that panels were backfitted at Rancho Seco with feedwater controls on the console segment left of the center panel as is described here?

A No, that was our original design as best I recall. They were not relocated. The controls that I think he is referring to here are auxiliary feedwater controls that are not grouped with main feedwater controls.

Q That situation persists today, is that correct?

A That is correct.

bfm4

Q Are you aware of any operator complaints about that particular arrangement of the feedwater controls?

A The operators, because of I think the familiarity with the simulator, have asked why aren't our auxiliary feedwater controls located by the main feedwater controls.

Q Are they located at the B & W simulator?

A Yes, they are.

Q What response has SMUD made to the questions of the operators about why they are not located together?

A Because the auxiliary feedwater controls essentially control a different system than the main feedwater controls.

(Pause.)

Q In a situation where you require the use of auxiliary feedwater, would it be normal procedure at Rancho Seco for the operators to monitor the operation of the main feedwater system?

A In a situation where the auxiliary feedwater system is required, you would not either want to have the main veedwater system available to you, or you would not be using it. The auxiliary feedwater system is not for all cases, but essentially is required when you have lost main feedwater or when you have lost reactor coolant pump flow.

In the case of loss of reactor coolant pump flow, you want the feedwater coming in at the upper part of the steam generator to enhance natural circulation flow.

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Therefore, you would not need main feedwater.

Q Taking a situation where auxiliary feedwater is necessary because of a loss of main feedwater upon enunciation of the actuation of auxiliary feedwater, would operators confirm that main feedwater was not operating?

A No, their instructions are to confirm that it is operating.

Q I am referring to main feedwater.

A Excuse me.

Q Would you like me to repeat the question?

A Yes.

Q Upon enunciation of the automatic actuation of auxiliary feedwater, would operators confirm that means feedwater is not operating?

A What an operator would do is that he would determine first of all why auxiliary feedwater, as initiated. He would do that one by monitoring main feedwater and see if he has lost that.

That would tell my why it initiated. Then his next look would be at reactor coolant pump flow. As far as confirming that main feedwater has stopped operating, there is no requirement to do that.

(Pause.)

Q Further down the same page, Mr. Rodriguez, on page 5-42, the very next paragraph, the author states, "In the tm6

same control room"-- which I assume is referring to Rancho Seco --"the safeguards panel is functionally grouped into an A segment to the left and a B panel to the right."

"However, a B element has found its way into the A panel as shown in figure 5-54. An A element is now in the B panel. These exceptions compromise the initial logic of the panel lay-out and should be corrected at the first opportunity."

Referring to page 5-54, which is on the previous page, does this accurately depict the current lay-out of the A and B segments of the safeguards panel at Rancho Seco?

A Does what?

Q Does 5-54 and that description that I just read from page 5-42.

A In my copy, I cannot read the labels on the switches, so I cannot tell you whether it is accurately depicting the layout.

Q Do you know whether -- are you familiar with the A and B segments of the safeguards panel at Rancho Seco?

A Yes, I am.

Q Do you know whether, in fact, the description that is given at page 5-42 is accurate?

A Yes, I do.

Q Is it accurate?

A Yes, it is.

Q How difficult would it be to move the indication that we are referring to hear the A and B panel so that all of the A panels are grouped together and all the B panels were grouped together?

A What they are referring to is two switches. They are the only two switches that are backwards. At the time that design was made, there was a great deal of discussion about what you can do to keep it consistent.

In order to maintain fire protection, separation of the cabling. That was the only way that could be put on those panels.

Q This was something that was retrofitted to the facility, is that correct?

A Yes. When I say retrofitted it was not in the original design lay-out. It came on, I think, late in the licensing.

Q So, is it your testimony that it would be very difficult to make the -- to group those functionally related switches together?

A For those two switches, yes. That's what I recall was the reason engineering stated that they could not maintain the A and B logically out for those two switches.

MR. SHON: Excuse me, Mr. Ellison. There is one thing, Mr. Rodriguez. You said they could not maintain fire separation for A and B. It seems putting an A switch

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bfm8

in with the Bs and a B switch in with the As is not a move in the direction of maintaining fire separation but just quite the opposite.

Just off the top of one's head one would think so.

THE WITNESS: Yes, sir. That was the exact statement I used when engineering said this was the way we had to do it. The details of keeping cable separation, I cannot recall, but the supervising electrical engineer and I talked about this at some length.

He finally convinced me that these were the only two switches that were going to occur. He did not have another way of doing it.

MR. SHON: In other words, they're right and we're wrong, but you don't remember why?

THE WITNESS: Yes.

MR. SHON: Thank you.

BY MR. ELLISON: (Resuming)

Q Would I be correct based upon the question that you asked the engineering group that you would prefer to have the panels grouped together?

A That is true.

Q I would like you to refer to page 7-6 and figure 7-9. Figure 7-9 appears at page 7-5. First of all, examining figure 7-9, is that part of the Rancho Seco control room?

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A The Rancho Seco control rod panel looks like this. It could be from some other utility, though. I suspect it is ours.

Q On page 7-6 in the second paragraph, the author states, "Figure 7-9 shows two large selectors with option markings located on a plate that rotates with the knob. The selector switches are part of the rod panel controls located on the reactor control console.

"A stationary reference arrow is used to designate the selected control option associated with each switch. The shape of this knob is inappropriate. It is pointed at both ends and gives the semblance of a pointer knob."

"The stationary small fixed reference arrows are not very prominent by comparison. A round, non-pointing, fluted, or knurled knob with a more priminent stationary reference arrow would be more effective in promoting error-free operation of these controls."

First of all, do you agree with that statement?

A Not necessarily.

Q Could you explain in what you would disagree with it?

A The design of that switch is that the indicator is on the base of the knob as described, and that indicator is to be matched up with that arrow, not the knob.

Q I understand that. However --

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bfm10

A Then, I guess you understand why I disagree with the statement.

Q I understand the statement to say that there would be less chance of somebody confusing that situation and using the shape of the knob rather than the appropriate method that you described if the knob was round. Do you agree with that?

A No, I do not agree with that.

Q Do you think it would be as likely that somebody would point a round knob in the wrong direction as to point a pointed one?

A Again, I come back to the design of the switch. The knob is not the pointer, the knob is simply a group so youcan turn the switch.

Q I understand that. However, I am discussing the situation where somebody accidentally does not understand that and the likelihood that they would do that. My question is is it not more likely that somebody would mistake the knob for a pointer if it were not pointed?

(Laughter.)

If it were ground.

(Laughter.)

A You know, that may be how you look at round knobs and straight knobs. A trained operator is operating these knobs. The only information he has available to him as to

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bfm11

whether or not he is positioning that in a proper place just to look at the base of it.

He has got to match up the base numbers whether it is rod 1-12 or group 1-7 with the arrow. The shape of the knob has nothing to do with that. He has a different indication that he is looking for.

MR. SHON: Am I correct in assuming, Mr. Rodriguez, that there is no way you can change the position of the knob, the pointer-like knob with respect to the scale. The scale moves with it, doesn't it?

THE WITNESS: That is right.

MR. SHON: So, there is no way that he can move that thing to make it point at another rod or another rod bank.

THE WITNESS: There are one or two positions on there where the end of the knob might match up with the arrow, but the function is to select a particular rod or rod group, depending on which one of those knobs.

He has to look at that number and match that with the arrow shown. He can't make the knob move to make it point at anything other than what it points with to begin with as far as the numbers are concerned. Isn't that true?

THE WITNESS: The knob is integral with that label.

BY MR. ELLISON: (Resuming)

Q So the operator could not move the knob to move at

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a different number than is on the base now, but could he move it to a point at the arrow, for example?

(General laughter.)

MR. SHON: Sure.

THE WITNESS: This is a selector switch, and there are distinct contact points that lock it in place. Whether or not the point on the knob matches one of those contact points or not I do not know, because whenever I operated that knob or knobs like that, I look for the number and the arrow and the knob is the device that I use to turn the base.

CHAIRMAN BOWERS: Mr. Ellison, it is time to quit, and plan to resume tomorrow morning at 9:00 o'clock.

(Whereupon, at 5:04 p.m., the hearing was recessed, to reconvene at 9:00 a.m. of 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: 5/6/80

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.

Official Reporter (Typed)
Suzanne R. Babineau

Suzanne R. Babineau

Official Reporter (Signature)

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David S. Parker

Official Reporter (Typed)

Official Reporter (Signature)