

NUCLEAR REGULATORY COMMISSION

ORIGINAL

In the Matter of:

HOUSTON LIGHTING & POWER COMPANY)
)
(Allens Creek Nuclear Generating Station, Unit 1)) DOCKET NO. 50-466 CP
)

DATE: October 6, 1981 PAGES: 17800 thru 18013

AT: Houston, Texas

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1 UNITED STATES OF AMERICA

2 BEFORE THE

3 NUCLEAR REGULATORY COMMISSION

4 In the Matter of:)

5 HOUSTON LIGHTING & POWER)
6 COMPANY)

) Docket No. 50-466 CP

7 Allens Creek Nuclear Generating)
8 Station, Unit 1)

9 Sun Belt Room
10 Eleventh Floor
11 Ramada Inn
12 7787 Katy Freeway
Houston, Texas

13 Tuesday,
14 October 6, 1981

15 PURSUANT TO ADJOURNMENT, the above-entitled
16 matter came on for further hearing at 9:00 a.m.

17 APPEARANCES:

18 Board Members:

19 SHELDON J. WOLFE, Esq., Chairman
20 Administrative Judge
Atomic Safety and Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

21 GUSTAVE A. LINENBERGER
22 Administrative Judge
Atomic Safety and Licensing Board Panel
U. S. Nuclear Regulatory Commission
23 Washington, D. C. 20555

24 DR. E. LEONARD CHEATUM
25 Administrative Judge
Route 3, Box 350A
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APPEARANCES: (continued)

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-and-

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I N D E X

		VOIR			BOARD	
<u>WITNESSES</u>	<u>DIRECT</u>	<u>DIRE</u>	<u>CROSS</u>	<u>REDIRECT</u>	<u>RECROSS</u>	<u>EXAM.</u>
Marvin W. Hodges (Resumed)						
By Mr. Doherty			17,805			
By Judge Cheatum						17,828
By Judge Linenberger						17,830
By Judge Cheatum						17,841
By Judge Wolfe						17,844
By Mr. Copeland					17,846	
By Mr. Doherty					17,849	
By Mr. Dewey				17,850		
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By Judge Cheatum						17,882
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By Mr. Culp	17,887					
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By Mr. Sohinki	17,913					
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			VOIR				BOARD
<u>WITNESSES</u>	<u>DIRECT</u>	<u>DIRE</u>	<u>CROSS</u>	<u>REDIRECT</u>	<u>RECROSS</u>	<u>EXAM.</u>	

Marvin W. Hodges
(Resumed) and
Tai L. Huang
(A Panel)

By Mr. Sohinki	17,938						
By Mr. Doherty		17,941					
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By Mr. Doherty					17,980		
By Mr. Sohinki				17,983			

Charles M. Ferrell
and Leonard Soffer
(A Panel)

By Mr. Dewey	17,986						
By Mr. Doherty		17,989					
By Mr. Doherty			17,994				

P R O C E E D I N G S

9:00 a.m.

JUDGE WOLFE: All right. The hearing is resumed.

In attendance this morning, representing Applicant, Mr. Copeland; Mr. Doherty is present; and representing the Staff, Mr. Sohinki and Mr. Dewey.

Mr. Dewey.

MR. DEWEY: Yes, sir. At this time, Staff would like to present Mr. Wayne Hodges as a witness concerning Board Question 6, RHR System.

JUDGE WOLFE: Let me interrupt just a moment.

I notice our schedule here has the afternoon of October 9th open. Obviously, that's a safety padding there.

The Board would like to leave between 2:30 and 3:00 o'clock on Friday. So in an effort to -- and without tiring the parties and the Board, where necessary we will run beyond 5:00 o'clock in the evening in an effort to keep on schedule, but not where we'll prejudice anyone's physical wellbeing.

As I say, the Board would like to leave between 2:30 and 3:00 o'clock on Friday afternoon, October 9th.

All right, Mr. Dewey.

MR. DEWEY: At this time we would present

1-2 1 Mr. Hodges for cross-examination.

2 Whereupon,

3 MARVIN WAYNE HODGES

4 was recalled as a witness and, having been previously duly
5 sworn to testify the truth, the whole truth and nothing but
6 the truth, was examined and testified further as follows:

7 JUDGE WOLFE: Mr. Copeland, is there cross?

8 MR. COPELAND: No, sir.

9 JUDGE WOLFE: Mr. Doherty.

10 MR. DOHERTY: Yes, Your Honor.

11 CROSS-EXAMINATION

12 BY MR. DOHERTY:

13 Q Mr. Hodges, have you been a participant in any
14 meetings or any -- yeah, any meetings with regard to this
15 issue outside of perhaps a meeting about Allens Creek in
16 this issue?

17 A Well, on Board Question 6 there really are two
18 separate issues, but to answer your question, yes.

19 Q Which ones did you attend?

20 A I've attended several meetings that related to
21 the testing of the relief valves for the alternate method
22 for shutdown cooling.

23 In addition, I've had a number of discussions,
24 although not attended meetings, with vendors or applicants,
25 on the steam condensing problem.

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1 Q Well, is this issue the same as that called
2 Task Analysis Plan 45, or TAP-45, or is this in your mind
3 separate, a different issue?

4 A There is a Task Action Plan on water hammer. I
5 don't recall the number of it. It may be 45. I don't
6 think it is, but I don't recall the number.

7 There is not a Task Action Plan item for the
8 shutdown cooling, not a concern that was raised by the
9 ACRS at any rate.

10 Q You didn't attend any such thing or haven't
11 been involved in any such thing, anyway; is that right?

12 A For the shutdown cooling I have attended
13 meetings that discussed the testing of the valves to
14 demonstrate the viability of that mode.

15 Q Okay. Well, the question sets up the
16 possibility of remedial measures.

17 In your opinion, have there been any remedial
18 measures taken or is the entire thing mainly a matter of
19 demonstration?

20 A For the water hammer problem or the shutdown
21 cooling problem? Again, there really are two separate
22 issues.

23 Q Yes, okay.

24 For the shutdown cooling problem.

25 A Okay. Yes, there has been a remedy proposed

1 and there has been testing done on the relief valves which
2 were involved in using this approach which have shown that
3 this is a viable approach, and then it has been accepted
4 as a reasonable approach by the NRC Staff.

5 Q What about the water hammer problem? What
6 remedial measures have gone on with that?

7 A The water hammer problem, we are still in the
8 investigatory stage of it. There have been no remedial
9 actions imposed by the NRC.

10 There are actions that are taken by the various
11 utilities to try to prevent the water hammer from occurring;
12 but at this stage we have not proposed a solution.

13 Q On page 17 there is toward the bottom -- I had
14 a question.

15 What does this equipment consist of, this
16 residual heat removal equipment? What is it and where is
17 it?

18 A Okay. The residual heat removal system itself
19 consists of three low pressure pumps that are the same pumps
20 that are used with the low pressure coolant injection
21 system.

22 It consists of two heat exchangers and in the
23 normal shutdown cooling mode the residual heat removal
24 system takes suction from one of the recirculation loops
25 coming off the reactor vessel.

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1 There is a single letdown line coming from
2 this recirculation loop that has two valves, two isolation
3 valves in that line so that it would take the water from
4 the reactor vessel, hump it through the residual heat removal
5 heat exchangers and back into the vessel.

6 That's the normal mode of operation.

7 The concern was that a single failure in either
8 one of those isolation valves in the letdown line coming
9 from the reactor vessel would incapacitate the system; and
10 the NRC has required that the reactor be capable of being
11 brought to a cold shutdown using only safety grade
12 equipment and considering a single failure.

13 So we take a failure in one of these isolation
14 valves in the letdown line and then the normal mode of
15 residual heat removal cooling is inoperable.

16 The alternative that has been proposed by the
17 BWR Owners is to fill the reactor vessel with water. You
18 are at low pressure conditions.

19 They would use the ADS valves to open up, at
20 least one of the ADS valves, and pump water using an RHR
21 pump and the low pressure coolant injection mode so they
22 would be pumping from the suppression pool through the
23 heat exchanger and into the reactor vessel. And then you
24 would overflow the reactor vessel through a safety/relief
25 valve and that would discharge back into the suppression

1 pool.

2 Q Where are the heat exchangers themselves? Where
3 are they physically located?

4 A They are located in the reactor building, but
5 not inside containment.

6 Q They work on the same principle as the steam
7 generators in a PWR?

8 A In the normal mode you have a surface water
9 system that is supplying water from your ultimate heat
10 sink, whether it's a cooling pond or the river, and that
11 is circulating inside the tube of the heat exchanger, and
12 then on the shell side of the heat exchanger you have the
13 water that's coming from the suppression pool, or in the
14 normal RHR mode it's coming from the reactor vessel itself
15 going through the heat exchanger, and heat is being
16 removed by the surface water system.

17 There is normally no steam generation in that
18 heat exchanger; but otherwise, it works somewhat similar
19 to a steam generator.

20 Q Now, toward the foot of 17 you talk about the
21 alternate shutdown cooling method.

22 When would it be -- on what would it be
23 decided to use the alternate method instead of the normal
24 method?

25 MR. COPELAND: He just explained that, Your

L-7
1 Honor.

2 MR. DOHERTY: No, I don't recall him doing that.
3 Maybe I can make my question a little tighter.

4 BY MR. DOHERTY:

5 Q On what evidence would the operators decide to
6 use the alternate method instead of the normal one?

7 A If he found he could not open one of the two
8 valves in the letdown line coming from the reactor vessel.

9 Q It states there at the foot that, "It is to
10 flood the vessel to the elevation of the steam lines."

11 Are those below the -- Are the steam valves
12 below or above the -- they are called ADS valves, I guess.
13 I'm having a little trouble with that.

14 A Okay. The ADS valves are a subset of the
15 safety/relief valves.

16 Q Yes, okay.

17 A They are located on the steam lines themselves,
18 and when we talk about them flooding up to the elevation
19 of the steam lines, the steam lines are located near the
20 top of the reactor vessel.

21 They edge at the reactor vessel and then drop
22 vertically about 40 or 50 feet and then they run horizontally
23 for some distance. The safety/relief valves themselves
24 are located on that horizontal run of the pipe.

25 So we are talking about flooding the reactor

1 vessel up to the elevation of the steam lines. You are
2 talking about up to near the top of the vessel.

3 Q The water source is suppression pool water for
4 the flood, right?

5 A Yes, that's one source. That's not the only
6 source, but that's one source.

7 You can also fill it up from the condensation
8 storage tank using a spray system.

9 Q Which spray system is that, part of the ECCS
10 or something additional?

11 A You could use -- in this case you would be
12 using the low pressure core spray system, which is part
13 of the ECCS.

14 Excuse me, I'm sorry. That does not -- basically,
15 yes, the water is coming from the suppression pool.

16 Q In a normal shutdown, say for refueling, is
17 the residual heat removal system used at all, or does it
18 just sit quietly and wait?

19 A In the normal shutdown, that is the normal
20 method of cooling using the residual heat removal system.

21 Q So then it's operated fairly frequently as a
22 system; it gets used?

23 A Yes.

24 Q Okay. Well, has this ever occurred, this
25 failure to open a valve when they went to a normal shutdown,

1 to your knowledge? Has anyone ever reported that?

2 A I'm not aware of it happening. It may have; I'm
3 not aware of it.

4 Q Do you know what started this concern? Did it
5 just come out of someone's head one morning, or was there
6 an incident, or do you know of anything like that?

7 A In reviewing the Safety Analysis Reports for
8 the various plants, we are always looking for the effects of
9 single failures and this was one single failure that we
10 located through the process of review, which would not
11 permit the use of the normal shutdown cooling equipment,
12 and so was a violation of our interpretation of General
13 Design Criterion 34.

14 I don't think it was necessarily prompted by
15 any particular event.

16 Q Now, at the top of 18 there, "Residual heat
17 removal system operating in the low pressure injection
18 mode would thus remove the residual heat discharged to
19 the suppression pool via the ADS valves."

20 Is this going to wind up a slower process than
21 the normal process, to your knowledge?

22 A It would be somewhat slower because the
23 temperature of the suppression pool would be lower, so
24 the heat exchangers would take the heat out of it at a
25 slower rate.

1 Q Now, do some of the accident analyses, such as
2 the design basis LOCA and -- well, that one, for example --
3 don't they forecast that there will be -- prior to using
4 the residual heat removal system, that there will be
5 blowdown into the suppression pool?

6 A Yes.

7 Q Would that tend to warm the suppression pool
8 water up?

9 A Yes.

10 Q Was that investigated as a possible problem,
11 the fact that the suppression pool would be heated up
12 from perhaps a previous event, such that to call on this
13 alternate system you would really be calling on warmer
14 water than was normally in the suppression pool?

15 A Okay. A couple of points. First off, you are
16 talking about a normal type of shutdown with no accident
17 having occurred, the only problem being that you could not
18 open one of these valves in the letdown line.

19 Secondly, there are technical specifications on
20 the temperature in the suppression pool, and so whenever for
21 any reason the suppression pool temperature had been
22 elevated due to a discharge through a relief valve or
23 whatever, then the operator would be required by the
24 technical specifications to put the residual heat removal
25 system in the suppression pool cooling mode, which just

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1 pumps water from the suppression pool through the RHR
2 heat exchanger and then back to the suppression pool to
3 remove that heat.

4 That's a normal mode of operation of that
5 system.

6 Q Well, if there is a design basis event and
7 then the event got to the position or place where the
8 operator decided that the residual heat removal system
9 should be called in use, that the other systems had done
10 their work, that the pressure was low enough, whatever
11 those problems would be, would it then be somewhat of a
12 problem if indeed you had this blocked letdown line, if
13 he had to call on the warmer water in the suppression pool
14 because it had been warmed previously?

15 MR. COPELAND: I'm going to object to that
16 question.

17 As I understand the witness' last answer,
18 that the concern that is a part of the Board question
19 relates to using the RHR system during normal shutdown.

20 What Mr. Doherty is now doing, which is
21 obviously by this witness' testimony quite remote and
22 speculative anyway, he is now stacking another fairly
23 remote event on top of that and assuming a double failure,
24 not a single failure.

25 He is assuming a LOCA followed by a single

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1 failure of the RHR system, and it seems to me that's beyond
2 the scope of what the ACRS' concern was and beyond the
3 scope of the Board's question.

4 (Bench conference.)

5 MR. DOHERTY: Looking at Criterion 34, which
6 is the single failure criterion for residual heat
7 removal, I think in the question I've only assumed a single
8 failure within the residual heat removal system. So I
9 don't think the objection is valid.

10 JUDGE LINENBERGER: I think the Board needs a
11 clarification here.

12 Is a blockage of this line for whatever reason
13 stacked on top of a design basis accident considered to be
14 a single failure situation or not? I don't really know.

15 THE WITNESS: In the Staff's evaluation of a
16 design basis accident, we normally consider one single
17 failure.

18 Now, a single failure may be a failure that
19 affects more than one system, but it could be a common
20 failure, a power supply, for example, that knocks out
21 several systems; but we always consider one single failure.

22 If he chooses to take that valve as a single
23 failure, we can do so. It's not a serious problem.

24 JUDGE WOLFE: What leads you to argue that
25 the ACRS concerns were so limited? Do you have the ACRS

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1 letter?

2 MR. COPELAND: I was basing that on the
3 witness' testimony, his written testimony, and what he
4 said on cross-examination, which was that the concern
5 related to a single failure in the RHR system.

6 Now, obviously, the witness has read the
7 letter and is familiar with the concern expressed there.
8 So that's all I have before me, Your Honor.

9 He cites it in his testimony at page 17.

10 JUDGE WOLFE: You may have a point, Mr. Copeland,
11 but the Board is interested in getting knowledge on this
12 particular point.

13 Objection overruled. Answer the question,
14 Mr. Hodges.

15 THE WITNESS: Okay.

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-14 1 MR. DEWEY: Your Honor, I'd like to make a
2 Staff objection.

3 JUDGE WOLFE: Yes.

4 MR. DEWEY: I believe that the witness just
5 previously testified that you would not get to the
6 situation that Mr. Doherty is describing because of the
7 fact that the suppression pool water would be in a cooling
8 phase from the RHR system.

9 So that problem wouldn't exist that Mr. Doherty
10 is talking about, the way I understand it.

11 JUDGE WOLFE: Well, let the witness answer.

12 All right, Mr. Hodges.

13 THE WITNESS: Yes, and I also have a copy of
14 the ACRS letter, if you are interested.

15 JUDGE WOLFE: All right.

16 THE WITNESS: But for the design basis
17 accident that Mr. Doherty proposed, you would still be
18 using the low pressure coolant injection system for the
19 long-term cooling mode.

20 The water would run out the break and back to
21 the suppression pool, and then you would pump from the
22 suppression pool through the RHR heat exchanger back to
23 the vessel.

24 BY MR. DOHERTY:

25 Q Well, do you know if during a design basis

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1 accident the operator is expected to do suppression pool
2 cooling through the residual heat removal system?

3 A He -- yes, he is expected to do suppression
4 pool cooling through the RHR system; but then that can also
5 be effectively done by pumping into the reactor vessel
6 through the RHR heat exchangers and accomplish the same
7 thing.

8 He will initially put his RHR system in the
9 suppression pool cooling mode and then after some period of
10 time after things have settled down, he will then switch
11 to injecting into the vessel through the RHR heat
12 exchangers.

13 Q Is there any practical difference between these
14 two things right now in your mind?

15 MR. COPELAND: I'm sorry, Mr. Doherty. What
16 two things?

17 MR. DOHERTY: Well, he's describing what
18 apparently are two different ways in which water would be
19 getting back to the vessel.

20 THE WITNESS: With the residual heat removal
21 system you can pump to the reactor vessel either through
22 the heat exchangers or not through the heat exchangers.

23 The normal low pressure coolant injection mode
24 that is called upon by the ECCS signals immediately following
25 a design basis event, the low pressure coolant injection

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1 system would pump directly into the vessel without going
2 through the heat exchangers. But then at a reasonable time
3 later, a half hour or so, then the operator would switch
4 to injection through the heat exchangers.

5 BY MR. DOHERTY:

6 Q. So the only difference is that it gets a run
7 through this exchanger in one and that might get some
8 cooling done?

9 A. That's correct.

10 Q. Now, on page 18 I had a question. We may have
11 got an idea of this answer, but I still want to ask it.

12 That is with regard to your answer to the
13 second question on page 18. In the first line you talk
14 about "primary water."

15 That water is the water that's being used,
16 circulated through the vessel? That's what you mean by
17 "primary water" in that sentence?

18 A. That's what I mean by "primary water," yes.

19 Q. Okay. Where you say in that same long
20 answer, "Steam is admitted at the top of the shell," do you
21 mean there's an inlet for steam there or do you mean -- it
22 looks like you are nodding yes.

23 A. I'm waiting for you to finish your question

24 Q. Okay. Or do you mean there's just steam there
25 because you've presented a space where steam could occur

1-17 1 because you've dropped the water level apparently there?

2 A. There is a pipe at the top of the heat
3 exchanger shell that can be supplied with steam from the
4 steam lines, yes.

5 Q. I don't understand how there could be any
6 possibility of cold water being admitted to the shell side
7 in the situation we are in. How could that happen? As
8 an operator error?

9 A. The piping that comes into the top of the vessel
10 is also connected to a source of water so that if a valve
11 in that pipe were inadvertently opened, then you could
12 dump liquid water into the steam space in the top of the
13 heat exchangers.

14 Q. Is it your -- well, I think you said earlier
15 that the water hammer problem is still being investigated.

16 A. Yes.

17 Q. Do you know if there's any consideration being
18 given to simply making it impossible for that ever to
19 happen by some type of an automatic cut-off of the water
20 supply to the shell or anything like that? Does that seem
21 like a reasonable proposal, or have you ever heard of such
22 a proposal?

23 A. Well, there are already valves that have key
24 lock switches so that the only way they can be opened is
25 for the operator to go to a shift supervisor and say, "I

1 need the key for that valve," and the supervisor would
2 possibly question him as to why he needed to open that
3 valve in this mode.

4 To my knowledge, there is no way to completely
5 preclude the opening of -- an operator error.

6 You can take extensive measures to minimize
7 them, but you can never completely preclude them.

8 Q Well, let's see. In normal operation of the
9 residual heat removal system, is there any steam condensing
10 done by that system?

11 A Well, the steam condensing mode is one of the
12 normal operation modes, but that's not any different than --
13 the normal residual heat removal mode, there is no steam
14 in there and there is no steam condensing going on.

15 Q I notice you say, "Water hammer has never been
16 reported during the steam condensing mode."

17 How would water hammer come to the attention of
18 the Commission?

19 A It would come to our attention if the results
20 of the water hammer had been such there has been damage to
21 pipe hangers or to supports or something of that nature
22 where there have been some noticeable effects.

23 Q So you are not saying the process hasn't
24 happened? There could have been a bang in there, but there
25 just wasn't any --

1 A. If there are no visible effects that can be
2 detected either when the heat exchangers are inspected
3 routinely or they are obvious, then if it occurred, it's
4 mild enough that it's obviously not a problem.

5 Q. Now, does water hammer occur in other heat
6 exchangers sometimes?

7 A. It has, yes.

8 Q. Like large exchangers like in PWR's, is that
9 a common problem?

10 A. It's not just a heat exchanger problem. It
11 has occurred in a number of different systems.

12 It's occurred in PWR's and BWR's.

13 Q. Uh-huh. When this has occurred, has this ever
14 actually resulted in breaking the tubes?

15 MR. COPELAND: Your Honor, I object to any
16 further questions along the lines of some unrelated piping
17 system in a PWR.

18 It seems to me to have absolutely no relevance
19 to the question that's raised here.

20 MR. DOHERTY: He stated it occurred in both
21 PWR's and BWR's in heat exchangers, and I'm asking him --
22 not speaking of PWR's.

23 MR. COPELAND: You are asking if water hammer
24 in the RHR system has ever caused any damage in a BWR?

25 MR. DOHERTY: No, I'm not asking him that. I

1 don't know how tight it is, but he said that they had
2 occurred in the heat exchangers.

3 I'm not certain this is the only heat exchanger
4 in the BWR so I'm not trying to limit him to that.

5 MR. COPELAND: I guess I just don't understand
6 the question then, because his testimony says on page 19
7 that, "Water hammer damage has not appeared to result in
8 any unsafe conditions in a RHR system."

9 I don't understand how your question is any
10 different than what he's already said here.

11 MR. DOHERTY: Well, I don't have anything
12 further to say, Your Honor, to the objection.

13 JUDGE WOLFE: Well, precisely what are you
14 seeking to explore?

15 MR. DOHERTY: Well, I'm trying to find out
16 essentially what damage has occurred in heat exchangers
17 in BWR's from water hammer.

18 (Bench conference.)

19 MR. COPELAND: Well, I don't have any objection
20 to that question. I think that's understandable.

21 I don't know what the point of it is if the
22 witness' testimony is that it hasn't resulted in an unsafe
23 condition.

24 I mean, it seems to me a rather pointless
25 question.

1 JUDGE WOLFE: Well, it may lead to something.
2 Let's see how we go.

3 Objection overruled. Answer that question.

4 THE WITNESS: The last question?

5 JUDGE WOLFE: Yes.

6 THE WITNESS: Okay. In -- for the RHR heat
7 exchangers there have been damage to piping supports;
8 there have been some cracks in some welds of piping.

9 For the isolation condenser for some of the
10 older boiling water reactors, it's basically also a
11 heat exchanger, there has been, again, damage to pipe
12 supports.

13 To my knowledge, there has been no damage to
14 the tubes from the water hammer. There have been water
15 hammers in RHR heat exchangers due to other things than --
16 like the steam condensers.

17 There have been water hammers in the surface
18 water systems that have caused some bending of a plate in
19 the heat exchanger itself.

20 But to my knowledge, that's been the extent of
21 the damage.

22 BY MR. DOHERTY:

23 Q On page 19 you speak about steam pockets caused
24 by leaking. Are these very large things or what are they?
25 What size?

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1 A Well, I don't know the absolute size, but as
2 I mentioned earlier, there is a steam line that's connected
3 where the water line is near the top, or on the top of the
4 heat exchanger shell. It's got a series of valves in those
5 and sometimes these valves leak slightly, and so you can
6 get a pocket of steam.

7 It's not going to fill the whole heat exchanger
8 by any means, but if it's a pocket of steam, when they
9 start up the normal RHR pump to pump water through there,
10 then it puts a slug of cold water in there with the steam
11 and it collapses. The steam bubble collapses, excuse me.

12 Q I notice in one of the results that you give
13 under Unresolved Safety Issue One, "Total avoidance of the
14 potential for water hammer phenomenon is not practical."

15 Let me ask you this. How long has this study
16 been going on?

17 A I think the study was started back in something
18 like 1978. There has been a NUREG published, essentially
19 like a status report, on the water hammer problem; but
20 the work is still continuing.

21 I think it's scheduled to be completed in
22 late '82 or '83, something like that.

23 Q I notice back on page 18, the way you've
24 worded this, at the middle of the large answer there in the
25 bottom quarter of the page there roughly, "If cold water

1 were admitted to the shell side of the heat exchanger..."
2 and then you go on. Has that ever happened, to your
3 knowledge?

4 A That has happened in the sense that we talked
5 about these steam pockets being formed and you start up
6 a pump; but where you have the large steam void, as you
7 would have for the steam condensing mode, that has never
8 happened.

9 Q So then would you say that the ACRS concern is
10 still sort of futuristic? It may happen someday kind of
11 fear or concern?

12 A The ACRS, just like the NRC Staff, tries to
13 look at all the possibilities and protect against them.
14 The fact that it has not occurred doesn't mean that we're
15 not going to question it.

16 Q Okay. Is there any kind of use of small
17 pumps to make up these steam spaces which have a potential
18 for water hammer? Is there any possible way for doing
19 that that you know of?

20 A You mean to fill them with water first or --

21 Q Yeah, something like that.

22 A There are small pumps connected to the system
23 that are basically used as keep-filled pumps. They are
24 very low capacity pumps, but that still wouldn't solve the
25 problem, because you still have large pumps available and

1 if somebody made a mistake, it could still happen.

2 You have to guard against it with procedures
3 and other things, but the large pumps are still there.

4 MR. DOHERTY: Thank you. Your witness, sir.

5 JUDGE WOLFE: Is there redirect, Mr. Dewey?

6 MR. DEWEY: No, sir.

7 JUDGE WOLFE: Board questions?

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JUDGE CHEATUM: I have one question.

BOARD EXAMINATION

BY JUDGE CHEATUM:

Q At Page 19, Mr. Hodges, the middle of the page

A Yes.

Q You say, "The ACRS's concerns regarding potential damage induced by hydrodynamic forces are being addressed as one of several types of water hammer effects being generically studied"

Water hammer effects is one effect of potential damage induced by hydrodynamic forces? This sentence leaves me a bit confused.

A Okay. Now, there are several types of water hammers. And so what we're trying to say is we're addressing a water hammer that is resulting from the collapsing of a steam void by the injection of cold water.

You can also get a water hammer by closing a valve too rapidly or having an unfilled line and starting up a pump and pumping against the closed valve or something like that.

There are several types or sources of water hammers.

And all I was trying to say with that sentence is that that water hammer that might result due to the condensing of the steam is just one one of several types

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1 of water hammers that we're looking at, and we're going to
2 try to come at a resolution to.

3 Q Okay, I guess that clarifies that a little
4 bit.

5 You described, in response to Mr. Doherty's
6 question about the types of damage which may occur as a
7 result of one -- and among the types of damage,
8 mechanical damages -- which one you said -- you men-
9 tioned cracked pipe.

10 A I think a crack in the weld.

11 Q A crack in the weld, okay.

12 A Yes.

13 Q A crack in the weld. Is this the kind of thing
14 that can be remedied through improved welding or -- so
15 as to -- in other words, aren't there things that can be
16 done to make this kind of damage less likely from water
17 hammer effects?

18 A I think you could probably improve the supports,
19 improve maybe some of your inspection techniques on the
20 welding to minimize that type of welding or that type of
21 failure.

22 But when I was trying to respond to his earlier
23 question, as far as eliminating it, I was talking about
24 the root cause was just the water hammer itself. That's
25 almost impossible to eliminate completely.

1 But, yes, you can design to try to minimize the
2 effects of the water hammer.

3 Q Well, is that being done?

4 A Yes.

5 Q I see. Okay.

6 JUDGE CHEATUM: I have no more questions.

7 BOARD EXAMINATION

8 BY JUDGE LINENBERGER:

9 Q I guess we might as well stay with water hammer
10 for a moment more.

11 In the first place, what is the method by which --
12 or phenomenon by which a plant operator would know that
13 a water hammer event had occurred?

14 A In the RHR heat exchanger or anywhere?

15 Q Let's confine it to this part of the system.

16 A Okay.

17 He would really only know it because of any
18 damage that might result from it; because it's located
19 remote from the control room, he wouldn't hear the sound
20 that you would get with it, unless there was like
21 an auxiliary operator nearby.

22 For the water hammer that have occurred, for
23 example, in the isolation condensers on some of the
24 older plants, they actually hear it from the control
25 room when it occurs.

1 I wouldn't expect he would hear the water
2 hammer and RHR heat exchanger unless there just happened
3 to be someone nearby.

4 So it would have to be picked up with the
5 inspection -- say, during a plant shutdown of piping
6 supports, welds, just the general condition of the heat
7 exchanger.

8 And, in fact, there have been water hammers
9 in the service water systems that were not detected until
10 after the plant was shut down, and they had removed the
11 heat exchanger for routine maintenance; and they located
12 damage.

13 Q Well, that's related to my concern, which is
14 the possibility that a water hammer event may have occurred,
15 may have caused damage, and if its occurrence were un-
16 known, there would be no reason to do any surveillance
17 for -- or inspection for that damage until perhaps the
18 next routine shutdown.

19 Now, let's assume for a moment that -- let's
20 get away from the RHR system for just a moment and assume
21 that a water hammer has occurred in an audible way, such
22 that a plant operator realizes that, and thinking back
23 over what has been done to control the system sees,
24 "Well, yes, I shouldn't have done it that way, but there
25 it is. We had a water hammer bang in the system."

1 Now, is there any kind of requirement in the
2 tech specs that says, "Thou shalt shut down and look for
3 damage, having detected this occurrence"?

4 A. I don't think so.

5 Q. Do you think it would be prudent to have
6 acoustic pick-ups strategically placed in remote parts
7 of the system, such that the occurrence of a water hammer
8 would be made audible?

9 I'm asking for your professional opinion here.

10 A. Yes. I'm trying to think for just a minute.
11 If you install something in such a way that it would give
12 you the signal that you wanted on the water hammer and
13 not a lot of spurious indications, I think it would be a
14 good system.

15 I'm a little bit concerned that every time you
16 start a pump up in the system or a neighboring system,
17 that if you have it sensitive enough to detect -- say,
18 a mild water hammer or whatever level of water hammer
19 you think you need to detect against, it might also be
20 activated a number of times spuriously.

21 So the concept, I think, is a good idea. In
22 practical applications it may be difficult to implement.

23 Q. Okay. Finally, at least with respect to water
24 hammers, that is, at the end of your testimony on Page 19,
25 you recite three results or conclusions. Are you aware --

1 Well, you also have adverted to certain locked valves
2 that would have to be knowingly and intentionally opened
3 as a measure to prevent or reduce the occurrence of water
4 hammer.

5 Are there any considerations with respect to
6 general operating instructions not involving key-locked
7 valves, just modus operandi that operators are asked
8 to -- or are cautioned to observe in the operation
9 of a plant to help minimize the occurrence of water
10 hammers?

11 A. There are some general instructions, I think,
12 for example, for the steam condensing mode, that instruct
13 the operators to open the valves very slowly.

14 This is at least in part to help with the water
15 hammer problem. It also helps with the problem of --
16 I think on the wire drawing of some of the valves that
17 they have.

18 So it's an operational convenience problem,
19 in addition to the water hammer problem. I think this
20 may be one area where the operating procedures need to be
21 strengthened and the training improved, to instruct and
22 caution the operator against the hazards of water
23 hammer and the steps he needs to do to try to minimize
24 it.

25 I think that may be one weak area that we need

1 to work on.

2 But the procedures that I'm familiar with, they
3 do not explicitly say, "Do this to prevent water hammer."

4 Q At the top of Page 18, Mr. Hodges, in the first
5 answer that appears there, there are two sentences, one
6 of which refers to testing of relief valves to demonstrate
7 certain capabilities.

8 And the second sentence refers to calculations
9 with respect to the existence of excess capability.
10 Now, I just want to understand how the tests and the cal-
11 culations relate.

12 But, first, let me ask you: Do they relate?
13 Are they related? And if so, how?

14 And then finally, who is doing the testing and
15 who is doing the analytical calculations?

16 A Okay. First off, this particular piece of
17 testimony was written two or three months ago. And since
18 this testimony has been written, the tests have now been
19 completed.

20 And the tests have demonstrated that the relief
21 valve will pass water in sufficient quantities for a single
22 valve to be effective in this mode of operations. So we
23 now have the test data in hand.

24 Q From valve vendors?

25 A The tests were conducted by General Electric,

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? 1 but they had valves from Crosby, from Dakers and from
2 Target-Rothen -- all of the valves that are used on boiling
3 water reactors; and they actually conducted the test pro-
4 gram.

5 And so the program has now been completed as
6 far as this testing is concerned. The NRC just recently,
7 in the past few weeks, received a data report.

8 But at least the first review of the data shows
9 that all of the valves performed satisfactorily. The rea-
10 son for doing the tests, of course, is because the valves
11 are designed to pass steam.

12 And now you're calling on them to pass water,
13 and you're looking to make sure the valves will close
14 satisfactorily after it passes, there's no damage to the
15 valves and you get the flow rates that you would antici-
16 pate through the valves.

17 Those are the purposes of the tests. And they
18 have been successful.

19 And the calculations that we're referring to
20 really relate to looking at the resistances through the
21 system, the pump that's pumping the gas and trying to
22 determine how many valves would need to be available --
23 would need to be open in order to get sufficient flow.

24 Those are normally done by the architect en-
25 gineer.

1 Q You say normally. With respect to Allens Creek --

2 A I would suspect they have not been done yet at
3 the CP stage because all of the piping details have not
4 been determined. But they would be done by the architect
5 engineer at the operating license stage.

6 Q Does the kind of testing that you're talking
7 about here include such considerations as whether -- if
8 there's a power or motor operated valve involved, that
9 the power available from the closing mechanism functions --
10 is adequate for liquid flow, as well as for vapor flow;
11 is that one of the kinds of things that's looked at? Or
12 is the valve so designed that there's adequate power regard-
13 less of whether it's liquid or vapor phase in the line?

14 A I'm trying to remember whether they've tested
15 whether or not the valves would close with water flow
16 going through them.

17 I know that they retested the valves. They
18 tested them in steam at 1000 pounds pressure, and they were
19 tested in water at low pressure, and again in steam at
20 high pressure to make sure there had been no damage to
21 the valves, and it still operated normally at the high
22 pressure mode.

23 I don't recall for sure whether they tested for
24 reclosing with the water flow going through there or not.
25 I just don't remember.

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1 But I know that they did make -- did test the
2 valves again at 1000 pounds with steam to show that they
3 would perform -- would seat adequately and they would open
4 and close at the right pressures.

5 Q In response to a question by Mr. Doherty, you
6 indicated that if the reactor heat removal system were
7 operating in a mode that involved flow via the automatic
8 depressurization system valves, that that mode of operation
9 would be -- I'm not sure I'm characterizing your words
10 correctly -- I think you said would be slower because of a
11 different temperature regime which would result in a lower
12 rate of heat removal; is that --

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THE WITNESS: I think he was asking about the cool down rate of the reactor vessel --

BY JUDGE LINENBERGER:

Q Yes.

A -- the water in the vessel.

And since the -- at the point where you would normally cut in the residual heat removal system, the temperature of the water is about 340 -- 300 to 340 degrees in the reactor.

And the temperature of the water in the suppression pool is normally limited by tech specs to 90 or 95 degrees. And the driving force for the removal through the heat exchangers -- the thermal driving force is lower going through the suppression pool; and that's all that I was referring to.

Q Right.

Now, what's the comparison, roughly, so far as volumetric flow rates are concerned, between the two paths via the ADS valves or through the line that would normally be used if there were not an obstruction?

Or are the volumetric flow rates are comparable?

A I don't think they're identical, but they're comparable.

Q Not a factor of ten difference?

A They're both in the neighborhood of, say, 7000

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1 gallons per minute.

2 So, you know, there would be maybe a ten percent
3 difference or something like that. But they're comparable.

4 Q Okay, thanks.

5 Now, finally, at Page 18 again, the second
6 answer, talking about the -- what happens to water in the
7 shell side of the RHR heat exchanger. And that water is
8 drained until the shell side level, your testimony says,
9 is about 75 percent of the level set point.

10 Okay. All I really want to know here is what's
11 the possibility or confusion here about how much water
12 there is? That 75 percent level set point, how is the
13 water level determined? Might there be some two-phased
14 flow or something that could mislead somebody, or mislead
15 the detectors in the system itself as to where the water
16 level is?

17 A No. At the point where he's draining the level
18 down, basically he has vented the top of the heat ex-
19 changer to the atmosphere, and so he doesn't draw a vacuum
20 into the heat exchanger as he's reducing the level. He
21 hasn't admitted steam yet.

22 He's just lowering the water level and draining
23 water out.

24 Q And how is the level indication?

25 A It will be a differential of pressures, an

1 indication similar to what you would use on a level in the
2 vessel or something.

3 Q Okay. I was going to ask you if it was that
4 kind of system --

5 A It's a differential pressure system, yes.

6 JUDGE LINENBERGER: All right, sir. Thank you.
7 That's all I have, Mr. Chairman.

8 (Pause.)

9 BY JUDGE LINENBERGER:

10 Q Mr. Hodges, back to an earlier question where I
11 was improperly venting an acoustic detection system to
12 tell the plant operators that a water hammer had occurred,
13 let me approach that from a slightly different direction.

14 I'm just concerned about the possibility that
15 subsequent to a water hammer event, or because of a water
16 hammer event, there may have been enough damage somewhere
17 that would warrant a fairly immediate shutdown and inspection.

18 And if the event occurs unnoticed -- or noted --
19 whether thereby there might arise a potentially dangerous
20 situation. Is this kind of thing something that the NRC
21 is concerned about?

22 A I personally have concerns on that because I
23 know of some cases on operating plants where even when the
24 operator was aware that a water hammer had occurred, he
25 continued to use the system.

1 There are other people on the NRC staff who
2 are also asking similar types of questions. So we're not
3 totally happy with the current status.

4 (Bench conference.)

5 BOARD EXAMINATION

6 BY JUDGE CHEATUM:

7 Q Following your comment there, after Judge
8 Linenberger's question, I am wondering whether -- or what
9 your comment would be to this, that you recognize there
10 are damages possible which might be serious.

11 And the question is: Are the systems that are
12 subject to water hammer and the types of damages which
13 can occur from water hammer which have been observed
14 now, are they critical systems, which could lead to major --
15 if the damage is undertaken could lead to major damage
16 or accidents?

17 This is really, I guess, what I have in mind.
18 Do you understand?

19 A No -- I understand what you're asking. That's
20 a good question.

21 If you take the RHR heat exchangers themselves --

22 Q Yes.

23 A -- they are used for long-term cooling; they are
24 used in the normal shutdown cooling; they're used for
25 suppression pool cooling. They're a very important piece

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2-15 1 of equipment.

2 And if you make them operable, say, if you make
3 both of them inoperable, then that, I think, is serious.
4 There has been within the past year one case that I'm aware
5 of where both RHR heat exchangers have experienced damage
6 due to water hammer. It was not a steam condensing
7 water hammer.

8 It was just a water hammer in the service
9 water system. But in the plenum where the service water
10 system enters the RHR heat exchanger, it flows up through a
11 set of tubes and back out with -- the same plenum with
12 just a plate to separate the two halves of the plenum.

13 The plate that separates the two halves has
14 been displaced about a foot up, so there was a leakage
15 path, and the plate was bent. So the amount of service
16 water that you could get -- to get the cooling function
17 would be greatly diminished.

18 A lot of it would just bypass the heat ex-
19 changer. As far as I'm concerned, that's a serious
20 problem that we're going to have to work on.

21 There are alternative cooling modes that can be
22 called upon and were called upon in that case, because
23 they discovered the failure in one heat exchanger during
24 the routine maintenance while they were out.

25 After a plant has been shut down for a period of

1 time, the decay heat is low enough that they can remove
2 both heat exchangers for a period of hours and just let
3 the reactor heat up before they put it back in service.
4 And they sometimes do this for maintenance purposes.

5 They took the other heat exchanger out of ser-
6 vice and inspected it as well, and found that the same
7 damage had occurred on both heat exchangers. So although
8 they were getting some cooling function from the two,
9 they had not really observed it at normal operation.

10 Both heat exchangers -- the integrity of the
11 heat exchangers (if you want to call it that) had been com-
12 promised. What they wound up doing in that case was in
13 using the fuel pool heat exchangers in place of an RHR
14 heat exchanger while they repaired the RHR heat ex-
15 changers.

16 So there is another heat exchanger available
17 that can be used as a back-up. It's not the one that is
18 normally counted upon in all of our safety analyses. But
19 it's there.

20 If you are at some elevated temperature and
21 pressure, normally you could use the condenser as a heat
22 sink while you did some repair work on the heat ex-
23 changer.

24 So there are alternative cooling paths that can
25 be used. But when we're saying we're relying upon the RHR

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1 heat exchanger for a number of modes, it becomes a very
2 important piece of equipment.

3 And in my mind, when you damage it seriously,
4 then that's a big problem; and yes, we're very much con-
5 cerned about that.

6 JUDGE CHEATUM: Thank you.

7 BOARD EXAMINATION

8 BY JUDGE WOLFE:

9 Q Mr. Hodges, you say the NRC is concerned
10 and is looking into this problem.

11 A Yes.

12 Q Just exactly what is the NRC doing along these
13 lines to ameliorate or prevent such a problem?

14 A It's a combination of several different types
15 of things. It's working on the procedures that are used
16 to operate the equipment to try to minimize the water
17 hammer.

18 Q Such as?

19 A Such as opening valves closely, cautioning the
20 operators about the effects of water hammer and how to
21 prevent them.

22 Q Yes.

23 A It gets into trying to determine just what could
24 be the more serious causes of water hammer, where you need
25 to install additional restraints to prevent the pipe

1 motion so you get the damage -- things of that nature.

2 There may be other things. Those are the ones
3 I'm aware of.

4 Q As a rule, when you speak of routine main-
5 tenance, what is this? Every so many months? What's
6 the plan, say, for Allens Creek --

7 A Well, on an RHR heat exchanger, routine main-
8 tenance is only done when the plant is shut down for re-
9 fueling. That would be like an 18-month interval.

10 Q Every 18 months?

11 A Something of that nature for an RHR heat ex-
12 changer, yes.

13 Q Along those lines, would you recommend routine
14 maintenance to be done -- routines to be made monthly or
15 every two months, rather than at the end of 18 months?

16 A Well --

17 Q In this area of the water hammer?

18 A -- I think maybe you could change that slightly
19 and say maybe you do an inspection every time you'd use
20 the system.

21 You'd check the restraints and welds and things
22 every time you used the system. You wouldn't expect any
23 damage to occur to the system when it had not been used
24 for something.

25 So I would see no need for a monthly inspection,

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1 for example, when the plant was just operating normally
2 and the RHR system was in standby.

3 But I can see some need for changes in pro-
4 cedures -- maintenance procedures and inspections.

5 That's my personal opinion.

6 Q Uh-huh. Now, this plate displacement that you
7 spoke of at some plant, that was or was not visual to the
8 eye? Or was this discovered just upon routine main-
9 tenance?

10 A It was visual once you dismantled the heat ex-
11 changer. It's just some of the internals of the heat ex-
12 changer.

13 So if you just walked past the heat exchanger
14 and looked at it, you wouldn't see it.

15 JUDGE WOLFE: All right. Cross on Board
16 questions, Mr. Copeland?

17 MR. COPELAND: Yes, sir.

18 RECROSS-EXAMINATION

19 BY MR. COPELAND:

20 Q Just to follow up a little bit, Mr. Hodges, as
21 I understand your testimony, the whole subject of water
22 hammer is now a generic issue that is being investigated by
23 the Staff; is that correct?

24 A That's correct.

25 Q And is it -- Am I correct in understanding

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that as a result of that investigation, that you will indeed develop recommendations as to what steps should be applied to all reactors to help alleviate this problem?

A. That is correct.

Q. All right, sir.

And do you see any reason at this time to have to, for example, shut down all reactors that are in operation because of the potential for water hammer before the Staff reaches a resolution of those steps?

A. No. I think that at least in the interim that the steps that are being taken, in terms of trying to advise the operator as far as procedures, to try to improve his operation of the system to prevent them, is adequate.

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1 BY MR. COPELAND:

2 Q And would it also be true that if you lost the
3 RHR system, that you do have other methods available for
4 bringing the plant to cold shutdown?

5 A That is correct.

6 Q All right, sir. And to make sure that I under-
7 stand what you've said in your direct testimony, it's my
8 understanding that as of this date, there has been no RHR
9 system water hammer damage that has resulted in an unsafe
10 condition in your opinion on any plant?

11 That's what you say on Page 19 of your testi-
12 mony.

13 MR. DOHERTY: Well, then I object. It has al-
14 ready been asked and answered.

15 MR. COPELAND: Well, I want to know if that's
16 still your testimony.

17 MR. DOHERTY: I'll withdraw the objection.

18 THE WITNESS: Okay. The -- In each case
19 where it has been observed, the plant has been capable of
20 being brought to a shutdown and being cooled.

21 MR. COPELAND: All right, sir, thank you.
22 Those are all the questions I have.

23 JUDGE WOLFE: Mr. Doherty?

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RE CROSS-EXAMINATION

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BY MR. DOHERTY:

Q In answer to one -- I guess the second round of Judge Linenberger's questions, you stated that one plant had cooled itself down by using the spent fuel pool heat exchangers because of damage to the normal RHR.

A I know that it had already cooled down using the normal RHR. They just maintained shutdown cooling with the fuel pool heat exchangers.

Q Do you know if Allens Creek has these same heat exchangers?

A Yes.

Q Does it?

A It's a standard piece of equipment.

Q Is it always available for this?

A Any of these could be out for maintenance --

Q Say it again.

A It could be out for maintenance, any piece of equipment has some maintenance period.

Q Does it have any use whereby it would be pre-empted that you can think of?

A It's being used to cool the fuel pool, if you had a freshly discharged core, it may be operating at near capacity to keep the fuel pool cooled.

Q So it would be needed for that; is that your --

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29
1 A With a freshly discharged core.

2 Q All right.

3 A You know, a full core freshly discharged.

4 MR. DOHERTY: Okay, thank you. No further
5 questions.

6 JUDGE WOLFE: Redirect, Mr. Dewey?

7 MR. DEWEY: Yes, sir.

8 REDIRECT EXAMINATION

9 BY MR. DEWEY:

10 Q Mr. Hodges, when will the Staff study on the
11 water hammer be completed, do you think?

12 A I think it's scheduled for December of '82.

13 Q Uh-huh. Well, in your opinion, will this be
14 enough time for the Applicant to make any necessary
15 changes to comply with -- before the OL stage?

16 A Yes.

17 MR. DEWEY: Thank you.

18 JUDGE WOLFE: All right.

19 We'll recess until 25 of 11:00.

20 (A short recess was taken.)

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1 JUDGE WOLFE: All right, Mr. Dewey.

2 MR. DEWEY: Yes, sir. At this time we wish
3 to offer Mr. Hodges for cross-examination on Board Question
4 17.

5 Before we do this, we have one small change in
6 the testimony at page 20, and I would like to ask
7 Mr. Hodges about this change so he can read it in the
8 record.

9 DIRECT EXAMINATION

10 BY MR. DEWEY:

11 Q Mr. Hodges, do you have any changes with
12 respect to Board Question 17 in your testimony?

13 A Yes, I do. On page 20 of my written
14 testimony, the fourth line from the top, and at the end
15 of that line it refers to "non-safety grade equipment."

16 The "non" should be deleted so that the
17 sentence should now read, "In August 1979, Westinghouse
18 informed their customers that the performance of non-
19 safety grade equipment subjected to an adverse environment
20 could impact the protective functions performed by safety
21 grade equipment."

22 MR. COPELAND: How about the next sentence; is
23 that still correct?

24 THE WITNESS: The next sentence is still
25 correct.

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BY MR. DEWEY:

Q. Do you have any other changes?

A. No other changes.

Q. Thank you.

MR. DEWEY: At this time we offer Mr. Hodges for cross-examination.

JUDGE WOLFE: Mr. Copeland.

MR. COPELAND: No, sir.

JUDGE WOLFE: Mr. Doherty.

MR. DOHERTY: Yes, sir.

CROSS-EXAMINATION

BY MR. DOHERTY:

Q. While we're on page 20, and with the change that you made there, is this a kind of linkage problem between non-safety and safety grade equipment then; is that the way you interpret this issue?

This one change makes a fairly significant difference in how you interpret the problem.

A. Yes. The concern as expressed was that a problem with a piece of non-safety grade equipment could impact the operability of a piece of safety grade equipment.

Q. Now, you've indicated here at the foot of 20 that breaks outside of containment are not a problem for BWR and then you give some exceptions. One of them is the

3-3
1 scram discharge system. That's non-safety grade in part
2 or not?

3 A There are parts of that that -- that is correct,
4 but I think it's important to note that for the BWR-6
5 design, which is for Allens Creek, that that system is
6 located totally inside the containment, and so the concern
7 there that would arise for earlier design boilers does not
8 come up for Allens Creek.

9 Q Then you are saying that a break in some
10 instrument lines and a scram discharge volume is not
11 isolated automatically; is that right?

12 MR. COPELAND: It's what the testimony says in
13 the very first sentence, Your Honor.

14 BY MR. DOHERTY:

15 Q How long would it take, do you know, to
16 isolate a break in the scram discharge volume?

17 A There is a valve that would need to be reset.
18 It's an air-operated valve, but if that can be reset, it's
19 a very quick operation. It can be done in a half a minute
20 or a minute or something like that.

21 If that cannot be reset, then it's a
22 considerably more difficult operation. It would take, to
23 get total isolation, several hours.

24 Q What about discovery of that? I mean, how would
25 someone know they needed to make that simple quick action?

-4

1 A. In this case, basically what you would be
2 experiencing would be a small loss of coolant accident
3 inside the containment. So they are going to be seeing a
4 change in the containment atmosphere as far as temperature,
5 pressure; they are going to see a heat up of the suppression
6 pool.

7 Typical LOCA type conditions would indicate a
8 break in the pipe.

9 Q. Could you foresee a loss of water level for
10 such a break, too?

11 A. I think that there would be an initial loss of
12 water level until you had systems on to compensate for it,
13 yes.

14 Q. Do you have any idea -- well, okay.

15 Do you have any idea how many gallons, say,
16 per minute would be the maximum you could lose?

17 MR. COPELAND: Your Honor, I think we are going
18 down a rabbit trail now. It seems to me that Mr. Doherty
19 is straying off the point of the Board question.

20 The question the Board asked was for the Staff
21 to present evidence as to the acceptability of using
22 non-safety grade equipment for the mitigation of transients,
23 and I don't think that the line of questions he's now
24 pursuing is focusing in on that question.

25 MR. DOHERTY: I'll try to rephrase it. I think

1 I can.

2 BY MR. DOHERTY:

3 Q Do you think there would be an unacceptable
4 loss of water from a break like this? Could that happen?

5 A The water loss is not so large that you can't
6 make it up. So no, it's not an unacceptable loss.

7 Q Would the loss be severe enough that you would
8 expect one of the emergency systems to start automatically?

9 MR. COPELAND: Your Honor, I think that question
10 has been asked and answered. The witness said it would
11 be like a small break LOCA and all the normal equipment
12 that would function in a LOCA would function.

13 (Bench conference.)

14 MR. DOHERTY: Where does he say that, Counsel?

15 MR. COPELAND: He testified to that, Mr. Doherty,
16 in response to a question you asked him.

17 MR. DOHERTY: No, I don't think he did. I
18 think he testified only to the first part of what you
19 said, Counsel.

20 JUDGE WOLFE: I think that's right,
21 Mr. Doherty.

22 We'll overrule the objection. Go ahead.

23 THE WITNESS: It is like a small break LOCA
24 and if the, for example, depending on what the leakage
25 rate was. For a break in the scram discharge volume,

3-6
1 the leakage rate depends on how much leakage can go past
2 the seals on the rod drives, and that is a function of the
3 wear on the seals. It could be anywhere from a fairly
4 small leakage on the order of three to four hundred gpm,
5 which could easily be made up by the reactor core isolation
6 cooling system, up to about a thousand gpm where you would
7 need the high pressure core spray to provide makeup.

8 BY MR. DOHERTY:

9 Q What were some of the instrument line breaks
10 you had in mind in your testimony at the foot of 20?

11 A These were lines coming off the level sensors.

12 Q Those are the only instruments? That's the
13 only type instrument referred to in that phrase?

14 A Yes, basically. You are talking about any
15 tubing that can be carrying water here, your thermal
16 couplers on the recirculation loops and other instruments;
17 you are talking about wires coming out. So basically you
18 are talking about instrument tubing for level indicators.

19 They are small pipes rather than tubing.

20 Excuse me.

21 Q You have calculated there or there is presented
22 probability at the top of 21 of a break in the scram
23 discharge volume.

24 Did that calculation apply to BWR-6's?

25 A The way the calculation was performed and

3-7 1 because of the number of systems that were ignored or as
2 mitigating systems -- excuse me, this is just on the
3 probability of a break. I'm sorry.

4 Q Uh-huh.

5 A Basically, this was a typical design. I
6 believe the piping for Browns Ferry was used, so that's a
7 BWR-4, but the things that went into the input of that
8 would not be drastically different for Allens Creek.

9 Q You state that, "The operator has sufficient
10 time and information to depressurize..." in the event of
11 this type of break.

12 Where is he going to get this information
13 from? This is what I don't understand. He might know
14 something is wrong, but I don't -- or do you presume he
15 would have to know something like that?

16 A Well, first off, his first indication is if
17 he can reset his scram, which part of his normal scram
18 procedure is to reset his scram fairly soon after it
19 occurs.

20 If he cannot reset his scram, then he
21 recognizes that there may be a problem. He then, also,
22 would be getting indications of a higher temperature in
23 the suppression pool. He will be getting some heat up
24 in the containment because you are blowing down inside the
25 containment with a small break.

8-8 1 He will probably be getting radiation alarms.
2 He'll have a number of indications that he has a small
3 break in progress, and he will have an indication that it's
4 very likely in the scram discharge volume because of the
5 incapacity to reset.

6 He will also have temperatures in the scram
7 discharge volume itself, which will give him some
8 information.

9 Q Okay, now, resetting. On a small break LOCA
10 like this, if there were a scram typically? Is that expected?

11 A In the first place, you are already scrammed.

12 Q All right. You are already scrammed. We got
13 to that.

14 A For this break you have to scram in order to
15 get it.

16 Q Would it scram automatically -- I need to
17 clear this out.

18 It scrammed automatically following some
19 indications of some kind of break somewhere; isn't that
20 right?

21 A It may have scrammed for a number of reasons,
22 whether it's a break, whether it's a normal type of
23 transient, whether it's just normal shutting down of the
24 plant.

25 You would have had to have a scram before that

3-9 1 becomes a break.

2 Q But you said the first way he would discover it
3 was this kind of break would be to reset the scram, and I
4 don't understand what you mean.

5 His reactor is now shut down by the rods
6 inserted.

7 A The rods are inserted.

8 Q So what's he doing?

9 A He's realigning some valves. It's called
10 reset of the scram. He's just realigning a valve in the
11 scram discharge piping.

12 Q All right.

13 A To stop the flow of water to the scram discharge
14 volume. That's an air-operated valve.

15 If he can close that valve, then that would
16 terminate any water going out that break. If he cannot
17 close it and he sees a high temperature in the scram
18 discharge volume and he sees conditions in the containment
19 that are indicative of a break, he knows what his break
20 is and he knows what actions he has to take.

21 Q Okay.

22 JUDGE LINENBERGER: Excuse me, Mr. Doherty.

23 I'm missing possibly something here. When you
24 speak of resetting the scram, I was thinking of that in
25 terms of re-engaging the drive mechanism or relatching the

3-10 1 drive mechanisms to the control blades.

2 Is that or is that not part of the scram reset
3 operation?

4 THE WITNESS: The control blades should still
5 be latched to the drive mechanism. They are in the core,
6 and the primary thing that is done on the resetting is the
7 closing of these valves so that you can drain the scram
8 discharge volume, be capable of accepting another scram if
9 need be.

10 The rods are already inserted and this has
11 nothing to do with the motion of the rods.

12 JUDGE LINENBERGER: Okay. Go ahead,
13 Mr. Doherty.

14 BY MR. DOHERTY:

15 Q Well, this resetting process itself, is there a
16 separate process for each control rod or are there group
17 rod closures or....

18 A It can be done as a group and there's, also,
19 I think, a separate valve that can be used on each rod.
20 But normally, it's done as one operation.

21 Q For all the --

22 A For all the rods.

23 Q But there's also available some type of
24 several or many or perhaps 200 -- I think there's something
25 on the order of 200 control rods.

8-11

1 A A hundred and eighty-five, something like that.

2 Q All right. Is there a separate reading of the
3 temperature for each control rod's outlet?

4 A No. We're talking about the temperature in the
5 scram discharge volume, which is a common volume to all of
6 them.

7 Q Not in the volume. Further up in the piping
8 to the head or --

9 A I don't think there's a temperature reading in
10 those. I'm not aware of one.

11 Q But the break of one would provide a
12 sufficiently abnormal temperature to say there's a break in one
13 of the -- in the system; is that right?

14 A The worst case is where a break occurs right
15 at the scram discharge volume piping where one of these
16 other pipes comes in, and that gives you the worst
17 conditions, and in that case, you also get a fairly large
18 flow.

19 If you can isolate all but one, it's a very
20 small flow and it's an insignificant problem. You are
21 talking about three to five gallons per minute through
22 each rod.

23 So if you isolate all but one, that's an
24 insignificant amount.

25 Q Now, at the foot of 21, you state the operator

8-12 1 is cautioned to look for an effect in emergency procedures
2 if there is an instrument line break.

3 That appears to take about a half an hour.
4 How is this to be done? Are you telling him to check in
5 30 minutes or what?

6 MR. COPELAND: I'm going to object to that
7 question, Your Honor. I think Mr. Doherty misread the
8 testimony, to begin with.

9 He said that -- He related this statement to
10 an instrument line break and it's not clear to me that
11 this testimony has anything to do at this point on page
12 21 with an instrument line break.

13 Maybe it does. It's just not clear.

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1 BY MR. DOHERTY:

2 Q Does the testimony at the foot of 21 refer to
3 instrument line breaks?

4 A Not specifically.

5 Q Well, then, what do the instrument lines refer
6 to in the last sentence there?

7 A These are the instrument lines for the level
8 sensors themselves. I'm not assuming a break -- For
9 that paragraph, I'm not assuming a break in those lines.

10 I'm assuming an effect of the ambient
11 conditions causing the change in the density of the water
12 in those lines.

13 Q These are conditions in the containment
14 building itself?

15 A That's right.

16 Q Well, is the statement at the top, sort of, of
17 21, about five lines down, just the last part of it, "The
18 operator has sufficient time and information to depressurize
19 and thus reduce the effect of the break," is that taken
20 from NUREG-0803?

21 A I did not copy the words from NUREG-0803, but
22 I wrote some of the words in NUREG-0803, and so they may
23 sound very similar.

24 Q Do you have a copy of NUREG-0803 with you?

25 A I did not bring a copy, no.

1 Q I see.

2 MR. DOHERTY: May I approach the witness,
3 Your Honor?

4 JUDGE WOLFE: Yes.

5 BY MR. DOHERTY:

6 Q Mr. Hodges, did I show you a copy of NUREG-
7 0803?

8 A Yes, you did.

9 Q Now, on page 4-3 there's a statement that I'd
10 like to read to you with regard to a section called
11 "Diagnostics."

12 It says, "The sources of SDV, piping break
13 detection signals for Mark-1 and Mark-2 containments
14 are," and they are so listed on page 4-3.

15 You can see that that listing stops at the
16 foot of the page. Do we have the same break detection
17 signals for a Mark-3 containment, to your knowledge?

18 A They would not be the same because the scram
19 discharge volume is located inside the containment for
20 the Mark-3; and if you'll notice, a number of these are
21 reactor building indicators.

22 So it's a different list.

23 Q In your opinion, are the sources for these
24 piping break detections equal to the Mark-1 and Mark-2?
25 Is the Mark-3 equal in ability in this regard?

3-15 1 A Yes. Here you have personnel observation of
2 leakage; that one may not be quite on the same level, but
3 the rest of them are essentially the same level.

4 Q Okay.

5 MR. DOHERTY: No further questions, Your Honor.
6 Thank you.

7 JUDGE WOLFE: Redirect, Mr. Dewey?

8 MR. DEWEY: No, sir.

9 JUDGE WOLFE: Board questions?

10 JUDGE CHEATUM: I have none.

11 BOARD EXAMINATION

12 BY JUDGE LINENBERGER:

13 Q Coming back to page 21, at the bottom of page
14 21, Mr. Hodges, is the implication of that final paragraph,
15 or should I infer from that final paragraph, let's say,
16 that there's a possibility that the operator might be
17 misled as to where water level is in the reactor pressure
18 vessel because of this temperature density kind of effect
19 you are talking about?

20 A The water level indicators are calibrated with,
21 I think, it's assuming a containment temperature of 135
22 degrees.

23 As you go above that and remain above that for
24 any period of time, then you start decreasing the density of
25 the water in the reference legs for the indicators and that

B-16

1 difference in density will result in an error in the
2 indicated water level.

3 The time that I have listed here is if you had
4 like a step increase up to some number above 135 degrees,
5 how long it would take for that instrument to respond.
6 So that would be the shortest time that you would see a
7 significant change in the accuracy of the level indicators
8 themselves.

9 The operators are made aware of this effect.
10 It's listed explicitly in the Emergency Procedure
11 Guidelines and in the emergency procedures that are
12 developed from those guidelines, and in fact there is a
13 listing of what the error is as a function of temperature,
14 and it cautions the operator to be aware of this.

15 Q Okay. Well, further refinements on this, I
16 think, can come later when we get into testimony on the
17 water level indicators, but I just wanted to make sure I
18 understood the significance of this last paragraph.

19 A Let me retract one thing I said. I said it
20 gave errors as a function of temperature.

21 There's another table that shows where it can
22 be wrong, but it's not in this one; but it does caution
23 the operator to be aware of the problem and alerted that
24 he can be given a misreading signal.

25 Q Okay. Perhaps I have a misconception, but at

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3-17 1 the top of page 20 in the first answer there, I would ask
2 you the same question Mr. Copeland did.

3 Items 1, 2, 3 and 4 in the context of the
4 testimony, I presume are indeed non-safety grade systems?

5 A That is correct. Those are non-safety grade
6 systems, and this particular paragraph is taken directly
7 from the little information letter that Westinghouse sent
8 out to its customers.

9 Q All right, sir.

10 Let me just probe one or two little things here.

11 The term "safety grade system," it seems to
12 me, is rather a broad designation that could stand some
13 refinement with respect to the kinds of conditions you are
14 expecting a system of components to survive or operate
15 during, such as seismic events on the one hand, such as
16 perhaps high temperature or high pressure environments or
17 other components.

18 Now, when the word "safety grade" or "non-
19 safety grade" is used in the context of your testimony
20 here, in the first place let me ask you does it or does it
21 not include the designation Seismic Category I, for example?

22 A That is one of the things for safety grade, yes.
23 The non-safety grade equipment would not satisfy the
24 Seismic Category I requirement necessarily. They may, but
25 they have not been demonstrated to.

B-18

1 Q Okay. Now --

2 A The reason I add that last sentence is quite
3 often it's the same instrument; just one has a piece of
4 paper certifying it will satisfy the requirements, the
5 other one doesn't.

6 Q There have been considerations involving
7 the question of whether certain pieces of equipment have
8 been properly environmentally qualified.

9 I don't know that any of those concerns are
10 related to BWR's or particularly to Allens Creek, but I
11 do know that generally environmental qualification of
12 some components is important, apparently.

13 Now, again, should I conclude from what you've
14 said at the top of page 20 that these four subsystems have
15 not had any kind of -- or been subjected to any requirement
16 for environmental qualification?

17 A At the time that the notice was issued by
18 Westinghouse, they environment that they had been designed
19 for did not include, for example, the effect of a break in
20 one of these systems spewing steam or hot water on it, so
21 that that was not a consideration in that design.

22 It is now a consideration in the design, and
23 the equipment located in the reactor buildings are having
24 to be designed for -- the boiling water reactors. I'm not
25 that familiar with what's being done on the PWR.

8-19 1 On the boiling water reactors, we're talking
2 about qualifying equipment in the reactor building to 212
3 degrees, 100 percent humidity for up to an hour's period of
4 time. So this is fairly severe environment.

5 Q I'm not sure I heard the answer to my question.
6 This may have to do with me, not you, but again, I need to
7 ask, are these Items 1 through 4 at the top of page 20
8 items that do satisfy or are required to satisfy
9 environmental qualification criteria?

10 A I'm not sure they all are now even, because
11 they are, quote, non-safety equipment, unquote. The
12 safety equipment is required to.

13 Q Let me ask you specifically with respect to
14 automatic rod control system, what's the basis for not
15 requiring that to be a safety grade system?

16 Why is it not important that it be safety
17 grade?

18 A I can speculate because it's not in my area
19 of review so I don't get into it directly. I can tell you
20 what I think, if you want that, but it's only a speculation.

21 Q Well, all right. Give me your opinion, if
22 you would, please, sir.

23 A The same rods are used in the safety mode.
24 The scram system is separate from the control system, and
25 so they can be scrambled independently from a malfunction in

3-20 1 the rod control system.

2 Q A small point of terminology here. I've seen
3 the acronym P-O-R-V frequently used. I could derive
4 P-O-R-V from your Item 2, power operated relief valve.

5 I've also heard P-O-R-V referred to as pilot
6 operated relief valve. Now, is there a distinction here?
7 Are the terms used interchangeably?

8 A I don't use the two interchangeably. I'm not
9 aware of their being used interchangeably.

10 Q What does P-O-R-V mean to you?

11 A That's power operated or power actuated.

12 Q Okay. Thank you.

13 JUDGE LINENBERGER: Thank you, sir. That's
14 all I have.

15 JUDGE WOLFE: Is there cross on Board questions,
16 Mr. Copeland?

17 MR. COPELAND: I have just one question,
18 Your Honor.

19 RE-CROSS-EXAMINATION

20 BY MR. COPELAND:

21 Q I'm a little confused, Mr. Hodges, as a result
22 of the exchange with Judge Linenberger.

23 On the top of page 20 where you list the
24 non-safety grade systems, it's my understanding that those
25 are all systems within a Westinghouse PWR; is that right?

1 A That is correct.

2 Q And do those same exact systems, are they
3 also in a BWR, and are they non-safety grade systems within
4 a BWR?

5 A There is a main feedwater control system in a
6 BWR which is a non-safety grade. The rod control system is
7 quite a bit different. Obviously, things with a steam
8 generator and a pressurizer would not be in a BWR.

9 Q Okay.

10 MR. COPELAND: Thank you.

11 JUDGE WOLFE: Mr. Doherty.

12 MR. DOHERTY: No questions, Your Honor.

13 JUDGE WOLFE: Redirect, Mr. Dewey?

14 MR. DEWEY: No, sir.

15 JUDGE WOLFE: All right. We'll now proceed,
16 Mr. Dewey, to what, or Mr. Schinki?

17 MR. SOHINKI: I think next we are scheduled to
18 proceed to Doherty Contention 42, which is position
19 indication for SRV's.

20 As the Board will recall, that was a separate
21 piece of testimony which was also incorporated into the
22 record at the same time as the previous piece of testimony
23 we've been discussing.

24 I have no additional direct examination, so
25 Mr. Hodges is available for cross-examination.

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JUDGE WOLFE: You say this was incorporated into the record as if read?

MR. SOHINKI: Yes. I believe it was at the same time that Mr. Hodges' testimony with regard to the other --

MR. COPELAND: Yes. It's following Transcript Page 15128, Your Honor.

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1 JUDGE WOLFE: Is there cross, Mr. Copeland?

2 MR. COPELAND: I just have one clarifying
3 question, Your Honor.

4 CROSS-EXAMINATION

5 BY MR. COPELAND:

6 Q On the top of Page 3 of your testimony, Mr.
7 Hodges, the first answer -- the first sentence in the first
8 answer, you say the pipe pressure should be near the con-
9 tainment pressure.

10 Which side of the valve are you speaking of
11 there?

12 A I'm talking about downstream of the safety/
13 relief valve.

14 Q Okay. That's what I thought. Thank you.

15 JUDGE WOLFE: Mr. Doherty?

16 MR. DOHERTY: Yes, Your Honor.

17 CROSS-EXAMINATION

18 BY MR. DOHERTY:

19 Q You state on Page 2, "An alarm indicating the
20 a safety/relief valve is open will be provided in the con-
21 trol room."

22 Would the operator know if there were more
23 than one opened?

24 A There would be an alarm for each.

25 Q For each.

4-2
1 JUDGE WOLFE: Excuse me, Mr. Doherty, Mr.
2 Sohinki, do you have a spare copy or copies of Mr. Hodges'
3 testimony on Doherty Contention 42?

4 MR. SOHINKI: I think I have at least one extra
5 copy.

6 JUDGE WOLFE: All right. That would be help-
7 ful. One copy would be fine.

8 (Document handed to Judge Wolfe.)

9 JUDGE WOLFE: Thank you.

10 BY MR. DOHERTY:

11 Q Do you know of any performance records for
12 this pressure sensor valve indicator?

13 A Are you talking about the particular type of
14 installation that's proposed?

15 Q Yes.

16 A It has been installed, I think, on one plant --
17 one operating plant. I don't know that it has been called
18 upon to operate yet, except maybe in testing.

19 Q Is there more than one path for the sensor
20 signal to travel to the control room, or is there a
21 single --

22 A It's redundant.

23 Q Okay. What's the power source for it? Do you
24 know?

25 MR. COPELAND: For the signal or the valve, Mr.

1 Doherty?

2 MR. DOHERTY: For the signal.

3 THE WITNESS: No, I do not.

4 BY MR. DOHERTY:

5 Q Do you know how high the pressure must go
6 above the containment pressure to activate the sensor
7 at this time?

8 A I don't know what the set point will be, but
9 when the valves open for discharge, it will go up sub-
10 stantially above the containment pressure. So there's
11 a very strong signal.

12 Q All right. So that's in -- Well, has
13 anyone discussed, or are you aware of any figure, other
14 than that or any ball park estimate other than that?

15 A I know that when the valves open, the pressure
16 in the discharge pipe will go up above 250 pounds, for
17 example, where normally it's in the range of, you know,
18 15 pounds or less.

19 So if it's a large signal, it's easy to detect.
20 I have not discussed the actual set point of the signal
21 with anyone.

22 Q Is there any concern of a valve being opened
23 and the pressure dropping sufficiently to sort of cease
24 the alarm function and just -- it becoming unnoticeable
25 that, indeed, the valve was still open, say, manual

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1 detectors.

2 Q In your opinion, is this a superior system to
3 acoustic?

4 A Yes, it is.

5 Q Do you have any idea how long -- Well, first
6 of all, would there be any lag of the open alarm following
7 closing of an SRV?

8 A The pressure should drop very rapidly, whether
9 it's milliseconds or ... you know, half a second, or some-
10 thing like that would maybe depend upon the set point.
11 But it would drop very rapidly.

12 Q Are there just two valve position indications:
13 open and closed?

14 A That's basically what the system is telling
15 you is opened and closed.

16 Q Uh-huh. But are there any other signals to the
17 control room, to your knowledge?

18 A Are you asking if there's something similar to
19 what they had at Three Mile Island where a signal was
20 sent -- I don't know what you're asking.

21 Q Well, I'm trying to find out what's available
22 in the way of information. It sounds like there will be
23 only two information sources on each valve: open or
24 closed.

25 MR. COPELAND: Well, Your Honor, I'm going to

4-4
1 operation of the valves for some reason?

2 A The valves are normally operated in the full
3 open or full closed mode. They're not a throttling valve.
4 There is a potential for a valve to leak, and the system
5 not detect it.

6 But if the valve is open, as opposed to leaking,
7 the system would detect it.

8 Q But it would detect it only by the fact that
9 there was pressure in the pipe, right?

10 A That's right.

11 Q Do you perceive of any circumstance where the
12 pressure might drop, and it would still be critical to
13 know that that valve is open; yet, because the pressure
14 drops sufficiently, the valve indicator ceased -- just
15 went off ... no longer alarmed?

16 A Unless you postulate an unlikely event like
17 the pipe breaking so that you don't have that as a boundary,
18 and with the valve open I see no reason for the pressure
19 to drop down.

20 Q You said the pipe. Did you mean --

21 A The discharge pipe.

22 Q Okay. In arriving at this solution to the
23 problem, do you know any of the other suggestions that were
24 made?

25 A I know a number of plants were using acoustic-type

1 object to any further questions along those lines. The
2 witness has testified this isn't a throttle-type valve,
3 that it is either opened or closed. That is all the pres-
4 sure sensor is supposed to tell.

5 So I don't understand where Mr. Doherty is
6 going with this.

7 MR. DOHERTY: I'll withdraw the question.

8 BY MR. DOHERTY:

9 Q When you refer to a throttle-type valve, does
10 that mean there are just opened and closed positions and
11 nothing else?

12 A A throttle-type valve would be similar to a
13 valve on a faucet; there are intermediate positions by
14 which you can control the flow rate.

15 I'm saying this is not a throttle-type valve.
16 It functions at either full closed or full opened.

17 Q I notice at the top of three you state, "When
18 the safety/relief valve is closed, the pipe pressure
19 should be near the containment pressure."

20 Is that just a caution there? The word,
21 "should," is a little bit indefinite sounding, but maybe
22 it's just a caution.

23 A It's a little bit of caution. If you've got some
24 air in there and the containment heats up slightly, it's a
25 closed volume, it can heat up and you can also have a

1 leg of water; and it's going to be slightly above the con-
2 tainment pressure, but not tremendously above -- a few
3 pounds above.

4 And so that's just caution.

5 Q That could be easily controlled by the setting;
6 is that right -- in your opinion?

7 A It's still well below the pressure that you would
8 get from opening the valve, so the set points that are
9 used for your signal would compensate for that, yes.

10 Q Okay. In your opinion, would it make any dif-
11 ference, the actual placement of the sensor in the --
12 downstream of the valve? Would it have to be close to the
13 valve, or would it be quite far down or --

14 A You would want to get it reasonably close to the
15 valve to get the highest pressure signal. But it doesn't
16 have to be adjacent to the valve, but reasonably close.

17 Q In the last paragraph at Page 3, how long would
18 it take for the reactor to appear to be off normal, if
19 there were a five percent flow in one -- through one of
20 these discharge pipes and not detected as a pressure
21 increase?

22 A I think we have a requirement that says they
23 have to be able to detect something like that within one
24 hour.

25 Q And they're supposed to detect that through some

4-8

1 alternate means like loss of inventory or something of
2 that --

3 A Or temperature in the suppression pool, water
4 level in the suppression pool.

5 Q Now, you've given a figure in percents, and
6 you've also given one in terms of gallons per minute.

7 Would it be fair to say that this wouldn't
8 detect a five-gallon-per-minute flow?

9 A No, that would not detect a five-gallon-per-
10 minute flow.

11 Q I see. How many gallons approximately is a
12 five percent flow per minute? Do you have any idea?

13 A You're talking about a steam flow, and if you
14 want to make it in terms of an equivalent make-up flow in
15 gallons per minute, you can do that; and that comes out to
16 be -- for one of these valves ... let's see ... I believe
17 it's around 1100 gallons per minute make-up required to
18 satisfy -- This is not the five percent. This is full
19 flow through the valve.

20 One hundred percent steam flow through the
21 valve requires roughly 1100 gallons per minute make-up to
22 maintain water level at 1000 pounds pressure.

23 So if you want to take -- for the five percent
24 number, if you take five percent of roughly 1100 --
25 that's what? 55 gallons or something like that, if I'm

4-9 1 doing the arithmetic right in my head.

2 Q Yes. I think it's less than 50, just doing it
3 in mine.

4 Is the pressure sensor similar in principle
5 to the one used to initiate scram?

6 A They're similar. Whether it's the same manu-
7 facturer or such, I don't really know.

8 Q But the principle --

9 A The principle is -- You're looking for an
10 increase in pressure, and, yes, it's the same.

11 Q You indicate it was used in different types of
12 industrial facilities, at the foot of Page 4. What I'm
13 wondering is were the pressures as severe as this sensor
14 would encounter in some of those applications?

15 A These sensors have been used over a very wide
16 range, anywhere from low pressure to several thousand
17 pounds pressure. So it's a pretty -- These types of
18 indicators are used over multiple conditions, over a very
19 wide range.

20 Q All right.

21 MR. DOHERTY: No further questions, sir, thank
22 you.

23 JUDGE WOLFE: Redirect?

24 MR. DEWEY: No, sir.

25 JUDGE WOLFE: Board questions?

1 JUDGE CHEATUM: Yes, I have one question.

2 BOARD EXAMINATION

3 BY JUDGE CHEATUM:

4 Q In your response to Mr. Doherty's question
5 regarding other types of sensors for determining valve
6 positions, you said that there was an acoustic sensor.

7 A That's correct.

8 Q Now, did I understand you to say that in your
9 opinion, or is in the Staff's general view, that the
10 acoustic-type sensor is not as desirable as the pressure
11 sensor?

12 A From the Staff's general view, the acoustic
13 sensor has been accepted as being a means of doing
14 this. For my personal opinion, I don't consider it to be
15 as reliable as the pressure sensor.

16 Q That's your personal opinion?

17 A That's my personal opinion.

18 Q All right. Just how is the acoustic sensor --
19 What is its mechanism? How does it operate?

20 A It's basically when you have a flow -- the steam
21 flow going through the pipe, it makes a noise; and so you're
22 trying to pick up -- like with a microphone, the noise.

23 The problem is you can have discharge through
24 an adjacent pipe. And if the sensitivity is set too
25 large, you pick it up from that. Or if you set it large

4-11 1 enough, you don't pick up that sensitivity, you don't even
2 detect the fact that you've got flow through -- you know,
3 the pipe.

4 And so it's the problems with the sensitivity
5 of the setting that makes me think that the acoustic is
6 not as good.

7 Obviously, it will pick up the noise and it
8 will work. But you have to be very careful of how the
9 sensitivity is set for the sensors.

10 So my personal preference would not be the
11 acoustic ones.

12 - - -

1 BY JUDGE CHEATUM:

4-12
2 Q All right. Could I infer from what you've said
3 that low flow, say, five percent of the wide open volume,
4 if you had a leak, say, using five percent of your steam
5 flow through the valve, the acoustic sensor would be no
6 more likely, or perhaps even less likely, to pick up that
7 leak than the pressure?

8 A I think that's true, yes. I think it would
9 probably be less likely.

10 JUDGE CHEATUM: That's all I have.

11 BOARD EXAMINATION

12 BY JUDGE LINENBERGER:

13 Q Mr. Hodges, no real substantive problem with
14 your testimony, but I'd like to just knitpick one
15 thing.

16 In a couple of places it seems to me that you
17 said -- well, specifically in the last paragraph on Page
18 4 -- that this system will provide the direct indication
19 of flow through the valves.

20 It seems to me that it is an inferred or in-
21 direct indication of flow through the valves, because I
22 can conceive of blocking the downstream end of the pipe,
23 letting the pressure build up, and this sensor is going to
24 give a pressure reading, and there will be no flow.

25 So is that --

4-13

1 A It's -- Since -- Actually in the strictest
2 sense, it's very difficult to get a direct measure of
3 flow, in most cases -- in an orifice meter, for example,
4 which you often think of as a direct measure of flow,
5 but, as you point out, it's not that direct. It's de-
6 rived from a differential pressure across the orifice.

7 But -- So I suppose in the absolutest
8 sense, you're right: it's not a direct.

9 Q Okay. I just wanted to make sure that I was
10 not missing something.

11 A No.

12 Q -- about the way it functioned. I'm not trying
13 to correct your testimony.

14 JUDGE LINENBERGER: Thank you, sir, that's
15 all.

16 JUDGE WOLFE: Cross on Board questions? Mr.
17 Copeland?

18 MR. COPELAND: No, sir.

19 JUDGE WOLFE: Mr. Doherty?

20 MR. DOHERTY: No, sir.

21 JUDGE WOLFE: All right. What next, Mr.
22 Sohinki?

23 MR. SOHINKI: Well, I think we are scheduled
24 next to take up the cold shutdown question. However, I
25 understand from the Applicant that their witness has some

4-14

1 travel constraints; and I'm sure Mr. Hodges would just as
2 soon have a little break. So I would propose that the
3 Applicant's witness on that question be taken next.

4 MR. COPELAND: That's fine with me, Your Honor.
5 There's only one problem. I've got to go get Mr. Culp.

6 Can we take a 10-minute break?

7 JUDGE WOLFE: We'll have a five-minute recess.

8 (A short recess was taken.) (Witness excused.)

9 JUDGE WOLFE: All right.

10 Do you have the Staff's report in support of
11 the Applicant's motion for reconsideration?

12 MR. SOHINKI: I have not. I had understood
13 from Mr. Black that he was going to transmit that
14 response to Mr. Culp's firm, who was going to have it
15 expressed down here.

16 I have not as yet received it. I don't know
17 whether Mr. Culp has.

18 MR. CULP: It's my understanding that it's at
19 Mr. Copeland's office, and that we will have it here after
20 lunch.

21 JUDGE WOLFE: I see.

22 Well, as I understand it, the Doherty
23 Contention No. 38(b) was the subject of the Board's
24 second order ruling on summary disposition. And it was
25 also the subject of Applicant's motion for reconsideration.

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4-15 1 Obviously, we have not ruled out the motion
2 for reconsideration at all. Do you wish to go forward,
3 regardless?

4 MR. CULP: I think so.

5 JUDGE WOLFE: All right.

6 MR. CULP: The Applicant calls Mike K. Mitchell
7 to the stand and ask that he be sworn.

8 JUDGE WOLFE: Would you stand, please, and
9 raise your right hand.

10 Whereupon,

11 MIKE K. MITCHELL

12 was called as a witness by the Applicant and, having been
13 first duly sworn, was examined and testified as follows:

14 JUDGE WOLFE: Please be seated.

15 DIRECT EXAMINATION

16 BY MR. CULP:

17 Q Mr. Mitchell, do you have a document before
18 you entitled "Direct Testimony of Mike K. Mitchell Regarding
19 Doherty Contention No. 38(b) - Cold Shutdown Within 24
20 Hours"?

21 A Yes, I do.

22 Q Did you prepare this testimony, or was it pre-
23 pared under your supervision?

24 A Portions of it were prepared under my super-
25 vision. Portions of it were prepared directly by me.

4-16
1 Q Do you have any corrections or additions to make
2 to the testimony?

3 A No.

4 Q Is the testimony true and correct to the best
5 of your knowledge and belief?

6 A Yes.

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

9 A Yes, I do.

10 MR. CULP: Mr. Chairman, at this time I move
11 that the testimony of Mr. Mitchell regarding Doherty
12 Contention 38(b), which was just identified, be incorporated
13 into the record as if read.

14 JUDGE WOLFE: Any objection?

15 MR. SOHINKI: No, sir.

16 MR. DOHERTY: Your Honor, I would like to take
17 the witness on voir dire.

18 JUDGE WOLFE: All right.

19 VOIR DIRE

20 BY MR. DOHERTY:

21 Q Mr. Mitchell, did you submit an affidavit on
22 this issue in 1980?

23 A No, sir.

24 Q To your knowledge, did you discuss your testi-
25 mony with anyone who did submit an affidavit in 1980 with

4-17 1 regard to this issue?

2 A Yes, I discussed it with -- an affidavit that
3 was presented by Mr. Joe Fray of the General Electric
4 Company.

5 Q Okay. Now, you are currently Senior Lead
6 System Engineer; is that right -- properly, your title?

7 A That's correct.

8 Q And do you supervise other engineers, sir?

9 A Yes.

10 Q About how many?

11 A It varies from between two and five individuals.

12 Q Have you ever authored any articles on heat
13 transfer?

14 A No, sir.

15 Q Fluid flow?

16 A No, sir.

17 Q Computer methods?

18 A My Master's thesis involved computer methods,
19 but it was not heat transfer.

20 Q Okay.

21 JUDGE CHEATUM: Mr. Mitchell, will you either
22 speak more closely to the microphone or louder, please?

23 THE WITNESS: Sure.

24 BY MR. DOHERTY:

25 Q Have you ever testified before an Atomic Safety

1 and Licensing Board before?

2 A. No, sir.

3 Q. I notice your undergraduate degree is in
4 civil engineering; and your MS is in engineering. Did
5 you specialize in any field of engineering in getting your
6 Master's of Science degree?

7 A. Yes, sir. My specialty was structural dynamics.

8 Q. Did that involve the study of thermodynamics?

9 A. No, sir.

10 Q. Did it involve the study of heat transfer?

11 A. No.

12 Q. Did your course in civil engineering involve
13 heat transfer?

14 A. Yes, sir.

15 Q. Was that like a year course -- basic course,
16 or what did you have?

17 A. The basic course, one semester.

18 Q. Did you study computer methods in either of
19 these -- Well, I think you answered that.

20 Okay -- Well, perhaps you didn't. Did you
21 study a course in computer methods at all in your graduate
22 work?

23 A. Yes, sir.

24 Q. How long was this GE Advanced Engineering
25 Program? What is that like, in size and number of hours?

1 A. That's a two-year program involving approxi-
2 mately 20 hours a week, covering interdisciplinary sub-
3 jects, electrical engineering, mechanical engineering,
4 including heat transfer and thermodynamics, some
5 structural analysis.

6 It's not a specialty program. It's a broad
7 training program that GE has.

8 Q. Is it aimed primarily at engineers involved
9 with nuclear plants?

10 A. No, sir. It's aimed primarily at engineers
11 for the General Electric Company, which includes turbines
12 and other equipment other than nuclear reactors.

13 Q. Uh-huh, okay.

14 MR. DOHERTY: All right. No further questions,
15 Your Honor.

16 No objections.

17 JUDGE WOLFE: If there are no objections, the
18 direct testimony of Mike Mitchell regarding Doherty Con-
19 tention 38(b), including an attachment of his background --
20 professional qualifications, are incorporated into the
21 record as if read.

22 (See attached pages.)

23 /

24 /

25 /

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	§	
	§	
HOUSTON LIGHTING & POWER COMPANY	§	Docket No. 50-466
	§	
(Allens Creek Nuclear Generating Station, Unit 1)	§	
	§	

DIRECT TESTIMONY OF MIKE K. MITCHELL
REGARDING DOHERTY CONTENTION NO. 38(b) -
COLD SHUTDOWN WITHIN 24 HOURS

Q. Would you please state your name and your position, and describe your educational and professional background?

A. My name is Mike K. Mitchell. I am employed at General Electric Company (GE) as Senior Engineer, Plant Design and Analysis. A statement of my educational and employment history is attached as Attachment MKM-1.

Q. What is the purpose of your testimony?

A. The purpose of my testimony is to address Doherty Contention 38(b), which alleges that:

"Contrary to NUREG-0578, the ACNGS reactor cannot be brought to cold shutdown in 24 hours."

Q. To your knowledge, is there any NRC requirement that specifies that Allens Creek must be designed to be capable of being brought to cold shutdown in 24 hours?

A. No. Following the TMI accident, there was a tentative proposal in NUREG-0578 for such a requirement.

1 However, no such requirement was imposed upon the near-term
2 CP plants in NUREG-0718.

3 Q. What is "cold shutdown"?

4 A. The phrase "cold shutdown" is defined in the BWR-6
5 standard technical specifications to mean that the reactor
6 temperature is below 200°F at atmospheric pressure and the
7 reactor mode switch is in the shutdown position.

8 Q. How is a reactor such as Allens Creek normally
9 brought to cold shutdown?

10 A. Normally, the initial phase of nuclear system
11 cooldown for ACNGS is accomplished by dumping steam from the
12 reactor vessel to the main condenser. When nuclear system
13 pressure has decreased to a point where steam supply pressure
14 is not sufficient to maintain the turbine shaft seals,
15 vacuum in the main condenser cannot be maintained and Shutdown
16 Cooling Mode of the Residual Heat Removal (RHR) System is
17 started to complete the task of placing the reactor in cold
18 shutdown.

19 The RHR System has several modes of operation, but
20 the mode of concern to achieve cold shutdown is the Shutdown
21 Cooling mode. In this mode, reactor coolant is pumped from
22 the recirculation loops by one of the RHR pumps and is
23 discharged to one of the RHR heat exchanger loops where
24 cooling occurs by transferring heat to the essential service
cooling water. The RHR heat exchangers are sized for

1 operation in the RHR mode of Suppression Pool Cooling
2 following a Loss of Coolant Accident (LOCA). Because the
3 heat load is much greater for this mode than for Shutdown
4 Cooling, the RHR System is considerably oversized for
5 achieving a normal cold shutdown condition.

6 Q. How long will it take to bring Allens Creek to
7 cold shutdown?

8 A. To determine the effectiveness of the ACNGS design
9 to achieve cold shutdown, decay heat load must be determined.
10 The maximum decay heat load after reactor shutdown calculated
11 for ACNGS is derived from the 1971 American Nuclear Society
12 formula as required by 10 CFR 50, Appendix K.^{1/} Using this
13 decay heat load, General Electric has determined that the
14 main condenser will cool the system to a temperature of
15 approximately 344° F at 110 psig in two hours. The system
16 is maintained at this temperature and pressure for an
17 additional two hours while the RHR System is flushed with
18 reactor grade water. At this point, one loop of the RHR
19 System is placed in service. At this time the heat load
20 is approximately 284.6×10^6 BTU/hr and decreasing. With the
21 temperature difference between reactor coolant and service

22 ^{1/} As an extra measure of conservatism, Appendix K requires
23 that an additional 20% heat load be added to the decay heat
24 load determined by the ANS formula.

1 water that exists at this time,^{2/} one RHR heat exchanger
2 loop is capable of removing approximately twice the amount
3 of heat being generated. During the initial phases of
4 shutdown cooling (to avoid cooling the Reactor Pressure
5 Vessel (RPV) down too rapidly), the heat exchanger discharge
6 flow is usually throttled such that the cooldown rate does
7 not exceed 100° F/hr. Subsequent to this initial gross
8 overcapacity period, the second heat exchanger loop can be
9 brought on line if needed to continue the cooldown process.
10 Based on analysis which has been correlated with heat
11 exchanger systems used on operating BWRs, the normal shutdown
12 cooling mode of the RHR System is fully capable of achieving
13 a reactor coolant temperature of less than 200° F in less
14 than seven hours with two hours conservatively allowed for
15 flushing of the RHR System.

16 If a single failure occurs during this normal
17 shutdown sequence of events, the alternate shutdown flow
18 path may be initiated such that suppression pool water is
19 injected directly into the RPV. The time to reach 200° F
20 using this alternate mode is significantly less than the
21 normal mode because the water being returned to the RPV in
22 the alternate mode is the cooler pool water (~ 150° F) rather
23 than the warmer heat exchanger discharge water (~ 300° F).

24 ^{2/} Essential service cooling water is assumed to be 95° F,
thereby making the difference in reactor coolant and service
water temperatures equal to 249° F.

1 Consequently the normal mode is the more limiting mode when
2 considering time to reach cold shutdown. Thus, even assuming
3 any single failure, the ACNGS reactor can achieve cold
4 shutdown in much less than 24 hours.
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MIKE K. MITCHELL

Mr. Mitchell is a Senior Engineer in the Plant Design & Analysis Section working for the General Electric Company, Nuclear Engineering Division, in San Jose, California, U.S.A. His employment with GE began in 1975 in the Piping Design Section. Subsequent assignments have been in the areas of Seismic & Dynamic Analysis, MK III Containment Technology & Heat Exchanger Design. Immediately prior to his present position, Mr. Mitchell was the supervisor of the Division's Engineering Training Program, which included technical responsibility for training of entry level engineers in the areas of heat transfer, fluid flow, and computer methods.

As a Senior Lead System Engineer, Mr. Mitchell is the person in GE with the primary responsibility and authority for the correct and complete design of the Residual Heat Removal System.

Mr. Mitchell is a member of the National Society of Civil Engineers and a registered Professional Engineer in the State of California. Mr. Mitchell is a 1975 graduate of the University of Arizona with a B.S. Degree in Civil Engineering. He is also a 1977 graduate of GE's Advanced Engineering Program and a 1978 graduate of the University of California, Berkeley, with a M.S. Degree in Engineering.

4-20 1 MR. CULP: The witness is available for cross-
2 examination, Mr. Chairman.

3 JUDGE WOLFE: Mr. Sohinki?

4 MR. SOHINKI: We have no cross-examination,
5 Mr. Chairman.

6 JUDGE WOLFE: Mr. Doherty?

7 MR. DOHERTY: Yes, Your Honor.

8 CROSS-EXAMINATION

9 BY MR. DOHERTY:

10 Q Starting on Page 2 of your testimony, you
11 speak of the initial phase of nuclear system cooldown
12 is accomplished by dumping steam, and that has kind of
13 gotten out of my understanding.

14 Would you tell me what the colloquial phrase
15 means?

16 A "Dumping steam"?

17 Q Yes.

18 A That means directing the steam that's being
19 produced by the decay process to the main condenser.

20 Q Bypassing the turbine?

21 A Yes.

22 Q Okay. How long does this initial phase take
23 place, as it's mentioned here on Page 2?

24 A It usually takes two hours.

25 Q And does that presume -- That presumes use of

4-21

1 the relief valves as needed?

2 A. I wouldn't expect the relief valves to lift
3 during this process because the reactor is being depres-
4 surized by the condensation that's taking place.

5 Q. Uh-huh.

6 A. The pressure initially starts at 1000 psi and
7 drops down to around 125 psi at the end of this two-hour
8 period.

9 Q. Okay. Now, there is a shift, you seem to
10 describe here, of moving from a dumping steam mode to
11 shutdown cooling mode of the residual heat removal system.
12 Is that something that must be done by the operators, or
13 does it happen automatically?

14 A. It requires operator action.

15 Q. I notice you used -- Now, in calculating this
16 cooling process, you used the essential service cooling
17 water system. What do you assume for a temperature of
18 that?

19 A. 95 degrees Fahrenheit.

20 Q. And that's the -- Where did you get that
21 number from?

22 A. That is the peak expected service water tempera-
23 ture. I believe that number is in the FSAR.

24 Q. Is it the temperature of the on-site cooling
25 lake?

1 A. That's correct.

2 Q. Okay. Do you know for a fact that that is
3 the highest temperature presumed for that lake, or do you
4 know if it's an average, or just a hot day; or do you know
5 what that is?

6 A. That is the peak expected temperature of the
7 cooling lake. It's not an average.

8 Q. Let me ask this: Is this source for this
9 essential service cooling water, is that like piped down
10 beneath the surface quite a distance in the lake?

11 A. You mean the suction source?

12 Q. Is it at the bottom of the lake?

13 A. I'm not sure. Normally it is close to the
14 bottom, such that there is enough suction head available
15 for the service water pumps.

16 Q. Okay. Now, you state at the top of Page 3,
17 "Because the heat load is much greater for" -- I'm missing
18 a term here. Which mode is that that you're speaking of
19 there?

20 A. Suppression pool cooling mode. This refers to
21 Suppression pool cooling mode.

22 Q. Okay. "... is much greater than for Shutdown
23 Cooling, the RHR System is considerably oversized for
24 achieving a normal cold shutdown condition."

25 This is based on the steam dumping initial

1 process having been run first; is that right -- this
2 comparison?

3 A. The comparison is based on the requirements for
4 heat removal in the suppression pool cooling mode versus
5 the requirements of removing heat in the shutdown cooling
6 mode.

7 I'm not sure I fully understand what you're
8 asking.

9 Q. Well, you made a comparison, and I'm trying to
10 figure out if determining how much heat load there is in
11 this shutdown cooling mode is -- assumes that the steam
12 dumping, described earlier in the testimony was done?

13 A. Yes, it does.

14 Q. It does assume that.

15 Can you think of any way in which a situation
16 would occur where steam dumping would not be possible?

17 A. Steam dumping to the main condenser?

18 Q. Yes.

19 A. There could be an isolation event whereby the
20 steam would be dumped to the suppression pool.

21 Q. Okay. Now, you did state earlier that the
22 turning off of steam dumping and the moving to shutdown
23 cooling mode was an operator decision or a manual action
24 anyway; is that right?

25 A. That's correct.

1 Q Can you think of any possibility where that
2 might be initiated too early for some reason?

3 A An operator error.

4 Q Uh-huh. Do you know if the operator instructions
5 generally ask him to wait -- what is he asked to wait
6 for?

7 A He's asked to wait until the pressure in the
8 reactor drops below the permissive, such that none of the
9 low pressure piping or equipment in the RHR system is
10 damaged.

11 Q So there's a number usually for each reactor?

12 A That's correct.

13 Q And he just has to watch until it gets to that
14 point?

15 A (Nods head, "Yes.")

16 Q Okay. In determining decay heat load -- I'm
17 on Page 3 still.

18 A Okay.

19 Q Is the decay heat load assumed to begin on
20 control rod insertion?

21 A Yes.

22 Q Okay. I guess I don't understand why all this
23 flushing on Page 3. I don't -- it states that it's
24 flushed with reactor grade water. That sounds like a
25 cleaning process, but I can't --

4-25
✓

1 A. That's correct. It's a cleaning process for
2 the heat exchangers and piping.

3 Q. Would this also be a cooling process?

4 A. No. It's a warming process actually where the
5 pumps and heat exchangers are warmed up before it's
6 placed into shutdown cooling.

7 Q. Why must this process be done?

8 A. It doesn't have to be done.

9 Q. Okay. So it could be skipped in an emergency
10 situation, if someone felt it was --

11 A. That's correct.

12 Q. I see.

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5-1 1 Q Well, you've put at the bottom there, I guess
2 that's the decay heat load at line 19 there. Would that
3 be a proper term for that? The 284.6, is that a decay
4 heat load?

5 A That's correct. That includes decay heat and
6 some heat removal to cool the metal down in the reactor.
7 It's a small portion of that heat load number that's
8 there.

9 Q Now, the figure 344 degrees Fahrenheit, looking
10 above where we were a minute ago, I don't know if you've
11 examined the testimony of the Staff's Applicant or not --

12 JUDGE WOLFE: The Staff's what?

13 MR. DOHERTY: Pardon me. Of the Staff's
14 witness.

15 JUDGE WOLFE: Yes.

16 BY MR. DOHERTY:

17 Q On page 2 of that he gives the same figure,
18 344 degrees.

19 Is that figure a calculation or is that
20 something more of a reading from a chart?

21 A That's a reading from a chart. The saturated
22 temperature of water at around 120 degrees psi -- 120 psi,
23 not 120 degrees psi.

24 Q Yeah, I thought that's what you meant.

25 You state one loop is placed in service. How

5-2 1 many loops are there?

2 A Two.

3 Q Two. Well, is there -- there are two. Now,
4 if one of those is inoperative for some reason, to your
5 knowledge is the reactor user permitted to go ahead and
6 operate?

7 A I believe he's allowed to operate for a certain
8 length of time. If the loop that's inoperable is not
9 placed back in service, he's required to bring the plant
10 down.

11 Q I see. Would that include an inoperable pump
12 as part of that?

13 A Yes, that would.

14 Q Now, do you know if there is shutdown required
15 if both loops are inoperative? Do you know what the rule
16 is there?

17 A Could you restate your question?

18 Q Yes. We asked a minute ago if one pump were
19 inoperative you felt that there was a shutdown in a certain
20 length of time.

21 I just have a feeling that if both of them are
22 off, I just wanted to find out what your understanding of
23 the rule was then.

24 MR. CULP: Mr. Chairman, I object to that
25 question. I think it's vague. I'm not sure what he means

5-3
1 by -- the witness may know, but I'm not sure what he means
2 by one of the requirements.

3 Is he asking is the plant allowed to operate
4 if both loops are inoperative, or is he asking some other
5 question?

6 MR. DOHERTY: No, that's what I was asking.

7 MR. CULP: Well, I'm going to object to the
8 question. If that's the question, I think it's outside
9 the scope of the contention.

10 MR. DOHERTY: Okay, I'll withdraw the question.

11 JUDGE LINENBERGER: Mr. Doherty, I thought
12 your question was whether the process of achieving cold
13 shutdown would be compromised if both recirc pumps were
14 inoperable.

15 From what you said to Mr. Culp, I gather that
16 was not your question.

17 MR. DOHERTY: That's right. It wasn't.

18 JUDGE LINENBERGER: Okay, thank you.

19 MR. DOHERTY: Thank you.

20 BY MR. DOHERTY:

21 Q There's a statement on page 4 where you give
22 kind of a summary where you say, "The RHR System is fully
23 capable of achieving a reactor coolant temperature of
24 less than 200 degrees in less than seven hours."

25 Do you assume there one or two loops in

5-4 1 operation?

2 A Two loops.

3 Q How about with just a single one, do you have
4 any --

5 A It might be slightly longer than that, but I
6 would not expect it to be significantly longer.

7 Q You state here that, "The heat exchanger
8 discharge flow is usually throttled such that the cooldown
9 rate does not exceed 100° F/hr."

10 Do you know why that is?

11 A That limitation is primarily to limit the
12 temperature induced stresses in the reacto. pressure
13 vessel. This is my understanding of that limitation.

14 If the reactor were cooled at a rate faster
15 than that every time it was shut down over the life of
16 the plant, the usage factor may increase above allowable.

17 Q Okay. You also mention, I guess an alternative,
18 suppression pool water may be directed into the RPV, and
19 you head that by saying, "If a single failure occurs." Is
20 that a single failure in the residual heat removal system
21 itself that you are speaking of there?

22 A It can be any single failure.

23 Q Are there any situations where the pool
24 water (looking at the bottom of 4) might be warmer than
25 150 degrees that you can think of?

5-5
1 A Yes, but not when you would be initiating
2 shutdown, the alternate shutdown cooling mode.

3 Q So if the reactor had had some type of transient
4 where it had discharged steam into the suppression pool,
5 there would be an increase of pool water there; isn't
6 that right?

7 A That's correct.

8 Q Are you then basing the idea on that if that
9 transient necessitated shutdown or it was decided to have
10 shutdown, that these other processes, the dumping steam
11 and so forth, would take so long that by then the
12 suppression pool temperature would go down to about 150
13 again? Is that your idea?

14 A I'm sorry, would you repeat the scenario again?

15 MR. DOHERTY: Would you repeat it, Mrs. Bagby,
16 for me?

17 (Record read.)

18 THE WITNESS: The 150 degrees Fahrenheit is
19 based on an initial suppression pool temperature at its
20 maximum tech spec limit which I believe is about 120
21 degrees, after which, if there was a transient such as
22 isolation that caused steam dump to the pool, it would heat
23 the pool up to something around 150 degrees, at which time
24 the steam would continue to dump to the pool, if it was
25 an isolation event, until the reactor was depressurized

5-6
1 to the shutdown cooling cut-in pressure permissive, at
2 which time the alternate mode could be placed into
3 operation, drawing suppression pool water at around 150
4 degrees.

5 BY MR. DOHERTY:

6 Q I think earlier in your written testimony
7 you spoke of the initial phase of nuclear system cooldown
8 being dumping steam to the main condenser.

9 In that answer it sounded like you were
10 suggesting that that would be to the suppression pool,
11 and that's....

12 A No. Under normal operation you would dump
13 steam to the main condenser.

14 Q Yes.

15 A Under a non-normal scenario, you might dump
16 steam to the suppression pool.

17 Q Okay. Then is seven hours sort of a ballpark
18 figure you'd say to go down to cold shutdown?

19 A Yes, sir.

20 MR. DOHERTY: No further questions. Thanks
21 very much.

22 JUDGE WOLFE: Is there redirect, Mr. Culp?

23 MR. CULP: No, sir.

24 JUDGE WOLFE: Board questions?

25 JUDGE CHEATUM: I have none.

BOARD EXAMINATION

6-1
bm
1
2 BY JUDGE LINENBERGER:

3 Q Mr. Mitchell, proceeding with a normal shut-
4 down, I believe you indicated the steam bypasses the
5 turbine and goes to the main condenser?

6 A Yes, sir.

7 Q Where does it go when it exits the main con-
8 denser? Back to the RPV?

9 A Yes. I'm trying to think of the routing that
10 it takes to get back to the RPV. I believe it's through
11 the feedwater line, after it's condensed, but I'm not sure
12 of the routing to get back to the reactor.

13 Q Well, are you saying possibly the routing is
14 different than it is for power operation; or are you
15 saying definitely the routing is different?

16 A It might be different.

17 Q All I was thinking about here is that the
18 turbine is no longer there to extract energy from the
19 steam. And so the rate of cooldown I would think might
20 be slower than if the turbine were allowed to spin
21 freely, so to speak, or spin with whatever driving force
22 there is in the steam.

23 Now, is there a particular merit to taking the
24 turbine out of the circuit at this point? I don't under-
25 stand why that's done.

1 A I believe it's taken out because it's outside
2 of the design requirements for the turbine. The reactor
3 power is something less than one percent, and it may not
4 be producing the quality of steam to run the turbine.

5 Q Oh, I see what you're saying. The turbine
6 would be experiencing off-spec steam quality --

7 A That's right.

8 Q -- and this could, you're saying, perhaps be
9 detrimental to the turbine?

10 A I believe that's why they trip it off line
11 and just go to the main condenser.

12 Q In several places in your testimony, for example,
13 at -- in the answer beginning on Page 3, you quote
14 some quantitative stake points and heat rate numbers.
15 And it's not clear to me where these have come from.

16 I understand the ANS formula for the shutdown
17 heat load. But when you talk about achieving a temperature
18 of 344 degrees in two hours, and at this time the heat
19 load is such and such -- various numbers are given
20 here ... where do these numbers come from?

21 How did you arrive at them?

22 A The heat load, 284.6, that's directly from the
23 curve, plus some sensible heat to cool down the metal in
24 the reactor.

25 Q Okay.

1 A The 110 psig is a ball park number for the
2 cut-in pressure permissive for shutdown cooling. The
3 actual number will depend on the design of the plant,
4 depending on how much pressure head there will be above
5 the low pressure portions of the RHR system.

6 344 is just saturation -- saturated tempera-
7 ture at that pressure.

8 The two hours to reach there is -- The
9 reactor normally operates about 550 degrees Fahrenheit,
10 down to 344 is about 200 degrees, and a 100 degrees
11 per hour.

12 So that's approximately two hours there.

13 The cooldown rate of 100 degrees per hour is
14 an operational number that's used. That's a historical
15 number, I believe.

16 Also, by using that cooldown rate, the fatigue
17 on the RPV is reduced.

18 Q Well, these all sound like reasonable and
19 plausible numbers. But I guess what I'm really looking
20 for is some assurance that somewhere, somebody has satis-
21 fied themselves that these numbers all fit together.

22 Is there an analysis report of some sort where
23 one could find these numbers and the derivation of them?

24 A Yes, sir.

25 Q Can you give a citation?

6-4 1 A I have a design record file -- an internal GE
2 document, actually for every nuclear plant, and specifically
3 for Allens Creek, documenting the computer code and also
4 the hand calculations that show that the heat -- that the
5 numbers given in the report are accurate.

6 Q You say that the numbers given in the report
7 are accurate. Which report do you -- Oh, you mean the
8 testimony?

9 A I'm sorry. I mean the testimony.
10 Specifically, the seven hours -- the seven-hour
11 number is the only number there that was really derived.

12 Q Is there a General Electric report of some sort
13 in the open literature that would allow one to -- that
14 could be consulted to get a feeling for whether these
15 behavioral considerations are reasonably what GE intends
16 for this system?

17 A I'm not sure. There are operating plants
18 that we've correlated our analyses with. But an external
19 document, I don't know whether there is or there isn't.

20 Q All right. I think what you've just said is
21 significant. I interpret what you've just said as
22 follows: That GE has looked at system performance
23 parameters in operating plants during, let's say, normal
24 shutdown, which is what we're talking about here, to
25 verify that these kinds of numbers really go with their

1 hardware?

2 A. That's correct.

3 Q. Okay. Finally, I think you indicated that
4 the seven hours you've calculated was based on the as-
5 sumption of two recirc loop operation, that if one were
6 lost, the number would not be significantly increased --
7 the time would not be significantly increased.

8 A. Yes, that's --

9 Q. I don't know whether it's a fair question or
10 not, but I'll ask it anyway. Suppose both recirc pumps
11 were unavailable. How would you handle the system
12 then?

13 A. Well, the recirc pumps aren't used -- You
14 mean the RHR system pumps --

15 Q. The RHR --

16 A. How would you cool the plant down if that were
17 to happen?

18 Q. Yes. How would you achieve your cold shut-
19 down?

20 A. The cold shutdown would take longer to ac-
21 complish, but it would be done by either the pool cooling
22 heat exchangers, or the reactor water clean-up system
23 heat exchangers, which are small.

24 Depending on when in the scenario you postulated
25 losing both loops, that would determine how long it would

1 take to reach cold shutdown.

2 Q Let's postulate that as soon as called upon,
3 they were not available; then what would the seven hours
4 stretch out to be?

5 A I don't know. If you didn't have any -- you
6 didn't have either RHR loop; is that what you're asking?

7 Q Right.

8 A I won't venture a quantitative guess on that.
9 I don't know. It would depend on the size of the heat
10 exchangers in the other two systems.

11 Q I'm not sure this represents a credible cir-
12 cumstance, but I'm wondering if GE has indeed looked at
13 this kind of an eventuality. You say you don't know.
14 But I'm curious whether it might take a week to cool it
15 down, or seven hours as opposed to 36 hours, or what we're
16 talking about here.

17 And I gather you can't offer any --

18 A You're probably looking at something around a
19 week, in that ball park -- just to give you a ball park
20 number.

21 GE has, in reference to your other -- GE has
22 looked at off-design basis scenarios, but when those
23 occur, generally we don't -- if it's an incredible event,
24 we don't take the incredible starting point numbers --
25 for example, rather than taking the peak suppression pool

6-7
1 temperatures, we take a more reasonable value, say 85
2 degrees rather than 95, and a few other initiating factors
3 in the transient to satisfy ourselves, just from an
4 engineering point of view, that the plant would be
5 safe.

6 Q I think it's implicit in what you've said,
7 but I need to ask you if you can explicitly state this --
8 not in your personal opinion, but as a GE Company position
9 that it is not credible to anticipate the early loss of
10 both RHR systems here during an attempt to bring a plant
11 to cold shutdown?

12 A Well, my own personal opinion is it's not
13 credible. As far as a GE position, I think it's implicit
14 in the design requirements that we have agreed to work
15 under that we do not feel it's implicit; we do not feel
16 that it's a credible event to lose both loops.

17 I don't know if I want to go so far as to say,
18 "That is the GE position."

19 It would clarify things if I would say, "Yes,
20 that is the GE position." But I don't know if I can go
21 that far.

22 Q You have to be candid with us here. And I
23 appreciate that.

24 JUDGE LINENBERGER: Thank you, Mr. Mitchell.
25 That's all I have.

1 JUDGE WOLFE: Cross on Board questions, Mr.
2 Sohinki?

3 MR. SOHINKI: No, sir.

4 JUDGE WOLFE: Mr. Doherty?

5 MR. DOHERTY: Yes, Your Honor.

6 RE-CROSS-EXAMINATION

7 BY MR. DOHERTY:

8 Q What is the power source for the residual
9 heat removal pumps?

10 A If on-site -- If off-site power is not avail-
11 able, the power source is diesel generators. The primary
12 source is off-site power.

13 MR. DOHERTY: Okay. No further questions.

14 JUDGE WOLFE: Redirect, Mr. Culp?

15 MR. CULP: Yes, sir, I have one question.

16 REDIRECT EXAMINATION

17 BY MR. CULP:

18 Q Mr. Mitchell, do you know whether the NRC
19 requires you to consider the loss of both trains of the
20 RHR system in -- whether you're required to consider
21 that as a design basis in bringing the plant to a cold
22 shutdown -- the loss of both trains of the RHR system?

23 A That is not a requirement.

24 MR. CULP: Thank you. That's all the questions
25 that I have.

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JUDGE WOLFE: All right. Is the witness to be excused permanently?

MR. CULP: Yes, sir.

JUDGE WOLFE: The witness is excused permanently.
(Witness excused.)

JUDGE WOLFE: We will recess until 2:00.

(Whereupon, at 12:40 p.m. the hearing was recessed, to reconvene at 2:00 p.m. of the same day.)

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AFTERNOON SESSION

2:00 p.m.

JUDGE WOLFE: All right.

MR. SOHINKI: I would like to recall Mr. Hodges to the stand, Mr. Chairman.

JUDGE WOLFE: You are still under oath, Mr. Hodges.

MR. SOHINKI: Mr. Hodges will be testifying with regard to the 24-hour cold shutdown.

JUDGE WOLFE: All right. Whereupon,

MARVIN W. HODGES

was recalled as a witness and, having been previously duly sworn to testify the truth, the whole truth and nothing but the truth, was examined and testified further as follows:

DIRECT EXAMINATION

BY MR. SOHINKI:

Q Mr. Hodges, do you have before you a three-page document entitled, "NRC Staff Testimony of Marvin W. (Wayne) Hodges on Doherty Contention 38B"?

A Yes, I do.

Q Was this document prepared by you or under your direct supervision?

A It was prepared by me.

-2
1 Q Do you have any additions or corrections to
2 make to the document at this time?

3 A No.

4 Q Is everything contained therein true and
5 accurate, to the best of your knowledge, information and
6 belief?

7 A Yes, it is.

8 MR. SOHINKI: Mr. Chairman, Staff would move
9 at this time that Mr. Hodges' testimony with regard to
10 Doherty Contention 38B be incorporated into the record
11 as if read and accepted as evidence on behalf of the
12 Regulatory Staff.

13 JUDGE WOLFE: Any objection?

14 MR. CULP: Applicant has no objection.

15 MR. DOHERTY: I would like to ask some
16 questions on voir dire of the witness.

17 JUDGE WOLFE: All right.

18 VOIR DIRE EXAMINATION

19 BY MR. DOHERTY:

20 Q Did you write any of the SER, Supplement No. 2,
21 particularly Section 5.4.5?

22 A No, I did not.

23 MR. DOHERTY: That's all the questions I
24 have. No objections.

25 I just wanted to check on that.

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JUDGE WOLFE: All right. The testimony of Marvin Hodges on Doherty Contention 38B is incorporated into the record as if read.

(See attached pages.)

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)	
HOUSTON LIGHTING AND POWER COMPANY)	Docket No. 50-466
(Allens Creek Nuclear Generating)	
Station, Unit 1))	

NRC STAFF TESTIMONY OF
MARVIN W. (WAYNE) HODGES ON DOHERTY CONTENTION 38B

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

A. My name is Marvin W. (Wayne) Hodges. I am employed by the U.S. Nuclear Regulatory Commission as a Section Leader, Section B, in the Reactor Systems Branch of the Division of Systems Integration. A copy of my professional qualifications has previously been submitted.

Q. What is the purpose of your testimony?

A. The purpose of this testimony is to respond to Doherty Contention 38B which states:

Contrary to NUREG-0578, the reactor cannot be brought to cold shutdown in 24 hours.

Q. What is meant by "cold shutdown"?

A. "Cold shutdown" means that the average reactor coolant temperature is less than 212°F and the reactor mode switch is in the shutdown position with the reactor subcritical.

Q. Does the NRC require that the reactor be brought to cold shutdown in 24 hours?

A. No. Branch technical position RSB 5-1, given in the Standard Review Plan (NUREG-75/087) states that the RHR system(s) shall be capable of bringing the reactor to a cold shutdown condition, with only off-site or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure. A reasonable time has generally been interpreted to be about 36 hours. The 36 hours has been based on the availability of high quality water to the steam generators of PWRs. Times longer than 36 hours (up to 72 hours) have been accepted by the Staff.

Q. Can ACNGS be brought to cold shutdown in less than 24 hours?

A. Yes. Based on information provided in the ACNGS PSAR, my calculations show that the ACNGS can be brought to cold shutdown in less than 10 hours. For these calculations, I used the curve for heat removal capability of the RHR heat exchanger given in section 5 of the ACNGS PSAR and I conservatively assumed that the decay heat remained constant at 38.3 Mw, which is the decay heat after being shutdown for two hours.

The initial phase of the cooldown, which consists of dumping steam to the condenser, is limited by technical specifications to 100°F/hr. Therefore, it takes at least two hours to reach the pressure (125 psia) at which the RHR system is normally used for shutdown cooling. At 125 psia, the saturation temperature is approximately 344°F while the initial operating temperature was 544°F. Even with allowing time to flush the RHR system, my calculations show that cold shutdown can be achieved in less than 10 hours.

Q. What is your conclusion regarding Mr. Doherty's allegation?

A. The contention appears to be based upon a misunderstanding of regulatory requirements. While the Allens Creek reactor will not be required to reach cold shutdown within 24 hours under the postulated conditions in the Standard Review Plan, my calculations demonstrate that cold shutdown can be achieved in substantially less than 24 hours. Thus, there is no basis for the concern expressed by Mr. Loherty.

7-4
1 MR. SOHINKI: We have no further direct
2 examination, Mr. Chairman. The witness is available for
3 cross-examination.

4 JUDGE WOLFE: Mr. Culp.

5 MR. CULP: No, sir, we have no questions.

6 JUDGE WOLFE: Mr. Doherty.

7 MR. DOHERTY: Yes, I have some cross-
8 examination, Your Honor.

9 CROSS-EXAMINATION

10 BY MR. DOHERTY:

11 Q There's a definition of cold shutdown on page
12 1 of your testimony. Can you tell us where you got that
13 or did you get that from anyplace or what?

14 A Yes, sir. The definition that I quoted here
15 I took from the technical specifications for the La Salle
16 plant.

17 There is also a definition in, I think, the
18 Standard Technical Specifications that uses 200 degrees,
19 rather than 212 degrees Fahrenheit, as I have.

20 Since they are reasonably close, I stayed with
21 the La Salle definition.

22 Q So cold shutdown varies from plant to plant?

23 A There are minor differences. Generally, the
24 212 is used on PWR's and the 200 is used on boiling water
25 reactors, and why the difference, it's probably tradition.

1 I really don't know.

2 Q What about pressure? Is there any pressure
3 technical specification on cold shutdown?

4 A No.

5 Q Is there anything about cold shutdown that
6 makes -- well, let's ask this.

7 Can you attempt repairs on a reactor when it's
8 in cold shutdown?

9 MR. CULP: Your Honor, I'm going to object to
10 that question. I don't understand the relevance of that
11 question to the contention.

12 MR. DOHERTY: Okay, I'll skip that.

13 BY MR. DOHERTY:

14 Q I think the sum of some of your testimony is
15 that, and some you may have heard this morning, is that
16 the reactor can be brought down to cold shutdown fairly
17 quickly, without arguing the number.

18 What are the advantages, if any, in doing that?

19 A In doing it quickly?

20 Q Yes.

21 A There are no large advantages. The requirements
22 that the NRC has had in its Standard Review Plan have been
23 based primarily upon the availability of high quality
24 water for pressurized water reactors, and not so much on
25 the need to get the plant itself to cold shutdown immediately.

7-6 1 Q Did you attend any of the post-TMI conferences?

2 Well, I think you indicated earlier that you had been
3 involved in some of the post-TMI activities of the Commission;
4 isn't that right?

5 A That's correct.

6 Q Do you recall the bringing up at any time at
7 any of those meetings of the idea of having 24 hours as a
8 limit?

9 MR. SOHINKI: Objection. Mr. Chairman, I
10 thought we had already decided to strike the portion of
11 the contention which refers to the TMI requirements -- or
12 TMI recommendations, excuse me.

13 MR. DOHERTY: I don't think that's an important
14 objection. I don't think it is an objection.

15 I don't think I have to be talking about the
16 words in the contention or stay away from some that were
17 rejected earlier completely.

18 MR. SOHINKI: The contention as originally
19 worded said, "contrary to the requirements of a document
20 which was issued as a result of the TMI accident."

21 That's the portion that was struck.

22 MR. DOHERTY: I just don't think that's that
23 important in terms of the acceptability of the question.

24 MR. SOHINKI: As I understand it, Mr. Chairman,
25 all we're discussing now is whether in fact the reactor

1 can be brought to cold shutdown in less than 24 hours,
2 period.

3 JUDGE WOLFE: Your question is what was the
4 discussion? Again, your question?

5 MR. DOHERTY: Yes --

6 JUDGE WOLFE: What was the discussion as a
7 result of TMI-2 about the period of shutdown; was that
8 your question?

9 MR. DOHERTY: Yes.

10 JUDGE WOLFE: I think that has a bearing on the
11 contention. Objection overruled. Answer the question.

12 THE WITNESS: You were asking me if I had
13 participated in discussions?

14 BY MR. DOHERTY:

15 Q Yes.

16 A Okay. I was not a member of the Lessons Learned
17 Task Force which came forth with the recommendation for the
18 24 hours. However, I did have several discussions with
19 members of that Task Force who were advocating that, and
20 I'm familiar with what they had in mind.

21 Q What did they have in mind?

22 A They were concerned that if you had a loss of
23 a safety function -- for example, you found that your
24 high pressure injection system were inoperable so that you
25 did not have that safety function, they wanted the plant to

7-8 1 be brought down very quickly and for the utility to have
2 to come into the NRC and explain why it was okay for them
3 to go back to operation.

4 The 24 hours was somewhat arbitrary.

5 Q Uh-huh. You speak of the Standard Review Plan
6 at the top of page 2. Does that have any time requirement
7 at all?

8 A It doesn't have a numerical value in it. It
9 just says "a reasonable time period."

10 As I said, based upon the availability of high
11 purity water for the PWR's for the auxiliary feedwater,
12 that has been interpreted to be about 36 hours.

13 In cases where they could show a source of
14 high purity water for a longer period of time, they have
15 been allowed to go to a longer period of time.

16 Q Has there been any interpretation or tradition
17 or anything about BWR's?

18 A BWR's have not really had a problem in satisfying
19 that time requirement because they could generally get to
20 cold shutdown much quicker than that.

21 Q Do BWR's operate at higher or lower temperatures
22 than PWR's?

23 A They operate at a little lower temperature.

24 Q And I take it they operate at a lower pressure?

25 A Yes.

7-9 1 Q Okay. There is a calculation on page 2 which
2 is an answer to the question right above of whether it can
3 be brought to cold shutdown in less than 24 hours.

4 It's an answer in megawatts, I guess.

5 How many heat exchangers are assumed operating
6 in that calculation; do you know?

7 A For that calculation on the heat removal
8 capability of the heat exchangers, I just used a curve
9 that's in Section 5 of the Allens Creek PSAR.

10 It was not clear from the legends on that
11 curve whether that was one or two heat exchangers. I
12 assume it's probably two.

13 Q You are taking that from the testimony of the
14 witness this morning? Is that the reason for your
15 assumption?

16 A Well, if it were one, they would probably want
17 to wave the flag and say they were taking credit for only
18 one.

19 Q Do you remember any better what this -- I
20 mean, Section 5 is a large section of the PSAR. Do you
21 remember any better what it was, or what that figure was,
22 where you got this information?

23 A I don't recall the figure number. If you need
24 it, we can get a copy of the PSAR and point to it.

25 Q I notice that the next paragraph, some of your

7-10 1 testimony seems to track the Applicant's witness.

2 You state there's a limitation of 100 degrees
3 Fahrenheit per hour for decrease, but then you state, "It
4 takes at least two hours to reach the pressure...."

5 When someone says "at least," I'm kind of
6 wondering just how conservative a figure that is.

7 A Okay. If he were to cool down at a rate of
8 100 degrees F. per hour, which is the technical specification
9 limit, then that would take two hours.

10 The operator in trying to do that will probably,
11 over that two-hour period, average slightly slower rate in
12 order not to exceed the cooldown rate.

13 So it might take him, rather than two hours
14 to get to that point, it might take him two-and-a-half
15 hours.

16 Q Did you do the calculation of the decay heat?

17 A Yes, I did.

18 Q That's a measure of heat, not power; is that
19 right?

20 A Okay. When you said "decay heat," I assumed
21 you meant did I do the calculation of the cooldown rate.

22 I did not generate the decay heat curve itself.
23 I used a normalized decay heat curve that I pulled out of a
24 computer program.

25 Q Well, there's a statement here that's confusing

7-11 1 me. It says -- it's somewhat in the middle of page 2.

2 "I conservatively assumed that the decay heat remained
3 constant at 38.3 megawatt."

4 I'm having trouble even placing that.

5 A. Okay. What I did was -- you can correlate the
6 decay heat in terms of a decay heat fraction, a fraction of
7 full operating power as a function of time.

8 To simplify my calculation, rather than treat
9 the decay heat as a variable input, which is a decreasing
10 heat input, I recognized the fact that you would not
11 start using the RHR heat exchangers for the first two
12 hours because you would be cooling down by bypassing steam
13 through the main condenser.

14 So I took the decay heat level, in this case a
15 normalized power level fraction, and I multiplied that by
16 the full operating power to get 38.3 megawatts, which would
17 be the decay heat existing two hours after shutdown.

18 So then I assumed that was the heat source,
19 the decay heat source, that the RHR heat exchanger would
20 have to remove for the remainder of the time.

21 Q. Okay. Now, in doing that sort of calculation
22 is there any requirement that one of the control rods remain
23 out?

24 A. There is a requirement that the reactor has
25 to remain subcritical with the highest worth rod, I think,

7-12 1 out; yes.

2 Q That's subcritical, though? It's different.
3 This is not getting to the criticalness?

4 A That's correct.

5 Q Okay. Do you have a copy of the SER, the
6 Supplement No. 2, the brownish one?

7 A Yes, I do.

8 Q Would you look at page 5-21, please? Excuse me,
9 5-9.

10 A Okay.

11 Q The statement there states, "The plant will
12 have Seismic Category I systems capable of bringing the
13 plant to cold shutdown within approximately 36 hours,
14 taking credit only for those actions that can be performed
15 from the control room and assuming a single active failure
16 in the systems."

17 Taking the end of it first, what do they
18 mean, "assuming a single active failure in the systems";
19 do you know?

20 A My interpretation of that paragraph is that
21 he's considering the fact that he's got two independent
22 RHR heat exchangers and two recirculation loops, and he's
23 also looking at alternate shutdown cooling method we
24 discussed this morning where you are not pumping from the
25 reactor vessel, but pumping from the suppression pool

7-13 1 through the RHR heat exchangers and you fill the reactor
2 vessel up with water so you are discharging through the
3 relief valves.

4 So my interpretation of that paragraph is that
5 he's talking about both those modes.

6 Q Okay. Then is he really saying that the
7 alternate system is going to be slower in that then?

8 A The alternate system would be slower, yes.

9 Q So you believe, then, that he's talking about
10 that system, really, not talking about the RHR system?

11 A That's most likely true, yes.

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7-14 1 Q You state at the last page of your testimony
2 that it can be brought to cold shutdown in substantially
3 less than 24 hours.

4 What does "substantially" mean to you in that
5 case, less than half?

6 A Well, I think earlier in my written testimony
7 it said in less than ten hours it could be brought to cold
8 shutdown. My calculations actually show somewhat less than
9 ten hours, but because of some uncertainties in some of
10 the input parameters, I chose to pick a higher number.

11 Q Did you allow two hours for flush time as the
12 GE witness did earlier?

13 A Yes, I did.

14 Q Was there anything that you heard from the
15 witness this morning that you disagree with at all with
16 regard to Contention 38B?

17 A In substance I don't disagree with what he
18 said.

19 There was one point where he was talking about
20 when the operator gets down to cutting on the RHR system,
21 I think he left the impression that the operator can
22 commit an error and start the system at a higher pressure
23 than 125 pounds.

24 There is an interlock that would not allow
25 him to do that. So it would take a failure of that

7-15
1 interlock, plus the operator error of committing that
2 act, in order to cut the system in at a higher pressure.

3 The system is designed for low pressure
4 operation. So in that sense, I think he was in error;
5 but in substance, I agree with his testimony.

6 Q Something like, you called it, an interlock.
7 I don't quite understand what that is. Would that be a
8 warning light saying, "Don't do this yet"?

9 A No. This is if he tried to open the valve, the
10 valve would not open.

11 Q I see, so he would remain on the steam
12 dumping mode?

13 A Until he got down to the lower pressure, yes.

14 Q I see. Are these systems generally designed to
15 start operating at roughly the same pressure and not at
16 higher pressures?

17 A These systems being --

18 Q RHR.

19 A -- RHR systems?

20 Q Uh-huh.

21 A The design pressures are very close on all of
22 them, yes. They don't vary markedly.

23 Q Now, I think you stated these were available
24 even if there were a loss of on-site power because there
25 were diesel engines to back them. Is there a single diesel

7-16 1 for each unit or one diesel for both; do you know?

2 A. There are three diesels for one unit.

3 Q. Three diesels for one unit?

4 A. Yes. However, one of those is committed to
5 the high pressure core spray. So I don't believe it would
6 be available for the RHR.

7 So, basically, for the RHR systems, there are
8 two diesels available.

9 Q. Okay.

10 MR. DOHERTY: No further questions, Your Honor.
11 Thank you very much, Mr. Hodges.

12 JUDGE WOLFE: Redirect, Mr. Sohinki?

13 MR. SOHINKI: No, sir.

14 JUDGE WOLFE: Board questions?

15 JUDGE CHEATUM: I have no questions.

16 BOARD EXAMINATION

17 BY JUDGE LINENBERGER:

18 Q. Mr. Hodges, with respect to this 125 psia value
19 that must be reached before activating the RHR system to
20 continue the shutdown cooling, you indicated two, if you
21 will, barriers there to prematurely starting it; one,
22 an interlock, which I presume is tied to a pressure
23 sensor somewhere.

24 A. That's correct.

25 Q. And secondly, operator awareness, if you will,

7-17 1 of the problem.

2 Does the Commission really lock upon this as
3 two barriers? The reason I ask the question is that I can
4 envisage myself sitting there saying, "Well, that interlock
5 is not going to let this thing come on prematurely. So I
6 don't have to watch the clock too closely. It seems like
7 it's been about an hour and a half. I'll go try and see if
8 I can activate."

9 So what I'm saying is that it seems to me human
10 nature being what it is, the interlock may be really the
11 only thing that's helping out here.

12 How does the NRC look at this?

13 A. We would consider the operator error as a
14 separate failure, so that it would be two failures to breach
15 the system.

16 Q. So you really do feel that operator discipline
17 is active here in preventing --

18 A. He has usually very explicit procedures on how
19 to start up the RHR system, at what pressure to do it, and
20 to deviate from that is directly an error and --

21 Q. That must be reported to the NRC; do you know?

22 A. I'm not certain whether that would have to be,
23 that particular one would be or not.

24 A fair number of them would be, but I just
25 don't know about that one.

7-18 1 Q Belaboring this just a bit more, how critical
2 is that 125 psia? Suppose there were some -- we've talked
3 at other times about set point drifts or whatever, and if
4 the RHR systems were started at 150 psia or 175, is that
5 serious? How critical is the 125?

6 A The RHR system itself is generally designed for
7 450 to 475 pounds of pressure. So if you open the valve
8 at 200 pounds, rather than at 125, you are not going to fail
9 the system.

10 Q Okay. What is the value -- what is the purpose
11 of flushing the RHR system?

12 A I would think that the RHR system could have
13 been operating in the suppression pool cooling mode. That
14 water may not be quite as clean as the water that's in the
15 vessel and just to try not to introduce the impurities.

16 I also heard in this morning's testimony about
17 heating up the system, which would be nice, but I was not
18 aware of that prior to this morning's testimony.

19 Q Okay, so it's a matter of maintaining some
20 additional control on contaminants going into the RPV;
21 is that it?

22 A That's my understanding.

23 Q I believe Mr. Mitchell testified that there was
24 nothing go or no-go about that, that if needed, the RHR
25 system could be put into operation without this pre-flush

7-19 1 activity; is that your understanding?

2 A I would agree with that, yes.

3 Q Finally, Mr. Mitchell was questioned this
4 morning about what kind of handicap the plant shutdown
5 operation might be under if neither RHR system was
6 functional.

7 Can you comment on that, please?

8 A If neither system were operable, then if you
9 had to shut the reactor down, it would be preferable not
10 to stay in a hot shutdown mode, for example, where you would
11 use the reactor core isolation cooling system to remove
12 decay heat.

13 There are alternative sources of heat removal
14 available, as we discussed this morning. The fuel pool
15 heat exchanger can be used, although it's not the one
16 that's normally considered.

17 Q Where would you get your pumping capability
18 from if you used the fuel pool heat exchanger?

19 A The RHR pumps will pump -- they are piped so
20 they will pump through that heat exchanger. It's a lot of
21 piping you are going through, but you can do it; and he
22 also mentioned in his testimony this morning about the
23 reactor water cleanup system, but that's a very small heat
24 exchanger, so it would only be worth about maybe half of
25 a percent in the decay heat and would not be a very usable

1 one for some period of time.

2 So with neither RHR heat exchanger available,
3 the preferable mode would be to either keep operating or to
4 be in a hot standby condition where you could dump heat
5 to the condenser.

6 If in the event that you could not dump to the
7 condenser and you had to shut down, you do have the fuel
8 pool heat exchanger, and for some period of time you could
9 pump water from the surface water system to the reactor
10 itself and dump it into the suppression pool to remove heat
11 from the vessel, but you would be limited in the amount of
12 time you could do that because of filling up the containment.

13 Q I'm not quite clear in my memory of what
14 Mr. Mitchell said this morning with respect to the initial
15 mode of cooldown in which you are bypassing the turbine
16 and dumping reactor steam directly into the main condenser
17 with respect to the point of whether or not the steam
18 returns to the reactor via the same piping circuit as it
19 does in normal operation or whether it must be valved
20 through some other circuit or pumped in some other way.

21 Would you address that point, please?

22 A The answer is both. If you have the normal
23 feedwater available, then you would be pumping from the
24 condenser through the feedwater system back to the reactor
25 vessel.

7-21 1 If for some reason the feedwater system were
2 not operating, you would be using the reactor core
3 isolation cooling system to supply makeup water to the
4 vessel. It would draw its suction from the condensate
5 storage tank.

6 However, you could take the water from the
7 condenser and pump it to the condensate storage tank.

8 Q Okay, so there's multiple choice there?

9 A There's multiple choice.

10 Q You indicated --

11 A Just one comment. I think in normal
12 circumstances you would be using the reactor core isolation
13 cooling system rather than the feedwater, because of
14 having to throttle the feedwater back drastically. So that
15 would be the normal, I would believe.

16 MR. DOHERTY: Excuse me. Could you repeat
17 yourself, "because of --" and then I lost what you said.

18 THE WITNESS: Okay. I'm saying normally you
19 would use the reactor core isolation cooling system,
20 because to use the feedwater you would have to throttle
21 back a fair amount on one of your pumps.

22 So the preferred source under shutdown
23 conditions would be the reactor core isolation cooling
24 system.

25 MR. DOHERTY: Thank you very much.

7-22 1 BY JUDGE LINENBERGER:

2 Q Is this throttling capability readily
3 achievable or does it take a bit of finagling to be able
4 to throttle back?

5 A It's available. It's achievable.

6 Q Finally, Mr. Hodges, on page 2, the second full
7 answer on that page, you discuss your use of the heat
8 removal capability curve for the RHR heat exchanger taken
9 from Section 5 of the Allens Creek PSAR.

10 Have you or the NRC independently assured
11 itself or satisfied itself that that heat removal curve in
12 the PSAR is reliable, is accurate, is what it ought to be?

13 A At this point we have not, because I tried to
14 find detailed information on the heat exchangers in order
15 to do the calculation, and it just is not available in the
16 PSAR.

17 This is something that would normally be done
18 at the FSAR stage when you have a fair amount of detail on
19 the tube area and the flow velocities. You could verify
20 the heat transfer coefficients.

21 But at the PSAR stage it's not there and we
22 are probably fortunate to even have that curve. I don't
23 think everyone even supplies that curve.

24 But at the PSAR stage there's not the detail
25 to do that.

7-23 1 Q All right. Then just one final detail on
2 that.

3 Is there any way you have of knowing if that
4 curve were grossly inaccurate? In other words, such that
5 the 28 megawatt -- I mean, such that the results of your
6 calculation would be distorted because of gross inaccuracy
7 in that curve.

8 Do you have any way of telling that it's
9 correct at least within some ballpark number?

10 A Well, we can compare it with a comparable
11 design and see if we've got roughly the same curve. That
12 would be the only way at this point.

13 Q Compare it with another plant?

14 A Compare it with another plant, yes.

15 Q And has that been done?

16 A I did not do that.

17 JUDGE LINENBERGER: All right. Thank you, sir.
18 That's all.

19 JUDGE WOLFE: Cross, Mr. Culp?

20 MR. CULP: No, sir.

21 JUDGE WOLFE: Mr. Doherty?

22 MR. DOHERTY: Yes, I have a couple of questions.

23 RE-CROSS-EXAMINATION

24 BY MR. DOHERTY:

25 Q You stated that you didn't compare this to

7-24 1 another plant. Was that because there was no plant
2 available like this sufficiently, or was it because of some
3 other reason?

4 A No, I did not compare it to another plant. I
5 don't think the times are drastically different from what
6 we've seen on other plants, but I did not sit down and make
7 a direct comparison.

8 I was not surprised by the answer, but in
9 answering his question I had to say I did not make a
10 direct comparison.

11 Q Okay. Did you say a moment ago that -- I
12 think you called it SWCU, reactor water cleanup, would take
13 a long time if it were used instead of the RHR? Was that
14 what you said roughly? I may have a letter wrong there.

15 A It's just the reactor water cleanup system that
16 I think you are referring to, yes, and it would indeed
17 take a long time if you were using that.

18 MR. DOHERTY: Okay, thank you very much.

19 JUDGE WOLFE: Any redirect, Mr. Sohinki?

20 MR. SOHINKI: I have one or two questions.

21 REDIRECT EXAMINATION

22 BY MR. SOHINKI:

23 Q Mr. Hodges, you were being questioned by
24 Judge Linenberger with regard to the postulated situation
25 in which neither RHR exchanger was available, and if you'll

7-25
1 recall the testimony from this morning, Mr. Mitchell stated
2 that in his opinion a ballpark estimate for bringing the
3 reactor to cold shutdown was in the nature of a week.

4 Q Would you agree with that assessment?

5 A Without having done the calculation, that would
6 seem reasonable, but I haven't calculated it to be
7 certain.

8 Q All right, and with regard to this 38.3 megawatt
9 decay heat, if the decay heat was in fact after two hours
10 ten or twenty percent higher than that, how would that
11 affect your calculation with regard to the time it takes
12 to reach cold shutdown?

13 A It would not affect it by ten or twenty percent.
14 The heat removal capability of the heat exchangers was
15 very large, so it would come down fairly quickly.

16 Even on the calculation, once you get to the
17 heat removal with the RHR system, initially you are limited
18 by 100 degrees F. per hour cooldown rate.

19 MR. SOHINKI: All right. That's all the
20 questions I have, Mr. Chairman.

21 JUDGE WOLFE: All right. We now proceed to
22 Doherty 41; is that correct?

23 MR. SOHINKI: Yes, sir, and for that contention
24 I would like Dr. Huang to join Mr. Hodges at the witness
25 table.

1 MR. SOHINKI: Your Honor, I would ask that
2 Dr. Huang be sworn.

3 JUDGE WOLFE: Would you remain standing and
4 raise your right hand.

5 Whereupon,

6 TAI L. HUANG

7 was called as a witness and, having been first duly
8 sworn, was examined and testified as follows:

9 JUDGE WOLFE: You may be seated.

10 DIRECT EXAMINATION

11 BY MR. SOHINKI:

12 Q Dr. Huang, do you have before you a six-page
13 document entitled "Supplemental Testimony of Tai L. Huang
14 Regarding Reactor Water Level Indicators," in paren,
15 "Doherty Contention 41 and TEXPIRG Additional Contention
16 54," together with Attachments A, B and C, Attachment A
17 being entitled "Professional Qualifications - Tai L.
18 Huang - July 1981"; and Attachments B and C being figures
19 representing the Oyster Creek and Allens Creek designs?

20 BY WITNESS HUANG:

21 A Yes, I do.

22 Q Was this testimony prepared by you or under
23 your direct supervision?

24 BY WITNESS HUANG:

25 A Yes.

8-2
1 Q And do you have any additions or corrections
2 to make to this document at this time?

3 BY WITNESS HUANG:

4 A No.

5 Q And is everything contained therein true and
6 accurate to the best of your knowledge, information and
7 belief?

8 BY WITNESS HUANG:

9 A Yes.

10 Q And do you adopt this testimony as your testi-
11 mony in this proceeding?

12 BY WITNESS HUANG:

13 A Yes.

14 Q Mr. Hodges, have you reviewed the testimony
15 submitted by Dr. Huang?

16 BY WITNESS HODGES:

17 A Yes.

18 Q And were you consulted with regard to this
19 testimony by Dr. Huang?

20 BY WITNESS HODGES:

21 A Yes.

22 Q And in further explanation of your appearance
23 on the stand with Dr. Huang, would you explain for the
24 benefit of the Board and the parties the part that you've
25 had with the NRC Staff with regard to review of reactor

1 water level instrumentation?

2 BY WITNESS HODGES:

3 A Following the Oyster Creek event in May of
4 1979, I was involved in evaluating what occurred at Oyster
5 Creek and calculating the water levels within the core
6 region and trying to determine whether the fuel had become
7 uncovered during the event and trying to explain why the
8 differences in the reading.

9 Also, as a part of the Builders and Owners
10 Task Force that was formed following the Three Mile Island
11 event, in reviewing the boiling water reactor systems I
12 was involved to a considerable extent in reviewing the
13 water level indication systems on all of the boiling water
14 reactors.

15 Q All right, sir. And are you satisfied that Dr.
16 Huang's testimony is accurate?

17 BY WITNESS HODGES:

18 A Yes.

19 Q And do you adopt the statements made by Dr.
20 Huang as your testimony in this proceeding?

21 BY WITNESS HODGES:

22 A Yes.

23 MR. SOHINKI: Mr. Chairman, the Staff would move
24 at this time that the testimony of Dr. Huang be incorporated
25 into the record as if read and accepted as testimony on

8-4
1 behalf of the Regulatory Staff.

2 JUDGE WOLFE: Any objection?

3 MR. COPELAND: No objection.

4 MR. DOHERTY: I'd like to take the witnesses on
5 voir dire, Your Honor.

6 JUDGE WOLFE: All right.

7 VOIR DIRE

8 BY MR. DOHERTY:

9 Q I'm going to ask you questions about your per-
10 sonal qualifications. You may want to put them in front
11 of you.

12 Dr. Huang, have you ever testified before a
13 Licensing Board?

14 BY WITNESS HUANG:

15 A No.

16 Q How many years have you been employed by the
17 NRC?

18 BY WITNESS HUANG:

19 A Six years.

20 Q Did you read the testimony of the Applicant on
21 this issue?

22 BY WITNESS HUANG:

23 A I did.

24 Q Did you, Mr. Hodges? Did you read that testi-
25 mony?

8-5
1 BY WITNESS HODGES:

2 A. I have read the written testimony.

3 Q. That's what I mean, yes.

4 Dr. Huang, are you familiar with the technical
5 specifications on operating when the water level indicators
6 have malfunctioned?

7 BY WITNESS HUANG:

8 A. I know that they are -- Are you referring
9 to that operating procedure type of an inspection or
10 what?

11 Q. All right, let's skip the question. I don't
12 think it's that good right now.

13 In your -- In the second paragraph, Dr. Huang,
14 of your personal qualifications, you state, "I was
15 responsible for the review of the thermal hydraulic aspect
16 of containment designs in my previous work assignment"

17 When did you leave that? Do you know what year
18 you left that?

19 BY WITNESS HUANG:

20 A. A year and a half ago; last year.

21 Q. Have you authored any publications for the
22 NRC?

23 BY WITNESS HUANG:

24 A. With regard to internal reports or what --
25 NUREG's?

1 Q Yes, NUREG's particularly.

2 BY WITNESS HUANG:

3 A No.

4 Q Have you authored any internal reports?

5 BY WITNESS HUANG:

6 A It's a generic report, yes.

7 Q What was the title of that, or subject?

8 BY WITNESS HUANG:

9 A "Containment in the" -- let me refresh my --
10 "Heat Conduction Between the Containment Wall."

11 Q Okay.

12 MR. DOHERTY: I have no further questions,
13 Your Honor; and the witnesses -- I think we should continue
14 with them.

15 JUDGE WOLFE: All right. Absent objection,
16 the supplemental testimony of Tai Huang regarding reactor
17 water level indicators, inclusive of Attachments A through
18 C, are incorporated into the record as if read.

19 (See attached pages.)

20 - - -

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)
HOUSTON LIGHTING AND POWER COMPANY) Docket No. 50-466
(Allens Creek Nuclear Generating)
Station, Unit 1)

SUPPLEMENTAL TESTIMONY OF
TAI L. HUANG REGARDING REACTOR WATER LEVEL INDICATORS

[Doherty Contention 41 and
TEXPIRG Additional Contention 54]

Q. Please state your name and position with the Nuclear Regulatory Commission.

A. My name is Tai L. Huang. I am employed as a Nuclear Engineer in the Thermal-Hydraulic Section of the Core Performance Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. A statement of my professional qualifications is attached (Attachment A) to this testimony.

Q. What is the purpose of your testimony?

A. The purpose of my testimony is to respond to two consolidated contentions which are basically concerned with the possibility of spurious water level indication at Allens Creek based primarily on incidents at Three Mile Island and Oyster Creek. The consolidated contention (Doherty 41 and TEXPIRG Additional Contention 45) reads as follows:

Intervenor's health and safety interests are endangered due to inadequate water level indicators for the reactor vessel for the proposed atomic

plant. That such indicators are often defective and mislead operators into actions which aggravate reactor incidents are evidenced by two recent incidents at U.S. facilities. At Three Mile Island, Unit II, spurious water level indications in the pressurizer and the reactor vessel resulted in operator errors which aggravated the event (March 29, 1979); and spurious water level indications in the Oyster Creek Nuclear Power Plant, resulted in operators failing to take action until the water level was dangerously low (May 2, 1979) - specifically the operator failed to open valves which would have allowed coolant to be pumped from the condensor to the reactor vessel. Intervenor contends Applicant must develop an alternative whereby the water level is sensed more reliably by redundant as to type level indicators and redundant as to function water level indicators. Intervenor contends an accident where a core uncovering results from unreliable water level sensing can lead to a release of radioactivity in excess of 10 C.F.R. 100, endangering his health and safety interests.

Intervenor further contends that inadequate water level indicators will lead to serious accidents for ACNGS, as at Three Mile Island, because the reactor systems are sufficiently similar in design being both dependent on safety systems actuated when reactor water level threatens to reach the top of the fuel rods. Because the proposed ACNGS has a higher power core density than any BWR this contention is particularly relevant to this proceeding. The Oyster Creek event provides a basis for showing much of the accident sequence has occurred in a BWR system.

Q. Is the reactor water level indication system to be installed at Allens Creek the same or similar to that employed at Three Mile Island, Unit 2?

A. No. The systems are completely different.

Q. Would you please explain the typical water level indication system which has been used in PWRs?

A. For TMI and other PWRs, the normal water level range in the reactor coolant system is within the pressurizer and is maintained by the pressurizer control system. Under normal circumstances, if there is some level indication in the pressurizer, the rest of the system should be full of coolant. However, under TMI conditions, i.e., stuck open PORV, steam flow into the pressurizer prevented drainage of pressurizer coolant such that the pressurizer indicated a water level while the primary coolant system was not full.

Q. What is different about the system you have just described compared to the one which will be installed at Allens Creek?

A. It should be apparent from the previous answer that PWRs presently have no reactor water level instruments in the reactor vessel itself. However, all BWRs, including Allens Creek, have pressure taps on the reactor vessel so that vessel level indications can be received by the operators in the control room.

Q. Explain briefly how the BWR water level indication system operates.

A. In BWRs, water level is measured by the operation of differential pressure sensing devices which have had a long and reliable inservice history in BWRs. Condensing chambers connected to the steam space in the reactor vessel are used as the reference leg. Pressure taps at different levels in the water space of the reactor vessel are used as the variable leg sensing taps for narrow and wide range instruments. Narrow range instruments and associated control room indicators and recorders monitor water level approximately between the bottom of the steam dryer skirt and five feet above that point. Wide range instruments and associated control

room indicators and recorders monitor water level approximately between one foot above the top of the active fuel and five feet above the bottom of the steam dryer skirt.

The differential pressure in the two legs permits determination of reactor pressure vessel water level, since the water level is a function of the differential pressure.

Q. Are the pressure sensing devices and associated control room indicators and recorders described in your previous answer fully redundant as to function.

A. Yes. There are eleven separate differential pressure sensing channels and control room indicators and recorders. Each water level range in the reactor vessel is overlapped by more than one separate sensing/indicating channel. There are two wide range level indicators/recorders and one wide range indicator, one narrow range level indicator/recorder and three narrow range indicators, one fuel zone indicator and an indicator/recorder, a high water level upset range indicator/recorder (overlaps the narrow range and wide range indicators and recorders) and a shutdown wide range level indicator (overlaps the upset range recorder).

The narrow range instruments are used to indicate water level for normal plant operation and the wide range instruments are used for ECCS initiation as a result of a low water LOCA transient. All of differential pressure devices and associated readout instruments in the control room will have to comply with the applicable provisions of Regulatory Guide 1.97, Revision 2, specifically those set forth in Part C, Section 1.3.1, "Design and Qualification Criteria-Category 1." These criteria include, among others, redundancy, single failure protection, and environmental and seismic qualification.

Q. If BWRs all have in-reactor vessel pressure taps for direct water level indication, and Oyster Creek is a BWR, why couldn't the spurious water level indication incident which occurred at Oyster Creek occur at Allens Creek?

A. Oyster Creek is a BWR-2 plant. The reactor coolant flow path in that design is through the annulus, recirculation lines and core area. For the level instrumentation to work properly, there must be an unrestricted and direct flow path between the annulus and core area so that the level indication will be consistent in both areas. For a non-jet pump reactor design such as Oyster Creek, there is a circumferential core shroud which acts as a buffer and restricts good fluid communication between the annulus and core region when all five recirculation loops are isolated. (See Attachment B). With all loops isolated, in a reactor scram the water level in the annulus might be higher than in the core because the steam generated by decay heat would condense back into the annulus region (but not into the core). Operating procedures at Oyster Creek have since been modified to eliminate this problem. However, for a jet-pump BWR-6 reactor design, such as Allens Creek, there is always good fluid communication between the two regions, since nothing restricts water flow whether the recirculation pumps are operating or not. (See Attachment C). Therefore, the reactor level instruments for Oyster Creek did provide a discrepant vessel level indication, while for Allens Creek there will always be a consistent and accurate level indication for both regions.

Q. What are your conclusions regarding this contention?

A. The reactor water level indication system to be installed at Allens Creek is different in critical respects from those used at TMI-2 and Oyster Creek, and the incidents at those facilities provide no cause for concern over the adequacy of the Allens Creek design. The Staff is confident that the water level indication system at Allens Creek will perform its intended function properly because:

(1) it is based on pressure taps on the reactor itself and differential pressure sensing devices which have been used reliably in BWRs for many years

(2) it is employed in a reactor design which virtually eliminates the possibility of discrepant level indication in the annulus and core areas

(3) it will be designed in accordance with the stringent provisions of Regulatory Guide 1.97, Rev. 2.

Attachment A

Professional Qualifications

Tai L. Huang

July 1981

I am presently employed with the U.S. Nuclear Regulatory Commission as a Nuclear Engineer in the Thermal-Hydraulics Section of the Core Performance Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation.

In my present work assignment at the NRC, I am responsible for the review of the reactor core thermal-hydraulic design, reload, and the functional requirements for core monitoring systems to provide capability for detection and response to inadequate core cooling conditions. I was responsible for the review of the thermal hydraulic aspect of containment designs in my previous work assignment at the NRC.

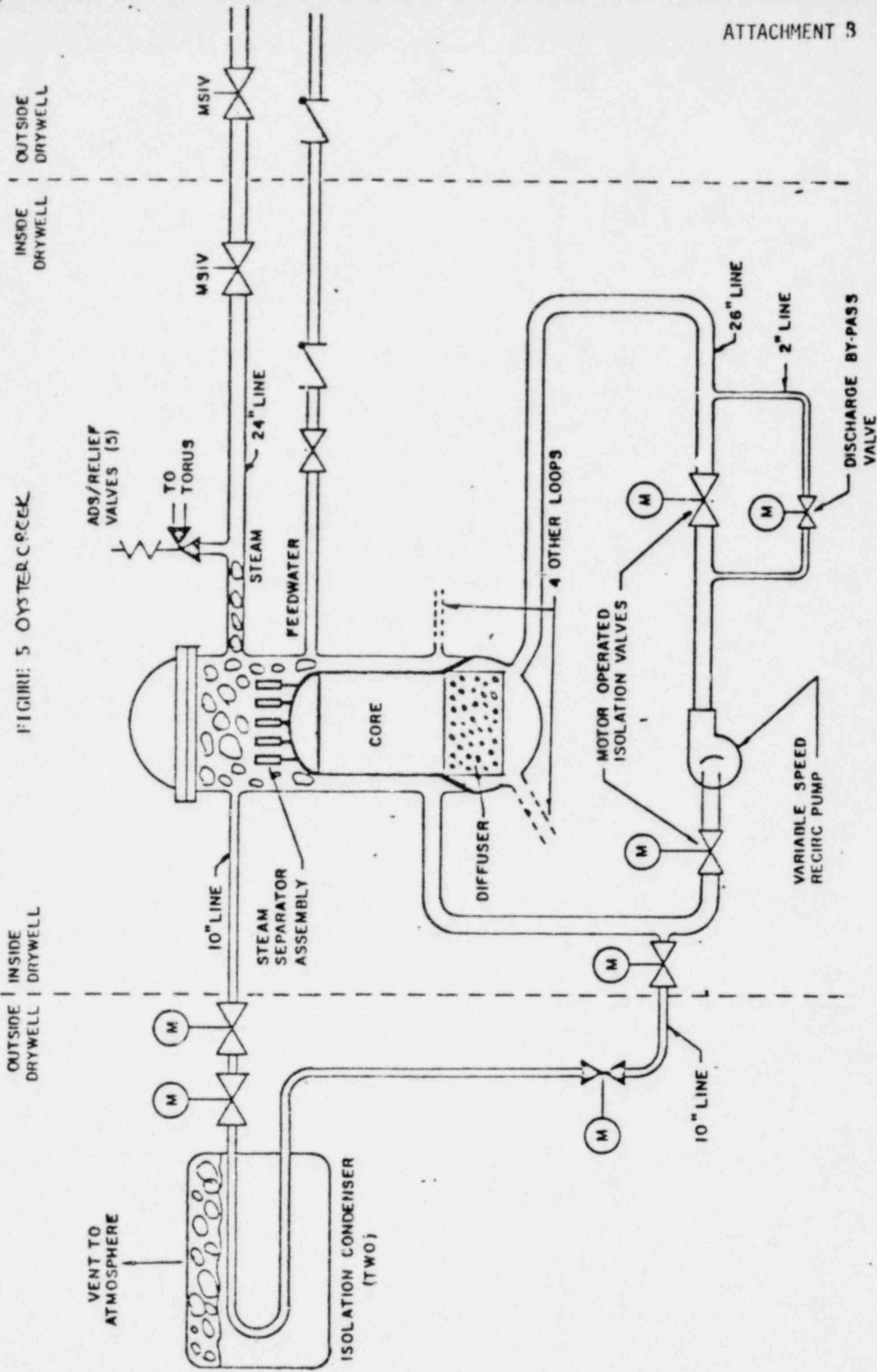
Prior to joining the NRC Staff in March, 1975, I was employed by Boeing Company as a Senior Mechanical Engineer (from 1972 to 1975). I was responsible for the thermal and fluid flow analysis for improving the aircraft engine performance and the environmentally controlled system design.

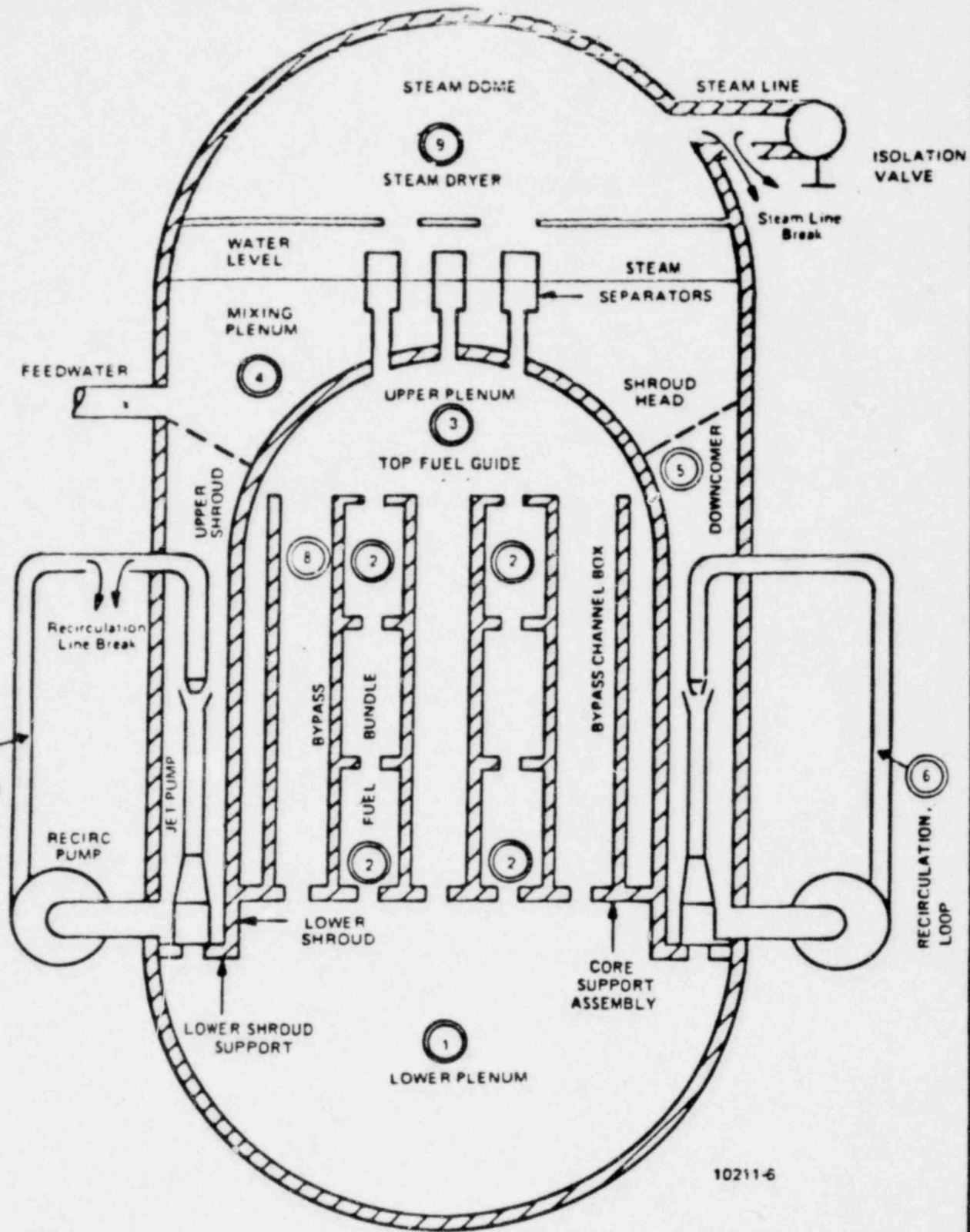
In 1972 I was employed by the Radiation Biology Laboratory of Smithsonian Institute as a Mechanical Engineer to be in charge of environmentally controlled chamber design.

In 1971 I was employed by the Research Laboratory for Engineering and Science (RLES) of the University of Virginia as a Senior Scientist to investigate the thermal-hydraulic properties of fluids.

I graduated from the University of Virginia with a Ph.D. degree in Aerospace Engineering, 1970. I received a M.S. degree in Mechanical Engineering from the University of Iowa, 1967 and a B.S. degree in Mechanical Engineering from Cheng Kung University, Taiwan, 1964. I am a registered Professional Engineer in the State of Maryland.

FIGURE 5 OYSTERCREEK





10211-6

HOUSTON LIGHTING & POWER COMPANY
 Allens Creek Nuclear Generating Station
 Units 1 & 2

PRESSURE NODES USED FOR
 DEPRESSURIZATION ANALYSIS

FIGURE 4.2-8

8-7
1 MR. SOHINKI: I have no further direct examina-
2 tion, Mr. Chairman. The witnesses are available for cross-
3 examination.

4 JUDGE WOLFE: Cross, Mr. Copeland?

5 MR. COPELAND: No, sir.

6 JUDGE WOLFE: Mr. Doherty?

7 MR. DOHERTY: Yes, Your Honor.

8 CROSS-EXAMINATION

9 BY MR. DOHERTY:

10 Q There were several corrections you made --
11 that were made on the testimony -- let's call them changes --
12 earlier. I guess it was yesterday that they were made.

13 One was on Page 3 and was a fairly small one,
14 but I wonder what the significance of it was. Are the
15 pressure taps which indicate the position of the water in
16 the reactor -- are they -- it seems as if for a while the
17 testimony said they were inside. Now they're said to be
18 on.

19 Now, what -- how was this determined to
20 change or --

21 BY WITNESS HODGES:

22 A The pressure taps -- or primarily they are
23 small holes that are drilled in the wall of the vessel
24 with connections -- for the piping to be connected to
25 them.

1 The tap itself is the hole that's on the wall
2 or through the wall of the vessel, rather than being
3 located inside of the vessel. So it was felt that would
4 be a little more precise, to say "on the vessel," rather
5 than "inside the vessel."

6 Q Okay. Now, is the pressure tap merely a
7 pipe, essentially leading out?

8 BY WITNESS HUANG:

9 A A capillary tube. It's a tube. It's a pipe ...
10 big.

11 Q Okay.

12 BY WITNESS HODGES:

13 A May I add a comment?

14 Q Yes, sir.

15 BY WITNESS HODGES:

16 A The pipes are -- they are small diameter pipes.
17 I think they're in the neighborhood of half an inch in
18 diameter, as far as the pipe leading from the vessel.

19 The hole in the vessel wall itself would not be
20 that large. It's just a -- It's like flush --
21 The pressures in the reactor are transmitted into the
22 pipe -- into the piping or tubing leading from the reactor
23 vessel.

24 And the fluid in that pipe being relative
25 incompressible -- being water, any changes in pressure in

1 the vessel itself is transmitted down the present trans-
2 mitter very quickly.

3 Q Now, when water drops below one of these taps --
4 because I've never seen one of these or had a chance to
5 work with one -- does the water then run out of the
6 tube; or does it stay in?

7 BY WITNESS HODGES:

8 A The tap should come in essentially horizontal
9 at the vessel itself. And from that point, the instrument
10 line would tend to slope towards the instrument -- or
11 down at a lower elevation.

12 So for the most part, the fluid would tend to
13 stay in the instrument line.

14 Q Has --

15 BY WITNESS HODGES:

16 A That's true for, for example, the variable
17 leg.

18 Q Well, then it's a situation where the level
19 goes up and goes -- passes and then drops below and then
20 rises again, around a tap point. Is there any chance of
21 losing accuracy in this device, or will it keep its
22 accuracy regardless of this sort of up or down ...

23 BY WITNESS HODGES:

24 A During the time period that the water level
25 is below the tap, the tap will indicate essentially a

8-10 1 constant level. The pressure transducer will indicate
2 essentially a constant level, when the water level in-
3 creases above the tap and then it'll show an increasing level
4 again.

5 Q Is this type of level indicator which you've
6 described, is this one that has been in use for many
7 years? Or has this been modified since Three Mile Island
8 or any recent events?

9 BY WITNESS HODGES:

10 A The same basic system has been in use for many
11 years. Now, on the older plants there was a variation
12 on the system, called the Yarway system. It was a Yarway
13 design.

14 I suppose Yarway was a corporation -- where
15 they enclosed both the variable leg and the reference
16 leg inside of a shroud, so that they would be heated
17 to basically the same temperature to provide some tempera-
18 ture compensation.

19 This was done on some of the instruments, but
20 not all of the instruments. The newer designs, like the
21 BWR-6, does not use the Yarway design.

22 But, otherwise, they're essentially the same
23 instrument.

24 Q Uh-huh. Further down on Page 3, you speak of
25 the steam space. Is that simply the space above the water

8-11 1 level in the reactor vessel in that --

2 BY WITNESS HODGES:

3 A. Yes.

4 Q. Okay. The last complete sentence you speak
5 about narrow range instruments and that these monitor
6 water level approximately between the bottom of the steam
7 dryer skirt and five feet above that point.

8 Are these the highest point in the reactor?

9 BY WITNESS HUANG:

10 A. This is the range for -- narrow range to indi-
11 cate a level. Not the highest point of the level.

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BY MR. DOHERTY:

Q It is the highest in the vessel --

BY WITNESS HUANG:

A. No, not the highest in the vessel.

JUDGE LINENBERGER: Mr. Doherty, with your indulgence here, I should like to ask this panel if they could somehow make verbal reference to Attachment C of Dr. Huang's testimony and tell us approximately where on that diagram this steam skirt -- steam dryer skirt is located.

It's not quite clear to me where it would be. And because we are being verbally recorded here, you can't very well point to the diagram and get that into the transcript.

So if you would give a verbal description of about where the steam skirt is, it would be appreciated.

WITNESS HODGES: Attachment C is a very simplified diagram. It's used more for computer code modeling rather than trying to locate the instruments.

But the bottom of the skirt -- the dryer skirt extends to below the larger diameter section of the steam separators and would be, roughly, on a line -- horizontal line with the top of the upper plenum dome ... not exactly, but approximately on that type of a line.

So the water level would be somewhere inside

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1 the dryer skirt, the skirt providing a seal -- a water
2 seal between the inside and the outside of that whole
3 annulus area.

4 JUDGL LINENBERGER: Okay, thank you.

5 BY MR. DOHERTY:

6 Q When you say it monitors between two points,
7 are you saying there's one of these taps at each end?
8 Or are you saying -- What does that mean?

9 BY WITNESS HODGES:

10 A The taps actually extend beyond the range
11 for the readout. That's just indicating the range that's
12 given on the scale of the instrument.

13 Q Where is this scale on the instrument?

14 BY WITNESS HODGES:

15 A That's located in the control room.

16 Q That's in the control room?

17 BY WITNESS HODGES:

18 A That's right. That's an indication in the con-
19 trol room. There's a scale that covers that range.

20 Q Okay. So the water can move between these taps
21 up or down, and it will be indicated in the control room;
22 is that correct -- if there's a change in the level between
23 the taps?

24 BY WITNESS HODGES:

25 A That's correct.

1 Q Okay. Now, you state at the top of four that
2 differential pressure in the two legs permits determina-
3 tion of the reactor pressure vessel water level, since
4 the water level is a function of the differential pres-
5 sure.

6 So then is this really an indirect measurement,
7 in a sense?

8 BY WITNESS HODGES:

9 A Really what you're doing is you're measuring
10 the weight of a column of water between the two points,
11 and taking that as representative of the water level.

12 In actuality, the two-phase water level could
13 be somewhat higher because the void content in the two-
14 phase mixture would cause an increase in the level.

15 So this is a measure of the level without
16 void.

17 Q Now, the bulk of Page 4 seems to be a sort of
18 description of instruments and monitors, how many. Now,
19 is it true -- factually true, that some of the wide-range
20 indicators overlap some of the narrow-range indicators?
21 In other words, they're both measuring sort of the same
22 thing, only you wouldn't really need to know that for
23 sure, but there's like a check --

24 BY WITNESS HODGES:

25 A There is overlap.

8-15
1 Q Now, in this description down here, you have
2 something called an indicator/recorder. Is that a unit
3 that does both -- indicate and record?

4 BY WITNESS HUANG:

5 A The recorder is something that ... paper
6 roll -- with a scale, so you can indicate what the level
7 is.

8 Okay. So it's also recording on that paper,
9 so it has two functions.

10 Q So then where you state, "There are two wide
11 range level indicators/recorders and one wide range
12 indicator," then you're saying on two of these places
13 we get an indication and a record, and in one place we just
14 get an indication?

15 BY WITNESS HUANG:

16 A Yes.

17 Q Okay. Now, in reading the accounts of the
18 Oyster Creek event, they spoke about different levels.
19 And there, apparently, is a level -- a last level below
20 the top of the fuel.

21 How high will that be at Allens Creek? Do you
22 have any idea yet, for where the lowest level monitoring
23 will be done above the fuel?

24 BY WITNESS HODGES:

25 A No. In fact, I think you may have it reversed,

8-16 1 if I'm understanding your question.

2 The lowest level indication at Oyster Creek
3 was above the top of the fuel. The lowest level indication
4 at Allens Creek is below the bottom of the fuel.

5 Q Okay. Is it correct to state this about the
6 current plans for this plant: It does have the capability
7 to record vessel water level over the range from the top
8 of the vessel dome to the lowest pressure tap?

9 MR. SOHINKI: Could I ask what -- Mr. Doherty
10 appears to be reading from something. Can you please
11 identify that document?

12 MR. DOHERTY: If you want to know, yes. It's
13 NUREG-0626, called "Generic Evaluation of Feedwater
14 Transients in Small Break Loss-of-Coolant Accidents in GE-
15 Designed Operating Plants and Near-Term Operating License
16 Applications."

17 MR. SOHINKI: This is for near-term operating
18 license applications?

19 MR. DOHERTY: "GE-Designed Operating Plants
20 and Near-Term Operating License Applications."

21 MR. SOHINKI: Perhaps you could show that to
22 the witnesses so that they don't take one statement out
23 of context. I don't know if, in fact, it is being taken
24 out of context, but at least they can read the --

25 MR. DOHERTY: I'd be glad to, no problem with

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1 that.

2 (Pause while document is shown to witnesses.)

3 WITNESS HODGES: Okay. I'm familiar with the
4 words you showed me there because I wrote them.

5 And it's my understanding that the Allens Creek
6 instrumentation will measure -- well, they have a commit-
7 ment to be able to measure from the center line of the
8 steam line down to below the elevation of the fuel,
9 which is not quite the same as that recommendation.

10 But it should still be sufficient for what we
11 were trying to get. That recommendation was written to
12 cover the full gamut of BWR's, from the old BWR-1, which
13 may have -- at least one of them has a steam line coming
14 right out of the top of the dome and going vertically up-
15 ward for some distance, up to the most modern BWR-6.

16 And what we were trying to get at is to be able
17 to measure -- and if you had water up to through the
18 steam lines, and we're getting water down the steam
19 lines, so the -- if I understand the commitment from Allens
20 Creek correctly, they will be able to do that, although
21 they will not be able to measure all the way to the top
22 of the steam dome, as was recommended in that report.

23 BY MR. DOHERTY:

24 Q Uh-huh. Well, taking a look at the other end
25 of this, in your opinion would this -- having said that

1 this is true of Allens Creek with that small exception,
2 would you say that would be giving the kind of coverage
3 against uncovering the fuel that was sought after in this
4 document?

5 BY WITNESS HODGES:

6 A. I don't recall the exact words of the commitment,
7 whether it says "down to the lowest half" or "down to the
8 bottom of the fuel." Either one of them -- Either of
9 those commitments would provide the coverage we were look-
10 ing for.

11 Q. Okay. At the foot of Page 4, it mentions
12 Regulatory Guide 1.97, Revision 2, and it mentions the
13 criteria -- the subjects of the criteria.

14 What are the requirements set forth therein
15 with regard to redundancy?

16 BY WITNESS HUANG:

17 A. Regarding the redundancy here, this is to cover
18 the single failure criterion. If you have a single
19 failure throughout the accident monitoring systems, and then
20 we had to have a two separate system to cover that, not
21 just for the redundancy.

22 This is specified in the Reg Guide 1.97 of
23 Class I -- Category 1.

24 Q. Is that also -- Have you also given the
25 requirement for single failure protection just now?

1 BY WITNESS HUANG:

2 A Yes.

3 Q I see.

4 BY WITNESS HUANG:

5 A That's what Allens Creek commits to do.

6 Q Okay. And what is the environmental qualifi-
7 cation for --

8 BY WITNESS HUANG:

9 A They have to qualify for -- also specified in
10 Reg Guide 1.97 -- and actually it has to be qualified
11 for Reg Guide -- to meet Reg Guide 1.89 -- 100.

12 1.89 is for that environmental qualification,
13 and 100 is seismic qualification.

14 Q In the seismic qualification, would that require
15 it to qualify with regard to the safe shutdown/earthquake
16 or something more --

17 BY WITNESS HUANG:

18 A Yes.

19 Q Just a safe shutdown?

20 BY WITNESS HUANG:

21 A For earthquake.

22 Q And for the environme cal does it give a parti-
23 cular environment expected in the containment or in that
24 area that the containment --

25 /

1 BY WITNESS HUANG:

2 A. Yes.

3 Q. -- for a design basis loss-of-coolant accident?

4 BY WITNESS HUANG:

5 A. Yes, for service condition, yes.

6 Q. Okay. Are they -- At Oyster Creek the indi-
7 cators were located outside the shroud; isn't that right?
8 That is, they were inside, I guess, in some way or on.
9 They certainly were not inside the shroud; is that
10 right?

11 BY WITNESS HODGES:

12 A. Are you talking about the location of the pres-
13 sure taps themselves?

14 Q. Yes.

15 BY WITNESS HODGES:

16 A. They were located on the vessel wall, which in-
17 dicated the level in the annulus region between the vessel
18 and the core shroud.

19 Q. And that's true of Allens Creek also, right?

20 BY WITNESS HODGES:

21 A. Yes.

22 Q. Are there any kinds of -- Is there any data
23 on possible differences between the water level in the
24 annulus and the water level in the reactor vessel?

25 MR. SOHINKI: I'm not sure I understand that

1 question. What kind of data are you talking about?

2 MR. DOHERTY: Well, counsel, data that would
3 diccate measurement differences.

4 MR. SOHINKI: Where? At any operating plant
5 they're familiar with or --

6 MR. DOHERTY: No -- Well, at least try to keep
7 it to BWR's for sure.

8 MR. SOHINKI: At operating BWR's?

9 MR. DOHERTY: Yes.

10 JUDGE LINENBERGER: And I think it needs to be
11 refined a little bit more, if I'm going to have a chance
12 to understand the answer. That is, under what operating
13 condition: normal operating condition with -- well, normal
14 operating condition in which the two-phase surface is well
15 above the core, up in the steam separator area, Mr.
16 Doherty?

17 MR. DOHERTY: Yes, let's try that first anyway.
18 We might want to try something else.

19 That would be a normal operation.

20 WITNESS HODGES: The only thing approaching
21 direct data on operating reactors that I'm aware of would
22 be the -- Well, I'm not aware of any direct comparisons
23 there are on the old boilers of the Oyster Creek vintage.

24 There's one tap that's located on the spray
25 sparger inside the shroud just above the fuel. That

8-22
1 instrument provides only an alarm and not a direct level
2 indication into the control room.

3 And the other level instruments do not go
4 down to that range. But indirectly, you can compare them as
5 long as you can track the transient water level and compare
6 it with when you would get the alarm.

7 I'm not aware of any direct data comparing
8 them.

9 BY MR. DOHERTY:

10 Q But this, you said, was in a -- I think you
11 said older -- or in Oyster Creek or something like that.

12 BY WITNESS HODGES:

13 A In plants of the Oyster Creek vintage.

14 Q How about plants of the vintage where there were
15 jet pumps and --

16 BY WITNESS HODGES:

17 A I'm not aware of any data that would give you a
18 direct comparison during the operation. There are no in-
19 strument taps inside to provide that.

20 Q Okay. Are you aware of any situations where it
21 was found that the water level indications were different
22 in normal operation of a jet pump BWR?

23 BY WITNESS HODGES:

24 A Well, again, with no direct data comparisons,
25 it would be difficult to answer your question. The one

1 case that does come to mind on a jet pump plant, the
2 fuel zone range instrument, the tap is not located on the
3 vessel wall, as are the other taps.

4 It's actually located in the throat -- in the
5 nozzle, not the throat -- but in the nozzle of the jet
6 pump, which is more directly connected to the core region
7 than the others.

8 But I don't think there has ever been a cor-
9 relation of those two even, because they're calibrated
10 for different temperatures from what the other instru-
11 ments are calibrated for.

12 Q Have you ever heard of any off-normal event
13 where it was discovered later that there was -- or that
14 there was strong evidence that the water levels had --

15 BY WITNESS HODGES:

16 A In a jet pump plant?

17 Q In a jet pump plant.

18 BY WITNESS HODGES:

19 A No.

20 Q Can you conceive of any accident condition
21 where that might occur?

22 BY WITNESS HODGES:

23 A I believe we discussed briefly yesterday two
24 situations where you might get a discrepant water level
25 indication. One would be when you would first turn on

8-24

1 the core spray system. If you could support flooding at
2 the upper part of the core and build up of the level of
3 water, then there is a possibility -- let's say, for
4 example, after a loss-of-coolant accident where you've
5 lowered the level in the core itself.

6 So you have a water level in the core and a
7 separate water level above the core, then the indicated
8 level would not give you the core level. But that would
9 be -- Even if that situation could exist, it would be a
10 very transient situation because recent test data from
11 the Lynn Test Facility, looking at the effects of core
12 spray and countercurrent flow at the top of the core
13 shows that this flooding breaks down very quickly as soon
14 as the sprays are turned on and the water drains down
15 into the lower plenum in the core region.

16 And so this would be an -- And that occurs
17 in the first fraction of a second. It's very quick. So
18 that would be a very transient time when you might not get
19 a good indication.

20 If you were in an accident and it proceeded
21 to the point that you remain uncovered for a substantial
22 period of time, that it got extensive core damage, then
23 the blockage in the core would have to exceed something on
24 the order of 95 percent.

25 If you did that, then you could get a difference

8-25

1 in the level indication. But it takes something of
2 that nature.

3 Q Let's see, that was the other one you mentioned
4 yesterday?

5 BY WITNESS HODGES:

6 A Yes.

7 Q There's a kind of puzzling statement here on
8 Page 5, about six lines up from the bottom. It states,
9 " ... since nothing restricts water flow whether the re-
10 circulation pumps are operating or not," I guess I can't
11 conceive of them not operating. How might they not
12 operate?

13 BY WITNESS HODGES:

14 A Turn the power off.

15 Q Just the power off.

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1 MR. DOHERTY: No further questions.

2 JUDGE WOLFE: Redirect, Mr. Sohinki?

3 MR. SOHINKI: No, sir.

4 JUDGE WOLFE: Board questions?

5 JUDGE CHEATUM: I have none.

6 BOARD EXAMINATION

7 BY JUDGE LINENBERGER:

8 Q Gentlemen, despite previous testimony on this
9 subject and your testimony, written and oral, there are
10 many things about this system that I don't understand, and
11 it's not important that I understand them, but it is
12 important that they be understood from what appears in the
13 record.

14 So I need to ask some questions.

15 First off, let's talk about how in theory a
16 schematized system would work. Let's discuss first a
17 series of vertical standpipes all of the same diameter,
18 incrementally increasing in height until you come to a
19 tallest standpipe that represents the full potential for
20 water level swing in the reactor vessel.

21 Now then, connected with each of those stand-
22 pipes or connected to each of those standpipes, as I
23 understand the way the system in theory performs, there is
24 a pressure transducer associated with each of those
25 standpipes, and the tallest one is designated as the

1 reference standpipe.

2 We put all these pipes in a vessel of water
3 that's obviously tall enough to contain them all and we
4 start lowering the water level in that vessel.

5 The tallest standpipe retains the water that
6 was in it when the vessel was full; the shorter standpipes
7 while the vessel was full all give the same pressure
8 reading in the pressure transducer at their base, let's say
9 because even though the pipe itself does not extend to the
10 top of the vessel, there is the equivalent of a water
11 column above it. When the vessel is full, I would
12 assume that all pressure transducers are reading the same.

13 Now, am I on track so far?

14 BY WITNESS HODGES:

15 A. In theory, basically you are on track. They
16 may --

17 Q. I want to understand the theory first, and
18 then come back.

19 BY WITNESS HODGES:

20 A. They may not read exactly the same because of
21 being calibrated for different temperature conditions.

22 Q. I'm coming to that ultimately. Fine.

23 But for now, let's have everything as near the
24 same as possible. I just need to understand some things
25 here.

3-3 1 Now, you start lowering -- we put all these
2 standpipes in a closed cylinder which is full of water,
3 and we start lowering the water in that cylinder, and when
4 the water gets down to the level of the next to highest
5 standpipe, we'll stop the lowering process. We'll read
6 the pressure from the transducer at the base of the
7 next tallest standpipe, compare it with the pressure
8 reading from the transducer of the tallest standpipe, and
9 we'll say the difference in those two pressures is
10 proportional to the amount of decrease that has occurred
11 in the water level.

12 Is this theoretically a correct analog of this
13 system you are describing?

14 BY WITNESS HODGES:

15 A That's very close, yes.

16 Q Okay. Now, before we lower the water level
17 any more, let's address how this information is presented
18 to the operator.

19 It is clear that in my simplified model here
20 were I to start to lower the water level further, in my
21 model the next to highest standpipe would continue to
22 contain the same amount of water.

23 So let's say for the moment I have some way in
24 the control room to read out what are the differences in
25 pressures from these two pressure transducers, the tallest

3-4 1 one, the reference leg, and the next tallest one, which
2 is a lower level.

3 Now I start lowering the water level further,
4 and so far as the reference leg and the second highest
5 standpipe are concerned, nothing is going to change about
6 those readouts.

7 So if I looked at only those two pieces of
8 information, I would not know that anything has happened
9 to the water level, but it is indeed dropping, and the
10 pressure transducer at the base of the second-from-tallest
11 standpipe is going down, and I can compare it with the
12 pressure transducer in the tallest standpipe, look at that
13 difference and say, "Aha, I can ignore standpipe number
14 two, because something is happening in standpipe number
15 three, and therefore, I had better believe that something
16 is happening to the water level."

17 BY WITNESS HODGES:

18 A. That's correct.

19 Q. And so on down through levels as low as I want
20 to go, I presume this process can be carried out.

21 Now, here's the first thing that's bothering
22 me. What happens to the information in the control room
23 such that the operator -- the readouts in the control room
24 such that the operator knows he is looking at the output
25 from the right standpipe?

-5
1 If he continues to focus on number one and
2 number two and the water level starts to drop, he won't
3 know it from that information.

4 Are the outputs from the other standpipes
5 multiplexed into a display system of some sort so that he
6 does not have to sample each channel independently to find
7 out whether anything is happening, or how does the operator
8 know that something is going on there?

9 BY WITNESS HODGES:

10 A Basically, on his narrow range instruments,
11 which would be the first ones to indicate that there's
12 no longer a fall in the water level, the taps are actually
13 located below the level of the range indicated on the
14 instrument.

15 So before you ever got to the point where the
16 instrument itself could not read because of the water
17 level being too low, you've already reached the end of the
18 scale, and the operator is trained not to believe his
19 instrument when it's pegged at the bottom of the scale.

20 He would then go to other instruments in order
21 to get his level indication.

22 Q Does this mean, then -- let me interrupt you
23 here.

24 Does this mean he has to scan several channels
25 of instrumentation until he finds one that has not

9-6 1 bottomed and then says, "Okay, that's where the action is
2 right now"?

3 BY WITNESS HODGES:

4 A He's required to look at more than one level.
5 There's some aspects of the new NUCLENET control system
6 that I'm not that familiar with and they may have some way
7 of multiplexing them in as you say, but at the normal
8 displays on the control panels he would have to scan
9 those.

10 Q He would have to scan them, okay.

11 BY WITNESS HODGES:

12 A Yes. Now, say, on the NUCLENET, and I'm just
13 not that familiar with what they use there. They may
14 have some way of bringing them all up on a display at one
15 time and comparing them there. I'm not certain.

16 Q All right. Well, but on plants where this
17 system is currently in use, the operator does have to
18 scan several --

19 BY WITNESS HODGES:

20 A That's correct.

21 Q -- output channels of information?

22 BY WITNESS HODGES:

23 A They are reasonably close together, but he
24 has to scan more than one. Yes.

25 Q Okay. Now, let's start making this schematic

1 arrangement look more like the actual arrangement.

2 Let's take this tallest standpipe, which we've
3 designated as the reference leg. Let's take it out of
4 that pot of water that the rest of them of varying heights
5 are still in with their pressure transducers.

6 Let's take that one outside and put it off
7 somewhere and we say okay, that's the reference leg. It's
8 not going to change anyway because it represents the highest
9 level the water will ever go, and now we'll start doing
10 things inside the cylinder that holds the rest of the
11 standpipe.

12 Let's start raising the temperature of the
13 water surrounding the standpipes and at a certain point in
14 some of the pipes there may be a two-phase condition.

15 You are lowering the water level slowly, you are
16 raising the temperature of the water that's being lowered;
17 you get a two-phase condition and the pressure transducer
18 at the bottom of the pipe really does not sense a difference
19 in pressure, but because of its being two-phase condition
20 in a particular standpipe, the actual surface level may rise
21 with no change in pressure transducer readout.

22 Now, is this a possibility in the actual
23 application?

24 BY WITNESS HODGES:

25 A Yes, because of the differences in the --

1 Q The density?

2 BY WITNESS HODGES:

3 A -- the density. The standpipes you are talking
4 about have been removed from the vessel, are out in the
5 reactor building now.

6 Q Now, in the actual application, if there is
7 a two-phase condition anywhere, not in my analogy, but in
8 the actual installation, if there is a two-phase condition
9 anywhere within the pressure vessel, is there also a
10 two-phase condition in the standpipe, the reference leg,
11 or is it always totally liquid phase?

12 BY WITNESS HODGES:

13 A That's totally liquid phase.

14 Q Okay.

15 BY WITNESS HODGES:

16 A You say "always." That covers a broad range.

17 If you are depressurizing, say going through a
18 rapid depressurization, and the temperature in the reactor
19 building is, say, 135, 150 degrees, something like that.

20 As you come down through a pressure where you
21 are at the saturation -- get to the saturation temperature
22 of the water, then water in that leg can start to flash and
23 can give you an erroneous reading, also; but under normal
24 operating conditions and most other conditions, you are
25 correct. That is normally single-phase.

9-9 1 Q Okay. I meant to interpose an intermediate
2 step there before getting to the two-phase that just
3 involved change of water density with temperature without
4 the complication of two phases.

5 Now, in the actual installation, not my analog,
6 are there circumstances where there are sufficient
7 temperature differences, forgetting phase change r w,
8 between -- but maybe we can't forget phase change.

9 Are there circumstances where there are
10 sufficient temperature differences between the equipment
11 of the individual standpipes and the reference leg so that
12 there will be a notable error with respect to a determination
13 of where the surface of the water is because of density
14 differences?

15 Let's go ahead and include the two-phase
16 situation, too. Can the operator be significantly misled
17 as to where the water level is because the reference leg
18 may be cooler, may be single-phase, and some of the other
19 standpipes are less dense because of higher temperature,
20 less dense because of two-phase; therefore, the water is
21 actually higher than it appears to be -- is higher than it
22 would appear to be from the transducer reading?

23 BY WITNESS HODGES:

24 A Normally, it's not a two-phase problem. It's
25 primarily a temperature problem.

-10

1 Q Okay, a temperature problem.

2 BY WITNESS HODGES:

3 A And there can be differences in the temperature
4 under abnormal conditions that will cause it.

5 Even on some of the older plants where they
6 use the Yarways, they had the temperature condensation
7 there. The vertical run of the standpipe was inside the
8 containment rather than outside the containment, and so it
9 was subject to whatever the variations in the containment
10 temperature.

11 And they found on those an error, depending
12 on which instrument you were talking about, that could be
13 up to as much as 27 inches.

14 That was the largest error, but yes, the
15 temperature effects can be substantial if the temperature --
16 and the ambient temperature around the instrument itself
17 varies widely.

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0-11 1 Q In the circumstance I have described, it seems
2 to me as though the error would be in a safe direction,
3 namely that because of decreased density in the pipes
4 inside the vessel compared with the reference leg, this
5 may read out a lower water level than actually exists, and
6 this would be an error in a safe direction.

7 Now, do I interpret this correctly?

8 BY WITNESS HODGES:

9 A That's correct. Our concern is when the error
10 goes in the other direction, and that comes with the heating
11 of the reference leg rather than the variable leg.

12 Q Well, okay. Right.

13 Now, before we get to that situation, let's
14 make my analog one step even more realistic.

15 There are not these standpipes inside the
16 RPV. There are pressure taps and there are sloping lines
17 going, I guess, to pressure transducers somewhere; is that
18 correct?

19 BY WITNESS HODGES:

20 A That's correct. There will be a sloping line
21 that will lead outside the containment building, and then
22 you have the vertical standpipes that you're talking about
23 basically leading to the transducers.

24 Q Now, these sloping lines, I guess, can have
25 any kind of slope you want as long as the transducer is a

1 certain elevation below the pressure tap; it's going to
2 read just a pressure equivalent to it --

3 BY WITNESS HODGES:

4 A That's right.

5 Q -- an equal standpipe, so it doesn't make any
6 difference what the slope of these lines is; is that
7 correct?

8 BY WITNESS HODGES:

9 A Basically, that's correct.

10 Q Okay. Now, let's get to the situation you
11 were just talking about where you might have the inverse
12 situation, namely the water in the reference leg being
13 less dense than the water in the measurement legs or, or
14 the other standpipes.

15 First off, how does that come about? Secondly,
16 what is done to compensate for it or prevent its causing
17 an operational error or misunderstanding on the part of
18 the operator?

19 BY WITNESS HODGES:

20 A Okay. The biggest problem, again, is on the
21 older plants that have these Yarway type of instruments
22 which are inside of the containment; and there, basically,
23 what has been done is to change the actuation set points
24 of emergency equipment so that even with the largest
25 expected temperature differences you would actuate the

1 systems. There wouldn't be a problem with actuating the
2 systems.

3 For an Allens Creek type of plant where these
4 are outside the containment itself, you are really
5 restricted to the environment of the reactor building.
6 You are not subject to quite as large swings, but if you
7 got up above the design temperature in the reactor
8 building, it could cause some error. How large would
9 depend on the temperature.

10 Q Well, you say "some error." Is it -- Has it
11 been demonstrated that that error is sufficiently small
12 that the operator doesn't need to worry about it, or must
13 it be compensated for or what?

14 BY WITNESS HODGES:

15 A For very wide swings in temperature inside the
16 containment on these older plants, the errors range from
17 about six inches for a narrow range instrument to twenty-
18 seven inches for a fuel zone ring instrument.

19 So the difference being for the fuel zone
20 instrument you have a much longer leg, standpipe, so to
21 speak, and so you are looking at the effect of that
22 density change over a longer distance.

23 Q In the Allens Creek type of facility
24 installation, where will the reference leg be located?

25 //

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9-14 1 BY WITNESS HODGES:

2 A. It will be outside of the containment.

3 Q. It, too, will be outside of the containment?

4 BY WITNESS HODGES:

5 A. Yes.

6 Q. Now, actually, I don't understand why it
7 isn't, if it's outside of the containment always, why the
8 water leg in the reference standpipe isn't always cooler
9 than water in the legs coming from the pressure taps?

10 BY WITNESS HODGES:

11 A. It is, but that difference in temperature is
12 taken into account in the calibration of the instrument.

13 Q. Okay. Now, we've got to refine the reference
14 leg a little bit, I think, because I've oversimplified it.

15 What is it with respect to the design for
16 an Allens Creek type installation that assures that the --
17 well, the only way I know to say it is that the water level
18 in the reference leg really represents the point within
19 the reactor pressure vessel geometry that you want it to
20 represent?

21 In other words, how do you --

22 BY WITNESS HODGES:

23 A. How do you assure it's filled with water.

24 Q. To the right place.

25 //

9-15 1 BY WITNESS HODGES:

2 A. You start out by filling it with water by
3 filling the reactor vessel all the way to the top and
4 that's when you are shut down.

5 Then as you lower the water level in the
6 reactor to commence operation, you are starting out with
7 a level that's full.

8 There is a condensing pot at the top of this
9 reference leg so if, for example, evaporation or whatever
10 should tend to take water out of the reference leg, the
11 standpipe that you are referring to is at a cooler
12 temperature. The steam from this condensing pot condenses
13 and replenishes whatever might leave and keeps it full.

14 If it would tend to overflow, this condensing
15 pot also would carry it over into the variable leg.

16 Q Carry it over into what?

17 BY WITNESS HODGES:

18 A. The variable leg.

19 Q I'm sorry. I'm missing the significance of
20 that.

21 The standpipe is full when the reactor vessel
22 is full.

23 BY WITNESS HODGES:

24 A. That's right.

25 Q But what would be the cause of an overflow or

9-16 1 an attempt?

2 BY WITNESS HODGES:

3 A. The variable leg is also connected to the
4 condensing pot, but it also would drain to the reactor
5 vessel.

6 So it's not just a simple single tube.

7 Q Okay.

8 JUDGE LINENBERGER: Incidentally, Mr. Sohinki,
9 this long involved questioning process here in a sense
10 illustrates why some diagrams are sometimes helpful.

11 MR. SOHINKI: Well, I think at the time that
12 the Applicant testified with regard to this issue, an
13 attachment to their testimony --

14 JUDGE LINENBERGER: That's right and that
15 really confused me.

16 (Laughter.)

17 MR. SOHINKI: Oh, that's what confused you?

18 JUDGE LINENBERGER: Yes.

19 JUDGE WOLFE: You can't win.

20 (Laughter.)

21 MR. SOHINKI: It may be there's not a diagram
22 in existence that's any clearer than that.

23 JUDGE CHEATUM: I think you'd have to have an
24 animated diagram with moving projections and whatever in
25 order to really illustrate this, because I'm more confused

1 now than when Judge Linenberger began.

2 (Laughter.)

3 WITNESS HODGES: I'm glad we can be so helpful.

4 BY JUDGE LINENBERGER:

5 Q Well, let's talk about an accident, loss of
6 coolant accident situation now, and for whatever reason --
7 and I don't want to get into accident sequences in any
8 detail, but for whatever reason the core starts to become
9 uncovered.

10 The ECCS is activated and I can envision a
11 mixed-up two-phase fog of water droplets and water vapor
12 inside the RPV.

13 Some of these things are flashing against the
14 wall and maybe the water level is -- oh, I see the answer
15 to my question already. I won't waste your time.

16 That's right. These pressure taps lead to
17 lines that really always stay full, even when the water
18 level drops below the orifice so --

19 BY WITNESS HODGES:

20 A That's correct.

21 Q And really the phase condition of the water in
22 these lines attached to the pressure taps is not very
23 directly influenced by temperature and pressure in the RPV
24 because they are sort of moving away from it out to another
25 building.

1 BY WITNESS HODGES:

2 A That's right. You are looking -- particularly
3 on the reference leg, it has very little communication.
4 The variable leg, some water from the vessel moves in there
5 to keep it warmed a little bit more, so there's a little
6 bit more communication, but basically, you are looking at
7 two columns of water essentially free of voids.

8 You are looking at what I call a collapsed
9 water level, a level with no voids in it, and so usually
10 the water level in the core is going to be higher than
11 that because of the presence of the voids.

12 JUDGE LINENBERGER: Okay, thank you very much,
13 gentlemen. I think I understand it this time.

14 JUDGE WOLFE: Cross, Mr. Copeland, on Board
15 questions?

16 MR. COPELAND: I wouldn't dare try, Your Honor.
17 (Laughter.)

18 JUDGE WOLFE: Well, Mr. Doherty.

19 MR. DOHERTY: Here goes.

20 RE-CROSS-EXAMINATION

21 BY MR. DOHERTY:

22 Q Somewhat through this there was a question as
23 to whether the operator could be misled by the water level
24 indicators because, I believe, mainly you said temperature
25 difference between the reference legs and the taps. Do

9-19

1 you follow me so far?

2 BY WITNESS HODGES:

3 A Yes.

4 Q Then I think you said that if there was any
5 misleading it would go in, I think you said, the safe
6 direction?

7 BY WITNESS HODGES:

8 A For the case he was talking about, that's true.

9 Q Are there any cases where it might go the
10 other direction?

11 BY WITNESS HODGES:

12 A When you heat up the variable leg -- excuse
13 me, the reference leg.

14 Q So in that case are you saying that you would
15 get an indication indicating that the water level is
16 actually higher than it is?

17 BY WITNESS HODGES:

18 A That's right.

19 Q In your mind, is it conceivable that a
20 reactor might get in a situation where the operator thinks
21 the fuel is just covered, just barely covered, and in
22 reality it's just not barely covered?

23 BY WITNESS HODGES:

24 A By "just not barely covered," do you mean only
25 a few inches uncovered or --

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9-20 1 Q I mean a few inches uncovered at the top.

2 BY WITNESS HODGES:

3 A It is conceivable that he could have an error
4 of a few inches. I don't know that that's terribly
5 significant.

6 Q And could that, then, lead to the generation of
7 hydrogen?

8 BY WITNESS HODGES:

9 A If it were only a few inches, no.

10 Q Why not?

11 BY WITNESS HODGES:

12 A Because you have very good cooling from the
13 boiling that is going on from the water that it is covered
14 and only very shortly the steam that would be surrounding
15 the fuel rods would be not superheated very much because
16 you'd have to travel a very long distance, and also, you
17 are at a low power end of the rod.

18 Q How far down would that water have to go before
19 you would get some hydrogen, would you say?

20 BY WITNESS HODGES:

21 A The calculation, I think, shows that you have
22 to get down to about the mid-plane of the fuel.

23 Q And the fuel is 12 feet long?

24 BY WITNESS HODGES:

25 A It's 150 inches, twelve-and-a-half feet.

9-21

1 MR. DOHERTY: No further questions. I hope
2 I carried the standard at least a little further.

3 JUDGE WOLFE: Is there redirect, Mr. Sohinki?

4 MR. SOHINKI: Yes. I think I have one or
5 two questions based on what's gone just previously.

6 REDIRECT EXAMINATION

7 BY MR. SOHINKI:

8 Q With regard to this situation where the fuel
9 would be barely covered, I'm not sure -- you may have
10 covered this previously -- but could you explain when the
11 operator would begin to take action in response to an alarm
12 of a low water level? At what point?

13 BY WITNESS HODGES:

14 A The first actions are automatic and the
15 operator is primarily just confirming that automatic
16 actions have occurred, or if it did not occur, then he
17 himself is trying to initiate them.

18 But the first one would come when the water
19 level -- I'm trying to remember the exact elevation. It's
20 at least six feet above the fuel.

21 I don't remember the exact elevation, but it's
22 considerably above the top of the fuel. That's the location
23 where the emergency core cooling systems are turned on.

24 If you get down to a level that's approximately
25 one-and-a-half feet above the top of the fuel and you have

9-22
1 indication that you've got a high containment pressure,
2 which shows you've got a break or you are losing water,
3 also, then your automatic depressurization system would
4 actuate and depressurize the system so that the low pressure
5 systems could fill it up very rapidly.

6 All of this emergency core cooling equipment
7 is actuated well before the water level reaches the top of
8 the fuel.

9 Q And when you say that at six feet above the
10 fuel the ECCS would be activated, that's in comparison
11 with what level above the fuel for normal operation?

12 BY WITNESS HODGES:

13 A It's about 15 feet.

14 Q All right, so --

15 BY WITNESS HODGES:

16 A It may be closer to 18 feet. It's at least
17 15 feet.

18 Q Do I hear you correctly then that there would
19 be several alarms which would alert the operator to a low
20 water level condition before he would get to a situation
21 where the fuel was uncovered at all?

22 BY WITNESS HODGES:

23 A Correct.

24 MR. SOHINKI: No further questions.

25 //

10-1 1 JUDGE WOLFE: Are the witnesses now to be ex-
m 2 cused permanently?

3 MR. SCHINKI: Dr. Huang is. Mr. Hodges, I
4 believe, will be back in the future, although not this
5 week.

6 JUDGE WOLFE: All right. Dr. Huang, you're
7 excused permanently; and we'll expect you back, Mr.
8 Hodges.

9 (The witnesses were excused.)

10 JUDGE WOLFE: All right. What's next?

11 MR. DOHERTY: Your Honor, may we have a short
12 break?

13 JUDGE WOLFE: Yes. But I'd like to know what
14 is next?

15 MR. DEWEY: We have the testimony of Charles
16 M. Ferrell and Leonard Soffer regarding population density
17 projections. That's Bishop Contention 1.

18 JUDGE WOLFE: All right. We'll recess until
19 ten after four.

20 (A short recess was taken.)

21 JUDGE WOLFE: All right.

22 Mr. Dewey.

23 MR. DEWEY: Our witnesses at this time are
24 Charles M. Ferrell and Leonard Soffer. They have not
25 been sworn in. They will testify with respect to Bishop

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Contention 1.

I'd offer them now to be sworn in.

JUDGE WOLFE: Who is who, Mr. Dewey?

MR. DEWEY: Mr. Soffer and Mr. Ferrell.

JUDGE WOLFE: Gentlemen, would you please rise, and raise your right hands.

Whereupon,

CHARLES M. FERRELL

and

LEONARD SOFFER

were called as witnesses and, having been first duly sworn, were examined and testified as follows:

JUDGE WOLFE: Please be seated.

DIRECT EXAMINATION

BY MR. DEWEY:

Q Gentlemen, do you have a document before you entitled "NRC Staff Testimony of Charles M. Ferrall and Leonard Soffer Regarding Population Density Projections"?

BY WITNESS SOFFER:

A Yes, sir.

BY WITNESS FERRELL:

A Yes.

Q Does this document include a statement of your professional qualifications?

/

1 BY WITNESS SOFFER:

2 A Yes, it does.

3 BY WITNESS FERRELL:

4 A Yes, it does.

5 Q Does this document consist of 18 pages?

6 BY WITNESS SOFFER:

7 A Yes, it does.

8 BY WITNESS FERRELL:

9 A Yes.

10 Q Was this document prepared by you or under your
11 direct supervision?

12 BY WITNESS SOFFER:

13 A Yes, it was.

14 BY WITNESS FERRELL:

15 A Yes.

16 Q At this time do you have any changes to make
17 to the document?

18 BY WITNESS FERRELL:

19 A Yes, sir, there's two typographical corrections
20 to make. The first is on Page 2, the last line of the
21 page, the word "by" should be "be."

22 Okay. The second one is on Page 4, the last
23 paragraph, the third line down, there should be a bracket
24 between "area" and "work."

25 Q You're closing the paren after the word, "area"?

1 BY WITNESS FERRELL:

2 A Yes, sir. A paren, yes.

3 JUDGE CHEATUM: Where is the bracket now, Mr.
4 Ferrell?

5 WITNESS FERRELL: It's between the words "area"
6 and "work."

7 JUDGE CHEATUM: The last paragraph?

8 WITNESS FERRELL: Yes, sir.

9 JUDGE CHEATUM: Okay.

10 BY MR. DEWEY:

11 Q Do you have any other changes?

12 BY WITNESS FERRELL:

13 A That's it. That's all.

14 BY WITNESS SOFFER:

15 A No.

16 Q With those changes, do you attest that the
17 testimony that you have prepared is true and accurate to
18 the best of your knowledge and belief?

19 BY WITNESS FERRELL:

20 A Yes, sir, it is.

21 BY WITNESS SOFFER:

22 A Yes, I do.

23 Q All right.

24 MR. DEWEY: Your Honor, I move that the testi-
25 mony of Charles Ferrell and Leonard Soffer be accepted into

1 the record and admitted as evidence as if read.

2 JUDGE WOLFE: Any objection?

3 MR. CULP: No, sir.

4 MR. DOHERTY: Your Honor, I would like to take
5 the witnesses on voir dire.

6 JUDGE WOLFE: All right.

7 VOIR DIRE

8 BY MR. DOHERTY:

9 Q I'm going to ask about the personal qualifica-
10 tions part of the testimony.

11 Mr. Ferrell, did you participate in NUREG-0625
12 at all?

13 BY WITNESS FERRELL:

14 A Let's see. That's the NUREG on population?
15 0625 -- no, sir.

16 No, sir, I did not.

17 Q You did not. Okay.

18 Did you participate at all in Regulatory Guide
19 4.7?

20 BY WITNESS FERRELL:

21 A No, sir, I did not.

22 Q Did you read the Applicant's testimony by Mr.
23 White?

24 BY WITNESS FERRELL:

25 A Yes, sir, I did.

10-5

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10-6

1 Q -- on this.

2 Did you testify in the early Allens Creek hear-
3 ings? There were some held in 1974, I think.

4 BY WITNESS FERRELL:

5 A No, sir, I did not.

6 Q Okay. Mr. Soffer --

7 BY WITNESS SOFFER:

8 A Yes, sir.

9 Q -- did you participate in the writing of the
10 SER Supplement No. 2 (I guess it is) for the Allens Creek
11 Nuclear Plant?

12 BY WITNESS SOFFER:

13 A Yes, it was prepared under my supervision.

14 Q That particular part, Section 2.2, was that --

15 BY WITNESS SOFFER:

16 A Yes, I believe so.

17 Q Okay. Did you participate in NUREG-0625 in
18 any way?

19 BY WITNESS SOFFER:

20 A Yes, I was a member of the working group.

21 Q Okay. And did you participate in Regulatory
22 Guide 4.7?

23 BY WITNESS SOFFER:

24 A No, sir, I did not.

25 Q Have you read the testimony of the Applicant's

1 witness, which was filed, I think, earlier this year by
2 Mr. White on this issue?

3 BY WITNESS SOFFER:

4 A I have read the written testimony, yes.

5 Q Did you participate, by any chance, in the
6 earlier hearings on Allens Creek?

7 BY WITNESS SOFFER:

8 A I believe I was a member of a panel on site
9 suitability a number of years ago. I can't recall pre-
10 cisely, however.

11 Q Uh-huh. Did you present testimony then?

12 BY WITNESS SOFFER:

13 A I may have. My memory fails me.

14 Q I think you mentioned being in Greece on a siting
15 mission.

16 BY WITNESS SOFFER:

17 A Yes, sir.

18 Q Was that as a U. S. Government employee?

19 BY WITNESS SOFFER:

20 A Yes. I was -- It was as a representative
21 of the IAEA, that's the International Atomic Energy Agency.

22 Q I see.

23 BY WITNESS SOFFER:

24 A My expenses were paid by IAEA. However, my
25 time was donated by the NRC.

10-8 1 Q I see. You stated at the bottom that you've
2 written about 12 technical papers on various topics related
3 to radiological safety aspects. Were any of these also
4 related to population?

5 BY WITNESS SOFFER:

6 A No, they were not.

7 Q Were they with regard to the radiological con-
8 sequences of accidents to population?

9 BY WITNESS SOFFER:

10 A Some of them would have involved shielding
11 activating studies, computer codes involving shielding.
12 There was -- I would like to amend my earlier remark.
13 I was a co-author on a NUREG involving demographic
14 statistics involving nuclear power plants.

15 That's NUREG-0348, which is a compilation of
16 population data and statistics.

17 Q Okay. This is just a general question. What
18 has been the general accuracy of predictions of population
19 within five miles of a plant, when you compare -- oh,
20 the early draft -- or early final construction license
21 environmental statements with the operating license
22 statements?

23 MR. DEWEY: Mr. Chairman, I'm going to object
24 to that question. It seems to me that's beyond voir
25 dire. He's asking cross-examination questions.

10-9

1 MR. DOHERTY: I think it's a general knowledge
2 question. I don't think I can ask it very well in the
3 cross-examination part. I don't -- think it's going to
4 not have relevance to the specific question here, but it
5 does have relevance to his knowledge as a population
6 expert.

7 (Bench conference.)

8 JUDGE WOLFE: Objection overruled. We'll hear
9 it.

10 WITNESS SOFFER: I don't believe there has ever
11 been a systematic study made of that. And my belief
12 is -- and at this point it's just a general feeling based
13 upon my knowledge -- that the results would be quite a
14 mixed bag.

15 MR. DOHERTY: Okay, no further questions, Your
16 Honor; and no objections.

17 JUDGE WOLFE: All right. Absent objections,
18 the testimony of Messrs. Ferrell and Soffer regarding
19 population density projections, inclusive of their
20 professional qualifications, are incorporated into the
21 record as if read.

22 (See attached pages.)

23 /
24 /
25 /

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY LICENSING BOARD

In the Matter of)	
HOUSTON LIGHTING AND POWER COMPANY)	Docket No. 50-466
(Allens Creek Nuclear Generating)	
Station, Unit 1))	

NRC STAFF TESTIMONY OF CHARLES M. FERRELL
AND LEONARD SOFFER REGARDING POPULATION DENSITY PROJECTIONS

[Bishop Contention 1]

Q. Please state your names and positions with the NRC.

A. My name is Charles M. Ferrell. I am a site analyst in the Siting Analysis Branch, Division of Engineering. My name is Leonard Soffer. I am a Section Leader in the Siting Analysis Branch, Division of Engineering. Copies of our professional qualifications statements are attached to this testimony.

Q. What is the purpose of this testimony?

A. The purpose of this testimony is to respond to Bishop Contention 1 which states:

The projected population density within a 50 mile radius of the proposed nuclear plant at Allens Creek is greater than the applicant estimates and exceeds criteria set by the Nuclear Regulatory Commission.

Q. In general, what will you attempt to show through this testimony?

A. This testimony will discuss the applicable NRC criteria, which are 10 C.F.R. Part 100 and Regulatory Guide 4.7, and will show that the present population densities meet these criteria, and that the projected population densities are expected to meet these criteria over the lifetime of the plant. The testimony will also present the bases for the staff's conclusion that the applicant has made reasonable projections of the population in the vicinity of the Allens Creek site.

1. NRC Siting Criteria

Q. What are the NRC siting criteria?

A. The Commission's criteria for determining the suitability of proposed sites for nuclear power plants are contained in 10 C.F.R. Part 100. Proposed sites are required to meet certain tests related to the surrounding population. The objective is to assure that the potential consequences of postulated accidents do not pose an undue risk to the health and safety of the public.

Q. What does 10 C.F.R. Part 100 require with respect to population criteria around a proposed site?

A. 10 C.F.R. Part 100 requires that in selecting the site for a proposed nuclear power plant that an exclusion area, low population zone and nearest population center ~~be~~^{be} defined and selected.

Part 100 also requires that the distance from reactor to the nearest population center^{1/} be at least one and one-third times the low population zone outer radius and, in addition, that the radiological consequences of an assumed hypothetical fission product release meet certain dose guidelines to an individual located at the boundaries of the exclusion area and low population zone. It should be noted that Part 100 contains no specific requirement relating to population density near a proposed site. The regulation does state that, with respect to the one and one-third rule, where very large cities are involved, a greater distance may be necessary. In the statement of considerations that led to Part 100, the Commission enunciated the policy that power reactors should be located away from densely populated centers, and stated that the population center distance criterion was added as a site requirement in order to provide for protection against excessive exposure doses to people in large centers, where effective protective measures might not be feasible. The Commission, however, issued no specific requirements on population density near a proposed site.

^{1/} 10 C.F.R. 100.3(c) defines a population center distance as the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents. Section 100.11(a) indicates that the boundary of the population center should be determined upon considerations of population distribution and that political boundaries are not controlling.

Q. In the absence of specific Commission requirements on population density, has the staff established any population density criteria to act as guidance to applicants?

A. Criteria on population density have been published in USNRC Regulatory Guide . (Revision 1, November 1975), "General Site Suitability Criteria for Nuclear Power Stations." These criteria, which are not part of the Commission's regulations but which do offer guidance on staff review practices, state with respect to population considerations the following:

"Areas of low population density are preferred for nuclear power station sites. High population densities projected for any time during the lifetime of a station are considered during both the NRC staff review and the public hearing phases of the licensing process. If the population density at the proposed site is not acceptably low, then the applicant will be required to give special attention to alternative sites with lower population densities."

"If the population density, including weighted transient population, projected at the time of initial operation of a nuclear power station exceeds 500 persons per square mile averaged over any radial distance out to 30 miles (cumulative population at a distance divided by the area at that distance), or the projected population density over the lifetime of the facility exceeds 1000 persons per square mile averaged over any radial distance out to 30 miles, special attention should be given to the consideration of alternative sites with lower population densities."

"Transient population should be included for those sites where a significant number of people (other than those just passing through the area) work, reside part time, or engage in recreational activities and are not permanent residents of the area. The transient population should be taken into account by weighing the transient population according to the fraction of time the transients are in the area."

- Q. In general, why were these population density values selected?
- A. The population density values were selected on the bases of allowing a good degree of site availability in all regions of the U.S., including the North-eastern U.S., while simultaneously implementing the Commission's policy that power reactors should be located away from densely populated centers.

For sites with population densities below these guidelines, it was considered unlikely that numbers of substantially better sites (from a population density standpoint) would be found in the north-east, reasonably near load centers. The population density at distances greater than 30 miles from a potential site was considered to have relatively little impact on siting. Supporting results of this view can be found in the Reactor Safety Study (WASH-1400) which indicate that, even in the event of a large accidental release of radioactivity, the consequences to the public would be expected to be low at distances greater than about 20 to 30 miles.

It should be pointed out that the population density levels mentioned do not represent upper bound limits of acceptability, but are merely "trip" levels which if exceeded, a site must be determined to have significant offsetting advantages as compared with available alternate sites of lower density.

- Q. Are any NRC efforts underway to establish new siting criteria?
- A. 10 C.F.R. Part 100 and Reg. Guide 4.7 are presently the only NRC siting criteria with regard to population. The Siting Policy Task Force, in its report (NUREG-0625) gave a numerical example merely to illustrate the concept. The examples in NUREG-0625 are not criteria, nor even proposed criteria. Although the Commission has announced its intention of revising 10 C.F.R. Part 100 (45 Fed. Reg. 50350) to incorporate population density and distribution limits, staff efforts are still underway in this area, and no new proposed criteria have been issued.

2. Compliance of the Allens Creek Site with 10 C.F.R. Part 100

- Q. What has the applicant concluded regarding the compliance of the Allens Creek site with the requirements of 10 C.F.R. Part 100 and the guidelines set forth in Regulatory Guide 4.7?
- A. The applicant has presented information in the PSAR and ER on the site including a discussion of the exclusion area, low population zone (LPZ), population center distance, nearest population center and has also presented information on the present as well as projected population in the site vicinity out to 50 miles. This information has also included a discussion of the methodology and sources used to develop the population projections as well as a discussion of the transient population in the site vicinity. The

applicant concluded, based upon the information obtained and submitted, that the site met the requirements of 10 C.F.R. Part 100, and was below the "trip" levels of Regulatory Guide 4.7, as well.

- Q. What has the staff previously concluded with respect to the site meeting the population requirements of 10 C.F.R. Part 100?
- A. The staff has independently evaluated the compliance of the site with respect to 10 C.F.R. Part 100, and the staff reported its conclusions in the original Safety Evaluation Report (SER) issued November 1974, as well as in Supplements 1 and 2, issued June, 1975 and March 1979, respectively. The SER and its Supplements noted that the site has an exclusion area, that the minimum distance from the plant to the exclusion area boundary is 4330 feet (1320 meters), that the LPZ outer radius is 3.5 miles and that nearest population center has been designated to be the city of Rosenberg located about 20 miles southeast of the site. The population center distance is at least one and one-third times the LPZ outer radius, as required by 10 C.F.R. Part 100.

The staff concluded, in the SER and in SER Supplement No. 1, that the site met the criteria of 10 C.F.R. Part 100. This Licensing Board also found in its Partial Initial Decision dated November 11, 1975, (LBP-75-66, 2 NRC 776 at 797) that the site met the criteria of 10 C.F.R. Part 100.

- Q. Since these previous evaluations and conclusions as noted in the SER and Supplement No.1, have the applicant and staff reevaluated population and population distribution in the vicinity of the site in light of more recent population data?
- A. Yes. As a result of delay in the application as well as a change from a two-unit to a single unit application, the applicant resubmitted population as well as other pertinent data in 1977. The staff reported its findings in SER Supplement No. 2 issued March 1979.

As the staff noted in SER Supplement No. 2, "Because high growth rates have been reported in areas east of the site, we reevaluated populations and population distribution in the vicinity of the site to determine whether our conclusions were still valid."

The staff, in the same SER Supplement 2, noted that Ft. Bend and Harris Counties had shown population increases of 53 percent and 17 percent, respectively, in the period from 1970 to 1976. The staff thereupon reviewed the designation of Rosenberg as the nearest population center. The potential growth of communities located closer to the site than Rosenberg was evaluated. These included the town of Katy, located about 19 miles east-northeast of the Allens Creek site, as well as the town of Sealy, located about 7 miles north-northwest. SER Supplement 2 noted that the 1978 estimated population for Katy was 5000 persons and the 1975 estimated

population for Sealy was 3211 persons. The staff noted, in view of the relatively low population of these communities compared with the value of about 25,000 persons defined in 10 C.F.R. Part 100 for the designation of a population center, that very rapid growth of these communities would be required within the lifetime of the plant before either could be considered as the nearest population center. The Staff further noted, based upon population projections at that time, that the possibility of either Sealy or Katy becoming population centers, as defined by 10 C.F.R. Part 100, could not be ruled out, although the staff considered it unlikely that Sealy would become the nearest population center during the plant lifetime. However, the staff concluded that even in the event Sealy or Katy become the nearest population center, the population center distance would still be greater than one and one-third times the LPZ outer radius. Therefore, the staff concluded, in SER Supplement 2, that the present exclusion area and present LPZ conform to the requirements of 10 C.F.R. Part 100 regardless of whether Rosenberg, Katy or Sealy is the nearest population center.

- Q. Has the staff reexamined its conclusion in the SER Supplement 2 as the result of the publication of the preliminary 1980 Census data?
- A. Since the publishing of SER Supplement 2, preliminary results of the 1980 Census have become available. The 1980 populations for the towns of Rosenberg, Katy and Sealy are shown in the table below.

<u>Town</u>	<u>Population (1980 Census)</u>
Rosenberg	17,707
Katy	5,677
Sealy	3,888

The staff, after evaluating this recent information concludes that our evaluation and conclusion reported in SER Supplement 2 remains unchanged and that the nearest population center is considered to be the city of Rosenberg, based upon its expected growth within the lifetime of the plant. The staff reaffirms, based upon data from the 1980 Census, that the exclusion area, low population zone and population center distance meet the requirements of 10 C.F.R. Part 100.

3. Comparison of Present and Projected Population Densities with Regulatory Guide 4.7

- q. What has the staff examined to determine if the applicant made reasonable population projections and whether present and projected densities will exceed the "trip" levels of Regulatory Guide 4.7?
- A. The staff has examined the applicant's population data, including population projection sources and methodology and has independently made assessments aimed at comparing the present and projected population densities around the Allens Creek site with the "trip" levels of Regulatory Guide 4.7, and determining whether the applicant has made reasonable population projections.

Efforts by the staff included the following:

- (1) The 1970 Census data in the Allens Creek site vicinity was independently confirmed by the staff using its own copy of a 1970 census computer tape.
- (2) The staff, using 1980 preliminary Census data, has prepared estimates of the 1980 population and population densities within 5, 10, 20 and 30 miles of the Allens Creek site.
- (3) The staff has assessed the population projections used by the applicant both with regard to the sources of data and the use of methodology, and has also compared the applicant's projections with those obtained from independent sources.

Q. What were the results of the staff's independent confirmation of population in the site vicinity based on the 1970 census?

A. Population data in the vicinity of the Allens Creek site based upon the 1970 census was prepared by the applicant and has been presented in the PSAR and ER. The staff, making use of its own computer program employing a copy of the 1970 census tape, has independently confirmed that the data presented by the applicant is reasonable. Table 1 presents the 1970 cumulative population and population densities in the Allens Creek site vicinity made both by applicant and staff.

TABLE 1-1970 POPULATION AROUND ALLENS CREEK

Distance, Miles	Applicant		Staff	
	<u>Population</u>	<u>Pop. Density</u> (people/mi ²)	<u>Population</u>	<u>Pop. Density</u> (people/mi ²)
0-5	1,844	23	2,471	31
0-10	7,999	25	7,327	23
0-20	34,000	27	35,647	28
0-30	94,000	33	97,662	35

It can be seen from a comparison of applicant's and staff's values that the agreement is very good (within 10%), except for the 0 to 5 mile distance, where the values differ by about 35%. This is explained by the fact that the applicant used an actual house count within this distance, while the staff computer program uses a technique which counts all of a census tract as being included whenever the center of the tract is within the circle in question. The staff has noted this phenomenon many times, and considers an actual house count to be more reliable at relatively close-in distances. It should also be noted that the 1970 population densities are well below the trip levels of Regulatory Guide 4.7.

- Q. How does the preliminary population data obtained from the 1980 Census compare with applicant's projections for the year 1980?
- A. Population data in the vicinity of the Allens Creek site for the year 1980 was projected by the applicant and has been presented in

testimony before the Licensing Board by W. T. White (following Tr. 8910). Preliminary data from the 1980 census for Texas Counties and county subdivisions has recently become available. The staff has used this data to estimate the 1980 population within the vicinity of the site by allocating the same fraction of population as that fraction area of a county or county subdivision lying within a given circle. Table 2 presents the 1980 cumulative population and population density around the site made by both applicant and staff.

TABLE 2-1980 POPULATION AROUND ALLENS CREEK

Distance, Miles	Applicant*		Staff**	
	<u>Population</u>	<u>Pop. Density</u> (people/mi ²)	<u>Population</u>	<u>Pop. Density</u> (people/mi ²)
0-5	2,260	29	2,545	32
0-10	11,120	35	10,156	32
0-20	46,830	37	56,828	45
0-30	198,630	70	216,037	76

*From testimony of W.T. White using Rice/Dames & Moore Projections

**Based upon 1980 Census Preliminary Report - PHC80-P-4b, Texas

A comparison of applicant's and staff's 1980 values indicate very good agreement within 10 miles of the site. Beyond 10 miles the staff estimates somewhat higher values than the applicant although we judge the overall agreement to be good. (The staff is about 20% higher at 20 miles, and about 10% higher at 30 miles.) It should

also be noted that the 1980 population densities are well below the trip levels of Regulatory Guide 4.7.

A comparison of the data of Tables 1 and 2 indicates that the major population growth around the Allens Creek site from 1970 to 1980 has occurred at distances of about 20 miles and beyond. Population growth within 10 miles of the site for this period was about 30 percent, while within 20 and 30 miles, the growth rates were about 65 percent and 130 percent, respectively.

- Q. In the staff's assessment of the reasonableness of the applicant's population projections, did the staff review the applicant's sources of data?
- A. Yes. The applicant's original population projections presented in the PSAR were based upon the 1972 study for the Houston-Galveston Area Council (HGAC). In 1977, the applicant provided revised projections based generally upon projections for Texas Counties made by the Texas Water Development Board (TDWB). Finally, in 1980 the applicant provided revised projections prepared originally by the Rice Center for the Houston-Galveston region and subsequently modified by James and Moore.

The staff notes that the sources used by the applicant are governmental groups or private institutions which are independent of the applicant. Such groups are typically interested in

examining future population growth for a variety of reasons and routinely prepare population projections making use of regional economic activity, observed growth patterns, transportation networks and other data considered appropriate and incorporating these by means of a suitable methodology. We further note that two of these projections concentrated on growth in the Houston area which is expected to be the major factor influencing future population growth in the vicinity of the site.

Q. Did the staff compare applicant's sources of data with those obtained from independent sources?

A. Yes. In the staff's SER dated November 1974, we compared the applicant's projections (at that time based upon the 1972 HGAC study) with independent projections made by the U.S. Department of Commerce, Bureau of Economic Analysis (BEA), for BEA Area No. 141, a 17 county area including the Houston-Galveston area and surrounding counties. As we noted in the SER, on page 2-8:

"The applicant projects population increases of about 122% and 208%, by the years 2000 and 2020 respectively, for the region within 50 miles of the plant. The BEA projects population increases of 79% and 154% by the year 2000 and 2020, respectively, for BEA Area No. 141. We find the applicant's population projections to be in reasonable agreement with those of the Bureau of Economic Analysis (BEA)."

Since our original comparison of the applicant population projections was made in 1974, the staff has obtained more recent projections made by the Texas Department of Water Resources (TDWR) which were published in January, 1980 as Report No. LP-126. The

report contains projections for all Texas Counties up to the year 2000. In addition, unpublished projections for the year 2020 for a number of counties of interest were obtained by telephone from a representative of the TDWR. We compared the projected increase in total population in all Texas counties within 50 miles of the Allens Creek site with the revised population projections presented by the applicant's witness W. T. White in his recent testimony before the Board and using the population projections labeled Rice/Dames and Moore.

Q. How do these sources of data compare?

A. The applicant's revised population projections forecast increases of about 55% and 79%, by the years 2000 and 2020 respectively, for the region within 50 miles of the proposed plant. Using data from the January 1980 report issued by the TDWR, the population for all Texas counties within 50 miles of the site is projected to increase by 57% and 116% by the years 2000 and 2020, respectively. We find that the applicant's most recent population projections are in reasonable agreement with recent projections made by an independent source.

Q. What does the staff conclude with respect to the reasonableness of the applicant's population projections?

A. After an assessment of the applicant's population projections, the staff has determined that:

- (1) They are based upon sources of data and studies from groups that are independent of the applicant, that such groups customarily prepare population projections for a variety of business and government users, and that these groups employ models using regionally applicable data and appropriate methodologies, and
- (2) A comparison of the applicant's projections with those from completely independent sources indicates there is reasonable agreement between the two.

The staff, therefore, concludes that the applicant has made reasonable projections of the population in the vicinity of the Allens Creek site.

- Q. How do the applicant's population projections compare with the "trip" levels of Regulatory Guide 4.7?
- A. In the staff's comparison of the applicant's most recent population projections, we assumed 1990 to be the estimated beginning of plant life and 2030 to be the end of plant life. The staff extrapolated the applicant's projections from year 2020 to 2030 by assuming the same growth rate. The cumulative population and population densities are shown in Table 3.

TABLE 3
Population Projections for Allens Creek Site

Distance, Miles	<u>1990</u>		<u>2030</u>	
	Population	Density (people/mi ²)	Population	Density (people/mi ²)
0-5	3,630	46.	5,500	70.
0-10	18,060	57.	31,200	99.
0-20	71,030	57.	109,180	87.
0-30	311,130	110.	519,520	184.

From Table 3, it can be seen that the projected population density is well below the "trip" level of 500 persons per square mile in 1990, and also well below the value of 1000 persons per square mile at estimated end of plant life, in 2030.

We conclude that the present and projected population densities are well below the trip levels of Regulatory Guide 4.7.

- Q. What is your overall conclusion regarding this testimony?
- A. On the bases of the above testimony, the staff concludes that the applicant has made reasonable projections of the population in the vicinity of the Allens Creek site, and that the site meets the siting criteria set by the NRC.

CHARLES M. FERRELL

PROFESSIONAL QUALIFICATIONS

SITING ANALYSIS BRANCH

DIVISION OF ENGINEERING

I am a site analyst in the Siting Analysis Branch, Division of Engineering, U.S. Nuclear Regulatory Commission. My present duties in this position include the evaluation of site related environmental safety aspects of nuclear power generating facilities and design basis accident analysis. I graduated from Salem College in West Virginia in 1950 with a B.S. degree in physics and a teaching field in chemistry, biology, and mathematics. Upon graduation, I was drafted, and after completion of armored infantry training at Fort Knox, Kentucky, was assigned as a military physicist to the Radiological Division of the U.S. Army Chemical Corps at Edgewood, Maryland. I spent approximately two years in research involving nuclear weapon thermal radiation, nuclear radiation shielding studies and fallout analysis. I was released from active duty and worked for two years as a civilian physicist in Aerosol Physics (Aerobiology) Research at the U.S. Army Chemical Corps Biological Warfare Laboratory at Fort Detrick, Frederick, Maryland. In 1954, I applied for and was granted an AEC Fellowship in Radiological Physics at Vanderbilt University and the Oak Ridge National Laboratory in Tennessee. An additional year of graduate work in physics as taken at West Virginia University. Night school classes in Nuclear Engineering from the University of Maryland plus short summer courses from MIT in Air Pollution, Heat Transfer, and Nuclear Power Reactor Safety constitute the remainder of my formal education. In April, 1974, I completed a two week course in Pressurized Water Reactor Systems at the Westinghouse Training Center in Monroeville, Pennsylvania. I am a charter member of the Health Physics Society.

I have been a member of the AEC's (now NRC's) Regulatory Staff since 1956. Of these twenty-four years, five years were spent in duties involving the safe industrial and medical use of radioisotopes, in the evaluation of spent reactor fuel shipping casks and the promulgation of reactor fuel shipping regulations. Eight years were served as the Technical Assistant to the Office of Hearing Examiners, U.S. Atomic Energy Commission in which I assisted in approximately 40 hearings on nuclear power reactors, fuel reprocessing plants, and in addition contract appeals hearings on nuclear submarine components and nuclear equipment.

In January, 1969, I transferred to my present position. Since that time I have served as the site analyst on forty two nuclear power plants, two U.S. Navy nuclear submarine reactors and a proposed nuclear powered crude oil tanker. I served as one of the technical reviewers of Chapter 7, "Assessment of Reactor Safeguards" in Applied Radiation Protection and Control by J. J. Fitzgerald, published under the auspices of the Division of Technical Information United States Atomic Energy Commission. I am one of the co-authors of the report "Demographic Statistics Pertaining to Nuclear Power Reactor Sites" NUREG-0348, and the report "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612, published by the U.S. Nuclear Regulatory Commission.

I have testified in licensing hearings on six nuclear facilities. These include San Onofre 2/3, Beaver Valley Unit 1, Hutchinson Island (now St. Lucie 1), Yellow Creek 1 and 2, Duane Arnold Unit 1 and Trojan Unit 1.

LEONARD SOFFER
PROFESSIONAL QUALIFICATIONS
SITING ANALYSIS BRANCH
DIVISION OF ENGINEERING
OFFICE OF NUCLEAR REACTOR REGULATION

I am Section Leader of the Site Analysis Section, Siting Analysis Branch, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. My duties in this position include responsibility for the review and evaluation of the population characteristics of nuclear power reactor sites as well as the evaluation of potential hazards posed by nearby man-related activities.

I received a B. S. Degree (with honors) in Physics from the City College of New York in 1952 and attended graduate school at Case Western Reserve University in Cleveland, Ohio.

Before joining the Commission, I was employed for 21 years as a Physicist and Nuclear Engineer with the National Aeronautics and Space Administration (NASA) at the Lewis Research Center in Cleveland, Ohio. In this capacity, I performed analyses on radiation shielding and nuclear safety requirements for nuclear power systems intended for lunar and space applications. I assisted in the radiation shielding design of the NASA Plum Brook reactor, served on an agency-wide study team investigating the radiological safety aspects of using radioisotopes for space power generation, and was section leader of a group responsible for research on radiation shielding and radiological safety concerns. I also monitored contracts and occasionally lectured on radiological physics and shielding to others within NASA.

I joined the Commission staff in July 1973, and have participated in the detailed review of over 20 nuclear power plants. My responsibilities in this regard have included evaluation of the demographic characteristics and nearby facilities of sites as well as the independent assessment of the likelihood and consequences of various postulated accidents. I have prepared and presented testimony at hearings on the population density and use characteristics of sites as well as the radiological consequences of accidents. In my capacity as Section Leader, Siting Analysis branch, I am responsible for reviewing the results of similar efforts by others.

Pertinent experience has also included participation in development of a draft standard entitled "Guidelines for Estimating Present and Forecasting Future Population Distributions Surrounding Power Reactor Sites", membership in the NRC Working Group that wrote the "Report of the Siting Policy Task Force" (NUREG-0625), and membership in a Siting Mission to Greece, to assist that Government in the development of demographic criteria for nuclear power plants.

I have also lectured on accident consequence assessment at several courses sponsored by the IAEA, have attended conferences devoted to population projection methodology for small geographic areas and have had discussions with expert demographers on this subject.

I have written about 12 technical papers on various topics related to radiological safety aspects of nuclear reactors. I am a member of the American Nuclear Society and the Population Association of America, which is the professional society of U. S. demographers.

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JUDGE WOLFE: Are the witnesses ready for cross, Mr. Dewey?

MR. DEWEY: Yes, sir.

JUDGE WOLFE: Mr. Culp?

MR. CULP: Applicant has no cross-examination.

JUDGE WOLFE: Mr. Doherty.

CROSS-EXAMINATION

BY MR. DOHERTY:

Q On Page 3 in a discussion of siting criteria, you state, "It should be noted that Part 100 contains no specific requirement relating to population density"

Then you state, "The regulation does state that, with respect to the one and one-third rule, where very large cities are involved, a greater distance may be necessary."

Is that -- I presume that's a paraphrase of the rule. But is it paraphrased with regard to the words, "may be necessary," or is that literally what it says?

BY WITNESS SOFFER:

A That's what the regulation says.

Q Uses just those exact words, "may be necessary"?

BY WITNESS SOFFER:

A I don't have the exact words of the regulation in front of me. But I believe that is, indeed, what the

1 regulation says.

2 If you'll allow me, I can get it.

3 Q All right.

4 (Pause.)

5 BY WITNESS SOFFER:

6 A Yes, sir, the regulation does state, "may be
7 necessary," literally.

8 Q Is there any interpretive history of that,
9 in terms of siting? Is that -- "may be necessary," how
10 has that been worked out? Has it just been a very --
11 I mean obviously we'll not set it on Staten Island.

12 But, you know, beyond that ... is there any
13 sort of a loose --

14 BY WITNESS SOFFER:

15 A There has been a long interpretive history
16 in that beginning about the 1960's and through about the
17 late 1960's, there evolved a general Staff policy that
18 sites having populations -- cumulative populations
19 greater than, for example, the Indian Point and Zion
20 sites were not suitable for nuclear power plants.

21 Beginning in the early 1970's, there began to
22 be some feeling on the part of the Staff that perhaps
23 there ought to be some kind of a trigger mechanism -- or
24 a trip mechanism (if you will) that looks at sites at a
25 still lower level, with the intent of asking an applicant

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to justify sites beyond that population even more.

This eventually culminated in the publication of Reg Guide 4.7 in October 1975 where the present guidelines are used, not as upper limits of acceptability, but basically as trigger levels, or trip levels, as I have referred to them -- which are intended to trigger an additional level of review with regard to alternative sites.

That's a brief rundown of the history of how this has been interpreted.

- - -

1 BY MR. DOHERTY:

2 Q Thank you. Do you have a copy of NUREG-0625
3 with you, by any chance -- either of you?

4 BY WITNESS SOFFER:

5 A I do, yes.

6 Q Perhaps you could turn to Page 46. There's
7 a recommendation section there beginning on Page 46.
8 And what I'm wondering is what is your understanding as
9 to whose recommendation -- who we a these recommendations
10 to?

11 BY WITNESS SOFFER:

12 A These were recommendations that were made
13 by an NRC Staff Task Force for upper NRC management and
14 for attention to the Commission as well.

15 Q I see. But nothing has happened with these
16 since the recommendation; is that correct?

17 BY WITNESS SOFFER:

18 A No, that's not true. The Commission has
19 issued an advance notice of proposed rulemaking indicating
20 that it intended to begin revising the siting criteria.

21 However, there has been no proposed criteria
22 issued at the present time. The Staff is still studying
23 the matter and is still considering possible changes.

24 Q I see. I see one of the recommendations
25 says, "Incorporate specific population density and

1 distribution limits outside the exclusion area that are
2 dependent on the average population of the region."

3 Would that tend to take over the language I was
4 asking you about earlier -- the "may be considered" type
5 language? Would you consider that recommendation and sort
6 of make firm that language?

7 BY WITNESS SOFFER:

8 A. Well, if that recommendation is incorporated
9 into the regulation, then obviously that would take pre-
10 cedence, yes.

11 Q. I see.

12 On Recommendation 4 it says, "Remove the re-
13 quirement to calculate radiation doses as a means of
14 establishing minimum exclusion distances at low population
15 zones."

16 If that's removed, what would take its place?
17 Anything?

18 BY WITNESS SOFFER:

19 A. The Task Force envisioned that there would
20 be, first of all, a specification of a minimum exclusion
21 distance. And at the same time there would be a require-
22 ment in reactor designs that there would be at least a
23 minimum complement of engineer safety features.

24 And with the accomplishment of a -- specifying
25 a minimum standard list of engineer and safety features,

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1 plus a minimum exclusion distance, that would accomplish
2 the same purpose.

3 Q Okay. I notice on Page 6 you speak in the
4 second -- the last answer on Page 6 -- about information
5 in the PSAR and ER ... the population center distance;
6 you mention that.

7 What is that distance? Do you recall, or do
8 you recall a place?

9 BY WITNESS SOFFER:

10 A I'm sorry, I -- the population center distance
11 is approximately 20 miles.

12 Q I see.

13 What's the significance of this population
14 center? You know, it seems like you find it. Is it just
15 sort of -- What is it? I mean ...

16 BY WITNESS SOFFER:

17 A The only thing that I can say is that it was
18 incorporated into Part 100 at the time by the Commission
19 as an additional requirement because -- and we indicated
20 the reasoning that was listed in the statement of con-
21 siderations at the time, since accidents of greater
22 consequence than might be hypothesized ... the Commission
23 indicated that it would be desirable to place population
24 centers at somewhat greater distances than merely just
25 outside the plant boundaries.

1 And they settled this on one-and-a-third
2 distance times the LPZ.

3 Q There must be something that defines what a
4 population center is or is not. It's -- You're saying
5 it has to be a certain distance from the reactor site.
6 That's clear.

7 BY WITNESS SOFFER:

8 A The regulation says that it shall be densely
9 populated, and that it shall be of about 25,000 residents
10 or more, and that political boundaries are not con-
11 trolling.

12 Q It's about 25,000 persons. Does it talk
13 about -- Does the regulation talk about 25,000 at the
14 start of a plant's operation, or does it take into account
15 predictions or --

16 MR. DEWEY: Your Honor, it seems to me that
17 he's asking directly what the regulation says. And if
18 he wanted to get that information, he could read it
19 himself, rather than asking the witnesses.

20 MR. CULP: And, moreover, on Page 3 in the
21 Footnote No. 1, the witness describes what's in the
22 regulations.

23 JUDGE WOLFE: Well, that question has already
24 been gone beyond now: Asked and answered. The question
25 put to the witness now is: Is this population center

10-17

1 distance calculated on the basis of present inhabitants
2 or future inhabitants as projected.

3 Is that indicated in the regulation, Part 100,
4 Mr. Soffer?

5 WITNESS SOFFER: That is not specifically
6 indicated in the regulation.

7 JUDGE WOLFE: What deference or what con-
8 sideration is given to that, if at all, in fixing the
9 population center distance?

10 WITNESS SOFFER: There is a precedent. The
11 statement of consideration that was published in the
12 FEDERAL REGISTER at the time Part 100 was promulgated
13 does say that the Commission will give consideration to
14 the extent possible, and will review future land uses.

15 The Staff has interpreted that to mean that
16 population centers should be defined not only on the
17 basis of present population, but that projections of
18 population should be made to the extent possible, and
19 that where there is an indication that an area will meet
20 the definition reasonably in the life of the plant, then
21 it should be designated as a population -- as the
22 population center, excuse me.

23 And the Staff has generally followed that
24 practice.

25 JUDGE WOLFE: I'm sorry, Mr. Doherty, I didn't

1 mean to take the questioning away from you.

2 MR. DOHERTY: Thanks. That's all right.

3 BY MR. DOHERTY:

4 Q Now, in the event the population center is
5 predicted to -- excuse me.

6 It has to be the nearest -- the nearest popula-
7 tion center with 25,000 residents, apparently either at
8 the time of operating in the beginning or predicted; is
9 that a fair summary?

10 BY WITNESS SOFFER:

11 A Yes, that's right.

12 Q And what significance would it be if a popula-
13 tion center turned out from predictions to appear to
14 have -- excuse me.

15 All right. Let me put it this way; let's just
16 make it as concrete as I can.

17 I believe at this moment -- this is just a
18 belief -- that there are predictions that a town slightly
19 closer to the plant has predictions of greater than
20 25,000.

21 I think it's two miles closer than the Richmond
22 area, which is the current population center.

23 Does that make any difference or not?

24 BY WITNESS SOFFER:

25 A In the real case such as you're talking, no,

10-19
1 it does not.

2 Since the only criterion that the population
3 center has to meet to satisfy Part 100 is that it be at
4 least one-and a third times the LPZ distance, since the
5 LPZ outer radius for Allens Creek is about 3 1/2 miles,
6 a population center could be, theoretically, as close
7 as about five miles and still satisfy the requirements
8 of Part 100.

9 That does not necessarily mean that it might
10 satisfy the criteria of Reg Guide 4.7. However, Reg
11 Guide 4.7 is applied only at the time of a construction
12 permit and applied in a prospective way, and they're
13 not applied retrospectively after the licensing situation --
14 after licensing becomes effective, so that if you're
15 talking about a population center which presently is 20
16 miles away, and if you were to say that a new population
17 center would develop, say, 18 miles away at sometime in
18 the future, but within the life of the plant, as a
19 practical matter that would not affect the status of
20 the plant or any actions that we would require on the
21 plant.

22 BY MR. DOHERTY:

23 Q Now, just to get this clarified -- and I hate
24 being obtuse -- using a figure on Page 7, the LPZ, the
25 outer radius there is given as 3.5 miles, right in the
middle of the page practically.

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1 The population center, would we take 3.5 and
2 multiply it by one-and-a-third to get the closest place
3 from which the population center could be without disturbing
4 the site?

5 BY WITNESS SOFFER:

6 A. Without changing the LPZ, yes, sir.

7 Q. That is the actual real number?

8 BY WITNESS SOFFER:

9 A. Yes, sir.

10 Q. Okay. There was presented in the testimony of
11 the Applicant several figures using sectors, and maybe we
12 won't need to pull those out and look at them, but they
13 use sectors of population. I think there were 16 sectors
14 and it looks like a very common procedure.

15 Now, do you have at this moment any 1980
16 Census data in that form, that is in that form of sectors?

17 BY WITNESS SOFFER:

18 A. No, sir, we do not.

19 Q. I see, and that kind of data would be very
20 difficult to obtain, I gather?

21 BY WITNESS SOFFER:

22 A. We have a computer program that enables us to
23 print out population in the form of sectors and annular
24 elements; however, at present we only have a 1970 Census
25 computer tape.

12-2

1 The Census has not issued the 1980 Census on a
2 computer tape yet, so we are unable to get it in that
3 format.

4 Q I see, but it seems to me there was a figure,
5 too, in here, or just a table that did use 1980 Census
6 data. Am I right about that?

7 BY WITNESS SOFFER:

8 A Yes, you are right, but we obtained that
9 manually without the aid of a computer program, and what
10 we did basically was to take a map of the Allens Creek
11 area.

12 We drew circles of five miles, ten miles,
13 twenty miles, thirty miles radius, and then by using
14 a printed copy of the 1980 Census information for the
15 State of Texas, which included counties and minor civil
16 divisions of Texas counties, we were able to allocate
17 appropriate portions of those counties within those
18 circles so that we were able to estimate what population
19 resided between zero and five miles and what population
20 resided between five and ten; but it is quite difficult
21 to allocate them within sectors and we did not attempt
22 to do that.

3 Q Okay. From reviewing the 1980 Census figures,
4 did you look back at any of the studies that were submitted
5 by the Applicant with any fresh ideas of their accuracy?

12-3

1 In other words, did the Census figures change
2 your ideas of the accuracy of those reports in any way?

3 BY WITNESS SOFFER:

4 A I did not check to see how the 1980 Census
5 numbers compared with the original projections made by the
6 Applicant.

7 I don't know if Mr. Ferrell did or not.

8 BY WITNESS FERRELL:

9 A I believe that the first data that was submitted
10 by the Applicant back in '72 or along in there, that the
11 different source that they used projected even higher than
12 the later Census; but it's been several months since I
13 looked at it, but that was my understanding at that time.

14 So originally they predicted higher, and then
15 they came in later and lowered the projections.

16 Q This was in 1972?

17 BY WITNESS FERRELL:

18 A I believe that's right, yes, but it's been a
19 while since I've looked at that data.

20 Q You don't recall it by name?

21 BY WITNESS FERRELL:

22 A I'm not sure. I think it was the Texas Water
23 Development Board, but I'm not sure.

24 It was whatever they used for their first
25 analysis. They had several of them listed and then they

2-4

1 chose one, but at thi. time I'm not sure which they used.

2 Q But you are pretty sure of the year?

3 BY WITNESS FERRELL:

4 A I think it was around '72. It was whenever they
5 came in with their first application.

6 I was not working on that plant at that time,
7 but I later reviewed it and it looked like the original
8 data was higher.

9 Q You said the first application. That would be
10 1974?

11 BY WITNESS FERRELL:

12 A Do you remember, Len?

13 BY WITNESS SOFFER:

14 A I believe the date of the PSAR was 1974 but
15 the Applicant's projections were of 1972 data.

16 Q Okay. So it would have to be before 1974
17 anyway, wouldn't it?

18 BY WITNESS FERRELL:

19 A Yes, sir.

20 Q Okay. Turning to page 15 of your testimony,
21 please, there's a quotation from the SER on that page,
22 and I was wondering -- you might want to read that again,
23 but maybe you can answer this without it.

24 Is the Staff required to use independent
25 sources, other than the U.S. Census in determining

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1 population suitabilities for siting?

2 BY WITNESS SOFFER:

3 A I'm not sure I understand your question. You
4 used the word -- there were two parts of it.

5 The Staff is not required to do anything in
6 that regard. The Standard Review Plan that the Staff goes
7 by suggests that the Staff reviewer should review the
8 Applicant's sources of data and methodology, and if
9 possible, check them with independent sources.

10 There is no requirement that those sources be
11 U.S. Government. Their desire is to use independent
12 sources, sources that can be used to confirm whether the
13 Applicant has done a reasonable job.

14 Q In that paragraph you state, first, Applicant's
15 projections for the region within 50 miles, and then you
16 speak about the BEA and what they project for BEA Area
17 No. 141.

18 Who went to the BEA's figures, you or the
19 Applicant?

20 BY WITNESS SOFFER:

21 A We did.

22 Q Okay. And do you mean for us -- in looking
23 at that paragraph, there's two percentages there for two
24 different years, 2000 and 2020.

25 Are you saying there that 122 percent and 79

1 percent are a reasonable agreement in your estimation; is
2 that what you mean?

3 BY WITNESS SOFFER:

4 A Over a 20-year period, I would say yes, that
5 that is what I would call a reasonable agreement.

6 Q And 154 and 208 percent; is that the same?

7 BY WITNESS SOFFER:

8 A When you consider the lifetime of the projection,
9 over a 40-year period, yes, I would consider that reasonable
10 agreement.

11 Q Do you consider Census figures better than
12 independent sources for projections in this area particularly?

13 BY WITNESS SOFFER:

14 A Not necessarily. They are usually more
15 convenient for us to get, but they are not necessarily
16 better.

17 Q How does the Bureau of Economic Analysis obtain
18 its statistics? It doesn't do house counts, does it?

19 BY WITNESS SOFFER:

20 A Well, it starts with basic Census data. It
21 divides the country up into a number of regions that are
22 basically all within the same general labor market or same
23 economic regions.

24 These consist of multiple-county areas, and then
25 it basically performs an economic analysis, looking at

12-7
1 labor market trends, growth in employment, industry,
2 transportation, and attempts to base population projections
3 on those.

4 There are basically two methods of making
5 population projections. One is by using demographic
6 techniques, which is to consider a given area and to look
7 at the components of change within that area.

8 This would include things like fertility or
9 birth rate, mortality or death rate, and migrations.

10 Such a model is called the demographic model.

11 An entirely different model is usually an
12 economic model, which merely looks at the employment and
13 the labor market in a given area, what the trend in
14 employment has been, what's happening with transportation,
15 key industries and things of this nature; and it tends to
16 make projections on the basis of economic activity.

17 People at the Census Bureau tend to rely on
18 demographic models. The Bureau of Economic Analysis, which
19 is, incidentally, part of the same agency, the Department
20 of Commerce, tends to rely on economic models; and there is
21 a little bit of interservice rivalry between them.

22 There's no clear-cut superiority, in my
23 opinion, between these two.

24 Q I see.

25 MR. DOHERTY: Your Honor, during the last

12-8
1 break Mr. Copeland mentioned to me that he expected to
2 have following this panel the panel on technical
3 qualifications available, not until, I think he said,
4 afternoon or noontime.

5 He also said he didn't think there was anything
6 to put in that space tomorrow morning. So it appears that --
7 I would like to stop now and continue tomorrow, but I
8 want to be certain I've represented Mr. Copeland correctly
9 here, what he said and so forth.

10 I don't want to -- we can give an option. I
11 know you are not feeling a hundred percent and I'm not
12 either. I'm tired.

13 Maybe we could stop if there's no real reason
14 to push on.

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1 MR. COPELAND: I have been scrambling, Your
2 Honor, to see what we could do for tomorrow. I, frankly,
3 was caught by surprise that we would get this far ahead
4 on the schedule.

5 I have checked with Mr. Oprea and Mr. Goldberg,
6 and I can have them both here by 1:00 tomorrow, but I can't
7 get them here any sooner than that tomorrow.

8 I'm sorry, I wish I could do otherwise.

9 JUDGE WOLFE: Well, then certainly we can recess
10 for tonight. It's 5:00 -- or close to.

11 We'll recess until tomorrow morning at 10:00.

12 MR. COPELAND: Well, Your Honor, I -- That's
13 fine with me, but I wonder if the Board would prefer to
14 start a little later in the morning, because it looks like
15 to me if Mr. Doherty is anywhere close to through that we
16 might end up with a big hole in the middle of the day,
17 waiting until 1:00. I just don't know.

18 JUDGE WOLFE: Yes, Mr. Doherty. Do you have
19 much more cross?

20 MR. DOHERTY: Probably an hour, an hour is
21 being conservative.

22 MR. COPELAND: In my mind, Your Honor, it might
23 be preferable to start a little later and have everybody
24 a little fresher and maybe we could run over a little bit
25 longer tomorrow evening.

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1 JUDGE WOLFE: All right. What's the suggestion?
2 10:00, 10:30, 11:00?

3 MR. COPELAND: I would suggest we start at
4 10:00.

5 JUDGE WOLFE: The suggestion is 10:00. 10:00
6 will be fine.

7 We'll recess until 10:00 a.m. -- Mr. Sohinki,
8 do you have something in hand?

9 MR. SOHINKI: Yes, sir, the Staff's response
10 to the Applicant's motion for reconsideration has been
11 delivered to me and I have copies for the Board and for
12 the parties.

13 JUDGE WOLFE: All right. Would you hand
14 them out.

15 (Pause while documents are distributed.)

16 JUDGE WOLFE: All right. There being nothing
17 more, we will recess now until 10:00 a.m.

18 (Whereupon, at 5:05 p.m. the hearing was
19 recessed, to reconvene at 10:00 a.m., Wednesday, October 7,
20 1981, in the same place.)

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This is to certify that the attached proceedings before the
NUCLEAR REGULATORY COMMISSION

in the matter of: HOUSTON LIGHTING & POWER COMPANY

Date of proceedings: October 6, 1981

Docket Number: 50-466 CP

Place of proceedings: Houston, Texas

were held as herein appears, and that this is the original
transcript thereof for the file of the Commission.

Mary L. Bagby
Official Reporter (Typed)

Mary L. Bagby
Official Reporter (Signature)