NUCLEAR REGULATORY COMMISSION

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U.S. NUCLEAR REGULATORY COMMISSION

In the Matter of:

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

DATE: October 29, 1981 PAGES: 19331 thru 19592

AT: Houston, Texas

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ALDERSON REPORTING

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...ne: (202) 554-2345

UNITED STATES OF AMERICA

BEFORE THE

NUCLEAR REGULATORY COMMISSION

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

Advocacy Auditorium
South Texas College of Law
1303 San Jacinto Street
Houston, Texas

Thursday,
October 29, 1981

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

APPEARANCES:

<u>Board Members</u>:

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

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

DR. E. LEONARD CHEATUM
Administrative Judge
Route 3, Box 350A
Watkinsville, Georgia 30677

APPEARANCES: (continued)

For the NRC Staff:

LEE DEWEY, Esq.
-and-
RICHARD L. BLACK, Esq.
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

For the Applicant - Houston Lighting & Power Company:

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For the Intervenors:

JOHN F. DOHERTY
4327 Alconbury
Houston, Texas 77012

I N D E X

I N D E X (continued)

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P R O C E E D I N G S

9:00 a.m

JUDGE WOLFE: All right.

In attendance this morning are Mr. Copeland representing the Applicant; Mr. Doherty; Mr. Black and Mr. Dewey representing the Staff.

Before we proceed with the cross-examination by Mr. Fields, the Board has been conferring, reviewing the transcript. At the top of page 19,324 of the transcript, Mr. Fields stated: " ... it is general policy to combine seismic loads with LOCA loads for evaluation of all safety-related structures."

And he proceeds to say, "Now, I would interpret that to mean that since pool swell loads are the result of LOCA loads, that the froth loads at the HCU floor would be combined with the seismic load at the HCU floor."

"However, I can't say I have read that particular statement in the PSAR."

We would ask Staff and Applicant, either through pointing to documentation or via the presentation of a witness, to confirm that the HCU supporting platform and the HCU's were or are being designed to withstand combined forces of seismic loads and LOCA pool swell loads at Allens Creek.

MR. COPELAND: All right, sir.

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JUDGE WOLFE: Now, do we have this document?

MR. COPELAND: We have the PSAR here. I think we can make a quick check. I think you may have misread what he stated, Your Honor, if I may.

You inserted the word "not" at line --

JUDGE WOLFE: What line was that, Mr. Copeland?

MR. COPELAND: At line 2.

You read his testimony as saying, " ... it is general policy not to combine ...."

JUDGE WOLFE: Oh? No, it reads: "However, it is general policy to combine seismic loads with LOCA loads for evaluation of all safety-related structures."

MR. COPELAND: Yes, sir.

JUDGE WOLFE: If I did put the "not" in there, it shouldn't have been in there.

Will we need some time for this?

MR. COPELAND: Yes, sir. I'd suggest that we just proceed ahead. That was Mr. Lugo, and he tells me that he knows in fact they are designed that way. But he is going to try and find the citing in the PSAR.

JUDGE WOLFE: All right.

(Bench conference.)

JUDGE WOLFE: All right. You may proceed with your cross-examination, Mr. Doherty -- no, let's see ... it's -- yes, your cross-examination. All right.

MR. DOHERTY: Thank you.

Whereupon,

MEL B. FIELDS

the witness on t' . stand at the time of adjournment, resumed the w' ͜ ss stand and, having been previously duly sworn, ' s examined and testified further as follows:

FURTHER CROSS-EXAMINATION

BY MR. DOHERTY:

Q. Mr. Fields, is there any way a load would be in water phase when it struck the HCU platform, to your knowledge?

A. Based on what we've seen from the PSTF tests, the water will break up into froth well below the HCU floor. The 18-foot specification is actually three or four feet over what we realistically expect the break-through to occur.

Q. Do you expect this break-through to always be at the same distance above the level of the pool throughout the entire 360-degrees?

A. There will be some variations as far as the break-through height for a particular accident, as you go around the circumference of the pool. The 18-foot specification was specified to bound all the possible variations in localized break-through, as well as a maximum injection of steam into the suppression pool --

steam and air.

Q Is there anything in the structure of the reactor building that would cause this variation?

A. There is a couple of structures, approximately 10 feet over the suppression pool, which will help the break-through process. The Staff and GE ignored this particular structure, this catwalk, in the development of the break-through height, which is conservative.

In actuality, this catwalk will probably cause break-through much sooner than the expected elevation.

Q Is that a 360-degree circle catwalk?

A. I believe it is, yes.

Q But you're not certain?

A. I'm not certain.

Q Is the suppression pool a uniform distance from its inner circle, let's call it, to its outer circle -- at the surface of the water for the full 360 degrees?

A. Are you asking is the pool width constant?

Q Yes, that might be one -- But at the level of -- Yes, all right. Let's try that and see --

A. The pool width is constant.

JUDGE LINENBERGER: By the way, do you recall what that dimension is?

THE WITNESS: Approximately 20 feet.

JUDGE LINENBERGER: Twenty. Thanks.

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

JUDGE WOLFE: Yes.

MR. BLACK: What are you going to approach him with?

(No response.)

BY MR. DOHERTY:

Q Mr. Fields --

A. Yes.

Q -- did I just show you two figures from the PSAR, one marked Figure 1.2-8, Section A-A, and the other marked Figure 2.2-2 of Revision 2 dated 12-20-79?

A. Yes, you did.

Q All right. I'd like to ask you a question from Figure 1.2-8. On this lower left side, which shows the suppression pool in cross-section, there appears to me to be a kind of structure, which indeed seems to shorten the distance across the suppression pool at that particular point, when compared to the suppression pool directly -- diametrically opposed to it.

That is, the width appears lesser in the left side of that diagram than on the right. Can you explain that?

A. Yes. There are certain structures which enter the pool, such as piping, which, of course, would reduce the surface area of the pool at that point.

In this particular point, you're looking at a baffle structure, which is directly under the personnel lock to prevent any impact loads from hitting the personnel lock. This is a very limited structure in size, several feet out from the drywell wall, and maybe four or five feet in width.

Q. Well, in your opinion, regardless of its size, does it reduce the distance across the pool at that location?

A. At that location it would reduce the level -- the width of the pool in that location.

Q. Now, looking at Figure 2.2-2, does that illustrate the same baffle to your mind or not?

A. One baffle is for the personnel lock, and the other baffle is for the TIP drive unit.

Q. I see. How far around in degrees does the baffle for the TIP drive units extend? Do you recall?

A. I don't recall. I'm trying to determine that from the drawings.

GE has performed some tests to show that blockage directly over the vents do not affect the vent clearing aspects of the containment, as far as the contention at hand, which is froth impact on the HCU floor.

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These structures, if anything, would reduce those loads.

Q Now, in the drawing marked Figure 1.2-8, Section A-A, are there hydraulic control unit modules above the baffle for the personnel lock?

A When you say "above," do you mean just anywhere above; or do you mean directly above?

Q I mean directly.

A I couldn't tell from this diagram whether they're directly above the personnel lock. They are not, certainly, below the HCU floor, which means it would be another 10 or 15 feet above the personnel lock in elevation.

But exactly if it's over the personnel lock is something I could not determine at this point.

Q Okay.

(Pause.)

MR. DOHERTY: I'm sorry for the delay. One of the answers made me look up something.

BY MR. DOHERTY:

Q What is the duration of these pool swell loads?

MR. COPELAND: Excuse me, Your Honor, I'm going to object to that. I don't believe that's within the scope of Mr. Doherty's contention.

He says, as I read his contention -- it's an allegation that the loads themselves have been underestimated, not how long they will last.

MR. DOHERTY: Well, I think there's two reasons it's relevant. There is an interpretation, I think, going here by counsel, which -- I mean I didn't specify here, other than by hydrodynamic forces, I think is the term used -- how the damage might occur.

And I didn't mean in filing this to exclude any durational aspects.

MR. COPELAND: I'll withdraw the objection.

JUDGE WOLFE: Mr. Fields.

THE WITNESS: The duration for the froth load at the HCU floor is slightly over 3 seconds. That's a specification.

BY MR. DOHERTY:

Q. You say that's a specification. What --

A. That means that the duration specified bounds the expected duration.

JUDGE LINENBERGER: Mr. Fields, when you say that, let me give you the impression it has on me, and then you -- if I'm wrong, please correct me -- but when you say there's a specification with respect to the 3-second duration of froth load, it sounds as though somebody is, in a sense, legi ating that the load will not

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endure more than 3 seconds, and we won't tolerate anything that says it may exceed that.

Now, I well recognize this may not be the way things are here with respect to this specification, but that's the kind of reading I hear out of your words. So can you comment about that?

THE WITNESS: Yes. Based on the velocity of the pool and the maximum amount of interaction between the air and the water, which creates the froth, the maximum amount of uplift force you can have has been determined, and also the maximum duration that you can conceivably have that froth going in an upward direction causing a loading on the HCU floor.

That's how the number 3 seconds was calculated. Once 3 seconds is over with, there is no more froth to be impinged upon the HCU floor.

JUDGE LINENBERGER: Well, then, my next question has to be: What is the point of this specification and how is it -- to what does it apply? This 3-second load-duration specification.

THE WITNESS: The specification begins with a triangular impulse load which lasts for approximately 100 milliseconds, followed by the 3-second froth drag load.

This specification is applied to any structure

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that is within 19 to 30 feet above the initial pool surface.

JUDGE LINENBERGER: Okay. Then, in other words, you're saying the specification is not something that ordains that the pool swell phenomenon can't last any longer than that? It is something that is derived from an analysis of the pool swell phenomenon and is imposed on the design of those structures, such that they must be able to survive that long; is that correct?

THE WITNESS: Basically. When you're talking about a load of 3 seconds duration, you're talking about a static load.

And 3 seconds, 30 seconds is not going to make any difference with respect to the design of that structure.

Once you pass the point of any dynamic effects, then it's just a static load, and it would make no difference to the designer whether the load was 3 seconds or 30 seconds.

JUDGE LINENBERGER: All right, sir, thank you.

BY MR. DOHERTY:

Q. At this point is the pool swell mass considered acceptably determined by the Staff?

A. The pool swell mass? The amount of water

1-11

that's in the suppression pool?

Q I was trying to get at some kind of a weight sort of thing.

A The critical parameter, as far as it relates to pool swell mass, is the height of the suppression pool over the top vent. And the NRC has determined that the current height of the top pool vent of 7 1/2 feet is acceptable with respect to pool swell load definitions.

Q How does the load take into account the amount, or does it?

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A. You're asking how does the load take into account variations?

Q. The definition defining load, how do you take in amount?

MR. COPELAND: The amount of water, Mr. Doherty? The weight of the water?

MR. DOHERTY: The amount of water.

THE WITNESS: The load specification is based on a maximum vent submergence of 7 1/2 feet. The Applicant is not allowed to have a vent submergence depth any greater than 7 1/2 feet.

If it's less than 7 1/2 feet, the pool swell loads will be reduced.

BY MR. DOHERTY:

Q. So is the assumption that all of that 7 1/2 feet reaches that height?

A. That is correct. It's conservatively assumed that none of the water drops back into the pool during pool swell and after break-through. Instead, we conservatively assume that all of the water mixes uniformly with the air and continues to rise.

In reality, a lot of the water would just have dropped back to the suppression pool resulting in froth densities much lower than what we use as a licensing basis.

1-13

Q What is the froth density you used for this?

A Approximately 19 pounds per cubic feet. It's 18 point something.

Q In General Electric's submittal, have they filed a computer-type of code for calculating this uplift?

A You're referring to the froth loads?

Q Yes.

A Froth loads were determined based on the tests performed at the PSTF facility.

Q I see. Are you all in agreement on methodology of how to make these calculations?

A As I stated earlier, the Staff has not quite finished its review of the specification for the pool swell velocity. At this point we don't see that it's going to require a major change in the pool swell velocity.

However, a change in the pool swell velocity will change the froth loading. And as I also stated earlier, for plants at the stage that Allens Creek is in, we don't feel that there's going to be major design changes that need to be made, if there are any changes.

Q Is that mainly on the idea of feeling that the pool velocity is pretty well calculated out and just unlikely to be much faster than you think at this point?

A Basically, yes.

1-14

Q. Okay. I'm looking at page 3 here where I think the Board had a question with regard to SRV's.

In the language of your reply, you said that -- to borrow the top there -- "GE has provided arguments to show the single-failure criteria should not be applied."

How would that be applied here? What would be the failure? The pipe burst?

A. In this case, single failure would be an SRV inadvertently opening.

Q. Now, is the statistic of probability there one just taken from normal plant operation; is that your understanding?

A. Basically, you look at the time interval in which an SRV has to open, in order to combine the SRV loads with pool swell loads. That time interval is approximately [illegible] d.

If you just do a fairly simple statistical approach, you can show that that reduces the magnitude of the probability by a factor of 100 or so.

Q. Didn't you say a moment ago that the magnitude of duration of those pool swells was 3 seconds, approximately, in the calculations at least?

A. That is correct. However, to combine the SRV loads, you have to have the SRV actuating while there is still water that the SRV can add the load to before it's

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sent up by the LOCA air bubble.

Q Okay. The Board asked -- This is kind of an evasive question. The Board stated that absent final results of vibrational effects on the HCU's, they had concluded that something more needed to be said in the hearing.

Are impact loads that we've been talking about and vibrational loads the same thing in your mind at this point?

A The impact duration could cause vibrational loads on the structures.

Q Well, as I hear you say that, you're saying one is the cause of the other, that is, and that the Board is interested in an effect.

In other words, my contention talked about the effect of sort of this, and they're talking about the effect of that; that is, the -- Is that right? Then we do have a difference here.

A I'm not sure if I understand the thrust of your question.

Q Well, it is a cause-and-effect situation, isn't it, that impact load means -- would be the cause of a vibrational load? Is that the way you ...

A It could cause a vibrational load, yes.

Q And that would be the proper terminology?

We're using the proper terminology?

A. Yes.

Q. Now, is the design based LOCA the largest pipe-type of LOCA?

A. Yes.

Q. -- in the drywell?

A. It's a double-ended rupture of the main steam line, which is the largest pipe contained in the drywell.

Q. Is there any way that the LOCA might be a smaller pipe -- Well, let's put it this way: Is there any way a LOCA with a smaller pipe, combined with operation of some of the high-pressure system -- safety system -- might cause a greater pool swell?

A. No.

Q. And is that something the Commission inquires into --

A. It is something you could determine from looking at the phenomenon involved. The pool swell is basically a function of how fast the pressure inside the drywell can rise.

The pressure rise inside the drywell is determined by how much energy can get out of that pipe. Therefore, the larger the pipe break, the larger the energy into the drywell, and the faster the pressure rise.

A smaller pipe break will result in less energy

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into the drywell, and, therefore, less drywell pressure and a less pool swell velocity.

JUDGE LINENBERGER: Mr. Fields, on this point you have described things in terms of amounts of energy as though, perhaps, time were not a factor. Now, is that a proper -- I don't trust that inference from your words. I don't think it's a proper inference from your words, that time is not a factor here.

THE WITNESS: Time is definitely a factor. When I say energy, I should say energy rate.

JUDGE LINENBERGER: Okay. That's what I thought was involved, but I wanted to hear you confirm it. Thank you.

BY MR. DOHERTY:

Q. Well, at this poin is it your understanding that the froth load will also hit the traversing in-core probe control unit , or are they really out of the way in current plans, such that they would not be loaded?

A. The TIP station is not in the froth zone. It's in the solid water zone. Liquid water, I should say.

Q. Well, would those loads be direct on those units; or is it your understanding there are concrete -- yes, concrete structures in the path of the rods?

A. It's my understanding that the TIP station is enclosed by a concrete structure, which also has a baffle

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which extends into the suppression pool to eliminate liquid water impact on the concrete structure itself.

JUDGE LINENBERGER: Mr. Fields, are you in a position to know for a certainty whether the traversing in-core probe system is essential or necessary to the safe shutdown of the reactor system?

THE WITNESS: It's my understanding that's not an essential system. It has no functions except for mapping the core during normal operation.

JUDGE LINENBERGER: Then should I conclude from that statement that if, following a LOCA, pool swell or froth forces associated therewith completely destroy the traversing in-core probe assembly that one might consider this a relatively negligible consequence, in terms of managing the shutdown of the reactor following such a LOCA?

THE WITNESS: As long as that destruction did not affect other safety-related equipment.

JUDGE LINENBERGER: Thank you.

BY MR. DOHERTY:

Q Now, a short while ago you spoke about a catwalk that was between the HCU level and the pool suppression -- or suppression pool surface. You mentioned it ... said it was annular, apparently.

How many feet does that stick out?

1-19

A. I'm not sure. Three or four feet.

Q. Uh-huh. So the personnel ...

MR. DOHERTY: I just need a minute or two to check over my notes before I finish.

(Pause.)

BY MR. DOHERTY:

Q. You spoke that a report would be out quite soon on the load -- I think -- am I correct in this, load definitions, that part?

A. For which type of accident? I mentioned two reports.

Q. I think one of them was due in November, next month.

A. Yes.

Q. Will that report contain any information on vibrational effects on the HCU's?

A. No. The load is basically -- will present the load definitions due to SRV actuation.

Q. Is there any research going on on vibrational effects at the moment, to your knowledge?

A. The effects that vibrational motions have on structures?

Q. The vibrational results of impact loads from pool swell. Is there anyone studying that, do you know?

A. That's a little bit outside my area.

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Q. Well, I'm not asking you what results those are, but I was asking if you know if there was someone in the NRC studying that, to your knowledge.

A. It's my understanding that the methdology used to take these load definitions and calculate a structural response is fairly standard and has already been approved by the Staff.

Q. So it's already done?

A. The methodology, I think, has already been accepted.

Q. But then is it true that you don't know if anyone has gone ahead and applied that yet? It's just ...

A. I imagine the Mark III's at the operating licensing stage have done it for their HCU modules that are in place.

Q. Okay.

MR. DOHERTY: No further questions. Thank you very much.

JUDGE WOLFE: Is there redirect, Mr. Black?

MR. BLACK: No questions.

JUDGE WOLFE: Board questions?

JUDGE CHEATUM: I have no questions.

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BOARD EXAMINATION

BY JUDGE LINENBERGER:

Q Mr. Fields, I think the general subject here has been pretty well covered by you, but a couple of details.

Getting back to safety/relief valve actuation and your discussion of it at the bottom of page 3 of your prefiled testimony, there's something I need to understand.

The pool swell and frothing phenomenon that we've been talking about -- or your testimony addresses -- is the result of, presumably, a large pipe break that results thereafter in a fairly significant amount of energy being delivered in a fairly short time to the water in the pressure suppression pool.

That energy comes from, it seems to me, the depressurization of the reactor pressure vessel. In order for the pool swell and frothing to reach a magnitude or, if you will, an elevation such that it's getting close to the HCU's support platform, to me implies that a significant amount of depressurization of the reactor pressure vessel has occurred.

If that is true -- and I'm going to give you an opportunity to correct the premise as I've stated, leading up to the question -- but if what I've said so far is

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true, then I have a problem seeing how the actuation of a safety/relief valve, or a couple of safety/relief valves, can significantly alter the behavior of the suppression pool water during this uplift phenomenon because I don't see how the safety/relief valves can provide significantly more venting or more energy release to the suppression pool water than has already occurred as a result of the large pipe break.

Therefore, despite the Board's question about this matter, I guess I have to ask: Why is an SRV actuation during pool swell a substantive dynamic consideration at all with respect to the behavior of the pool swell?

A. I don't think that we are saying that it is a substantial load. It's just that the method that we approach to resolve this issue is to show that it would not occur.

Now, if we wish to go back and say that combine the loads and see what the effects are, it gets very complex because the SRV loads are based on a bubble oscillating in liquid water.

Now, if you had an SRV actuating during the same time you had a LOCA bubble entering the pool, you would probably not have the same kind of loads. In fact, I'm sure it would be totally different, and I imagine much

1-23

less than the SRV loads you would have if you just had no LOCA.

So the methodology that could be developed to calculate what the combined loads were would require a tremendous amount of analysis to see what the possible effects are.

I imagine they are minor. But I guess it was felt that the easiest way to resolve this issue was to show that -- from a probabilistic standpoint and from a mechanicalistic standpoint, it would not occur.

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Q Okay. Your answer to my question about the effect of pool swell and froth loads on the traversing incore probe assemblies, you indicated that even the deactivation or destruction or disruption of those TIP units would not compromise the safe shutdown of the reactor so long as their destruction didn't result in any damage to a safety related system.

I think this was essentially what you said; is that correct?

A That's true.

Q Now, one way of looking at that is that all you are saying is all safety related systems must remain operational.

Another way of looking at what you said is there may be a possibility of an interaction of some sort following from the destruction of these TIP assemblies, as though maybe disruptive forces could break one of the assemblies loose and hurl it at something that is depended on for the safe shutdown of the reactor.

Are you aware of any kinds of interactions like this that need to be worried about?

A No. The TIP station is enclosed in a concrete structure. Now, assuming that the pool swell load caused significant enough loads to be transmitted to the concrete structure and put the TIP's out of commission, it would not

-2

take these TIP's and, say, hurl them up 25 feet because of vibration loads and run into the HCU floor.

So as far -- when I said as long as it did not cause damage to safety related equipment, to cause damage you would actually have to break them off somehow, and since they are not exposed to pool swell loads directly, it would be transmitted to the concrete structure, and I don't see any way of that happening.

JUDGE LINENBERGER: Thank you, sir.

That's all the questions I have.

JUDGE WOLFE: Cross on Board questions, Mr. Copeland?

MR. COPELAND: No, sir.

JUDGE WOLFE: Mr. Doherty?

MR. DOHERTY: No, Your Honor.

MR. BLACK: And I have none either.

JUDGE WOLFE: I think at this time I'll put the question to you, Mr. Copeland, Mr. Lugo is here?

We might as well recall him and put him on the stand with regard to the outstanding Board request or question.

MR. COPELAND: We can do that, Your Honor. I think I can tell you where the information is in the PSAR, whichever way you want to do it.

I would now determine that we'll also have to

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put Mr. Malec on.

JUDGE WOLFE: Pardon me?

MR. COPELAND: We will also have to call Mr. Malec, because I wasn't sure whether your question related to the HCU floor or the HCU itself.

JUDGE WOLFE: Is Mr. Malec here?

MR. COPELAND: Yes, sir. Let me go get him, Your Honor.

(The witness was excused.)

MR. COPELAND: Your Honor, at this time I would ask that Mr. Lugo be resworn and Mr. Malec be sworn, also.

Mr. Lugo is on the left and Mr. Malec on the right.

JUDGE WOLFE: We had excused, Mr. Lugo?

MR. COPELAND: Yes.

Whereupon,

MIGUEL A. LUGO

-and-

WALTER F. MALEC

were called as witnesses and, having been first duly sworn to tell the truth, the whole truth and nothing but the truth, were examined and testified as follows:

JUDGE WOLFE: Please be seated.

MR. COPELAND: Your Honor, Mr. Malec has previously filed his testimony in this case. Although he has not testified yet, I would assume voir dire would not be necessary at this point.

I will explain to the Board what his role is and have him explain what his role is in the design of the plant.

JUDGE WOLFE: Yes, would you.

MR. DOHERTY: Excuse me, Your Honor.

Did you say, Counsel, that Mr. Malec had been previously sworn?

JUDGE WOLFE: No. He has written direct testimony which will be present subsequently.

MR. DOHERTY: Yes. I don't want to waive any voir dire rights. That's all.

DIRECT EXAMINATION

BY MR. COPELAND:

Q Let me start with Mr. Lugo first.

Mr. Lugo, do you know whether the HCU floor is designed to withstand both seismic loads and the LOCA pool swell loads?

BY WITNESS LUGO:

A Yes, I do, and this is stated in our PSAR, Section 3.8.3, which has to do with the drywell and the internal containment steel structures.

On page 3.8-28B it appears the list of load combinations to which these platforms must be designed;

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and load combination 1(b)(8) shows that we do consider pool swell loads together with seismic loads.

Q Now, Mr. Malec, are you employed by Ebasco?

BY WITNESS MALEC:

A I am.

Q And what is your title at Ebasco?

BY WITNESS MALEC:

A My title is Mechanical Supervising Engineer.

Q What is your basic responsibility?

BY WITNESS MALEC:

A It includes the technical and administrative responsibility for mechanical fire protection, plumbing, HVAC, stress analysis, supports and restraints, water treatment, inservice inspection, and the design groups associated with those engineering groups.

Q In that capacity are you familiar with the design requirements for the hydraulic control units?

BY WITNESS MALEC:

A I am.

Q Can you tell me, sir, whether the hydraulic control units are to be designed to withstand both LOCA pool swell loads and seismic loads?

BY WITNESS MALEC:

A The hydraulic control units are designated as Safety Class II. They are in the General Electric scope of supply.

They are also designated Seismic Category I.

One of the roles of the mechanical engineering in Ebasco is to interface directly with General Electric. They provide us with the dynamic capability of that equipment.

Houston Lighting & Power Company has contracted with General Electric to analyze these components for those loads.

THE REPORTER: Excuse me, Mr. Copeland. May we have Mr. Malec's first name for the record.

MR. COPELAND: I'm sorry.

WITNESS MALEC: Walter.

JUDGE WOLFE: Is there cross-examination, Mr. Black?

MR. BLACK: No cross-examination.

JUDGE WOLFE: Mr. Doherty?

MR. DOHERTY: Yes, Your Honor, just one or two.

JUDGE WOLFE: Would you like to have just a moment or two to look at the provisions of the PSAR cited by the two witnesses?

MR. DOHERTY: Your Honor, I don't think that would be necessary.

JUDGE WOLFE: All right.

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CROSS-EXAMINATION

BY MR. DOHERTY:

Q Mr. Malec, I think you said a moment ago both the seismic and the LOCA loads. When you say "both," do you mean a single load made up of a combination of those?

BY WITNESS MALEC:

A Ebasco will supply to GE the responses for those particular items and General Electric will combine them in their analytical process to verify the adequacy of the design of the HCU units to withstand those forces.

The exact loading combination is cited in the Containment Structures Design Report.

MR. DOHERTY: No further questions.

JUDGE WOLFE: All right. We'll turn to Board questions.

JUDGE CHEATUM: I have no questions.

BOARD EXAMINATION

BY JUDGE LINENBERGER:

Q I gather, Mr. Malec, that you personally, then, have not involved yourself with the manner in which these loads are combined in this analysis?

BY WITNESS MALEC:

A That's correct, Your Honor.

I am peripherally aware of how the program will proceed within General Electric. That will be their area of responsibility to verify the adequacy of the design of the HCU modules.

Q. So far as the supporting platform is concerned, Mr. Lugo, are you familiar with how these loads are combined such that one can then assess the adequacy of the floor design?

BY WITNESS LUGO:

A. Your Honor, I'm not 100 percent familiar with this. This will be addressed by a future witness from Ebasco from the structural point of view.

I am familiar from the point of view of exposure to this, being in the same group, and I do know that these loads are considered in the design of the platform.

JUDGE WOLFE: Mr. Lugo, what will be the Ebasco witness' name?

WITNESS LUGO: It's Mr. Nuta, N-u-t-a.

BY JUDGE LINENBERGER:

Q. One final wrap-up here, Mr. Malec. I believe you referred to -- maybe these words are not quite right -- containment design report?

BY WITNESS MALEC:

A. That's close, Your Honor. It's Containment Structures Design Report, Revision 2. It's incorporated into the PSAR by reference. If you'd like, I'll find the page.

Q Incorporated by reference. Has this report been published as a GE document that has a number identification to it?

BY WITNESS MALEC:

A No, sir. It's an Ebasco published document.

Q All right, an Ebasco document. Forgive me.

BY WITNESS MALEC:

A It does not have a specific number, simply the title. It's cited in Chapter III. I'll find the page for you.

Your Honor, it's cited in several places. One place that we found very quickly is on PSAR page 3.8-26.

Q Excuse me, dash what?

BY WITNESS MALEC:

A Two six.

JUDGE CHEATUM: Two point six?

WITNESS MALEC: No, sir, 3.8-26. That deals specifically with structures.

However, there is a section in there that does cite it for equipment and loads for equipment in the area of the pool swell.

Your Honor, I can give you that citation now. It's Paragraph 3.9.2.2, PSAR page 3.9-5.

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BY JUDGE LINENBERGER:

Q And is it cited there for the proposition of addressing this analysis of load combination?

BY WITNESS MALEC:

A It says, "The design loading combinations are considered in the design of ASME Code Class II and III components, or categorized as normal upset, emergency or faulty plant conditions in Table 3.9-2. Additional loading combinations for piping and suppression pool area, however, are presented in Chapter VII of Revision 2 of the Containment Structures Design Report."

JUDGE LINENBERGER: Ms. Bagby, were you able to get that or was it too fast?

THE REPORTER: Yes, sir.

(Bench conference.)

JUDGE LINENBERGER: Mr. Chairman, I think that satisfies my inter t, at any rate, in this matter, but perhaps it would be appropriate to just back off a ways and comment about the line of questioning that began last evening with Mr. Doherty and led to an obje tion in part on the basis of relevancy by Mr. Copeland, wherein Mr. Copeland correctly observed that the Doherty contention we are addressing does not explicitly refer to seismic loads simultaneous with a LOCA and pool swell loads, and that is indeed an accurate observation on Mr. Copeland's part.

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The Board in conferring on this matter, as the Chairman indicated earlier, on further reflection felt that given the backdrop of the regulations, which require that safety related components and structures must withstand a simultaneous application of LOCA and seismic loads, we felt that it better served the record and the interests of the public here to at least establish at this phase of the Allens Creek proceeding that this obligation to consider both types of loads simultaneously is being met in the design efforts.

So that's the reason for, if you will, dragging the proceeding through this subject again this morning.

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JUDGE WOLFE: I would also add that the Board's interest was also generated by Mr. Fields' testimony, as I stated earlier today, at page 19,324 when he stated that, "It is the general policy to combine seismic loads with LOCA loads for evaluation of all safety related structures."

We wanted to pin that down to this specific plant and its specifications.

All right. Are there cross-examination on Board questions, Mr. Black?

MR. BLACK: No questions.

JUDGE WOLFE: Mr. Doherty?

MR. DOHERTY: No, Your Honor.

JUDGE WOLFE: All right. The witnesses -- Mr. Lugo, I take it you are apparently now excused again, and Mr. Malec, you are temporarily excused.

(Witness Lugo was excused.)

(Witness Malec temporarily excused.)

MR. COPELAND: Mr. Lugo, I think, wants to catch an airplane as quickly as he can, Your Honor.

MR. DEWEY: Staff's next witness is Mel Fields to testify regarding hydrogen monitoring.

Mr. Fields has previously been sworn in and has previously testified.

JUDGE WOLFE: You remain under oath. I have told you that before, Mr. Fields.

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MR. FIELDS: Yes.

Whereupon,

MEL B. FIELDS

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

DIRECT EXAMINATION

BY MR. DEWEY:

Q Mr. Fields, do you have before you a copy of a document entitled, "NRC Staff Testimony of Mel B. Fields Relative to Hydrogen Monitoring"?

A Yes, I do.

Q Does this document consist of seven pages?

A Yes.

Q Is there an attachment listed as Figure 1?

A Yes.

Q Is there another attachment listed as a July 15, 1974, memorandum, entitled, "Westinghouse Topical Reports on Electric Hydrogen Recombiner"?

A Yes.

Q Is this report two pages?

A Two pages, plus an enclosure.

Q How many pages is the enclosure?

A Eight pages.

Q Is there also attached to your testimony a

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May 1, 1975, letter from the Nuclear Regulatory Commission to the Nuclear Safety Department of Westinghouse Electric Corporation?

A. Yes.

Q. Is there an attachment to this letter?

A. Yes.

Q. Is this attachment three pages?

A. Yes.

Q. Is there a -- a June 22nd, 1978, letter from John Stolz of the Nuclear Regulatory Commission to Thomas Anderson of Westinghouse Electric Corporation?

A. Yes.

Q. Is there a seven-page attachment to this letter?

A. Yes.

Q. Mr. Fields, at this time do you have any changes to make with respect to your testimony?

A. Yes. On page 3, on the fourth line from the bottom of the page, where it reads "1200°F," that should be "1600°F."

MR. DOHERTY: Excuse me. You said page 3?

THE WITNESS: Page 3, fourth line from the bottom.

MR. DOHERTY: And this was out of the most recent submittal?

THE WITNESS: No. Page 3 of my testimony.

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JUDGE CHEATUM: What is the correction again, Mr. Fields?

THE WITNESS: Instead of "1200°" it should read "1600°."

BY MR. DEWEY:

Q Are there any other corrections?

A No.

Q Mr. Fields, with these corrections, do you attest that the statements made in your testimony are true and correct to the best of your knowledge and belief?

A Yes.

MR. DEWEY: Your Honor, at this time the Staff wishes to offer the testimony and attachments of Mel Fields into evidence.

JUDGE WOLFE: Any objection?

MR. COPELAND: I have no objection, Your Honor, but I need to be excused from the room for a minute so I can go get a copy of his testimony.

There is no problem with proceeding in my absence.

MR. DEWEY: Do you want a copy of mine?

MR. COPELAND: Do you have an extra one?

MR. DEWEY: Yes.

MR. COPELAND: Thank you.

JUDGE WOLFE: Mr. Doherty.

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MR. DOHERTY: Your Honor, I just have one or two questions of the witness on voir dire.

JUDGE WOLFE: All right.

VOIR DIRE EXAMINATION

BY MR. DOHERTY:

Q Have you studied the behavior of lighter than air gases in any of your work, any of your studies, schooling?

A No.

Q Do you consider yourself an expert in the behavior of hydrogen in enclosed structures?

A I believe I have the knowledge that will allow me to determine whether or not the distribution of hydrogen inside the containment will be adequate enough to prevent pocketing of hydrogen inside containment.

Q What is the basis of that belief, please?

A The basis for this is I have been reviewing this type of material since I have been with the NRC, for the last six years. It's one of my jobs.

Q When you began reviewing this work, were you given any supervision with regard to hydrogen in containment structures, anything of that order?

A Yes. There were informal conversations with other members of my Branch, seminars within the Branch, various reports to read, such as the Standard Review Plan,

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and the 10 CFR Part 50.44, which contains information on the hydrogen generation release rates allowed.

Q Have you been with the Containment Systems Branch for six years?

A Approximately a year and a half of that time I spent in the Power Systems Branch, where one of my duties was to review the environmental qualifications of the Westinghouse thermal recombiner.

MR. DOHERTY: Okay. No further questions and no objections, Your Honor.

JUDGE WOLFE: Absent objection, the testimony of Mel Fields relating to TexPirg Contention Amended 40, inclusive of the attachments identified by Mr. Dewey.

These documents are incorporated into the record as if read.

(NRC Staff's Testimony of Mel B. Fields on TexPirg Contention 34 follows:)

- - -

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)	)	

NRC STAFF TESTIMONY OF MEL B. FIELDS
RELATIVE TO HYDROGEN MONITORING

[TexPirg Contention ~~A-40~~ 34]

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

A. My name is Mel B. Fields. I am employed at the U. S. Nuclear Regulatory Commission as a Containment Systems Engineer in the Containment Systems Branch. I have testified previously in this hearing on Board Question 4B, Compliance with GDC 50; Board Question 9, Bypass Leakage; and Board Question 4A, Combustible Gas Control.

Q. What does TexPirg Contention ~~A-40~~ 34 allege?

A. TexPirg Contention ~~A-40~~ 34 states as follows:

> TexPirg contends that the Applicant monitoring of in containment building events during LOCA or similar events is not adequate to detect immediately the occurrences of hydrogen explosions. That the recent Three Mile Island incident shows that current approved containment building monitoring apparatus did not bring such an event to the attention of operators immediately, and that therefore the strong possibility existed that actions which would prevent a second hydrogen explosion were not taken. There is danger that hydrogen explosions will endanger TexPirg members because the containment building during a LOCA is likely to contain radioactive gases which would be released from the building damaged even lightly by the explosion and in excess of 40 CFR 190 or 10 CFR 20.

Q. What is the purpose of this testimony?

A. The purpo: of this testimony is to respond to board questions contained in the September 1st Order on this contention. I will address each of the Board's questions separately.

Board Question #1

Supply test results supporting the adequacy of the type and size of thermal recombiners to be used;

Response

The recombiners currently planned for installation inside the ACNGS containment are Westinghouse thermal recombiners with a flow capacity of 100 scfm.

The staff has been reviewing this recombiner model since 1972. Westinghouse has described this recombiner in WCAP-7709-L, Electrical Hydrogen Recombiner for Water Reactor Containments (July 1971) and in Supplements 1 through 7 to this report. Attached are three letters (R. L. Tedesco to R. C. DeYoung, dated July 15, 1974; D. B. Vassallo to C. Eicheldinger of Westinghouse, dated May 1, 1975; and J. F. Stolz to T. M. Anderson of Westinghouse, dated June 22, 1978) that provides the staff's detailed evaluation of Westinghouse's test program to qualify its thermal recombiner. These letters contain the type of tests run, the standards that the recombiner was required to meet, and the performance characteristics of the recombiner.

Board Question #2

Effects of poisoned recombiner surfaces and convective circulation in reducing recombiner effectiveness;

Response

The recombiner was exposed to severe environmental effects such as steam, containment spray, radiation, temperature, and the performance of the recombiner was not degraded. The details of these tests and their results can be found in the above mentioned three letters.

The effect of convective circulation on recombiner performance has two aspects. The first aspect is the possibility of uneven hydrogen concentrations inside containment leading to possible unacceptably high local concentrations and reduced recombiner efficiency if the H_2 concentration around the recombiner is low. No stratification or pocketing of hydrogen is expected because of various mixing mechanisms present inside the containment such as heat sources, heat sinks and containment sprays. Also, experiments have shown that when a gas lighter than air is introduced at the bottom of a container, as is the case for ACNGS where the hydrogen would be introduced through the suppression pool vents, very rapid mixing occurs. Extensive analysis on this topic is contained in section 6.2.5 of the ACNGS PSAR.

The second aspect is the possibility of convective air currents affecting the performance of the hydrogen recombiner by interferring with the convective air flow through the recombiner or causing recirculation of air that just left the recombiner. Convective circulation of air throughout the containment is caused primarily by temperature differences. Because the temperature difference between the recombiner surface (~~1200~~ 1600°F) and the entering air is so much greater than the temperature differences expected between the containment atmosphere and other heat sources (or sinks), the staff expects the convective air circulation outside the recombiner to have little or no effect on the air flow rising through the recombiner. This expectation was verified by tests performed by Westinghouse (see page 3 of the July 24, 1974 letter reference in the response to Board Question #1).

Board Question #3

Sufficient recombiner dynamic analysis to demonstrate that 3% concentration of hydrogen is a conservative alarm set-point;

Response

The applicant has provided analysis, which has been confirmed by the staff, to show that the hydrogen generation rate using current regulatory requirements (Regulatory Guide 1.7) is far less than the recombination capability of the recombiners once the short-term hydrogen generation from metal-water reaction is over. At the time the hydrogen concentration is reaching 3% inside the containment (approximately 8 days) the hydrogen generation rateis so low that the operator has many hours in which to get one of the two recombiners in operation.

Board Question #4

Relationship - functional and geometrical - between alarm sensor and the eight monitoring samplers;

Response

The location of eight monitoring sample points within the drywell and containment are shown on Figure 1. The locations were determined using two different models. The first model assumes hydrogen diffusion to be identical to neutron diffusion. Buoyancy effects were neglected in applying isotropic diffusion. The use of this model yielded locations above the suppression pool and at the bottom of the drywell. The second model considers the effects of free convection. The buoyancy forces lift the hydrogen from the lower regions of the containment and drywell to higher regions. The influence of trapping was also considered. This model provided five locations: 1) The top of the containment, 2) Near the top of the pressure vessel, 3) Top outside of drywell, 4) Top outside of drywell (opposite), 5) Near the Reactor Water Cleanup Pump area. Both models which provided the sampling locations assumed that no mechanical mixing occurred.

The hydrogen monitoring system consists of sample and return lines, isolation valves, hydrogen analyzers and sample pumps. The equipment excluding the isolation valves and piping is located in the reactor auxiliary building. Each sample line can be monitored by either analyzer through a sample selection manifold. The hydrogen concentration is determined in the analyzer and the volume percent is recorded in the Control Room. The analyzer has a range of 0-5 percent hydrogen with an accuracy of $\pm$ 2.0 percent of full scale and a minimum sensitivity of 0.2 percent hydrogen by volume. The concentration is recorded during sampling and an alarm is automatically actuated if the concentration at any sample point exceeds 3.0 volume percent.

The hydrogen monitoring system is manually actuated from the control room within 30 minutes of a safety injection signal. If Regulatory

Guide 1.7 assumptions are used in the generation rates of hydrogen, operator action, and thus hydrogen monitoring, is not needed for up to 9 hours after a LOCA.

Board Question #5

Ability to periodically test the operability of the monitoring, alarm and recombiner systems;

Response

The hydrogen monitoring and alarm system can be tested and calibrated by introducing low concentration H_2 and N_2 mixtures for zero adjustment and scale calibration. This calibration can be completed from the control room. The recombiners have the capability to be periodically energized to confirm their operability requirements. These tests will be performed at the power levels needed to perform their function of recombining hydrogen with oxygen and for a long enough period to demonstrate stability of the system.

Board Question #6

Basis for confidence that pockets of high hydrogen concentration will not elude the monitoring and alarm systems; and

Response

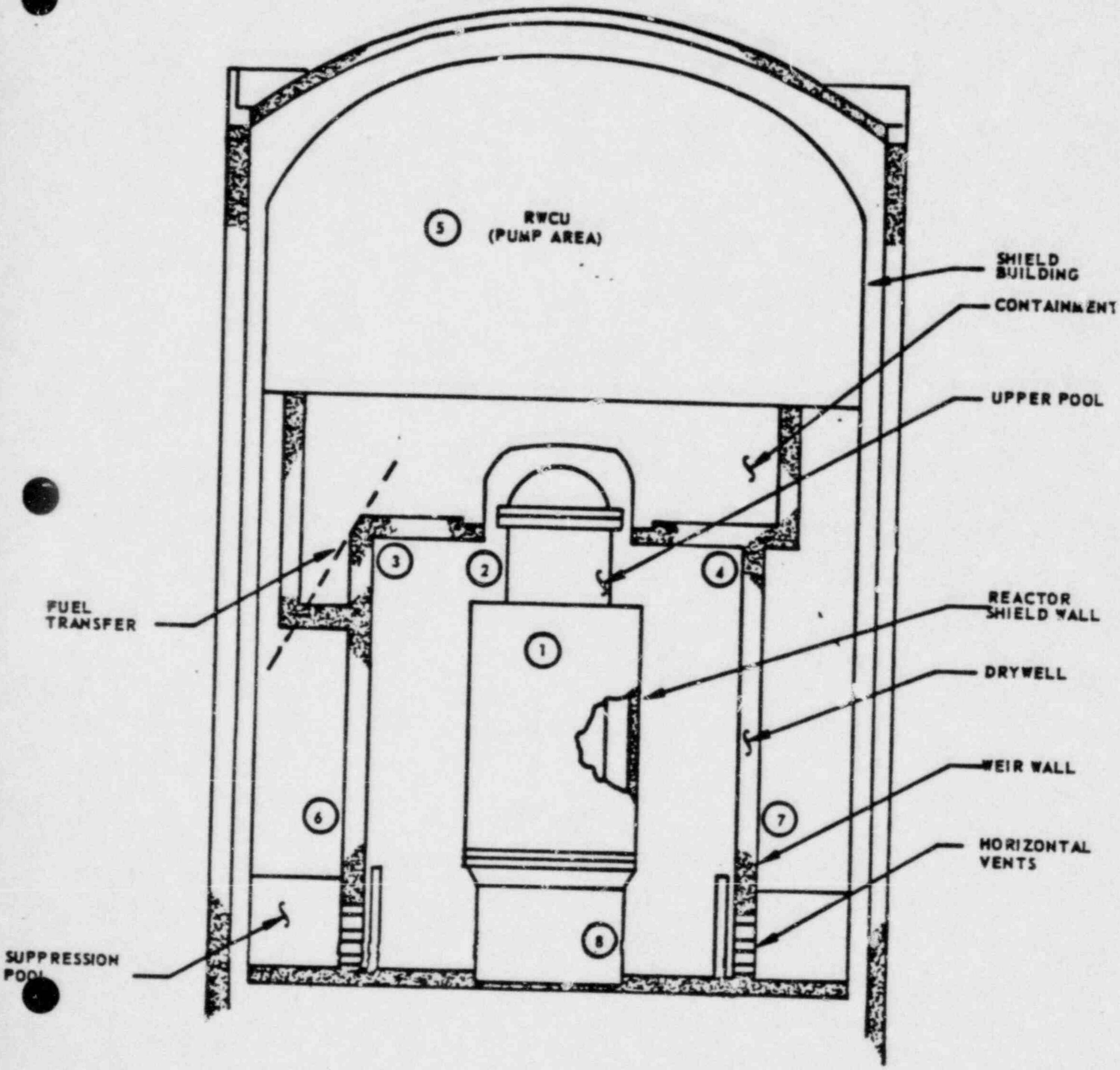
Because of the relatively open area inside the containment and because of the mixing mechanisms (as detailed in the response to Board Question #2) there will be no pocketing of hydrogen inside containment. In addition, the location of the hydrogen monitors was based on where hydrogen could collect if it was possible to do so.

Board Question #7

Nature of the backup containment hydrogen purging system that may be required to function at a time when the containment atmosphere is radioactive.

Response

The backup containment hydrogen purge system consists of a 2" supply line and a 2" exhaust line that would purge the containment atmosphere by exhausting the gas into the annulus. After being recirculated in the annulus to allow for radioactive decay, the gas would then be released through the Standby Gas Treatment System to the environs.



5
RWCU
(PUMP AREA)
SHIELD
BUILDING
CONTAINMENT
UPPER POOL
FUEL
TRANSFER
3
2
4
1
REACTOR
SHIELD WALL
DRYWELL
WEIR WALL
6
7
HORIZONTAL
VENTS
SUPPRESSION
POOL
8

Figure 1

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

JUL 15 1974

R. C. DeYoung, Assistant Director for Light Water Reactors, Group 1, L

WESTINGHOUSE TOPICAL REPORTS ON ELECTRIC HYDROGEN RECOMBINER (TAR-167 & 199)

Report Numbers and Names:

Proprietary	Non-Proprietary
1A. WCAP-7709-L, Electrical Hydrogen Recombiner for Water Reactor Containments (July 1971)	1B. WCAP-7820, Electrical Hydrogen Recombiner for Water Reactor Containments (December 1971)
2A. WCAP-7709-L, Supplement 1, Electric Hydrogen Recombiner for PWR Containments (April 1972)	2B. WCAP-7820, Supplement, Electric Hydrogen Recombiner for PWR Containment (May 1972)
3A. WCAP-7709-L, Supplement 2, Electric Hydrogen Recombiner for PWR Containments Equipment Qualification Report (September 1973)	3B. WCAP-7820, Supplement 2, Electric Hydrogen Recombiner for PWR Containments Equipment Qualification Report (October 1973)
4A. WCAP-7709-L, Supplement 3, Electric Hydrogen Recombiner for PWR Containments Long Term Tests (January 1974)	4B. WCAP-7820, Supplement 3, Electric Hydrogen Recombiner for PWR Containments Long Term Tests (February 1974)
5A. WCAP-7709-L, Supplement 4, Electric Hydrogen Recombiner for PWR Containments (April 1974)	5B. WCAP-7820, Supplement 4, Electric Hydrogen Recombiner for PWR Containments (May 1974)

Originating Organization: Westinghouse Electric Corporation
Responsible Branch: LWR 1-1
Principal Projects Reviewer: E. A. Licitra
Requested Completion Date: July 12, 1974
Review Status: Complete

In accordance with your request of November 3, 1973 the Containment Systems Branch, Directorate of Licensing, has reviewed the subject topical reports. These reports are applicable to many of the license applications that are currently under review. Enclosed is our evaluation of these reports.

In summary, these reports describe the initial development of the electric hydrogen recombiner[1], testing on the prototype recombiner[2], qualification testing on the production recombiner[3], long-term tests on the production recombiner[4], and confirmatory tests on the production recombiner[5]. (We reported on the adequacy of the prototype recombiner in my memo to you dated November 30, 1973.) The results of these tests demonstrated that the Westinghouse hydrogen recombiner should perform satisfactorily for the intended service conditions and therefore, we have concluded that these recombiners are acceptable as part of the combustible gas control system to control the hydrogen concentrations in PWR containments. Review of instrumentation, controls and the seismic analysis of the prototype and the production unit will be conducted by the Electrical and Instrumentation Branch, and the Mechanical Engineering Branch, respectively, as part of the review of license applications of the plants at which these units are to be installed.

We have concluded that the above topical reports (both proprietary and non-proprietary) can be referenced for specific plants that are being reviewed for license applications.

The non-proprietary version presents an adequate representation of the proprietary reports.

RLTedesco

Robert L. Tedesco, Assistant Director
for Containment Safety
Directorate of Licensing

Enclosure:
As stated

cc: w/o encl.
A. Giambusso
W. McDonald

w/encl.

J. Hendrie	R. Klecker	T. Greene
S. Hanauer	D. Eisenhut	O. Parr
J. Glynn	S. Varga	A. Dromerick
D. Vassallo	J. Carter	
E. Licitra	G. Lainas	

TOPICAL REPORT EVALUATION

Report Numbers and Titles:

Proprietary

1A. WCAP-7709-L, Electrical Hydrogen Recombiner for Water Reactor Containments (July 1971)

2A. WCAP-7709-L, Supplement 1, Electric Hydrogen Recombiner for PWR Containments (April 1972)

3A. WCAP-7709-L, Supplement 2, Electric Hydrogen Recombiner for PWR Containments Equipment Qualification Report (September 1973)

4A. WCAP-7709-L, Supplement 3, Electric Hydrogen Recombiner for PWR Containments Long Term Tests (January 1974)

5A. WCAP-7709-L, Supplement 4, Electric Hydrogen Recombiner for PWR Containments (Apirl 1974)

Non-Proprietary

1B. WCAP-7820, Electrical Hydrogen Recombiner for Water Reactor Containments (December 1971)

2B. WCAP-7820, Supplement, Electric Hydrogen Recombiner for PWR Containment (May 1972)

3B. WCAP-7820, Supplement 2, Electric Hydrogen Recombiner for PWR Containments Equipment Qualification Report (October 1973)

4B. WCAP-7820, Supplement 3, Electric Hydrogen Recombiner for PWR Containments Long Term Tests (February 1974)

5B. WCAP-7820, Supplement 4, Electric Hydrogen Recombiner for PWR Containments (May 1974)

Originatin Organization: Westinghouse Electric Corporation, Nuclear Energy Systems

Reviewed By: Containment Systems Branch, Directorate of Licensing, July 1974

Summary of Topical Reports

Westinghouse Electric Corporation has developed an electric hydrogen recombiner as part of the combustible gas control system to control hydrogen concentration within a pressurized water reactor containment following a loss-of-coolant accident. The recombiner consists essentially of a thermally insulated vertical metal duct with metal sheathed electric resistance heater provided to heat a continuous flow of containment gas mixture up to a temperature which is sufficiently high to react the hydrogen and oxygen. The gas mixture enters the recombiner and flows up through the heater section and out the top by natural convection. No circulation fans are required and the air flow rate is established by an orifice plate at the bottom of the recombiner. The recombiner is designed to circulate 100 scfm of air through the recombiner and has a power rating of 75 kilowatts. The above reports describe the recombiner and the various tests that have been conducted.

WCAP-7709-L and WCAP-7820 present the analytical basis for selection of the design requirements and a description and results of the proof-of-principle tests which demonstrated the basic feasibility of the thermal recombiner. These tests were performed by flowing various mixtures of air, nitrogen, and hydrogen through a tubular assembly containing an electric resistance heater to determine heater gas temperature limits and recombination efficiency. The results of these proof-of-principle tests showed a recombination efficiency of essentially 100% was obtained for heater gas outlet temperatures greater than approximately 1150°F and that the recombination efficiency was not affected by gas mixture composition over the range of interest.

The description of the electric hydrogen recombiner and the test program for the proof-of-principle tests are repeated in WCAP-7709-L, Supplement 1 and WCAP-7820, Supplement. These reports also describe the tests that were conducted on the full-scale prototype recombiner. The tests were conducted in a silo type of facility to simulate an actual PWR containment building. A spray system was provided in the top of this building and fans were utilized in some tests to simulate various air currents around the recombiner.

The following type of tests were conducted on the full-scale prototype recombiner:

a. Air tests to establish the natural convection flow characteristics of the recombiner and to measure internal temperature.

b. Air and hydrogen tests to determine the recombiner electric power requirements and operating temperature for a PWR containment.

c. Tap water (with and without hydrogen) and a 24-hour sodium tetraborate (with hydrogen) spray tests to confirm that the containment spray would have no significant effect on the ability of the recombiner to function properly.

d. Steam tests to confirm that steam would have no significant effect on the recombiner operations.

e. Air current tests utilizing fans to determine the effect of various air currents on the performance of the recombiner and to check for any tendency for recirculation.

The results of these tests showed that the prototype recombiner performed satisfactorily.

WCAP-7709-L, Supplement 2 and WCAP-7820, Supplement 2 describes the tests conducted on the production unit electric recombiners. The production recombiner is essentially the same as the prototype except for some minor design changes. The following types of tests were conducted:

a. Air flow tests on three units and temperature distribution tests on five units. These two tests were performed on a production recombiner to demonstrate that the orifice configuration which controls the air flow through the recombiner was correct and permitted a minimum of 100 scfm of air flow and that the temperature in the recombiner reached 1150°F.

b. Thermal cycle tests were conducted to prove the recombiner can sustain repeated cycling during normal service life. The thermal cycling is expected due to periodic in-plant heatup tests to demonstrate availability of the recombiner.

c. Seismic tests to demonstrate the adequacy of the recombiner to perform their intended purpose following an earthquake. Vibration testing was chosen as the method for verifying the performance of the equipment under earthquake conditions. The equipment tested included both the prototype and production recombiner, power supply and control panel.

d. Containment environment tests were conducted to demonstrate that the recombiner will function properly in the containment post-LOCA pressurized steam and spray environment. A secondary purpose was to estimate the amount of reserve life left in the recombiner system. The test facility consisted of a large pressure vessel, boiler and control devices. Various equipment that had been subjected to 80 heatup and cooldown thermal cycles were also tested. Heaters were tested at high pressure, at moderate pressures, and at low pressures with contianment spray added to the steam. Tests were conducted using both sodium tetraborate and sodium throsulfate spray with steam. After six simulated post-LOCA pressure transients, no functional failure was produced. The heater banks were completely disassembled and tested. Visual inspection indicated that 11 out of 240 heater elements showed nondisabling sheath damage at the cold end. To confirm that the sheath splits occurred after a number of simulated post-LOCA transients, the steam chamber tests were repeated on another set of four heater banks. No damage and no clad splits were found after the first post-LOCA transient. To confirm the reserve life left after a post-LOCA transient, these heater banks were subjected to a series of further transients that showed that at least four post-LOCA pressure transients are required to initiate this type of nondisabling damage.

e. Ground fault tests were conducted to demonstrate that a single ground fault in the system will not result in failure of the recombiner.

f. Irradiation tests were performed to demonstrate that the electrical components in the recombiner will perform their functions after irradiation. All components except one were preaged by subjecting them to 80 heatup and cooldown thermal cycles and then all components were subjected to six post-LOCA steam pressure and spray cycles.

Tests confirmed that the electrical components of the recombiner will withstand and perform satisfactorily after exposure to radiation levels up to $2x10^8$ rods. These reports contain two appendices; one which describes the electric hydrogen recombiner and the other elaborates on certain topics which have been covered in earlier reports.

All tests shown that the production electric hydrogen recombiner with its associated equipment will satisfactorily perform its intended functions.

WCAP-7709-L, Supplement 3 and WCAP-7820, Supplement 3 describes the long-term tests that were conducted on a production recombiner. The following three separate tests were performed:

a. High temperature heater test on 12 production heater elements that were inserted into a special constructed oven with their cold ends protruding through the oven wall. This was to simulate the recombiner heater bank in the recombiner heater frame. This test demonstrated that the heaters will perform satisfactorily at temperatures much in excess of their requirements. The test was conducted for 21 days.

b. Long-term recombiners and heater element tests were performed on a production recombiner for 60 days. This test demonstrated that the recombiner will operate successfully at temperatures well in excess of those expected after a LOCA with four percent containment hydrogen for an extended period of time.

c. Long-term steam chamber tests were conducted in the same test facility as used in previous pressure transient tests. This facility consists of a large pressure vessel, boiler, and control devices. Two heater banks that were subjected to one containment LOCA pressure transient in which at the end of 20 hours the pressure was reduced to 20 psia and held for 20 days. One heater bank was energized 24 hours after the simulated LOCA and the controls set to 100% power for 20 days. The test demonstrated satisfactory operation of the heaters in a post-LOCA steam atmosphere.

WCAP-7709-L, Supplement 4 and WCAP-7820, Supplement 4 describes two tests that were performed on the production recombiner to confirm results obtained on earlier tests of the prototype recombiner that was reported in Supplement 1. The two tests were:

a. A hydrogen test to confirm the production recombiner will perform its intended function. This test was conducted in the silo test facility with a 4.6 v/o hydrogen atmosphere.

b. A spray test was conducted by spraying sodium tetraborate spray on the recombiner while it was operating at the recombiner temperature. This test was run for ten days and confirmed the two days test on the prototype recombiner.

SUMMARY OF REGULATORY EVALUATION

The results of the tests conducted on the prototype and production recombiner demonstrated that the recombiner should be capable of controlling the hydrogen concentration in a post-LOCA PWR containment environment. Review of instrumentation, controls and the seismic analysis of the prototype and the production unit will be conducted by the Electrical and Instrumentation Branch, and the Mechanical Engineering Branch, respectively, as part of the review of license application of the plants at which these units are to be installed.

REGULATORY POSITION

We have concluded that the Westinghouse's electric hydrogen recombiner is acceptable as part of the combustible gas control system to control the hydrogen concentration in PWR containment buildings as required by Regulatory Guide 1.7. The above topical reports (both proprietary and non-proprietary) should be referenced for specific plants that are being reviewed for license applications. The staff does not intend to repeat its review of WCAP-7709-L and its supplements when it appears as a reference in a particular license application except for the instrumentation, controls and seismic capability of recombiner.

Should Regulatory criteria or regulations change, such that our conclusion concerning these topical reports are invalidated, you will be notified and given the opportunity to revise and resubmit your topical report for review, should you so desire.

The non-proprietary versions present an adequate representation of the proprietary reports.

MAY 0 1 1975

Mr. C. Eicheldinger, Manager
Nuclear Safety Department
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Eicheldinger:

The NRC staff has completed its review of the following Westinghouse Electric Corporation reports:

1. WCAP-7709-L (Proprietary) and WCAP-7820 (Non-proprietary) entitled, "Electrical Hydrogen Recombiner for Water Reactor Containments,"

2. WCAP-7709-L Supplement 1 (Proprietary) and WCAP-7820 Supplement 1 (Non-proprietary) entitled, "Electric Hydrogen Recombiner for PWR Containments,"

3. WCAP-7709-L Supplement 2 (Proprietary) and WCAP-7820 Supplement 2 (Non-proprietary) entitled, "Electric Hydrogen Recombiner for PWR Containments Equipment Qualification Report,"

4. WCAP-7709-L Supplement 3 (Proprietary) and WCAP-7820 Supplement 3 (Non-proprietary) entitled, "Electric Hydrogen Recombiner for PWR Containments Long Term Tests,"

5. WCAP-7709-L Supplement 4 (Proprietary) and WCAP-7820 Supplement 4 (Non-proprietary) entitled, "Electric Hydrogen Recombiner for PWR Containments."

A summary of our evaluation is enclosed.

Report WCAP-7709-L, as supplemented, satisfies the criteria of our "Elements of the Regulatory Staff Topical Report Review Program" dated August 26, 1974 and thus is considered a topical report. We consider WCAP-7820, as supplemented, an acceptable non-proprietary version of WCAP-7709-L, as supplemented. When either of these reports is used as a reference, both the proprietary report and its non-proprietary version must be referenced.

As a result of our review, we have concluded that WCAP-7709-L, as supplemented, describes an acceptable design and environmental and seismic qualification for the prototype and production models of the Westinghouse electric hydrogen recombiner. Therefore, WCAP-7709-L, as supplemented, may be referenced in license applications as an accepted topical report when used to support this conclusion. We note, however, that for plants that are required to perform environmental qualification in accordance with IEEE Std. 323-1974, we shall require that the electric hydrogen recombiner also be qualified to this standard.

We do not intend to repeat our review of WCAP-7709-L and its supplements and WCAP-7820 and its supplements when they appear as references in a particular license application, except to assure that the material presented in these reports is applicable to the specific plant involved.

In accordance with established procedure, we request that within three months of receiving this letter, you issue revised versions of WCAP-7709-L and its supplements and WCAP-7820 and its supplements to include this acceptance letter.

If you have any questions about our evaluation of this report, please contact us.

Sincerely,

Original signed by
D. B. Vassallo

D. B. Vassallo, Chief
Light Water Reactors Project Branch 1-1
Division of Reactor Licensing

Enclosures:
Topical Report Evaluation

ENCLOSURE

TOPICAL REPORT EVALUATION

Report and Date: WCAP-7709-L (Proprietary), "Electrical Hydrogen Recombiner for Water Reactor Containments" (July 1971)

WCAP-7820 (Non-proprietary), "Electrical Hydrogen Recombiner for Water Reactor Containments" (December 1971)

WCAP-7709-L Supplement 1 (Proprietary), "Electric Hydrogen Recombiner for PWR Containments" (April 1972)

WCAP-7820 Supplement 1 (Non-proprietary), "Electric Hydrogen Recombiner for PWR Containments" (May 1972)

WCAP-7709-L Supplement 2 (Proprietary), "Electric Hydrogen Recombiner for PWR Containments Equipment Qualification Report" (September 1973)

WCAP-7820 Supplement 2 (Non-proprietary), "Electric Hydrogen Recombiner for PWR Containments Equipment Qualification Report" (October 1973)

WCAP-7709-L Supplement 3 (Proprietary), "Electric Hydrogen Recombiner for PWR Containments Long Term Tests" (January 1974)

WCAP-7820 Supplement 3 (Non-proprietary), "Electric Hydrogen Recombiner for PWR Containments Long Term Tests" (February 1974)

WCAP-7709-1 Supplement 4 (Proprietary), "Electric Hydrogen Recombiner for PWR Containments" (April 1974)

WCAP-7820 Supplement 4 (Non-Proprietary), "Electric Hydrogen Recombiner for PWR Containments" (May 1974)

Originating Organization: Westinghouse Electric Corporation

Reviewed By: Electrical, Instrumentation and Controls Systems Branch, Mechanical Engineering Branch and Containment Systems Branch, Office of Nuclear Reactor Regulation

MAY 0 1 1975

Summary of Topical Report

Westinghouse Electric Corporation has developed an electric hydrogen recombiner as part of the combustible gas control system to control hydrogen concentration within a pressurized water reactor containment following a loss-of-coolant accident. The recombiner consists essentially of a thermally insulated vertical metal duct with a metal sheathed electric resistance heater provided to heat a continuous flow of containment gas mixture up to a temperature (1150°F) which is sufficiently high to react the hydrogen and oxygen. The gas mixture enters the recombiner and flows up through the heater section and out the top by natural convection. No circulation fans are required and the air flow rate is established by an orifice plate at the bottom of the recombiner. The recombiner is designed to circulate 100 standard ft^3/min of air and has a power rating of 75 kilowatts.

Report WCAP-7709-L describes (1) the initial development of the electric hydrogen recombiner (2) testing on the prototype recombiner (3) qualification testing on the production recombiner (4) long term tests on the production recombiner and (5) confirmatory tests on the production recombiner.

Staff Evaluation

The results of the tests performed demonstrated that the Westinghouse electric hydrogen recombiner should perform satisfactorily for the intended service conditions. One exception taken by the staff to the test results was the seismic test described in Section 3.3 of WCAP-7709-L Supplement 2. Section 3.2.2 of IEEE Std 344-1971 - "Trial-Use Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations" states "The device being tested should demonstrate its ability to perform its intended function ... before, during, and following the test". The hydrogen recombiner was not energized during the vibration test. It was our concern that the heater elements are more likely to be subject to failure from vibration when they are energized because of potential thermal stresses than when not energized. It was not demonstrated that the hydrogen recombiner had the capability to perform its intended function following a seismic event, given that it had been energized during the event. In response to this concern, Westinghouse provided additional information which included a seismic analysis to demonstrate the adequacy of the electric hydrogen recombiner heater elements for seismic conditions when the recombiner is operating. This analysis has been found to be acceptable. However, it should be noted, that acceptance of these reports does not provide generic acceptance of the Westinghouse vibration testing philosophy. Whereas the use of the sine beat - single axis vibration testing is adequate for the particular items of equipment described in the referenced reports, other items may require different techniques.

MAY 0 1 1975

Neither IEEE Std 344-1971 mentioned above nor IEEE Std 323-1971 "General Trial-Use Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations" were referenced in WCAP-7709-L for qualifying the recombiner for service inside the containment following a LOCA. However, the report does contain sufficient information to conclude its acceptability on the basis of the requirements of the above standards.

The recombiner outlet temperature ranges from about 1150°F, a temperature sufficient for recombination, to 1450°F associated with rated sheath temperature of 1600°F. The test temperature exceeded the recommended maximum sheath temperature for the heater elements. The maximum allowable power that resulted in a sheath temperature of 1600°F was not noted. Westinghouse subsequently provided this power level as approximately 66Kw. This is the upper operating limit of the power supply.

The power supply for the recombiner consists of a 3 phase 75 KVA transformer, silicon-controlled rectifiers and control circuitry. The instrumentation for the recombiner is contained in a control panel and like the power supply is located outside the containment. The panel is used to control the power supply and to read out temperature from the three thermocouples located in the recombiner. The instruments mounted on the panel include a power meter, thermocouple readout, potentiometer, off-on switch, and power available light. The environmental limits for the control panel and power supply were not stated in the original submittal. The environmental conditions for which the power supply and control panel have been designed were subsequently provided.

Staff Position

We find that WCAP-7709-L and its supplements provide an acceptable design and environmental and seismic qualification for the prototype and production models of the electric hydrogen recombiner. For plants that are required to perform environmental qualification in accordance with IEEE Std. 323-1974, we shall require that the electric hydrogen recombiner also be qualified to this standard. These plants will require more documentation, than is supplied in WCAP-7709-L and its supplements, in their applications.

We find WCAP-7820 an acceptable non-proprietary version of WCAP-7709-L.

MAY 0 1 1975

JUN 22 1978

Mr. Thomas M. Anderson, Manager
Nuclear Safety Department
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Anderson:

SUBJECT: EVALUATION OF WCAP-7709L, SUPPLEMENTS 5, 6, AND 7

We have completed our review of Westinghouse Electric Corporation report Supplements 5, 6, and 7 to WCAP-7709L (Proprietary) and WCAP-7820 (Non-Proprietary) entitled "Electric Hydrogen Recombiner for PWR Containments". Our evaluation is enclosed.

As a result of our review, we have concluded, subject to the conditions in our enclosed evaluation, that the Westinghouse electric hydrogen recombiner is acceptably qualified for the seismic and environmental conditions identified in Supplements 1 through 7 of WCAP-7709L in accordance with the requirements of IEEE 323-1974. Applications using the Westinghouse recombiner must include in their Final Safety Analysis Report information to demonstrate either (1) that accident environmental conditions and plant seismic response spectrum are either within the accepted envelope conditions in WCAP-7709L or (2) that the recombiner is acceptably qualified on some other analytical or experimental basis.

Accordingly, topical report WCAP-7709L and its Supplements 1 through 7 are acceptable for reference in license applications. Topical report WCAP-7820 and its Supplements 1 through 7 is an acceptable non-proprietary version of WCAP-7709L. When either of these reports is used as a reference, both the proprietary report and the non-proprietary version must be referenced.

In accordance with established procedures, it is requested that Westinghouse issue revised versions of these reports within three months of receipt of this letter to include this acceptance letter, the enclosed evaluation, and any changes resulting from our review.

We do not intend to repeat our review of these reports when they appear as references in a particular license application except to assure that the material presented in these reports is applicable to the specific plant involved.

Should Nuclear Regulatory Commission criteria or regulations change, such that our conclusions concerning these reports are invalidated, you will be notified and given an opportunity to revise and resubmit your topical reports, should you so desire.

Sincerely,

Original Signed by
John F. Stolz
John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management

cc: Mr. D. Rawlins
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

ENCLOSURE

SAFETY EVALUATION REPORT

SUPPLEMENTS 5, 6, AND 7 OF WCAP-7709-L
"ELECTRICAL HYDROGEN RECOMBINER LWR CONTAINMENT"

Summary of Topical Report

Westinghouse Electric Corporation has developed and tested an electric hydrogen recombiner to limit hydrogen concentration within a pressurized water reactor containment following a loss-of-coolant-accident. This recombiner is located inside the containment and consists of a metal sheathed electric resistance heater provided to heat a continuous flow of containment gas mixture to about 1150°F. At this temperature hydrogen reacts with oxygen in the environment to form steam, thereby reducing the hydrogen content in the containment atmosphere. The control panel and power supply are located outside the containment.

The recombiners are designed to be permanently installed inside of containment and are not intended to be used for sharing between two or more units. Therefore the design criteria for these recombiners do not take into account vibratory and impact loads that would be imposed during transportation in addition to the loads that would be imposed during a seismic event.

WCAP-7709 L provides a description of the electric hydrogen recombiner, design criteria, design bases and performance analyses. Supplement 1 to WCAP-7709 L provides a description, analysis and results of performance tests of a prototype recombiner under conditions simulating post-LOCA conditions inside containment. Supplement 2 to WCAP-7709 L provides a description, analysis and results of tests to qualify the recombiner for seismic loads and loss-of-coolant-accident environments. Supplement 3 provides a description, analysis and results of long term tests of the electric heater elements in air (60 days) and in a post-LOCA steam environment (21 days). Supplement 4 provides a description, analysis, and results of performance tests of a production unit to demonstrate its capability to operate when sprayed with sodium tetraborate and to successfully recombine hydrogen and oxygen.

The staff has previously reviewed WCAP-7709 L through Supplement 4, and found the Westinghouse recombiner functionally acceptable for use in nuclear power plants. In addition, environmental and seismic qualification was found to be acceptable based on the requirements of IEEE 323-1971, "General Trial - Use Guide for Qualifying Class IE Electrical Equipment for Nuclear Power Generating Stations" and IEEE 344-1971, "Trial-Use Guide for Seismic Qualification of Class IE Electrical Equipment for Nuclear Power Generating Stations". Our safety evaluation was transmitted to Westinghouse by letter dated May 1, 1975 from D. B. Vassallo to C. Eicheldingen. In that evaluation we concluded that additional documentation would be required for plants committed to meet IEEE-323-1974 "IEEE Standard for Qualification of Class IE Electrical Equipment for Nuclear Power Generating Stations". This standard includes both seismic and environmental qualifications.

Supplements 5, 6, and 7 to WCAP-7709-L provide additional documentation to demonstrate conformance of the Westinghouse electric hydrogen recombiner to the requirements of IEEE 323-1974. Supplement 5 provides the results of tests to demonstrate design margin, capability to withstand containment leakage tests, and capability to operate during an earthquake. Supplement 6 compares the tests and analyses performed for the recombiner with the requirements in IEEE 323-1974 to demonstrate conformance. Supplement 7 provides results and analyses of additional tests to demonstrate acceptance of auxiliary equipment for the recombiner (power supply, control panel, power cables, cold reference junction box, and automatic temperature controller).

Our evaluation of Supplements 5, 6, and 7 to WCAP-7709-L are provided below.

Summary of Regulatory Evaluation

Information in Supplements 5 and 6 is intended to show that the Westinghouse electric hydrogen recombiner is in conformance with IEEE 323-1974. Type testing (recommended in IEEE 323-1974 as the preferred method), was primarily used to qualify the Westinghouse recombiner. The tests and analyses performed by Westinghouse adequately, demonstrate that the recombiner, excluding the control panel and power supply, meets the following specific requirements of IEEE 323-1974.

1. The equipment shall be operated to the extremes of performance and electrical characteristics. . . The recombiner was operated at higher than normal temperatures (1450°F versus the normal operating temperature of 1200°F). We noted in our May 1, 1975 evaluation that 1450°F gas temperature corresponded to a maximum sheath temperature of 1600°F (rated sheath temperature) and that this temperature was achieved with 66 kilowatts power supplied to the heaters.

   In Supplement 5, additional over-temperature tests were successfully run with the heater at maximum power level and sheath temperatures up to 1750°F. We conclude based on the tests, that the heaters will operate satisfactorily with the maximum power of 75 kilowatts supplied to the recombiner.

2. Equipment shall be aged in accordance with Section 6.3.3 of IEEE 323-1974 to put it in a condition which simulates its expected end-of qualified life condition . . . The recombiner inside containment is composed primarily of metallic structural material, metal-enclosed thermal insulation, metal clad ceramic heater elements, and power cables. Since the recombiner is in a normal containment atmosphere and subjected to periodic testing, Westinghouse concluded that the most significant aging factor was the fatigue life of the structure, due to thermal stresses induced by the periodic heat up and cool down tests (i.e., the recombiner would not deteriorate significantly due to normal atmospheric conditions alone). The recombiner structure was subjected to 80 thermal cycles, corresponding to 40 years of expected periodic testing, and was found to be in good operating condition.

   We conclude that the recombiner structure was satisfactorily tested to demonstrate acceptable end of life condition. The power cable inside containment was tested in accordance with IEEE Std 383-1974 and after reviewing the details of the tests performed, we conclude that the irradiation, steam, and alkaline spray conditions were sufficiently severe and the cables were acceptably qualified.

3. The aged equipment shall be subjected to mechanical vibration. . .

   The Mechanical Engineering Branch has evaluated the mechanical vibration tests conducted on the "aged" equipment. The concept of aging was addressed explicitly for the first time in IEEE-Std. 323-1974. The aging guidance therein reflects the requirements of IEEE Std. 279-1971 Sec. 4.4. The objective of aging is to put samples in a condition equivalent to the end-of-life condition.

   For the initial seismic tests reported in WCAP-7709-L, Supplement 2, it was assumed that the recombiner is in the de-energized mode since, for PWR containments, the recombiners are not energized for approximately 24 hours after the DBA. A seismic analysis of the recombiner heater element is presented in Appendix B of Supplement 5 to WCAP-7709-L which demonstrates analytically that the recombiner would function adequately under seismic conditions while it is energized and is in operation. In this analysis the natural frequency of the heater elements are calculated to be 250.5 cps for built-in ends and 112.0 cps for simply supported ends. Static loadings equal to 5.6g horizontal and 2.5g vertical (1.5g seismic + 1g weight) are applied in the analysis. The stresses are determined to be 1322 psi and 607 psi in the horizontal and vertical directions, respectively, which are much less than the yield strength of 13500 psi for Incoloy 800 tubing at 1600°F. This tubing forms the metal cladding of the heater element assembly and since it is the most highly stressed part of the assembly, heater elements are acceptable for the hot seismic condition. The midspan deflections and the clearance between heater elements and holes in the separation plates have also been analyzed and shown to have a negligible effect on recombiner performance.

An additional vibration test of a production recombiner is described in WCAP-7709-L, Supplement 5 in which the recombiner was energized and at temperature before, during and after the vibration test. This test confirms the analysis of the heater elements discussed earlier. The equipment was vibrated in 3 directions, horizontal side-to-side, horizontal front-to-back and vertical. The recombiner was maintained at 1250°F throughout and after the test. The test input was of the sine beat wave form type and was performed at resonant frequencies, determined by a frequency search test performed from 1 to 35 Hz plus additional frequencies described in the report. The test method used is a single frequency method (described in IEEE 344-75 Section 6.6.2.3). The single frequency sine beat method is justified for this application on the basis that the resonances are widely spaced and do not interact to reduce the fragility level, as permitted in Section6.6.2 of IEEE 344-75. The single axis test is justified on the basis that the tests conservatively reflect the seismic loadings at the equipment mounting locations. A commitment is made in the report that for each plant application, the required seismic response spectrum for that plant will be checked against the test response spectrum to verify that the test response spectrum envelopes the required response spectrum. This is consistent with the requirements of Regulatory Guide 1.100.

4. The aged equipment shall be operated while exposed to a simulated DBA. . .

   A series of tests were performed on the portion of the production recombiner that is located inside the containment, including several post-LOCA pressure transients (69 psia, 302°F) and long term steam tests to demonstrate that the recombiner can successfully withstand the post-LOCA environment. In addition, alkaline solution was sprayed on the recombiner during operation. These tests have been accepted by the staff for qualification of the Westinghouse electric hydrogen recombiner because the recombiner has no temperature sensitive electrical components required to operate during the portion of the post-LOCA pressure transient wherein high temperatures exists and the maximum expected steam temperature following a steam line break (420°F) is not likely to cause structural failure of the recombiner.

5. The equipment shall be operated while exposed to the simulated post-accident conditions. . . To show the long term capability of the heater banks to operate in the post-LOCA environment, two heater banks were subjected to a DBA plus 12 months of simulated post-LOCA environment. The test showed that the individual heater elements and banks plus thermocouples, electrical cabling, and thermocouple junction boxes which are susceptible to steam would perform satisfactorily.

Supplement 7 to WCAP-7709-L is the last in the series of reports for the Westinghouse electric recombiner and contains qualification results for the recombiner power cable located inside containments, the recombiner control and power supply panels located outside containments, and additional optional features including a cold reference junction box and an automatic temperature control device which may be selected by an applicant.

The qualification of the control panel and power supply located outside the containment does not meet our interpretation of the aging requirements set forth in IEEE Std 323-1974. However, tests performed on the control panel and power supply located outside the containment included short-term high temperature exposure (10 days at 155°F for the control panel and 10 days at 135°F for the power supply). We found the qualification of the control panel and power supply acceptable, based on these tests and also based on the accessibility of these components for repair following a LOCA. The recombiner will not be needed for several days following a LOCA and since these components will be easily accessible, repair of components that may fail can be accomplished.

Seismic tests of the control panel and power supply were performed to demonstrate conformance to IEEE 344-1975 "Recommended Practice for Seismic Qualification of Class IE Equipment for Nuclear Power Generating Stations". IEEE 344-1975 recommends that seismic tests be performed using biaxial motion and both random frequency and sine beat input. The power supply and control panels were mounted on the drive plate of a vibration table and energized. The test series consisted of resonance frequency search plus five OBE's followed by an SSE. The input for the five OBE's was a biaxial, random frequency while the SSE was a biaxial sine beat input, the maximum "g" level being 0.2. The magnitude of the vertical acceleration was kept to two-thirds the magnitude of the horizontal acceleration. The input was made of decaying sinusoids covering the frequency range of 1.25 to 3.50 Hz. The sine beat test was performed at each resonance frequency and at eleven other frequencies ranging between 1.25 and 33.5 Hz. These tests were run four times (once for each equipment mounting direction) without component failure. We find these tests acceptable.

The power cables for the recombiner were tested along with the heater banks in the post LOCA steam and spray environment and seismically tested with the recombiner. The testing did not completely conform to the procedure outlined in IEEE 383-1974, "Standard for Type Tests of Class IE Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations". To meet the requirements of Section 2.4 of this standard, which deals with environmental exposure, a series of tests were performed on the power cables which included thermal aging, irradiation, post LOCA containment steam and spray exposure and voltage tests. We find these tests acceptable.

The cold reference junction box is for use in those containments which have copper conductors through containment penetrations already installed. The usage of a compensator in the junction box allows the chromel-alumel leads from the recombiner to be connected to copper leads inside the junction box. The copper leads can then be run through a typical copper penetration to the control panel, thus eliminating the need to replace installed copper penetrations with chromel-alumel penetrations. The cold reference junction box, with the exception of the compensator, has been tested for the same range of conditions as the tests that were performed on the recombiner. The compensator itself was irradiated and placed in a steam environment for a short period of time. Since the compensator (a wire-wound resistor encapsulated in a ceramic type material) does not have temperature sensitive elements in it and since the compensator is used only to provide the operator with an approximation of the temperature of the heater inside the recombiner and has no control functions, we find the qualification tests of the cold reference junction box to be acceptable.

The automatic temperature control feature is an option which allows the power level to be controlled by feedback signal from the recombiner thermocouples. It consists of minor wiring modifications within the control panel and addition of a printed cicuit card to the temperature indicator. Because the changes that would have to be made in the design of the control panel to add the automatic temperature control feature are minor, we find this concept acceptable from a qualification standpoint. However, the use of this device to control a recombiner system that also incorporates the cold reference junction box would mean that a compensator in the junction box would be relied upon for control purposes. To alleviate this problem Westinghouse has agreed not to allow the use of the automatic temperature control device except during periodic tests for those plants that choose to use the cold reference junction box. We find this approach acceptable.

Regulatory Position

Based on our review of WCAP-7709-L, we have concluded as follows:

(1) The Westinghouse electric hydrogen recombiner, (excluding the control panel, power supply and the optional automatic temperature control and cold reference junction) meets the requirements of IEEE 323-1974.

(2) The control panel and power supply are acceptable on the basis of high temperature exposure tests and also because there would be adequate accessibility and time for repair, if necessary, following a loss of coolant accident before they would be required to operate.

(3) The recombiner, control panel and power supply meet the requirements of IEEE-344-1975.

(4) Power cables meet the requirements of IEEE-383-1974.

(5) The optional automatic temperature control feature is acceptable for use on all plants except those which use the cold reference junction box. For plants using the cold reference junction, automatic temperature control may be used for periodic tests but must be disconnected at other times during plant operation.

(6) The cold reference junction box is acceptably qualified to provide approximate heater temperature indication to the operator; however, it is not qualified for control functions.

Westinghouse report WCAP-7709-L and Supplements 1 through 7 may be referenced in applications to support the above conclusions where the calculated accident environmental conditions and plant seismic response spectrum are enveloped by the conditions for which the recombiner is qualified. Each application referencing this topical report shall either include information to demonstrate that environmental and seismic conditions for that plant fall within the accepted envelope conditions of WCAP-7709-L, or provide further analyses or tests to demonstrate acceptability.

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JUDGE WOLFE: Is there cross, Mr. Copeland?

MR. COPELAND: No.

JUDGE LINENBERGER: I have a couple of clean-up things here.

First off, is it really 40 or is it 34, the contention number? I thought I detected some confusion on this, inconsistency let's say.

MR. DOHERTY: Yes, I have seen this, too. I meant to ask that.

MR. COPELAND: I believe it is 34, Your Honor.

JUDGE LINENBERGER: I have the impression it is.

MR. DEWEY: I think this is true now. I see Mr. Fields' previous affidavit was entitled Contention 34, too, so with the Board's leave, we would like to amend the cover page of Mr. Fields' testimony to read, "TexPirg Contention 34," rather than "TexPirg Contention 40."

JUDGE LINENBERGER: Okay. Now then, Mr. Dewey --

MR. COPELAND: Your Honor, I might note that I think there was a typographical error in the Board's order.

JUDGE LINENBERGER: I think you are right, Mr. Copeland. Our order of September 1, 1981, second order ruling on summary disposition also mislabeled that contention.

MR. COPELAND: I think we picked that up and started scheduling matters and writing testimony.

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JUDGE LINENBERGER: Mr. Dewey, the next thing I wanted to understand is that addressing only for the moment the prefiled direct testimony separate from its attachments, the TexPirg contention is mentioned a couple of times on the first page and the contention itself is quoted on the first page of that testimony, but all of the succeeding portions of that testimony go exclusively to questions the Board raised when that contention was admitted.

Now, is it the position of the Staff that Mr. Fields in answering the Board's questions, as this prefiled testimony seems to do, also addresses the Staff's position with respect to the contention itself?

MR. DEWEY: Well, no, sir. I think that our position here would be were the only items that were left were the Board's concern and so this is all we put in.

If the Board prefers, we will -- the other parts of his testimony would be considered the affidavit. We have not included the affidavit with this testimony.

If the Board would prefer, we would at this time also include the affidavit as part of his testimony; that was filed previously by Mr. Fields.

JUDGE LINENBERGER: Before I answer that, I should like to just quickly re-read what our September 1 order said.

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(Bench conference.)

JUDGE WOLFE: We will have a 15-minute recess until a quarter of 11:00.

(Recess taken.)

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JUDGE WOLFE: The Board has conferred with regard to your question, Mr. Dewey --

MR. DEWEY: Yes, sir. Could I add one thing that I don't think I related during our previous discussion? And that is, I guess the main reason why we didn't add Mr. Fields' affidavit with this testimony was that we felt that the subject matter was sufficiently covered in the Applicant's witness' testimony regarding this subject matter.

Therefore, we just addressed the Board's concerns. However, we are prepared at this time, if you so desire, to give you copies -- to submit copies of the affidavit which would cover that.

JUDGE WOLFE: Yes. Well ...

(Bench conference.)

JUDGE WOLFE: No, we do not require or ask that Mr. Fields' affidavit attached to Staff's Motion for Summary Disposition of this contention be incorporated into the record as if read.

As indicated in our September 1, 1981 Order, we indicated we didn't think the affidavit was adequate. And along these lines, Mr. Doherty, we would indicate that -- we will first tell you, Mr. Doherty, as you're well aware, that Mr. Fields' testimony that has now been incorporated into the record addresses several questions that were posed by the Board in its September 1, 1981 Order. These were minimal questions that we suggested that Staff and/or Applicant should address.

Most certainly, and obviously, you may cross-examine this witness on his answers to these several questions posed by the Board.

In addition, you may ask questions of this witness that perhaps are directed to matters that are outside of the Board's questions, but are still within the framework of the TexPirg contention.

In other words, your questions are not restricted to the direct testimony of this witness. You may ask questions of this witness within the framework, within the scope of the contention itself.

Do I make myself clear, Mr. Doherty?

MR. DOHERTY: Yes, you do, sir.

JUDGE WOLFE: All right. Is there cross, Mr. Copeland?

MR. COPELAND: Yes, sir.

CROSS-EXAMINATION

BY MR. COPELAND:

Q. Mr. Fields, at the time you prepared your testimony, did you have available to you the testimony of the Applicant's witnesses on this contention?

A. Yes, I did.

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Q Did you review that testimony prior to the time you wrote your own testimony?

A Yes, I did.

Q In your opinion, was the Applicant's testimony dispositive of the contention?

A Yes.

Q Do you feel like in answering the Board's questions that you have also disposed of the contention, as well as having addressed the Board's questions?

A Yes.

Q Would you explain why, sir?

A Basically, the TexPirg contention deals with the inadequacy -- potential inadequacy of the hydrogen monitoring system.

By responding to the Board's questions on the capability of the monitoring system, on the capability of the recombination system, I felt that it also addressed the contention question.

Q Do you believe there is any merit to TexPirg's contention?

A No.

MR. COPELAND: Thank you. That's all I have.

JUDGE WOLFE: Is there cross, Mr. Doherty?

MR. DOHERTY: Yes, Your Honor.

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CROSS-EXAMINATION

BY MR. DOHERTY:

Q Mr. Fields, was there a hydrogen explosion at Three Mile Island?

A There was a hydrogen burn at Three Mile Island. Whether or not the burn speeds reached supersonic speeds is something that was not able to be determined.

Q Would you say that the fact that it was impossible to determine whether there was an explosion, which is the word I use, or a burning is an indication that monitoring equipment in the containment was adequate?

A Are you asking me if the monitoring equipment that was at TMI was adequate to detect the incident that occurred?

Q You stated in your previous answer that there's still today an inability to determine whether there was a burn or an explosion of hydrogen. I believe you said that. Is that right?

A That is correct.

Q What I'm asking is -- Well, let me ask this first. There is the word in the contention, "monitoring," referring to monitoring apparatus and also -- well, it does use the word, "monitoring."

What does that conjure up in your mind,

"monitor"?

A. Monitoring would be following that particular parameter as it changes and measuring that parameter.

Q. Have you ever inquired as to what hydrogen monitoring apparatus there was at Three Mile Island?

A. Yes.

Q. Have you inquired as to what hydrogen monitoring apparatus there will be at Allens Creek?

A. Yes.

Q. Is there any difference?

A. There is considerable difference.

Q. What difference?

A. Basically, the monitoring system that was available at TMI-2 required the operator to manually take a sample from the containment air space, carry the sample to another room in which measurements were made of the hydrogen concentration.

At Allens Creek this will be done automatically by two redundant systems, which have the ability to take samples from eight different locations within the containment.

This will be done remotely from the control room and does not require any transportation by physical -- by personnel.

Q. Can that type of apparatus detect a hydrogen explosion?

A. This hydrogen is designed to measure hydrogen levels. A hydrogen explosion would be detected by the changes in the pressures and temperatures inside the containment.

Q. Okay. Now --

JUDGE LINENBERGER: Excuse me, Mr. Doherty, but there is a point I would like clarification on with respect to the answer to your previous question about differences between Allens Creek and TMI.

Mr. Fields, what you described with respect to TMI-2 sounded as though it was a process of taking what are sometimes called grab samples from the containment atmosphere, taking them somewhere and analyzing them.

THE WITNESS: That is correct.

JUDGE LINENBERGER: What you've described as proposed for Allens Creek sounds to me as though the only difference is that a person does not have to go physically somewhere and get that grab sample and take it somewhere to analyze it, that the sample is being piped from the containment atmosphere to the analyzer, but that otherwise, the decision of when and whether to do an analysis, from what you've said, sounds no different than at TMI-2.

In other words, at Allens Creek, as you have described it so far, somebody still has to make a decision to actuate the analyzer.

He doesn't have to carry the sample for sure, because it's piped. But that to me still leaves a significant similarity between Allens Creek and TMI-2.

Now, do you have something to add to that?

THE WITNESS: Yes. The hydrogen monitoring system will be actuated within 30 minutes of a safety injection signal, and continuously record the hydrogen levels inside the containment throughout the duration of the accident.

JUDGE LINENBERGER: I see. By safety injection signal, is that equivalent to a scram signal?

THE WITNESS: A scram is insertion of the control rods into the core. Safety injection is the actual injection of coolant water into the vessel, which would only come about if your pressure inside the vessel went down for some reason.

JUDGE LINENBERGER: I see. So it's actuation of the ECCS -- to put words in your mouth -- that triggers the initiation of this 30-minute interval following which hydrogen analysis will be automatically performed?

THE WITNESS: Yes. The operator will manually turn on the hydrogen monitoring system 30 minutes after an ECCS actuation. And the monitor will automatically examine

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the hydrogen levels inside containment.

JUDGE LINENBERGER: Now, that's a decisional and seemingly voluntary obligation on the part of the operator. Are you aware of whether there are administrative controls or electronic controls that make it difficult for the operator to forget that he has got to do this 30 minutes later?

THE WITNESS: This procedure of actuating the monitoring system is in the emergency procedures -- or will be in the emergency procedures, which the operator follows after an accident.

JUDGE LINENBERGER: So it is, in a sense, an administrative control?

THE WITNESS: Yes.

JUDGE LINENBERGER: Okay.

Mr. Doherty, I apologize for this long interruption, but I didn't feel we had completely pinned down these differences.

BY MR. DOHERTY:

Q. In the event of a hydrogen explosion, would the hydrogen monitoring system detect a decrease in hydrogen?

A. First of all, the first question is whether the monitoring system will survive the hydrogen explosion. Now, that is not a design basis accident for Allens Creek.

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The Applicant has taken steps to assure that a hydrogen explosion will not occur.

However, to answer your question from a theoretical standpoint, after an explosion or a burn, the hydrogen levels will be reduced. And, therefore, if you do have monitoring systems that are available, they will show a lowering of the hydrogen levels.

Q Well, at Three Mile Island, did they have a pressure monitoring system in the containment building?

A Yes.

Q When this explosion or burn occurred, did that show any change on that?

A Yes. There was a pressure spike measured, which caused containment isolation. I believe it was approximately six hours after the accident began.

Q And is it true that the operators didn't attach any significance to the spike at the time?

A That's not true. I'm not sure exactly what the operators did then, but I'm sure they knew they had a problem.

Q Why are you sure? What makes you feel that way?

A When the operator sees that containment is isolated due to a pressure rise inside containment, he knows that something is not going correctly, whether it's

3-10

due to instrument error or due to a real situation, he still has to take the same procedures to find out what's wrong.

Q. Okay. I think you went from the specific to the general there.

Specifically at Three Mile Island, did the operators identify that something had gone wrong in the containment, as you've described it at that time?

A. When the pressure peak occurred?

Q. Yes.

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A. I'm not sure exactly what the operators were doing at TMI following that period.

Q. Okay. Would you say that as currently designed -- Well, as currently designed, would you say the Allens Creek hydrogen monitoring system is capable of providing unambiguous information that hydrogen has burned or exploded in the containment?

A. The monitoring system is designed to determine the hydrogen levels inside containment, yes. Now, the Applicant has not designed this equipment, nor has the Staff required them to design this equipment to survive a burn or explosion.

Q. Do you know if the Staff is considering making specs on that system such that it would survive burn or explosion?

MR. COPELAND: I object to that, Your Honor. It seems to me to be beyond the scope of the contention. And I think perhaps in light of the witness' last answer, that's dispositive of the whole contention.

It seems to me TexPirg's contention is that you are required to monitor for an explosion. And his answer is that that is not a position of the Commission or the Staff.

And it seems to me that's the end of the discussion.

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MR. DOHERTY: I think the question is relevant, Your Honor. I don't think it disposes of the whole issue.

JUDGE WOLFE: And your question once again, Mr. Doherty?

MR. DOHERTY: I'm afraid I've forgotten it, Your Honor, in the lapse.

MR. COPELAND: His question, Your Honor, was whether the hydrogen monitoring devices have to be designed to withstand the effects of a hydrogen explosion, which is clearly not part of TexPirg's contention.

MR. DOHERTY: Well ...

(Bench conference.)

JUDGE WOLFE: I'll sustain the objection. Certainly that episode is not contemplated within the four corners of this contention.

BY MR. DOHERTY:

Q What is the Commission's position as to what constitutes an adequate hydrogen monitoring system?

A A system that can accurately give the operators information on the hydrogen levels in the containment following an accident.

Q In your opinion, if such a system could not survive a hydrogen burn, do you believe that would be an adequate system for giving hydrogen information to the operators?

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MR. COPELAND: The same objection I had a minute ago, Your Honor.

JUDGE WOLFE: The same ruling. Sustained.

MR. DOHERTY: Your Honor, the contention states that: "TexPirg contends that" -- excuse me. I'm sorry. You've ruled.

MR. COPELAND: Your Honor, I wonder if I might ask the witness a clarifying question here at this point, and maybe it would help speed things up.

JUDGE WOLFE: Mr. Doherty?

MR. DOHERTY: All right, go ahead, counsel.

MR. COPELAND: Mr. Fields, do you know whether in NUREG-0718 there is any requirement to monitor the containment building in order to be able to detect a hydrogen explosion?

THE WITNESS: No.

MR. COPELAND: Well, Your Honor, it seems to me, in light of that fact, that where we are in this contention is the Commission has decided, in adopting its near-term CP requirements in NUREG-0718, that no hydrogen monitor is required to monitor for a hydrogen explosion.

And in light of that, it seems to me that the contention really doesn't mean anything anymore. We are sort of at a point where the Commission has

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decided what is required.

And I think what the witness has done in his testimony is say -- well, he's saying that he doesn't think such is necessary because, you know, the hydrogen monitoring system that is there is going to prevent such a buildup.

But I wonder if we even need to pursue that anymore. Am I making sense?

JUDGE LINENBERGER: You're making sense, Mr. Copeland, but -- and I don't want to sound like a lawyer here -- but really you asked the witness a question: Does he know whether something is in 0718, and his answer was no.

Now, that answer could be interpreted to mean he doesn't know whether it's in there, or that answer could be interpreted to mean that no, it is not in there.

And --

MR. COPELAND: I'm sorry if I asked the question that poorly, Your Honor. I didn't mean to. You're certainly right, if that's the way the answer came out, Your Honor, if that was my question. I didn't think it was.

THE WITNESS: I could rephrase my answer.

MR. COPELAND: Well, it seems to me the

question is: is there a requirement in --

MR. DOHERTY: Excuse me --

MR. COPELAND: -- NUREG-0718 that requires a monitoring system for the detection of a hydrogen explosion.

THE WITNESS: To the best of my knowledge, there is no such requirement in NUREG-0718.

JUDGE LINENBERGER: I just wanted to get that clear.

MR. DEWEY: Well, Your Honor, the Staff doesn't believe that that would be dispositive of this, just because that NUREG doesn't have that requirement. It would not be dispositive of this contention, just because this is not addressed in the NUREG.

JUDGE WOLFE: The NUREG, in other words, is not a regulation?

MR. DEWEY: That's right.

MR. COPELAND: I had understood, Mr. Dewey, that the Commission had adopted that as a regulation.

MR. DEWEY: If that's correct, then I'm incorrect.

JUDGE WOLFE: When, Mr. Copeland?

MR. COPELAND: I can't recall the date now, Your Honor. We've been through this before. I don't recall exactly when it was.

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JUDGE WOLFE: Well, without anymore definitive citation to that effect, the objection is overruled.

MR. COPELAND: Well, I don't believe that --

JUDGE WOLFE: Excuse me. The --

THE WITNESS: The NUREG-0718 --

JUDGE WOLFE: Wait just a moment. Let me re-phrase that.

We heard the statement of the witness and argument of counsel and NUREG-0718 is not a regulation, so you may now proceed, Mr. Doherty, with your cross-examination.

MR. COPELAND: Your Honor, I might say before we present our witnesses, I intend to re-raise that argument; and I will have the citation available, because I think my point is well taken. And I think prior to the time we present our witnesses, I certainly intend to re-raise that point.

MR. DOHERTY: Are you finished?

MR. COPELAND: Yes.

JUDGE WOLFE: I think you had raised that with -- either with respect to that NUREG or another NUREG; you had made such an objection.

MR. COPELAND: I just can't remember, Your Honor.

JUDGE WOLFE: All right, Mr. Doherty.

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MR. DOHERTY: Mr. Chairman, I have gotten -- I must apologize, I lost track of things a little bit.

I asked the witness' personal -- his opinion as to the adequacy of a hydrogen monitoring system which could not survive a hydrogen burn or hydrogen explosion.

And I think that was objected to and sustained at that point. Does that close off that line of questioning? Is that your recollection, that no more questions can be asked about the adequacy of the hydrogen monitoring system to survive a hydrogen explosion or hydrogen burn?

JUDGE WOLFE: That's the Board's ruling.

MR. DOHERTY: Okay. Thank you.

BY MR. DOHERTY:

Q. Is there anything in the Applicant's plans that would prevent hydrogen burns more than what was available at Three Mile Island?

A. Other than what was available at Three Mile Island?

Q. Yes.

A. The systems are a lot different. Three Mile Island had recombiners, outside containment. For Allens Creek there will be inside containment.

The monitoring system is different, as I earlier pointed out. In addition, the -- Allens Creek will have a system in place to mitigate the consequences of a degraded core accident, which TMI did not have whatsoever.

Q [illegible] the Commission set up requirements to mitigate [illegible] -- think -- what did you call them before?

A Degraded core?

Q Degraded core.

-- at this time for the Applicant to follow?

MR. COPELAND: I'll object to the question. We're getting beyond the scope of the contention. The scope of the contention is very narrow; and that is that monitors ought to be present to detect a hydrogen explosion.

MR. DOHERTY: First of all, it was the testimony of the witness that there will be systems available to mitigate core degradation, which is relevant to whether there will be any hydrogen generated or not.

MR. COPELAND: Well, we've already addressed that issue, Your Honor. That was testimony of the witnesses several weeks ago. We --

MR. DOHERTY: Were those Staff or Applicant witnesses --

MR. COPELAND: -- were asked the question of whether we met the current requirements for hydrogen control. We addressed Section 50.44(e), I believe, and the

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current requirements.

MR. DOHERTY: If "we" is the Applicant, I don't think the objection is taken.

MR. COPELAND: Well, I think Mr. Fields was also a witness on that, as I recall.

JUDGE WOLFE: Well, regardless of that, the objection, I take it, is the question is outside the scope of this contention.

Now, address that, Mr. Doherty.

MR. DOHERTY: He has mentioned systems that mitigate core degradation, which would mean hydrogen released to the containment building, which is directly relevant to the occurrence of hydrogen explosions.

Obviously, if the systems for mitigating core degradation stopped hydrogen removal, as a for instance, then we are gathering information on the occurrence of hydrogen explosion.

MR. DEWEY: Your Honor, we could go on and on from core degradation to the next point to the next point. I think he has gone too far with this core degradation on hydrogen monitoring.

(Bench conference.)

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JUDGE LINENBERGER: I knew it was going to be one of these weeks.

Mr. Doherty, before we can rule here, we have to really understand what your question is. Is your question directed toward determining whether the monitoring system has the capability to detect whether an explosion has occurred?

MR. DOHERTY: Well, it's more directed toward finding out if there's even going to be any hydrogen at all any more, if that's the Staff's position, that they can have no -- I have to apologize.

I didn't prepare cross this way. This morning I suddenly found out I could ask some questions I thought I was precluded from asking.

So I don't have -- this is my opportunity to cross. I don't get a chance to go home and figure out a few things. So it's not coming out the way I'd like. That's why it's coming out garbled and that's why it's tough all around.

JUDGE WOLFE: Well, exactly what is your question now, Mr. Doherty?

MR. DOHERTY: Where he stated that there's no requirement that the system survive hydrogen explosions, I'm trying to find out if indeed what the Commission has determined is that there are these systems to mitigate

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degraded core accidents will be of such that there won't be any hydrogen so there won't be any reason to worry.

JUDGE WOLFE: I don't understand that question, what your question is as addressed to the witness now.

MR. DOHERTY: I don't have that solidly in my mind any more as to exactly the words I used.

What I'm trying to find out is has the Commission decided there's no reason to have a system for monitoring hydrogen which will survive a hydrogen explosion because there won't be any hydrogen in a degraded core accident to cause an explosion.

That's what it's directed at.

(Bench conference.)

MR. COPELAND: Your Honor, may I object. I thought the witness had answered that question already by saying they felt like the existence of the recombiners and the CO_2 system were sufficient to control whatever hydrogen would be generated to prevent an explosion, which begs the question of whether there ever will be any hydrogen. That's for anybody to guess.

The fact is, the requirement is there to have those things on the assumption there will be some hydrogen produced under certain events. So I think he has answered the question as best anybody could answer the question.

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JUDGE WOLFE: So your objection to this question, Mr. Copeland?

MR. COPELAND: Is asked and answered, if that is now his question.

JUDGE LINENBERGER: Let me just review something here, Mr. Doherty, and see if this will assist you in perhaps rephrasing your question.

The witness has testified that the philosophy of the design approach here, rather than being one of being able to determine whether there has been a hydrogen explosion, is an approach that will attempt to assure that there is never enough hydrogen left hanging around in the wrong places to build up to a concentration that will permit an explosion.

In other words, the witness has said that they are not critiqueing the Allens Creek design on its ability to know that an explosion has occurred. They are critiqueing the Allens Creek design on the basis of its ability to prevent hydrogen concentration to build up anywhere near to where an explosion can occur.

Now, I've put words in the witness' mouth. Let me ascertain before finishing here if that is the thrust of your statement, Mr. Fields?

THE WITNESS: Yes, it is.

JUDGE LINENBERGER: All right, sir.

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Now, given that reiteration of that position of the Staff, Mr. Doherty, it seems to the Board that however the contention is worded (and I realize it's not your contention), however the contention is worded, the permissible line of questioning for you is really how well is the Applicant approaching the job of assuring that hydrogen can never get to the explosion point, rather than given an explosion, what equipment will live through it.

(Bench conference.)

JUDGE LINENBERGER: Well, Mr. Doherty, the Board having recapitulated what it thinks the current situation is here, why don't you now restate your question however you want to and let's then see if it flies or gets shot down by objection.

BY MR. DOHERTY:

Q. Would you say at the moment that the Commission acknowledges that in event of a loss of coolant accident there will be some hydrogen generated?

A. There is that possibility, yes.

Q. All right. Is it part of the design based loss of coolant accident that some hydrogen will be generated, to your knowledge?

A. Yes.

Q. Okay. Would you say that the Commission is placing reliance on systems to prevent hydrogen burn or

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hydrogen explosions, rather than attempting to follow them?

A. Yes.

Q. I believe earlier in the hearings you testified that the systems to control level of hydrogens in -- I believe earlier you testified on systems to keep the level of hydrogen in the containment below the burn level.

I believe you testified that there were such systems, is that right, in these hearings?

A. Yes.

Q. Has the NRC held any evaluations of that system so far?

A. Yes.

Q. Has the Advisory Committee on Reactor Safeguards expressed any interest in this, to your knowledge?

A. Are you referring to the system that is used to mitigate consequences of hydrogen generation due to the current regulations or the proposed regulations for degraded core rulemaking?

Q. Well, why don't you answer it both ways.

A. Okay. As far as the current regulations in 10 CFR Part 50, the amounts and types of control of the hydrogen for a design basis accident was looked at many years ago.

I imagine the ACRS looked at it then and found

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it acceptable.

Currently, there is a proposed rule before the Commission on degraded core scenarios, the near-term CP rule.

Specifically, the post-accident inerting system that Allens Creek is proposing has been looked at by the ACRS.

Q. Do they consider it adequate?

A. Do they consider what?

Q. Do they consider it adequate? You say they have looked at it.

A. They have not yet made a finding. They have asked the Staff to provide them with more information.

Q. Do you know of any plants that are Mark III containments that are in the operating license stage that use a system such as the Applicant's?

A. I don't know of any.

Q. Do you know of any alternative systems to the Applicant's?

MR. COPELAND: Your Honor, this whole line of questions has been pursued before with this witness in another hearing.

That's one of the reasons we are taking so long is we just keep replowing old ground here. That's all we've been doing all morning long with this witness.

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He's talking about things that have all been discussed. We have talked about the compliance with 50.44. We have talked about compliance with degraded core. We have talked about the hydrogen recombiners. We have talked about the CO_2 system. We have talked about the status of review by the ACRS.

There's not one single thing that's come out here that hasn't already been discussed.

(Bench conference.)

MR. DOHERTY: So what's your objection, asked and answered?

MR. COPELAND: Waste of time.

(Bench conference.)

JUDGE WOLFE: Yes, Mr. Doherty.

MR. DOHERTY: This is an asked and answered objection, I take it? There's an objection outstanding at the moment, or merely a complaint?

MR. COPELAND: The bottom line is, Your Honor, obviously he's gone way outside the scope of the contention.

The contention is a very narrow and simple contention, and he has used the contention to back through and go back through the entire discussion that we have already had, and I just think it is time to terminate the cross-examination.

It's not going anywhere and it's not doing

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anything but replowing old ground.

MR. DOHERTY: Well, I've asked some questions with regard to monitoring of hydrogen explosions, which certainly, I don't believe that is old ground.

We may have a case where two contentions should have been solidified or combined in some way instead of given independence; but I still think -- to me it's a source of concern that we have the possibility that hydrogen monitoring will not survive an explosion and, therefore, a second explosion or a second buildup of hydrogen, which is expressed in the contention, could occur and there won't be any monitoring for that at all if indeed there's no protection against hydrogen explosion in the first.

MR. DEWEY: Your Honor, Mr. Doherty has just rewritten a new contention. It is entirely new. It should have been done a long time ago if that was his concern.

MR. DOHERTY: What was that?

MR. DEWEY: A breaking of the hydrogen monitoring system whereby you --

MR. DOHERTY: That's what he said.

JUDGE WOLFE: You are arguing now, Mr. Doherty, with the Board's previous ruling.

Now there's an objection to this last question that you are going outside the scope of the contention. What is your response?

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MR. DOHERTY: Well, my response is that the contention cannot be logically dealt with if we say that since the monitoring system cannot survive an explosion, we cannot talk about monitoring anymore, which is the way I find this argument coming out.

JUDGE LINENBERGER: The problem is that we're getting right back to something that the Board reviewed with you a few minutes ago; namely, and I'll say it in different words this time, your contention, the allegations of the contention are inconsistent with the design and review approach that Applicant and NRC are taking.

Applicant has chosen and NRC is evaluating that choice to approach hydrogen on the basis that they will do a good enough job detecting and suppressing the concentration of hydrogen, such that an explosive situation will not arise; and, therefore, have not seen a need to provide a monitoring system that has to survive an explosion, because the design approach which is proposed and which the Staff is reviewing is that the system will work well enough that there will never be an approach to an explosive configuration.

So for you to come back and keep questioning about something that is not being proposed is fruitless, Mr. Doherty.

As I indicated earlier, it's fair ground for

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you to try to assure yourself that the monitoring and recombiner systems will indeed prevent you ever getting to the explosive point; but to keep hammering away that monitors must survive an explosion when that is not a design objective is fruitless and we can't permit it.

MR. DOHERTY: All right, Your Honor.

JUDGE WOLFE: I'll sustain the objection.

MR. DOHERTY: Yes. All right.

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BY MR. DOHERTY:

Q Well, you might want to turn to page 2, Mr. Fields. We'll work by page numbers a bit here.

Do you think it makes any difference that Westinghouse has been developing this thing, this recombiner that you talk about on page 2? They are in the business. They make something different. Do you think that makes a difference?

A In respect to the safety?

Q In respect to what the recombiner will be used for.

A No.

Q I have a question here. There is a letter from Mr. Vassallo that you mention halfway down in your testimony, to Mr. Eicheldinger, who has my sympathies.

Must the hydrogen recombiner be qualified under IEEE standards?

A Which one?

Q 323.

A Which year?

Q '74.

A Yes.

Q Now, are the pressure and spray environments the same for pressurized water and boiling water reactors post-accident?

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A. No.

Q. Now, it's my understanding that the Staff did a detailed evaluation of Westinghouse's test program to qualify its thermal recombiner with PWR environments.

That would seem to indicate that if used in a BWR situation, they might question. Do you have any --

A. Yes, I think I can explain that. The environmental effects in a PWR are much more severe than it would be for a BWR outside the drywell.

For instance, the recombiner was qualified to a temperature of 300 degrees; in a Mark III containment, the absolute maximum you'd ever see was 180. Likewise, with pressure, they qualified it to 60 to 70 psig; and in a Mark III containment you won't see any more than 11 or 12 psig.

We looked at all the parameters that could possibly change and concluded that the environmental conditions inside the containment of a Mark III are quite a bit more mild than they are inside the PWR.

Q. There is a figure, 180 degrees, is the maximum temperature. Where is that for a BWR containment?

A. Where is the figure?

Q. Yes, sir.

A. It's in Chapter 6.2 of the PSAR for Allens Creek. I don't know the exact figure number; I'm sure I

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can find it.

Q Well, no. So you've satisfied yourself that that's the --

A. The peak temperature.

Q -- peak temperature for BWR's?

A. Yes. Inside the containment.

Q Right, but not inside the drywell.

A. That's a different story.

Q Yes, okay.

Excuse my slowness here.

You enclose some pages of the Safety Evaluation Report. The device is said to heat containment gas mixture to 1150 degrees. Is that what it does? It heats up whatever it draws through, right?

A. Right. That's the minimum temperature it heats up. That's minimum temperature needed for complete recombination of oxygen and hydrogen.

Q Now, when you say "recombination," that presumably makes water?

A. Yes.

Q And does that -- is that figure of 1150, is that an optimum number for water production, so to speak, or is it a highest number you dare go to before you might run the risk of igniting something there; do you know?

A. There's no problem with igniting. It's the

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temperature, the minimum temperature by which you have complete recombination.

You can have higher temperatures and you won't have any detrimental effects.

Q. Would that be, in your mind, much, much higher or much greater, 300 degrees or something of that order?

A. How high would it have to be before you had problems?

Q. Yes.

A. I don't know of any problems you would have with any temperature.

I imagine if you raised it to an absurdly high level you might do strange things.

Q. Yes. They talked about on page 3 in answer to the Board's Question 2, poisoned recombiner surfaces. That's at the foot of 2. By "they," I believe I mean the Board here. Yes, that's correct.

Then you listed a series at the top of page 3 of things that the recombiner was exposed to.

The only thing I can think of that might be poisoned was containment spray. Is the containment spray -- Is there a containment spray in this Allens Creek system?

A. Yes.

Q. Is the containment spray pure water or tap water or whatever, something of that?

A. I believe it is. The tests that were run on the recombiner included an additive to reduce the iodine content of the air, some sodium chemical.

That was what was exposed to the recombiner.

Q. Do you have any idea how long this system will have to operate to do a job which it's maximumly expected to have to do?

A. I don't think that number has been firmed up exactly. They have done tests on the heater banks, which is the critical element in this recombiner, to show that it will operate satisfactorily for a year. I don't suspect that it will be needed that long.

Q. In the second paragraph on page 3 you talked about heat sources as, I gather, providing a buoyancy effect, perhaps lifting hydrogen up into the ceiling and getting it an even distribution?

A. Basically. In this case the heat sources would heat up the air and, of course, the air would rise; not only just the hydrogen, the air.

Q. Yes, everything.

A. Yeah.

Q. These heat sources then would tend to promote mixing; is that the conclusion here?

A. Yes.

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Q So that -- well, would the suppression pool typically be a heat source in this kind of situation?

A. Yes.

Q Okay. Now, if we had a situation where the recombiner had to operate for several days, though, wouldn't the suppression pool tend to cool off and contribute less and less so that this beneficial turbulence would tend to decrease with time?

A. Yes.

Q I think you mentioned that sprays would have a beneficial effect, containment sprays would reduce stratification, also.

Is that presuming the containment sprays are colder?

A. It's basically because the spray will cause the air to move, just the fact that you have spray droplets going through the air will cause air currents, not necessarily the temperature difference.

Q I see. Was the change that you made in the fourth line from the bottom on page 3 of the temperature, is that a typographical correction or is there another story on that?

A. I believe when I first wrote that down I took the temperature of the air, instead of the temperature of the recombiner surface.

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Now, the 1600 degrees is the maximum temperature of the recombiner surface at the rated power.

It really doesn't make any difference in the final analysis, whether it's 1200 or 1600 degrees does not affect the conclusion.

Q. Moving ahead to -- the page numbers are hard to read. I guess it's page 5.

With regard to the model for locating hydrogen there, is there a plan to go beyond modeling, to your knowledge, or is that going to be what you are going to rely on?

A. Modeling is sufficient, considering the number of sample points. If the modeling was done to justify maybe one location, then perhaps the Staff would say, "Why don't you add a few more locations in order to cover any possible uncertainties."

Q. I think you provided a figure down here, yes, at the very end of your numbered pages.

There are two locations there that I wanted to inquire about.

MR. DEWEY: What page are you referring to, Mr. Doherty?

MR. DOHERTY: It's the first page following page 7, which is the --

MR. DEWEY: Figure 1?

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MR. DOHERTY: Yes, Figure 1 of the NRC Staff testimony. It's a figure of the containment.

BY MR. DOHERTY:

Q. Now, there is an area marked "5" that has a large line drawn, right, to form the base of that area.

It appears to me there are two cavities at either end next to the shell of that one line drawn across; do you see that?

A. I'm not really sure. Would you point it out to me.

Q. Sure.

A. This area right here and this area right here.

Q. Yes.

A. Okay.

Q. Now, is that line that I mentioned that seems to be the base of that Area 5, is that a solid barrier, to your knowledge?

A. No, it is not.

Q. Does it limit gas in any way at all from rising, to your knowledge?

A. To my knowledge, it does not.

Q. Okay. I notice at the foot of 5 there is a discussion of the monitoring system. You state that an alarm is automatically actuated at three percent.

A. Yes.

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Q Can you give an idea -- well, let's ask this. Is four percent the percentage at which burning may occur?

A It's the theoretically lower limit at which you can have upward burning.

Q Has the Staff worked out any estimates of how rapidly the concentration might rise from three percent to four percent?

A Yes, we have. Basically, the rise at this point is due solely, or almost totally to radiolysis, which is a fairly slow process and well within the capability of the hydrogen recombiners.

Q Okay. At the top of 5 you talk about Regulatory Guide 1.7 assumptions --

A Page 6, you mean?

Q Yes, right. About Regulatory Guide 1.7 assumptions used in the generation rates of hydrogen.

Has the Commission's view remained the same since Three-Mile Island with regard to rate of hydrogen generation?

A There is some complexity in the situation in that as far as design basis accidents, Reg. Guide 1.7 is still valid.

Now there is some further work, or additional conservatisms, that are presently before the Commissioners which would increase that generation rate as far as the

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zirconium steam reaction.

Q. I take it the result of that would be that possibly the recombiner might be needed sooner after a LOCA; is that --

A. If you assume the worst cases that we are currently contemplating for a degraded core scenario, the recombiners are not adequate to control the hydrogen levels, which is why the Applicant had proposed the post-accident inerting system.

Q. Well, assuming the inerting system, won't -- in the event of a hydrogen generating accident, won't hydrogen have to be recombined to be removed?

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

Q. Okay.

A. Assuming that the inerting system works?

Q. Oh, yes.

A. If the inerting system works, then there is no need to recombine hydrogen.

Q. Well, the inerting system does not destroy the hydrogen; isn't that correct?

A. That is correct.

Q. Will it be just left? I mean, won't something have to be done with that hydrogen eventually?

A. Yes. The long-term solution and alternatives that can be used to take care of the degraded core situation

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is still being looked at by both the Applicant, the ACRS and the Staff.

We have come up with no firm conclusions on what to do about it.

Q. I think I might have one or two more questions.

Does the recombiner operate some kind of a motor or is it more like a heater?

A. There are no moving parts in the recombiner. It just heats up the air.

Q. So would it be fair to say that starting is not a difficulty; would you feel comfortable with that?

A. We are very comfortable, yes.

MR. DOHERTY: All right. Thank you very much, Mr. Fields.

JUDGE WOLFE: Redirect, Mr. Dewey?

MR. DEWEY: Yes, sir, just one question.

REDIRECT EXAMINATION

BY MR. DEWEY:

Q. Mr. Fields, Mr. Doherty mentioned the fact that the suppression pool after several days, for example, might cool down and, therefore, you would not have that as a basis for heat to provide some of the convective forces that would be utilized to push the air up to the recombiners.

Is it your opinion that this suppression pool as a heat source would be necessary to make the recombiner effective?

A. No, it's not needed. The recombiner, because of the temperature difference, has its own motive power, so to speak, to cause air to enter the recombiner.

Q. So the recombiner would be sufficient in and of itself without the suppression pool?

A. That is correct.

MR. DEWEY: Thank you.

JUDGE WOLFE: Board questions.

BOARD EXAMINATION

BY JUDGE CHEATUM:

Q. I would like to just review a little of the chemistry of the atmosphere that is monitored by the monitoring system.

I understand that the monitor will record the percent of hydrogen in the containment atmosphere and that an alarm will sound off when it reaches a level of, say, three percent.

Now, beyond that level, at some point if there are the right combinations of other elements of the atmosphere to sustain an explosion, the hydrogen will explode if it's ignited in some way, right?

A. If it reaches a certain concentration level. For explosive concentrations you have to be 18 percent.

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Q. Eighteen percent?
A. That's for detonation.
Q. Detonation?
A. Yes.
Q. What's the difference between explosion and detonation?
A. Really none.
Q. Or a burn and detonation?
A. A burn is where the burn speed is less than sonic speeds. A burn speed propagates at less than the speed of sound, and what happens is after about 4.1 percent you have the possibility of some burning; and as you increase the hydrogen concentration, burn speed will increase until you reach 18 percent, which is defined as detonation, because the burn speed goes supersonic.
Q. Is burning dependent to an extent on the amount of oxygen in the atmosphere?
A. Yes.
Q. Is it totally dependent?
A. It's dependent on the composition of the atmosphere. For instance, the CO_2 will, even though there's hydrogen and oxygen present, will prevent any hydrogen burn.

So while you do need hydrogen and oxygen, it's possible to prevent hydrogen burn through other factors.

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Q. By adding carbon dioxide?

A. Yes. That's one method.

Q. This would not be a purging method, but a --

A. Inerting.

Q. A suppression method?

A. Yes.

Q. To prevent. I remember we went into that in the last testimony.

What initiates a burn, what kind of situation? Is an electrical spark or something like that required that really sets off burn or the explosion?

A. That really depends on the concentration. The lower the concentration, the stronger the ignition source you have to have.

Perhaps it's even possible that very high concentrations do have auto-ignition without a spark present.

Q. So-called spontaneous combustion?

A. Yes.

Q. Is an electrical spark or something like that assumed to generally be the most probable ignition source?

A. Yes, although for the purposes of design, we assume that once the hydrogen reaches a certain level it's going to ignite.

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We don't go around looking for ignition sources.

Q I notice in the attachments to your testimony the Westinghouse system for recombining hydrogen and reducing its amount in the containment, that it all refers to PWR containments.

I am sure there's probably no difference really in relation to how this operates whether it's a PWR or BWR containment. Is that true?

A. That's true. The only differences are the fact that the environment that's contained in the BWR is milder and, therefore, has less detrimental effects on the recombiner than it is in the PWR.

The temperature and pressures are lower.

Q I see.

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BY JUDGE CHEATUM:

Q. I see. So that is what you meant by "milder"?

A. Yes, sir.

Q. It may be in the testimony, but perhaps I may have missed it. Is the combiner automatically actuated when hydrogen reaches a certain concentration, or does it have to be manually automated?

A. It has to be manually actuated.

Q. -- from the control room?

A. Yes. And the reason that the Staff accepts this procedure is because the generation of hydrogen is very small. And the concentration levels go up very, very gradually, allowing plenty of time for operator action.

Q. What is the maximum amount of time that might be required following a DBA, a LOCA, core damage, before you might have to start actuating the recombiner?

I'll admit that probably depends on what your monitors have told you; is that right?

A. True. We are talking about days. The number I have in my testimony states that the recombiner would not be needed until approximately eight days after the accident.

The latest information provided by Allens Creek suggests three to four days. And there will be some operator action required before that, to initiate the

5-2

drywell mixing system, which is not the same thing as the recombiner system.

That will be required eight to nine hours after the accident.

JUDGE CHEATUM: Thank you, Mr. Fields; I have no further questions.

BOARD EXAMINATION

BY JUDGE LINENBERGER:

Q. Sir, when you said that the -- in answer to Dr. Cheatum's question -- that a recombiner had to be manually actuated, I assume you meant by that that somebody had to turn on some equipment that, in essence, resulted in the heating up of the surface plates in the recombiner to bring them up to temperature; is that what has to be done to manually actuate them?

A. Yes, and that can be done from the control room.

Q. Once those surfaces in the recombiner have reached their operating temperature, from that point on does an operator have to do anything to cause them to start to recombine --

A. No.

Q. -- hydrogen and oxygen?

A. The process is automatic.

Q. You indicated, first, no moving parts and,

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secondly, 100 standard cubic feet per minute volumetric flow rate through the recombiner.

Is the driving force for this 100 cubic feet per minute flow rate convective in nature arising from the heat on these surfaces in the recombiner?

A. Yes, it is.

Q. Have tests indicated whether these recombiner surface temperatures increase if hydrogen is present, compared with their operation if there's no hydrogen present -- compared with their temperature if there's no hydrogen present?

What I'm asking is: Does the act of recombining hydrogen in the vicinity of these heated surfaces, since it is an exothermic reaction, cause the surfaces to increase in temperature?

A. I saw no mention of that in the Westinghouse reports. If it was significant, I'm sure it would have been noticed.

However, that particular possibility was not explicitly addressed.

Q. Okay, let's back off now and look at the overall operational philosophy of dealing with hydrogen after a loss-of-coolant accident.

First off, let's start with prior to a loss-of-coolant accident, with normal operation. Under the

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condition of normal operation of the facility, is there any reason for making use of the hydrogen monitoring system, under normal operating conditions?

A. No.

Q. There is radiolysis of water in the core, and when you say no in answer to that question, I would infer that the hydrogen deriving from the radiolysis of water occurring as a result of normal operation is dealt with in some other way not requiring the use of the hydrogen recombiner.

A. That is correct. There is a gas tripper that's off the main condenser, which recombines the hydrogen, but it's not the same -- it's not a post-accident system.

Q. Right.

Okay. So we need neither the monitors nor the recombiners during normal operation. Now, let's go to a loss-of-coolant accident of such a nature that the emergency core cooling system is called on to operate.

Let's assume that it responds to that demand for duty, as it was designed to do. You've indicated that approximately 30 minutes after this initiation of operation of the emergency core cooling system, that the operators will then start monitoring for hydrogen build-up; is that correct?

A. That is correct.

Q. So far as the operating procedures that the Commission will require, are the recombiners turned on at the time the hydrogen monitoring system is turned on; or is it permissible for the operators to wait until some hydrogen buildup is detected before the recombiners are turned on?

A. The current requirement is to allow the operator to wait until he feels that the levels are at or approaching three percent.

Q. Okay. Let's continue this sequence of events now by saying next that the monitor indicates, some hours into the -- minutes or hours into the event -- that the monitors show -- the hydrogen analyzer monitors, or at least some of them or one of them, show a hydrogen concentration approaching the three percent.

And so the recombiner is turned on. Now, you have the potential for two branch points in the sequence I'm developing: One, that as time proceeds, the emergency core cooling system continues to keep the core covered.

The recombiners have been turned on. The reactor is in a shutdown configuration, and things are just waiting, presumably, for the core to ultimately cool down, however many days, weeks or whatever that may take.

Now then, what is the analyzed or anticipated behavior of hydrogen concentration from the point where the recombiners are turned on because the concentration has reached approximately three percent, the ECCS continues to keep the core covered, things just sit there then for several days or weeks -- what is the then time-dependent shape of the hydrogen concentration curve?

Can you address that qualitatively or quantitatively?

A. At least qualitatively, using our analytical models and the fairly conservative bounds on how much radiolysis we can think that would happen and also how much reaction of any zinc-based paint. We have determined that the amount of hydrogen generation is less than the recombination rate of one recombiner.

Q. One recombiner.

A. Yes.

Q. Okay. So what you're saying -- I can conclude from that -- you tell me if I'm correct in so concluding -- that the capability of one recombiner to eliminate hydrogen exceeds the source term for generating hydrogen; therefore, if the concentration got up to three percent when the recombiners were turned on, it would gradually decrease with time then, so long as the

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core remains covered?

A. That is correct.

Q. Okay. Now, let's go back to the possibility of a branch point. We've had the accident, the ECCS has come on.

But for whatever reason -- and it's not directly relevant to my question -- the ECCS does not maintain coverage of the core with water. And again, for whatever reason, the water level begins -- in the reactor vessel begins to drop.

The upper portion of the core begins to uncover; the water level continues to drop. Now, I really should put some sort of rate of lowering of the water level on here, but I'm not quite sure how to do it -- yes, let me do it.

Let's take a number like one foot per hour that the water level is gradually dropping below the top of the core.

Now, what -- with respect to people who are concerned about hydrogen in that facility, under such a circumstance what would be the typical sequence of events that they would look for; what kind of information might they seek; what kind of remedial action might they take, starting from this point where it's known that the ECCS is deficient in delivering water to the core,

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such that the level is lowering about a foot an hour?

A. Okay. This will put you in a degraded core situation. And the operator has instrumentation inside the reactor vessel which will allow him to determine the water level.

Once he sees the water level not covering the core, his training and emergency procedures will tell him that the possibility of generating a significant amount of hydrogen is possible, and he will actuate the post-accident inerting system.

Q. Well, that sounds good. But, presumably, somebody is keeping an eye on what the hydrogen analyzer is saying during this stage of the scenario I've proposed here.

And to my way of thinking, inerting is possibly a sort of last-ditch measure. So I would be inclined to think that as long as the hydrogen analyzer says -- not getting much above or any above three percent concentration, let's hold off on inerting because maybe we can overcome the problem with the water dropping and so forth.

Now, what I'm getting at here is: Are the operators going to be instructed to inert when they know the water level is dropping below the top of the core, irrespective of what the hydrogen analyzers read?

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Or are they going to be guided by what they read from the hydrogen analyzers?

A The initiation parameters that will be used to actuate the post-accident inerting system has not been completely defined.

However, looking at the hydrogen analyzers inside containment may not provide a quick enough turn-around, because you can get into some very fast hydrogen generation rates when you lower the water.

Therefore, because the inerting system needs approximately 45 minutes for complete inerting of the containment, it hasn't been determined yet whether or not we can rely solely -- or rely in part on the hydrogen monitors.

That is something that is still being examined. It will be looked into, because we agree that we don't want the post-accident engineering system being used unless it's necessary.

However, we can't say at this point.

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Q Okay. You anticipated my question and very good.

Now, let's go to page 5 of your testimony. Near the bottom of the one full paragraph on that page, you talk about range, accuracy and sensitivity of the hydrogen analyzer.

A range of zero to five percent, and does the five percent represent full scale on the analyzer, so far as you know?

A This represents full scale. I should point out that a requirement that was contained in NUREG-0737, which has only been applied to OL's and operating plants, requires that the range be zero to ten percent for PWR's and for BWR's Reg Guide 1.97 is requiring zero to thirty percent.

Q To 30 percent?

A Correct. So the bottom line is they will have to put in at the OL stage instrumentation that will range between zero and thirty percent hydrogen concentration.

Q Okay.

A And they have already committed to do so in their response to NUREG-0718.

Q All right. Now, the quoted accuracy of plus or minus two percent of full scale, will that still obtain when full scale is thirty percent?

-2

A. It will be in that range. We may not require quite as tight a requirement for the larger monitor as we do for the range from zero to five; but it will be adequate to detect trends, to detect generation rates and to provide the operator with the information he needs.

Q. Okay. Your Figure 1 is a nice clean pictorial of a portion of the facility.

If I look at Figure 1.2.8 of the PSAR, the inside of that facility is a mess. Excuse me, I'm not referring to housekeeping.

I'm referring to there's just all sorts of stuff in there. There's partitions, there's subfloors, there's maintenance people platforms, there's cubicles; there's just all kinds of things in there.

Now, when I look at your Figure 1 and your words about how well things are going to be mixed because of convection, I find it relatively convincing.

When I look at Figure 1.2.8 of the PSAR and all the stuff that is stuff in there, I get really concerned about how well the mixing is going to be.

Now, your discussion talks about a model used in an analysis to define locations for the hydrogen analyzer to take samples.

Did that model in any way take into account all these structures and stuff that's inside that building,

-3

such as is illustrated in Figure 1.2.8, or did it look at something like your Figure 1, which is a real straightforward geometry?

A. I would suspect it looks more like the figure you have. However, I don't know how detailed the model was.

Q. I see. All right.

There are two types of concerns here. Is the monitor looking at the right places to find higher than desirable concentrations of hydrogen -- or to look f them; and secondly, are the recombiners themselves pl ed in positions where if there's not complete uniform hydrogen, the higher concentrations might occur.

Let's go to that second question. With regard to your Figure 1, approximately where would the two recombiner stations occur?

A. I believe it's in the region in the containment that is at an elevation higher than the reactor vessel head. So it's towards the upper portion of the containment volume, one on either end, I believe.

Q. Say again that last sentence.

A. There's a recombiner on either end, you know, approximately 180 degrees apart.

Q. Both at the same elevation, but about 180 degrees?

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A. I believe so.

Q. On your drawing there appears the word "containment" and a line with an arrow that points down into a volume there, which volume I have the impression was filled with water.

Now, where that word "containment" and the arrow therefrom terminates, is that water in that?

A. This drawing should only be used for a limited purpose. There is water in there that will be dumped into the suppression pool following an accident.

However, that line you see going across that separates the RWCU pump area from the area that's called the containment really shouldn't be there.

Q. Are you saying the pump area near the upper dome, upper end of the steel containment shell really communicates with the annulus inside the containment building above the suppression pool?

A. That is correct.

Q. Okay. So there is really no line across there, no physical thing across there.

A. No, sir.

Q. But there is water above his pressure vessel dome or something?

A. Yes, upper pool.

Q. Well, now, let's go back and talk once more about where these recombiners are located, now that we've seen where the water is and where the lines aren't.

Tell me again about where the recombiners are located?

MR. COPELAND: I understand, Your Honor, they may be on that figure from the PSAR that you were looking at.

JUDGE LINENBERGER: Well, okay. That's going to require me to find my magnifying glass.

Can somebody provide Mr. Fields with a copy of Figure 1.2.8 to look at.

Maybe Mr. Fields will need a magnifier, too, I don't know.

MR. COPELAND: Is it all right, Your Honor, if Mr. Malec points it out to Mr. Fields?

JUDGE LINENBERGER: Fine with me.

THE WITNESS: Your Honor, it is not shown on the diagram you have.

What was shown to me was a planed section at Elevation 232 feet that has the recombiner on it.

If you look at your section you will see that 232 feet --

JUDGE LINENBERGER: Excuse me. Give me the elevation again.

THE WITNESS: Two hundred thirty-two.

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JUDGE LINENBERGER: Okay. I find that elevation.

Does that elevation represent the elevation of some sort of a platform upon which the recombining units sit?

THE WITNESS: It sits on some sort of a platform, yes.

JUDGE WOLFE: We will recess until 2:15.

(Whereupon, at 12:49 p.m., the hearing was recessed, to reconvene at 2:15 p.m., the same day.)

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AFTERNOON SESSION

2:15 p.m.

JUDGE WOLFE: All right.

Judge Linenberger.

BY JUDGE LINENBERGER:

Q Mr. Fields, I should like to continue just a bit longer on the consideration involving the placement of hydrogen analyzer intake locations, hydrogen recombiner locations and the consideration of the possibility of an accumulation of hydrogen somewhere that might reach a flammable concentration without adequate advance warning of this.

Let me clear up one question right at the outset. You indicated that operation of containment spray would assist in the mixing and homogenizing of hydrogen concentration throughout the containment, as I recall.

A I think what I meant to say is the containment sprays will cause air currents, whether it tries to homogenize the air and hydrogen is something I did not mean to say. I'm not sure if that occurs. It probably does, because of the air movement.

Q Well, I don't understand why you would mention it unless that were one of the benefits of it.

A The benefits of the sprays, in that it does cause the air to mix --

Q Right.

A Now, from what we can see, the hydrogen is going to be very well mixed in the air anyway and does not need to rely on the sprays to mix the hydrogen with the air. To cause the air to circulate throughout the containment is the primary benefit of the containment sprays.

Q All right. The primary benefit of the containment spray, you say, is to cause the air to mix throughout the containment.

But I thought you were taking credit for that phenomenon to assist in avoiding the occurrence of pockets of higher than average concentrations of hydrogen within the containment then.

A Yes.

Q Okay. Now, the question I was leading to here -- let's go back to the sequence of considerations we were discussing just before lunch. We've had an event that has caused the emergency core cooling system to be activated. Under the circumstance where -- for the duration of time we're talking about for the purposes of this discussion, the core remains covered.

And the hydrogen analyzer system is activated 30 minutes after the ECCS has been activated. Now then,

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what determines in that set of circumstances whether containment spray will be turned on?

A. The primary reason for having containment sprays is not to prevent hydrogen pocketing, although it does aid in that respect --

Q. Right.

A. Containment sprays are there to reduce the pressure that could possibly build up inside the containment if you have some small steam leakage from the drywell to the containment.

Now, the initiation parameter for containment sprays is 10 minutes, plus a certain pressure set point, which I can't reall offhand. The 10-minute --

Q. Excuse me. Ten minutes plus a certain --

A. Plus a pressure set point.

See, a pressure set point must be reached before the containment sprays will be actuated and -- automatically. Of course, the operator does know that he can use the sprays to mix the hydrogen -- mix up the air.

Q. Well, for the purposes of this discussion, we'll consider that effect of the containment spray sort of a fringe benefit. The primary purpose, as you've said, is to keep containment pressure from exceeding a certain level.

Now, referencing your Figure 1, into what part

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of the various sub-volumes indicated there does containment spray enter into, the spray itself, not --

A. The spray comes out of the spray headers at the top of the containment. And the RWCU --

Q. Excuse me, I have to stop you right there, because from this drawing, I'm not sure that I know what is meant by the "top of the containment."

A. Okay. The area that is labelled RWCU, this is pump area. There are ring headers -- ring spray headers located in that region that will spray down, and it will spray into the atmosphere located underneath and to the sides of the -- this annulus area down into the pool.

Q. All right. But, presumably, as I look at that drawing, there is no direct access for that spray water into the drywell area; is that correct?

A. That is correct.

Q. Okay. Now, if I understood your prior testimony, or oral comments correctly, the recombiners themselves are supported at an elevation of approximately 232 feet, which is above where that -- in your Figure 1, about where that horizontal line occurs that you said doesn't exist.

A. When I say the horizontal line doesn't exist, I'm saying that it doesn't represent a division

7-5

between one volume and the other.

Q Right. Okay --

A There are platforms there.

Q There are things there --

A Yes.

Q But at any rate --

A If I could, I have a drawing here from the -- a report referenced in the PSAR that shows the locations of the recombiners from a section ... a plainer view.

Earlier I stated that the recombiners were approximately 180 degrees on either side. And this diagram will show that they are approximately 135 degrees on either side of the containment.

Q Is that a figure from the PSAR?

A It's a figure from a report that was referenced by the PSAR.

MR. COPELAND: It's the report that Mr. Malec referred to this morning, the Reactor Systems Containment Report.

THE WITNESS: It's called "The Containment Structures Load Report."

BY JUDGE LINENBERGER:

Q Okay. For the moment, let's try to do without that figure, since it's only implicitly in evidence. What I'm leading up to is that with both recombiners

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sitting at or just above that elevation you indicated at which they were located, and looking at this Figure 1.2.8 from the PSAR, I would conclude that the recombiners will be most effective in operating on the air/hydrogen/vapor mixture -- whatever it is -- above the 232 foot elevation, just because -- I mean I reach that conclusion only because below that elevation in this PSAR figure that I've been speaking about, there seems to be an awful lot of stuff installed, and above that elevation much less stuff installed.

So it seems to me that circulation patterns would be less tortuous above the 232-foot elevation than below it. Is that a reasonable observation for me to make?

A. The amount of open area below the thermal recombiners is approximately 25 percent of the total area available to be open.

So, yes, it is not as open as the area above the thermal recombiners. But you still have a couple of thousand square feet of open area for the containment atmosphere to move freely.

Q. On page 5 of your testimony you indicate five stations from which containment atmosphere will be sampled for analysis. The first one says the top of the containment, which is a volume that is -- that readily communicates with the two recombiners.

The second station says near the top of the pressure vessel. Now, if I look at your Figure 1 or the PSAR figure we've been referring to, those words, "near the top of the pressure vessel," to me cause me to consider the region between the top of the pressure vessel and the -- what is labelled in another figure similar to your Figure 1; namely, Figure 6.2-1 from the PSAR, removable drywell head.

In other words, above the dome of the reactor pressure vessel, there is a space which is confined by a curved member of some sort labelled "Removable Drywell Head."

Now, is it in that area that this number two station is located, when it says "near the top of the pressure vessel"?

Is my question intelligible?

A. I understand the question. Unfortunately, I can't give you any more specifics on where the location is. It's close to that area. Certainly, it's close to that area.

But whether or not it's the exact top of that dome is something I couldn't say at this point.

Q. Well, I only wanted to illustrate a point here, that above the pressure vessel dome and below the removable drywell head -- and those are words, as I said, that come from Figure 6.2-1 of the PSAR -- there is a region of volume where it seems to me things can get relatively stagnant with respect to circulation that might be counted upon to homogenize the hydrogen concentration throughout the containment vessel.

And, sure enough, that's the place where you have a monitoring intake station, and that's a logical place for one, it seems to me.

On the other hand, if, because that is a local high point, hydrogen begins to collect there, I find it awfully difficult to see how the recombiners up near the top of the containment are going to do much good with respect to cleaning up the hydrogen in that relatively -- what I think is a relatively stagnant volume right above the pressure vessel upper head.

Well, I'm just illustrating the kinds of things that I'm concerned about with respect to hydrogen pocketing. I believe you said that you are not sure yourself to what extent the analysis looked at the actual arrangement of things inside the containment in determining where these monitoring station intakes should be. Is that a --

A. That's correct. I'm not sure where they assume that there was equipment in place. I feel fairly

7-9

assured that they did put the major structures into the model when they developed it.

I'd like to point out that the hydrogen levels, as you go from lower elevations to higher elevations, really remain fairly much the same.

Q. How do you know that?

A. Hydrogen doesn't stratify, once it's mixed. That's from test results.

Q. Let me probe that point just a moment. Are you saying that there are test results that indicate that if you thoroughly mix a certain amount of hydrogen with air, then stop the mixing and let that volume sit undisturbed for some period of time, the hydrogen does not tend to diffuse upward?

A. When you say "undisturbed," you're referring to a completely stagnant condition. That does not exist in this situation.

There is -- For this particular purpose, there is a fairly large air current flow inside the dry-well, which would -- which dominates the diffusion characteristics of hydrogen.

It's pretty weak. The diffusion characteristic of hydrogen is pretty weak. And I believe in the Section 6.2.5 of the PSAR, they develop some analyses to show the use of Grashof numbers at fairly low

7-10

velocities will maintain the diffusion of hydrogen, or will keep the concentration of hydrogen fairly constant. It doesn't require much velocity at all.

Q Now, did you indicate that this conclusion is based on some test results of some sort?

A Based on test results -- I believe they were done at Los Alamos.

The analysis is just based on dimensional analysis to show that. So there's two different ways.

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7-11

THE WITNESS: I should also point out that we have fairly extensive experimental programs being developed right now to look into all aspects of hydrogen generation, transportation, whether it can -- how well it will mix with the air in the first place and other such items.

BY JUDGE LINENBERGER:

Q Does the existence of these programs, that you say we have now, imply that the Staff has some uneasiness about how well the hydrogen concentration will reach a uniform distribution within a containment, such as the Allens Creek containment?

A The thrust of the programs is not so much because we have concerns about that particular aspect. It's more -- mainly due to the fact that we have possibilities of much higher hydrogen generation rates than were previously assumed.

The behavior of fairly high concentrations of hydrogen is the major aspect of this study.

Q Well, I can rephrase what you've said in the following way, that in the event of a metal/water reaction, that gives rise to a more rapid rate of evolution of hydrogen than would be the case with the core covered, the lack of adequate mixing could have considerably more serious consequences. And, therefore,

7-2

if one is worried about the lack of an adequate mixing, these kinds of programs you're talking about would help either determine whether those worries are real or not.

Now, I don't know whether that really is the thrust of what you're saying. But it seems to me that's something I can infer from what you've said.

A. That can be inferred, yes.

Q. Have you ever blown up any hydrogen?

A. No.

Q. It can be impressive when you're not expecting it especially.

MR. COPELAND: I suspect that's true for anything that blows up, Judge.

BY JUDGE LINENBERGER:

Q. On page 6 regarding our question directed to testing the operability of the monitoring, alarm and recombiner systems, you indicate certain things can be tested and calibrated, and the calibration can be completed from the control room.

This does not, per se, indicate whether there will be a requirement that things be periodically tested, analyzers recalibrated and so forth.

Do you know whether there will be -- for the operation of a facility such as Allens Creek, is currently the Staff's position to require periodic testing and

7-13

recalibration?

A. Yes. It will be in the standard tech specs.

Q. Okay. Just one more question to -- a further illustration of the basic concern that the Board has here.

If the monitoring station near the -- above the top of the reactor pressure vessel upper head -- and, presumably, below the removable drywell head, started to indicate that the hydrogen concentration was beginning to exceed three percent, or even was approaching three percent with no evidence of leveling off, I would be concerned because I don't see a diffusion or flow or mixing path that connects that region above the pressure vessel with the region up above where the recombiners are located.

So I don't understand how certain regions there can communicate with the recombiners to allow them to do their job.

And --

A. Perhaps if I explained the drywell mixing system, it will help.

Q. All right.

A. There is -- penetrating the top of the drywell not -- I don't think it's at the vessel head cover, but at the top portion, there are redundant lines that penetrate

7-14

the drywell that after the initial phase of the accident is over, the operator will turn on some fans. This will take air from the containment at approximately 500 standard cubic feet per minute, and blow it into the drywell.

This will cause the drywell air pressure to increase until the suppression pool level goes down to the first vent. Then you have a mixture of the fresh containment air with the drywell air, bubbling through the suppression pool, where it can be recombined by the recombiners.

That has been analyzed to show that you can maintain the hydrogen concentrations inside the drywell below four percent, using just one of the drywell mixing systems.

As far as the area that is above the reactor vessel head, I couldn't say with certainty that the drywell mixing system enters at that point, but when you consider that that area is fairly small, that the temperatures almost certainly would have to be below the auto ignition point.

And there are, as far as I know, no equipment that could cause an electrical spark. All of these things combined would make me feel somewhat assured that there is not a problem with that particular area.

7-15

The drywell itself is a problem. That's why we have the drywell mixing system.

Q Has there been a specification or criterion developed yet that will determine when the drywell mixing system is to be activated?

A. Yes.

Q And what is that?

A. The current analysis shows that the drywell mixing system has to be initiated no later than eight to nine hours after the LOCA. It can be initiated before, once the initial blowdown is over, and you build up the reactor vessel with water.

So you do have plenty of time for actuation. This is assuming 10 CFR Part 50.44 release rates.

Q I had not before thought about that drywell mixing system, in terms of its aiding the hydrogen -- smoothing out of the hydrogen concentration or homogenizing the hydrogen concentration.

So I'm glad you mentioned that. I don't think it is mentioned in your testimony.

A. I thought I had mentioned it.

Q Maybe I missed it.

A. Let me see if I can find it.

(Pause.)

I'm surprised I left it out. I guess I was

7-16

concentrating too much on answering the specific Board questions.

But, yes, that's -- The reason why we have the drywell mixing system is to take care of that problem, and that problem alone, because it was a concern raised at the CP stage for GESSAR.

Q Gee, I would think that would deserve more emphasis than the containment spray system, for example, in assisting with the --

A Certainly, it's much more important.

JUDGE LINENBERGER: Well, I guess that's all the questions I have for now, Mr. Chairman. I still have some residual concern about the effectiveness of this mixing, because I am aware of a number of types of industrial accidents that have occurred because there has not been adequate mixing.

But I think we've gone as far as we can go here with this. So that ...

(Bench conference.)

MR. COPELAND: Your Honor, our witnesses on this issue are still yet to testify; and I would expect they would be able to address the problem, when we get to that.

I am reminded that Mr. Weingart did, in fact, discuss this problem to some extent the last time he testified.

(Further Bench conference.)

JUDGE WOLFE: All right. Is there cross on Board questions, Mr. Copeland?

MR. COPELAND: No, sir.

JUDGE WOLFE: Mr. Doherty?

MR. DOHERTY: Yes, Your Honor.

RECROSS-EXAMINATION

BY MR. DOHERTY:

Q. Can you give us a better idea of the volume of this space beneath the drywell head that we've been discussing?

A. The space that is beneath the drywell head cover and the -- I'm sorry -- the reactor vessel head cover and the reactor vessel head itself, that dome shape --

Q. Yes. I think you know what we mean.

A. I don't know the volume offhand.

Q. Is this the first time you've ever had this brought to your attention, just now?

A. Had what brought to my attention?

Q. That there might be a problem with the hydrogen eddying up there above the vessel?

A. The Staff has looked at the possibility for pocketing and the consequences thereof for this plant, and conclıded that the potential for pocketing, to the extent where you'd have any problems, was negligible.

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Q Has the Applicant committed to use the hydrogen recombiner by Westinghouse that's in your testimony, described?

A. That's the model referenced in the PSAR.

MR. DOHERTY: Okay. No further questions.

JUDGE WOLFE: Redirect, Mr. Dewey?

MR. DEWEY: Yes, just a couple of questions perhaps.

REDIRECT EXAMINATION

BY MR. DEWEY:

Q In this area of concern under the dome, would it be -- if that area became a problem, wouldn't the hydrogen inerting system, the CO_2 system, couldn't that ultimately take care of the problem?

A. That would be -- could take care of the problem, yes.

Q Okay. You also speak in your testimony of a backup containment hydrogen purging system; is this the same as the CO_2 system?

A. No.

MR. DOHERTY: Objection, Your Honor. I think it's outside the scope of the Board's question.

MR. DEWEY: That was one of the questions, the nature of the backup containment --

JUDGE WOLFE: Well, hold it. Hold it.

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MR. DEWEY: Excuse me.

JUDGE WOLFE: The witness had already answered no. Was that your answer?

THE WITNESS: The answer was no.

JUDGE WOLFE: He had answered no before your objection was heard.

MR. DOHERTY: Then I move that the answer be struck.

JUDGE WOLFE: All right.

MR. DEWEY: Well, Your Honor, the Board Question No. 7 says, specifically calls for the "nature of the backup containment hydrogen purging system that may be required to function at a time when the containment atmosphere is radioactive."

MR. DOHERTY: But that's not a Board question coming from this panel today.

MR. DEWEY: I think it's a followup of this line of questioning, of Judge Linenberger's questions regarding this area underneath the -- that specific area that he was concerned of where there might be a hydrogen backup.

MR. DOHERTY: Well, Your Honor, I don't think it is because I don't think purging system was at issue. The issue there seemed to me to be the possibility of a pocket where hydrogen would eddy, and discussion following seemed to be whether that would be in fact the case.

(Bench conference.)

JUDGE WOLFE: Since the answer is no, the motion to strike is sort of extraneous, and the motion is denied.

All right. Go ahead, Mr. Dewey. Anything more?

MR. DEWEY: That's all.

JUDGE WOLFE: You are excused temporarily, Mr. Fields.

(The witness was temporarily excused.)

MR. COPELAND: Your Honor, at this time the Applicant could like to call Mr. Guy Martin, Jr., and Mr. Walter F. Malec regarding McCorkle Contention 17 on bypass leakage.

I believe Mr. Malec has already been sworn earlier today, and I would ask Mr. Martin be resworn. He previously testified in this case, but I think he was dismissed.

Whereupon,

WALTER F. MALEC

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

JUDGE WOLFE: You may be seated, Mr. Malec. You remain under oath.

Mr. Martin, you will be sworn.

Whereupon,

GUY MARTIN, JR.

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

JUDGE WOLFE: Please be seated.

DIRECT EXAMINATION

BY MR. COPELAND:

Q Mr. Martin, do you have in front of you a document entitled, "Direct Testimony of Guy Martin, Jr. and Walter F. Malec Regarding McCorkle Contention No. 17 - Bypass Leakage"?

BY WITNESS MARTIN:

A Yes, I do.

Q And was that testimony prepared under your direct supervision?

BY WITNESS MARTIN:

A Yes, it was.

Q Do you have any changes to make?

BY WITNESS MARTIN:

A No, I do not.

Q All right, sir.

I believe that you are the person who has answered all the questions contained in the direct testimony on pages 1 through 4; is that correct?

BY WITNESS MARTIN:

A. Yes, that's correct.

Q. Are the answers contained therein true and correct to the best of your knowledge, information and belief?

BY WITNESS MARTIN:

A. Yes, they are.

Q. Did you also prepare -- assist in preparation of Attachment GM-1, which is the affidavit of Guy Martin and Walter F. Malec?

BY WITNESS MARTIN:

A. Yes, I did.

Q. And as to those portions of the affidavit that you prepared, are the statements contained therein still true and correct, as you swore on the 28th day of July 1980?

BY WITNESS MARTIN:

A. Yes, they are.

Q. And do you adopt the written direct testimony and the attachments thereto, which is your prior affidavit, as your testimony in this proceeding?

BY WITNESS MARTIN:

A. Yes, I do.

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Q. Mr. Malec, did you likewise participate in preparation of the affidavit of Guy Martin, Jr., and Walter F. Malec?

BY WITNESS MALEC:

A. I did.

Q. And are the statements contained therein that you helped prepare still true and correct, as you swore them on the 29th of July 1980?

BY WITNESS MALEC:

A. Yes. That was amended by Attachment GM/WFM-2.

Q. All right. Do you adopt that prior affidavit as your testimony in this proceeding?

BY WITNESS MALEC:

A. I do with one correction.

Q. All right.

BY WITNESS MALEC:

A. It's on page 2 of Attachment 2 in the next-to-the-last paragraph. It begins, "All containment isolation valves which have Type 'e' --" change to Capital "E" to be consistent with the notes in Table 6.2-12 of the Allens Creek PSAR.

Q. Would you repeat that change.

BY WITNESS MALEC:

A. On page 2 of Attachment 2.

Q. That's Attachment GM/WFM-2?

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BY WITNESS MALEC:

A. That is correct.

The next-to-the-last paragraph starting out, "All containment isolation valves," the letter "e" that is lower case within the quote should be changed to capital "E" for consistency with the notation of Table 6.2-12 of the Allens Creek PSAR.

Q. Are there any other changes?

BY WITNESS MALEC:

A. No, sir.

MR. COPELAND: At this time, Your Honor, I would ask that the testimony of Mr. Martin and Mr. Malec, together with the Attachment GM-1 and Attachment GM/WFM-2 be incorporated into the record as if read.

JUDGE WOLFE: Any objection?

MR. DEWEY: The Staff has no objection.

MR. DOHERTY: Your Honor, I wish to take the witnesses on voir dire.

JUDGE WOLFE: All right.

VOIR DIRE EXAMINATION

BY MR. DOHERTY:

Q. Mr. Malec, I would like to start with you.

You state your education is at Polytechnic Institute of Technology. I've never heard of the place. I don't mean to say you went to an obscure place, but can

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you tell me where that is?

BY WITNESS MALEC:

A. Yes. It's a recent name change. When New York University sold its School of Engineering and Science to Brooklyn Polytechnic Institute, subsequent to that change of schools, the Institute was renamed Polytechnic Institute of New York.

I started the program at New York University and finished up at the old Brooklyn Poly, whose name is now Polytechnic Institute of New York.

MR. COPELAND: Your Honor, if I might interrupt for just a second.

I hope it's clear to the Board that the exhibits that are contained in here as well as the professional qualifications of these two gentlemen are part of the prior affidavit.

So whenever it is incorporated into the record, that will include their professional qualifications as well.

Excuse me, Mr. Doherty.

BY MR. DOHERTY:

Q. In your position as supervising engineer, how large a staff do you have?

BY WITNESS MALEC:

A. I have two roles as supervising engineer. I supervise a staff of about 22 engineers directly, and approximately 125 engineers and designers indirectly.

Q. And then you just list the Allens Creek plant there. What was that intended to mean? You wrote, "Houston Lighting & Power Company - Allens Creek."

What were you saying there? Is that saying that's your sole responsibility in terms of these things below it?

BY WITNESS MALEC:

A. That's correct.

Q. What is HVAC, please?

BY WITNESS MALEC:

A. Heating, ventilating and air conditioning.

Q. Okay, and on what components have you done stress analysis?

BY WITNESS MALEC:

A. I beg your pardon?

Q. On what components have you done stress analysis, or are you responsible for the stress analysis for?

BY WITNESS MALEC:

A. The stress analysis for the piping on Allens Creek.

I have not done it directly. I have administrativ supervision responsibility, project execution for the

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Stress Analysis Group. They have their own technical supervisor.

Q. Okay. Prior to that, I see you were a principal engineer, and you list as your foremost function, "Responsible for preparation and maintenance of ECCS and BOP flow diagrams."

Now, maintaining a diagram to me implies some kind of keeping up with changes; is that essentially what that is?

BY WITNESS MALEC:

A. That's correct.

Q. And then prior to that you were a senior engineer. Was that a supervisory position or not?

BY WITNESS MALEC:

A. No, sir. Those are administrative grades. Senior engineers and principal engineers are administrative pay grades.

Lead engineer is a functional title. It's an assignment.

In those two assignments I did in fact as a lead engineer supervise as many as four other individual engineers; but I did report to a supervising engineer.

Q. And as a supervising engineer, did you do direct kinds of engineering work without -- not administrative work, but what you'd call more hand's-on type work?

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BY WITNESS MALEC:

A. I do both administrative and technical work as a supervisor.

Q. All right. Is bypass leakage in a containment vessel similar to leakage in a ship hull or are they really too remote to have much transfer of skill?

BY WITNESS MALEC:

A. There is an analogous relationship, not direct. One could make some type of analogy there.

It's not directly applicable one to one, but it's analogous.

JUDGE LINENBERGER: Sir, could you pull your microphone much closer, please. We are having trouble hearing you up here, too.

Thank you.

BY MR. DOHERTY:

Q. Mr. Martin, I notice that your emphasis seems to have been in radiological engineering or something of that order.

There is a statement in the affidavit of you both that states -- it's on page 2. It states, "However, for practical purposes, the containment must be penetrated by piping and other openings."

This is talking about leak type barriers.

How is it that your expertise in radioactive

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fission products fits into leakage testing?

BY WITNESS MARTIN:

A. Leakages that you get from the containment, post-loss of coolant accident or any other accident which occurs inside of containment, result in off-site radiological exposures.

So, therefore, the nature of my work, as far as like you called it, appears to be a radiological expert, ties in to the leakages from a containment structure in the sense that if it were not for the leakages from the containment, there would be no need to do those calculations.

Or conversely, because of the leakages from the containment, then off-site radiological exposures have to be performed.

Q. Okay. Mr. Malec, did you contribute to the PSAR? Was that some of your duties?

BY WITNESS MALEC:

A. That's correct.

MR. DOHERTY: I have no further questions, Your Honor, and no objections.

JUDGE WOLFE: Absent objection, the testimony of Guy Martin, Jr. and Walter Malec regarding McCorkle Contention No. 17, inclusive of the attachments identified by Mr. Copeland are incorporated into the record as if read.

(Applicant's Testimony of Guy Martin, Jr. and Walter F. Malec on McCorkle Contention No. 17 follows:)

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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 GUY MARTIN, JR. AND WALTER F. MALEC REGARDING McCORKLE CONTENTION NO. 17 - BYPASS LEAKAGE

Q. Mr. Martin and Mr. Malec, have you reviewed your affidavit on McCorkle Contention No. 17, which affidavit is attached hereto as Attachment GM/WFM-1?

A. Yes.

Q. Are the statements contained in the affidavit still true and correct?

A. Yes, except for the changes described in the errata attached hereto as Attachment GM/WFM-2.

Q. Mr. Martin, have you reviewed that portion of the Board's Order of September 1, 1981, wherein the Board calculated the amount of unfiltered leakage (0.0195%) to be approximately 40% of the 0.5% total leakage?

A. Yes, I have.

Q. Would you please address the questions raised by the Board at pages 4 and 5 of the Order?

A. A review of the values presented in the Board's

Order reveals that an arithmetical error has been made. The bypass leakage is 0.0195% of the containment volume per day or approximately 4% of the 0.5% containment leakage rate value. The Board's statement concerning the calculation methodology used to arrive at the bypass leakage value is correct. However, it should be noted that the presently calculated bypass leakage value of 0.0195% of the containment volume, if it were to occur, would result in a thyroid dose value equal to one-half of the 10 CFR Part 100 thyroid dose limit. As stated in the Supplement No. 2 of the Staff Safety Evaluation Report, the atmospheric dispersion factor at the exclusion zone boundary has decreased. However, the bypass fraction of 0.0195% is based on a previously calculated atmospheric dispersion factor which is 67% higher than the dispersion factor which would have been calculated using current NRC guidance and site meteorological data. Consequently, offsite doses would be significantly lower than previously determined if they were calculated using this bypass fraction in conjunction with the current NRC Staff atmospheric dispersion factors. At the Operating License stage the bypass leakage value will be recalculated to reflect the latest NRC methodology and site meteorological data to calculate the site-specific atmospheric dispersion factors.

The Board's mention of the containment leak rate which was specified as both a percentage of weight and volume denotes that the presentation of this value as a function of these two parameters has caused a degree of confusion which warrants some clarification.

In the calculation of the offsite radiological doses to show compliance with the siting criteria of 10 CFR Part 100, the containment is assumed to leak at a constant leak rate of 0.5% of its volume per day. From a dose calculation standpoint, the radionuclides, uniformly mixed in the containment atmosphere, are assumed to leave the containment at this constant leakage rate regardless of the flow rate of carrier air in which they are assumed to be mixed. The maximum containment airborne concentration of these radionuclides will occur at standard temperature and pressure (STP) conditions. Therefore, the air leakage expressed in terms of a fraction of the containment air volume at STP conditions will have the same radionuclide concentration and hence will be selected as the technical specification value to be met, in testing, in order to remain within the dose criteria of 10 CFR Part 100. The leakage rate can be expressed as a percentage of weight per a unit of time by converting volume to weight. Under test conditions, the containment will be pressurized, the leak rate measured and compared to this technical specification value. The Board's statement is correct in that there is no difference in percent by weight and percent by volume no matter how it is expressed, since, ultimately, the actually measured quantity will be either a mass or a volume of air per a unit of time.

Attachment GM-1

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF GUY MARTIN, JR.

State of New Jersey
County of Bergen

I, Guy Martin, Jr., Supervising Radiological Assessment Engineer, Allens Creek Project, for Ebasco Services Incorporated, of lawful age, being first duly sworn, upon my oath certify that I have reviewed and am thoroughly familiar with the statements contained in the attached affidavit addressing intervenor Brenda McCorkle's Contention 17 regarding filtration system leakage. All statements contained therein,which relate to Ebasco Services Incorporated scope of supply for the Allens Creek Nuclear Generating Station, are true and correct to the best of my knowledge and belief.

[signature]

Subscribed and sworn to before me this _____ day of __________, 1980.

[signature]

CAROL A. OPITENOK
NOTARY PUBLIC OF NEW JERSEY
MY COMMISSION EXPIRES SEPT. 18, 1983

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 No. 1)	)	

AFFIDAVIT OF WALTER F MALEC

State of New Jersey
County of Bergen

I, Walter F Malec, Supervising Mechanical Nuclear Engineer, Allens Creek Project, for Ebasco Services Incorporated, of lawful age, being first duly sworn, upon my oath certify that I have reviewed and am thoroughly familiar with the statements contained in the attached affidavit addressing intervenor Brenda McCorkle's Contention 17 regarding filtration system leakage and that all statements contained therein are true and correct to the best of my knowledge and belief.

Walter F Malec

Subscribed and sworn to before me this ____ day of ____ ,1980.

CAROL A. OPITENOK
NOTARY PUBLIC OF NEW JERSEY
MY COMMISSION EXPIRES SEPT. 13, 1983

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	§	
No. 1)	§	

AFFIDAVIT OF GUY MARTIN, JR.
AND WALTER F. MALEC

My name is Guy Martin, Jr. My business address is Two World Trade Center, New York, N. Y. I am the Supervising Radiological Assessment Engineer for the Allens Creek Project employed by Ebasco Services Incorporated. The statement of my background and qualifications is attached as Exhibit I to this testimony.

My name is Walter F. Malec. My business address is 160 Chubb Avenue, Lyndhurst, N. J. I am the Supervising Mechanical Nuclear Engineer for the Allens Creek Project employed by Ebasco Services Incorporated. The statement of my background and qualifications is attached as Exhibit II to this testimony.

This affidavit addresses the issues raised in McCorkle Contention No. 17. The contention states that the Allens Creek containment as designed will allow 20 percent of the containment leakage to bypass the filtration systems.

I. Introduction

The Allens Creek containment consists of a free-standing steel shell 1 1/2 to 1 3/4 inches thick which enclo · the reactor vessel holding the reactor fuel. The containment is designed to protect the public from the release of radioactive fission products by providing a leak-tight barrier. However, for practical purposes, the containment must be penetrated by piping and other openings. Although these penetrations are sealed by some means such as redundant valving, a certain quantity of leakage is inevitable. NRC regulations (10 CFR, Part 50, Appendix J) limit the quantity of leakage allowed.

II. Containment Leakage Expected for Allens Creek

The Containment Vessel is a seismic Category I steel shell designed to confine the radioactive materials, gases under pressures and temperatures associated with a loss-of-coolant accident and all other abnormal operating conditions. The design leak rate will be 0.5 percent by weight of the contained atmosphere per day at calculated peak pressure. The Containment Vessel will be designed to contain any leakage from the drywell and the noncondensable gases from reactor vessel blowdown by the safety/relief valves or from the rupture of the largest pipe inside the drywell.

To determine the type of leakage which can be expected, a list of all potential leakage paths through containment penetrations was compiled (Table 6.2-12a of the Preliminary Safety Analysis

Report). This list is reproduced as Exhibit A. From this list, only six penetrations constitute potential unfiltered leakage paths. These six penetrations are listed in Table 6.2-13 of the PSAR and the table is reproduced as Exhibit B.

In arriving at the list contained in Exhibit B, an evaluation was made of all lines which penetrate the containment to determine the number and types of barriers to bypass leakage provided for each line. The types of bypass leakage barriers considered were as follows:

(a) Isolation valve outside containment.

(b) Isolation valve inside containment.

(c) Closed Category I piping system inside containment.

(d) Closed Category I piping system outside containment.

(e) Water seal in line.

(f) Line beyond isolation valve outside containment vented to annulus for filtration by the Standby Gas Treatment System (SGTS).

(g) Line terminates outside containment in filtered ECCS Area of Auxiliary Building.

Leakage barriers of types (c) through (g) effectively eliminate any bypass leakage. Leakage barriers of types (a) or (b) limit but do not eliminate bypass leakage. Therefore, lines containing any of the bypass leakage barriers (c) through (g) were not considered as potential bypass leakage paths. Lines containing only types (a) or (b) were included in Exhibit B as potential unfiltered leakage paths.

III. Unfiltered Leakage

The amount of containment leakage allowed in the Technical Specifications will be significantly less than that which would produce total off-site doses equal to the 10 CFR 100 limits. The contributors to this total leakage include the Standby Gas Treatment System releases, leakage to the controlled ventilation ECCS area of the Auxiliary Building and all unfiltered bypass leakage. The actual value of the bypass leakage technical specification will be determined as a result of LOCA dose calculations performed when the FSAR is prepared for submittal. However, a value of .0195 percent/day of the containment volume is the present best estimate of the maximum total unfiltered bypass leakage based on preliminary LOCA dose calculations. These dose calculations are provided in detail in Section 15 and Appendix 15.A of the PSAR.[1/]

IV. Tests and Inspections

In order to assure that the containment will maintain its expected level of leak-tightness, Applicant will conduct a leak testing program in accordance with

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1/ The fraction of total containment leak rate technical specification which will be released via potential bypass leakage lines is quoted at PSAR, p. 15.A-4b as 2.9×10^{-2}. This number is a typographical error. The correct value is 3.9×10^{-2}.

Appendix J of 10 CFR 50. As required by Appendix J, three types of tests will be performed:

Type A - This test will measure the primary reactor containment overall integrated leakage rate. It will be conducted after the containment is completed and ready for operation and again about once every three and one-third years thereafter. In addition, any major modification or replacement of components of the primary reactor containment performed after the initial leak rate test shall be followed by either a Type A test or a Type B test of the area affected by the modification.

Type B - Appendix J defines these tests as those:

> intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for the following primary reactor containment penetrations:
>
> 1. Containment penetrations whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetrations fitted with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies.
>
> 2. Air lock door seals, including door operating mechanism penetrations which are part of the containment pressure boundary.
>
> 3. Doors with resilient seals or gaskets except for seal-welded doors.
>
> 4. Components other than those listed above which must meet the acceptance criteria in III.B.3 of Appendix J.

Except for containment air locks, Type B tests will be conducted during each reactor shutdown for major fuel reloading but in no case at intervals greater than two years. The seals of the personnel air locks will be tested after each opening or, if left unopened, at an interval not to exceed one year.

Type C - Type C tests are those intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:

1. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum, relief, and instrument valves;

2. Are required to close automatically upon receipt of a containment isolation signal in response to controls intended to effect containment isolation;

3. Are required to operate intermittently under post-accident conditions; and

4. Are in main steam and feedwater piping and other systems which penetrate containment of direct-cycle boiling water power reactors.

Type C tests shall be performed for isolation valves during each reactor shutdown for major refueling.

V. Conclusion

The Allens Creek containment will be designed to limit leakage to 0.5 percent by weight of the containment atmosphere per day at calculated peak pressure. Applicant has calculated that, under loss of coolant accident conditions, a maximum of .0195 percent per day of containment volume may escape via the potential bypass leakage lines and that the resulting doses will not exceed the limits of 10 CFR Part 100. Hence, Intervenor's claim that 20 percent of the containment leakage will bypass filtration systems does not reflect the present plant design and the updated bypass leakage fraction calculations contained in PSAR, Section 15 and Appendix 15.A. Finally, the projected containment integrity will be assured by performing the leak-rate tests called for by 10 CFR, Appendix J.

EXHIBIT A

EVALUATION OF POTENTIAL BYPASS LEAKAGE FOR CONTAINMENT PENETRATIONS

System Service	Line Size (in.)	Bypass Leakage Barriers*	Considered Potential Bypass Path
Main Steam Lines A, B, C, and D	26	A, B, H	No
Feedwater A and B	20	A, B, E	No
RHR Pump A, B, and C Suction from Suppression Pool	24	A, D, E, G	No
RHR Shutdown Suction From Recirculation Loop	20	A, B, D, E, G	No
RHR Return A and B to Recirculation Loop	12	A, B, D, E, G	No
RHR A, B, and C LPCI	12	A, B, D, E, G	No
RHR A, B, and C Pump Test Lines to Suppression Pool	18	A, D, E, G	No
HPCS Pump Suction from Suppression Pool	24	A, D, E, G	No
HPCS Pump Discharge	12	A, B, D, E, G	No
HPCS Test Line to Suppression Pool	12	A, D, E, G	No
HPCS Minimum Flow Line	4	A, D, E, G	No
LPCS Pump Suction from Suppression Pool	24	A, D, E, G	No
LPCS Pump Discharge to Pressure Vessel	12	A, B, D, E, G	No
LPCS Test Line		A, D, E, G	No

EXHIBIT A

System Service	Line Size (in.)	Bypass Leakage Barriers *	Considered Potential Bypass Path
Steam Supply the RCIC Turbine and RHR Heat Exchanger	10	A, B, D	No
RCIC and RHR to Head Spray	6	A, B, D, E	No
RCIC Pump Suction from Suppression Pool	6	A, D, E	No
RCIC Turbine Exhaust to Suppression Pool	12	A, D	No
RCIC Pump Discharge Minimum Flow Bypass	2	A, D, E	No
RCIC Vacuum Pump Discharge	2	A, G	No
CRD Pump Discharge	2	A, B, E	No
Station Air Supply	2	A, B	Yes
Instrument Air Supply	2	A, B	Yes
Reactor Building Closed Cooling Water Supply	14	A, B, E	No
Reactor Building Closed Cooling Water Return	14	A, B, E	No
Reactor Water Clean-up to Condenser and Radwaste	4	A, B, E	No
Reactor Water Clean-up Backwash Transfer Pump Discharge	4	A, B, E	No
Main Steam Drains to Condenser	3	A, B, E	No

EXHIBIT A

System Service	Line Size (in.)	Bypass Leakage Barriers *	Considered Potential Bypass Path
LPCS Minimum Flow Line	4	A, D, E, G	No
RHR Pump Minimum Flow Line (Typ 3)	4	A, D, E, G	No
Chilled Water System Supply	4	A, B, E	No
Chilled Water System Return	4	A, B, E	No
Containment Purge Supply	4	A, B, F	Yes
Hydrogen Purge Exhaust	4	A, B, D	No
Containment Vacuum Relief A and B	18	A, B, F	No
Fuel Transfer Tube	32	A, B, E	No
Demineralized Water Supply to Containment	4	A, B, E	No
Discharge from Fuel Pool Cooling and Cleanup to Containment Pool	6	A, B, E	No
Inlet to Fuel Pool Cooling and Cleanup from Containment Pool	10	A, B, E	No
Condensate Makeup Supply	2	A, B, E	No
Drywell Floor Drain Discharge Header	3	A, B, E	No
Containment Floor Drain Discharge	3	A, B, E	No

EXHIBIT A

System Service	Line Size (in.)	Bypass Leakage Barriers*	Considered Potential Bypass Path
Containment Ventilation Air Supply and Exhaust	36	A, B, F	No
Drywell Containment Equipment Drains	3	A, B, E	No

* Possible Bypass Leakage Barrier Designation :

A. Isolation valve outside containment
B. Isolation valve inside containment
C. Closed Category I piping system inside containment
D. Closed Category I piping system outside containment
E. Water seal in line
F. Line beyond isolation valve outside containment vented to annulus
G. Line terminates outside containment in filtered ECCS area of auxiliary building

EXHIBIT B

POTENTIAL UNFILTERED CONTAINMENT BYPASS LEAKAGE PATHS

Description	Line Size (in)
Station Air Supply	2
Instrument Air Supply	2
Containment Purge Supply (2)	4
Main Steam Line Guard Pipe	
Feedwater Line Guard Pipe	
Personnel Air Lock	

GUY MARTIN, JR
Supervising Engineer
Radiological Assessment

SUMMARY OF EXPERIENCE (Since 1965)

Total Experience - Fifteen years participation in Safety Analysis Reports, Environmental Reports, SAR amendments, licensing documents, and cost analysis for insurance premium determination.

Professional Affiliations - American Society of Mechanical Engineers
Health Physics Society
American Nuclear Society
Intern Engineer in New York State, Certificate No. 022127

Education - MS, Polytechnic Institute of New York, 1976 Nuclear Engineering
BE, City College of the City of New York, School of Harvard University School of Public Health, 1977 - Radiological Surveillance Course.

REPRESENTATIVE EBASCO PROJECT EXPERIENCE (Since 1973)

Supervising Engineer

Participate in the coordination, technical review and preparation of Safety Analysis Reports (SAR), Environmental Reports (ER), SAR amendments and other licensing documents (e.g., Appendix I to 10 CFR 50 studies) for submittal to the Nuclear Regulatory Commission as part of the application for Construction Permit and Operating License of nuclear power plants.

Areas of complete responsibility include sections of the SAR dealing with the radiological dose assessment work associated with normal and hypothetical accident conditions. In this regard, conduct safety reviews of systems, specifications and operation from a nuclear safety viewpoint and check their compliance with established nuclear safety criteria.

Furnish technical support in the preparation of testimonies for safety hearings and ACRS presentation. Study, develop, maintain and use appropriate methods, including computer programs for evaluating radiological exposures.

PRIOR EXPERIENCE (3 years)

Equitable Life Assurance Society of the US
Cost Analyst

Work involved calculating and analyzing cost of various activities performed throughout the company; assisting departmental managers in their budget preparation work. Made statistical studies for determination of activity costs and providing company's actuaries support information for premium determination.

Dividend Specialist

Reviewed and analyzed dividend and claim reserve calculations. Prepared d ursement authorizations and dividend information r orts for policy holders. Participated in training programs for new employees.

Publications

Martin, G and J Thomas 1978. Meeting the dose requirements of 10 CFR 100 for site suitability and general design criteria 19 for control room habitability: a parametric approach. Transactions of American Nucl ar Society 24th Annual Meeting, Vol. 28.

Martin, G, D Michlewicz and J Thomas 1978. Fission 2120: a program for assessing the need for engineered safety feature grade air cleaning systems in post - accident environment. Proceedings of 15th DOE Nuclear Air Cleaning Conference.

Letizia, A P, G Martin and J F Silvey 1979. - Implications for nuclear facilities of changes being initiated in the NRC standard atmospheric diffusion model. Proceeding of the 41st Annual Meeting of the American Power Conference.

Bhatia, R K, Mauro, J, Martin, G. Effects of Containment Purge on the Consequences of a Loss-of-Coolant Accident. Transactions of American Nuclear Society 1980 Annual Meeting.

Born — Philadelphia, Pennsylvania

Education — Polytechnic Institute of Technology, degree of Engineer in Nuclear Engineering - 1978
Massachusetts Institute of Technology, MS in Nuclear Engineering - 1970
U.S. Coast Guard Academy, BS - 1968

Member — American Nuclear Society

Licensed — Registered Professional Engineer in the State of New York (No. 56673)

Experience:

1980 — Ebasco Services Incorporated, Lyndhurst (NJ) Office; Supervising Engineer, Mechanical-Nuclear Engineering Department:

Houston Lighting & Power Co - Allens Creek NGS - Unit No. 1 - 120^ MW(e) BWR

Technical and administrative responsibility for mechanical, fire protection, plumbing, HVAC, stres: analysis, hangers and supports, and inservice inspection ac: ities. Includes schedules, budgets, and client relatio. :.

1978-1980 — Ebasco Services Incorp-ated, Lyndhurst (NJ) Office; Principal Engineer, Mechanical-Nuclear Engineering Department

Houston Ligh. ng & Power Co - Allens Creek NGS - Unit No. 1 - 1200 MW(e) BWR, Lead NSSS Engineer

Responsible for preparation and maintenance of ECCS and BOP flow diagrams, piping layouts, system design descriptions, inservice inspection provisions, Nuclear Island building general arrangements, PSAR and FSAR preparation, equipment sizing and specification, NSSS vendor interface for correspondence, drawing review, and contract administration.

1976-1978 — Ebasco Services Incorporated, New York Office; Senior Engineer, Mechanical-Nuclear Engineering Department including:

Houston Lighting & Power Co - Allens Creek NGS - Unit No. 1 - 1200 MW(e) BWR, Lead NSSS Engineer

Louisiana Power & Light Co - Waterford SES Unit No. 3 - 1165 MW(e) PWR. Lead NSSS Engineer

(Same responsibilities as listed for 1978-1980 above.)

1976-1978 (Cont'd)	Responsible for preparation and maintenance of ECCS and BOP flow diagrams, piping layouts, system design descriptions, inservice inspection provisions, Nuclear Island building general arrangements, PSAR and FSAR preparation, equipment sizing and specification, NSSS vendor interface for correspondence, drawing review, and contract administration.

* * * * *

1974-1976	United States Coast Guard, Marine Inspection Office, New York; Lieutenant - Supervisory Boiler Inspector. Responsibility for supervision, assignment and training of Marine Inspectors in largest Marine Inspection Office in country. Inspection of hull and machinery material condition of U.S. flag and foreign merchant vessels, and pressure vessels under construction. Application of various laws and regulations of the United States, ASME Code, ANSI, TEMA, NEC and NFPA Standards. Review of engineering plans and alterations, reports from field and resident inspectors.
1973-1974	United States Coast Guard, USCGC Spencer (WHEC-36), Lieutenant - Chief Engineer. Responsibility for operation, maintenance and repair of hull and engineering plant of 6200 slip twinscrew steamship. Direct supervision of 40 officers and men. Duties included preparation of repair specifications and maintenance of vessel records. Received Coast Guard Achievement Medal for superior performance of duty.
1970-1973	United States Coast Guard, Marine Inspection Office, New York, Lt and Ltjg - Marine Inspector. Inspection of hull and machinery of U.S. and foreign flag merchant vessels.
1968-1969	United States Coast Guard, USCGC Mellon (WHEC-717), Ensign, Assistant Engineer Officer.

ERRATA

Section II, p. 2, delete the second sentence in the first paragraph. Add the following in its place:

> The maximum allowable leakage rate of 0.5 weight percent per day at the calculated peak internal pressure related to the design basis accident is as specified for pre-operational tests in the Technical Specifications.

Section II Containment Leakage Expected for Allens Creek, page 2: Delete the second paragraph in its entirety and the third paragraph up to "The types of bypass leakage barriers....."

Substitute the following:

"To determine the type of leakage which may be expected, all containment penetrations are initially considered."

A) Mechanical Penetrations are those penetrations through which piping or tubing enters or leaves the containment. The penetration assemblies themselves are not considered as potential bypass leakage paths since they are of welded construction. Potential leakage through the pipe itself was considered. A listing of piping penetrations is included in updated Table 6.2-12 of PSAR (Amendment No. 59 dated June 1981). Potential unfiltered leakage paths are also indicated on this table. Potential unfiltered bypass leakage paths through piping were arrived at by considering the types of bypass leakage barriers for the pipe.

- Pg. 3 add: "(h) Main Steam Isolation Valve Leakage Control Systems" after item g.

  Change the first sentence to read "Leakage Barriers of types (c) through (h)....."

- Pg. 4 change "(c) through (g)" to (c) through (h)".

- Pg. 4 add at the end of Section II the following:

  "Instrument tubing, other than the list in table 6.2-12, which penetrates the containment are designed considering the guidelines of NRC Reg. Guide 1.11. Instrument tubes, other than those indicated otherwise in table 6.2-12, are not considered bypass leakage paths since they have a Type "c" or Type "d" barrier".

On page 4, at the conclusion of Section II, add the following:

B) Electrical Penetrations are not considered bypass leakage paths since any leakage would be into the Shield Building Annulus. This annulus is served by the Standby Gas Treatment System.

C) The Personnel Air Lock and Equipment Hatch will be considered as potential unfiltered leakage pathways and will be tested to 10 CFR § 50 Appendix J Type B criteria.

- Section IV, p. 6 Type C: Delete the second sentence to the end of the section and add the following at that point:

  "The containment isolation valves are indicated in Table 6.2-12 of the PSAR (Amendment 59 dated June 1981).

  All containment isolation valves which have Type "E" bypass leakage barriers will be leak tested in accordance with ASME - B&PV Code Section XI, subsection IWV, category A requirements for leak tightness".

- Delete Exhibits A&B and replace with Table 6.2-12 of the Allens Creek PSAR.

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MR. COPELAND: The witnesses are tendered for cross-examination.

JUDGE WOLFE: Mr. Dewey, cross?

MR. DEWEY: Staff has no cross.

JUDGE WOLFE: Mr. Doherty.

MR. DOHERTY: Yes, Your Honor.

MR. COPELAND: I would like to note for the record, Your Honor, that this was one of Ms. McCorkle's contentions, and she has not shown up here today.

JUDGE WOLFE: Duly noted.

CROSS-EXAMINATION

BY MR. DOHERTY:

Q. Well, at page 2 of the direct testimony, there is mention of a 0.5 percent containment leakage rate.

Are we talking about a volume there or are we talking about a weight percentage?

BY WITNESS MARTIN:

A. They are synonymous.

Q. Okay.

JUDGE LINENBERGER: Mr. Martin, I did not understand your answer.

WITNESS MARTIN: They are synonymous. I just want to add that that was one of the concerns of the Board, also, and that was responded to in this direct testimony.

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JUDGE LINENBERGER: I believe you discuss that later on in the testimony.

WITNESS MARTIN: Yes, I do.

JUDGE LINENBERGER: Thank you.

BY MR. DOHERTY:

Q. Now, was that five percent drawn from technical specifications?

BY WITNESS MARTIN:

A. No, it was not. That .5 percent becomes the technical specification.

Q. Now I'm having trouble hearing you. I think you are actually a little too close to the mike, oddly enough.

Could you repeat that?

BY WITNESS MARTIN:

A. The .5 percent was not drawn from technical specifications. It will become one.

Q. I see. There are no technical specifications on this at the moment; is that correct? Is that what you mean by, "It will become one"?

BY WITNESS MARTIN:

A. Well, to my knowledge, technical specifications for the Allens Creek Project have not been prepared yet.

Q. I see what you mean.

You state that the Supplement No. 2 to the SER

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shows that the atmospheric dispersion factor at the exclusion zone boundary has decreased.

Can you identify where in the SER it says that? I was looking for it a couple of days ago; I didn't locate it. Did you by any chance?

BY WITNESS MARTIN:

A. It's in Section 2.3 of the SER.

Q. 2.3?

BY WITNESS MARTIN:

A. Of the SER Supplement No. 2.

Q. You submitted, I think what you called Attachment 2, or anyway it's called "Errata," and let's see here. I'm sorry, I didn't mean to get you off. I think that my own notes were confusing there.

Is there meant in that -- just that phrase at line 15, page 2, "current NRC guidance," does that mean to imply that there was a recent change in NRC guidance, Mr. Martin?

BY WITNESS MARTIN:

A. Yes, there was.

Q. What was that change, please?

BY WITNESS MARTIN:

A. The NRC has within the past year and a half to two years come up with a new or revised methodology, which would be acceptable to them in the evaluation of

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the so-called atmospheric dispersion factors which are used for calculating off-site doses post-accident.

Q You state then that as a consequence of that change, the "doses would be significantly lower," as a result of this bypass fraction under the new method.

What is significantly lower? Is that on the order of ten times or --

BY WITNESS MARTIN:

A On line 13 of page 2.

Q Yes.

BY WITNESS MARTIN:

A Sixty-seven percent.

Q All right. That's what you were referring to.

You state, "At the Operating License stage --" that's further down, at line 19, "-- the bypass leakage value will be recalculated to reflect the latest NRC methodology and site meteorological data."

Does that mean there's got to be some additional data, meteorological data, collected between now and the operating license?

BY WITNESS MARTIN:

A That is correct.

Q I see. So you don't know for sure there will be a new NRC methodology, right? That might stay the same.

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BY WITNESS MARTIN:

A. I couldn't comment on that.

Q. But there will be some new meteorological data?

BY WITNESS MARTIN:

A. Yes, definitely.

Q. Even though that might be the same, also. Okay.

Now, going to page 3, you state that the radionuclides are uniformly mixed in the containment atmosphere and are assumed to leave regardless of flow rate of the carrier air.

Is that an attempt to be conservative or what?

BY WITNESS MARTIN:

A. No, it was not an attempt to be conservative. It just reflects the calculational method.

Q. I see. Okay.

Then the sentence directly below that, starting at line 12, is this standard temperature and pressure -- what is that? Is that standard temperature and pressure for in a nuclear containment or room temperature?

BY WITNESS MARTIN:

A. Standard temperature and pressure is defined as 14.7 psi and 70 degrees Fahrenheit.

Q. Did you say 14.7?

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BY WITNESS MARTIN:

A. 14.7.

Q. All right. You say, "The maximum containment airborne concentration..." will occur at those conditions. Is that guidance from the Regulatory Guide that makes that statement?

BY WITNESS MARTIN:

A. No, it's not.

Q. Would you explain the basis of the statement then?

BY WITNESS MARTIN:

A. If you were to take a volume of air --

Q. Take a what?

BY WITNESS MARTIN:

A. A volume of air, and to that you were to add X quantity of radionuclides in, for example, standard temperature and pressure conditions, if you were now to increase the quantity of air that you had for mixture, you would further, then, dilute that concentration of radionuclides.

Therefore, in any given STP conditions, you have, then, what you would think of as average conditions for mixture; therefore, your quantity of mass would be smaller than at higher temperature and pressure.

Therefore, the concentration that you would

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have at STP conditions would be more because you would be diluting the radionuclides in a lesser quantity of mass of air.

Q. Okay. I think I heard you; I'm still having some trouble.

On page 3 of the affidavit there is a listing of types of bypass leakage barriers, and even though that page -- unless I'm mistaken, that listing remains as part of the testimony submitted here.

Now, are (a) and (b) there, are those referring to the main steam isolation valves, on that page?

BY WITNESS MALEC:

A. Excuse me, Mr. Doherty, may I respond to that?

Q. Certainly.

BY WITNESS MALEC:

A. Not necessarily. That refers to any containment isolation valve.

Q. Okay. Now, then, (c) through (h) refer -- there's no valves involved there. That's the non-valve type barriers?

BY WITNESS MALEC:

A. That's not necessarily true.

Q. Would the primary barrier be a non-valve barrier? Perhaps there's a valve barrier secondarily as a backup?

BY WITNESS MALEC:

A. The valve is a secondary barrier. I should note that in some cases the valve is a necessary portion of the barrier, though.

Q. Then I notice that a correction is added, leakage control systems.

How did that happen to be overlooked, or was it merely overlooked, or what?

BY WITNESS MALEC:

A. It was overlooked.

Q. Are welded joints on penetrations also considered potential bypass leakage barriers?

BY WITNESS MALEC:

A. No, they are not.

Q. Is that because --

BY WITNESS MALEC:

A. Excuse me. They are bypass leakage barriers, but they are not considered as part of the bypass leakage path.

Q. Are there any such welded joints on penetrations, I think sometimes called guard pipes, are there any in the Allens Creek design?

BY WITNESS MALEC:

A. There are on some of the penetrations.

Q. Why are they not considered bypass leakage

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potential?

BY WITNESS MALEC:

A The weld is tested. It's pressure tested. It comes under Type A test.

It will be included in the over-all integrated leak rate test. However, it's not considered a bypass leakage path. The weld is solid.

If I can take you back to your analogy of the ship's hull.

- - -

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ed

Q Go ahead.

BY WITNESS MALEC:

A. When you put the ship in the water, if you have done the pressure test on a pressure vessel or when you make that welded joint, many times they will apply a water spray to the outside to assure that that weld is intact and will not leak when the ship goes in the water.

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

JUDGE WOLFE: Yes.

WITNESS MALEC: The welded joint --

BY MR. DOHERTY:

Q Hold it just a minute.

Mr. Malec, did I just show you a copy of Branch Technical Position CSD 6-3?

BY WITNESS MALEC:

A. You did.

Q And did I draw your attention to Section B, Branch Technical Position, particularly one part which says, "The following leakage barriers end paths, which do not terminate within the secondary containment should be considered potential bypass leakage paths around the leakage collection infiltration systems of the secondary containment," and with particular attention to Part C, "Welded joints on penetrations; e.g., guardpipes, which pass through both the primary and secondary containment barriers."

Do you have a copy of that with you that you can consult?

BY WITNESS MARTIN:

A. Yes, we do.

Q. Good. Would that seem to indicate that tha·· should be one of the paths to be considered on page 3 of your affidavit?

BY WITNESS MALEC:

A. If this were to be a leakage path, it would be picked up in the 10 CFR 50, Appendix J, Type A integrated leak rate tests for the containment.

JUDGE WOLFE: Are you turned on, Mr. Malec.

THE REPORTER: Yes, he is, but that microphone, you have to get very close to it for it to come through.

JUDGE LINENBERGER: I didn't hear a single word of that answer.

WITNESS MALEC: If there were to be leakage through any of the welded penetration areas, for instance the guardpipe as referenced in this particular document that you showed to me, it would be picked up as a portion of the 10 CFR 50, Appendix J, Type A integrated leak rate tests for the containment.

I don't think that they will be a problem

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because they have welded construction; however, they will be included in the testing that Houston --

JUDGE CHEATUM: Mr. Malec, you have a very soft voice. If you could harden it up a bit, we could hear you better.

WITNESS MALEC: They will be picked up as part of the testing that Houston Lighting & Power will have to perform on a periodic basis.

BY MR. DOHERTY:

Q. Okay. Turning on to page 4, does the errata that is meant to go at the end of the first paragraph on 4, the last sentence says it refers to Type "c" or Type "d" barrier.

Does that refer to page 3 actually? Are those the same thing as closed Category I piping systems?

BY WITNESS MALEC:

A. I'm sorry, I don't follow --

Q. It's fairly confusing. You need three addresses.

First of all, page 4 of the affidavit.

BY WITNESS MALEC:

A. Page 4 of the --

Q. Of the affidavit.

BY WITNESS MALEC:

A. Of the affidavit, yes.

Q. Right, and you indicated on the errata sheet at the foot of it that you wanted to add to Section II on page 4 this paragraph.

BY WITNESS MALEC:

A. Starting with "Instrument tubing"?

Q. Yes.

BY WITNESS MALEC:

A. I understand.

Q. Now, at the foot of that you refer to "Type 'c' or Type 'd' barrier." That's the last sentence in the errata.

BY WITNESS MALEC:

A. Yes.

Q. What I'm wondering, does that Type "c" or "d" really refer to -- Does it refer to Type "c" and "d" bypass leakages which are enumerated on page 3?

BY WITNESS MALEC:

A. Yes.

Q. I see.

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

JUDGE WOLFE: Redirect, Mr. Copeland?

MR. COPELAND: No, sir.

JUDGE WOLFE: Board questions?

JUDGE CHEATUM: I have no questions.

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BOARD EXAMINATION

BY JUDGE LINENBERGER:

Q. Gentlemen, beginning with your prefiled direct testimony filed under date of September 18, 1981, you refer at the bottom of page 1 and top of page 2 to an arithmetical error in the Board's 1 September 1981 Second Order dealing with Summary Disposition Motions.

A few weeks ago when you were not here, the Board indicated that you are quite right about that. There was an arithmetical error made, and the implication of that is that -- or the implication of our statement with respect to that error is that we agree with your calculation of approximately four percent of the half percent containment leakage as being the value that will be unfiltered.

Now, then, beginning with the sentence that begins at line 4, you have indicated that, "The Board's statement concerning the calculation methodology used to arrive at the bypass leakage value is correct."

Now, since I want to go into the implications of that a little later, I want to make sure I understand what it is you are agreeing with the Board on here.

BY WITNESS MARTIN:

A. I just like to agree with the Board, basically. That's all it is.

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Q. Pardon?

BY WITNESS MARTIN:

A. I just like to agree with the Board. That's all.

(Laughter.)

Q. Let me be explicit here. The thing that the Board noted in the 1 September 1981 Order was that it appeared that rather than stacking up a collection of individual estimated leakage rates to give a total expected leakage rate and therefrom calculating exclusionary doses, it looked to the Board as though what had been done was to start with the permissible exclusionary dose and back into what would be considered acceptable leakage rates.

Now, is that the thing you are agreeing with?

BY WITNESS MARTIN:

A. It is exactly what I am agreeing with.

Q. Okay. Now, let's leave that. I want to come back to it in just a moment, but I want to go now to, again, the same page 2 of your prefiled direct, the sentence beginning with the word "However," at line five and a half, wherein you indicate that a "bypass leakage value of 0.0195 percent of the containment volume, if it were to occur, would result in a thyroid dose value equal to one-half of the Part 100 dose limit."

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Now, my impression is that for these kinds of analyses, a factor of two is not a -- well, for many things like this that one has to calculate, to be accurate within a factor of two is pretty good.

I guess what I'm asking is if this bypass leakage value brings you to within one-half of the Part 100 thyroid dose limit, my impression is that's getting awfully close, because it seems to me that the error in the methodology and the error in inputs to the calculational method could easily lead you to an answer that could, depending on how you put numbers together, vary by a factor of two.

So I want you to tell me why it is you can take comfort from coming within one-half of that Part 100 value in this kind of a calculation?

How can it be so precise that that's a comfortable result?

BY WITNESS MARTIN:

A. This could be a very long answer to your observation.

Let's just take a look at the various portions of parameters which are involved in this calcualtion, and we'll just talk about errors associated with each portion.

Let's just talk about the conservatisms

involved in the calculations themselves.

The first conservatism is in the so-called source term assumption itself. In the calculation of these doses which are applicable in this case of a loss of coolant accident, you assume, number one, that the ECCS has not operated.

You have full core melt indicating that 100 percent of your noble gases have been released to the containment atmosphere, and 25 percent of your iodine has also been released to the containment atmosphere, and thereby available for release to the environment.

That in itself is the fundamental highly conservative assumption that one makes in these calculations.

Number two, the containment structure itself is assumed to leak at its design basis leak rate; in this case, the .5 percent per day.

Once these releases reach the environment, the dispersion, which the cloud now is afforded, is assumed to be only at that level which occurs five percent of the time; or conversely, which is only exceeded only 95 percent of the time.

And lastly, the receptor of that dose is assumed to stay there for the full two-hour period without any regards to the potential emergency type of action

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which could be taken on his behalf.

Now, having accepted all these assumptions as ground rules for performance of the calculations, I don't think there is any doubt in anybody's mind that they are very conservative numbers, and keeping in mind th they are basically performed, calculation studies, for the purpose of showing the site suitability of 10 CFR Part 100.

Then having thrown in all these conservatisms in the calculation, the factor of two within Part 100 indicates that there is a high level of confidence that if such an accident would occur, then the Part 100 doses would not be exceeded.

So, therefore, a factor of two below the guidelines is a very comfortable margin to have.

Q. All right, sir.

Don't let me put words in your mouth here. I think from what you've said that even if you came out with a factor of two above the Part 100 dose limit, you would still consider it acceptable.

BY WITNESS MARTIN:

A. No.

Q. No? Is that because of the legality of the matter, because Part 100 shall not be violated, or is it in terms of the realism of the calculations?

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BY WITNESS MARTIN:

A. Well, your first assumption, yes, is correct, because of the legality of the question.

However, the second question is from a health physics viewpoint, you would not want anybody offsite or onsite to receive a dose of 300 grams through the thyroid or 25 grams through the whole body.

Q. Certainly. All right.

Let's get back now to the earlier consideration of how you approached the unfiltered bypass leakage values.

You have listed in the earlier affidavit a list of leakage paths that are candidates for consideration in terms of permitting bypass leakage and other parts of the structure that are considered barriers to bypass leakage.

Okay. Let's assume the barriers remain barriers for now and look only at those paths that are candidates for leakage.

In the affidavit, starting at page 4 and continuing on through the balance of the text of the affidavit is a section labeled, "Tests and Inspections."

Now, without going into detail, I read that Section IV as saying here's the kinds of tests and test procedures that are going to be undertaken in the future to determine just what really is the leakage rate from

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these various paths that this listing has indicated are potentially available for leakage; is that correct?

BY WITNESS MARTIN:

A. Yes, that's correct.

Q. Approximately on what calendar schedule would you anticipate that these tests will be undertaken?

BY WITNESS MALEC:

A. May I respond to that, Your Honor?

Q. Incidentally, my questions go to either or both of you, so I don't....

BY WITNESS MARTIN:

A. Mr. Malec can handle that.

Q. Fine.

BY WITNESS MALEC:

A. It varies by the type of test. Generally, they are specified in 10 CFR 50, Appendix J. Type A test has one interval, which I believe is three times in ten years.

Q. Oh, excuse me, sir. I am getting at something just ahead of that.

What I'm really asking about is in terms of what will be tested and when it will be tested in order to establish the readiness of Allens Creek to operate?

BY WITNESS MARTIN:

A. I think you mean the preoperational testing

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phase; is that correct?

Q. Say again, please?

BY WITNESS MARTIN:

A. Preoperational testing phase?

Q. All right. So these kinds of things will be done during this preoperational testing phase when the plant is built, assembled and pretty much everything is in place; is that correct?

BY WITNESS MARTIN:

A. Yes, that's correct.

Q. So since neither you nor we know whether the plant is going to be built, or when if it is, you wouldn't be able to give me a calendar schedule at this time, I gather?

BY WITNESS MARTIN:

A. I don't think there is one in existence right now.

Q. Okay. But in essence, the situation that the Applicant is faced with is there comes a time to do these tests and assess the results and start adding up the leakage rates that are associated with each of these identified leakage paths.

Is it not possible that these measured leakage rates will add up to something greater than the four percent of the half percent?

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BY WITNESS MARTIN:

A. Let's just take the following case, just a hypothetical case, for a second.

It's possible, yes, that in the first cut of testing that the rate may be exceeded, just as it's also possible it may be below the technical specification limit.

However, I just want to point out that there is more than one objective for performing those tests. Primarily it's to verify or to measure the integrity of all these areas which are being tested; and whenever a leak is detected, whether in a valve or weld seam, these leaks will be corrected.

So ideally, after the testing regime is over, you will have yourself a, quote, ideal type of containment, which if all is done correctly, you should not have any further leaks.

In the extreme cases you will be at best below the specified technical specifications. In this case you will be below the four percent number.

Q. Okay. The kind of thing I'm concerned about, and I realize these ultimate answers have to wait till the pre-op testing phase, the kind of thing I'm concerned about is what's the likelihood that during this testing phase you will arrive at some measured leakage rate values

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that in essence say, "My gosh, we've got to rip out a tremendous amount of piping here and redo it because the leakage is unacceptable."

In other words, have you had experience that would indicate that if the leakages are not quite what's expected, in the sense of being greater than (certainly not less than), that there are practical remedial measures that can be taken to pull these leakage rates down without tearing down the building and starting over again or something?

BY WITNESS MARTIN:

A. Let me just offer you a two-phase response, with the first phase I will respond to and the second half I will defer to Mr. Malec.

Concerning the practical experience, I do not have any practical experience in the actual performing of tests. However, I do make it a point to get the test results from Ebasco's Plant Operation and Betterment Department whenever they go out and perform such tests.

The last one I have seen was the one performed at the St. Lucy Unit I facility down in Florida, which is, incidentally, a pressurized water reactor.

That plant happens to have a 24 percent bypass leakage fraction technical specification and --

Q. Excuse me. Does that 24 percent correspond to

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th four percent that we are talking about?

BY WITNESS MARTIN:

A. It's comparable to our four percent.

Q. Analogous to it, okay.

BY WITNESS MARTIN:

A. Yes. And the measured fraction after tests was ten percent.

Q. Say again, what was ten percent?

BY WITNESS MARTIN:

A. The measured number was 10 percent, so we are talking about a factor of two below the technical specifications.

Q. You are saying the 24 percent would have been allowable, but 10 percent was actually measured?

BY WITNESS MARTIN:

A. Was measured.

Q. I see. So what you are saying is there is some experience to indicate that things can come out all right?

BY WITNESS MARTIN:

A. Yes.

Q. Mr. Malec, did you have something to add here?

BY WITNESS MALEC:

A. Yes, Your Honor.

We take several steps in the specification of,

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for instance, the containment isolation valves, penetration assemblies, and so forth, that those components which may in fact contribute to any bypass leakage, and we take steps in the shop, and we specify to the manufacturer, actually have him prove that certain leakage rates will be met, and these are typically very, very low.

We are also looking at industry improvements in things like seating ability of valves. Check valves in particular have been a source of leakage in the past, and we are looking at some of the things that the vendors are coming out with to cut down on the potential bypass leakage paths.

We will go into the shop for all our containment isolation valves and we will see that they are hydro-tested and leak rate tested, and our specifications are typically well below that which would normally be allowable under, for instance, ASME Code, Section 11, Subsection IWV 3,000, where leak rates for different valve types are specified.

Q Have you, sir, had experience with -- well, let's drop the word experience.

Have you had an opportunity to review historically how well manufacturers have been able to meet these specifications.

I'm saying only it's one thing to put nice

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sounding specs on it and say, "We don't want it to be any worse than this," but it's another thing if those specs are not realistic within the state of technology.

So do you have some historical performance to look at here to assure you that the specs are realistic?

BY WITNESS MALEC:

A. I do not have it at my fingertips. One of the engineers who works for me has had extensive valve experience, and based on his discussions with the various valve vendors, and he is on one of the industry subcommittees for this type of thing, indicated to me last week before I left that there would not be a problem with the numbers we have specified in the specifications.

Q. So this is the kind of thing your organization considers independent of whether you have gotten involved in it or not?

BY WITNESS MALEC:

A. Yes, sir. We recognize that the valve will not always perform at a given rate after it's been in service.

By starting it out low, it gives us additional margin to allow for wear and so forth.

We also specify things like double packings, leak off connections, those types of features that will assist the Applicant in holding any bypass leakage down

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to a bare minimum.

JUDGE LINENBERGER: Thank you, gentlemen. I believe that's all.

JUDGE WOLFE: Cross, Mr. Dewey?

MR. DEWEY: No, sir.

JUDGE WOLFE: Mr. Doherty?

MR. DOHERTY: Yes, Your Honor.

RECROSS-EXAMINATION

BY MR. DOHERTY:

Q. A question of you, Mr. Martin, you mentioned the St. Lucy nuclear plant in Florida as being familiar to you.

Do you know if this reactor has an equal number of penetrations of containment as the Allens Creek?

BY WITNESS MARTIN:

A. No, I do not.

Q. Do you know if this plant has the same ratio one to another of the eight types of barrier penetrations which you put on page 2 of the affidavit -- or that was placed on page 2 of the affidavit?

BY WITNESS MARTIN:

A. The ratio of what now? I'm sorry, I didn't understand.

Q. Of one to another.

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BY WITNESS MARTIN:

A No, I do not.

MR. COPELAND: Excuse me. Are you asking, Mr. Doherty, if they are the same types of leakage paths?

MR. DOHERTY: No, same ratios of numbers one to another.

WITNESS MARTIN: I do not know if it's the same ratio.

MR. DOHERTY: Thank you. No further questions.

JUDGE WOLFE: Redirect, Mr. Copeland?

MR. COPELAND: No, sir.

JUDGE WOLFE: Now, with respect to the witnesses, they are to return?

MR. COPELAND: Mr. Malec is to stay, Mr. Martin is to stay, and we need one more gentleman, Mr. Chiou.

JUDGE WOLFE: We will recess now until about 4:20.

(Recess taken.)

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MR. DEWEY: Staff's next witness is Mel Fields.

Whereupon,

MEL B. FIELDS

was recalled as a witness and, having been previously duly sworn, was examined and testified as follows:

MR. DEWEY: He's going to testify regarding bypass leakage.

Mr. Fields is still under oath, I believe.

JUDGE WOLFE: Yes.

DIRECT EXAMINATION

BY MR. DEWEY:

Q. Mr. Fie'ds, do you have a document before you entitled "NRC Staff Testimony of Mel B. Fields Relative to Bypass Leakage"?

A. Yes, I do.

Q. This is a two-page document; is that correct?

A. That is correct.

Q. At this time do you have any changes to make in your testimony?

A. Yes, there's one change. On page 2, approximately the middle of the page, the beginning of the answer to the question on reasonableness of the bypass leakage, the sentence begins: "Yes. The amount of unfiltered leakage assumed for ACNGS (0.095%)" -- that

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should be 0.0195.

Q Is this your only change?

A. It is.

Q With this change, do you verify that the facts contained therein are true and correct to the best of your knowledge?

A. They are.

MR. DEWEY: Your Honor, at this time I request that the testimony of Mel Fields regarding McCorkle Contention 17 be admitted into evidence in this proceeding.

JUDGE WOLFE: Any objection?

MR. COPELAND: No objection.

MR. DOHERTY: No objection, Your Honor.

JUDGE WOLFE: All right. The testimony of Mr. Fields relating to McCorkle Contention 17 is incorporated into the record as if read.

(NRC Staff's Testimony of Mel B. Fields on McCorkle Contention 17 follows.)

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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)	)	

NRC STAFF TESTIMONY OF MEL B. FIELDS
RELATIVE TO BYPASS LEAKAGE

[McCorkle Contention 17]

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

A. My name is Mel B. Fields. I am employed at the U. S. Nuclear Regulatory Commission as a Containment Systems Engineer in the Containment Systems Branch. I have testified previously in this hearing on Board Question 4B, Compliance with GDC 50; Board Question 9, Bypass Leakage; and Board Question 4A, Combustible Gas Control.

Q. What is the purpose of this testimony?

A. The purpose of this testimony is to respond to McCorkle Contention 17.

Q. What does McCorkle Contention 17 allege?

A. McCorkle Contention 17 states as follows:

> The containment as *designed* will allow excess leakage to bypass the filtration systems. The power company admits that 20 percent of the leakage would not even be filtered.

Q. Has the staff reviewed the amount of containment leakage that will bypass the filtration systems?

A. Yes. The staff has reviewed the penetrations and lines penetrating the ACNGS containment for the potential of having bypass leak paths against the criteria set forth in Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants," which is part of NUREG-0800, (Standard Review Plan) Section 6.2.3, "Secondary Containment Functional Design." The criteria used by the applicant to classify the lines into potential and nonpotential bypass leak paths follow our guidelines and are acceptable.

Q. Is the amount of bypass leakage assumed by the applicant reasonable from a hardware performance capability standpoint?

A. Yes. The amount of unfiltered leakage assumed for ACNGS (0.~~095%~~ .0195% of containment volume per day) is approximately 4% of the total leakage allowed (0.5% of containment volume per day). This kind of percentage ratio between total leakage and bypass leakage is reasonable for dual containments, and has been shown to be achievable in the periodic leak tests of operating plants.

Q. What measures will be taken to assure that containment leakage at ACNGS will not exceed the technical specification limits for both total leakage and unfiltered leakage?

A. Appendix J of 10 CFR Part 50 requires extensive pre-operational leak tests and periodic leak tests during the life of the plant to assure that the containment will maintain its expected level of leak-tightness. Type A leak tests (total containment leakage) will be performed three times during each 10 year service period while Types B and C leak tests (for containment penetrations), which will provide a measure of expected unfiltered leakage, will be performed at intervals not to exceed 2 years of duration.

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

JUDGE WOLFE: Is there cross, Mr. Copeland?

MR. COPELAND: No, sir.

JUDGE WOLFE: Mr. Doherty?

MR. DOHERTY: Yes, sir.

CROSS-EXAMINATION

BY MR. DOHERTY:

Q Were you present for the cross-examination and Board questions of the witnesses for the Applicant, Mr. Malec and Mr. Martin?

A Yes, I was.

Q Was there anything you heard from them that you disagreed with?

A I wouldn't say disagreed. There's one point that may provide some clarification to the parties involved here.

And that was with regard to the potential bypass leak paths insofar as whether or not welded joints are potential bypass leak paths.

As it states in our Branch Technical Position CSB 6-3, the Applicant has the choice of either identifying it and assigning a leakage rate to that welded joint, or during the Type A test, perform a soap bubble test, and if there is any detectable leakage, to correct it to

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zero.

So the discussion that was held earlier on why this penetration was not assumed to be a potential bypass is resolved because the Applicant has committed itself to show us that there is zero bypass leakage through this penetration.

Q Do you -- at this point are you saying that you've observed in the PSAR a commitment?

A. They have committed to the Branch Technical Position, which directly infers this type of commitment

Q Okay.

MR. DOHERTY: No further questions, Your Honor.

JUDGE WOLFE: Redirect, Mr. Dewey?

MR. DEWEY: No, sir.

JUDGE WOLFE: Board questions?

BOARD EXAMINATION

BY JUDGE LINENBERGER:

Q Mr. Fields, has the Staff -- Are you aware of the extent to which the Staff has assessed historically the ability of operating plants to achieve the level of unfiltered bypass leakage that the licensees for those plants had estimated prior to completion of the plants they would be able to achieve?

In other words, has the Staff reviewed

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pre-operational projections of Applicant's against post-operational or post-startup actual achievements with respect to unfiltered bypass leakage?

A. Yes, we have.

Q. And can you comment on what the results of that comparison have indicated?

A. In general, the predictions were conservative with respect to what was measured. There may be cases in which the actual measured leakage was greater than what was predicted.

But there has not been a case, to my knowledge, where the leakage could not be corrected with minor changes in the containment design to below what the requirement was.

Q. So only minor remedial matters, to your knowledge, have ever been required; no substantive, quite extensive backfitting kinds of things have been required to bring the plants into conformance?

A. That is correct.

Q. All right, sir.

JUDGE LINENBERGER: That really is the only question I have.

JUDGE WOLFE: Mr. Copeland, cross?

MR. COPELAND: No, sir.

JUDGE WOLFE: Mr. Doherty?

MR. DOHERTY: No, Your Honor.

JUDGE W -FE: Is the witness --

MR. DEWEY: The witness is to be permanently excused.

JUDGE WOLFE: All right. The witness is permanently excused.

(The witness was permanently excused.)

MR. CULP: Mr. Chairman, at this time the Applicant calls Chiou, Malec and Martin to testify on Doherty Contention 11.

To your left is Mr. Martin; Mr. Malec is in the middle; and Mr. Chiou is on the right.

JUDGE WOLFE: Mr. Malec, you have been sworn already today. You are still under oath.

You are still under oath, Mr. Martin.

And, Dr. Chiou, if you would remain standing and raise your right hand.

Whereupon,

WALTER F. MALEC

-and-

GUY MARTIN, JR.

were recalled as witnesses and, having been previously duly sworn, were examined and testified as follows,

and

CHUNG-YI CHIOU

was called as a witness and, having been first duly sworn, was examined and testified as follows:

DIRECT EXAMINATION

BY MR. CULP:

Q Gentlemen, do each of you have before you a document entitled "Direct Testimony of Chung-Yi Chiou, Walter F. Malec and Guy Martin, Jr. Regarding Doherty Contention 11 - Fuel Pool Accident," which consists of a document of six pages, plus a statement of each of your professional qualifications?

BY WITNESS MARTIN:

A Yes, I do.

BY WITNESS MALEC:

A Yes, I do.

BY WITNESS CHIOU:

A Yes, I do.

Q Mr. Martin, beginning with you, did you prepare this document; or was it prepared under your supervision -- those aspects of the testimony which are identified as your testimony?

BY WITNESS MARTIN:

A I prepared those portions.

Q Do you have any corrections or additions to make to the testimony?

BY WITNESS MARTIN:

A. No, I do not.

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

BY WITNESS MARTIN:

A. Yes, it is.

Q. Do you adopt this as your testimony in this proceeding?

BY WITNESS MARTIN:

A. Yes, I do. I just want to point out, however, that my personal qualifications were not included in here, although they have been included as part of the record on Doherty No. 40.

Q. All right.

Mr. Malec, was the testimony which has been identified in the testimony as yours, was that prepared by you or under your supervision?

BY WITNESS MALEC:

A. It was.

Q. Do you have any corrections or additions to make to that testimony?

BY WITNESS MALEC:

A. I do not.

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

BY WITNESS MALEC:

A. It is.

Q. Do you adopt it as your testimony in this proceeding?

BY WITNESS MALEC:

A. I do.

Q. Dr. Chiou, did you prepare this testimony; or was it prepared under your supervision?

BY WITNESS CHIOU:

A. I prepared it.

Q. Do you have any corrections or additions to make to the testimony?

A. Yes, I do.

Q. Would you give those to us, please?

BY WITNESS CHIOU:

A. On page 2, the ninth line, delete the word "pit." Add "structure with stainless steel liner."

MR. DOHERTY: I'm sorry. I missed what line. And would you repeat it? It's my error. I just missed what you said.

WITNESS CHIOU: Delete the word, "pit."

MR. DOHERTY: What line on --

WITNESS CHIOU: Add "structure with stainless steel liner."

MR. CULP: Mr. Doherty, that's on Line 8 1/2 on

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page 2.

MR. DOHERTY: All right. And it says, counsel, what does it say? "The floor of this pool consists of" --

MR. CULP: The sentence begins: " ... Building is a reinforced concrete pit." I believe the witness says strike the word, "pit," and insert "structure with stainless steel liner."

JUDGE WOLFE: "With stainless steel" what?

MR. CULP: "Liner."

BY MR. CULP:

Q Do you have any other corrections or additions?

BY WITNESS CHIOU:

A No.

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

BY WITNESS CHIOU:

A Yes.

Q Do you adopt it as your testimony in this proceeding?

BY WITNESS CHIOU:

A Yes.

MR. CULP: Mr. Chairman, at this time I move that the testimony identified by these three witnesses be

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incorporated into the record as if read.

JUDGE WOLFE: Inclusive of the attachments?

MR. CULP: Yes.

JUDGE WOLFE: Any objection?

MR. DEWEY: No, Your Honor.

MR. DOHERTY: Your Honor, I would like to take -- I guess Mr. Malec and Dr. Chiou on voir dire.

I'll start with Mr. Malec, just a few questions.

- - -

VOIR DIRE

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BY MR. DOHERTY:

Q Do you have -- in your supervising engineer position, Mr. Malec, do you have responsibility for stress analysis of the spent fuel pool?

BY WITNESS MALEC:

A. No.

Q Have you ever done stress analysis on a spent fuel pool?

BY WITNESS MALEC:

A. No, I have not.

Q You list here plumbing in your current administrative experience. Would I infer that that means water piping, as opposed to steam piping? Would that be fair?

BY WITNESS MALEC:

A. No, a better characterization would be typically drainage, both in terms of drainage to the sumps, soil lines, non-radioactive sump pumps, toilet facilities.

Q Uh-huh.

BY WITNESS MALEC:

A. In addition to the pool liner leak detection system at the exit from the liner welds.

Perhaps typically you could say these are non-pressurized systems.

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Q Well, does any of your training involve concrete construction?

BY WITNESS MALEC:

A. I have had perhaps one course in construction.

Q In designing hangers and supports and working with engineers who are working on that, would that include anchoring in concrete, that type of work, that type of evaluation?

BY WITNESS MALEC:

A. It's loosely included, yes. I have administrative responsibility for that area. There is a technical supervisor who looks over that work.

Q Okay.

BY WITNESS MALEC:

A. The anchoring of the pool liner is not within the mechanical scope, I don't deal ---

Q I'm sorry. I couldn't hear what you said.

BY WITNESS MALEC:

A. The anchoring of a pool liner is not within the mechanical scope in Ebasco Services, I don't deal with that.

Q Dr. Chiou, in your professional qualifications, you say you develop design criteria for stainless steel pool liners and impulsive/impactive analyses.

By that do you mean an impulsive/impactive analysis on the stainless steel pool liners?

BY WITNESS CHIOU:

A. Not only for stainless steel liners. It also includes buildings and building components.

Q. Uh-huh.

MR. DOHERTY: No further questions, Your Honor, and I have no objections.

JUDGE WOLFE: Absent objections, the testimony of Dr. Chiou and Messrs. Malec and Martin regarding Doherty Contention 11, inclusive of Dr. Chiou's professional qualifications and Mr. Malec's professional qualifications, are incorporated into the record as if read.

(Applicant's Testimony of Chung-Yi Chiou, Walter F. Malec and Guy Martin, Jr. on Doherty Contention 11 follows.)

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 CHUNG-YI CHIOU,
WALTER F. MALEC AND GUY MARTIN, JR.
REGARDING DOHERTY CONTENTION 11 -
FUEL POOL ACCIDENT

Q. Please state your names and positions and describe your educational and professional backgrounds.

A. My name is Chung-Yi Chiou. I am employed by Ebasco Services Inc. My business address is 160 Chubb Avenue, Lyndhurst, N.J. I am in charge of the design of the stainless steel pool liner for ACNGS. My educational and professional background is described in Attachment CYC-1.

My name is Walter F. Malec. My business address is 160 Chubb Avenue, Lyndhurst, N. J. I am the Supervising Mechanical Nuclear Engineer for the Allens Creek Project employed by Ebasco Services Incorporated. The statement of my background and qualifications is attached as Exhibit WFM-1 to this testimony.

My name is Guy Martin, Jr. and my business address is Ebasco Services, Inc., 2 World Trade Center, New York, N.Y.

I have previously discussed my position and background in connection with my testimony on Doherty Contention 40.

Q. Mr. Chiou, in his answer opposing the NRC's Staff Motion for summary disposition of this contention, Mr. Doherty alleges that the spent fuel pool could be breached by the dropping of a fuel assembly onto the floor. Is this possible?

A. No. The spent fuel pool in the Fuel Handling Building is a reinforced concrete structure with stainless steel liner. The floor of this pool consists of a 1/4" thick stainless steel liner and a six-foot thick concrete slab beneath the liner. A preliminary impact analysis has been performed on a spent fuel bundle assembly dropping onto the spent pool floor which indicates that the liner will not be penetrated. This analysis utilized conservative assumptions such as:

1. The fuel bundle assembly will strike perpendicular to the floor with fuel channel not removed.

2. No credit is taken for the concrete floor providing support for the fuel pool liner.

The Spent Fuel Pool including the pool floor is being designed as a Seismic Category I structure and will maintain its structural integrity for a fuel bundle assembly drop accident.

Q. Mr. Martin, supposing that a fuel bundle drop did penetrate the pool liner, would such an accident present unacceptable radiological consequences?

A. In the highly unlikely event that a spent fuel assembly accidentally drops to the spent fuel pool floor and penetrates the stainless steel liner, it may be assumed that some of the rods in the assembly will be damaged. In the evaluation of the offsite radiological consequences, two potential pathways for radioactivity releases were considered: 1) the spent fuel pool water escaping through the punctured liner and, 2) the unmixed gaseous fission products released from the pool surface.

The release to the environment of radioactivity, assumed to have mixed in the spent fuel pool water escaping through the damaged liner is improbable. Water leakages, due to liner plate weld damage will be collected by, and then routed to, the low purity system of the radioactive waste treatment system via the Pool Liner Leak Detection System. If the pool liner were damaged in an area not serviced by the Pool Liner Leak Detection System, it is reasonable to assume that the six foot thick concrete slab beneath the spent fuel pool would limit potential leakage. Consequently, no radioactivity is expected to escape to the environment via the liquid pathway.

The only possible pathway for releases of radioactivity to the environment is by the exhaust of gaseous fission products airborne in the Fuel Handling Building. The analysis of the consequences of such releases has been performed and is presented in Chapter 15 of the ACNGS Preliminary Safety Analysis Report (PSAR). The assumption that 98 rods of the fuel assembly having the highest fission product activity fail, makes the results of the accident postulated in the PSAR more conservative than those which would be obtained for only one assembly hitting the spent fuel pool floor. This conclusion is substantiated since one fuel assembly consists of only 62 fuel rods. Assuming that all the rods in one spent fuel assembly were to fail upon impacting the spent fuel pool floor, the radiological doses at the ACNGS exclusion area boundary have been calculated to be:

Whole body:	0.4 Rem
Thyroid:	0.3 Rem

These doses are a small fraction of the 10 CFR Part 100 dose limits and are well within the NRC guidelines for a fuel handling accident.

Q. Mr. Malec, given the accident hypothesized above has the Applicant considered the possible loss of cooling water from the spent fuel pool?

A. Yes. The loss of water inventory due to the postulated breach in the Spent Fuel Pool (SFP) boundary has been considered. Preliminary calculations indicate that the loss of water inventory from the SFP through the normally closed leakage detection system valve (which was also postulated to have been left open) as a result of a SFP boundary breach at a weld would be limited to less than 39 gpm. Hence, sufficient makeup capacity is available to maintain water coverage over the fuel.

Q. Does this loss of pool water inventory present a safety problem?

A. No. The spent fuel pool cooling water system has both safety grade and non-safety grade sources of make-up water. The make-up capability of the Safety Class 3, Essential Service Cooling Water System (ESCWS) will provide adequate water to the Spent Fuel Pool during a breach of the pool boundary. The ESCWS can provide approximately 100 gpm of makeup water from either safety train. The non-safety related Demineralized Water System can provide an additional 50 gpm of makeup water.

Q. Mr. Martin, are you aware of any fuel handling accident involving the actual dropping of a fuel bundle in the fuel pool at an operating nuclear power plant?

A. Yes. An incident of this type had occurred at Millstone Unit 1 in September of 1974.

Q. What were the consequences of the accident on the fuel pool and the fuel bundle?

A. It was reported that no visual damage was noted in the fuel pool and that no fuel pool leakage was observed. The fuel bundle was bowed considerably, but continued air samples indicated no release of activity. Activity analysis of the fuel pool water remained constant and showed no signs indicative of fuel pin failure.

Q. Gentlemen, what are your conclusions regarding the subject fuel pool accident?

A. The spent fuel pool boundary will maintain its structural integrity following a postulated fuel bundle drop impacting the floor. Even if pool leakage is assumed due to liner breach, radiological considerations are bounded by the PSAR Chapter 15 accident analysis.

Attachment CYC-1

CHUNG-YI CHIOU

SUMMARY OF EXPERIENCE (Since 1965)

Total Experience - Sixteen years experience in Civil Engineering including project team work for fossil and nuclear power plants; and research and engineering with emphasis on missile dynamic analysis, seismic analysis and design, soil-structure interaction analysis, earthquake engineering, stress analysis by NASTRAN and ANSYS; and site characteristics studies.

Professional Affiliations - Registered Professional Engineer in New York
American Society of Civil Engineers (ASCE).

Publications - "Structural Optimization by Methods of Centers", Ph. D Thesis - University of Illinois at Urbana - 1975
"A 3-D Solid Finite Element for Heterogeneous Materials", Seventh Symposium on Engineering Problems of Fusion Research - 1977

Education - BSCE - Chen-Kung University - Taiwan - 1965
MSCE - University of Illinois - 1970
Ph.D - Civil Engineering - University of Illinois - 1975

EBASCO EXPERIENCE (8 years - since 1973)

Principal Civil Engineer Civil Design Engineering Department

Allens Creek NGS - Unit No. 1 (BWR) - Responsibilities include the development of design criteria for Stainless Steel Pool Liners and Impulsive/Impactive Analyses; building design specification for Fuel Handling and Reactor Auxiliary Buildings; procurement specification and bid evaluation for FHB Bridge Crane, FHB Gantry Crane, Drywell Closure Head, Drywell Personnel Lock/Equipment Hatch and SST Pool Liners; design review and design engineering for FHB, RAB, and Drywell.

Lovett Units 4 & 5 Coal Reconversion - Prepared Civil specifications for concrete, reinforcing and structural steel; prepared the prefabricated metal building specification and reviewed architectural specifications; and contributed to the development of design criteria for Warehouse, Office and Lab Building on pile foundation.

Senior Civil Engineer - Consulting Civil Engineering Dept.

Engaged in the determination of safe shutdown earthquake using deterministic and/or probabilistic (risk analysis) approaches, generation of design response spectra by site independent approach or site-response analysis method, simulation of SSE accelerograms and adjustment of earthquake time-history so that computed response spectra envelope the design response spectra, study of lavered media to dynamic loadings, deconvolution analysis to define earthquake motion below surface, media-structure interaction analysis to generate floor response spectra, finite element seismic stability analysis of Class I dykes, and structural analysis and member design of PWR primary prestressed concrete containment and secondary steel frame containment for nuclear projects.

Engaged in stress analysis and evaluation of nondestructive test results of hydraulic turbine spiral casing, stability analysis of arch and gravity dams, and dynamic analysis of equipment components and off-shore mooring facilities for hydro and fossil projects.

Engaged in the development of 3-D solid finite element (HEXNL) for hetero-geneous materials and implementing it to NASTRAN, design of test specimens for determination of material properties and evaluation of test results, 3-D stress analysis of TF coil using HEXNL element, and elasto-plastic residual stress analysis of coil using ANSYS for fusion test reactor.

PRIOR EXPERIENCE (8 years - 1965 to 1973)

University of Illinois; Research Assistant, Civil Engineering Department (5 years)

Engaged in research related to the adjustment of earthquake acceleration records, design response spectra, seismic design of tall building and structural optimization. Research assistant in Aeronautical and Aerospace Engineering Department (1/2 year). Participated in research projects on the stability of stochastic processes and stresses due to rardom moving loads.

PRIOR EXPERIENCE (8 years)

China Airlines, Taipei, Taiwan; Airport Officer (2 years)
Responsible for works related to international flights, including aircraft load balancing.

Chinese Navy, Taiwan; Ensign, Engineering Officer (1 year)
Responsible for field inspection of construction, redesign structural columns, shear walls, and roof trusses of storm-damaged military facilities and preparation of drawings for contract bidding.

WALTER F. MALEC

Born Philadelphia, Pennsylvania

Education Polytechnic Institute of Technology, degree of Engineer in Nuclear Engineering - 1978
Massachusetts Institute of Technology, MS in Nuclear Engineering - 1970
U.S. Coast Guard Academy, BS - 1968

Member American Nuclear Society

Licensed Registered Professional Engineer in the State of New York (No. 56673) and the State of Texas (No. 48430)

Experience:

1980 Ebasco Services Incorporated, Lyndhurst, N.J. office; Supervising Engineer, Mechanical Engineering Department:

Houston Lighting & Power Co. - Allens Creek NGS - Unit No. 1 -1 200MW(e) BWR

Technical and administrative responsibility for mechanical, fire protection, plumbing, HVAC, stress analysis, hangers and supports, water treatment and in-service inspection activities. Includes schedules, budgets, and client relations.

1978-1980 Ebasco Services Incorporated, Lyndhurst, N.J. office; Principal Engineer, Mechanical-Nuclear Engineering Department

Houston Lighting & Power Co. - Allens Creek NGS - Unit No. 1 - 1200 MW(e) BWR, Lead NSSS Engineer

Responsible for preparation and maintenance of ECCS and BOP flow diagrams, piping layouts, system design descriptions, in-service inspection provisions, Nuclear Island building general arrangements, PSAR and FSAR preparation, equipment sizing and specification, NSSS vendor interface for correspondence, drawing review, and contract administration.

1976-1978 Ebasco Services Incorporated, New York office; Senior Engineer, Mechanical-Nuclear Engineering Department including:

Houston Lighting & Power Co. - Allens Creek NGS - Unit No. 1 - 1200 MW(e) BWR, Lead NSSS Engineer

Louisiana Power & Light Co. - Waterford SES Unit No. 3 - 1165 MW(e) PWR. Lead NSSS Engineer.

Same responsibilities as listed for 1978-1980 above.

1974-1976 United States Coast Guard, Marine Inspection Office, New York; Lieutenant - Supervisory Boiler Inspector. Responsibility for supervision, assignment and training of Marine Inspectors in largest Marine Inspection Office in country. Inspection of hull and machinery material condition of U.S. flag and foreign merchant vessels, and pressure vessels under construction. Application of various laws and regulations of the United States, ASME Code, ANSI, TEAM, NEC and NFPA Standards. Review of engineering plans and alterations, reports from field and resident inspectors.

1973-1974 United States Coast Guard, USCGC Spencer (WHEC-36), Lieutenant - Chief Engineer. Responsibility for operation, maintenance and repair of hull and engineering plant of 6200 slip twinscrew steamship. Direct supervision of 40 officers and men. Duties included preparation of repair specifications and maintenance of vessel records. Received Coast Guard Achievement Medal for superior performance of duty.

1970-1973 United States Coast Guard, Marine Inspection Office, New York, Lt. and Ltjg - Marine Inspector. Inspection of hull and machinery of U.S. and foreign flag merchant vessels.

1968-1969 United States Coast Guard, USCGC Mellon (WHEC-717), Ensign, Assistant Engineer Officer.

MR. CULP: The witnesses are available for cross-examination at this time.

JUDGE WOLFE: Mr. Dewey.

MR. DEWEY: The Staff has no questions.

JUDGE WOLFE: Mr. Doherty.

MR. DOHERTY: Yes, Your Honor.

CROSS-EXAMINATION

BY MR. DOHERTY:

Q Well, first of all, the spent fuel pool, is that in the spent fuel building; or is that in both the spent fuel building and the containment building?

BY WITNESS MALEC:

A The spent fuel pool is in the fuel handling building.

Q It's in the fuel handling building. Okay.

Now, if a fuel assembly is dropped, is it conceivable to you that the pool would be empty when that event occurred, have no water in it, let's put it that way?

BY WITNESS MALEC:

A In my opinion?

Q Uh-huh.

BY WITNESS MALEC:

A No.

Q All right. If, indeed, this event occurred

10-16

then and the spent assembly hit the water and proceeded on down with gravity -- gravitational force, would there be any way it could hit the floor without hitting some type of racking, some parts of the racks?

BY WITNESS MALEC:

A. Excuse me, Mr. Doherty, do I understand you to say that the spent fuel element was out of the water initially?

Q. No. Let's assume that the spent fuel is being moved. All right. And it's dropped -- Okay, let's get it clear.

From your question --

BY WITNESS MALEC:

A. I inferred that from when you said it hit the water, thereby being out of the water initially.

Q. Yes, that's what I meant. Perhaps you could explain what you have in your mind.

BY WITNESS MALEC:

A. I don't have anything in mind. I'm trying to understand what you'd like to know.

Q. Okay. Let me ask a question then: In re-fueling is the fuel assembly ev[illegible] lifted out of the water?

BY WITNESS MALEC:

A. No, sir.

10-17

Q Uh-huh. Okay. So it's always -- So we're talking about a drop initiated under some water --

BY WITNESS MALEC:

A That's correct.

Q -- continuing -- and occurring entirely under water?

BY WITNESS MALEC:

A Yes, sir.

Q All right. Now, the question I was going to -- and I appreciate your interjecting that, Mr. Malec -- is it possible for an assembly to hit the floor, or will it have to strike some part of a structure in the pool?

BY WITNESS MALEC:

A It's theoretically possible it could go directly to a floor, yes.

Q To hit the naked floor?

BY WITNESS MALEC:

A I'm sorry? To --

Q To hit the --

BY WITNESS MALEC:

A The bottom of the fuel pool?

Q Well, I think we've said it. The word "naked" upsets you, I guess.

BY WITNESS MALEC:

A I can't understand what you said.

10-18

Q All right.

BY WITNESS MALEC:

A The what floor? The naked floor? Was that your word?

Q Yes. That is --

JUDGE WOLFE: Without hitting any obstacles, Mr. Malec.

WITNESS MALEC: I just didn't hear the word "naked."

It is possible for the spent fuel element to go directly to the pool floor.

BY MR. DOHERTY:

Q Are there, as part of the fuel racks, any sort of ... oh, little -- I'd say small, anyway -- pieces or guides on the floor itself, to guide the -- to hold the fuel at the bottom?

BY WITNESS MALEC:

A The racks do not -- the pool floor does not have guides for the fuel on it.

Q So it's a smooth flat surface, nothing on it?

BY WITNESS MALEC:

A In general, I would say yes.

Q Okay. Now when the fuel is put in the -- is removed from the containment building and then moved to the spent fuel building -- that is, moved, is it at its

10-19

highest activity when it's at the spent fuel building?

BY WITNESS MALEC:

A No.

Q Can you give us an idea of how much less active it is by the time it gets to the spent fuel building?

MR. CULP: Your Honor, m going to object to that question. I don't see the relevance of that to Mr. Doherty's contention.

JUDGE LINENBERGER: Before we address your objection, Mr. Culp, I'd like to ask Mr. Doherty to just repeat that question, because I'm not sure I heard all of the words.

I'm not asking you to rephrase, but just repeat, because of the acoustic problem here.

MR. DOHERTY: Well, the question was, as I remember it: I asked him if the -- I think I asked him how much less active, that is, how much less radioactive is the fuel once it gets to the spent fuel building, than it is -- for comparison, in the reactor building?

MR. CULP: And I object to that on the basis that I don't think that is relevant at all to his contention of the fuel bundle dropping into the bottom of the spent fuel pool.

MR. DOHERTY: Well, the contention deals with

10-20

the consequences. This is as reworded on page -- well, of the Board's Order of September 1 -- under that contention.

"As Mr. Doherty points out, neither the SER nor the FS/FES describe the consequences of a spent fuel assembly drop into the spent fuel pool floor."

And the consequences would include possible release of radioactivity; therefore, it's relevant to ask how radioactive is this material.

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10-21

JUDGE WOLFE: Well, what's the relevancy -- it's either more or less active. We're only interested -- or you should only be interested in its activity at the time that it's in the -- it falls to the floor -- to the liner. So I'll sustain the objection.

MR. DOHERTY: Well, Your Honor, the question was: When it's moved to the spent fuel building -- the assumption being that will be its last movement -- and at that particular point, I think that's relevant. That's one time when it could well be dropped.

JUDGE WOLFE: Yes.

MR. DOHERTY: And I am presuming that the radioactivity of spent fuel decreases with time.

JUDGE WOLFE: Yes.

MR. DOHERTY: So its first arrival would be its highest radioactivity. That's why I think it's still relevant.

Otherwise ... the idea of making a comparison is fairly -- it was more of a suggestion of how to answer the question. But it might be more precise to just give the radioactivity itself in some way, in some measurement, rather than a comparison.

JUDGE WOLFE: Well, it's only relevant, as far as I can see, as to what its radioactivity is at the time it arrives at the spent fuel pool, period.

MR. DOHERTY: That was the question.

JUDGE WOLFE: No, you wanted to know ...

JUDGE LINENBERGER: That's why I asked you specifically to repeat the question, because I thought you were asking for a comparison --

MR. DOHERTY: I'm sorry --

JUDGE LINENBERGER: -- of the source term when the fuel element -- when the fuel bundle is in the reactor building, compared with the source term when the fuel bundle is over the pool available to be dropped.

And that is the question that was objected to, and --

MR. DOHERTY: That's the one you sustained.

JUDGE WOLFE: That's the one I sustained.

Now, proceed.

BY MR. DOHERTY:

Q. How radioactive, in any units you may choose, is the spent fuel when it arrives at the spent fuel building?

BY WITNESS MARTIN:

A. I could tell you that. A typical time -- rather a minimal time would be 24 hours of decay after the reactor has been shut down. So 24 hours of decay, then the time of decay that the fuel assembly would have undergone after removal from the reactor vessel.

10-23

Q How high above the pool floor is the spent fuel, when it's being shifted into the storage -- into the spent fuel pool for storage?

Perhaps the maximum height over the pool would be the most meaningful answer.

BY WITNESS MALEC:

A. This is not an exact number, but an approximation, I would say somewhere between 17 to 20 feet. I don't recall exactly. But 17 to 20 feet is an order of magnitude height.

Q And can you give an approximation of the weight of the spent fuel assembly?

BY WITNESS MALEC:

A. As I recall -- again, this is an order of magnitude -- it's about 700 pounds.

Q Does it include the fuel channel?

BY WITNESS MALEC:

A. The 700 pounds?

Q In that 700 pounds, is the --

BY WITNESS MALEC:

A. As I recall, it does, although I would have to refresh my memory on the exact weight.

Are you referring to the fuel when it's being shifted, or the weight I just gave you?

Q When it's being shifted, I think.

10-24

BY WITNESS MALEC:

A. It may or may not have the channel attached to it.

Q. Do you think the 700 pounds does not include a channel, which you gave?

BY WITNESS MALEC:

A. My recollection is that it does, although, again, I would have to refresh my memory to the exact weight.

Q. What are the approximate dimensions of a fuel assembly, the end particularly?

BY WITNESS MALEC:

A. The length?

Q. Right, the length, yes.

BY WITNESS MALEC:

A. About 12 feet.

Q. And what about that at end, from the end?

BY WITNESS MALEC:

A. As I recall, it's about 8 1/2 inches.

Q. Square?

BY WITNESS MALEC:

A. Approximately.

Q. And is that -- Does that assembly include a core support plate?

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10-25

BY WITNESS MALEC:

A. No, the core support plate remains in the reactor.

Q. So then is it a blunt-ended assembly at this point?

BY WITNESS MALEC:

A. No, there's a nose piece on the front -- on the bottom, I should say. At the core flow inlet portion.

Q. Okay.

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1-1

ed

BY MR. DOHERTY:

Q Is the perpendicular strike the greatest impact per unit area?

BY WITNESS CHIOU:

A To the structure damage, that's most conservative impact.

Q I notice at page 2 in the analysis you state, at line 17, "No credit is taken for the concrete floor providing support for the fuel pool liner."

So that in this analysis, is this an unsupported structure then that this impact is striking?

BY WITNESS CHIOU:

A The concrete floor usually has a certain flexibility. In my impact analysis I didn't take into account the flexibility of the concrete floor. It gives more conservative results.

Q Are you saying -- this concrete floor, is that something that normally sits under the steel liner? In other words, the steel is set on top of it? Is that the way the construction will be?

BY WITNESS CHIOU:

A The liner is on top of the concrete floor.

Q All right. Now, if that's taken away, won't that give the floor additional space to give on impact?

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1-2

BY WITNESS CHIOU:

A. The spent fuel pool floor is resting on the foundaticn. It is resting on the soil.

Q. Are you talking now about the actual design or the way the analysis has to be viewed now?

BY WITNESS CHIOU:

A. The physical condition of the spent fuel pool, that's what I'm talking about. I did not take into account the flexibility of the concrete floor in analysis, in order to give a more conservative estimation.

Q. Does your analysis assume any flexibility in the steel liner then?

BY WITNESS CHIOU:

A. Only at the welded seam; not at other areas.

Q. Is the floor, the stainless steel floor, a one-piece sheet?

BY WITNESS CHIOU:

A. The floor is made up of a lot of -- many pieces welded togeth

Q. So, then, there is some flexibility where those pieces join across the floor.

BY WITNESS CHIOU:

A. At the welding seam we had to provide the leak detection channels.

Q. Had to provide what? Excuse me.

1-3

BY WITNESS CHIOU:

A. Leak detection channel.

Q. Yes.

BY WITNESS CHIOU:

A. That's why the liner has certain flexibility at the welding seam only.

Q. And the welding seams, how many of those are there in the pool?

BY WITNESS CHIOU:

A. I do not recall.

Q. What is the approximate area of the pool?

BY WITNESS CHIOU:

A. It's about 40 times 40, I guess.

Q. All right. That's feet, I take it, 40 feet by 40 feet.

BY WITNESS MALEC:

A. That's approximately correct. The units are feet.

Q. You use the term "structural integrity" here at line 21. Does that mean no leakage paths are initiated?

BY WITNESS CHIOU:

A. The stainless steel core liner is designed as a big tight structure.

Q. Are you saying, then, it will remain as designed,

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1-4

then?

BY WITNESS CHIOU:

A. Yes.

Q. And that's what structural integrity means, then?

BY WITNESS CHIOU:

A. Yes.

Q. To your knowledge, has there been any probability calculations worked out of dropping the spent fuel pool from previous experience -- dropping a spent fuel assembly?

Does anyone have those kinds of figures among the panel?

BY WITNESS MARTIN:

A The answer to your question is yes. There have been probability calculations done for that.

However, I don't have the figures right now, but there have been published figures on these probabilities.

Q. I guess this question should go to you, Mr. Malec.

Do you know if the crane for moving spent fuel has been selected at this point, the design?

MR. CULP: Your Honor, I'm going bject to that. I think the testimony assumes that there is a drop within the fuel bundle, and I don't understand how

1-5

the crane that is moving the fuel bundle is relevant to the contention.

MR. DOHERTY: I think Counsel is correct. Thank you.

MR. CULP: You are welcome.

BY MR. DOHERTY:

Q Well, Mr. Martin, this seems to have come to you. On page 3 you state at line 16, "If the pool liner were damaged in an area not serviced by the Pool Liner Leak Detection System," then there's something more; but I don't understand.

What do you mean by not serviced by a leak detection system? Is the leak detection system only partial?

BY WITNESS MALEC:

A I'll respond in lieu of Mr. Martin. The leak detection channels run under the weld seams. They do not cover the entire bottom of the pool, only in those areas where there are welding seams.

Q So that, if I can get this straight, are you saying here, starting at line 12, "Water leakages, due to liner plate weld damage," is that assuming that the assembly was dropped at a seam, and you are covering two cases there, at a seam and not at a seam? Is that how that's to be looked at?

1-6

BY WITNESS MARTIN:

A. This paragraph was to assume that in the event that the water leakages bypass the leakage collection system, the paragraph was written for the purpose of assuming that wasn't the case, the presence of the six-foot thick spent fuel pool floor would prevent any water leakages or release of water to the environment.

Q. I had trouble hearing that. Did you say six-foot --

BY WITNESS MARTIN:

A. -- thick concrete floor.

Q. All right. So you do take credit for the concrete floor in your analysis here?

BY WITNESS MARTIN:

A. Yes, I did.

Q. At the top of 4 there's a reference to an assumption about 98 rods of the fuel assembly.

Now, since a fuel assembly has a smaller number than 98 fuel rods, I'm wondering how that number comes out or what led to the choice of that number, 98?

Would you stay a little bit away from the microphone, Mr. Martin. You have a very explosive speech. It tends to get lost in the static to me.

BY WITNESS MARTIN:

A. The number of 98 rods -- is that okay? Maybe

1-7

we should go through a testing phase, one, two, three.

The number of 98 rods comes from the impact analysis of a dropped spent fuel assembly, which is presented in the PSAR.

It basically assumes that if you have a dropped assembly, it goes through the calculations of the kinetic energies of a dropped fuel assembly, which would result in the failure of the rods in that complete assembly, then upon further impact of the neighboring assemblies, you would have further failures of fuel rods in the neighboring assemblies, based upon the residual kinetic energy of the dropped spent fuel assembly.

The total which has been calculated is 98 fuel assemblies.

Q. Is this analysis of a fuel assembly drop that strikes other fuel assemblies?

BY WITNESS MARTIN:

A. Yes, it is.

Q. I see.

BY WITNESS MARTIN:

A. It is in the PSAR, incidentally.

Q. Okay.

BY WITNESS MARTIN:

A. I beg your pardon. I think I said "98 assemblies." I meant 98 rods.

1-8

Q Yes. What would be the consequences if the fuel became uncovered ingross terms?

MR. CULP: Wait a minute.

I don't understand that question. I object to the question, that it is vague.

MR. DOHERTY: I will try to be a little more specific by asking him:

BY MR. DOHERTY:

Q What would be the consequences to the fuel cladding if this fuel became uncovered?

MR. CULP: I object to the question, now that he has rephrased it, on the grounds that it's not relevant to his contention.

MR. DOHERTY: There's a question asked by Counsel here, "Does this loss of pool water inventory present a safety problem?" They go into various systems that would stop that from happening.

I think they've introduced into the record statements with regard to uncovering the fuel or protecting themselves from uncovering the fuel, so it's relevant to talk about why are they doing this. What's this protection for?

That's at page 5, line 6.

(Bench conference.)

JUDGE LINENBERGER: Mr. Doherty, does your

1-9

question go to a complete loss of water from the fuel?

MR. DOHERTY: No, I didn't specify and that wasn't what I had in mind at the moment I asked it.

JUDGE LINENBERGER: It's a little difficult to know how to rule on the objection when you say that you don't know what you had in mind.

MR. DOHERTY: Well, one thing at a time. It certainly makes sense to ask about a complete loss.

In the end, the issue, though, is uncovered fuel. It will be uncovered, whether it's totally uncovered or half uncovered, certainly there will be some uncovered, and that's what the question was aimed at.

MR. CULP: Mr. Chairman, the purpose of this piece of testimony is to indicate what would happen if there were a breach in the spent fuel pool and whether there would in fact be enough water to keep the fuel assemblies covered; and there's nothing in the testimony to indicate that would ever happen, that the fuel would be uncovered.

MR. DOHERTY: The testimony doesn't say it is impossible, which is the only grounds, I think, it could be objected to that we should not even talk about it.

There would have to be a statement of impossibility.

(Bench conference.)

1-10

JUDGE WOLFE: Well, we've heard the objection. We are not ruling on that, but the Board is interested in asking you, Mr. Doherty, if you are tying that question into the testimony by these witnesses at page 5, the witnesses are saying at this portion of their testimony that the spent fuel pool cooling water would not escape because there is a makeup capability.

Now, you are asking the question what would happen if the spent fuel assembly is uncovered, and your question goes contrary to the testimony of the witnesses that, I take it, the assembly would not be uncovered at any time, once in the pool or hitting the liner of the pool.

So I think you had better back off of that and ask -- we'll permit you to ask whether they affirm what they said in this testimony, that a spent fuel assembly would never be uncovered, because your question flies in the face of this testimony.

The Board on its own will not allow the question because it flies in the face of the testimony.

You can delve more into their testimony seeking to discredit what they've said insofar as uncovering of the assembly is concerned.

MR. DOHERTY: All right.

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1-11

BY MR. DOHERTY:

Q. Is 150 gallons per minute the maximum makeup water for the pool in the event of an accident -- or breach, excuse me -- of the pool?

BY WITNESS MALEC:

A. From all possible means?

Q. Yes.

BY WITNESS MALEC:

A. No.

Q. What other means are available?

BY WITNESS MALEC:

A. We can use some makeshift rigs from fire protection system, from the potable water system we have installed to mineralize the water makeup, and we have the essential surface cooling water.

Q. All right. Now, haven't you already stated, though, that the essential surfaces cooling water system will provide approximately a hundred gallons in your testimony here?

BY WITNESS MALEC:

A. Is that a question?

Q. Yes.

BY WITNESS MALEC:

A. Yes.

Q. In order to do these makeshift arrangements,

1-12

would somebody have to be at the pool side?

BY WITNESS MALEC:

A. Yes. Not necessarily pool side, but in the vicinity of the spent fuel.

Q. Would there be any hazard to that person?

MR. CULP: Objection, Your Honor. I object to that question.

JUDGE WOLFE: What grounds?

MR. CULP: It's not relevant.

MR. DOHERTY: He has testified there would be other sources available, but he has stated there would have to be someone there, and I'm asking, well, indeed, would that be a safe place for a person. We have had previous testimony that this is radioactive material, and I think it's a fair question to ask if standing near the pool might be hazardous under those conditions.

JUDGE WOLFE: Standing near the pool might what?

MR. DOHERTY: Might be hazardous under those conditions.

(Bench conference.)

JUDGE WOLFE: All right. You've established the relevancy.

Objection overruled.

WITNESS MALEC: Perhaps you didn't understand

1-13

my first response.

It may be somewhat hazardous to the individual; however, the Safety Class III, Essential Surface Cooling Water System, is operated remotely from the control room.

Since it's about double the capacity we calculate that the maximum leak rate would be out of a channel, it's unnecessary for anyone to go to pool side.

These other makeshift sources, for whatever reason, are available, should anyone want to use them. They are not absolutely essential, nor are they safety class.

BY MR. DOHERTY:

Q. You say calculated a breach; is that correct?

BY WITNESS MALEC:

A. No, I did not calculate a breach.

Q. There is a statement in here, though, that during a breach of the pool boundary, the essential surface cooling water system will provide adequate water.

What is this hypothetical breach?

BY WITNESS MALEC:

A. Any breach.

Q. Any breach?

BY WITNESS MALEC:

A. Given that a breach occurs in a seam area, we

1-14

calculated what the flow rate would be to determine whether or not the safety related makeup sources were adequate to handle such a flow rate; not that we calculate there would be one. It's our safety related makeup capability to the pool.

Q. Why do you keep the spent fuel underwater?

BY WITNESS MALEC:

A. There are several reasons. It's thermally hot and radioactively hot.

Q. Any other reasons?

BY WITNESS MARTIN:

A. No.

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bm

BY MR. DOHERTY:

Q Is it so thermally hot that it might melt?

BY WITNESS MALEC:

A In the water or in the air?

Q Well, I asked you why you kept it covered.

BY WITNESS MALEC:

A Yes. There would be a potential for lack of cooling where there would be a breach of the cladding, if it were not kept covered with water.

Q Can you give me any idea how big this breach would be here at all, what you had in mind?

BY WITNESS MALEC:

A No, the breach size is immaterial, because the flow rate is limited by the channel.

Q I think I've allowed a word to get in here that I don't understand, and that's "channel." There is -- Was that -- You said "channel," and I take that literally as a sort of grooved thing underneath at the weld seam.

BY WITNESS MALEC:

A If you're referring to my last response, yes.

Q I see. I think you may have referred to it at page 2 also. Is that -- when you describe the pool liner leak detection system.

BY WITNESS MALEC:

A. Could you be more specific where you mean?

Q. When you describe the pool liner leak detection system at line 15.

BY WITNESS MALEC:

A. Yes. There is a channel under the liner weld.

Q. And that's the same channel we're talking about here?

BY WITNESS MALEC:

A. That's correct.

Q. Okay. Well, going to the foot of five, the question asked of you, Mr. Martin, did that dropping of a fuel bundle in the fuel pool impact the floor?

BY WITNESS MARTIN:

A. I'm sorry, but I don't understand your question.

Q. I didn't hear you, I'm sorry.

BY WITNESS MARTIN:

A. I did not understand your question.

Q. The question is: You referenced an incident at Millstone Unit 1 on page 5.

BY WITNESS MARTIN:

A. Yes.

Q. Are you answering that yes, it did hit the

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floor?

BY WITNESS MAPTIN:

A. Yes, it did hit the floor.

Q. In its dropping, was it impeded by the racks in any way?

BY WITNESS MARTIN:

A. The incident report, which I read, did not mention that.

Q. Mention what?

BY WITNESS MAPTIN:

A. You asked me if it hit the rack in any way. I responded that the incident report which I have read did not mention if it did or did not.

Q. Did it say it was a vertical drop?

BY WITNESS MARTIN:

A. Yes, it was.

Q. Was there -- Were you able to determine if the spent fuel pool was constructed in the same -- with the same steel liner as Allens Creek's will be designed?

BY WITNESS MARTIN:

A. The stainless steel 304 was used in that particular nuclear plant. However, you might want to ask either Dr. Chiou or Mr. Malec what type of stainless steel will be used for the liner.

That's your question?

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Q That's fine. I'd like to get the answer to that, yes.

BY WITNESS CHIOU:

A The materials for the Allens Creek project stainless steel liner is A-240, type 304.

Q Now, was that the same type of steel, Mr. Martin, or was it different, or were you uncertain?

BY WITNESS MARTIN:

A It's stainless steel 304.

Q Three -- It's hard to hear you. Are you saying "three or four"?

BY WITNESS MARTIN:

A Three, zero, four.

Q Three, zero, four. Type 304?

BY WITNESS MARTIN:

A Yes.

Q Okay. I'm sorry, it's the acoustics.

Now, was that fuel bundle a 7x7 or 8x8? Was that given you?

BY WITNESS MARTIN:

A I think it was a 6x6.

Q Was it of the same weight as the one that would be used here, to your knowledge?

BY WITNESS MARTIN:

A I don't know.

Q Okay. And do you know -- in relation to your statement about samples indicated no release of activity, do you know if the fuel bundle was burned to its maximum or the amount of burn-up that it had received?

BY WITNESS MARTIN:

A No, sir, I don't think it was.

Q Okay. Was it dropped from the same height? Could you determine?

BY WITNESS MARTIN:

A I'm sorry, the incident report did not mention the height it was dropped from.

Q It was not mentioned?

BY WITNESS MARTIN:

A No, it was not.

Q Can you give me the name of the report that you got on that or --

BY WITNESS MARTIN:

A It's from "Nuclear Power Experience" dated February 1976.

JUDGE LINENBERGER: Mr. Doherty, I'm having trouble understanding the relevance of this line of questioning.

Had the analysis of these gentlemen relied on the Millstone event to support a premise that no

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damage would occur at a pool drop accident at Allens Creek, then I would see the relevance. They did not rely upon that at all.

They, in fact, tell you in their testimony that they assumed damge to quite a few fuel pins, so I'm really having difficulty seeing the importance of taking up everybody's time to go through the Millstone event, which was not used as a basis for the analysis of this event.

MR. DOHERTY: Well, the Millstone event was offered -- and I don't think there's any way anyone could construe it, except as perhaps supporting the Applicant's position here.

I think I have a right to attack Applicant's position --

JUDGE LINENBERGER: Mr. Doherty --

MR. DOHERTY: -- and I have just about concluded my questions on this area, too.

So it's more in that line. I just ... why it was done.

JUDGE LINENBERGER: Why don't you find out why they quoted that event? It doesn't support anything. They assumed a much more serious incident than occurred at Millstone. And it's just possible that the only reason that they quoted Millstone at all was to show you that their premise was quite conservative.

But we're going through a lot of details here that don't get at the Allens Creek accident that was analyzed by these gentlemen.

I'm just --

MR. DOHERTY: Yes. Well, I'm concerned if I've caused you and your colleagues to feel like we've been wasting time.

I can assure you that I don't intend to pursue the Millstone event any further. However, I don't feel like asking questions which will make them look good either. They have counsel who can do that.

So at this point I feel like dropping off on the Millstone testimony entirely.

You might like to sit back a minute. I need to look in an envelope for possible other questions.

(Pause.)

JUDGE WOLFE: While Mr. Doherty is looking through his papers, I will ask you the question: Whoever wrote this -- or responded here, I guess it was Mr. Martin.

Of course, you were responding to questions, but what was your purpose -- I take it -- in getting into this area of the Millstone event ... to what purpose?

WITNESS MARTIN: The only purpose there, Your

Honor, is to indicate that there has been some incidents in that area, and just to offer what indeed had happened at other operating nuclear plants. And the experience indicates that there was some damage to the fuel assembly.

There was no damage to the spent fuel pool liner. There were no measurable radiological releases from the incident.

So the purpose of including that portion was just to indicate what has been gained, in terms of actual experience from an operating nuclear plant, which is also, incidentally, a boiling water reactor.

JUDGE WOLFE: But the report that you described or named was not fully particularized as to what, if any, other bundles the dropped bundle might have hit, or whether it hit other portions of the -- whether it hit brackets in the pool? Those sorts of particulars were not provided; is that correct?

WITNESS MARTIN: That is correct, sir.

(Bench conference.)

JUDGE WOLFE: With this understanding or this statement by Mr. Martin, if you have some more questions in this area, Mr. Doherty, you may proceed.

MR. DOHERTY: I have no further questions in that area.

JUDGE WOLFE: What?

MR. DOHERTY: To be honest with you, I didn't hear, so I don't -- I've missed something.

JUDGE WOLFE: Well, whether you heard or not, what I'm telling you is that in light of what they did say, you may proceed to ask several more questions, if you so desire in this area, as to the Millstone Unit 1 episode.

BY MR. DOHERTY:

Q. Is the spent fuel pool, with regard to make-up capability, with regard to just that part, is that in compliance with any regulatory guides of the Commission at this point?

BY WITNESS MALEC:

A. It's in compliance with regulatory guidance, yes. I'm not able to recall exactly where it appears. There would be several potential places.

Q. What about 1.13? Does that ring a bell?

BY WITNESS MALEC:

A. I just said that I don't recall exactly where it is.

Q. I see. I didn't hear you say that.

BY WITNESS MALEC:

A. I recognize that 1.13 deals with spent fuel pool, that area, but I don't recall exactly where it may appear.

Q Okay.

MR. DOHERTY: No further questions, Your Honor.

JUDGE WOLFE: All right.

Redirect, Mr. Culp?

MR. CULP: No, sir.

JUDGE WOLFE: Board questions?

JUDGE CHEATUM: I have none.

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BOARD EXAMINATION

BY JUDGE LINFNBERGER:

Q Gentlemen, on page 4 of the prefiled testimony, in the middle of the page there are some -- there are two numbers representing whole body and thyroid doses at the exclusion area boundary that have been calculated, based on the assumption of a failure of 98 fuel rods.

Are any of you able to tell us what assumptions went into that calculation with respect to burn-up of the rods, fraction of volatile fission products that escaped, the kinds -- what kinds of input parameters went into the calculation that resulted in those numbers?

BY WITNESS MARTIN:

A Yes, sir, if I may just make one correction to your statement. It was not 98 rods, it was 62 rods.

Q It was how many rods?

BY WITNESS MARTIN:

A Sixty-two.

Q Sixty-two?

BY WITNESS MARTIN:

A Yes. This is -- These doses are --

Q Okay, fine.

BY WITNESS MARTIN:

A Now, in response to your question, the first assumption is that your reactor has been operating three years at the maximum power level of 3758 megawatts thermal.

Therefore, all the radionuclides of concern have reached their equilibrium level.

Number two, 10 percent of the nuclides in the reactor core are assumed to be present in the gas cap of the rods, except for Crypton-85, which is 30 percent.

All of the activity in the gas cap is assumed to be released to the water of the spent fuel pool.

Out of the activity, which is released to the spent fuel pool, 100 percent of that activity for the noble gases is assumed to escape to, one, the spent fuel pool atmosphere and then to the environment.

The iodine which is released to the spent fuel pool, due to its high affinity to mix with the water and remain in solution, a conservative factor of 100 has been used to reduce the activity, which is now in the spent fuel pool, which then becomes airborne, or look at it that way -- there's a factor of 100 between what is assumed that has been released in the water to what is sub-secretly released to the spent fuel pool atmosphere.

The releases to the environment are assumed to have been made through the standby gas treatment system charcoal absorbers, which have a design iodine-filter efficiency of 99 percent.

And lastly, upon reaching the environment the cloud of radionuclides is assumed to mix at dilution factor equal to what would be occurring only five percent of the time.

And ultimately the receptor of the dose -- the thyroid dose, is assumed to breathe at a rate equal to that of organ activity, and is assumed to remain there for a period of two hours.

Q. Two hours?

BY WITNESS MARTIN:

A. Two hours.

And the whole body dose is assumed that -- the assumption is made for the whole body dose that the person is submerged in a cloud of such radionuclide, also for the duration of the release.

Q. It sounds as though you assumed the most adverse meteorological circumstances for this calculation?

BY WITNESS MARTIN:

A. It's not the most adverse. It's the level which is considered adverse enough for these types of calculations, because it's only exceeded five percent of the time.

Q. Okay.

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BY WITNESS MARTIN:

A. The most adverse, of course, would be the 100 percent number.

Oh, I forgot to include one thing: There is also a 24-hour decay period because the assumption is made that it will take at least 24 hours before the fuel is moved from the reactor vessel after shutdown.

Q. Okay. By the way, is there a -- normally is the operational restrictions on plants such as Allens Creek, is there a tech spec or administrative control limitation that requires that at least 24 hours of cooldown occur before fuel is transported?

BY WITNESS MALEC:

A. I'm not aware of any, Your Honor. However, pragmatically, by the time the reactor is shut down, the primary system is depressurized and cooled down; the reactor head is removed and the fuel is transferred. It's a minimum of 24 hours.

I think probably a more realistic number may be as high as 40 hours, until the time the spent fuel actually reaches the fuel pool area.

Q. Well, all right. You're telling me that practically you really can't get at the fuel in much less time.

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BY WITNESS MALEC:

A. Perhaps in emergency conditions it would be possible. We're looking now at a normal refueling period. This is what we assume, based on the time durations of the activities that must be performed to get to the fuel and the times required to reach particular fuel elements, lift them and transport them over.

Q. I guesr I was asking a slightly different question. Is there a tech spec or administrative control limitation that prohibits star-ing fuel transfer to the spent pool in less than some amount of time?

BY WITNESS MARTIN:

A. I do not think there is.

Q. Pardon?

BY WITNESS MARTIN:

A. I do not think there is.

JUDGE LINENBERGER: All right, I have no more questions.

JUDGE WOLFE: Cross, Mr. Dewey?

MR. DEWEY: No, sir.

JUDGE WOLFE: Mr. Doherty?

RECROSS-EXAMINATION

BY MR. DOHERTY:

Q. Mr. Martin, I think you said that one of the assumptions in calculating your dosage at page 4 was that

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this was after three years of operation?

BY WITNESS MARTIN:

A. Yes.

Q. Would three years of operation create the maximum radioactivity content of the core for the Allens Creek plant?

BY WITNESS MARTIN:

A. Yes, it would.

Q. Do you know what the planned duration of a fuel cycle is for the Allens Creek plant?

BY WITNESS MALEC:

A. As it's planned today, it's a one-year operation between refuelings.

Q. Do you at this moment, Mr. Malec, expect that the final operational cycle will be one year?

BY WITNESS MALEC:

A. That's our current plans. I have no reason to think that there would be any other number.

Q. Maybe I didn't hear you ... you said you have heard no other number mentioned?

BY WITNESS MALEC:

A. I have no reason to believe it would change from one year at this point.

Q. Uh-huh. I had a little trouble hearing it, but I think the last exchange, there was talk about emergency removal of the core. Would that -- Would you be required to place that in the spent fuel pool, or would you have any residence time in a storage pool in the containment?

BY WITNESS MALEC:

A. My response dealt with the amount of time to get to the fuel elements themselves in the reactor. I said that under a normal type of condition, a normal refueling outage, we looked at typical durations. And they would be on the order of about 40 hours.

I said that it might be possible, under some other emergency-type conditions, to get in sooner. However, for the purposes of our analysis for a normal transfer, we assumed 24, where a more realistic and more pragmatic number would be about 40 hours.

Q. Okay. I'm not really too worried about the numbers. But you said "transfer," that was your word just a minute ago. What were you talking -- transfer from where to where, please?

BY WITNESS MALEC:

A. Transfer of spent fuel elements out of the reactor.

Q. To?

BY WITNESS MALEC:

A. Ultimately to the fuel handling building.

Q Would that imply a stopping point along the line at any other spent fuel pool?

BY WITNESS MALEC:

A There's a temporary storage pool provided in the upper elevation of the containment. It is possible that we would either remove it directly from the reactor and transfer it into the fuel transfer tube, or because there's a finite time required to transfer down through the fuel transfer tube, we may -- or HL&P may elect to go back and take out another spent fuel element and put it in this temporary storage position in the upper fuel storage pool.

Q That storage pool, though, couldn't hold a whole core, could it?

MR. CULP: Your Honor, I'm going to object to any more questions along this line. The Board's question was very specific: Are there any technical specifications as to the time limit for removing the fuel bundles to the spent fuel pool.

And I think Mr. Doherty has gone way beyond the Board question.

(Pause.)

MR. DOHERTY: I have no further questions.

JUDGE WOLFE: Redirect, Mr. Culp?

MR. CULP: No, sir. The witnesses are to be excused permanently -- I take that back. Mr. Malec is to be excused temporarily.

JUDGE WOLFE: Mr. Malec, you are temporarily excused. The other witnesses are excused permanently.

(The witnesses were excused.)

JUDGE WOLFE: We will have a short recess.

(A short recess was taken.)

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EVENING SESSION

6:10 p.m.

JUDGE WOLFE: All right. During the recess, I had occasion to query Mr. Culp on any agreements that had been made to setting over either Mr. Hamilton or Mr. Gotchy's testimony.

MR. DOHERTY: Excuse me, Your Honor, I didn't hear the one word that seemed most important. You were going to --

JUDGE WOLFE: I said during the recess I talked to Mr. Culp with regard to any agreement that I understood may or may not have been made setting over Mr. Gotchy's or Mr. Hamilton's testimony until such time as Mr. Doggett was available.

Mr. Culp advised, I believe, that you had said something with regard to the fact that Mr. Doggett would not be here.

MR. DOHERTY: I received a note, I think it was from Dr. Gotchy, which indicated that, one, Mr. Doggett had a jury trial today and, two, there was a death in the family. His mother-in-law died in Canada and he will not be here tomorrow.

MR. DEWEY: This was not from Dr. Gotchy, I believe.

MR. DOHERTY: I'm sorry. It was someone. I

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just couldn't tell who it was, but in any case, that's the only thing I know about Mr. Doggett at this point.

JUDGE WOLFE: I see. All right. Well, in any event, we must proceed.

Mr. Culp.

MR. CULP: Your Honor, at this time the Applicant calls Leonard D. Hamilton to the stand to testify on Cummings Contention 9.

Dr. Hamilton has previously testified in this proceeding and has previously been sworn.

JUDGE WOLFE: You are still under oath.

Whereupon,

LEONARD D. HAMILTON

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

DIRECT EXAMINATION

BY MR. CULP:

Q. Dr. Hamilton, do you have before you a document which is entitled, "Affidavit of Leonard D. Hamilton Concerning the Health Effects of Low Level Radiation"?

A. Yes, I do.

Q. Did you prepare this document or was it prepared under your supervision?

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A. Yes, I prepared this document.

Q. Do you have any corrections to make to this document at this ti e?

A. Yes. I have a few corrections to make.

In the bibliography, on page 2 of the bibliography, there are two publications. No. 11, which is listed as "In press," by Ginevan, and I'd like to give the actual reference now. It's 38: --

JUDGE CHEATUM: Dr. Hamilton, what did you say?

THE WITNESS: I'm sorry, I'll repeat it.

Reference 11, instead of "In press," delete "In press," and insert, "38:129-138." Those are the page numbers. "1980."

Reference 13, "In press," cross out "in press," and insert "37:202-220, 1979."

Then on the next page there are some typos in the references. In Reference 14, the title of the second author is "Dolphin." The "o" should be omitted.

On the third line, it says, "T. Jan..." That should be struck, J-a-n, and you should put in N-a. The author is Najarian.

And on the next line, "National Radiology," the "y" should be stricken, and it's "National Radiological Protection Board." There should be a Capital "P."

3-4

In reference 14, the second author is W-a-x-w-e-i-l-e-r, and the last author is Cox, C-o-x.

Those are all the corrections I have in the bibliography.

In my personal qualifications, which are attached, I would like to make one update.

Where it says in the middle of page 1 of my personal qualifications, "The Biomedical and Environmental Assessment Division is the lead group in the --" and then I'd like to insert the following. This takes account of the reorganization of the Department of Energy under Dr. Edwards.

"-- in the Health and Environmental Risk Analysis Program," caps for each of those, comma, "Human Health & Assessment Division," and then it goes on, "Office of Health and Environmental Research, comma," and then instead of "of the Assistant Secretary of Environment," it should be, "Office of Energy Research, comma, U.S. Department of Energy."

That takes account of the reorganization of the program.

Then, of course, in the next paragraph, it's now, since this affidavit, we've been working for the last eight years instead of seven years.

And I think there's a final typographical error

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on page 4, on the one, two, three, four, five, sixth line down, there's an "o" missing in "Biological." "The National Academy of Sciences Committee on Biological Effects of Atomic Radiation."

With those corrections, it seems to be in order.

BY MR. CULP:

Q Dr. Hamilton, with those corrections, is this document correct to the best of your knowledge and belief?

A Yes, it is.

Q Do you adopt this as your testimony in this proceeding?

A I do.

MR. CULP: Mr. Chairman, at this time I move the affidavit of Dr. Hamilton concerning the health effects of low level radiation be incorporated into the record as if read.

JUDGE WOLFE: Any objection?

MR. DEWEY: No objection.

MR. DOHERTY: Your Honor, I'd like to take the witness on voir dire.

MR. CULP: Mr. Doherty, may I inquire whether you intend to challenge the qualifications of Dr. Hamilton to testify on this issue?

MR. DOHERTY: Yes, I do.

MR. CULP: I would like to point out

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Dr. Hamilton has been qualified as an expert witness by the Board on the issue of health effects of the uranium fuel cycle, which certainly includes low level radiation; and it seems to me it is a waste of time to ask questions on voir dire at this time.

MR. DOHERTY: Let's see, Counsel. You've represented to me that he was on a low level radiation effects issue last February or March?

MR. CULP: I said he had been qualified as an expert witness by this Board when he testified on the issue of the health effects of coal and nuclear, and I said he was qualified as an expert in health effects of the uranium fuel cycle, which would include low level radiation.

MR. DOHERTY: All right. I will pass voir dire.

It is getting late, and maybe we can move along.

JUDGE WOLFE: All right.

(Bench conference.)

JUDGE WOLFE: Mr. Culp.

MR. CULP: Yes, sir.

JUDGE WOLFE: We have noted that Dr. Hamilton's affidavit is directed to the original Cummings 9.

In light of our Order of September 1, 1981,

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we restricted the disputed issue and reworded the contention.

Now, what is your purpose now? How do you intend to utilize Dr. Hamilton's testimony to respond to the more restricted and limited issues set forth in our Order of September 1?

MR. CULP: Well, Your Honor, let me put this in perspective, if I can. I think, as I remember, we did inform the Board that if in fact the Summary Disposition Motions were denied, we would intend to go forward on the affidavits that had been filed in support of the Motions for Summary Disposition.

This particular issue was filed by the Staff seeking summary disposition. Dr. Hamilton's affidavit was filed in support of the Staff's Motion for Summary Disposition, and the affidavit basically supported the Staff's witness, Dr. Gotchy, on this issue; and the affidavit was directed towards the various studies that had been mentioned that were contrary, or apparently were contrary to the prevailing opinion there's no health effect from low level radiation.

When the Board issued its decision ruling on the Motion for Summary Disposition, the Board did re-formulate the issue and did it in terms of whether the Staff's figures for the health effects of low level radiation were accurate.

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We reviewed Dr. Hamilton's affidavit and believe that it addressed the important issues in the case. The important issues, that is, dealing with the -- and I don't want to use the word evidence, but dealing with studies that tended to say that there were some health effects from low level radiation.

We would intend to use Dr. Hamilton's testimony now, in effect, to support Dr. Gotchy's testimony that he has filed in this proceeding on this issue.

Dr. Gotchy's testimony is basically a rewrite of his affidavit, plus some additional information, and we have Dr. Hamilton here basically to support the testimony of Dr. Gotchy on this issue.

JUDGE WOLFE: In other words, it is not being offered into evidence as relating to the original contention, as such? It's offered now, even though it references the original contention, the original Cummings 9, it is offered now in support of Dr. Gotchy's testimony, which does deal with the Board's more restrictive reworded contention?

MR. CULP: That is correct.

JUDGE WOLFE: I see. We will admit it, then, absent objection, for that purpose, and we will not consider it, obviously, as evidence presented insofar as it relates

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to the original contention.

All right. The affidavit of Leonard D. Hamilton, dated 12-17-80 and the attachments are incorporated into the record as if read.

(Applicant's Testimony of Dr. Leonard D. Hamilton on Cummings Contention 9 follows:

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12/17/80

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)	)	

AFFIDAVIT OF LEONARD D. HAMILTON CONCERNING THE HEALTH EFFECTS OF LOW LEVEL RADIATION

My name is Leonard D. Hamilton. A statement of my personal qualifications is attached.

I have reviewed: (a) the contention that:

The health effects* of low level radiation emitted during normal operation of the plant, even though meeting the "as low as is reasonably achievable" standards of Appendix I, if included in the NEPA balancing of costs and benefits, would alter its benefit to the extent that costs would outweigh benefits.

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* Health effects include impacts upon humans, animals, and plants.

(b) The NRC Staff's statement of material facts as to which there is no genuine issue to be heard attached to the NRC Staff's motion for summary disposition of November 26, 1980, and

(c) The affidavit of Reginald L. Gotchy concerning the NEPA impacts of low level radiation of November 26, 1980.

I find the NRC Staff's statement of material facts with respect to the consolidated contention on the NEPA impacts of low level radiation to be reasonably conservative and concur with them. Likewise I find that affidavit of Reginald L. Gotchy concerning the NEPA impacts of low level radiation to be reasonably conservative and concur with the discussion on current health effects models, the validity of NRC health effects models, and the conclusion on the de minimus nature of Appendix I health risks.

As a physician with extensive experience in health effects and their assessments there are several additional points I would like to make. First, calculations of health effects must be based on risk estimates. Risk estimates are made by multiplying the estimated delivered dose of radiation by an established damage function. I have made such risk estimates for the health effects of the uranium and coal fuel cycles based upon the annual incidence of effects to be expected from operation of standard plants.

The damage functions for radiation that I have used were derived from the BEIR I (1972)[1], the United Nations Scientific Committee on the Effects of Atomic Radiation (1977)[2], and currently the BEIR III (1980) reports.[3] These are essentially the same reports relied on by the NRC Staff in their statement of material facts and by Dr. Reginald L. Gotchy in his affidavit. The very low dose and dose rates given by natural background radiation in the environment, and the considerably lower doses that would be given at low dose rates by nuclear power stations, are obviously very much lower than those for which there are data on damage e.g. tumor induction by radiation. However, to get a rough idea of risk one assumes that the linear proportional dose and tumor induction observed at much higher doses and dose rates can be extrapolated down to the lowest doses. The assumption of linear proportionality down to the lowest doses and dose rates undoubtedly over-estimates actual risk. As Dr. Gotchy states in his affidavit (p. 6), it produces "estimates of risk that are generally characterized by most radiobiologists as tending to be upperbound (i.e. overestimates of the actual risk). Indeed both BEIR committees (1972 and 1980) noted that the lower bounds of risk from exposure to low level and low LET radiation (the type emitted from LWR's) could include zero." This means that the actual numbers of cancers induced by these very low doses, given at very low dose rates will be lower than the estimates and may be zero. Nevertheless a conservative approach and one I use in making our risk estimates on the nuclear fuel cycle is to take the dose-effect values obtained at high-dose levels and to extrapolate them down to low-dose levels.

Several recent reports (Bross;[4] Mancuso, Stewart and Kneale;[5] and Najarian[6]) have been interpreted by some people to indicate that the commonly employed risk estimates, which are based on the UNSCEAR (1977)[2] and BEIR Committee Reports (1972 and 1980)[1,3], underestimate the risk of radiation at all levels. They especially emphasize that the linear theory (that the risk per unit dose as derived from available data at high levels of radiation dose holds all the way down to zero exposure dose) is not sufficiently conservative in estimating risk at low doses but rather underestimates it.

Bross believes he has identified subgroups in the population which are especially sensitive to radiation damage. His belief derives from his analysis of the Tri-State Leukemia Survey, wherein he studied an association between some "indicators of susceptibility" (viral infections, bacterial infections, and allergy) shown by the leukemic child from birth until diagnosis of leukemia. He concluded "the apparently harmful effects of antenatal irradiation are greatly increased in certain susceptible subgroups of children possessing the indicators associated with a slightly higher intrinsic risk of leukemia." However, reanalysis[7] of his findings shows that children with leukemia are simply more prone to viral and bacterial infections and allergies before the clinical onset of the disease, i.e., these indicators characterize the disease itself and do not relate to the child's inherent susceptibility to leukemia. The incidence of these diseases as part of the pre-leukemia phase of leukemia in children is well known in clinical hematology. Analysis of Bross' data shows that the incidence of these indicator diseases before the clinical onset of leukemia is the same in children who had received no irradiation in utero as in those who had.[7] The hypothesis of Bross, that there is a susceptible portion of the population at higher risk of leukemia, has also been challenged on the grounds that Bross' methods yield no way to identify susceptible individuals ahead of time and so no way to test his thesis.[8]

More recently, Bross has suggested that the relatively small radiation exposures from diagnostic X-rays in adults significantly increases the risk of leukemia.[9] It appears that Bross assumes, in coming to this conclusion, that in the absence of diagnostic X-rays, the incidence of heart disease and leukemia is zero.[9,10] Were this not the case the fact that the "dose-response" curves of adults exposed to diagnostic X-rays are flat below 10 rad exposure would suggest a threshold. Indeed, a more conventional relative risk analysis[11] found little or no increase in risk of leukemia from a small number of diagnostic X-rays. Bross also assumes here that relative risks are fixed and that the percentage of the population affected varies with dose, i.e., the basic response variable is the proportion of the irradiated population affected by radiation. Conventional relative risk analyses assume that everyone is affected and that the relative risks vary with dose. The improvement made by Bross et al.'s approach is unclear. The position taken here by Bross appears to be at odds with his earlier paper, in which he postulated the existence of a sensitive subgroup of fixed size whose relative risk of leukemia increased rapidly with increasing X-ray dose.

Finally, one should note that the leukemia risk (or "percent affected") does increase dramatically in males (females appear to be unaffected) after large numbers of diagnostic X-rays. However, the cause-effect relationship is uncertain in that large numbers of diagnostic X-rays -- 40 or more within 10 years -- implies the presence of a disease state perhaps deriving from heart disease or a preleukemic sensitivity to infections.

Mancuso, Stewart, and Kneale[5] have reported preliminary findings on the work and survival experience of 24,939 male workers with 3,520 certified deaths and of an unspecified number of female workers with 412 certified deaths at the Hanford Works, Richland, Washington between 1943 and 1971. The preliminary report, largely limited to analysis of data on the 3,520 male deaths for which death certificates were available, claims to demonstrate a radiation-induced excess of cancers, greater than linear models would indicate. Their analysis has been widely criticized. Their report does not state the actual individual doses received by Hanford workers who died of cancer, only mean cumulative radiation doses. Besides, their study did not take into account the calendar year in which the cancer began and made no correction for the fact that the incidence of the cancers they were observing in the Hanford workers also increased during the period of the study in the population at large. Thus, Table 11 in their publication, showing an increase in cancer with increasing dose accumulated over increasing time, fails to take into account that even in the absence of the increasing dose of radiation, there is a similar increase in cancers they were finding in the U.S. as a whole when plotted against increasing time. Other analyses of the same data published by Marks et al.[12] and by Hutchinson et al.[13] point to the possibility of an association with the work experience for two cancer types: cancer of the pancreas and multiple myeloma (multiple myeloma in whites is increasing in the U.S. for no known reasons). There is no reported radiation relationship for lymphatic or haemopoietic cancers other than myeloma, i.e., no excess of leukemias (which previous experience suggests should have been most observable where radiation is a factor).

Since the specified radiation doses were very small, perhaps on the order of a few rads, the cancer-doubling estimates found in the Mancuso, Stewart, and Kneale paper have been strongly disputed. If the postulated small dose actually caused a doubling of the spontaneous rate of cancers, then background radiation would produce more than the numbers of cancer observed in the population. It therefore appears that if these doubling doses are correct, something other than radiation was the cause of the observed cancers.

Najarian and Colton[6] estimated that since the Portsmouth Naval Shipyard (PNS) in New England began to service nuclear-powered ships in 1959, 20,000 people were employed there, of whom about 20% were exposed to radiation. From a search of death certificates 1959-77, 1,450 former PNS employees who had died below age 80 were identified in New Hampshire, Maine, and Massachusetts. To ascertain whether these ex-employees were radiation-exposed workers, attempts were made to contact near relatives by telephone. This was successful in 525 cases and it was established that 146 were probably exposed to radiation during their working life.

The authors show that, compared with mortality in U.S. white males for 1973, the observed numbers of cancers and leukemias were considerably greater than those expected for example, 56 cancer deaths were found in death certificates of 146 ex-workers exposed to radiation; only 34.5 were expected. In non-exposed workers there were 88 cancers; 79.7 were expected. For leukemias there were 6 in the former radiation workers; only 1.1 were expected.

Najarian and Colton listed some inadequacies in their survey. It was an analysis of deaths only; no information was available on the total population at risk. There could be a bias in the information supplied by relatives. They had no information on how long workers worked at the shipyard, how long nuclear workers were exposed to radiation, and the amounts of radiation they received. Consideration was not given to other toxic agents, such as asbestos, smoking, industrial solvents which could have acted alone or synergistically with radiation to cause the apparent excess deaths from cancer and leukemia.

There are other inadequacies in this survey. To exclude some of the effects of other carcinogens, one must show that cancer frequencies increase with increasing radiation exposure, but knowledge of the lifetime accumulated doses of the former employees was not available. More importantly, if the radiation work at PNS began only in 1959, it is unlikely that changes in overall cancer frequency induced by radiation would appear before at least 10 years after exposure, or after 5 years for leukemia, these being roughly minimum latent periods for cancer induction. The data given in Najarian and Colton can be divided into deaths during the periods from 1959-69, when radiation effects would not be expected to appear, and 1970-77, when effects might be expected. In 585 death certificates of persons who died between 1959-69, 24.6% had cancer listed as the cause of death. Considering the 33 radiation-exposed workers who died during this period, 13 or 39.4% of the deaths were recorded as due to cancer. In 865 death certificates 1970-77, 25.7% had cancer as the cause of death; hence there was no significant difference between the percentage of cancer deaths between the two periods for all workers. For the 113 radiation-exposed workers, 43 or 38.1% of deaths in the later period were due to cancer -- no more than in the earlier period (39.4%). The data are tabulated below:

				RADIATION EXPOSED		
	ALL DEATHS	CANCER DEATHS	% CANCER DEATHS	ALL DEATHS	CANCER DEATHS	% CANCER DEATHS
1959-69	585	144	24.6	33	13	39.4
1970-77	865	222	25.7	113	43	38.1
	1,450	366				

The absence of any apparent latent period effect casts doubt on conclusions about the contribution of radiation to the curiously high numbers of cancer deaths among the radiation workers.[14]

In the meanwhile, NIOSH made available to Drs. Najarian and Colton radiation exposure data supplied by the U.S. Navy. On February 2, 1979, at a symposium sponsored by the Johns Hopkins School of Public Health, Baltimore, Maryland, Drs. Najarian and Colton introduced these radiation exposures into their PNS Study. At this time, they announced that in contrast with the original Lancet data, where 6 leukemia deaths were observed instead of 1.1 expected, it was found that two of the cases of leukemia had no history of radiation exposure. One had less than 0.1 rem, which is what one receives after one year's natural background. One received 15 rem, one 5 rem, and one "not remembered" -- probably less than 5 rem. The number of leukemias is now 3 instead of 1.1 expected. For all cancers the new data are:

	CANCERS			
EXPOSURE	NUMBER	OBSERVED	EXPECTED	RATIO
Less than 0.1 rem	64	17	13.5	1.26
From 0.1 to 0.99	50	16	10.5	1.53
Greater than 1	49	19	10.2	1.58
No exposure	358	92	7.49	1.24

Chi-square test shows no significant difference in the ratio among the exposed levels at p = 0.10. Cochran's chi-square test for a linear regression, which considers that that ratios increase in the expected direction shows no statistical significance at p = 0.05 but <u>is</u> significant at p = 0.10.

In any event the final report of the U.S. Department of Health and Human Services, Public Health Service Centers for Disease Control, National Institute for Occupational Safety and Health (NIOSH's) Epidemiologic Study of Civilian Employees at the Portsmouth Naval Shipyard (PNS)[15] based on a total cohort of 24,545 civilian white males employed at PSN between January 1, 1952 and August 15, 1977 is now available (the study was referred to in Dr. Gotchy's affidavit at page 11). The report found no excess of deaths due to malignant neoplasms and due specifically to neoplasm of the blood and blood-forming tissue in civilian workers at PNS. "This NIOSH study had over a 99% probability of detecting the 5-fold increased risk of death due to leukemia reported by Najarian et al.[6] among radiation exposed employees at PNS if it had existed. Furthermore, had the true relative-risk of death from leukemia been 2.2, the likelihood of detecting such a risk would still have been 80%. However, when observed leukemia deaths at the shipyard were compared with expected deaths, derived from the

United States white male population rates, no excess was found. No relationship between exposure to radiation and mortality from any cause was observed among the PNS population when compared to the United States white male population. Furthermore, no excess in leukemia mortality was observed in the radiation exposed population when compared to the non-radiation exposed employees of PNS."

The report cautions that an insufficient number of years may have elapsed for most "radiation workers since their initial radiation exposure to permit manifestation of currently latent cancers. In addition, the number of workers with radiation exposure at PNS was relatively small, making the opportunity for observing a slight excess in mortality very unlikely."

Thus, although these claims of higher risks from the levels described by Bross, Mancuso, Stewart, Kneale, and Najarian have become the subject of considerable public debate, examination to date of their work does not support these claims.

Dr. Gotchy in his affidavit page 12, quotes the conclusions of BEIR III that "available data relative to the effects of low-dose or low-dose rate exposures on carcinogenesis in humans and experimental animals do not, in general, support the hypothesis of an increased probability of induction at low dose rates," and goes on to quote the conclusions of

BEIR III on possible synergism between cigarette smoking and exposure to radon-222 decay products. Recent data from miner groups and the Japanese A-bomb survivors indicate that smoking acts to shorten the latent period to the onset of bronchial cancer, but that combination of smoking and radiation leads to a cancer risk that is not much more than additive.

For these reasons I concur with the NRC staff statement of material facts and Dr. Gotchy's affidavit.

The foregoing affidavit was prepared by me and I swear that it is true and correct to the best of my knowledge, information and belief.

Leonard D Hamilton
Leonard D. Hamilton

Subscribed and sworn to before me
this 3rd day of December, 1980.

Marilyn L. Newton
Notary Public

My Commission Expires: 8/31/85

My Commission [illegible]

REFERENCES

1. BEIR, The Effects on Populations of Exposure to Low Levels of Ionizing Radiation, Advisory Committee on the Biological Effects of Ionizing Radiation (BEIR). Division of Medical Sciences, National Academy of Sciences. National Research Council, Washington, D.C., November 1972.

2. Sources and Effects of Ionizing Radiation, United Nations Scientific Committee on the Effects of Atomic Radiation 1977 Report to the General Assembly, with Annexes, United Nations, New York, 1977.

3. Committee on the Biological Effects of Ionizing Radiations. The Effects on Populations of Exposure to Low Levels of Ionizing Radiations, National Research Council, National Academy of Sciences, Washington, D.C., May 1980.

4. Bross, I.D. and Natarjan, N., Leukemia from Low-Level Radiation: Identification of Susceptible Children, New England Journal of Medicine 287:107-110, 1972.

5. Mancuso, T.F., Stewart, A., and Kneale, G., Radiation Exposures of Hanford Workers Dying from Cancer and Other Causes, Health Physics 33:369-85, 1977.

6. Najarian, T. and Colton, T., Mortality from Leukemia and Cancer in Shipyard Nuclear Workers, The Lancet, May 13, 1978, p. 1018.

7. Smith, P.G., Pike, M.C., and Hamilton, L.D., Multiple Factors in Leukaemogenesis, Letter to Editor, Brit. Med. J. 2:482-483, 1973.

8. Rothman, K.J., Review of Dr. Irwin Bross' Presentation on Radiation Exposure and Cancer Risk, prepared for a public meeting of the low-level effects of ionizing radiation sponsored by the U.S. Nuclear Regulatory Commission, Washington, D.C., April 7, 1978; see also Letters to Editor, Journal of the American Medical Association 238(1):1023-1024, 1977.

9. Bross, I.D.J. and Natarjan, N., Genetic Damage from Diagnostic Radiation, J. Amer. Med. Assoc. 237:2399-2401, 1977.

10. Bross, I.D.J., Ball, M., Rzepko, T., and Laws, R.E., Preliminary Report on Radiation and Health Disease, J. [illegible] d., 9:3-15, 1978.

11. Ginevan, M.E., Nonlymphatic Leukemias and Adult Exposure to Diagnostic X-rays: The Evidence Reconsidered, Health Physics, in press.

12. Marks, S., Gilbert, E.S., and Breitenstein, B.D., Cancer Mortality in Hanford Workers, IAEA Symposium on the Latent Biological Effects of Ionizing Irradiation, March 1978, IAEA SM-224.

13. Hutchinson, G.B., Jablon, S., Land, C.E., and MacMahon, B., Review of Report by Mancuso, Stewart, and Kneale of Radiation Exposure of Hanford Workers, Health Physics, in press.

14. Reissland, J.A. and Dolphoin, G.W., A Review of Mortality from Leukemia and Cancer in Shipyard Nuclear Workers by T. Janjarian and T. Colton, The Lancet, VI, p. 1018, 1978, National Radiology protection Board (NRPB), Harwell, Didcot, England, May 18, 1978.

15. Rinsky, R.A., Wasweller, R.J., Bierbaum, P.J., Terpilak, M., Zumwalde, R.D., Murray, W.E., Landrigan, P.J., and Cos, C. Epidemiologic Study of Civilian Employees at the Portsmouth Naval Shipyard, Kittery, Maine, U.S. Department of Health and Human Services, Public Health Service Centers for Disease Control, National Institute for Occupational Safety and Health, Division of Surveillance, Hazard Evaluations, and Field Studies, Cincinnati, Ohio, December 1980.

DR. L. D. HAMILTON
PERSONAL QUALIFICATIONS

My name is Leonard D. Hamilton. My address is: 6 Childs Lane, Setauket, New York, 11733. I am, among other responsibilities, Head of the Biomedical and Environmental Assessment Division in the National Center for Analysis of Energy Systems at Brookhaven National Laboratory, Associated Universities, Inc., Upton, New York, 11973. The Biomedical and Environmental Assessment Division is jointly sponsored by the Department of Energy and Environment and Medical Department at Brookhaven. The Biomedical and Environmental Assessment Division (BEAD) aims at developing a realistic assessment of biomedical and environmental effects of energy production and use. All forms of energy, including electric power generation using fossil fuels, hydro, nuclear, and new technologies, are assessed. The Biomedical Environmental Assessment Division is the lead group in the Office of Health and Environmental Research of the Assistant Secretary of Environment, U. S. Department of Energy, assessing the health and environmental effects of energy production and use and among other responsibilities is charged with producing a health and environmental effects assessment of the National Energy Plan.

I have been involved in assessing the risks of radiation for man for 35 years, specifically the health effects of nuclear energy for electric power generation for 20 years, and the assessment of the comparative health effects from various energy sources, for the past 7 years. The Biomedical and Environmental Assessment activity formally began in July, 1973; for the past and present year our level of effort is 204 man-months annually.

I received my Bachelor of Arts in 1943 and qualified in medicine from Oxford University in 1945. I am a registered medical practitioner in the United Kingdom and licensed physician in New York State. After several positions in University hospitals, which included a postion as Resident Medical Officer at the Radiotherapeutic Centre, Addenbrooke's Hospital, Cambridge, during which time I was concerned with the management of cancer patients undergoing treatment with radiation, I proceeded to research at Cambridge University on histological studies of the mechanism of the action of therapeutic doses of ionizing radiation for which I received my Ph.D. in experimental pathology in 1952. In the meanwhile, in 1951, I had received my Doctor of Medicine degree from Oxford; this is a senior medical qualification in the United Kingdom, roughly equivalent to Diplomate in Internal Medicine in the United States. I am also a Diplomate of the American Board of Pathology (Hematology).

From 1950-1964 I spent 14 years on the research staff of the Sloan-Kettering Institute for Cancer Research and on the clinical staff of Memorial Hospital in New York being Associate Member and Head, Isotope Studies Section at the Institute and Assistant Attending Physician, Department of Medicine at Memorial. During this time I was also a member of the faculty of Cornell University Medical College and a Visiting Physician, Cornell Division, Bellevue Hospital. Since then I have maintained a continuing association with the Sloan-Kettering Institute as Associate Scientist.

At the Institute my laboratory research was on the molecular structure of the genetic material (DNA) and the cells in man concerned with the immune mechanism. I provided the DNA on which the proof of the double-helical structure of DNA is based, and was one of the first to establish the long life of the immune cells in man. My clinical work in Memorial Hospital involved research on the treatment of patients afflicted with cancer and leukemia with new chemical agents and also with new applications of radiation therapy.

In 1964 I joined the scientific staff of Brookhaven National Laboratory as Senior Scientist and Head, Division of Microbiology, and Attending Physician, Hospital of the Medical Research Center. Since 1973 I have been Head of the Biomedical and Environmental Assessment Group which in 1976 became a Division of the National Center of Analysis of Energy Systems.

At Brookhaven I continued my laboratory research begun at Sloan-Kettering. In addition since my Visiting Fellowship at St. Catherine's College, Oxford 1972-73, I have been concerned with placing all risks in life in perspective; and since becoming Head of the Biomedical and Environmental Assessment activity in 1973, particularly with the assessment of the hazards associated with different energy sources and their use. Our group has the lead responsiblity to DOE for the assessment of health and environmental effects from various energy systems, and of coordinating such assessments in national laboratories, universities and research institutes in the United States.

My interest in the risks of radiation for man began with my Ph.D. work in Cambridge in 1946 and, since DNA and the immune system are prime targets of radiation damage has continued throughout my laboratory research. I was associated informally with the United Nations Scientific Committee on Effects of Atomic Radiation (UNSCEAR) almost since its inception in 1957, served as Consultant, Office of the Under-Secretaries for Special Political Affairs (UNSCEAR), 1960-62, and was responsible for the first draft of the somatic effects of radiation in the 1962 report. This section covers the effects of radiation in inducing leukemia and cancer in man. I have reviewed most of the working papers of UNSCEAR since then. I was a member of the National Research Council-National Academy of Sciences (NAS-NAS) Committee on Bilogical Effects of Atomic Radiation, Subcommittee on Hematologic Effects, 1960-64, the NRC-NAS Solar Energy Research Institute Workshop, 1975, the NRC-NAS Committee on Environmental Decision Making, Steering Committee on Environmental Monitoring, Panel on Effects Monitoring 1975-76, the NRC-NAS Health Effects Resource Group, Risk Impact Panel of the Committee on Nuclear and Alternative Energy Ssytems (CONAES) 1975-80, the NRC-NAS Panel on the Trace Element Geochemistry of Coal Resource Development Related to Health 1976-80, and the NAS-NRC Committee on Research Needs on the Health Effects of Fossil Fuel Combustion Products, 1976-80.

I was a member of the Mayor's Technical Advisory Committee on Radiation, New York City, since 1963 until its end, December, 1977 and have been a member of the Technical Advisory Committee on Radiation to the Commissioner of Health of the City of New York since August, 1978.

Since 1972, I was a Consultant to the Environment Directorate, Organization for Economic Co-operation and Development; since 1976 served as DOE (formerly ERDA) Representative in the U. S. Delegation to the Environment Committee and U. S. delegate to the Joint Environment-Energy Steering Group. I was a member of the United Nations Environmental Program (UNEP) International Panels of Experts on the Environmental Impacts of Production, Transportation, and Use of Fossil Fuel 1978, on the Environmental Impacts of Nuclear Energy 1978-79, on Renewable Sources of Energy and the Environment 1980, and on the Comparative Assessment of Environmental Impacts of Different Sources of Energy, 1980. I was a member of the Beijer Institute, UNEP, and USSR Commission for UNEP International Workshop on Environmental Implications and Strategies for Expanded Coal Utilization, 1980.

I am currently a member of the U. S. Department of Health and Human Services, Public Health Service Centers for Disease Control, National Institute for Occupational Safety & Health overview group, supervising the epidemilogical study of the employees at the Portsmouth Naval Shipyard where an alleged increase in leukemia was reported by Najarian and Colton in 1978, and a Consultant to the Division of Environmental Health, World Health Organization and the United Nations Environmental Program on the comparative health effects of different energy sources.

I have been Professor of Medicine, Department of Medicine, Health Sciences Center, State University of New York at Stony Brook, New York since 1968 and I am currently a member of the American Association for Cancer Research, American Society for Clinical Investigation (emeritus), American Association of Pathologists, Inc., the Harvey Society, and the British Medical Association.

I have published more than 100 scientific papers, including many reports assessing the hazards of various energy sources.

MR. CULP: Your Honor, I have a few questions to ask Dr. Hamilton on further direct.

JUDGE WOLFE: All right.

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BY MR. CULP:

Q Dr. Hamilton, your affidavit does not consider the non-cancerous effects of low-level radiation. Can you explain why that is so?

A Yes. That is simply -- it is because the Appendix I levels of radiation, cancer would be the main or the principal effect that we could calculate.

I believe that there's some misunderstanding about the risk of non-cancerous effects from Appendix I levels which arises from this quotation from the BEIR III Committee Report, the printed version, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation: 1980."

If you'll turn to page 478 of this document, there is a summary introducing Chapter VI, which is on "Somatic Effects: Effects Other Than Cancer."

MR. DOHERTY: Your Honor, may I see what the witness is going to read from?

JUDGE WOLFE: Certainly.

THE WITNESS: I'm going to read from this chapter of the BEIR Committee Report. I'm going to read the last sentence at this point, and I'm going to read several sentences.

Would you like a copy?

MR. DOHERTY: Yes, I would.

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THE WITNESS: I'm going to read from this last sentence here, I'm going to read from somewhere in the middle, and then I'm going to read from somewhere at the end. We don't have a complete ... but that will give you some idea, okay?

MR. DOHERTY: Yes.

THE WITNESS: And I'll reference the others, okay?

This summary deals with a number of other somatic effects, other than cancer -- developmental effects in the embryo or fetus, acute effects on the gonads, the testis and the ovary, and cataract.

And finally one sentence -- only one sentence -- on the life shortening.

And it says: "There appear to be no non-specific effects of radiation at low doses that lead to a shortening of life span, although the existence of specific effects in addition to cancer cannot now be excluded."

Now, unfortunately, that sentence carries with it, in rather extraordinary ambiguity, that as far as life-shortening effects are concerned, one might get the implication that the BEIR Committee isn't really excluding the occurrence of nonspecific life-shortening effects at low doses.

But in actual fact, if you go into the body of the report from which this summary is taken -- I'm sorry, Mr. Doherty, I don't have a copy of this, but I'll read it slowly.

On page 502, it says: "Lethal diseases have not been shown to be equally advanced by radiation; this suggests that the effects of such exposure are not directly equivalent to natural senescence. Although it is apparent that radiation advances the time of onset of some neoplaastic diseases, the only nonneoplastic diseases that have been shown to be accelerated by radiation are nephrosclerosis, which occurs only at high doses, and amyloid deposits in LAF mice."

That's capital L-A-F, that's a variety of mice.

"Mortality data statistically adjusted for competing risks by the method of Kaplan and Meir strongly suggest that nonneoplastic diseases are not advanced in time in animals exposed to radiation at doses that result in life-span shortening of less than 15%. On the basis of an empirical estimate of a 3-5% reduction in life span per 100 rads of whole-body exposure, no significant increases in the rate of induction of nonneoplastic diseases would be anticipated at doses of less than 300 rads."

And then if I go on to the conclusion of this

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paragraph -- of this chapter, sorry -- and Mr. Doherty you have a copy of this.

"There is no firm evidence that exposure to ionizing radiation causes premature aging in man or that the associated increased incidence of carcinogenesis is due to a general acceleration of aging. It may be concluded from the available data that ionizing radiation induces or accelerates some but not all diseases, depending on the genetic susceptibility of the subject and the exposure conditions."

Now, this is the crucial sentence: "For doses of less than approximately 300 rads of low-LET radiation, the principal mechanism of life-shortening is the induction or acceleration of neoplastic diseases. This conclusion is essentially in accord with that of the International Commission on Radiological Protection that the evidence of life-shortening from effects other than tumor induction is inconclusive and therefore cannot be used for quantitative risk estimates."

And then it goes on to say that "The United Nations Scientific Committee on the Effects of Atomic Radiation has taken a similar position that" -- and this again, is my emphasis -- "with the possible exception of high-dose exposures, life-shortening depends almost entirely on the induction of neoplasia."

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Now, I'd like to quote in actual fact from UNSCEAR, UNSCEAR-1977 -- and I have a copy of this for Mr. Doherty.

(Document handed to Mr. Doherty.)

MR. DOHERTY: Thank you.

THE WITNESS: Perhaps the Board would like a copy.

The paragraph I'm going to quote from is Paragraph 27 on Page 570 of the UNSCEAR 1977 Report. UNSCEAR is the United Nations Scientific Committee on the Effects of Atomic Radiation.

"Owing to inadequacies of the statistical treatment of the data, the conclusions of these early experiments were thus challenged and the non-specific life-span shortening attributed to a technical artefact (sic). Actually, other experimental series where appropriate death-rate analyses have been performed and where the effects of dose fractionation, chronic exposure and age at irradiation have been tested, have shown rather conclusively that the concept of a non-specific ageing effect of radiation is no longer tenable. At present, the consensus seems to be that life-span shortening is to be attributed almost entirely to the induction of neoplasia, especially at low doses and dose rates. A non-specific component may however become apparent in the high-dose range."

The point I'm making here is: There is no evidence that these life span -- non-specific life-shortening effects can take place at doses that one would consider to be Appendix I levels. They're all at 300 rad, which are thousands of times higher than the doses we're concerned with in the nuclear fuel cycle, and a thousand times higher than Appendix I levels.

And it's only the cancer risk that it's appropriate to include in this cost/benefit ratio.

BY MR. CULP:

Q Dr. Hamilton, would you turn to page 12 of your testimony. In the second paragraph on that page you refer to a final report of the U. S. Department of Health and Human Services. Do you know whether that report has been finalized or published, since the date your affidavit was filed?

A Yes. This is, of course, the final report. And it's the one that I gave reference to, Reference 15.

An article has appeared in the literature, and I'll give you the reference to that, by Rinsky, et al. The authors are essentially the same, although they occur in slightly different order.

Now, the title is "Cancer Mortality at a

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Naval Nuclear Shipyard."

It appeared in the "Lancet," L-a-n-c-e-t -- that is a British medical journal, Number 8214, page 231, 1981.

That was on Saturday, January the 31st of this year.

And, essentially, it comes to the identical conclusions that I have outlined in my report on pages 12 and 13 of my testimony.

MR. CULP: That's all the questions I have, Your Honor. The witness is available for cross-examination.

JUDGE WOLFE: Mr. Dewey?

MR. DEWEY: Staff has no cross-examination.

JUDGE WOLFE: Mr. Doherty.

CROSS-EXAMINATION

BY MR. DOHERTY:

Q Dr. Hamilton, on the study which you just looked at a minute ago with us called "Sources and Effects of Ionizing Radiation," the UNSCEAR Report. You read a paragraph from that on page 570, the last sentence of which was a "A non-specific component may however become apparent in the high-dose range."

And then it has parenthesis "227." Do you -- Was that high-dose range above 300 r-e-m?

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A. Yes. That's a reference to the work of Grohn on that -- you know, along the lines of the experimental work that I quoted from in the BEIR Committee, the BEIR III Committee Report.

Q. Yes. Well, I wanted to be sure what we have in mind.

Is it your testimony at this point that there should be no weighing of non-carcinogenic diseases in the cost/benefit analysis resulting from Appendix I releases?

A. Well, my testimony is that the -- all of the evidence we have for the non-carcinogenic effects of radiation, for their existence, would indicate that for each one of them, there is a threshold level which is considerably higher than the Appendix I levels.

So, therefore, they don't enter into the consideration. I mean, they're not something you would get, therefore, at Appendix I levels. And, therefore, they're not germane to -- There's no possible reason to consider them.

Q. Well, what diseases were studied for this -- Well, that may be incorrect. Was the approach to take a disease and see if there was a relationship; or was the disease to take exposed individuals and see if any peculiarly large number of non-specific diseases showed

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up, or non-cancerous diseases showed up?

A. No. There's a huge literature on the effects on radiation on both experimental animals and on people. And as a result of this, in addition to the possibility of the carcinogenic effects of radiation and the genetic effects of radiation, it has been found if you produce -- if you give animals or people enough radiation, you can produce a number of other specific effects.

One of them was believed to be life-shortening. And I have shown you the evidence that life-shortening, if it can exist at all, exists only with doses above 300 rad.

And even now there is some skepticism that there is something specifically that can be called life-shortening independent of the induction of a tumor. When you produce a tumor in an animal or person, it shortens their life.

The question is: Is there any other type of life-shortening?

Well, if there is, you don't see it with doses below 300 rad. That's the point.

That's point number one.

There are three other areas where people have described non-cancerous effects, again from doses that I maintain are above Appendix I levels and, therefore,

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there's no point in calculating the risk.

The effects on the embryo and the fetus, effects on the gonads -- the male testis and the female ovary -- and then effects on the lens of the eye. Of course, if you have a large enough dose of irradiation -- in a single dose, you can induce -- it can be fatal.

But, again, we're talking about doses that are thousands of times greater than Appendix I levels. And it's, therefore, inappropriate to me -- unequivocally inappropriate -- to sort of weigh them in any sort of NEPA-like rocedure because you're weighing something that has no -- zero weight.

Q. Well, you said there was a huge amount of studies. What specifically did they track down? What were the topics of these --

A. Well, if we're talking about effects on the gonads -- and this applies to animals, and also there have been some studies in human beings ... people gave animals, or in some cases human beings ... were given certain doses of radiation. And then the effects on sperm counts or the effects of the histology of the testis or the effects on the histology of the ovary, or the effects on fertility were measured as end points.

And from this it was deduced that you needed very large doses to produce some of these effects. And you cannot see these effects below doses of hundreds of rads.

That's the sort of study that is done and the sort of conclusion that is drawn.

There have been a number of studies on the fetus, different stages in the fetus in experimental animals. The fetus is probably one of the more radio-sensitive systems.

And there is some evidence for producing effects in the fetus in experimental animals down to five rad, a single dose of five rad.

But there's evidence, too, that protraction of the dose, of course, lessens the effect, as one would expect.

But there's nothing, even in the fetus which is the most radiosensitive of these systems -- there's nothing that produces any effect at Appendix I levels that we're talking about.

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Cataracts have been measured by direct observation, of course, in both man and experimental animals.

You give a certain dose of radiation and you see what it is that produces a cataract; but again, it's a question of hundreds of rads.

Q This disease or -- I think it was called life-shortening that you mentioned. Did anyone get to anything more than that?

I mean, life-shortening doesn't sound like a disease to me. It doesn't sound like you ever thought it was a disease.

It sounds like a statistic.

A. Well, the point about it -- let me explain as best as I can as a physician.

As one grows older, and we all do inevitably, there are certain changes in the tissues of the body. Some of them we associate with this process of getting older.

For example, arteriosclerosis, that is, you know, the gradual blocking up of tiny blood vessels, is something that one recognizes has a tendency to increase with age.

Of course, when you have arteriosclerosis in the heart, it will impair the function of the heart; and when you have arteriosclerosis in the kidney, it impairs the function of the kidney.

There are other things that will happen with age. Some people use their hair; that's another sign of age. Or your hair turns gray in the color, if you don't use certain formulas to maintain it.

I don't have to give you a lecture on the fact that we recognize certain changes that go on in the human body with age.

Now, people observed in experimental animals, anyway, that was the thing, that when they were radiated with large doses of radiation, it did appear in some experiments that -- and, of course, the animals tended to die early when they gave them large doses of radiation -- it did appear that there seemed to be this generalized aging effect.

But as people looked at the matter more carefully and they did more detailed histological studies, that is, studies of the sections of the tissues and things of this sort, on the whole the current thinking is, and it's certainly true, as I've indicated to you with doses -- there's still some query about what goes on when you have 300 rads or more in a single dose, but the general concensus now is that there's no such thing as non-specific life-shortening effects.

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The life-shortening that we see in experimental animals has been due to the induction of the tumor; and, of course, a malignant tumor induced by radiation curtails the animal's life by inducing fatality at an earlier age.

Q Okay. You stated on page 2 of the affidavit, which has become your testimony, that you reviewed Dr. Gotchy's affidavit and concluded it to be reasonably conservative.

Just what is your standard there? What do you mean yourself, "reasonably conservative" there?

A Well, what I mean by this is that Dr. Gotchy has used risk estimators, and I'm now talking about the thermatic effects, risk estimators, and particularly the carcinogenic effects risk estimators, and he did at this particular time derive them, I believe, if my memory is correct, from BEIR I.

But as I have indicated in previous testimony, risk estimators derived from BEIR I and NSCE in 1977 and BIER III all fall within the same ballpark.

Dr. Gotchy used the linear-linear relationship between the dose of radiation and the induced effect.

Now, this is a conservative calculation of the effect from low LET radiation, which is what we are concerned with here as far as this particular plant is

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concerned, low LET radiation, that's what the Appendix I levels are talking about, because the effects could be zero.

What he has given is an upper boundary, an upper limit of the effects, in his calculation. This linear-linear relationship which he's used and which I use is the, I believe, the most conservative way of calculating the damage.

It allows nothing for the very low dose rate at which these radiations would be carried out. There's nothing allowed for repair.

That's what I mean by reasonabl conservative.

Q Well, is this linear relationship, is it more conservative than a threshold type?

A Oh, yes, because a threshold type would assume that before you saw an effect, you've got to reach a certain dose.

Let's assume that the threshold says that you've got to have at least a level that's at least a hundred millirem or something of that sort.

That means that people would have to have a hundred millirem plus the other thing before you saw an effect; whereas, essentially, he has calculated the risk down to zero, as a matter of fact. Every dose has a slight risk. It's an incremental effect, so it's much

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more conservative than a threshold.

Q. But if you say every dose has a slight risk, you are kind of like -- it's almost philosophical, right? It's just that there's no way to say there's no risk. Is that right?

A. No, but it's very tiny. For example, this calculation that the Board would have in their ruling, 0.2 percent value that he calculated for the lifetime risk, I've recalculated that and I find that very conservative when I recalculated it using the BEIR estimate. I think the risk is -- if I've got my zeroes in the right place, it's .009 percent, not .02 percent.

So the value of .2 percent that Gotchy, I believe, had in his original affidavit is ultra-conservative. Reasonably is an English euphemism from my point of view.

Q. Say again? I didn't hear what you said.

A. When I said, "Likewise I find that affidavit of Reginald L. Gotchy concerning the NEPA impacts of low level radiation to be reasonably conservative," I was using the word reasonably in the English understatement of the word.

Q. But you also said in being conservative, there was no allowance for repair?

A. No, no.

Q. Didn't you say that?

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A No. You misunderstood what I said. I'm sorry.

Q I'm sorry.

A I must have failed to make myself clear.

The linear-linear relationship, when I say it makes no allowance for repair, it makes no allowance for the effect being a lot less because of repair.

We know that a dose of radiation given over a short period of time, when it's spread out over a long period of time, when it's fractionated and protracted, it's much less effective.

It produces many less tumors or much less of any particular effect, and that is because there is repair of the radiation damage.

So that when I say he is not making any allowance for repair, I'm using that as an example of his conservatism, because in actual fact, the numbers of effects would really be a lot less than those he calculates, if he were to make allowance for repair.

JUDGE LINENBERGER: Perhaps if you had said not taking credit for repair, that might be --

THE WITNESS: Very good, sir. I'm very happy to --

JUDGE LINENBERGER: It's just a semantic thing.

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THE WITNESS: -- be guided by Judge Linenberger. That's very helpful.

BY MR. DOHERTY:

Q Is there any literature here that you've included -- I don't recall any on repair, any citation in here.

I mean, how do we know that it's actually so that there is such a thing as repair?

A Well, I think I alluded to the repair, I believe, and I can refresh your memory, in my original testimony when I compared the health effects of the nuclear and coal fuel cycle, because I explained why I used the linear-linear model myself and I explained how this was not taking credit for repair and, therefore, was a reasonably conservative way of calculating the risk.

I again -- I don't want to burden you with further reading, because I know you are an avaricious reader, but there is in NSCE '77 an extremely good review of repair, an excellent review.

Q Okay.

A This would be absolutely a very good investment for you, from my point of view.

Q Well, when you talk about spreading the dose out, let's imagine a dose of 50 rems, you are saying if that were 50 one-rem doses, that would not be as severe

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as one 50-rem; is that that type of --

A. Correct. Correct. Absolutely.

Q. You are not saying 50 50-rem doses?

A. No. No. I'm just saying that 50 -- if you were to take -- just to be practical.

You give an individual a whole body dose of 50 rem, and that's sometimes done in radiation therapy. In all probability you would have some what's called radiation sickness for the first few hours after that single dose.

The next day, by the way, interestingly enough, you could give them another dose of 50 rem, and they would have adapted to that.

But if you were to give the same person 50 one-rem doses over 50 days, there wouldn't be any chance at all of them having any reaction whatsoever, and the effects would be, you know, completely diminished.

I think this is very important to keep in mind about these Appendix I levels, because we are talking about five millirem spread over a whole year as being the maximum dose, and that's a miniscule dose, absolutely miniscule.

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BY MR. DOHERTY:

Q Now, you state on page 2 that " ... calculations of health effects must be based on risk estimates." Is that your preference, or is that a kind of a statement of the limits of how health effects can be done at all?

A. Well, I think it's -- Yes. If you don't have a risk estimate, you can't calculate the effects.

Q Is there any research that you've seen that perhaps is contrary to your position here that doesn't use risk estimates? Or does everyone do that?

A. No, I don't see any ability to make any quantitative calculation on health effects between any agent and an effect, unless you have a risk estimate.

Q Uh-huh. Well, the two items you list in the next sentence, which is the most difficult to reliably quantify in -- well, let's put it this way.

In the studies, have they been more successful at determining the dose, or more successful in just determining a damage function? Which is the strongest link in the research typically?

A. Well, with radiation, of course, I think it depends on the epidemiological situation.

Q How about with experimental animals?

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A. I think it's more -- Well, obviously, it's very easy to -- when you're dealing with experimental animals to measure the dose. I mean, that's a very easy thing to do.

And the established damage function, of course, with experimental animals is not -- it's maybe just a little bit more difficult because you have to observe the animal and see the effect. So you have to keep the animals for a little longer.

And if you're thinking in terms of difficulty being the duration of the time of the effort, I think it's reasonable to say that you can more easily and quickly measure the dose in an experimental situation than you can the effect.

It's just that it takes more effort to keep the animals and watch for the appearance of the tumors or whatever result you're watching for, than it does to measure the dose.

Q. Well, in some of the human studies, did they -- did the authors set forth -- Did they just talk about cancer classifications in general terms, or did they attempt to relay or communicate --

A. No, in the human studies, if we're talking about the inductions of cancers --

Q. Yes.

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A. -- in humans, of course, there's a great deal of clinical diagnosis that has to go into the identification of the particular tumor that was identified in an individual.

So there's quite an enormous amount of labor that goes into identifying clinically the tumor and categorizing it, and looking after the patient with the tumor.

And, of course, there's a certain amount of effort that goes into determining what the dose was that the person got, depending on the situation.

Q. Is diagnosing a tumor very difficult these days?

A. Pardon?

Q. Is diagnosing a tumor a difficult thing to do?

A. Well, first of all, you've got to be a trained physician. And that, I think involves, as a physician myself, getting into medical school is very difficult in this country, as you know. That's the first difficulty.

Rather long training. And when we're talking about diagnosing a tumor for epidemiological things, that usually involves a number of people who are involved: a physician, possibly a surgeon who has done the actual surgery -- and that involves a tremendous amount of training for the person who actually did the surgery. And then, of course, you have to have the person who looks at the sample under the microscope; that's the pathologist.

And he has to be trained to identify the tumor. So I think that from the point of view of all the work that goes into this, this is a very -- the diagnosis and so on, it's a very difficult and laborious process, starting off, as I said, with getting into medical school.

Q. What about the reliability of death certificates and that sort of thing? Do you think that diagnoses are generally fairly accurate?

A. Well, we all know that there's a certain error in death certificates. But on the whole, I think they're pretty reasonable ... the cause of death.

Q. Would you fault any of the Shipyard studies or any of the Hiroshima studies for that at all?

A. Well, the Shipyard studies -- I'm only familiar, intimately, with one Shipyard study -- the Portsmouth Naval Shipyard --

Q. That's what I meant. I didn't mean to make that plural. I'm sorry.

A. Enormous labor has been done by the National

Institute of Occupational Health and Safety who track down, you know, whether there were people who died as a result -- not as a result, but coincidental with their employment at the Portsmouth Naval Shipyard.

They've actually looked at all these causes of death, those that have died, a lot of the death certificates. And they have a specialist. He goes under the extraordinary title of "Nose-ologist."

There's a whole art of verifying the diagnosis on the death certificate. And in certain cases they've actually gotten where they've been concerned about it; and if there's any doubt in their mind, they've gone actually back to the medical record of the individual concerned.

Q. So you feel that that's not a weakness in the different studies, that they are not tagging people with cancer when they didn't have it, or failing to see cancer when they did?

A. No, no, I don't think so. I think there's a great deal of labor that goes into -- particularly in the Portsmouth Naval Shipyard ... a great deal of labor has gone into being sure that the diagnosis that the person -- in this study that the person ... was alleged to have had was, in fact, the diagnosis that he really did have.

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Q About the Hiroshima studies though, some of those in Japan, they've been relied on, I think -- Have they been relied on very much in reaching these conclusions -- the Hiroshima study?

A Well, the Hiroshima studies are the studies that form part of the -- the Hiroshima/Nagasaki studies are studies that form part of our experience on what are the effects of high doses of radiation. The risk estimates have been in part drawn from the Hiroshima/Nagasaki studies and in part from these Englishmen, the ankylospondylitis of the spine, and in part lots of other people with various thyroid conditions were irradiated, and in part all sorts of other areas.

And the whole thing has been drawn together. And you want to ask me about the reliability of the Hiroshima/Nagasaki --

Q Perhaps the ankylospondylitis would be a better group, because they were at least not in war time, as I understand it.

A It was a group of peaceful Englishmen suffering from this rheumatic condition of the spine, who retreated with this, and then subsequently went on to develop an increased incidence of leukemia and other solid tumors.

But, of course, they had considerable relief

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from their ankylospondylitis as a result of the treatment.

Ankylospondylitis is a very painful disease, if untreated.

Q Though the -- Were the diagnositic abilities of physicians on that study, which -- I think it's 30 years old approximately now -- Well, when was the radiation done?

Maybe if you could give me a date, I could work a little bit on this. When were those people treated?

A. I believe they were treated in the thirties, if my memory is correct.

Q So there has been a follow-up since then?

A. Yes.

Q Would you say that the reliability with regard to diagnosing tumors and leukemia from that time -- at that time -- the thirties and forties -- is equal to today's or worse?

A. Well, of course, they were English physicians. And, of course, as an English physician, I feel they were impeccable at that time. Excellent. Very good.

I was trained in the 1940's in the United Kingdom at Oxford University. As a matter of fact, this study was done by -- on the ankylospondylitis was

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done Court-Brown and Dahl, both of whom -- well, Court-Brown is unfortunately dead. But I know Sir Richard Dahl very well.

And they are quite outstanding in the field of human epidemiology. If I criticize them at all, it's that they've been rather slow in producing really ... you know, exquisitely refined estimates of the doses, which the people got. But I think that they're reasonable ... reasonable estimates.

JUDGE LINENBERGER: I wanted to inquire about that very feature, the reliability of dosimetry at that time.

THE WITNESS: Well, I would say the following -- I don't want to in any way appear to be chauvinistic, but in the United Kingdom, as far as health physics was concerned -- I think you'll find that it existed -- it preceded the United States by at least 20 y ars.

People were paying far more attention to radiation, particularly for therapeutic purposes in the United Kingdom, in the thirties and the forties whereas it wasn't until later that they began to pay attention -- the same sort of attention in this country.

JUDGE LINENBERGER: And how good was their dosimetry compared to modern-day?

THE WITNESS: Well, I think you could say that

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obviously things are better now than they were then.

But I think the doses were reasonable, and I think one could go back -- by the way, and re-examine them and find, and improve them, as a matter of fact.

JUDGE LINENBERGER: But I gather you're saying that dosimetry was not sufficiently unreliable to place any significant question on the results?

THE WITNESS: No, but I mean the doses were very -- I would say they're reasonably reliable.

JUDGE LINENBERGER: Okay, thank you.

THE WITNESS: I mean the British Health Physics had a tradition that I think preceded this country by 20 years ... very reasonable.

MR. DOHERTY: Your Honor, I have probably another hour and a half or so of cross.

JUDGE WOLFE: All right. We'll recess until 9:00 a.m.

MR. COPELAND: Your Honor, one housekeeping matter: Mr. Congdon was due to follow Dr. Gotchy tomorrow. He has stayed here two days and couldn't stay any longer and had to leave.

So instead of taking up the WIGLE Code issue tomorrow, we will go on to IGSCC on schedule.

Instead of taking up the issue on Doherty Contention 15, the WIGLE Code, we'll move to welding and then IGSCC.

JUDGE WOLFE: Welding?

MR. COPELAND: Yes, with Mr. Litton.

On the schedule you have it shows Frazar and Gunther. Frazar and Gunther, as you will recall, testified last week.

JUDGE WOLFE: Yes.

MR. COPELAND: It looks like to me it's very doubtful that we will get to anybody other than Mr. Litton, Mr. Gunther and Mr. Malec, after we finish with Dr. Hamilton and Dr. Gotchy.

So I'm just assuming we will not get to any of the witnesses listed for October 30.

JUDGE WOLFE: Yes.

MR. COPELAND: So I have not had them come, they're all from out of town.

JUDGE WOLFE: All right, 9:00 a.m.

(Whereupon, at 7:10 p.m. the hearing was adjourned, to reconvene on Friday, October 30, 1981, at 9:00 a.m.

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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 29, 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)