UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMSSION

- - - - - - - - - - - - - - - - - - - -x
:
In the matter of: :
:
METROPOLITAN EDISON COMPANY : Docket No. 50-289
: (Restart)
(Three Mile Island Unit 1) :
:
- - - - - - - - - - - - - - - - - - - :

25 North Court Street,
Harrisburg, Pennsylvania

Thursday, December 18, 1980

The evidentiary hearing in the above-entitled matter was resumed, pursuant to adjournment, at 9:03 a.m.

BEFORE:

8012310014

IVAN W. SMITH, Esq., Chairman,
Atomic Safety and Licensing Board

DR. WALTER H. JORDAN, Member

APPEARANCES:

On behalf of the Licensee etropolitan Edison Company:

GEORGE F. TROWBRIDGE, Esq.
THOMAS A. BAXTER, Esq.
DELISSA A. RIDGWAY, Esq.
Shaw, Pittman, Potts and Trowbridge,
1800 M Street, N.W.,
Washington, D. C.

On behalf of the Commonwealth of Pennsylvania:

ROBERT ADLER, Esq.
Assistant Attorney General,
505 Executive House,
Harrisburg, Pennsylvania
WILLIAM DORNSIFE,
Nuclear Engineer

On behalf of Union of Concerned Scientists:

ELLYN WEISS, Esq.,
ROBERT D. POLLARD
Harmon & Weiss,
1725 I Street, N.W.
Washington, D. C.

On behalf of the Regulatory Staff:

JAMES TOURTELLOTTE, Esq.
JAMES M. CUTCHIN, IV, Esq.
Office of Executive Legal Director,

United States Nuclear Regulatory Commission, Washington, D. C.

Petitioners for leave to intervene pro se:

STEVEN C. SHOLLY, 304 South Market Street, Mechanicsville, Pennsylvania

On behalf of ANGRY:

GAIL BRADFORD

C O N T E N T S

WITNESS:	DIRECT	CROSS	REDIRECT	RECROSS	BOARD	CROSS ON BOARD
James H. Conran						
By Mr. Baxter		8669				
By Dr. Jordan					8670	
By Ms. Weiss				8691		
By Mr. R. Adler				8709		
Walter L. Jensen Jr.						
By Mr. Cuthchin	8710					
By Mr. Weiss		8720				
By Dr. Jordan					8731	
By Ms. Weiss						8735
By Mr. Cuthchin			8737			
Afternoon Session p. 8744						
Gary T. Urquhart						
James H. Correa						
Robert C. Jones						
By Mr. Baxter	8744					
By Mr. Dornsife		8763				
By Dr. Jordan					8768	
By Chairman Smith					8776	
By Dr. Jordan					8777	
By Mr. Dornsife		8782				
By Dr. Jordan					8783	
By Mr. Cutchin			8786			
By Mr. Jordan					8787	
By Mr. Dornsife						8792
By Mr. Baxter			8795			
By Dr. Jordan					8797	
By Ms. Weiss		8797				
By Mr. Baxter			8817			
By Ms. Weiss				8818		
By Mr. Dornsife				8818		
Walton L. Jensen, Jr.						
John J. Zudans						
By Mr. Cutchin	8820					
By Dr. Jordan					8826	
By Mr. Dornsife		8830				
By Dr. Jordan					8833	

C O N T E N T S

WITNESS:	DIRECT	CROSS	REDIRECT	RECROSS	BOARD	CROSS ON BOARD
By Mr. Baxter						8841
By Mr. Dornsife						8841
By Dr. Jordan					8842	
By Mr. Dornsife					8844	
By Dr. Jordan					8844	
By Ms. Weiss		8845				
By Mr. Baxter		8853				
By Mr. Dornsife		8854				

EXHIBITS

NUMBER	FOR IDENTIFICATION	IN EVIDENCE
USC 21	8707	
USC 22	8707	
USC 23	8707	
USC 24	8708	
USC 25	8708	
USC 26	8708	
USC 21-26		8709
UCS 2&3		8746
USC		8826

CHAIRMAN SMITH: Before we begin with the testimony, I would like to take advantage of the opportunity with both Mr. Tourtellotte and Mr. Trowbridge here to discuss the possibility of setting before too long a session to discuss perhaps a departure from traditional timing of filing proposed findings, with the idea being that typically at the close of a record boards maynot be as productive as they might be until proposed findings come in.

We would like to explore possibilities of having proposed findings come in perhaps in stages so we would have findings to work on immediately after the close of the record.

For example, it might be possible to have proposed findings by the licensee and the staff if they wish immediately at the close of the record on the procedural background.

That way we could give the parties the opportunity to look at it and see if they have any problems with it and maybe just adopt it; perhaps maybe even the licensee and the staff could agree upon proposed findings on procedural background,and maybe the board could adopt it exactly as submitted, which would free us to do some work on the substantive matters.

And then perhaps some of the substantive proposed findings could be submitted rather soon after the close of the record depending on when they are heard; the problems particular intervenors may have in presenting proposed findings; for example, I would think on Mr. Lewis's contention that since he only has one contention that a shorter period of time might be set aside for him to file proposed findings.

Then UCS with all of their contentions, for example.

On the other hand, consideration may be given to the fact that UCS's case closes earlier in the proceeding. That is the idea, and we invite recommendations and analyses.

But the basic idea is to keep the board as efficiently productive at the close of the record as possible.

One that occurs right off the bat is if we are happy with the proposed findings with respect to the procedural background, that could start right away in tuping production; that could just get that out of the way, and we could be free to work on the substantive aspects immediately.

That is the invitation the board is extending and we invite your thought and recommendations on it.

MR. TOURTELLOTTE: The staff has already started such a program. We had anticipated it might be possible to file partial proposed findings after each discreet segment of the proceedings is completed.

I realize that perhaps there is a chance that there may be one or two issues outstanding, fo instance, on plant design and modification. But for the most part, for example, it may be done by the end of January.

It seems to me that probably within 30 or 45 days after that, we could have partial proposed findings on at least as has been totally resolved of that segment of the case. It seems to me it would also benefit the board and the board could either start directly or indirectly.

We have already done the procedural part of it. At least it has been done and it has been submitted to me for review.

CHAIRMAN SMITH: We had hoped to have the procedural part of it done, too, as we went along, but it never came to pass. We were never able to do it. And of course another factor has to be considered, and that is, as I recall, Commission rulings and perhaps express language of regulations, which I cannot identify --I think all parties have a right to file proposed findings on all issues.

That would create a timing problem. However, that right also -- the timing of that can be scheduled. But I believe any intervenor can file and the Commonwealth can file proposed findings on any issue, notwithstanding whether that issue was reflected in their contentions or not.

If my impression of the procedural law of the Commission is correct, that would have to be taken into account in what I am proposing.

In any event, that is what we are thinking. I am not aware of any case where this has been done before, Mr. Tourtellotte.

I am pleased that you are thinking along those lines. We would certainly invite it.

MR. TROWBRIDGE: Mr. Chairman, we intend ourselves to not wait until the end of this proceeding to draft findings. I welcome the suggestion especially that we get together, and perhaps get together with the staff if we can on common procedural findings.

Perhaps I will have a discussion with Mr. Tourtellotte and perhaps we can come back to the board with a concrete proposal.

CHAIRMAN SMITH: That would be fine. Anything else before we begin?

MR. TOURTELLOTTE: Yes, Mr. Chairman, one preliminary matter, which is actually referring back to the 9th and 10th of December. I was reviewing the transcripts, and something occurred to me that I thought was possibly something we should not let pass or at least the staff felt it should not let pass without some comment to be sure that we are not establishing a precedent.

It concerns the letter which was sent to the board by UCS. The letter directly quotes a transcript of the Commission. I simply wanted to bring to the attention of the board section 410 CFR 9.103, which strictly prohibits the citation of such transcripts.

I did not want it inferred by reason of the fact that we did not mention it a that time that we in any way condone or acquiesce to the citation of the transcript by UCS.

It is contrary to the regulations.

CHAIRMAN SMITH: Would you give me that section, please.

MR. TOURTELLOTTE: 9.103. What it says about half way down is that statements of views or expressions of opinions made by the Commissioners or NRC employees at open meetings are not intended to represent final determinations or beliefs. Such statements may not be cited or relied upon before the Commission or in any proceeding under part 2 of these regulations, 10 CFR, Part 2. Except as the Commission may direct.

While I understand the concern that UCS may have had in that instance, it is nevertheless a strict violaton of the regulations to bring the matter up in that procedural fashion. How it may have been otherwise--I think it is possible to bring it up otherwise without violating the regulations.

MS. WEISS: I do not interpret the way in which we used the transcript reference as a violation of that regulation. I believe that regulation refers to citing such statements either as evidence as an expression of Commission opinion.

As the board is well aware, we cited it only because it was said: not for any evidentiary value. We also had people in that meeting who heard that comment. Rather than state from their memory, we believed it is better to state it from the transcript.

I do not believe that is a violation of that regulation.

MR. TOURTELLOTTE: Of course, I have given that some consideration, but I view that position as really untenable becaue basically UCS goes on after making that citation to ask for some kind of specific relief from the board.

They are asking the board to do something, and they are basing it upon that quote. Now, whether it is a letter of a formal pleading really does not make that much difference.

CHAIRMAN SMITH: I was of course aware of that section. And as Ms. Weiss, I thought of it in terms of its application in substantive matters. But as I read it, I see such limitation in the language on it.

I can appreciate your position on it. There is also another aspect of it which, when I addressed it orally a few days ago, which I did not give very thorough consideration to, and that is general counsel also has an interest in your motion, both in -- as the general counsel -- general counsel might view it as his right to seek counseling without violating the ex parte rules of the Commissin on procedural matters anywhere he can find it.

And I do not want to intrude upon any privilege he might feel. Nevertheless, I still stand behind the remarks that I made, that I have had no communication with any member of the Office of General Counsel on any substantive matter in this or any other proceeding as far as that is concerned.

It is thoroughly established practice and rule that there be no such communications, and I continue to stand by it. I think the point you are making -- notwithstanding section 9.103 -- should be addressed. I had not really taken into account that problem, that the OGC might have some feelings on it, too. It is not just something that is within our prerogative to dispose of entirely.

I wanted to bring that to your attention. I have had no communication from OGC that they feel that way; it is just something that occurred to me.

MS. WEISS: The letter to this board was sent out from my office while I think I was up here; I am not sure. I do not know if a copy of it went to the general counsel. I did communicate directly with the Commission, however, about the Indian Point proceeding.

CHAIRMAN SMITH: With respect to --

MS. WEISS: With respect to this comment as w ll as others made at that meeting. That certainly did go to the general counsel, so general counsel is aware of our concern about those particular words.

It may not be specifically aware that we wrote a letter to this board, although we did say in the letter to the Commission on Indian Point that we intended to communicate with this board about the subject.

CHAIRMAN SMITH: Referring to the actual communication you cent?

MS. WEISS: Yes. We sent a letter t the Commission first on Indian Point. It seemed as if they were ready to make a decision on procedural aspects of that, and in that letter we brought up these comments and stated in that letter that we did intend to communicate directly with this board.

CHAIRMAN SMITH: There is one other aspect, and that is I told you that I would give you a preliminary report, which we did, a thorough report over the Christmas vacation. I have promised so many things for that Christmas vacation that it is my inclination -- I will give you an opportunity to object -- but I will not even get around to it until the decision in this case issues.

I make that based upon the representation to you that I have had no communication with any Commission staff level person, suggesting a feeling on any of the issues in the case.

If you have information or beliefsa or perceptions to the contrary, I think you are going to have to bring them to my attention so I can express them.

I do not think I am going to have the time over the Christmas -- the days that we are out of hearing to address it. Unless you object and persuade me to the contrary, I am going to defer the whole thing until after the initial decision issues.

MS. WEISS: Based upon what you said orally in the hearing, I have no objection to that.

Before we begin, I would like to approach the bench for a couple of minutes, if that is all right.1

CHAIRMAN SMITH: All right.

CHAIRMAN SMITH: Let's constructively approach the bench -- constructively at some other place than at the bench.

(Bench Conference)

CHAIRMAN SMITH: Anything further?

(No Response)

Mr. Baxter has some questions, some cross.

Thereupon,

JAMES H. CONRAN

the witness on the stand at the time of recess, resumed the stand and was examined and testified as follows:

CROSS EXAMINATION (Resumed)

BY MR. BAXTER:

Q You have been with the staff for a number of years. Do you have a general familiarity with the role that a project manager performs in processing applications for construction permits and operating licenses on the staff?

A Yes.

Q Would you say that a project manager's main responsibilityis coordinating and planning the detailed technical review that is performed by other members of the staff?

A That is certainly a big part of the job. I would add understanding, understanding the technical review work that the staff -- that the experts in the technical review branch have done.

It is not just a bean counting, scheduling job. It entails understanding inputs made by people who are experts in narrow areas and taking their safety evaluations and understanding them and tying them together in what is called a safety evaluation report.

I do not think it is a complete characterization; I think it is a big part of the job: coordinating, planning, yes, but understanding is at least as big a part of the job.

MR. BAXTER: Thank you. Those are all my questions.

BOARD EXAMINATION

BY DR. JORDAN:

Q Mr. Conran, do you have a copy of Mr. Pollard's testimony on UCS 14?

A I think so; it will take me a moment to get it.

(Pause)

I am sorry. It looks like it might take me longer than a minute to get it.

Q No hurry.

(Pause)

CHAIRMAN SMITH: I will loan you mine if you cannot find it, and then I will share Dr. Jordan's with him.

THE WITNESS: I believe this is it here. Yes, this is it. I have it.

BY DR. JORDAN:

Q Would you first turn to page 1 of the testimony which begins UCS contention number 14. Do you have that?

A Yes.

Q I would like to go through this contention and see if you agree or disagree with the statements made.

Before I do, in order not to get involved in a harangue or problem with definitions, I am going to talk about safety grade and nonsafety grade. I will assume that other items which are important to safety are nonsafety grade at the moment unless they are specified as safety grade.

The first sentence there, Mr. Pollard says, "The accident demonstrated that there are systems and components presentlyclassified as nonsafety related which can have an adverse effect on the integrity of the core because they can directly or indirectly affect temperature, pressure, flow, and/or reactivity."

Do you agree with that statement?

A I do. I have agreed with it in my testimony explicitly.

Q "This issue is discussed at length in section 3.2 of NUREG-0578. The following quote from page 18 of the report describes the problem."

Then follows the quotation.

Now, do you agree with the statement from NUREG-0578 as quoted by Mr. Pollard in whole, or do we need to go through it line by line?

A I subscribed to it initially on our Lessons Learned Task Force when we published the report with those words in it. I would only say that some difference in the staff's perspective on those words may have evolved subsequent to that.

We have had longer to reflect on it and to digest the results. I think I still agree with the words here, though. If we got very specific about one or another word or phrase, in general I accept --

Q All right. If it turns out that you would like to as a result of further answers to questions on my part, if you would say, well, I would like to modify that line or something like that and go back to it, that is fine.

At the moment, let's continue.

The next sentence says, "The staff proposes to study the problem further."

Is that a true statement?

A I think it is a true statement as far as it goes. I do not think it goes far enough. We are not proposing to just study the problem.

Q You plan to do even more?

A We have already done more, and we propose to study even further.

Q All right. So therefore I guess that you say you would also therefore agree with the next part of the sentence: this is not a sufficient answer, in that you intend to do more than study the problem.

A Only in that sense would I agree that it is a correct statement. I do not think it accurately -- it implies something that does not accurately characterize the staff's posture or their program.

Q There is only one sentence left.

Do you agree or disagree with that? Let's read the sentence so it will be clear in the record: "All systems and components which can either cause or aggravate an accident or can be called upon to mitigate an accident must be identified and classified as components important to safety and required to meet all safety grade design criteria."

Now, then, there may be a slight problem with components important to safety, but by -- that may be a problem, but I suspect not.

A It is not for me; it is not a problem for me.

Q All right. Then do you agree entirely with the sentence, or do you have any reservations?

A I almost entirely disagree with that sentence. I thought some more last evening. I think it was Mr. Adler's or Mr. Dornsife's question in this regard yesterday.

I think -- I think my agreement with the point that he was pursuing and in fact Mr. Keaten, I think, has made the same point in his testimony -- I want to more stronglysupport the thought that they had in mind.

I think it certainly should be said that I profoundly disagree with this statement. I want to context a few additional remarks that I am going to make.

Q Let me first find out in what respect all systems and components which can either cause or aggravate an accident or can be called upon to mitigate an accident must be identified.

Would you say that is true?

A I would say identified.

Q So it is the part that they must all be classified as safety grade. That is your disagreement; is that correct?

A That is the part that I choke on.

Q All right. I think we have all heard pretty much the reasons for it, but if you would like to summarize the reasons, that is fine.

A I think it is important to give the right context.

On a number of different issues over the last few years on which Mr. Pollard has raised objections or criticisms to the staff's way of doing things in their regulation of reactors, I very frequently find myself in his direction if not in his corner. So it is not

Q A very understandable position.

A It is not that I think of Mr. Pollard as an extremist, that I disagree so profoundly with this particular formulation of -- latest formulation of his thoughts. I think it is awful important in that context to acknowledge his contributions in the area of development of standard, influencing the thinking of safety reviewers, safety regulations in the areas that he is most qualified in, instrumentation, control systems, that sort of thing.

But in saying this Contention, he has simply gone too far. Mr. Pollard in effect, if he is completely serious about this Contention, would automate and complicate and interlock and upgrade nearly every if not every system in the plant.

Q And therefore there must be a large number of systems that can be called upon --

A The things that can initiate or aggravate or quite possibly be called upon to mitigate an accident include, I am afraid, nearly everything, nearly every system in the plant.

Q I do believe Mr. Pollard would agree.

A That notion, that proposal would disturb me enough if he were simply talking about applying it to new designs, things that you could still design and build, but to contemplate applying that proposal to existing reactors is scary. If you get half a centimeter beyond the saying of it and you think about the doing of it, it involves a kind of modification, and cutting wires and pulling cables and cutting piping and rewelding, and with the best of intentions, I think it would be potentially dangerous.

The first thing that comes to mind is the Crystal River event. I feel some pangs about what happened down there. I was a party to the recommendation, and I think it was a very good one, that saturation meters be applied to reactors. In the process of doing that, a simple mistake was made that resulted in a subtle kind of interaction that led to the Crystal River event.

You could multiply that sort of thing by at least three orders of magnitude if you did what Mr. Pollard is contemplating doing to operating facilities, and if he ever succeeded in making his oint and it actually were implemented, it is not an exaggeration, and I am very serious in saying that for reasons of personal safety, I don't think I would want to go near the monster that would result from applying this Contention in a straightforward manner.

CHAIRMAN SMITH: Would you be more sensitive to Dr. Jordan's efforts to interrupt?

THE WITNESS: Yes, I'm sorry.

BY DR. JORDAN: (Resuming)

Q We will get back to this in somewhat more detail. You will have a chance. But would you say, then, that at the moment your problem is with the words "all systems and components?"

A Yes.

Q Would you concede that there may be some systems and components that should be identified and upgraded?

A Yes, I would agree with that. That's not a concession. I already agree with that, and I think our programs indicate that.

Q Perhaps, then, we have agreement with the Contention excepting for the word "all" in the last sentence.

All right.

Let's go on now and try and narrow the area whereby the word "all" does violence to some of your opinions concerning safety.

I have a few questions on your testimony which may bear on that topic. In fact, I guess on page 13 and 14 of your testimony, we will get into some of the systems which in some instances I am sure Mr. Pollard would have included, and perhaps we can see where areas of disagreement therefore do lie.

However, before doing that, there is a matter of clarification on page 14, in answer to question 18. Question 18 was "Does the staff have any long term plans or programs for evaluating possible safety effects of non-safety systems or components generally, and for reassessing the appropriateness of current non-safety classifications in view of the lessons learned from the TMI 2 accident?"

Your answer is, "Yes, that was an explicit objective of Recommendation 9 (Review of Safety Classifications)."

What are you referring to there?

A That is the final Lessons Learned Task Force Report, NUREG-0585.

Q I see. All right. It wasn't clear.

And then you go on to some other reports which we will perhaps get to.

Now, getting back to the items which you say have been upgraded but not to full safety grade, let's for the moment look at Item 4 on page 14 in which you say automatic initiation of auxiliary feedwater system, short term, long term requirement, to provide safety grade initiation.

Was there -- was it deliberate that you restricted your answer in the long term for safety grade initiation, that you intended to leave out control of the auxiliary feedwater system?

A That was really an explanatory parenthetical comment, to make sure that I didn't misrepresent the recommendation that the task force made. It was broken into two parts. On a short term basis, we thought that it was appropriate to upgrade in every way that could be -- that was practical on the time scale that was being contemplated in the NTOL plants, Sequoia, North Anna, Salem, the plants that were right on top of us in the licensing process, in other words, that it would be appropriate to do an interim sort of upgrade in those plants, if it was necessary.

For the long term, they should be full safety grade.

Q Both for initiation and control?

A I'm sorry, maybe I missed part of your question.

Q That's the point I am getting at. In the long term, you limit it to initiations, as to upgrading the safety grade initiation of the auxiliary feedwater, but not the control of the auxiliary feedwater. Should that be safety grade?

A I didn't venture into that area because, as a task force, for example, we had not thought about, we had not addressed the control area.

I think the staff's position is on that yes.

Q How's that?

A I think the staff's position on that is yes.

As I understand both the staff's position and the Licensee's position, that is going to be true.

Q So therefore you agree with Mr. Pollard with respect to Item 4, anyhow, that in the long term it should be upgraded to safety grade.

A Yes, but with this reservation. I wouldn't come to that conclusion on the basis of the lessons learned from the TMI 2 accident. But I think it is appropriate. I don't quarrel with the staff's plan to do that.

Q Would you come to that conclusion on the basis of a systems interaction study? Has there been any systems interaction study that would require such a commitment or conclusion?

A There hasn't been a systems interaction study in the sense of that word. A good deal of attention has been paid in what are called reliability interim, reliability looks at the AFW system. There has not been a final, systematic look at the AFW system that I am aware of. I think the IREP programs are approaching that on a much more formal basis, the Interim Reliability Evaluation Program. I think it is six units on which the studies are being done currently, or are contemplated, at least.

Q Do you know what those six units are, just as a matter of information?

A I don't know if I could tick them all off. I know one is Calvert Cliffs, for example, because one of the members of our branch is sitting in on and assisting in that program.

Q I see.

Are there any B&W plants on the list besides Crystal River that you know of?

A I don't know. I just can't recall at the moment.

Q I gather you feel that there should be in the long run a systems interaction study which would possibly confirm the need for, in this case, for safety grade equipment.

A Yes.

Q The next item down was auxiliary feedwater flow indication. I guess again I was wondering why you pulled out just the indication and not control.

A Again, it's because just the way, in the aftermath of TMI 2, the staff's various studies and investigations developed historically. There was already a great deal of work and attention being paid to auxiliary feedwater systems by the Bulletins and Orders Task Force and by at least one other group. So when the Lessons Learned group was assembled, the decision was made not to look at auxiliary feedwater systems unless there was something about the treatments that had been given the AFW systems that stood out in someone's mind where we could suggest additional improvements that had not already been identified. There had been many of them identified by that time.

Q In both 4 and 5, you refer to NUREG-0680. I guess I was a little concerned that you were basing your testimony on 0680 rather than your own evaluation, say, when the situation is much broader.

A Oh, yes, it is. As a criticism, just as to the format that the information is presented in here, I would accept that criticism. NUREG-0680, however, references very explicitly the appropriate sections of the work that I was involved in.

Q Does NUREG-0860 recommend full safety grade for the auxiliary feedwater system in the long run, in the long term?

A I have looked for that implication or that statement. I couldn't come to the conclusion out of 0680 that that is what was intended, full safety grade at some time in the future -- that is why I have asked people who are directly involved and have been for some time, Mr. Wermiel, for example, what is the staff's position with regard to whether or not the AFW is safety grade.

Q It has been brought up many times in this hearing. You still don't have a good answer that you feel comfortable with yourself.

A I wouldn't, and that is why I wouldn't try to give you a definitive answer on it. I think I know what the staff's position is with respect to whether or not the system is safety grade, and I have heard the Licensees say that they consider the system safety grade with the caveats that I mentioned. I don't know if or where the staff has required that it be safety grade. I have looked for it. I have not been able to find it.

It seems an obvious point that I should have gone to somebody and checked with before I came up here, but I didn't really expect to discuss the AFW system.

Q I have also looked for it, and I thought possibly you could help me on that.

Let's look now at the first three items that you quote in answer to Question 17, emergency power supplies for pressurizer heaters.

Is that a case -- I guess you are saying, I believe, that this is a non-safety component or system which hasn't been fully upgraded.

In what respects are the emergency power supplies for the pressurizer heaters not safety grade?

A Do you mean as --

Q What did you mean?

A The idea was -- this, of course, was not applicable just to TMI 1, but if in the designs -- and I think it is true -- I shouldn't even say I think it is true of most designs -- but to the extent that at the time of the TMI 2 accident the pressurizer heaters on any reactor plant, the power supplies for those pressurizer heaters were not Class 1E, that is on failure of offsite power, they could be connected to the diesel generator, the buses that are supplied by the diesel generators, if that was not in fact part of the design, then our recommendation was that you do at least that, improve the reliability of the power supply.

Q In the case of TMI 1, in what respects are the power supplies not upgraded to full safety grade, and -- well, let's concentrate on that part of the question first.

In what respects do they not have full safety grade at TMI 1 on emergency power supplies for the pressurizer heaters?

A I am not sure I can discuss that in great detail.

Q Is it because of your unfamiliarity with the --

A With the details of what is actually at TMI 1. I haven't looked at the ways in which -- the other ways in which those power supplies are not safety grade because as indicated here, we didn't think that they had to be full safety grade.

Q Your lack of familiarity with the full situation at TMI 1 I believe you explained is in part at least due to the fact that you had a very short time to prepare this testimony.

Does that give you yourself any concern with respect to the overall testimony on the amount of studies that have been done at TMI 1 on systems interactions?

We will come back to that. I guess at the moment I will just ask a yes or no.

Are you familiar with any studies on TMI 1 on systems interactions?

A No formal studies; I have heard Mr. Keaten's description of what he thought was implicit in that sort of activity. I think it is appropriate what they did, but that certainlyis not the extent of what we contemplate when our branch, for example, talks about doing systems interaction studies.

I think it should be pointed out that that is not necessarily a criticism because we have not required that sort of thing yet.

Q Let's consider that the -- your answers, then, are more generic than they are specific to TMI-1.

A Yes. To give a little more perspective on that, when the Lessons Learned Task Force made its final report in November of last year, I think it was, the group was then split up to several factions.

One group was given the responsibility of implementing the Lessons Learned recommendations in the context of the NTOL reviews that were coming down on us at that time.

Another group was assigned specifically to TMI-1 restart. So some people who were on the Lessons Learned group and others of course from various elements of the staff have been working on the TMI-1 restart review for over a year now; not because I was not interested in it, but because I did not have that assignment, I have been associated with that review for a couple of months now.

Q Let's go to the next item and consider it in a generic fashion, the emergency power supply for PORV and block valves.

Now, do you know what has been proposed for TMI-1 or proposed for -- generically for B & W plants? And do you know, therefore, in what respects it fails to be safety grade?

A I think the answer is the same. I know basically what the improvements that have been proposed are. I have read about them in the TMI-1 restart, but I do not know the ways in which they are not safety grade.

Q You don't know what?

A I do not know the ways in which they are not safety grade.

Q I see.

A Again, that does not particularly -- well, in this regard, I am afraid -- I do not want to leave the wrong impression with the board.

It does not disturb me in the same sense that memorizing the accident sequence didn't disturb me, and I do not think it affects the validity of my conclusions because other members of the staff who are qualified and competent have -- they know what the proposals are in detail and they have done the review.

I rely on other members of the staff and their work as a resource. I do not try to memorize it.

Q So you have confidence that the staff will really do a good job on that?

A Of course, yes. And I rely on the fact that they are doing that kind of a job. I take their conclusions and their work and in the context that I work in, I extend our integrate their conclusions.

I almost always make a point of gonig back and checking with them to see if I have missed some significant point;because I have not done these kinds of reviews in detail or because I cannot talk about the sequence, the accident sequence in great detail, that does not mean that -- I do not think it calls into question the validity of my conclusions.

Q One or two more questions only.

You and Mr. Pollard and I would say perhaps I have agreed that there is a need for systems interaction studies. You have also said, however, that as far as you know there are not any -- I believe -- no proposed for TMI-1 prior to restart, that you know of no studies --

A That is true.

Q I presume in the long term that there will be -- I believe that is --

A I do not know that, Dr. Jordan. I will tell you that the recommendation that was made by the Lessons Learned Task Force is the one I still prefer. Somewhere between the making of the recommendation and how it came out on the other end in the Action Plan, things got changed a bit.

Our recommendation was that all licensees be required to do systems interaction studies, and I think that is still appropriate. I did not approve the Action Plan. The Commission did.

There were a number of considerations involved in their deciding that things did not have to be done. Well, they have not speicified yet to my knowledge ever that all licensees have to do systems intereaction studies.

I think -- personally, I think it is appropriate that all licensees do systems interaction studies; systems interaction studies are not meant for the benefit of the staff.

They are meant for the benefit of the safetyof plants. The people who run those plants should know the most about them on a very detailed basis. I think theyhave the obligation to carry the major load.

It is for their information and the benefit of the safety of those plants, not the staff's benefit. I think the original recommendation is still the propoer one. The fact of the matter is the Commission felt differently for some reason.

The recommendation was modified. We may still get there. I am still involved in the process, and to the extent that I can influence what happens, everyone will do systems interaction studies on their plants. That is a personal opinion, but it is a strongly held personal opinion.

DR. JORDAN: I have no more questions.

CHAIRMAN SMITH: Ms. Weiss?

(Pause)

While you are preparing your notes, I had overlooked the fact that both for Mr. Pollard and for Mr. Conran, they had used diagrams on the chart board. I think that the procedure should be that that is fine. Use those. But then they should be reproduced.

They are simple enough; yours looks like a bloodshot eyeball, I think. Simple enough.

Reproduce it so it can be bound in the transcript. Would you ask Mr. Pollard if he would do the same thing with his?

We will bind them in the transcript when they are prepared at the same time so the record will reflect what the chart looks like.

MS. WEISS: It is actually the staff's turn for redirect.

MR. CUTCHIN: We can take care of that quickly. I have no redirect, Mr. Chairman.

RECROSS EXAMINATION

BY MS. WEISS:

Q I want to pursue a series of questions Mr. dornsife asked you yesterday.

Let me start with your question eight on page 6 of your testimony.

(Pause)

The question is: "has the staff identified those structures, systems, and components which must be safety grade."

The first two sentences of the answer are "Yes. They are listed in Regulatory Guide 1.29."

Mr. Dornsife directed your attention to item h, list of equipment in Regulatory Guide 1.29. That is cooling water and seal water systems or portions of those systems that are required for the functioning of the reactor coolant system, components important safety, such as reactor coolant pumps.

Do you know whether in current plants to which there is no question that this regulatory guide applies, whether reactor coolant pump, cooling water, and seal water systems are required to be safety grade?

MR. BAXTER: I am sorry, Ms. Weiss, the term "current plants" --

MS. WEISS: Any plant which applies now for a license; any plant to which Regulatory Guide 1.29 applies.

THE WITNESS: I do not have specific detailed knowledge, no. I cannot answer your question.

BY MS. WEISS:

Q You do not know?

A I can read the requirement and not know whether it has been applied properly or how it is applied to existing plants. I do not know.

Q Your testimony is that the staff has identified structures, systems, and components which must be safety grade and they must be listed in detail in RegulatoryGuide 1.29.

If it is the case that one of the systems listed in that regulatory guide is not required to be safety grade by the staff, isn't your testimony wrong?

A The staff has identified in detail in a regulatory guide in generic language the kinds of components and systems and structures that have to be safety grade.

With respect to plant specific identification of such structure, I did not mean to imply that it was in the regulatory guide.

Q No, no. I am not asking about plant specific. Just assume for the moment with me that the staff does not require on anyplant that reactor coolant pump, cooling water, and seal water systems be safety grade.

And that is not a question of interpretation. That system is identified clearly in Reg Guide 1.29. If they are not requiring that to be safety grade, then isn't your testimony, your answer to question eight wrong?

A No, no, it is not. It may go to the quesion of whether or not the staff is applying the regulations the way they should.

But the staff is in this document -- what I am saying is these are the requirements. If you are saying there is a discrepency between the requirements and the way that the staff has implemented the requirements, it is possible that there arethose kinds of discrepencies. I do not know of them.

Q Isn't it possible that it is not a requirement if they are not requiring any plant to have a safety grade reactor coolant pump seal water system, isn't the logical inference that it is not a requirement?

A I would not try to draw that logical inference.

Q You are telling me --

A If your interest is in the information that would answer your question, Ms. Weiss, what I have said was I cannot provide you that information.

If your interest is really in going into that degree of detail, someone else would have to answer the question.

Q I asked you to assume -- assume that the staff does not require that reactor coolant pump seal water systems be safety grade.

Then, isn't your testimony wrong in question eight?

Your testimony is: "The staff has identified the structures, systems, and components which must be safety grade. They are listed in detail in Reg Guide 1.29."

Isn't it correct that if this system which is quite explicitly called out in Reg Guide 1.29 is not required to be safety grade, then your testimony is wrong.

A I still would not agree with you evenif that were the case. I think that the staff has identified those systems that have to be safety grade.

If what you are suggesting is that there is a systematic slip up in the way these requirements, these criteria are met in plants, I cannot help you with that. I do not know the answer.

CHAIRMAN SMITH: Would you explain what you mean by the use of the word "must" in question eight? In the first place, I assume that you wrote that question?

THE WITNESS: Yes. "Must be safety grade" means required by regulation to be safety grade. I think maybe to carlify my answer to Ms. Weiss: if I were pursuing the same point she were and trying to make the point, my next question would be, has the staff assured in all existing plants that this requirement is met.

Then if I said yes and I did not know the answer to your question, I would say my testimony was wrong.

BY MS. WEISS:

Q I am going to pursue this more. I am not convinced that you are focusing on what I am asking you.

A All right.

Q You stated quite explicitly that Regulatory Guide 1.29 lists the components which must be safety grade.

A Yes.

Q Did you ask any project manager for any plant currently -- for any currently pending application whether he uses Reg Guide 1.29 as a list of all systems and components which must be safety grade?

A Yes.

Q And the answer you got was yes?

A I had very lengthy discussions with the project manager that is right across the hall from me who is currently involved in the Summer review, for example. He says, yes, I used this. We had lengthy discussions about what the various terms meant; what the implications of this, that, and the other were.

I did not focus on reactor coolant pump seals.

Q Now I am asking you to focus on reactor coolant pump seals.

A You are asking me to make an assumption.

Q And whether or not you agree with it, the rules of this question are that you assume it to be correct, that reactor coolant pump seals systems are not safety grade. The staff is not currently requiring them to be safety grade for currently pending applications.

Doesn't that mean that your testimony is wrong?

THE WITNESS: Is it true that I have to accept Ms. Weiss's assumptions?

CHAIRMAN SMITH: Yes.

THE WITNESS: On the condition that your assumption is correctly stated, I would have to agree with your statement.

BY MS. WEISS:

Q Dr. Jordan asked you about question 18 on page 14 of your testimony, which is a question about long term plans or programs that the staff may have.

Your answer references recommendation nine, review of safety classifications, and I believe your answer to Dr. Jordan was that that was recommendation nine, as contained in NUREG-0585, the final Lessons Learned document.

Is that correct?

A Yes.

Q Are you aware that that recommendation has been dropped out of the Action Plan?

A I thought I said a few words about that. It has not been dropped out; it has been changed so that a lot of people do not recognize it as such any more. It has not been dropped out.

Q Where does it appear?

A It is sort of scattered among the Action Plan item of IREP and whatever the extension of IREP is. there is a specific action called systems interaction; I think it is in the II.C section.

Q The II.C section is quite different from recommendation nine of NUREG-0585.

A I remember saying that myself.

Q Much less specific; is that correct?

A I would like to locate it, if I could.

(Pause.)

A What was your question again, Ms. Weiss?

Q Let me ask you one that may be more useful. You have looked at Recommendation 9 and you have looked at NUREG-0600. It is II.3, is the requirement that remains?

A II.c.3 is the one that specifically is labeled systems interaction.

Q Were you present in the room when I was questioning Mr. Keaton about -- I think it was Mr. Keaton -- about, specifically about recommendation 9 and the study that it calls for, and I asked him if the Licensee was performing such studies or believed itself required to perform studies in recommendation 9?

His answer was that they could not find a surviving requirement that they do those studies; is that your understanding?

A No, it wasn't. I think they found the surviving requirement. It is just that the schedule on which it has to be done and the ground rules under which it has to be done have changed considerably.

They made a recommendation that we thought went right to the point. Licensees should be required, all Licensees, for whatever the reasons, between the making of the recommendation and its acceptance and embodiment, incorporation into the action plan, obviously someone had second thoughts about whether the wording of our recommendation was appropriate and specific enough, whether it represented -- had enough thought been given to prioritization.

Implicitly, the answer, I think, was no, in recognition of the fact as further studies were completed in the aftermath of TMI-2, as the staff's recognition of the fact that operator error had in fact played a much more fundamental role in the core damage part of this accident than design problems --

MS. WEISS: Mr. Chairman, this is not responsive to the question.

THE WITNESS: It is responsive.

MS. WEISS: You have reiterated this four or five times.

BY MS. WEISS: (Resuming)

Q What remains of this recommendation? Not why, just what?

A I would like to know now if I am to be -- if I am to be allowed to complete answers to questions.

CHAIRMAN SMITH: There is a particular anomaly in the conversations on cross-examination and witnesses. Attorneys on cross-examination are permitted to be --

perhaps if you take simply the exchange of the moment -- to be unfair. They are permitted to be argumentative. They are permitted to be incomplete in their questions. They are permitted a great deal of latitude.

Witnesses have to pretty much accept it. Now, if you feel that an answer to Ms. Weiss' question needs further explanation after you answer the question directly and concisely, you should say that it requires further explanation. This is the way an organized record is developed.

Don't take it personally. The witness is the most important part of the hearing, but the person who is least protected. Unless you feel that you are personally being harassed by the questions, which I don't see that at all.

THE WITNESS: I don't feel that way.

CHAIRMAN SMITH: Try to give a direct, concise answer to the question, and if you believe an additional explanation is necessary, say so. Then either your counsel or the Board or more likely Ms. Weiss will ask you for a further explanation.

THE WITNESS: I believe a further explanation is required.

BY MS. WEISS: (Resuming)

Q Let me get the answer first.

CHAIRMAN SMITH: Let's begin the question and answer again.

MS. WEISS: I will start over again.

BY MS. WEISS: (Resuming)

Q I refer you specifically to Section 9 on page A-14 of NUREG-0585.

A Thank you.

Q That section is entitled "Review of Safety Classifications and Qualifications," and it reads, quote: "The owners of operating plants and all plants under construction should be required to evaluate the interaction of non-safety and safety grade systems during normal operation, transients, and design basis accidents, to assure that any interaction will not result in exceeding the acceptance criteria for any design basis event. The review should be systematic and include all non-safety components, equipment, systems and structures under all conditions of normal operations, anticipated operational occurrences, and design basis accidents, initiated both within the plant such as pipe breaks and from outside the plant, such as earthquakes, other natural phenomenon and offsite hazards."

Is TMI-1 required to perform that specific evaluation? Please give me a yes or no answer and then explain?

A I think the answer is no. No requirement has been imposed on any Licensee to date to my knowledge.

CHAIRMAN SMITH: Wait a minute. You said please give me an answer yes or no and then explain. And then when he begins to explain, if I heard it correctly, you said you had no further questions.

MS. WEISS: He continued after that. He can explain if he wants. It's fine with me.

CHAIRMAN SMITH: All right. You can explain your answer if you want.

THE WITNESS: It was sort of a reiteration of a previous answer. I think the answer is no. I don't personally agree with that answer.

CHAIRMAN SMITH: You don't agree with the answer? You don't -- it is the answer you don't agree with or the result? It was your answer.

THE WITNESS: The result, yes.

MS. WEISS: I have no further questions. If he wants to explain more, he can.

THE WITNESS: I didn't want to set myself up as a judge of a decision that the Commission has made on a basis which included considerations other than we in the Lessons Learned Task Force considered.

If there is any further explanation required, what I was trying to say before was, when we made this recommendation, the detailed and intensive studies of that accident have not been completed. As far as we knew --

well, we recognized even at that point that there were two factors involved: human and design problems.

To the extent that we could judge at that time which was the more serious or more compelling to be addressed, we couldn't make a choice. As the studies of the accident continued and the information began to come in and pile up in favor of operator error as the problem to address first on a very urgent basis, I think that is the kind of consideration that went into whether or not the implied equal priority that the Lessons Learned Task Force gave to both systems interaction studies and operator training, improvement, and that sort of thing.

Taken at face value, our words did not say that one should have great priority over the other. I think the Commission decision in the action plan and what is the requirements as they are imposed on TMI-1 and other Licensees reflects clearly a prioritization which ranks systems interaction studies below the other types of recommendations that came out of the study.

Even though I still personally favor strongly a requirement that all Licensees be required to do systems interaction studies, that is not the way the requirement came out, and it was very proper of Mr. Keaten to make that observation.

BY MS. WEISS: (Resuming)

Q Now I have to ask you one more question. I understand your testimony that you had to make decisions about what had to be done first, and it may be that the Commission's views of what had to be done first differed with the staff's views of what had to be done first, and maybe both of their views changed over time.

A Yes.

Q As to what needed to be done first. My question to you was not is there a time schedule or has that time schedule been changed, but whether there is any requirement at all that remains. And I understand your answer to be that there is none.

A I understand. I understand your dissatisfaction with my answer, then. In that regard, you're right.

At present it is not only a matter of the Commission saying it should be done on a different time schedule. At present, as I understand it, there is no explicit requirement that all Licensees ever be required to do a systems interaction study, and that is the part of the staff's position that I disagree with personally.

MS. WEISS: Before we leave, I have talked to the staff about how we get the two revisions of Regulatory Guide 1.29 and Section 10.4.7 of the standard review plan into evidence, all of which were referenced by Mr. Conran. I think the agreement is that we will offer all of the documents that Mr. Conran referenced in his clarification yesterday morning.

We will offer those all as UCS exhibits. We have three copies of each for the reporter. Maybe I should go through them at this point.

CHAIRMAN SMITH: You are completed with your examination of Mr. Conran?

MS. WEISS: Yes.

CHAIRMAN SMITH: I have a question before we go to the papers. Your use of Reg Guide 1.29 in your testimony, both in your direct testimony and cross-examination, does not seem to fit the disclaimer which appears on all reg guides, and that is: These are not regulations, they are not requirements. There may be substituted methods.

It just doesn't seem to fit. Could you comment on that? It would seem to me that the staff's position would be, if these must be safety grade, you are going to have a hard time coming up with a substitute for compliance with 1.29.

THE WITNESS: I think on a practical basis -- on a practical basis, first of all, reg guides, although they are not regulations, seem to be treated that way and thought of that way to a great extent. However, if a Licensee wants to get technical about it, then surely the point can be made that it is not a regulation.

The burden on him is to make the case that he has given adequate compliance or adequate safety.

CHAIRMAN SMITH: Your view as far as 1.29, the staff would be very, very skeptical about substitute compliance with those requirements.

THE WITNESS: I think we would give it one thorough review if somebody proposed otherwise. It occurs to me, however, that maybe the answer to the point that Ms. Weiss was pursuing is in fact reactor coolant pump seal associated cooling systems are not safety grade. The way that is specified in Reg Guide 1.29, it is conceivable that they are not that way, because the Licensees have made the case that they don't have to be or adequate protection is provided otherwise.

CHAIRMAN SMITH: That's fine. Thank you.

(Pause.)

CHAIRMAN SMITH: Would you proceed, very slowly.

MS. WEISS: Okay. The first document is Regulatory Guide 1.29, seismic design classification.

CHAIRMAN SMITH: Which one?

MS. WEISS: Revision 2, February 1976.

Let me give the reporter one of these packages while we are talking, so she can do the marking, and she can do the others later.

(Counsel hands documents to reporter.)

CHAIRMAN SMITH: UCS Exhibit 21.

(The document referred to was marked UCS Exhibit No. 21 for identification.)

MS. WEISS: The next one is Regulatory Guide 1.29, seismic design classification, revision 3, September 1978, and that should be UCS 22.

(The document referred to was marked UCS Exhibit No. 22 for identification.)

MS. WEISS: The next one is Section 11.2 of the standard review plan. It is labeled on the top right NUREG-75-087, and it is titled "Liquid Wastes Management Systems," and that should be UCS-23.

(The document referred to was marked UCS Exhibit No. 23 for identification.)

MR. CUTCHIN: It is labeled in the lower right-hand corner revision 1, Mr. Chairman.

MS. WEISS: Correct.

The next one is Section 11.3 of the standard review plan, also labeled in the upper right-hand corner NUREG-75-087, and in the lower right-hand corner revision 1, entitled "Gaseous Waste Management Systems." That should be UCS-24.

(The document referred to was marked UCS Exhibit No. 24 for identification.)

CHAIRMAN SMITH: There are two NUREG-75-087's in existence.

MS. WEISS: All of the standard review plan sections are 75-087.

(Pause.)

MS. WEISS: The next one is Section 11.4 of the standard review plan, also NUREG-75-087, also revision 1, titled "Solid Waste Management Systems."

(The document referred to was marked UCS Exhibit No. 25 for identification.)

MS. WEISS: The last one --

CHAIRMAN SMITH: One moment, please.

(Pause.)

MS. WEISS: The last one is Section 10.4.7 of the standard review plan, NUREG-75-087, also revision 1, titled "Condensate and Feedwater System." That should be UCS-26.

(The document referred to was marked UCS Exhibit No. 26 for identification.)

MS. WEISS: Mr. Conran also noticed a Federal Register notice; I do not see any need to put that into evidence; also because the version we have is virtually unreadable.

(Pause)

I will give the reporter the other two copies right now.

(Counsel handing documents to witness)

CHAIRMAN SMITH: These are offered?

MS. WEISS: Yes, Mr. Chairman. UCS offers them.

CHAIRMAN SMITH: Any objections?

(No Response)

CHAIRMAN SMITH: UCS Exhibits 21 through 26 are received.

(The documents previously marked UCS Exhibits 21 through 26 for identification, were received in evidence.)

MR. ROBERT ADLER: I have one point of clarification.

BY MR. ROBERT ADLER:

Q Did you state yesterday TMI-1 is not currently included in plans for IREP?

A I thought that was true. That is my state of knowledge on the subject, yes.

MR. ROBERT ADLER: Thank you.

CHAIRMAN SMITH: Anything further?

(No Response)

CHAIRMAN SMITH: You are excused,sir. Thank you.

(The witness was excused)

CHAIRMAN SMITH: Let's take our midmorning break.

(Recess)

CHAIRMAN SMITH: I guess our assembling ranks are present as much as they can be.

Are you ready, Mr. Cutchin?

MR. CUTCHIN: Yes, Mr. Chairman. I would ask Walton L. Jensen to take the stand.

Thereupon,

WALTON L. JENSEN, JR.

was called as a witness on behalf of the NRC, and having been previously duly sworn, was examined and testified as follows:

DIRECT EXAMINATION

BY MR. CUTCHIN:

Q Mr. Jensen, do you have before you a copy of a document consisting of seven pages plus a statement of your professional qualifications that bears the caption of this proceeding and is entitled "NRC Staff Testimony of W. Jensen Relative to the Classification of Pressurizer Heaters as Components Important to Safety (UCS Contention 3)?

A Yes, I do.

Q Was that document prepared by you?

A It was.

Q Do you have any corrections or modifications that you wish to make to the document?

A Yes, I have two corrections.

MR. CUTCHIN: These have already been put in the reporter's copy, Mr. Chairman.

THE WITNESS: The first is in the answer to my question 14. And it is in the last sentence of that answe . It should read, "Credit for operation of the pressurizer heaters is not assumed in the safety analysis of design basis accidents."

BY MR. CUTCHIN:

Q In other words, you have changed the word -- you have deleted the word "potential" and inserted the words "design basi " in its place?

A Yes. The second change is question 15. It should read, "With respect to the pressure control function of the pressurizer heater, should these componens be classified as components that are important to safety and that are necessary to perform a safety function specified in 10 CFR

100."

Q Very slowly. Start again and go very slowly on the new language.

A Excuse me. "With respect to the pressure control function of the pressurizer heater, should these components be classified as components that are important to safety and that are necessary to perform a safety function specified in 10 CFR 100."

Q As modified, is this testimony true and correct to the best of your knowledge and belief?

A Yes, it is.

MR. CUTCHIN: Mr. Chairman, I would ask that Mr. Jensen's testimony be received into evidence and bound into the record as if read along with the accompanying outline.

CHAIRMAN SMITH: Are there any objections?

MS. WEISS: No.

CHAIRMAN SMITH: The testimony is received.

(The testimony of Walten L. Jensen follows:)

MR. CUTCHIN: I have a few questions of Mr. Jensen in the way of rebuttal.

BY MR. CUTCHIN:

Q On page III-2 of Mr. Pollard's testimony, written testimony, the last sentence on the second paragraph reads as follows: "If a sufficiently high pressure is not

Lay-in #1 12/18/80 TMI-1

OUTLINE

This testimony of Walton L. Jensen, Jr., contains the NRC Staff's response to UCS Contention 3.

The purpose of this testimony is to demonstrate that, contrary to the assertions made in the contention, the pressurizer heaters are not components important to safety and thus need not satisfy safety-grade requirements.

that are necessary to perform a safety function specified in 10 CFR 100

Conclusions to be drawn from this testimony:

Pressurizer heaters are required to maintain hot standby.

Maintenance of hot standby is not a function important to safety.

Pressurizer heaters are not necessary to maintenance of natural circulation.

Normal cooldown procedures for the plant instruct the operator to turn off the pressurizer heaters to reduce reactor system pressure.

Loss of the pressurizer heaters would result only in slow depressurization of the reactor coolant system.

Operation of pressurizer heaters is not necessary to prevention or mitigation of accidents.

Pressurizer heaters are not components important to safety that are necessary to perform a safety function specified in 10 CFR 100

Pressurizer heaters need not satisfy safety-grade requirements.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter METROPOLITAN EDISON COMPANY (Three Mile Island Nuclear Station, Unit No. 1)	))))))	Docket No. 50-289 (Restart)

NRC STAFF TESTIMONY OF W. JENSEN RELATIVE TO THE CLASSIFICATION OF PRESSURIZER HEATERS AS COMPONENTS IMPORTANT TO SAFETY

(UCS CONTENTION 3)

Q1) Please state your name and position with the NRC.

A) My name is Walton L. Jensen, Jr. I am an employee of the U. S. Nuclear Regulatory Commission assigned to the Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. From June through December 1979, I was assigned to the Analysis Group of the Bulletins and Orders Task Force, Office of Nuclear Reactor Regulation.

Q2) Have you prepared a statement of professional qualifications?

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

Q3) Please state the nature of the responsibilities that you have had with respect to the Three Mile Island Nuclear Station - Unit 1.

A) The accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979, involved a feedwater transient coupled with the equivalent of a small break in the reactor coolant system, though the accident's ultimate severity resulted from a number of interacting elements including lack of complete understanding of system response, misleading instrument readings and inadequate operator training and procedures. Because of the resulting severity of ensuring events and the potential generic applicability of the accident to other reactors, the NRC staff initiated prompt action to:

(1) assure that other reactor licensees, particularly those plants such as TMI-1 which have a similar design to TMI-2, took the necessary actions to substantially reduce the likelihood of future TMI-2-type events from occurring, and (2) initiate comprehensive investigations into the potential generic implications of this accident on other operating plants.

To accomplish some of this work, the Bulletins and Orders Task Force (B&OTF) was established within the Office of Nuclear Reactor Regulation (NRR) in early May 1979. The B&OTF was responsible for reviewing and directing the TMI-2-related staff activities associated with loss of feedwater transient and small break loss-of-coolant accidents (LOCAs) for all operating plants to assure their continued safe operation.

I was assigned to the Task Force in June 1979. I participated in the preparation of NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants."

Following my assignment to the Reactor Systems Branch, I participated in the evaluation of potential feedwater transients at operating B&W plants and participated in the final preparation of the staff Safety Evaluation on the Three Mile Island 1 restart.

Q4) What is the purpose of your testimony?

A) The purpose of my testimony is to respond to the UCS Contention Number 3, which states:

> The staff recognizes that pressurizer heaters and associated controls are necessary to maintain natural circulation at hot standby conditions. Therefore, this equipment should be classified as "components important to safety" and required to meet all applicable safety-grade design criteria, including but not limited to diversity (GDC 22), seismic and environmental qualification (GDC 2 and 4), automatic initiation (GDC 20), separation and independence (GDC 3 and 22), quality assurance (GDC 1), adequate, reliable on-site power supplies (GDC 17) and the single failure criterion. The staff's proposal to connect these heaters to the present on-site emergency power supplies does not provide an equivalent or acceptable level of protection.

Q5) Are the pressurizer heaters and associated controls necessary to maintain natural circulation at hot standby.

A) No.

Q6) What is the function of the pressurizer heaters?

A) The pressurizer heaters are part of the normal control system which regulates primary system pressure. When the pressurizer heaters are activated, boiling occurs within the pressurizer producing steam which acts to increase reactor system pressure. The reactor system pressure may be reduced by operation of the pressurizer sprays which condense the steam in the pressurizer.

Q7) Are the heaters required to maintain hot standby?

A) Yes

Q8) Is it important to safety to maintain hot standby?

A) No.

Q9) What would be the consequences of a failure of the pressurizer heaters?

A) A failure of the pressurizer heaters would produce a slow decrease in reactor system pressure by heat transfer from the pressurizer to the surroundings. A startup test was recently conducted at Sequoyah which secured the pressurizer heaters during natural circulation (i.e., all reactor coolant pumps were also turned off). The rate of depressurization was measured at the Sequoyah Nuclear Plant Unit 1 to be 100 psi/hour. Pressurizer level was maintained utilizing the charging and letdown systems. Pressurizer heat loss data taken at TMI-1 indicates that the pressure reduction would be less than that at Sequoyah for a loss of pressurizer heaters. See Page C8-7 of the NRC SER for TMI-1 Restart NUREG-0680.

In the plant procedures for Pressurizer System Failure, Emergency Procedure 1202-29, the operator at TMI-1 is instructed to begin plant cooldown in the event that the pressurizer heaters fail to operate.

Q10) Is operation of the pressurizer heaters necessary to shutdown the reactor and maintain it in a safe shutdown condition?

A) No, the operating procedures for plant cooldown (OP 1102-11) instruct the operator to turn off the pressurizer heaters so as to reduce reactor system pressure. The goal is to reduce reactor system pressure sufficiently to reach the Decay Heat Removal System maximum operational pressure of 320 psig.

Q11) In the event that the reactor coolant pumps were also inoperable, would natural circulation be maintained?

A) Yes.

Q12) How?

A) The conditions required in the reactor system for natural circulation to be effective are discussed in the NRC response to UCS Contention 1. These discussions describe test data from B&W operating reactors which demonstrate that single phase natural circulation is an effective means of cooling the core when the reactor system temperature is below the boiling temperature. The discussions also describe the effect of steam bubbles in the reactor coolant loops as an effect which might retard natural circulation flow. Steam bubbles would begin to form if the reactor system coolant pressure dropped to the saturation pressure. For this reason, the operator is instructed to maintain the reactor system temperature below its boiling point with a 30°F margin by controlling the heat removal through the steam generators or if necessary, by activating High Pressure Injection (HPI) as discussed in Operating Procedures 1102-16 "RCS Natural Circulation Cooling."

Q13) What would be the effect of HPI activation?

A) The water added by the HPI system would act to prevent loss of pressurizer level and to increase the reactor system pressure so that boiling in the loops would not occur. The slow depressurization rate of the primary system following a failure of the pressurizer heaters (about 100 psi/hour) provides adequate time for the operator to prevent boiling of the primary system water. High Pressure System Injection is a safety grade system with redundant pumps and operates from emergency power busses.

Q14) Is operation of the pressurizer heaters necessary to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guidelines of 10 CFR 100?

A) No, protection for these accidents are provided by the Emergency Core Cooling and Emergency Feedwater systems. Credit for operation of the pressurizer heaters is not assumed in the safety analysis of design basis ~~potential~~ accidents.

Q15) With respect to the pressure control function of the pressurizer heater, should these components be classified as components that are important to safety and that are necessary to perform a safety function specified in 10 CFR 100?

A) No. As described in the above discussions, operation of the pressurizer heaters is not required for plant safety.

Q16) Then why does NUREG-0578 state that "...there is a need to consider the upgrading of those pressurizer heaters and associated controls required to maintain natural circulation at hot standby conditions to a safety-grade classification..."?

A) Section 2.1.1, Page A-2 of NUREG-0578 states "to achieve greater heater reliability and to decrease the number of demands for operation of the Emergency Core Cooling System." The repeated unnecessary actuation of the Emergency Core Cooling System is undesirable. The actuation of ECCS for a loss of pressurizer heaters would be an unlikely event at TMI-1 since adequate means is provided to the operator to control system pressure utilizing the charging and letdown systems and by controlling the cooldown rate of the steam generators. Protection from loss of pressurizer heaters due to loss of power supply will also be available at TMI-1 by connecting a bank of heaters to the emergency power supply with another bank of heaters available as a backup as discussed in the NRC SER for TMI-1 restart NUREG-0680, Pages C8-6 and C8-8.

These modifications will decrease challenges to ECCS. (See NUREG-0578, pp. A-1, A-2, and A-3).

WALTON L. JENSEN, JR.

PROFESSIONAL QUALIFICATIONS

I am a Senior Nuclear Engineer in the Reactor Systems Branch of the Nuclear Regulatory Commission. In this position I am responsible for the technical analysis and evaluation of the public health and safety aspects of reactor systems.

From June 1979 to December 1979, I was assigned to the Bulletins and Orders Task Force of the Nuclear Regulatory Commission. I participated in the preparation of NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants."

From 1972 to 1976, I was assigned to the Containment Systems Branch of the NRC/AEC, and from 1976 to 1979, I was assigned to the Analysis Branch of the NRC. In these positions I was responsible for the development and evaluation of computer programs and techniques to calculate the reactor system and containment system response to postulated loss-of-coolant accidents.

From 1967 to 1972, I was employed by the Babcock and Wilcox Company at Lynchburg, Virginia. There I was lead engineer for the development of loss-of-coolant computer programs and the qualification of these programs by comparison with experimental data.

From 1963 to 1967, I was employed by the Atomic Energy Commission in the Division of Reactor Licensing. I assisted in the safety reviews of large power reactors, and I led the reviews of several small research reactors.

I received an M.S. degree in Nuclear Engineering at the Catholic University of America in 1968 and a B.S. degree in Nuclear Engineering at Mississippi State University in 1963.

I am a graduate of the Oak Ridge School for Reactor Technology, 1963-1964.

I am a member of the American Nuclear Society.

I am the author of three scientific papers dealing with the response of B&W reactors to Loss-of-Coolant Accidents and have authored one scientific paper dealing with containment analysis.

maintained, the accumulation of steam will prevent operation of the reactor coolant pumps and will prevent natural circulation in the reactor coolant system.

Do you have anycomments you wold like to make in reference to that statement?

A Yes. As I testified fairly extensively in my answer to UCS contention one, a substantial amount of steam in the reactor coolant loop can temporarily block natural circulation.

However, if the level in the primary system is dropped down sufficiently, the two phase mode of natural circulation using boiling and condensation will be established in the steam generator. Since the emergency feedwater nozzles are higher than the elevation of the core, this two phase mode of natural circulation would be established before the core would become uncovered.

CHAIRMAN SMITH: Off the record.

(Discussion off the record)

BY MR. CUTCHIN:

Q Mr. Jensen, I now refer you to words at the bottom of page III-3 and at the top of page III-4 of Mr. Pollard's testimony.

And they read as follows: "The apparent purpose of this modification, meaning in the emergency power supply to the heaters, was to provide reasonable assurance that the pressure in the reactor coolant system could be maintained high enough to permit decay heat removal by natural circulation."

Do you agree with that statement, and would you please comment on it?

A That would be one of the purposes. The primary purpose of the modification is stated in NUREG-0578: it is to permit the -- is to prevent -- excuse me -- the unnecessary actuation of high pressure injection.

It is not desirable to actuate the high pressure injection system unless it is required to mitigate an accident since the actuation of the system introduces water from the fairly cool borated water storage tank very quickly into the four high pressure injection nozzles that are in the coolant loops.

These nozzles are heated to the primary system temperature and the introduction of cold water places the thermal cycle on these nozzles, and the plant is only designed for so many of these thermal cycles.

In the case the pressurizer heaters were lost, in the case of TMI-1, the plant would gradually depressurize over a period of several hours so that in about five or six hours the pressure would have descreased sufficiently so that high pressure injection system would be automatically actuated.

This would be before the system reached saturation, however, so this would then cut a thermal cycle on the high pressure injection nozzles.

In the pressurizer heaters, in the procedures, the plant procedures for pressurizer heater failures, however, the operator is instructed to begin cooling down the plant and shutting down the plant.

In this procedure, the pressure would be controlled using the makeup system which feeds into that one high pressure injection nozzle.

The makeup system is operated all the time during plant operation. So this particular nozzle has water from the makeup system flowing through it all the time.

So the use of the makeup system in controlling the plant pressure in the procedures would not place a thermal cycle on the high pressure injection system nozzle.

DR. JORDAN: Could I ask for a little clarification at that point: are you saying, then, that the procedure of using the makeup-letdown system would maintain the pressure so that you would stay in hot standby?

THE WITNESS: No. As I read the procedure, the procedure for pressurizer system failure call for the pumps to be shut down using the normal cooldown procedure.

And in this procedure, the plant would be brought to a cold shutdown condition using the makeup system as required to control system pressure.

DR. JORDAN: As the pressure is descreased -- and it would be descreased as you go for cold shutdown in order to meet the requirements --

THE WITNESS: Yes. It would be descreased by the fact that the pressurizer heaters were not in operation automatically by theheat loss through the insulation in the --

DR. JORDAN: But in the deliberate cooldown we are talking about now, I believe you said there would be loss of heaters. There would be a deliberate cooldown to -- and I presume with the idea of achieving cold shutdown.

THE WITNESS: Yes.

DR. JORDAN: They would adjust the pressure accordingly, and as the pressure decreased, then wouldn't the high pressure injection system come up?

THE WITNESS: In the cooldown procedures, I believe there are instructions on -- you would go down to a certain pressure. You lock out the high pressure injection system so it would not be actuated.

DR. JORDAN: There is a mechanism then for locking out the high pressure injection system?

THE WITNESS: Yes, I believe so.

DR. JORDAN: All right. That is all I was after now.

BY MR. CUTCHIN:

Q Isn't that referred to as the ECCS bypass?

A Possibly so.

Q On page III-11 of Mr. Pollard's testimony in the second full paragraph, on the last half of the page, the words appear, "Another example of the logical postion adopted by the staff at Met Ed is the failure to require conformance with general design criterion four by demonstrating that the pressurizer heaters will remain oeprable following a small loss of coolant accident."

Could you tell us whether pressurizer heaters are required to function in a small break LOCA scenario?

A Pressurizer heaters are not assumed to function in the analysis of these accidents. It is difficult to see what effect -- that they would have any effect on small break loss of coolant accidents because those that were analyzed by B & W showed that the pressurizer would be emptied very quickly in the event of a small break LOCA.

For breaks in the primary system, for the case of a stuck open PORV, the pressurizer would not be emptied, of course, but for this condition, all of the heat that would be generated by the pressurizer heaters would be carried out the stuck open PORV and in any case the primary system would be quickly brought to a saturated condition because of loss of fluid out of the valve.

So they really would have no effect on loss of coolant accidents.

Q One last question: on page III-14 of Mr. Pollard's testimony, the paragraph numbered two reads as follows: "Another principal lesson learned from the TMI-2 accident is that the frequency of events that lead to opening the PORV should be reduced and that the methods of assuring that a stuck open PORV can be isolated should be improved;to suggest that an anticipated operational occurrence should be handled by deliberately opening the PORV and turning a routine event such as a loss of offsite power into a loss of coolant accident is contrary to the lessons supposedly learned."

Could you comment on those statements, please?

A I do not believe that the PORV would be opened at TMI-1 in the event that the pressurizer heaters failed to operate. That is because the operator would be instructed to bring the plant -- to bring the plant down and shut it down, reducing the saystem pressure so that the pressure would never reach the point where the PORV would be opened.

If the operator took no action at all, the system would gradually decrease in pressure in any case to the point where the high pressure injection system would be actuated.

However, when the high pressure injection system is actuated, the pressure would increase again by the primary system at a pressure of about 1800 psi. The 50 degree subcooling criteria would be reached,and the operator could then thrrottle the highpressure injection system.

And also in this case the PORV would not be challenged.

MR. CUTCHIN: Thank you, Mr. Jensen.

I have no further questions. Mr. Jensen is available for cross examination.

CHAIRMAN SMITH: Ms. Weiss?

(Pause)

MS. WEISS: I ought to have the record note at this point that as I told the board and the parties earlier Mr. Pollard is unwell, and I think we had a demonstration of that earlier this morning.

And so I am sitting here by myself, and I am not competent to prepare questions on the rebuttal that we have just heard. I can certainly attempt to do the cross examination plan that has already been prepared and in the hands of the board.

I do not believe that I should leave this witness without having an opportunity to discuss this rebuttal with Mr. Pollard now. And I do not know how sick he is because I have not seen him since he left this morning.

MR. CUTCHIN: I was going to make a suggestion, Mr. Chairman. Mr. Jensen will be back in connection with an additional contention. To the extent that Ms. Weiss decides that she has querstions specifically going to this oral rebuttal, we will stipulate that Mr. Jensen can answer then at that time, unless there is a better way.

CHAIRMAN SMITH: However, if you have an opportunity to address it during this segment and it develops that you feel confident to start or to try, you should try.

And then if after Mr. Pollard is able to attend you feel there is need to address it again, we will approach it then.

MS. WEISS: I do not feel competent to ask questions on rebuttal.

CHAIRMAN SMITH: You do not even want to attempt it?

MS. WEISS: No, but I will try the rest of the cross examination.

CROSS EXAMINATION

BY MS. WEISS:

Q Mr. Jensen, would it be correct to state that as you described in connection with your previous pieces of testimony, the expertise and the analysis that you brought to bear on the B & W LOCA analysis, that that generally describes the way which you prepared this testimony as well?

And that is not terribly elegantly stated, but the question is: is this testimony also based on the B & W small break LOCA analyses and their other computer analyses and your review of those?

A It is not directly based on B & W's LOCA analysis, though they give an idea of how the system behaves to -- well, I based my testimony on-- I mentioned a test that was done on the Sequoyah nuclear reactor; calculations that have been made on the effect of heat loss on depressurization and on the effect of the pressurizer heaters on the primary system.

I do not think it is particularly based on LOCA analysis.

I was certainly influences by myreview of the LOCA analysis.

(Pause)

Q Can you please define "hot standby," "hot shutdown," and "cold shutdown."

A Yes, I believe so. Hot standby would in my view -- it is -- the reactor is critical and hot and at a fairly high pressure.

Hot shutdown is a similar condition, but the reactor would not be critical because of the -- the safety rods would be inserted into the core.

Cold shutdown would also be with the rods inserted, but with the reactor at a fairly cold temperature and pressure.

Q Do you know specifically for Three Mile Island Unit 1 what the temperatures and pressures are?

A Well, I looked at the technical specifications for Three Mile Island, and let's see, for hot standby the temperature was greater than -- I think it was a t-average or greater than 525 degrees.

For hot shutdown the temperature was also -- t-average temperature was greater than 525 degrees fahrenheit. And the reactor was shut down by a 1 percent criticality margin.

Cold shutdown, I believe, was also shutdown by 1 percent criticality margin. And I believe the temperature was 200 degrees fahrenheit or less than 200 degrees fahrenheit.

Q I was struck by contrasting the questions and answers to querstions five and seven.

Let me read them and ask you to explain what accounts for the difference in the answers.

Question five is "Are the pressurizer heaters and associated controls necessary to maintain natural circulation at hot standby?"

Your answer is "No."

And question seven is, "Are the heaters required to maintain hot standby?"

And your answer is "Yes."

Would you explain to me what accounts for the difference in those?

A Yes. I guess it is hard to see. My thinking was that if the pressurizer heaters were lost at hot standby, the natural circulation would be maintained.

That is the basis for my answer to question five. But question seven, I answered that the heaters were necessary to maintain hot standby, and I was -- I had in mind the depressurization of the primary system that would occur if the heaters were not operational so that over several hours the reactor would be tripped.

I guess then it would be in a hot shutdown condition.

(Pause)

Q Question 6 on page 3. You mention that the reactor system pressure may be reduced by operation of the pressurizer sprays. Do those sprays operate if the reactor coolant pumps are tripped?

A I don't believe they do.

Q And the operators are now instructed to trip the reactor coolant pumps immediately upon the onset of a small-break LOCA? That's correct, isn't it?

A He would trip the pumps in the event he received a high-pressure injection system caused by a low reactor system pressure, which he would get in the event of a small-break LOCA. But the small-break LOCA would likely depressurize the system anyway without the benefit of the pressurizer sprays.

Q On question 9 you describe the consequences of a failure of the pressurizer heaters. Does that describe the consequences of a failure of the pressurizer heaters during a LOCA?

A No, it doesn't. I really didn't address these answers to a LOCA. I was thinking just the failure to pressurize the heaters and their effect on natural circulation. I don't think the pressurizer heaters would have much effect on LOCA.

Q This question hypothesizes only a presssurizer failure and everything else is operating normally in the plant; correct?

MR. BAXTER: Pressurizer heater failure?

MS. WEISS: Pressurizer failure.

THE WITNESS: The test I am referring to at Sequoyah was done with the primary coolant pumps off.

BY MS. WEISS:

Q Reactor coolant pumps?

A Yes. However, the depressurization rate of the system caused by pressurizer heater failure would be about the same whether or not the reactor coolant pumps were operational or not.

Q What would the rate of depressurization be during a small-break LOCA within the capability of the makeup or high-pressure injection system?

A I guess it wouldn't depressurize. If the break was in the capability of the makeup or high-pressure injection system, it would come to some equilibrium pressure where the flow in was equal to the flow out.

Q Did the Sequoyah test simulate such a situation?

A There wasn't a break in the system at the Sequoyah test. It was done to observe the effect of the pressurizer heaters being off on natural circulation.

Q Then I am correct that for a small-break LOCA within the capability of the high-pressure injection system, the primary system would not depressurize?

A That's true.

Q For the Sequoyah test was the emergency feedwater system operating properly?

A I assume it was, because natural circulation occurred. If it wasn't operating, there would not have been natural circulation.

Q And were there any prohibitions against the use of a letdown system because of leak rate, or did you postulate a leak rate or a radiation level which would have resulted in a prohibition against the use of the letdown system?

A I believe the letdown system was used. Of course, it removes mass from the primary system, and it would act to depressurize the system. It would have the opposite effect of the makeup system. I don't see why there would be a particularly high radiation level in the letdown system water anyway as the result of a pressurizer heater failure.

Q During the TMI-2 accident, letdown system could not be used because of the high radiation level; is that correct?

A I really don't know.

Q Are you aware of any case of an operating reactor where the plant has gone from hot to col shutdown with the primary system solid throughout that entire period?

A Not completely. But Westinghouse plants do routinely go solid, both in startup and shutdown. And they are taken solid at pressures below 400 p.s.i.

Q Tell me the range over which the Westinghouse system is designed to be solid, what pressure range, where it is operated in a solid mode?

A It would be from cold shutdown to 400 p.s.i. The system is a good deal stiffer at low pressures than it would be at high pressures and temperatures.

Q Do you agree that the capability to maintain natural circulation is important to safety?

A Yes.

Q Do you agree that controlling pressure is important to achieving the conditions necessary for natural circulation?

A Yes. But, as I have already testified, if the pressure isn't controlled, even though natural circulation could be temporarily blocked, it would be reestablished in th two-phase condensation mode before it could become uncovered.

Q Is the two-phase condensation mode a feed-and-bleed mode?

A No.

Q Do you know what the effect would be -- let me strike that. You state that -- do you know what the effect would be on the number of demands for ECCS if the pressurizer heaters were made fully safety-grade?

A No, I don't.

Q Do you know what the effect will be on the number of demands for ECCS by adding one heater bank to emergency power or to adding to capability of connecting one heater bank to emergency power?

A No.

Q Question 16 --

A Excuse me. The effect would be to lessen the demands on the ECCS. I don't know quantitatively how much it would be lessened.

Q Question 16. You state that "In the unlikely event of loss of pressurizer heaters for TMI-1, the actuation of ECCS" --- that's not exactly what you said. Let me read exactly what you say: "The actuation of ECCS for a loss of pressurizer heaters would be an unlikely event at TMI-1, since adequate means is provided to the opportunity to control system pressure utilizing the charging and letdown systems and by controlling the cooldown rate of the steam generators."

Would you tell me, please, which pumps are used for the charging system?

A The charging system is a generic term. For Three Mile Island Unit 1, the charging system is the makeup system. And I believe that high-pressure injection pump or makeup pump number B is used.

Q The tech specs for Three Mile Island Unit 1 allows the plant to operate with only two of the three HPI pumps functional; is that correct?

A I think so.

Q Would it be accurate to say that the charging system uses the same pumps as the high-pressure injection system?

A Yes, the makeup system uses the same pumps as the high-pressure injection system.

Q I believe it is on the record that the letdown system is not safety-grade, is that correct, for Three Mile Island Unit 1?

A I don't think it is.

A And when you refer to controlling the cooldown rate of the steam generators, is that done by the use of the turbine bypass valves and/or the atmospheric dump valves?

A Yes.

Q And those are also not safety-grade for Three Mile Island Unit 1?

A No, they are not. But let me point out that these valves are located outside of containment. In the process of cooling down by heat loss from the pressurizer in the event that the pressurizer heaters were lost would be very small. So that it would take, at a depressurization rate of 100 p.s.i. per hour, it would take a fairly long time before the primary system would reach the saturation pressure. During this time the relief valve from the secondary system would be available to be serviced.

Q I think we have had previous testimony on the subject, and I don't want to get back into it again, particularly when I am by myself. Do the tech specs for Three Mile Island Unit 1 require the availability of both groups of heaters as a limiting condition of operation?

A I haven't looked at the tech specs. I haven't observed that in the tech specs. In fact, I haven't read them. But the procedures require, of course, that the plant be shut down if the pressurizer heaters are not available.

Q If the procedures require the plant to be shut down if the pressurizer heaters are not available --

A Yes.

Q -- the modifications will require that two of the banks of heaters out of the five, I think, have the capability of being connected to emergency power; is that correct?

A That is my understanding of what will be done.

Q With respect to those two banks of heaters, do you know whether the tech specs for Three Mile Island Unit 1 require both to be available as a limiting condition of operation?

A I don't know.

MS. WEISS: I have no further questions of the witness at this time.

CHAIRMAN SMITH: Mr. Adler is -- I don't see him. Do you want to cross-examine him, Mr. Dornsife?

MR. DORNSIFE: I am sorry, Mr. Chairman, Mr. Adler had some other business. He had to go to the office for a while. I am going to be exclusively representing the Commonwealth. I have no questions for the witness.

CHAIRMAN SMITH: Mr. Baxter?

MR. BAXTER: I have no questions.

BOARD EXAMINATION

BY MR. JORDAN:

Q Question 16, which you were just considering in your reply to Ms. Weiss, quotes a section of NUREG-0578, which says: "There is a need to consider the upgrading of those pressurizer heaters and associated controls required to maintain natural circulation at hot standby conditions to a safety-grade classification."

Are you saying there was a consideration made and, as a result of the consideration, they decided against upgrading to safety grade?

A Well, I guess that is my testimony.

Q That is your testimony. That is what I wanted to make sure. What I am asking is: There was consideration given, as required by NUREG-0578?

A Yes.

Q All right. In answer to Question 13, which is, "What would be the effect of high-pressure injection at division," you reply, "The water added by the HPI system would act to prevent loss of pressurizer level and to increase the reactor system pressure so that boiling in the loops would not occur."

How does the high-pressure injection system increase the system pressure? Is it by compressing the steam bubbles or by going solid?

A I guess in the scenario that I postulate, as heat was lost from the pressurizer, the pressure would be dropped in the primary system to the set point of the high-pressure injection system. During this time, the system would not be solid. And then when the high-pressure injection was actuated, the pressure would increase, and this would be by compression of the bubble in the pressurizer.

However, if the system was left at a high pressure and was not depressurized and the pressurizer heaters were not operational, the bubble would gradually condense and the system would be solid.

Q Would it go solid before you reached the cold shutdown condition?

A The procedures for going to cold shutdown call for bypassing the high-pressure injection system and bringing the plant down. So the high-pressure injection would not be actuated. The system would be gradually depressurized by removing heat through the steam generators, and then pressure could be controlled, if need be, by controlling the charging and letdown system. By using these systems, the pressurizer would not have to be brought solid.

Q I see.

A Only if the reactor system was left at a -- in the pressurized state and not brought down, without pressurizer heaters, the system would gradually go solid.

Q I guess I am puzzled a bit, because you are saying that you could control the pressure by the charging and the letdown system without going solid.

A Yes. This is the normal procedure. The idea, of course, is to decrease the pressure. The operator would be decreasing the pressure by removing heat from the steam generators, but he would also try to maintain the system in a subcooled condition. He would do this by adjusting his charging flow. As he brought the system down in pressure, there would be a shrinkage of water in the system. So that to maintain a constant pressurizer level, he would have to add water to the system, using the makeup pumps, during the time of cooldown.

Q I see. So you are saying that by maintaining the pressurizer level, he will maintain a pressure?

A Yes.

Q Even without heaters, because of the heat capacity of the water in the pressurizer?

A Yes. And it would require a small amount of steam all the time as he was decreasing the pressure, because the specific volume of the steam would be increasing. So, as he brings the system down, even though steam is condensed, he would need less steam.

Q Okay. With respect to the Sequoyah tests that you reference in Question 9, were those tests performed prior to operation and, hence, without any after-heat in the core?

A The test was done with the reactor at a power of 3 percent, and then the reactor coolant pumps and pressurizer heaters were tripped.

Q So the 3 percent power simulated the after-heat that might be in the core of another reactor?

A Yes.

(Pause.)

Q I think Ms. Weiss has already asked the other questions that I had on your testimony.

DR. JORDAN: I have no further questions.

MR. CUTCHIN: No further questions.

CHAIRMAN SMITH: Ms. Weiss?

MS. WEISS: I have one to follow up what Dr. Jordan asked.

CROSS-EXAMINATION ON BOARD EXAMINATION

BY MR. WEISS:

Q He asked you, Mr. Jensen, whether there was specific consideration made to upgrading the heaters to safety grade and you said, yes, there was. Could you tell me, please, where I could see that document anywhere?

A Well, what I was referring to basically was my testimony which is on the need to make the pressurizer heaters safety grade.

Q Your testimony references the statement in NUREG-0578 which is also referenced by UCS in our contention, particularly question 16. You quote from it. And it states, quote: "There is a need to consider the upgrading of those pressurizer heaters and associated controls required to maintain natural circulation at hot standby conditions to a safety grade classification." Quote.

Dr. Jordan asked you if that consideration had been made. I think your answer was yes; is that correct?

A I have certainly considered it. I have written this testimony about it.

Q Your answer is it was considered only by you in your testimony?

A That's all I have knowledge of. It may have been considered by many other people. That's all I have direct knowledge of.

Q Are you aware if any group on the staff did a specific analysis to consider the upgrading of the heaters and controls to safety grade?

A I'm not. But there may well have been such a group.

Q If there were one, you are not aware of it?

A I can't think of it at this moment.

Q Did the Licensee ever submit any documentation describing what would be required to do a full upgrading of the heaters at Three Mile Island Unit 1 to you in connection with your considerations?

A I haven't seen such a document.

Q Then you don't know specifically what design modifications would be required at Three Mile Island Unit 1?

A No, I don't.

MS. WEISS: No further questions.

(Pause.)

REDIRECT EXAMINATION

BY MR. CUTCHIN:

Q The question has been, was there consideration by others to the recommendation of NUREG-0578 that the pressurizer heaters be upgraded to full safety grade status. To your knowledge, is there a NUREG-0660 task action plan item which would reflect a requirement that these heaters be fully upgraded?

MS. WEISS: If that is what the question was, that is not -- I did not intend the question to imply that there is a requirement that they be upgraded, but merely that there is a statement of the need to consider upgrading.

MR. CUTCHIN: The reason I used the word "requirement" and differentiated between requirement and recommendation is I believe I heard Dr. Jordan perhaps misspeak when he referred to the 0578 requirement and I wanted to make sure that that was a recommendation.

My question is, is there a task action plan item, and I think that is a better indicator of whether these recommendations were picked up.

THE WITNESS: I just don't remember.

MR. CUTCHIN: I believe, Mr. Chairman, NUREG-0660 could speak for itself.

CHAIRMAN SMITH: Certainly it can. Why don't you go right to it?

MR. CUTCHIN: I do not have it, but I was hoping perhaps the witness would know, and we can make an effort to look and see. It is my understanding that it does not. Therefore, I couldn't cite you where it does.

CHAIRMAN SMITH: Anything further with Mr. Jensen?

(No response.)

CHAIRMAN SMITH: You're excused.

(Witness excused.)

MR. BAXTER: Mr. Chairman, I am going to have to -- I apologize. I have to ask for an early lunch break. One of the three members of my next panel stayed in his hotel room nursing a minor variation of what is going around here. We will be ready after the one-hour lunch break.

MS. WEISS: I am not sure whether you ought to call him back.

MR. BAXTER: He is on his way here now.

MS. WEISS: Let me just say, I feel at an extreme disadvantage on an issue that Mr. Pollard has testimony on, to go into the examination of the witnesses without him beside me. I don't think there is any way to get him back today. I wish it hadn't happened, but I don't really believe that I am prepared to go forward at this point.

MR. CUTCHIN: Mr. Chairman, am I understanding Ms. Weiss correctly that, with respect to the cross-examination plan, she was relying heavily on Mr. Pollard to do the examination himself? If that be the case, then I think it is understandable that she is unable to go forward.

But had she been planning to ask the majority of the questions herself, maybe we could make the attempt to go as far as she could. And then where she reached the point where she thought Mr. Pollard was the appropriate one to ask the questions, maybe we would have to run down.

MS. WEISS: It is not that I was thinking that he would do a lot of the questioning himself. But he prompts me. And it is extremely difficult to go forward on a subject like this next one without him here, as I think both of the other counsel would testify that they would not like to go forward without their technical advisors next to them.

MR. BAXTER: No, I wouldn't like to. On the other hand, a good deal of the consultation is done ahead of time in preparing the plan.

These witnesses fully expect to be here tomorrow, and I would hope that -- or ask whether we couldn't make an attempt at it, Ms. Weiss. And if, with the cross-examination that the Commonwealth has and the Board's examination, we don't fill the afternoon, we could break early. And perhaps Mr. Pollard will be well enough tomorrow that we could continue or come back to you.

MS. WEISS: It's fine with me. If you -- if other people, if the Board and the parties want to do the questioning, I have no objection to that. I do have an objection to my having to go forward with mine.

CHAIRMAN SMITH: They're going to be here. Why don't we begin and see -- I don't have the cross-examination plan, nor the testimony before me. Do you have cross-examination, Mr. Dornsife?

MR. DORNSIFE: Yes, sir. I believe we submitted a plan on 5, not 6.

CHAIRMAN SMITH: I think we should begin, and then when the problem actually comes up, we come face to face with the problem, we will deal with it then.

MR. BAXTER: It would seem to me that at least the planned question or the first question can be asked. I understand the problem with follow-up questions. At least that's where I need my technical assistance. And the witnesses would be back Friday for those follow-up questions.

I don't understand why Ms. Weiss couldn't ask at least the initial planned questions.

CHAIRMAN SMITH: I agree that you are going to have to have access to Mr. Pollard to develop the record fully on this point if you say you do. And I think that is quite clear, that you regularly depend upon him even when you are doing the examination.

I think you should look at your cross-examination plan just to see what you can do, and then we will worry about it when it actually becomes time.

MR. BAXTER: Mr. Chairman, I have another sort of extraordinary suggestion to consider. I note Ms. Bradford is here, and whether we can inquire whether she would be prepared to go ahead with the argument on Dr. Beyea this afternoon, which we would be if there is excess time available.

CHAIRMAN SMITH: Did you hear the suggestion, Ms. Bradford?

MS. BRADFORD: I would prefer to do that tomorrow.

CHAIRMAN SMITH: Other than preference, is it possible for you to do it this afternoon? Bear in mind, Ms. Bradford, we have been very, very accommodating.

As a matter of fact, on second thought, when the Board set this for the 16th we did it without hearing objections from the other people and it really was not fair timing for it. So if you can present your argument later this afternoon, it would be very helpful. If you cannot, okay. It's up to you.

MS. BRADFORD: I would prefer to do it tomorrow. I don't have my papers here today. I was not expecting to do it today.

CHAIRMAN SMITH: All right. Then I have another. Let's start and then see what questions the Board and others might have. And then we will see if you can start, and if you can't then perhaps Mr. Pollard might feel better. We have enough problems than to rule on problems before they arise.

Now, I have another preliminary matter. I wonder if we could enlist your aid in communicating with Mr. Jordan about what his preference is with respect to the intervention of PANE in this proceeding. There is a rather unusual situation. The Commission's order was predicated upon rejecting psychological stress contentions.

PANE has nothing except psychological stress contentions. A determination has to be made by someone whether or not the Commission's order itself was the action terminating the intervention of PANE, in which case then PANE has its remedies, or whether Mr. Jordan believes that the Board should issue an order effectuating the Commission's determination and rejecting the petition to intervene.

So before I decide on what we should do on it, Mr. Jordan should have an opportunity -- excuse me?

MR. CUTCHIN: Don't you mean Mr. Cunningham -- I'm sorry.

CHAIRMAN SMITH: There is an Intervenor here --

MR. CUTCHIN: I shouldn't have interrupted.

CHAIRMAN SMITH: Could you do that for us? If it is not convenient --

MS. WEISS: It is no problem. I have to call the office anyway.

CHAIRMAN SMITH: We would like to hear from him what his preference is or what his view is on the problem.

We will adjourn, then, until 1:00 p.m.

(Whereupon, at 11:55 a.m., the hearing was recessed, to reconvene at 1:00 p.m. the same day.)

AFTERNOON SESSION

(1:00 p.m.)

MR. BAXTER: We recall Robert C. Jones to the stand, and we call Gary T. Urquhart and James H. Correa.

Whereupon,

GARY T. URQUHART

JAMES H. CORREA

were called as witnesses on behalf of the Licensee and, having been first duly sworn, were examined and testified as follows; and

Whereupon,

ROBERT C. JONES

was recalled as a witness on behalf of the Licensee and, having been previously duly sworn, was examined and testified as follows:

DIRECT EXAMINATION

BY MR. BAXTER:

Q Going from my left to right, would each of you state your name, position, and place of employment?

A (WITNESS JONES) Robert C. Jones, Jr. Babcock & Wilcox Company, Lynchburg, Virginia.

Q What is your position?

A (WITNESS JONES) Supervisor engineer, ECCS analysis unit.

A WITNESS URQUHART) Gary T. Urquhart, the unit manager of the auxiliary equipment unit, nuclear power generation division, Babock & Wilcox Company.

A (WITNESS CORREA) James H. Correa. GPU, Parsippany. Mechanical engineer in the mechanical components section.

Q Gentlemen, I call your attention to two documents which bear the caption of this proceeding. The first one is dated September 15, 1980. It is entitled "Licensees Testimony of James H. Correa, Gary T. Urquhart, and Robert C. Jones, Jr., in Response to UCS Contentions 5 and 6, Valves and Valve Testing."

The second document is dated October 28, 1980. It is entitled "The Licensees Testimony of James H. Correa and Gary T. Urquhart, in Response to the Board Question on UCS Contention 6."

Does the testimony associated with your names in these two documents, including the attached statement of professional qualifications, represent testimony prepared by your or under your direct supervision for presentation at this hearing, Mr. Jones?

A (WITNESS JONES) Yes.

Q Mr. Urquhart?

A WITNESS URQUHART) Yes.

Q Mr. Correa?

A (WITNESS CORREA) Yes.

Q Do you have any changes or corrections to make to your testimony, Mr. Jones?

A (WITNESS JONES) No.

Q Mr. Urquhart?

A WITNESS URQUHART) No.

Q Mr. Correa?

A (WITNESS CORREA) No.

Q Is the testimony true and accurate, to the best of your knowledge and belief?

A (WITNESS JONES) Yes.

A WITNESS URQUHART) Yes.

A (WITNESS CORREA) Yes.

MR. BAXTER: Mr. Chairman, I move that the testimony identified be received into evidence and incorporated into the transcript as if read.

MS. WEISS: No objection.

MR. CUTCHIN: No objection.

CHAIRMAN SMITH: The testimony is received.

(The documents referred to were marked UCS Exhibits No. 2 and 3 for identification and received in evidence.)

MR. BAXTER: I have questions on oral rebuttal, but no one is representing the Commonwealth. I don't know what to suggest to do.

Lam-in #2

8746

LIC 10/28/80

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(Restart)
	)	
(Three Mile Island Nuclear Station, Unit No. 1)	)	
	)	

LICENSEE'S TESTIMONY OF

JAMES H. CORREA AND GARY T. URQUHART

IN RESPONSE TO THE BOARD QUESTION ON UCS CONTENTION 6

OUTLINE

This testimony supplements Licensee's Testimony of James H. Correa, Gary T. Urquhart and Robert C. Jones, Jr. in Response to UCS Contentions 5 and 6 (Valves and Valve Testing), dated September 15, 1980. In particular, this testimony responds to the Board Question relating to UCS Contention 6.

The testimony explains that while the pressurizer safety valves perform a safety function, the PORV does not. Because of the design of the safety valves, it is expected that they can perform the required safety function of opening and discharging liquid or two-phase fluid if necessary. In addition, the experience during the Crystal River transient of February 26, 1980, and in the fossil power industry generally, provides some assurance that the results of the EPRI test program will be favorable.

INTRODUCTION

This testimony, by Mr. James H. Correa, Engineer, Mechanical Components, GPU, and Mr. Gary T. Urquhart, Unit Manager, Auxiliary Equipment Unit, Babcock & Wilcox Company, is addressed to the following Board Question regarding UCS Contention 6:

> The board wants more than just a schedule for testing of reactor coolant system safety and relief valves, as is required pursuant to NUREG-0578. Is there reasonable assurance that the tests will be successful, e.g., that there is good evidence that the valves will indeed perform in an accident environment?

RESPONSE

BY WITNESSES CORREA AND URQUHART:

The original design and testing of the pressurizer power operated relief valve (PORV) and safety valves was described in Licensee's testimony in response to UCS Contentions 5 and 6 (Valves and Valve Testing) (pages 4-8). As also addressed in that testimony (pages 2, 3 and 7) the PORV does not serve a pressure relief safety function. The safety valves, however, do serve a safety function in that they provide Reactor Coolant System overpressure protection. The safety valves may also serve as a safety-grade discharge path for reactor coolant fluid during feed and bleed operation - see Licensee's testimony in response to UCS Contentions 1 and 2 (Natural and Forced Circulation) (page 12).

The only function required of the safety valves in order to provide overpressure protection or for feed and bleed operation is to open and discharge fluid. The disc lifts in response to the system pressure force on the disc face. The pressure at which the disc lifts - i.e., at which the valve opens, or functions - is dependent on the opposing force applied by the valve spring. Because of the construction of the valves there is no reason to expect that liquid or two-phase flow conditions would have a detrimental effect on the ability of the valves to perform their required function.

This conclusion is specifically supported by the experience at Crystal River on February 26, 1980, and the examinations subsequent to that transient - see Licensee's testimony in response to UCS Contentions 5 and 6 (pages 6 and 7). The valve opened at 2400 psig; was open for approximately 20 minutes; experienced saturated steam, two-phase fluid and water at 2400 psig, 410°F with a maximum flow rate of 700 gpm; and reseated at 2300 psig (4% blowdown). These conditions are similar to those in one of the valve tests in the EPRI test program, in which the valve is set to open at 2500 psig, pass 450°F water at a maximum flow rate of 1000 gpm, and reseat at approximately 2375 psig (5% blowdown).

Also, safety valves are used extensively in fossil power applications. Many of those valves are similar in basic design to the valves at TMI-1 and have experienced flow conditions other than steam. There is no known power industry incident of a properly set and maintained safety valve failing to open upon demand, even though liquid and two-phase flow through these valves has occurred.

JAMES H. CORREA

Business Address: GPU Service Corporation
100 Interpace Parkway
Parsippany, New Jersey 07054

Education: B.S., Mechanical Engineering, Rensselaer Polytechnic Institute, 1969.

Experience: Mechanical Engineer III, GPU Service Corporation, 1978 to present. Responsible for providing technical engineering on valves for GPU system nuclear power plants; providing technical support to resolve field problems, including repair recommendations and field technical guidance; providing technical support for plant modifications, including writing technical specifications for valves and modification documentation packages. Other responsibilities have included reviewing flow diagrams for proper valve selection; reviewing architect-engineer technical specifications for technical content including referencing the proper codes and standards and valve design features.

Mechanical Design Engineer, Foster Wheeler Corporation, 1972 to 1978. Performed engineering work on primary sodium valves for the fast flux test facility and steam generators for a high temperature gas cooled reactor. Responsibilities included preparing material and sub-contracted machining requisition packages; vendor surveillance; preparing and issuing shop fabrication releases which include drawings and shop procedures; and the resolution of vendor material and machining problems and shop fabrication problems in the areas of manufacturing, materials and quality control.

Cognizant Engineer, Machinery Apparatus Operation, General Electric Company, 1970 to 1972. Performed technical engineering work on Naval Nuclear Heat Exchangers and Pressurizers, including definition of specifications, vendor selection, design review and analysis, fabrication surveillance, and the resolution of installation problems. Engineering work included the solving of technical problems in a number of technical disciplines such as mechanical analysis, heat transfer, quality control, materials and welding, and manufacturing.

Engineer, Mechanical Facilities Planning, Missile and Space Division and Re-entry and Environmental Systems Division, General Electric Company, 1969 to 1970. Performed design and cost estimates for specific projects such as ventilation systems and piping systems. Provided design direction for construction and renovation projects.

Professional Affiliations: Registered Professional Engineer, New Jersey.

GARY T. URQUHART

Business Address: Babcock & Wilcox Company
Nuclear Power Generation Division
P.O. Box 1260
Lynchburg, Virginia 24505

Education: B.S., Mechanical Engineering, State University of New York at Buffalo, 1970. M.B.A., Lynchburg College, 1979.

Experience: Unit Manager, Auxiliary Equipment Unit, Equipment Engineering Section, Babcock & Wilcox Co., 1980 to present. Responsible for preparation of equipment specifications for equipment such as valves, heat exchangers, small pumps and tanks, evaluation of vendors' designs, review and approval of vendor submitted documentation, and resolution of field problems.

Senior Engineer and Supervisory Engineer, RCS Mechanical Design Unit, Component Engineering Section, Babcock & Wilcox Co., 1976 to 1980. Responsible for detail design and analysis, manufacturing liaison and resolution of shop and field problems for the reactor internals (core support assembly).

Various assignments in Quality Control (Assurance) and Materials Engineering for the Fossil Power Generation Division, Nuclear Equipment Division and Nuclear Power Generation Division, Babcock & Wilcox Co., 1970 to 1976. Responsibilities included preparation of manufacturing procedures such as non-destructive examination and welding, material selection, evaluation and analysis for fossil boilers and the performance of internal and vendor quality audits.

Lay-in #3

LIC 9/15/80

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear Station, Unit No. 1)	)	
	)	

LICENSEE'S TESTIMONY OF

JAMES H. CORREA, GARY T. URQUHART AND ROBERT C. JONES, JR.

IN RESPONSE TO UCS CONTENTIONS 5 AND 6

(VALVES AND VALVE TESTING)

OUTLINE

The purposes and objectives of this testimony are to respond to UCS Contentions 5 and 6, which assert that proper operation of the pressurizer power operated relief valve (PORV) is necessary to mitigate the consequences of accidents, that the failure of the PORV can create or aggravate a loss of coolant accident (LOCA) and that appropriate qualification testing has not been performed to verify the capability of the reactor coolant relief and safety valves. The testimony discusses that the PORV was not designed to fulfill a safety function and is not required for mitigation of design basis LOCA's. It is explained that while the PORV can be actuated and potentially remain open, creating or aggravating a LOCA, analyses have been performed to demonstrate that these transients can be safely mitigated. Changes to minimize the possibility of such an occurrence are also addressed. The testimony continues with a discussion of the original design and testing applied to the pressurizer relief and safety valves. Recent experience at Crystal River 3 during which a safety valve flowed steam, two-phase fluid and water is addressed. Modifications being made to the PORV, and the EPRI valve testing program are described.

INDEX

INTRODUCTION 1

RESPONSE TO UCS CONTENTION NO. 5 2

RESPONSE TO UCS CONTENTION NO. 6 4

INTRODUCTION

This testimony, by Mr. James H. Correa, Engineer, Mechanical Components, GPU, Mr. Gary T. Urquhart, Unit Manager, Auxiliary Equipment Unit, Babcock & Wilcox Company, and Mr. Robert C. Jones, Jr., Supervisory Engineer, ECCS Analysis Unit, Babcock & Wilcox Company, is addressed to the following contentions:

UCS CONTENTION NO. 5

> Proper operation of power operated relief valves (PORV's), associated block valves and the instruments and controls for these valves is essential to mitigate the consequences of accidents. In addition, their failure can cause or aggravate a LOCA. Therefore, these valves must be classified as components important to safety and required to meet all safety-grade design criteria.

UCS CONTENTION NO. 6

> Reactor coolant system relief and safety valves form part of the reactor coolant system pressure boundary. Appropriate qualification testing has not been done to verify the capability of these valves to function during normal, transient and accident conditions. In the absence of such testing and verification, compliance with GDC 1, 14, 15 and 30 cannot be found and public health and safety is endangered.

UCS withdrew its sponsorship of its Contention No. 6, which has been adopted as a Board Question (See Board Memorandum and

Order of Prehearing Conference of August 12-13, 1980, dated August 20, 1980).

RESPONSE TO UCS CONTENTION NO. 5

BY WITNESS JONES:

UCS Contention 5 states that proper operation of the pressurizer power operated relief valve (PORV) is necessary to mitigate the consequences of accidents and that the failure of the PORV can create or aggravate a loss of coolant accident (LOCA). Contrary to the contention, the PORV is not required for mitigation of design basis LOCA's and, while a LOCA would result if the PORV did not close after being actuated, such as occurred at TMI-2, the safety-grade Emergency Core Cooling System (ECCS) is designed to mitigate the event and to assure adequate core cooling.

The original design function of the PORV was to provide a pressure relief capability which, in conjunction with plant control system actions to reduce reactor power and/or adjust steam generator feedwater flow, would prevent a reactor trip on high primary system pressure during various operational transients. In this manner, unit availability would be enhanced. The relief capability of the PORV was not designed to fulfill a safety function. The high pressure trip function of the Reactor Protection System (RPS) and the pressurizer safety valves provide the required overpressure protection for the Reactor Coolant System. The RPS and the pressurizer safety valves are safety-grade equipment and comply with applicable criteria.

Since the TMI-2 accident the setpoints for the PORV and the high pressure reactor trip setpoint have been inverted. In the original design and operation of TMI-1, the opening pressure for the PORV was 2255 psig and the high pressure reactor trip setpoint was 2355 psig. These setpoints are now 2450 psig and 2300 psig, respectively. As a result, actuation of the PORV is not now expected during operational transients provided that feedwater is delivered to the steam generators in a timely manner. Thus, the frequency of PORV actuation has been reduced.

However, there are still circumstances where the PORV can be actuated and potentially remain open, creating or aggravating a LOCA. Analyses have been performed to demonstrate that these transients can be safely mitigated (as defined by 10 CFR Part 50, Paragraph 50.46(b)) by the ECCS. These analyses included both a stuck-open PORV case (i.e., the PORV causes a LOCA), and a scenario in which a small-break LOCA occurs simultaneously with a loss of all feedwater and results in a subsequent stuck-open PORV (i.e., the PORV aggravates a LOCA) - see Licensee's testimony on Additional LOCA Analysis in response to UCS Contention 8. Additionally, there have been several changes made to enhance the operator's ability to recognize and terminate a transient caused by a stuck-open PORV. Specifically, an accelerometer which senses discharge line flow and discharge line flow measurement instrumentation are being provided. These, along with PORV position demand indication and PORV discharge line temperature measurement, will provide additional assurance that PORV position will be recognized. Also, the PORV and block valve have power supplied by the emergency power system. This provides the capability for closing the block valve upstream of the PORV, in the event of a stuck-open PORV and a loss-of-offsite power.

In summary, and contrary to the above contention, proper operation of the PORV and associated block valve and the instruments and controls for these valves is not essential to mitigate the consequences of design basis LOCA's and, although the failure of the PORV can create or aggravate a LOCA, the consequences of such an accident can be safely mitigated by safety-grade equipment.

RESPONSE TO UCS CONTENTION NO. 6

BY WITNESS URQUHART:

UCS Contention 6 asserts that appropriate qualification testing has not been performed to verify the capability of the reactor coolant relief and safety valves. Contrary to this assertion, these valves - the pressurizer power operated relief valve (PORV) and safety valves have - been properly designed and tested pursuant to applicable criteria.

The pressurizer safety valves are components important to safety in that they are both part of the reactor coolant pressure boundary and functionally provide overpressure protection for the Reactor Coolant System (RCS). The valves were designed for and protect the integrity of the RCS at the design conditions of the primary system - 2500 psig and 670°F. Reference 1 describes in detail the pressure relief criteria for the valves, the method of analysis to develop the criteria, and the results and conclusions of the analysis. As is shown in the referenced document, the RCS is adequately protected by either of the two safety valves since each is capable of relieving the required capacity.

The relief capacity of the safety valves was established consistent with the applicable edition and addenda of Section 9 of Section III of the ASME Boiler and Pressure Vessel Code. This included certification by the valve manufacturer of the capacity of the valves utilizing prototypical testing to establish discharge factors and analytical verification of the ability of the valves to withstand design and operating pressures.

The safety valves were also designed in accordance with the requirements of Section III of the ASME Code to assure reactor coolant pressure boundary integrity. Testing and examination of the valves during and following manufacturing and testing included the following:

(a) Chemical and mechanical testing of the materials.

(b) Volumetric examination of the materials.

(c) Surface examination of the materials.

(d) Hydrostatic pressure testing of the completed valves at the manufacturer and after installation.

(e) Verification of set pressure.

(f) Seat leakage testing following opening and closing.

Also of significance with regard to the capability of the pressurizer safety valves is the transient which occurred February 26, 1980, at the Crystal River nuclear unit, a plant with a B&W nuclear steam system and components similar to TMI-1. During the transient, one of the two safety valves lifted at approximately 2400 psig and flowed saturated steam, two-phase fluid and liquid water. The water flow rate was up to 700 gpm and the valve reseated at approximately 2300 psig, a blowdown of about 4% below the opening pressure.

Subsequent to the transient, the affected valve was subjected to detailed laboratory inspection and testing to determine if any damage had been sustained. The set pressure of the valve was checked three times and determined to be approximately the 2400 psig experienced during the transient. Leakage was measured at about 1.1 gpm. Disassembly and inspection identified steam cutting of the valve disc and a damaged bellows assembly. The steam cutting was most likely caused by leakage that was present prior to the transient. The damage to the bellows did not appear to be due to the February 26, 1980 transient. Neither the steam cutting of the disc nor the damaged bellows impaired the intended pressure relief function of the valve. In summary, no damage detrimental to the proper operation of the valve was discovered even though it had experienced flow conditions other than saturated steam.

The pressurizer PORV was designed for the same system conditions as the safety valves - 2500 psig and 670°F. The valve design was governed by the same ASME Code requirements as the safety valves as it related to pressure boundary integrity, and the valve was tested and examined in a manner similar to the safety valves. Because the PORV is power operated in response to an independent pressure signal, verification of set pressure was not applicable. Verification of valve opening and closing was performed however, prior to shipment and following installation. Also, as discussed in the testimony above in response to UCS Contention 5, the PORV does not serve a pressure relief safety function. Therefore, certification of relief capacity was not required nor was such considered necessary, and an upstream isolation/block valve is allowed by design criteria and is provided. Relief capacity was established by design analysis. The General Design Criteria are applicable to the PORV only to the extent that it forms part of the reactor coolant pressure boundary.

BY WITNESS CORREA:

The PORV which will be installed in TMI-1 prior to restart is the TMI-1 spare PORV. This valve was ordered per the original PORV requirements, was manufactured in 1978, was "N" stamped per Code Case 1581, and in general satisfies the 1977 Edition with the Winter 1979 Addendum of Section III of the ASME B&PV Code for fabrication requirements.

The valve is being modified per the manufacturer's latest design features to improve seat tightness. The modification is being performed per the latest ASME B&PV Code, Section III, requirements. As part of the modification effort, the valve will be disassembled and all critical dimensions will be recorded and checked against drawing requirements. In addition, all moving parts will be inspected for surface finish and signs of wear caused by the original testing of the valve prior to its shipment in 1978. This inspection of the valve internals will ensure that the valve parts meet all requirements. After reassembly of the valve, it will be seat leak tested and opened at its set point. This will ensure that the valve will function properly.

Prior to being installed in TMI-1 the valve will again be seat leak tested. During hot functional testing the valve also will be actuated to ensure its functional ability and to test all downstream instrumentation.

A valve testing program is also in progress. This program is being conducted by the Electric Power Research Institute (EPRI). The purpose of the program is stated in the EPRI Program Plan for the Performance Testing of PWR Safety and Relief Valves, Revision 1, dated July 1, 1980 and is as follows:

> The primary objective of these tests is to evaluate the performance of each of the various types of reactor coolant system safety and relief valves in pressurized water reactor plant service for the range of fluid conditions under which they may be required to operate.[1] The requirements are that:
>
> 1. The safety and relief valves open and close on command, when subjected to simulated plant operational conditions calculated to result in valve actuation.
>
> 2. The flow capacity of the valves be established.
>
> The second objective of the program is to obtain sufficient piping thermal hydraulic and support reaction load data to permit confirmation of analytical models utilized for plant unique analysis of safety and relief valve discharge piping systems.

---

1 These conditions will be defined based on an evaluation of the transients specified in Regulatory Guide 1.70, Revision 2.

The program plan to be followed in evaluating the performance of PWR safety and relief valves includes a number of elements which are described in the following:

° A test program will be performend in which selected, actual safety and relief valves are tested under fluid conditions which are calculated to occur during anticipated operational transients and postulated accident sequences in PWR plants. These fluid conditions include steam, water and transition from steam to water. The primary purpose of these tests is to demonstrate that the valves will open and close as required when subjected to simulated transient conditions and that the flow capacity of the valves can be correctly predicted.... It is expected that all testing will be complete by July, 1981.

° A combined test and analysis program will be performed to evaluate the adequacy of analytical methods utilized for PWR safety and relief valve discharge piping response. First, the main valve test facility at Combustion Engineering will include prototypical upstream piping, including water seals, and a simplified discharge piping arrangement which simulates significant features of plant discharge piping systems. These systems will be instrumented to measure dynamic load, piping response and fluid conditions. In parallel with this effort, engineering evaluations are being performed to assess the adequacy of available methods for prediction of safety and relief vavle discharge piping loads. A key part of this effort is the analysis of a number of sample problems using state-of-the-art methods. These problems will include the upstream and discharge piping configurations and ranges of fluid conditions selected for use in the valve performance tests. In addition, analysis of piping configurations representative of actual PWR discharge piping installations has been initiated to demonstrate that the test configuration adequately represents all significant features important to safety and relief valve operation. The combined results of these analytical test programs will provide the data needed to confirm the analytical methods used for piping and support analysis. This information will then be available to utilities for use on a plant-specific basis for evaluation of installed discharge piping systems....

- An evaluation will be performed of available data and experience obtained in foreign valve test facilities, and any domestic test programs that may be applicable. Utilization of other related test experience is considered desirable in order to identify and minimize potential problem areas which might otherwise have an impact on the EPRI test program schedule....

- Effort is underway to evaluate the effects of postulated valve failure modes (e.g., excessive leakage, excessive blowdown, reduced flow capacity, etc.,) on reactor system performance in order to establish preliminary acceptance criteria and guidelines for evaluation of the significance of the valve test results.

- Evaluations of the Crystal River 3 safety and relief valves and piping will be performed. This will be a co-operative effort among EPRI, Florida Power Corporation and Babcock and Wilcox to examine the valves and piping at Crystal River 3 which were subjected to water discharge conditions in February 1980. This evaluation is expected to provide early information on the performance of the affected valves and discharge piping. It may also provide useful information on the effect of service history and aging on valve performance.

(See Mr. Urquhart's testimony above on the Crystal River inspection.)

Met-Ed has submitted its plant specific data (valve drawings and inlet and discharge piping drawings) to EPRI for inclusion in the testing program. One of the relief valve types chosen to be tested is the same model as the TMI-1 relief valve, Dresser model no. 31533VX-30. Also, one of the safety valves types chosen to be tested is the same model as the TMI-1 safety valve, Dresser model no. 31739A.

B&W has supplied operational transient and postulated accident sequence data to EPRI for 177-fuel-assembly reactors (TMI-1 type). This data is being used in defining test parameters for the EPRI test matrix. Therefore the EPRI test results can be directly applied to TMI-1.

As stated in the Restart SER, the EPRI test program is responsive to NRC short term recommendation 2.1.2 of NUREG-0578.

BY WITNESSES CORREA AND URQUHART:

In summary, contrary to the above contention, the TMI-1 pressurizer relief and safety valves have been appropriately designed and tested. In addition, actions are being taken to provide further assurance that the valves will function properly and reliably.

REFERENCE

1. Topical Report BAW-10043, "Overpressure Protection for Babcock & Wilcox Pressurized Water Reactors," May, 1972.

JAMES H. CORREA

Business Address: GPU Service Corporation
100 Interpace Parkway
Parsippany, New Jersey 07054

Education: B.S., Mechanical Engineering, Rensselaer Polytechnic Institute, 1969.

Experience: Mechanical Engineer III, GPU Service Corporation, 1978 to present. Responsible for providing technical engineering on valves for GPU system nuclear power plants; providing technical support to resolve field problems, including repair recommendations and field technical guidance; providing technical support for plant modifications, including writing technical specifications for valves and modification documentation packages. Other responsibilities have included reviewing flow diagrams for proper valve selection; reviewing architect-engineer technical specifications for technical content including referencing the proper codes and standards and valve design features.

Mechanical Design Engineer, Foster Wheeler Corporation, 1972 to 1978. Performed engineering work on primary sodium valves for the fast flux test facility and steam generators for a high temperature gas cooled reactor. Responsibilities included preparing material and sub-contracted machining requisition packages; vendor surveillance; preparing and issuing shop fabrication releases which include drawings and shop procedures; and the resolution of vendor material and machining problems and shop fabrication problems in the areas of manufacturing, materials and quality control.

Cognizant Engineer, Machinery Apparatus Operation, General Electric Company, 1970 to 1972. Performed technical engineering work on Naval Nuclear Heat Exchangers and Pressurizers, including definition of specifications, vendor selection, design review and analysis, fabrication surveillance, and the resolution of installation problems. Engineering work included the solving of technical problems in a number of technical disciplines such as mechanical analysis, heat transfer, quality control, materials and welding, and manufacturing.

Engineer, Mechanical Facilities Planning, Missile and Space Division and Re-entry and Environmental Systems Division, General Electric Company, 1969 to 1970. Performed design and cost estimates for specific projects such as ventilation systems and piping systems. Provided design direction for construction and renovation projects.

Professional <u>Affiliations</u>: Registered Professional Engineer, New Jersey.

GARY T. URQUHART

Business Address:	Babcock & Wilcox Company Nuclear Power Generation Division P.O. Box 1260 Lynchburg, Virginia 24505
Education:	B.S., Mechanical Engineering, State University of New York at Buffalo, 1970. M.B.A., Lynchburg College, 1979.
Experience:	Unit Manager, Auxiliary Equipment Unit, Equipment Engineering Section, Babcock & Wilcox Co., 1980 to present. Responsible for preparation of equipment specifications for equipment such as valves, heat exchangers, small pumps and tanks, evaluation of vendors' designs, review and approval of vendor submitted documentation, and resolution of field problems.
	Senior Engineer and Supervisory Engineer, RCS Mechanical Design Unit, Component Engineering Section, Babcock & Wilcox Co., 1976 to 1980. Responsible for detail design and analysis, manufacturing liaison and resolution of shop and field problems for the reactor internals (core support assembly).
	Various assignments in Quality Control (Assurance) and Materials Engineering for the Fossil Power Generation Division, Nuclear Equipment Division and Nuclear Power Generation Division, Babcock & Wilcox Co., 1970 to 1976. Responsibilities included preparation of manufacturing procedures such as non-destructive examination and welding, material selection, evaluation and analysis for fossil boilers and the performance of internal and vendor quality audits.

ROBERT C. JONES, JR.

Business Address:	Babcock & Wilcox Company Nuclear Power Generation Division P.O. Box 1260 Lynchburg, Virginia 24505
Education:	B.S., Nuclear Engineering, Pennsylvania State University, 1971. Post Graduate Courses in Physics, Lynchburg College.
Experience:	June 1971-June 1975: Engineer, ECCS Analysis Unit, B&W. Performed both large and small break ECCS analyses under both the Interim Acceptance Criteria and the present Acceptance Criteria of 10 CFR 50.46 and Appendix K.
	June 1975-Present: Acting Supervisory Engineer and Supervisory Engineer, ECCS Analysis Unit, B&W. Responsible for calculation of large and small break ECCS evaluations, evaluations of mass and energy releases to the containment during a LOCA, and performance of best estimate pretest predictions of LOCA experiments as part of the NRC Standard Problem Program. Involved in the preparation of operator guidelines for small-break LOCA's and inadequate core cooling mitigation.

MR. SMITH: Let me take up another matter first. Maybe they will arrive. I can't find the cross-examination plan.

MS. WEISS: I have got one. I don't specifically remember giving it to the Board. I don't remember exactly when I did it. I thought I did it. Let me see. I may have a copy.

CHAIRMAN SMITH: I am not suggesting you have it. I am just saying I can't find it and Mrs. Moran isn't here to help me. So that could very well be the problem.

MS. WEISS: It's only three pages long.

CHAIRMAN SMITH: All right, let me take it.

(Pause.)

MR. BAXTER: I notice there is a representative of the Commonwealth here. I propose to proceed. I am going to be referring to the direct testimony of Robert D. Pollard on behalf of the Union of Concerned Scientists regarding UCS Contention Number 5. The testimony is dated October 10, 1980.

MS. WEISS: One moment, please.

(Pause)

BY MR. BAXTER:

Q In item two, page V-4 of his testimony, Mr. Pollard states that both relief and safety valves have an alarming history of failing to reclose.

Earlier on page V-3 in the first sentence of the second full paragraph he states, "There is a history of relief and safety valve failures at operating plants."

Just below that sentence Mr. Pllard identifies three types of failures which he asserts have been experienced.

Mr. Urquhart, are you aware of experiences in operating pressurized water reactors where the pressurizer safety valves have opened below the set point, the first example of failures cited by Mr. Pollard?

A (Witness Urquhart) Let me first -- to answer your question directly, yes, there have been occasions when the pressurizer safety valves opened below the set point. However, I would not characterize that type of situation as being a failure of the valve.

I would characterize a failure of the valve as a valve no performing its overpressure protection function, which is to open and relieve the system overpressure.

The two cases I have direct knowledge of is Babcock and Wilcox reactors where on Rancho Seco a pressurizer safety valve did lift, and it lifted somewhat light, and also at Crystal River 3 where thesafety valve lifted and also lifted below the opening set pressure.

Q Is that what you mean when you say it lifted light, that it lifted below the set pressure?

A (Witness Urquhart) Yes. It lifted below the 2500 psig set pressure.

Q Moving to the next category of failures cited, are you aware of any instances in operating pressurized water reactors where the pressurizer safety valve has not opened at the set point?

A (Witness Urquhart) Other than the instances where they opened below the set point, I am not aware of any instance where the pressurizer safety valve when called upon has opened at a pressure exceeding the set point.

In the two instances that I stated on Rancho Seco and Crystal River 3, both valves opened below the set point which, as far as protecting the reactor coolant system, is in the safe direction.

In discussions with manufacturers of the safety valves -- namely, Dress & Crosby -- they are also not aware of where their safety valves have failed to open when called upon, where the pressurizer safety valves have failed to open when called upon; that is, above the set pressure.

Q Moving to the last category, are you aware of any instances in operating PWRs where the pressurizer safety valves have not reclosed after the pressure has decreased below the opening set point?

A (Witness Urquhart) The design of the pressurizer safety valve, of course, is that the pressure does decrease somewhat below the opening set pressure. A term called blowdown where the pressure actually has to decrease a certain percentage below the opening point before the valve will reclose; that is by design.

In the two instances I am aware of -- mainly, the Rancho Seco -- directly aware of -- the Rancho Seco safety valve lift and the Crystal River 3 safety valve lift, Rancho Seco opened somewhat below the set pressure and reclosed.

I do not know the exact closing pressure. Crystal River 3, the valve opened and reclosed within 4 percent of the opening set pressure, which is as designed.

DR. JORDAN: My mind was wandering a bit. Were you in those instances referring to the PORV or the safety valve?

WITNESS URQUHART: The safety valve.

DR. JORDAN: In both cases?

WITNESS URQUHART: Yes.

CHAIRMAN SMITH: Your first round of questions related in both facilities to PORV valves.

MR. BAXTER: No, sir.

WITNESS URQUHART: Safety valves.

CHAIRMAN SMITH: You have not come to PORVs?

MR. BAXTER: The testimony by Mr. Pollard refers to both relief and safety, but my questions so far have really dealt with safety valves.

BY MR. BAXTER:

Q Mr. Urquhart, as we just learned, the testimony of Mr. Pollard that I referred you to on pages V-3 and V-4 refer to the history of failures, asserted history of failures of relief and safety valves both.

In addition to that testimony on page V-6 Mr. Pollard refers to the relatively high probability of PORV failure; on page V-12 to the history of PORVs failing to reclose.

Have you reviewed the experience of PORV failures, and if so, what are your comments on Mr. Pollards observations?

A (Witness Urquhart) Yes, I have reviewed the history of PORV failures. On the Babcock & Wilcox PWRs, there have been three instances when the plant was at power when the PORV has failed to reclose.

Considering the number of times the PORV has been activated at power, I personally would not consider that an alarming history of failure. In addition, prior to the TMI-2 accident, the last previous incident where a PORV had failed to close, a PORV that is of the design TMI-1 has on their plant, a dresser PORV, was in November of 1975, from the period November 1975 until the TMI-2 accident; there was no failure of a dresser PORV on a B & W reactor to close with -- I believe there was in excess of 60 actuations in that time period.

Q On page 5-6 of his testimony --

DR. JORDAN: Let's clear this up now. You said no failures of dressers. Does that mean that Davis-besse was not a dresser valve?

WITNESS URQUHART: That is correct. Davis-Besse was not a dresser valve.

DR. JORDAN: All right.

BY MR. BAXTER:

Q On page 5-6 Mr. Pollard states that the staff has previous acknowledged that the probability of failure of the PORV in the open position contributes significantly to the probability of a small break LOCA.

He cites page 3-7 of NUREG-0565.

Mr. Jones, is that citation to NUREG-0565 relevant to an assessment today of the probability of PORV failure?

A (Witness Jones) No. The assessment in 0565, that statement was relative to the probability of a PORV sticking open pre-TMI, the pre-TMI experience with the valve. Since the TMI accident, the set point for the reactor trip and the PORV opening set point have been inverted and has reduced the frequency of PORV actuation.

So now in my belief it would not contribute significantly to the probability of a small break LOCA.

Q On page 5-9 of his testimony, Mr. Pollard addresses what he considdsered to be the reason for the change in set points you just described, Mr. Jones.

Do you agree with his explanation of the reasons for making those changes in set points?

MS. WEISS: I am trying to take notes; one moment, please.

(Pause)

BY MR. BAXTER:

Q In particular, I would like you to address the last sentence in the second paragraph: "The change in set points reflects a basic recognition of the inherent unreliabiity or inadequate qualification of the valve shown through a history of valve failure."

A (Witness Jones) The set point changes that were made were made shortly after the TMI accident, and were made basically to reduce the frequency of actuating the PORV because it had stuck open at TMI.

It was not made, to my knowledge, based on any study which stated that the PORV was an unreliable valve, but rather made as a prudent measure, in light of the fact that the transient had stuck open a valve at TMI.

As far as the set point being kept below the pressurizer safety valve set point, because of -- apparently Mr. Pollard is claiming that the safety valves are inherently unreliable or they have been inadequately qualified in his statement, it is my belief that the reason the PORV set point was kept below the safety valve set point was to provide additional defense in depth.

That is, you do not necessarily want to acuate the safety valve if it is not necessary, and by keeping the PORV below the pressurizer safety valve, you provide an additional buffer to safety valve set point.

It was not done because of any recognition of any inherent unreliability of the safety valve.

DR. JORDAN: Is there some basis for saying that the PORV is better able to handle openings and closings, that you would rather have it be the PORV than the safety valve?

Is it better designed to handle relief of pressure?

WITNESS JONES: I would not state that, nor would I state the latter, the counter-positive to that. I think it is simply a recognition that --a general recognition that you generally do not want to use safety systems if it is not absolutely necessary to provide a defense in depth concept with a nonsafety grade -- in many instances, a nonsafety grade piece of equipment to prevent hitting the safety grade piece of equipment.

Additionally, the PORV does have a block valve which can isolate that path should it stick open. It is not based on any unreliability.

Another way to reflect this is the PORV has never been claimed to be functionally operable during the plant life while it is up at power. It has never been treated as a piece of safety equipment.

There has never been a recognition of the safety valve being unreliable and there have been instances where plants have run with the block valve closed in the PORV path.

CHAIRMAN SMITH: Deliberately?

WITNESS JONES: Deliberately.

BY MR. BAXTER:

Q Beginning at the bottom of page 5 --

MR. BAXTER: I am sorry, Dr. Jordan.

DR. JORDAN: There is nothing in the specs, then, requiring the block valve to be open during operation?

WITNESS JONES: Not to my knowledge.

BY MR. BAXTER:

Q Beginning at the bottom of page 5-10, Mr. Pollard is describing pressure control during low temperature operation. He concludes that passage on page 5-11 with the statement, "During low temperature operation, the PORV clearlyperforms a safety function."

Mr. Jones, what is the role of the PORV during low temperature operation, and how is it designed to perform that function?

A (Witness Jones) When you are in low temperature operation, you do set the PORV at a low pressure set point. But the tech specs allow the PORV to be taken out of service if certain conditions are met, such as you have basically the HPI system racked out or lock out valves closed, and the level in the pressurizer being maintained at a volume.

The licensing basis for the low temperature operation of the plant was operator action to mitigate these transients. Basically, you have to show that there was better than 10 minutes for the operator to terminate an overpressure transient at low temperatures.

The PORV just serves as a backup to the operator action function; no credit was given to the PORV as a licensing basis.

Q On page 5-12 of the testimony, Mr. Pollard asserts that, "Reducing challenges to the emergency core cooling system is in itself a safety function, and therefore a goal that is important to safety."

Do you agree with that view, Mr. Jones?

A (Witness Jones) Well, reducing challenges to safety systems is an objective you try to strive for. You do not want to challenge safety systems if they are not necessary to perform an actual function.

That is not in my mind a safety objective; it is just an operational consideration. You do not -- there is no goal of how often you should challenge the safety systems, and in fact the plants are designed to have a certain number of actuations of safety equipment, including actuations which may be inadvertent.

CHAIRMAN SMITH: How do you reconcile that statement with your immediate past testimony on the purpose of the PORV?

WITNESS JONES: Like I said, in general, you do not want to hit safety systems. Gong back to, like, say, the safety valves on the pressurizer, if you challenge the safety valves, there is a potential that you may have a leak develop after they are challenged, which may result in shutdown of the plant while you refurbish the valves, et cetera.

It is an operational concern rather than a safety concern. If you actuate the PORV and it should make you close the block valve, you go right back up in operation. Actuation of th emergency core cooling system, for example, will cause a thermal shock transient on the nozzle,on the HPI nozzle. You do not want to do that if it is not necessary.

The nozzles are designed to withstand something on the order of 40 actuations; as long as you meet that design for the nozzle, you are not violating any safety limits. You set up your plant so that you will not have more than 40 safety actuations on a single plant.

It is not a safety goal, per se.

DR. JORDAN: This nozzle you are speaking of now is the nozzle from the safety valve on the discharge side?

WITNESS JONES: No. The example I was giving there was the ECCS injection nozzls, the HPI nozzles.

DR. JORDAN: The ECCS system, then, is designed to operate only 40 times during the life of the plant; is that right?

WITNESS JONES: The nozzles that the ECCS injects to have been analyzed to 40 cold water cycles hitting a hot nozzle. From that sense -- you can say in a sense the system as a whole is designed to be reliable, not just for 40 cycles, but the nozzle itself can only analytically have been only analyzed to withstand 40 cycles.

It is expected if you go look at actual actuations that if they are happening -- or higher water temperatures in the BWST, for example, than what was analyzed, you can withstand many more cycles than the original 40 whch was designed.

The 40 was chosen as that is the, say, expected type of actuation you would get, something on the order of 40 once a year.

DR. JORDAN: Has TMI-1 used up an appreciable fraction of its 40?

WITNESS JONES: I really do not know.

CHAIRMAN SMITH: I have a related question. I have been thinking about it ever since we have been talking about challenges to safety systems. What is the philosophy concerning testing these systems under the conditions under which they would have to operate when called upon to perform their safety function?

You say don't test them, dont wear them out. We just depend upon their design and quality assurance to assure they work?

WITNESS JONES: No. The systems are periodically tested; I am not sure of the exact frequency, but the HPI system, the pumps are started every -- I think it is six months; it may be less -- to assure that the will start. It does not inject into the reactor vessel, however. I am not sure of the exact layout and how it is done, but they do not -- I think they just open the recirculation line around the pump and they assure they pump starts and develops a proper head for that flow condition which verifies that the pumps will work.

They also stroke the valves and items like that and assure the circuitry itself works in a testing mode on a certain periodic basis which is set out in the technical specifications.

The safety valves, I believe, one safety valve is taken off at every refueling and tested as to whether it will pop at its proper set point.

BY MR. BAXTER: (Resuming)

Q Mr. Jones, looking at Mr. Pollard's testimony on feed and bleed cooling, which begins on page 5-13 of his written testimony, do you rely on PORV, as he asserts, to accomplish or maintain the feed and bleed cooling mode

A (WITNESS JONES) No. We rely on -- well, the word "rely" is too strong. We will use, if available, the PORV for feed and bleed cooling. But the analyses that have been performed to demonstrate the capability of feed and bleed cooling have been done using safety valves only, not the PORV.

Q On the issue of depressurizing the reactor coolant system under conditions of inadequate core cooling --

MS. WEISS: Just a second, please.

BY MR. BAXTER: (Resuming)

Q On the question of depressurizing the reactor coolant system under conditions of inadequate core cooling, Mr. Pollard testifies at page 5-17 that there is no alternative to use of the PORV for depressurization. So that therefore that is a PORV safety function.

Do you agree, Mr. Jones?

A (WITNESS JONES) No, I do not. If you look into the procedure which he has attached to his testimony, which is 1202-6B, it is page 25 of that procedure.

(Pause.)

What Mr. Pollard is interpreting as the PORV being the only means to depressurize the system is the note under Step 3.3 about the RCS will depressurize after the specific statement of open the pressurizer PORV. I would like to note Step 3.2 above it, which is depressurize the operative steam generator as quick as possible to atmospheric pressure.

That action will be much faster in depressurizing the primary system than using the PORV. We use both means to depressurize the system. The PORV itself is not fulfilling -- the PORV is an additional means to depressurize the plant, but will be -- have a smaller impact than use of the steam generator.

CHAIRMAN SMITH: You testified before what the equivalent square inches of opening was on the PORV. Did you say on the order of half a square inch?

WITNESS JONES: It is 1.04 square inches.

DR. JORDAN: This is smaller than the safety valves?

WITNESS JONES: Yes.

MR. BAXTER: I think the testimony in square feet was .007.

WITNESS JONES: .00739.

BY MR. BAXTER: (Resuming)

Q Looking at the same page of Mr. Pollard's written testimony, 5-17, he references emergency procedure 1202-39 on inadequate core cooling. Is there a reason why non-safety grade equipment might be empoyed in that procedure, Mr. Jones?

A (WITNESS JONES) Yes, there is. I would also like to note, the procedure which I was just reading from, which is attached to Mr. Pollard's testimony, attachment 3 to 1202-6B, is also the inadequate core cooling procedure. Both of those procedures rely on non-safety grade equipment because the event that we are dealing with here is an event beyond the design basis.

The Commission after the TMI accident directed the development of procedures for inadequate core cooling, although inadequate core cooling could not result from the design basis analyses which we have performed. In developing that procedure, we used all available equipment.

MR. BAXTER: The panel is available for cross-examination.

CHAIRMAN SMITH: Ms. Weiss, do you want us to go to the procedure that we discussed before lunch, that we will take other questions first?

MS. WEISS: Please, Mr. Chairman.

CHAIRMAN SMITH: Do you want to begin, Mr. Dornsife?

CROSS-EXAMINATION

BY DORNSIFE:

Q Mr. Jones, on page 4 of your testimony you say that the PORV and the block valve have been upgraded -- you don't say that, but that's what it means -- so now they are supplied by emergency power. Do you know what the distribution of that emergency power is, what bus they come from?

To make it more simple, are they from the same bus?

A (WITNESS CORREA) Yes, they are.

A (WITNESS JONES) I don't really remember.

A (WITNESS CORREA) On the emergency diesels, the valves are on the same bus. But if the bus does lose power, the block valve has the automatic transfer to the other bus. The block valve can receive power from either emergency diesel.

Q The PORV itself is not powered from the diesel?

A (WITNESS CORREA) The PORV is on the batteries and the batteries are charged. The chargers have power from the A diesel. As I said, both the PORV and the block valve are on the same diesel. But if that one fails, the block valve can be transferred to the other diesel.

Q Can the PORV -- say it were to fail open. Does it still have the capability of being manually closed, if it were to fail open? Can it be manually operated in addition to its automatic function?

A (WITNESS CORREA) No, it cannot.

Q It only has an automatic function?

A (WITNESS CORREA) It is electrically actuated. If it is open, to close the valve you have to cut the power to the valve and then it is supposed to close. It is supposed to close upon loss of power.

Q So it can't be independently operated of its control system?

A (WITNESS CORREA) No, it cannot.

Q The makeup of the valve, the way I understand it, is pretty much like a regular valve, that you could possibly control it; is that not true? It is not like a passive safety valve? There is some circuitry there that you could conceivably independently manually operate it; is that correct, through the control system?

A (WITNESS CORREA) There will be in Unit 1 a manual key lock switch which will provide for remote operation of the valve. This switch will be administratively -- was controlled, so the operator cannot open that valve any times he wants. They have to be in certain procedures to allow that key switch to be used.

Q From that standpoint, in a sense, there is single failure -- prevention of single failures from not allowing the valve to close.

Let me try to ask it from a different perspective. In the case that the valve were to fail open, there is in a sense single failure-proof, some single failure capability there, to withstand a single failure, in other words, to close off that particular flow path; is that not correct?

A (WITNESS CORREA) Yes, we do have alarms in the control room which will tell the operators that the valve is open, and then they can manually close the block valve.

Q The block valve can be powered from either diesel, so that it is also single failure-proof as far as power supplies are concerned?

A (WITNESS CORREA) Yes, it is.

Q Has Met Ed or B&W ever pursued, or have they been able to discover to any extent why the PORV failed during the accident, what happened to it?

A (WITNESS CORREA) No, we have not. From the Met Ed standpoint, no.?

A (WITNESS URQUHART) No, we have not.

Q I am sure there will be some, once the valve is available for examination, there will be some studies done to determine what the failure was, I am sure.

A (WITNESS CORREA) Yes, there will be, I am sure.

Q There was some confusion when we discussed the accident sequence whether the PORV itself was used to control pressure instead of the block valve. In an attempt to depressurize and remove decay heat, the block valve was being cycled. There was some thought that maybe at some point the PORV was also being cycled.

Does anybody have any thought on that? I know there was some controversy over whether it was actually used. Does anyone know?

A (WITNESS JONES) No, we don't know.

A (WITNESS URQUHART) No.

Q Mr. Jones, a procedure that you quoted from just a second ago, page 25 of 1202-6B --

(Panel conferring.)

Q You said that Step 3 -- you explained the rationale of Step 3. But you also said that Step 3.2 would perform that particular function of depressurization in a much quicker manner than Step 3.3; is that correct?

A (WITNESS JONES) That's correct.

Q Does Step 3.2 -- can you do Step 3.2 without reliance on non-safety grade equipment?

A (WITNESS JONES) I'm not really sure. The general way of doing it would be to use the turbine bypass system, which would be non-safety grade equipment. You could also use the atmospheric dump valves. I am not sure whether they are safety grade or not.

Q Assuming those two components are not safety grade, is that particular procedure any different from 3.3, from that standpoint, its reliance on safety or non-safety equipment?

A (WITNESS JONES) Throughout the inadequate core cooling procedure, we use non-safety grade. So it is not any different than 3.3 using non-safety grade.

Q Are the PORV and block valve environments, are they going to be environmentally qualified prior to restart?

A (WITNESS CORREA) The block valve is environmentally qualified. It is seismically qualified, also. The PORV is seismically qualified. And I believe that the solenoid operator is good up to 356 degrees Fahrenheit.

The control circuitry for the valves, for the block valve, it is environmentally and seismically qualified. For the PORV it is environmentally qualified.

MR. DORNSIFE: I have no further questions.

CHAIRMAN SMITH: Mr. Cutchin?

MR. CUTCHIN: I have no questions of these witnesses, Mr. Chairman.

(Pause.)

BOARD EXAMINATION

BY DR. JORDAN:

Q Mr. Pollard points out on page 5.6 of his testimony, a footnote which reads as follows: "A single failure in the PORV circuitry could cause the PORV to open inadvertently. I have noted that, although NUREG-0578, Section 2.1.2 specifically calls for qualification of the control circuitry associated with the PORV, restart evaluation for TMI page C8-10 does not include this requirement."

Have you observed this note of Mr. Pollard's, and have you a response to that?

A (WITNESS CORREA) This footnote actually refers to two separate items. The first is the single failure. There is a possibility that a single failure could open up the valve. It is more likely that the single failure in the control circuitry would cause the valve to lose power. Upon loss of power, if the valve is in the closed position it stays closed; if it is in the open position, it is supposed to go closed.

It is more likely to have an open circuit, which would cause a power loss to the valve, than a short circuit, which would cause power given to the valve.

The second item that it refers to is, I believe, in the restart report, the evaluation of the relief and safety valve test program. This is an item that the staff has commented on also to the EPRI staff, that the control circuitry is not included in the EPRI test program. I believe the EPRI position is that those components associated with the valve itself will be included in the test program. Those components associated with the valve installation at a specific plant are the plant's responsibility.

Q We have had this matter before us before, previously. The PORV indeed does -- is indeed part of the boundary of the primary. So in the sense that it is part of the boundary, it should be safety grade. Is it safety grade, in that sense? Has it qualified as safety grade, in that respect? Can anyone speak to that?

A WITNESS URQUHART) The PORV as it pertains to being a pressure boundary device is fully qualified in accordance with the appropriate requirements. That is, it has been designed, fabricated, and analyzed in accordance with ASME code as a Class 1 valve. It has also been seismically analyzed to assure that its pressure boundary integrity would be maintained during a seismic event.

Q Nevertheless, as you say, the circuits are not -- the circuits which control the opening and the closing are not safety grade, in that they do not meet the single-failure criteria.

A (WITNESS CORREA) That is true, but we do have the flow indications downstream of the PORV to show that it is open, and we do have the block valve to close that flow path.

Q I can't remember, but I suspect strongly it was Mr. Pollard that pointed out that the valves which are being added in the pressure vessel head to relieve gases, for example, are qualified safety grade in a different way than -- and that their control circuitry is safety grade. Is this part of Mr. Pollard's testimony? Can anybody check me on that?

A (WITNESS JONES) I believe what you are referring to is his quote on page 5-8 of his testimony.

Q Thank you.

A (WITNESS JONES) My understanding of the vent criteria is they will be manually controlled from the control room. But I believe the Commission has rescinded the single-failure proof part of the statement in their recent clarification.

Q That would be interesting to see, then. We can check that by looking at NUREG-0737. That has been changed?

A (WITNESS JONES) That is my understanding.

Q All right. If anybody has evidence to the contrary, let him speak now or later.

MS. WEISS: I would just like to say I don't know one way or the other. So my failure to speak up does not indicate --

DR. JORDAN: I understand. I mean now or later. I had you in mind.

BY DR. JORDAN:

Q So far as meeting the other requirements of section 2.1.2 of NUREG-0578, do you think otherwise you do meet all the requirements? Have you looked at them carefully and can you testify that the requirements have been met?

A (WITNESS CORREA) Are you referring to the item on page 7, item 2.1.2, performance testing for the valve?

Q Yes.

A (WITNESS CORREA) We meet all the other requirements in that area.

Q Mr. Jones, on page 2 of your testimony --

A (WITNESS CORREA) Excuse me, Dr. Jordan. In reading this a little more carefully than when I just skimmed through it, there is some controversy between the EPRI staff and the NRC staff about the two-phase flow testing. That is being resolved. On revision 1 of the EPRI test program, I believe that the staff has reviewed it, and they had six specific comments which the EPRI staff is trying to resolve for this test program.

Q I see. I hadn't come to that part, yet, but that is helpful. Thank you. On the last paragraph on page 2, you start off by saying: "The original design function of the PORV was to provide pressure relief capability which, in connection with the plant control system actions to reduce reactor power and/or adjust steam generator feedwater flow, would prevent a reactor trip on high pressure -- on high primary system pressure during various operational transients."

I was wondering if you could describe some of those transients.

A (WITNESS JONES) Well, one of the transients that the plant was originally set up to handle without causing reactor trip was a turbine trip. That has since been alleviated by the installation of the direct reactor trip function on turbine trip. The way the transient would have progessed, it would have had a turbine trip that would have bottled up the steam generator. The steam generator would increase in pressure, the primary system would heat up to approximately 2300 p.s.i. or so and would open the PORV and would relieve steam.

In the meantime, the integrated control system would start a rod insertion at a controlled rate, along with a feedwater runback, to decrease the plant power to roughly 15 percent, and then control it at that stable power level.

Q 15 percent?

A (WITNESS JONES) Yes.

Q And you control that by bypassing the turbine?

A (WITNESS JONES) Yes. And it could handle other types of operational transients, such as small changes in feedwater flow or a loss of a single feed pump. And it is basically the same kind of action: reduce power to certain values.

Q Yes. I think you do mention that transient later. Is that the chief transient that you are referring to at that point?

A (WITNESS JONES) The chief one I was referring to in there was the turbine trip. There were the others, but the turbine trip was one of the original features that we were trying to handle with the control system.

Q All right. So you say that is the original design function. Was that, though?

A (WITNESS JONES) Yes.

Q If that was the original design function and it no longer meets that design function, why don't you just block it out, leave it out?

A (WITNESS JONES) As I stated earlier, there are plants that run a lot of times with the PORV shut -- I mean with the block valve shut in the path. Following the TMI accident, that was one of the concepts which were brought up, but the Commission wanted the PORV to remain functional, if possible, and to provide that cushion to the safety valves for, in my opinion, better defense in depth.

Q So it is a protection for the safety valves?

A (WITNESS JONES) Yes, it can provide that. But again, while the Commission wants it, they have not imposed criteria, to my knowledge, where you keep the PORV open continuously.

Q But in view of the relative sizes of the valve, it really doesn't provide much backup in the event of a severe pressure transient; isn't that true? An atmosphere event, for example, ATWS.

A (WITNESS JONES) For an ATWS event, you use all of the valves, the PORV and the two safetys. That's how it's been analyzed. But its capacity, I guess, is roughly one-third of a safety valve.

Q I see. It's helpful to put it in those terms.

Are the safety valve capacities adequate to handle an ATWS event?

A (WITNESS JONES) I don't really know. ATWS is still an area that is still under generic review. I don't know where it's going.

Q We're not going to go into ATWS events. I think there has been some question as to whether it is or not, but that is outside of the scope.

CHAIRMAN SMITH: While Dr. Jordan is going over is notes, I want to ask Mr. Correa if he can clarify part of his testimony.

BY CHAIRMAN SMITH:

Q You were describing the likelihood of PORV failing open. You spoke in terms of it less likely to have a short circuit than a failure of power. That still leaves it open; that still leaves it unbounded in the likelihood. I was just wondering if you could add to your testimony some likelihood that gives you some measure of it. We don't know what the likelihood of it losing power is, either.

A (WITNESS CORREA) I don't have that, sir. I don't have those numbers.

Q Maybe I just don't understand your answer.

A (WITNESS CORREA) The PORV control circuitry is basically not single-failure proof. The valve is to close upon loss of power. If, for some reason, in the control circuitry or in the pressure transmitter that power is lost, then the valve is supposed to close. The way that the single failure could cause the valve to open is if there is a short circuit in one of these controls which would cause power to go from the pressure transmitter to the controller and then to the valve.

Q You have no way of knowing what the likelihood of that is?

A (WITNESS CORREA) No, I do not.

Q So your comparison to the loss of power didn't really intend to indicate the probabilities of that event?

A (WITNESS CORREA) No, it did not.

BY DR. JORDAN:

Q Has the change in set points between the reactor protection circuits and the PORV led to an increase in frequency of challenge to the reactor protection system? Does anybody know about that?

A (WITNESS JONES) Yes, it has.

Q How significant; have you any feeling at all? I know this will get us into the ATWS discussion again.

A (WITNESS JONES) I don't remember the exact numbers. There were some looks at the plant data following the change in the set points to look at the increased trip frequency. And on an overall average, the trip frequency would not cause us to exceed the design limits that we had set up for the -- or the projected number of reactor trips per year, which was 10 per plant per year, or 400 trips over the plant life. It did cause it to go up somewhat, and the numbers that I saw put the frequency more around the industry average than it had been previously.

Q It's not a large increase, then, in the absolute number of trips?

A (WITNESS JONES) I think it increased about 10 or 15 percent.

Q All right. Fine. That's what I thought.

You mentioned the Crystal River event, which was the opening of one of the safety valves. Do you know, was this just a failure in the safety valve and it opened at lower than normal pressure; or was it an increase in pressure, and, if so, wouldn't the PORV have opened first? Can anyone enlighten me on that?

A WITNESS URQUHART) During the Crystal River 3 event, the PORV was open, but it was blocked off during the transient. After it was blocked off, the -- and the HPI was continued, the set point on the safety valve was reached, although that was light, as I testified before.

Q So the block valve was operated, but the HPI continued then to cause discharge from the safety valve?

A WITNESS URQUHART) That's true.

CHAIRMAN SMITH: Be careful when you use the term "light." It also sounds like "late," and that would be different.

WITNESS URQUHART: Yes.

BY DR. JORDAN:

Q It is a new term to me.

Perhaps you have already addressed my next question, which was on page 8. The top paragraph that begins on that page, the last sentence says: "General design criteria are applicable to the PCRV only to the extent that it forms part of the reactor coolant boundary." My question that I had noted here is: So what does -- what is required by the general design criteria? And you said that was to meet ASME codes. Is that all that is required? I believe that is what the answer was.

A WITNESS URQUHART) Yes, it was. The valve was designed in accordance with the code.

Q Does that meet the general design criteria?

A WITNESS URQUHART) Yes, it does.

MR. DORNSIFE: I am taking up your invitation of speaking later. I have been looking at NUREG-0737 concerning the --

DR. JORDAN: Good for you.

MR. DORNSIFE: My looking at it, it doesn't necessarily agree with what the witness said.

DR. JORDAN: You have a little problem with that?

MR. DORNSIFE: Yes. If you have the document, on page 3-56 is where it is discussed.

DR. JORDAN: Give us time.

(Pause.)

WITNESS JONES: Which page was that?

MR. DORNSIFE: 3-56, paragraph 7, A-7. Redundancy is discussed. I think A-4 has some applicability, too.

DR. JORDAN: Just a moment.

CHAIRMAN SMITH: It is a short paragraph. Perhaps it might be helpful to read it into the record at this point.

MR. DORNSIFE: "Since the reactor coolant system vent will be part of the reactor coolant pressure boundary, all requirements for the reactor pressure boundary must be met. In addition, sufficient redundancy should be incorporated into the design to minimize the probability of an inadvertent actuation of the system. Administrative procedures may be a viable option to meet the single-failure criteria." That is the applicable portion.

CHAIRMAN SMITH: One more sentence.

MR. DORNSIFE: That is not in the context.

CHAIRMAN SMITH: You left out a word in the first sentence. "Since the reactor coolant system vent will be part of the reactor coolant system pressure boundary..." -- you left out the word "system."

MR. DORNSIFE: Paragraph 4 also has some applicability, the first sentence, particularly: "Where practical, the reactor coolant system vent should be kept smaller than the size corresponding to the definition of a LOCA." That takes it out of the realm of a PORV opening.

CHAIRMAN SMITH: Did you agree with that, Mr.

Jones?

WITNESS JONES: Let me add to that, you did not read all of 4. I think the remainder of 4 puts you right in line with the general design for the PORV. If you continue clarification number 4, it says: "On the PWRs the use of new or existing lines whose smallest orifice is larger than the LOCA definition will require a valve in series with the vent valve that can be closed from the control room to terminate a LOCA that would result if an open vent valve could not be reclosed."

That is essentially what you have with PORV and the block valve. It is totally consistent. Again, "Use of administrative procedures," in item 7, "may be a viable option to meet the single-failure criteria." That is what I was referring to in my earlier testimony. They have eliminated the direct necessity to automatically meet the single-failure criteria. You can meet it with administrative controls.

DR. JORDAN: Do you have further questions?

MR. DORNSIFE: No. Thank you very much. That does clear it up. It's good to do it now.

WITNESS JONES: Excuse me. I found one other point. If you go to page 3-55, item 4, it says: "Changes to previous requirements and guidelines on the specs." It says, on item 4: "Delete requirement of September 27, 1979, letter from Vassallo to Applicant, stating that 'Vents shall satisfy single-failure criteria of IEEE 279.' Vent systems are not required to have redundant pads. Degree of redundancy should be provided by power in different vents from different emergency buses."

MR. DORNSIFE: Dr. Jordan, I do have a question on that, concerning that.

DR. JORDAN: Fine.

CROSS EXAMINATION

BY MR. DORNSIFE:

Q Would you say paragraph 4 you just read from talks more about its safety function of providing a vent path rather than preventing a LOCA?

A (WITNESS JONES) I believe that's what item number 4 was referring to.

Q Paragraph 4, the next page, when you read beyond what I read, isn't the failure, when you say that it is adequate to meet the single-failure criteria by another valve, doesn't the failure of the one vent valve automatically assume the single failure and the other vent valve being powered from a different diesel would then satisfy the redundancy it is talking about?

A (WITNESS JONES) Well, it doesn't directly state in that line in that item that they have to be powered from separate diesels. It is possible, if you were using that sort of a system, that may be required. But again, what I was trying to point out with the remainder of that paragraph is that the Commission's decision here is very similar to the position on the PORV, that you have the PORV and block valve combination.

(Board conferring.)

REEXAMINATION BY THE BOARD

BY DR. JORDAN:

Q On page 11 you describe the test program. And you say in the third paragraph: "Effort is underway to evaluate the effects of postulated valve failure modes on reactor system performance in order to establish preliminary acceptance criteria and guidelines for evaluation of the significance of the valve test results."

I would like to understand a little bit what is meant by this and how you describe the "effort that is underway." I don't understand really, "to establish preliminary acceptance criteria and guidelines."

A (WITNESS CORREA) I believe that EPRI has slightly changed this. What they have now is valve screening criteria.

Q What?

A (WITNESS CORREA) Valve screening criteria.

Q Screening?

A (WITNESS CORREA) Screening criteria for the valve test results. The screening criteria is that the valve opens and stays open when it is supposed to; the valve closes when it is supposed to; and the valve sustains no internal or external damage which would prevent it from operating on the next actuation.

If a valve falls outside of these guidelines, then the plant that has that valve, the NSSS supplier of that plant and the valve manufacturer are all notified of a possible defect in the valve; and then it is up to them to evaluate whether this possible defect will affect their plant operation.

Q I see. Is there not also -- or is it perhaps part of the EPRI program -- is there not a program at TVA to test valves, safety valves? Are you aware of the nature of the program?

A (WITNESS CORREA) Not at TVA, no.

Q Not at TVA.

A (WITNESS CORREA) The EPRI test program does have three test facilities. It has the Marshall station from Duke Power. We are also using the Wylie facility at MARCO, and the CE facility is being modified for testing in the early part of 1981. And the CE facility is in Windsor, Connecticut.

Q One of the review plan -- the action plans -- does describe a TVA test program, but it must be something else.

It doesn't matter anyhow.

On page 12 of your testimony you say: "B&W has supplied operational transient and postulated accident sequence data to EPRI for the 177 fuel assembly reactors; namely, the TMI-1 type. This data is being used in defining test parameters for the EPRI test matrix."

Can you tell me the nature of this data that is being supplied to EPRI, what it says?

A (WITNESS JONES) The information being supplied to EPRI is the results of various analyses which have been performed for the normal safety analyses of this generic plant type. They include such items as a loss of main feedwater transient, a turbine trip, loss of electric load. I believe we have supplied them with a stuck-open PORV transient. And there are a few others.

Specifically, the type of stuff that is given is information as to the flow rates, the pressurization rate within the reactor coolant system, and the fluid qualities during these transients that would be exiting through the valve.

Q Okay. Is there any statistical information from past experience included in this, do you know?

A (WITNESS JONES) I don't believe so.

DR. JORDAN: I have some questions on Mr. Pollard's testimony, but I believe that Mr. Baxter has covered my questions there pretty well. So I believe that's all I have for these witnesses.

CHAIRMAN SMITH: Other than Ms. Weiss, are there any questions?

MR. CUTCHIN: I have one follow-up question based on some of Dr. Jordan's questions.

REDIRECT EXAMINATION

BY MR. CUTCHIN:

Q When you were discussing, Mr. Correa, with Dr. Jordan the fact that the operator, the solenoid operator for the PORV would be, I believe you said, qualified or is qualified in the EPRI program, but that the responsibility for qualification of the control circuitry associated with that valve falls to the owner ofthe plant, are you familiar with the requirements of Inspection and Enforcement Bulletin 79-01B relating to environmental qualifications?

A (WITNESS CORREA) No, I am not.

Q Then I can't ask my question.

MR. CUTCHIN: Thank you.

DR. JORDAN: You didn't introduce at this time, or did you, the license testimony of Correa and Urquhart in response to Board question on UCS Contention 6?

MR. BAXTER: Yes, I did do that.

DR. JORDAN: I neglected to go through that. I think I have hardly any questions.

BOARD EXAMINATION (Continued)

BY DR. JORDAN:

Q You mentioned again the Crystal river incident; following the incident was there an inspection of the safety valve?

A (Witness Urquhart) Yes, there was. The valves were removed from the plant and they were sent to a laboratory where they were first put on a bench tester where they were tested to assure that they still functioned as they are required to function.

They were tapped open three times; each time they opened at approximately 5400 psig which is where they opened on the plant.

The valves were then disassembled and subjected to a visual examination, the results of which I mentioned in my testimony. Briefly, the valve was in good shape with some damage to the disk; that is the pressure retaining portion that lifts off the seat due to steam cutting which mostly likely resulted from leakage prior to the incident.

There was some small leakage through the valve prior to the incident. The bellows -- the so-called bellows seal valve was also damaged, but it appeared to be an assembly problem not related to the incident.

Neither of these -- neither the steam cutting or the damaged bellows in any way affected the ability of the valve to function.

Q Those valves are being put back in at Crystal River?

A (Witness Urquhart) I believe that valve was refurbished. It is either on the plant or is now being used as a spare. It may very well be used as a spare now.

Q I see.

(Pause)

Certainly, the prime function of the valve is to relieve pressure in case the pressure gets too high. You mentioned that there has been a fair amount of experience from the fossil industry.

Have there been enough occasions -- have there been a significant number of failures in the fossil industry that you know what the situation there is?

A (Witness Urquhart) In my discussions with two manufacturers of safety valves -- that is, Dresser whose valves are on the TMI-1 plant and Crosby who still supplies a good deal of safety valves to the nuclear industry and also to the fossil power industry -- they do not know of an instance where a properly maintained and set valve has failed to open when it is required to open.

That is, that it has completely stuck shut and has not performed its protective function.

Q What is involved in properly maintaining a valve to make sure that it will open when called upon?

A (Witness Urquhart) In a nuclear power plant during each refueling, at least one of the valves is removed from the pressurizer. It is put on a bench test to assure that it is still functional.

Q About once a year?

A (Witness Urquhart) Approximately.

Q So the cycle -- about once every two years for each valve, then, I suppose.

A (Witness Urquhart) Yes, depending on the length of the fuel cycle, approximately that.

(Pause)

Q One of the things that I did ask for was: is there reasonable assurance that the tests will be successful; that is, that there is good evidence that the valves will indeed perform in an accident environment and just how do you address that?

What is your response then? There is a test program? I would like some assurance that the test program would indeed be successful.

A (Witness Urquhart) I think with respect to the safety valve, the history of safety valve usage, not only in the nuclear industry, but in the fossil power industry also, provides, I believe, assurance that valves will indeed function when they are called upon to function, which -- of course, the critical action there is that they open when they are called upon to open.

I believe there is very good evidence and experience with -- does provide assurance that these valves function properly in the test program.

Q You are not concerned that the test program is going to open up something that had not been thought of before?

A (Witness Urquhart) I do not believe so. As far as steam flow, I think the valves have been -- valves of this design and similar designs have been in use for many years and have worked very well.

Other flow conditions such as water, there is some limited amount of experience. In fossil supercritical units there have been occasions where they have been pumped up, and the valves have relieved the water.

They have functioned; that is not to say that they have been leak proof, leak tight when they reseated. In many cases leakage -- there will be leakage after a valve has passed water.

In the case of Crystal River, however, the increasing leakage after the valve did pass water was insignificant. The valve was leaking somewhat prior to the event. It flowed 700 -- I believe it flowed water at the rate of 700 gallons per minute for 20 minutes and reseated and upon reseating,it was leaking at about 1.1 gallons per minute.

And upon visual inspection, there was really no damage that could be attributed to the water flow through that valve.

Q You anticipated essentially my last question; that is, will the testing include two phase as well as water flow?

A (Witness Urquhart) My knowledge of the test program -- the program would include water. I am not sure to what extent it will include two phase flow, as Mr. Correa mentioned before. But I know it will include water flow through those valves.

CHAIRMAN SMITH: Ms Weiss, what is your position on continuing? There is a possibility that you might take into consideration; always before we have followed the cross examination plan and I have noted a question down there.

Even if it might have occurred, we have not asked you. It could be that after you exhaust your questions,that if you have no objection, Dr. Jordan could redo cross examination,and you could suggest questions to him, too. Don't take any pressure to take that course. Cross examination is more than just asking questions, I realize. Just take that into account.

MS. WEISS: I do not have any objection to Dr. Jordan looking at our questions.

DR. JORDAN: I have reservations about the suggestion. Mr. Pollard is very much better at knowing what he has in mind in picking up things than I am.

MS. WEISS: I would like to go as far as I feel comfortable, and I cannot -- there was extensive rebuttal, and I know that I am not competent to deal with that at this point.

I will go as far as I can on the cross examination.

DR. JORDAN: Let's have a short break. And I appreciate Ms. Weiss's offer to go as far as she can.

MR. DORNSIFE: I have one more follow-up.

CROSS ON BOARD EXAMINATION:

BY MR. DORNSIFE:

Q Mr. Correa, you testified -- you said that the block valve is environmentally qualified. it is powered -- it can be powered from either diesel. As far as a pressure boundary component, it is safety grade.

In your engineering judgment, how much would its reliability be improved, and what would be necessary to upgrade it to safety grade, and how much would it improve its reliabiility over the way it is currently -- will be installed prior to restart?

A (Witness Correa) I believe that is an item that I am not able to answer right now. I would have to look at more of the systems involved to see exactly what interactions making the block valve safety grade would have on other systems in that area.

Q Do you -- can you qualitatively address it?

Do you have an idea of what would be necessary to upgrade it?

Does it meet the safety grade criteria right now?

A (Witness Correa) It meets the criteria as a pressure boundary part in that it is seismically qualified. It has been built to the ASME code. To do more upgrading, we have to lookat the requirments of 279 for the single failure criteria, possibly having two valves in series, things like that.

Q Can the operator change the power supply from either diesel in the control room or is that done locally at a panel at the switch gear?

Do you know?

A (Witness Correa) I believe to go on the A diesel it is automatic. And then to go on to the other diesel, it is a swtich in the control room.

Q It is fairly accessible to the operator?

A (Witness Correa) Yes, it is.

DR. JORDAN: Which is on battery?

WITNESS CORREA: The PORV is on battery. That is a DC valve.

DR. JORDAN: Why is that? Do you need a particulary reliable supply for the PORV under certain circumstances?

WITNESS CORREA: That is the way the plant was originally designed. I do not know the reasons for the original design of the plant.

CHAIRMAN SMITH: Let's take a 10 minute midafternoon break.

(Recess)

MR. BAXTER: Mr. Chairman, I have a follow-up question on something Dr. Jordan raised.

REDIRECT EXAMINATION

BY MR. BAXTER:

Q The question of the EPRI valve testing program, Mr. Correa, since your response to the board question on UCS contention six was filed on October 28, 1980, has there been any testing accomplished so far by EPRI that is relevant to the valves at TMI-1?

A (Witness Correa) Yes, there has. This week the testing of the dresser PORV started at the Marshall Steam Station. This testing is on steam. The testing started Monday.

But they had slight problems with the power supply to the valve. The valve takes a fairlyhigh inrush current, and the EPRI power supply did not have a high enough current. So they modified it Monday night, and on Tuesday they actuated the valve without any problems.

They actuated the valve 14 times. The first four actuations were to shake the system down, and then they did a one minute blow at 400 psig backpressure.

Then they checked seat leakage, and there was zero seat leakage. And then they did four 10 to 15 second blows; again at 400 psig pressure.

MS. WEISS: I did not hear. They did four what?

WITNESS CORREA: Four blows, 10 to 15 seconds each. The seat leakage at that time was 50 milliliters per minute. After this series of five tests they did another series of five tests at 160 psig backpressure.

The seat leakage at the end of that series of tests was zero gpm. The flow through the valve at the 400 pound backpressure was 156,534 pounds of steam per hour. At the 160 pound backpressure, it was 156,259 of steam per hour.

The dresser calculated flow for the valve was 157,000 pounds of steam per hour. The opening time of the valve was 170 milliseconds, and the closing time was slightly slower, but not very much; it was in the range of 200 to 300 milliseconds.

The only problem which they found -- and this problem did not prevent valve operation -- was on one of the first actuations of the valve, the bellows on the pilot stem seal ruptured.

It allowed steam to leak from the pilot area to the atmosphere. As I said, this did not affect valve operation. This is only preliminary EPRI data. I was called on this by the EPRI test manager for the Marshall test.

He called me as soon as the testing was done yesterday to describe what had happened. The report on these valves will be issued, I believe it is March of next year, the full report for the Marshall testing.

In the meantime also EPRI will be looking into the cause of the bellows -- of the bellows rupture to determine if it is a generic defect or if it is a manufacturing defect only affecting this bellows.

MR. BAXTER: Thank you.

BOARD EXAMINATION (Continued)

BY DR. JORDAN:

Q This brings up something I had never considered. Why are they tested with various steam backpressures. I would have thought maybe they would test them only to atmospheric backpressure.

A (Witness Correa) They discharge into a drain tank, and that causes the backpressure on the valves.

Q It is more realistic to have a backpressure on them.

A (Witness Correa) Yes, it is.

DR. JORDAN: I see. all right.l

CROSS EXAMINATION

BY MS. WEISS:

Q On page 4 of Mr. Jones's testimony, at the top you describe the changes being made to enhance the operator's ability to detect and terminate a transient caused by a stuck open PORV.

Is all the instrumentation which you describe safety grade, and if not, could you tell me which is and which is not. That also goes to any circuitry involved.

(Pause)

A (Witness Jones) I believe there are going to be control grade indications. The differential pressure transmitter has been quaLified for operation in the post-LOCA environment.

That is, the dicharge line flow measurement instrumentation, and I am referencing in my testimony that it has been qualified to operate in the post-LOCA environment and operate after a seismic event.

The accelerometer is part of the loose parts monitoring system, and it has been seismically teted and has been environmentally qualified also.

Q For what environment?

A (Witness Jones) The accelerometer is steam line break and small break LOCA qualified.

Q With those exceptions, the instrumentation is control grade rather than safety grade?

A (Witness Jones) That is my understanding.

Q Can I direct your attention to NUREG-0578. Do you have a copy in front of you?

Do you have a copy of the document?

(Pause)

A (Witness Jones) I have some excerpts. If I have the right exceprt I will have it in front of me. Otherwise, I will have to get a copy.

Q Page 7, section 2.1.3, information to aid operators in accident diagnosis and control.

A (Witness Jones) Yes, I have it.

Q Recommendation A is for a direct indication of power operated relief valves and safety valve position for PWRs and BWRs; it then describes that such a direct position indication -- I will not go into the description.

Will TMI have a direct indication of the PORV and safety valve positions in the control room?

A (Witness Jones) Yes. And those are the accelerometer flow and the discharge lie flow measurement instruments because the position states that you can have either a direct position indicator or a reliable flow indicator.

The accelerometer is a device that senses the flow down the line which is an indication of whether the valve is open or closed and the elbow taps will be able to provide information as to whether there is flow going through the lines.

Q You meet that position by having flow indication devices rather than direct indications of valve positions? Correct?

A (Witness Jones) That is correct.

Q What changes have been made, if any, to the design of unit 1 to enhance the ability to terminate a LOCA caused by a stuck open PORV as opposed to changes mad to enhance the ability to detect that it is stuck open?

(Pause)

Perhaps I should be more specific with particular reference to the block valve. Has the block valve or valves been modified for two phase flow?

A (Witness Jones) Not to my knowledge. I guess to answer the question that you asked, I was just trying to run it through my mind.

It is my understanding that given a stuck open PORV that basically there is no changes made to the term ation of that. The physical aspects of terminating the [illegible] -- there is of course the training changes, the fact that we have reduced the probability of hitting the PORV in the first place.

Q The control circuitry for the block valve does not meet the single failure criteria; is that correct?

A (Witness Jones) I do not know.

Q Is the block valve environmentally qualified for any particular condition?

A (Witness Urquhart) I can answer that. The block valve -- the operator on the block valve is qualified, I believe, for 300 degrees fahrenheit, on the order of 2×10^8 or as far as radiation dosage.

Q Do you know if it meets the single failure criterion, the control circuitry for the block valve?

A (Witness Urquhart) I do not know if the control circuitry meets the single failure criterion or not.

Q This is in Mr. Urquhart's part of the testimony. I would like to direct your attention to NUREG-0737 if you have got it, the clarification of the Action Plan requirements.

Actually, this may be Mr. Urquhart or Mr. Correa. In particular, section II.D.1: I guess that is on page 3-72. I want to confirm if I can whether that description of the testing program conforms to current plans as to scheduling and scope of the testing program.

A (WITNESS CORREA) Are you referring to the clarifications listed under 2.D.1?

Q Yes.

A (WITNESS CORREA) As far as item A goes, I believe that the EPRI test program meets that. Item B, the qualifications -- yes, the qualification of the block valves, since this is a new requirement EPRI is still studying this to try to get it into the test program. They are having discussions with NRC staff on it.

Q And is the schedule for ATWS testing still to be completed by July 1981?

A (WITNESS CORREA) There is no ATWS testing at the moment. Even in 737, there is no ATWS testing.

Q Now, the item, then, under "Clarifications," directly beneath what we have just gone through, Item A calls for performance testing of relief and safety valves, and then lists a set of -- or lists some information which is required to be provided by October 1, 1981.

My question is, is that still the current schedule, and does that properly describe the scope of the program?

A (WITNESS CORREA) If you would just give me a moment, I have to look at the latest EPRI status on this.

Item A-1, it is still the schedule. For Item A-2, it is still the schedule.

(Pause.)

A (WITNESS CORREA) Item A-3, yes.

CHAIRMAN SMITH: Is "yes" a full answer.

WITNESS CORREA: It is going to meet the October 1981 schedule, which is what she asked, I believe.

CHAIRMAN SMITH: Your question was too parts?

BY MS. WEISS: (Resuming)

Q Yes. I assume you also mean that is still an accurate description of the EPRI program?

A (WITNESS CORREA) This is not intended, I believe, to be a description of the EPRI test program. It is a description of what they require. And yes, the EPRI program will meet with what the NRC requires.

Q You mean the EPRI test program will go beyond it? I am trying to understand whether it would be in any way consistent -- either less broad or inconsistent?

A (WITNESS CORREA) As far as clarification A, the EPRI program is consistent with this clarification except for six items, and I am not sure what the exact six items are. These are items that are under discussion between EPRI and the NRC staff.

Q I take it the NRC staff has not yet approved the test program? There are still some six outstanding items that remain to be resolved; is that correct?

A (WITNESS CORREA) Yes. Some of these items are minor, like for example one of the items was that the EPRI test program as submitted in July did not fully represent all of the safety and relief valves that were installed in the various plants.

Since that time, EPRI has expanded its test program to include more valves.

Q Do you know what any of the other open items are between the NRC and EPRI?

A (WITNESS CORREA) Offhand, I would have to refer to the EPRI information, which I have back in the office. I do not have that right here right now.

Q If you have a chance to do that before you appear again, because the panel will be back, I would like to have the information on exactly what the nature of the open items is.?

A (WITNESS CORREA) All right.

MR. BAXTER: He's not going to be back in his office tonight, though.

MS. WEISS: I am not even sure whether we will be here tomorrow, but Monday.

BY MS. WEISS: (Resuming)

Q Item number 8 on page 3-73 relates to qualification of PWR block valves. Is there a date for completion of that program?

A (WITNESS CORREA) This item is under discussion between EPRI and the NRC staff right now. Due to funding limitations and time constraints to get the safety and relief valve qualification testing done by July 1981, these block valves put a very big restriction into that test program. And as I said, EPRI is discussing this requirement with the staff.

When the resolution of this requirement comes about, I really don't know.

Q Is it EPRI's position that it cannot complete the program for qualification of block valves by July 1, 1982?

A (WITNESS CORREA) I would have to again look at the latest EPRI submittals to the participating utilities to determine what the EPRI position is on that item.

Q Again, I will ask you to check that.

(Pause.)

Q On page 5, Mr. Urquhart, of Mr. Urquhart's testimony, at the top you state that the PORV and the safety valves have been properly designed and tested pursuant to applicable criteria. I think you answered, in response to Dr. Jordan, that those were the reactor pressure boundary criteria.

Can you tell me specifically what GDC those are?

A (WITNESS URQUHART) Let me check. I believe they would be, as far as the pressure boundary is concerned, they would be Criteria 1, 14, 15, and 30.

Q I think that's right. But you can check it if you want.

A (WITNESS URQUHART) I just looked.

(Pause.)

Q You discussed the incident at Crystal River on page 10 -- excuse me, page 6 and 7 of your testimony. You state it was a valve similar to the one at TMI-1. Were they both the same manufacturer?

A (WITNESS URQUHART) Yes, they are.

Q Can you describe for me any differences between the two valves?

A (WITNESS URQUHART) To the best of my knowledge, I don't believe there are any design differences between the valve at Crystal River 3 and TMI-1. I believe they are both the same model of Dresser pressurizer safety valve in both locations.

I am not aware of any real design differences between the two valves, other than they were manufactured at different times.

Q Do you know if the same materials were used in the fabrication of both?

A (WITNESS URQUHART) I would say generally yes, the same materials. The bodies of the valves are stainless steel. They are generally an austenated stainless steel construction. I don't know that there is any difference in the material of construction of the two valves.

Q Can you tell me how long Crystal River Unit 3 has been operational?

A (WITNESS URQUHART) I believe it has been since 1970, just the best of my recollection.

Q You state that the leak rate after the event at Crystal River was 1.1 gallons per minute. Do you know what the leak rate was prior?

A (WITNESS URQUHART) I don't know specifically what the leak rate was prior. All I know was that it was leaking to some extent prior to the event.

Q Is there an allowable leak rate for that valve?

A (WITNESS URQUHART) Not specifically for the valve. But the plant technical specifications allow that you have ten gallons per minute known leakage out of the reactor coolant system. That can be either from, for example, a safety valve or any other source of leakage from the reactor coolant system.

Q Do you have any evidence that might lead you to an opinion of the magnitude of the leak rate prior to the accident?

A (WITNESS URQUHART) I don't know specifically what the magnitude was. I really don't know, other than it was leaking prior to the event.

Q You stated that -- let me ask you this. How many times during the Crystal River accidents did the safety valve open and close?

A (WITNESS URQUHART) To the best of my recollection, I believe it only opened once. I believe it opened and it stayed open for that 20 minute time period, in which it was going at approximately the rate of 700 gallons per minute of water. That's the best of my recollection.

Q You stated that you found, upon visual inspection, steam cutting of the disk and a damaged bellows assembly. Do you really have way of knowing when that damage occurred and why it occurred?

A (WITNESS URQUHART) The steam cutting of the disk -- as I said before, the valve was leaking somewhat prior to the event, in which case steam cutting of a disk would be a fairly prevalent type of damage. High-velocity steam going through a small leak path on the disk would tend to cause erosion of those materials.

The damaged bellows -- it was evident from the inspection that the damaged bellows very much appeared to be an assembly problem, an alignment problem between the bellows and the disk itself.

Q You mean when the valve was originally installed in the plant that damage was done?

A (WITNESS URQUHART) When the valve was originally assembled, put together. Not necessarily installed in the plant, but when the valve was originally put together, it appeared that there may have been some misalignment between the disk and the bellows assembly that caused the damage.

The bellows was not non-functional, let me put it that way. To accurately describe it, the nose of the bellows was extended. I don't know if that makes it any clearer, but it was definitely not due to an actuation of the device. It was due to some alignment problem between the disk and the bellows proper.

Q Was that apparent when you looked at the valve or did you have to take it apart?

A (WITNESS URQUHART) You have to disassemble the valve to see it.

(Pause.)

Q On page 8 of the testimony, Mr. Correa, you state that the spare valve, the spare PORV, will be installed in Unit 1 prior to restart. I am curious as to why they are doing that. What's wrong with the one that is in there now?

A (WITNESS CORREA) There is nothing wrong with the one that is in there now.

Q Why are they putting a spare?

A (WITNESS CORREA) The spare valve has been modified to incorporate the latest manufacturer's seat design on the main disk, which will provide a more leak-tight valve.

(Pause.)

Q And I understand that prior to restart you are going to test the spare valve; is that correct?

A (WITNESS CORREA) It will be actuated, yes, and the basic purpose of the actuation is to ensure the -- to ensure that the valve is functional and to test all of the downstream instrumentation.

Q Would you describe the test for me, how many times you are going to open and close the valve and how you test downstream?

A (WITNESS CORREA) The test procedure for this item has not been written yet. It is a restart item and still has to be done.

Q On page 12, by way of, I guess, summary, you state that: "The TMI-1 pressurizer relief and safety valves have been appropriately designed and tested." Quote. I just want to make sure I understand the conditions under which they have been tested.

Is it true that neither of those valves have been tested for two-phase or water flow?

A (WITNESS URQUHART) I would say that's true, neither valve has been tested specifically for water or two-phase flow, which will be accomplished during the EPRI test program.

Q Describing the changes in the EPRI program, I believe it was Mr. Correa, you discussed new valve screening criteria. I believe you stated that the screening criteria will require -- or defines as success that the valve opens and closes and sustains no damage sufficient to prevent it from operating on the next demand.

And the you state, if it doesn't pass, that EPRI will notify the manufacturers and the NSSS suppliers and the owners of plants, I guess, that have these valves. Was that a correct summary of what you said?

A (WITNESS CORREA) Yes. And what I should have also added is that the NRC staff will also be notified.

Q That was my question.

I think it was Mr. Urquhart who talked about speaking -- in response to a question of Dr. Jordan's about valve failures in the fossil industry. You said you had spoken to two valve manufacturers, Dresser and Crosby. I would like to get a handle on how many valve manufacturers there are.?

A (WITNESS URQUHART) As far as manufacturers of safety valves, I would have to say Dresser and Crosby are probably the major manufacturers, and they are the only two that I have ever dealt with. I believe there is another one named Linogrin and a company called Target Rock is also in the business of making safety valves, however not of the type that we are discussing here today, not of the spring-loaded self-actuating type.

Q What about power-operated relief valves, pilot-operated relief valves?

What about power operated relief valves -- pilot operated relief valves?

A (Witness Urquhart) Today there is probably numerous manufacturers: Target Rock, Dresser, Crosby, Air Research Corporation. Those come to mind right now. They currently make pilot operated relief valves that I am familiar with.

There are other PORVs that utilize a different actuating mechanism such as some of the other NSSS suppliers use. They use an air actuator.

Those are manufactured by companies such as Control Components; Fisher -- I believe Fisher Valve Company makes a valve similar to that...

A (Witness Correa) If I could add something based on the EPRI test program and the population of valves they have found. As far as safety valves go, the two major manufacturers are Crosby and Dresser. Target Rock only has one safety valve, and there are no other safety valve manufacturers listed for the pressurized water plants.

As far the PORV goes, as Mr. Urquhart started to say, there are basically two types: the globe type, which are either air or solenoid actuated, and there are the electromatic or the pressure-matic which are made by Dresser and Crosby.

Q Would you say that you made an exhaustive survey of the history of operation of PORVs and safetyvalves in nuclear and fossil plants?

A (Witness Urquhart) As far as the usage of PORVs, my research was purely limited to the experience with B & W pressurized water reactors, as far as PORVs. Those are the basis of my statements before.

As far as safety valves, the extent of the ressearch was mainly limited to conversations with the manufacturers of the valves; namely, Dresser, Crosby, and Target rock.

Q I recall -- and I probably cannot put my hand on it right now, but I recall a figure appearing in the Lessons Learned document, 0578, to the effect that PORVs had failed to reseat five times out of 230 actuations.

Do you recall that?

A (Witness Urquhart) I do not recall those particular numbers. To the best of my knowledge, for example, on B & W PWRs, as I stated before, there have been three failures to reclose the power oprations, including the TMI-2 event.

Q When you talked to Dresser and to Crosby about failures in the fossil industry, that was for safety valves?

A (Witness Urquhart) Spring loaded safety valves.

Q They stated that they knew of no instance where properly installed and maintained valves failed to open.

That involves an exercise of judgment about whether the valve was properly installed and maintained, that the company who owned the plant might have a difference of opinion with the manufacturer.

I am wondering whether you made any effort to check that. Did you ask for how many failures overall there had been and made any attempt to verify whether in fact those were due to installation and maintenance problems?

A (Witness Urquhart) No. In essence, the discussions took the tone of, do you know of any instances where your valves have failed to perform their protective function.

The answer in all cases was no, provided the valves were properly maintained and set. They did say the problems they have seen and know of stem mainly from improper sizing; that is, the valves that were installed on the facility were not properly sized to protect the plant.

Another type of problem they have encountered is when somebody would gag a valve. When I say "gag," that is to prevent it from lifting, actually physically prevent the valve from lifting by installing -- essentially installing a screw at the top of the valve, and you can gag it and prevent it from lifting.

Q And then you are simply passing on this information which you has from Dresser and Crosby? The essence of my question is whether you made any attempt to go beyond their statement to you to do any independent check.

A (Witness Urquhart) I would have to say that my research was limited to discussions with those two manufacturers.

Q I was curious about one answer that one of you gave to Dr. Jordan.

I believe you were asked whether the safety valve testing program will include two phase relief. And I beliee the answer was that you did not know.

Is that correct?

A (Witness Correa) Two phase flow was one of the six items that is under discussion between the NRC and EPRI which I will answer questions on tomorrow.

Q I take it EPRI does not interpret the Action Plan or the Lessons Learned as requiring testing of the valves on two phase flow?

A (Witness Correa) I will have to answer that tomorrow.

Q You described a phone conversation that you had with the EPRI people relating the results of tests done this week on a Dresser PORV.

I want to confirm that you stated that that test was for steam only. Is that correct?

A (Witness Correa) Yes, it was for steam only.

MS. WEISS: Those are all the questions I have at this time, Mr. Chairman. I have gone through most of the cross examination plan.

I have not done the rebuttal testimony.

CHAIRMAN SMITH: Is there anything further we can accomplish this afternoon? I understand that you regard your examination on the direct testimony complete? Do you want to keep your options open?

MS. WEISS: I would. There are a few questions that I need to confer about. We might as well continue, though.

CHAIRMAN SMITH: Mr. Baxter?

REDIRECT EXAMINATION

BY MR. BAXTER:

Q Mr. Urquhart, Ms. Weiss was asking you about the extent of your research on valve failures and your conversations with the manufacturers, Dresser and Crosby.

Would you expect that you would be in your professional capacity and role at Babcock and Wilcox, that you would become aware of any failures of safety valves in the nuclear power industry?

A (Witness Urquhart) I would think most definitely. If there was a failure of a safety valve to perform its function, that I would be aware of it.

MR. BAXTER: Thank you. That is all I have.

MS. WEISS: That raises one more.

RECROSS EXAMINATION

BY MS. WEISS:

Q I want to make it clear that I was questioning you about fossil experience as well as nuclear. You did understand that?

A (Witness Urquhart) Yes.

CHAIRMAN SMITH: Anything further with these witnesses?

MR. DORNSIFE: Yes, I have one additional question based on Ms. Weiss's.

BY MR. DORNSIFE:

Q Mr. Jones, do you recall Ms. Weiss's questions concerning the indications that have been added to verify whether the PORV is in fact open or closed and their qualifications?

A (Witness Jones) Yes.

Q Do you have a copy of NUREG-0737?

(Pause)

Would you please look at page 3-75.

Do you have that?

A (Witness Jones) Yes, I do.

Q Would you specifically look at clarification item number three and tell me whether, as far as its qualification, whether the TMI design would meet that clarification.

(Pause)

A (Witness Jones) I believe they do, yes.

Q Take a quick look at all those clarification items and see if there are any that the design will not meet.

(Pause)

A (Witness Jones) To the best of my knowledge, they meet them all; though, I would not want to -- do not know enough about human factors types analysis of control rooms to be absolutely sure. But certainly the first five they meet.

MR. DORNSIFE: Thank you. That is all I have.

CHAIRMAN SMITH: Gentlemen, at least for this afternoon, you are excused.

(The witnesses were excused)

CHAIRMAN SMITH: Is there any recommendation for the remaining one hour and 15 minutes of the afternoon? Can we start with one of your witnesses?

MR. CUTCHIN: I was going to suggest, Mr. Chairman, that in the interest of moving forward, I am prepared to put both of my witnesses on as a panel, and then we can go as far as we can with them.

CHAIRMAN SMITH: Okay, gentlemen, if you would come forward.

MS. WEISS: I wonder if we could use the same procedure on these witnesses; that is, have the other parties do the questioning first.

CHAIRMAN SMITH: Sure.

MR. CUTCHIN: I understood that would be the plan. We would have the same gap in their coverage as we have in these witnesses.

CHAIRMAN SMITH: I think that is reasonable.

Thereupon,

WALTON L. JENSEN, JR.,

was recalled as a witness, on behalf of the NRC staff, and having been previously duly sworn, was examined and testified as follows:

and

JOHN J. ZUDANS

was called as a witness, on behalf of the NRC staff, and having been duly sworn, was examined and testified as follows:

MR. CUTCHIN: Mr. Jensen obviously has previously been sworn. Mr. Zudans has not. They are a panel.

DIRECT EXAMINATION

BY MR. CUTCHIN:

Q First, Mr. Jensen, do you have with you a copy of a document consisting of five pages to which is attached a copy of your professional qualifications consisting of two pages. The document bears the caption of this proceeding, and it is entitled "NRC Staff Testimony of Walton L. Jensen, Jr. Relative to Primary System Relief and Block Valve (UCS Contention Five).

A (Witness Jensen) Yes, I do.

Q Was that document prepared by you?

A (Witness Jensen) Yes, it was.

Q Do you have any corrections or modifications you wish to make?

A (Witness Jensen) No.

Q Do you adopt it as your testimony in this proceedings?

A (Witness Jensen) Yes.

Q Is it true and correct to the best of your knowledge and belief?

A (Witness Jensen) Yes, it is.

MR. CUTCHIN: Mr. Chairman, I would ask that Mr. Jensen's testimony previously identified be received into evidence and be bound into the transcript as if read along with the outline accompanying it.

CHAIRMAN SMITH: If there are no objections, the testimony is received.

(The testimony of Walton L. Jensen, Jr., follows.)

BY MR. CUTCHIN:

Q Mr. Zudans, do you have before you a document consisting of seven pages plus two pages of your

Lay-in #4

OUTLINE

The testimony of Walton L. Jensen, Jr., contains the NRC Staff's response to UCS Contention 5.

The purpose of this testimony is to demonstrate that contrary to the assertions made in the contention, the power operated relief valve and block valve are not components important to safety and need not satisfy all safety grade design criteria.

Conclusions to be drawn from this testimony:

The function of the PORV is to prevent unnecessary opening of pressurizer safety valves and to provide a backup means of depressurization and of overpressure protection.

The function of the block valve is to permit isolation of a leaking or failed-open PORV.

Proper operation of the PORV and block valve is not required to mitigate the consequences of any design basis accident.

Failure of the PORV and block value to function can cause the equivalent of a small-break LOCA but if the failure occurred in conjunction with a LOCA the consequences would not be significantly altered.

An unisolated stuck-open PORV will not result in core damage.

The PORV and block valve are not components important to safety.

The PORV and block valve are being upgraded to reduce the number of challenges to the safety valves and ECCS during operation.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of METROPOLITAN EDISON COMPANY (Three Mile Island Nuclear Station, Unit No. 1)	)))))	Docket No. 50-289 (Restart)

NRC STAFF TESTIMONY OF WALTON L. JENSEN, JR., RELATIVE TO

PRIMARY SYSTEM RELIEF AND BLOCK VALVES

(UCS CONTENTION 5)

Q1) Please state your name and position with the NRC.

A) My name is Walton L. Jensen, Jr. I am an employee of the U. S. Nuclear Regulatory Commission assigned to the Reactor Systems Branch, Division of Systems Integration, Office of Nuclear Reactor Regulation. From June through December 1979, I was assigned to the Analysis Group of the Bulletins and Orders Task Force, Office of Nuclear Reactor Regulation.

Q2) Have you prepared a statement of professional qualifications?

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

Q3) Please state the nature of the responsibilities that you have had with respect to the Three Mile Island Nuclear Station - Unit 1.

A) The accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979, involved a feedwater transient coupled with the equivalent of a small break in the reactor coolant system, though the accident's ultimate severity resulted from a number of interacting elements including lack of complete understanding of system response, misleading instrument readings and inadequate operator training and procedures. Because of the resulting severity of ensuing events and the potential generic applicability of the accident to other reactors, the NRC staff initiated prompt action to:

(1) assure that other reactor licenses, particularly those plants such as TMI-1 which have a similar design to TMI-2, took the necessary actions to substantially reduce the likelihood of future TMI-2-type events from occurring, and

(2) initiate comprehensive investigations into the potential generic implications of this accident on other operating plants.

To accomplish some of this work, the Bulletins and Orders Task Force (B&OTF) was established within the Office of Nuclear Reactor Regulation (NRR) in early May 1979. The B&OTF was responsible for reviewing and directing the TMI-2-related staff activities associated with loss of feedwater transient and small break loss-of-coolant accidents (LOCAs) for all operating plants to assure their continued safe operation.

I was assigned to the Task Force in June 1979. I participated in the preparation of NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants."

Following my assignment to the Reactor Systems Branch, I participated in the evaluation of potential feedwater transients at operating B&W plants and participated in the final preparation of the staff Safety Evaluation on the Three Mile Island 1 restart.

Q4) Please state the purpose of this testimony.

A) The purpose of this testimony is to respond to UCS Contention 5 which reads: "Proper operation of power operated relief valves, associated block valves and the instruments and controls for these valves is essential to mitigate the consequences of accidents. In addition, their failure can cause or aggravate a LOCA. Therefore, these valves must be classified as components important to safety and required to meet all safety-grade design criteria."

Q5) What are the functions of the PORV and Block Valve?

A) The PORV is provided to prevent the pressurizer safety valves from being opened for mild transients. It is more desirable to open the PORV than the safety valves since the PORV is provided with an upstream block valve to isolate the PORV in the event that the PORV fails to reseat, whereas the safety valves do not have an isolating block valve. The PORV also gives the operator a means of depressurizing the primary system that is independent of the steam generators and provides a backup to operator action in preventing reactor system overpressure during low temperature operation. The function of the block valve is to permit the operator to manually isolate a leaking or failed-open PORV.

Q6) Is proper operation of the PORV or block valve essential to mitigate the consequences of accidents?

A) No, proper operation of the PORV, associated block valve, and instruments and controls is not required to mitigate the consequences of any design basis accidents.

Moreover, a stuck open PORV which is not isolated will not result in damage to the fuel element cladding. Therefore, the fission products contained in the fuel elements would not escape from the core. The only releases to the public would be from radioactive materials already contained in the primary coolant. This material would include activated corrosion products contained in the primary coolant and fission products which might have leaked into the coolant during operation.

Q7) What offsite doses would result from a stuck open PORV that was not isolated by the operator?

A) The releases to the public would be less than those calculated for the Large Break LOCA analyzed in the TMI-1 FSAR (Chapter 14) since for the large break LOCA all the fuel element cladding was assumed to have failed with a complete release of the fission product gas. The releases to the public for the Large Break LOCA were calculated to be a thyroid dose of 0.26 rem and a whole body dose of 0.0085 rem at the edge of the exclusion area; and a thyroid dose of 0.07 rem and a whole body dose of 0.0075 rem at the low population zone boundary. Those doses are less than the 10 CFR 100 guidelines by a factor of more than 1000. The releases to the public from a postulated stuck open and unisolated PORV would be less than for the Large Break LOCA.

Q8) Can failures of these valves, instruments and controls cause or aggravate a LOCA?

A) A failure of the PORV or associated instruments and controls which results in inability to isolate the flow path through the valve causes the equivalent of a small-break loss-of-coolant accident. The accident would be terminated by closure of the block valve which is an immediate action to be taken by the operator in the event of a small-break LOCA. Even if the block valve were not isolated the capability of the High Pressure Injection System is sufficient to permit safe shutdown of the reactor with no core uncovery or core damage. In the event that the PORV was to open inadvertantly following a small-break in the primary system piping, the effect on the reactor system would be equivalent to increasing the break size. The effect of an increase in break size would fall within the spectrum of small-break sizes already analyzed for TMI-1. The spectrum of small-break sizes analyzed for TMI-1 is discussed in the NRC's testimony in response to UCS Contention 8. The calculated consequences for all small-breaks are significantly below the limits of 10 CFR 50.46 so that no cladding failures would occur. Thus, the failure of the PORV, block valve or instruments and controls would not significantly aggravate a small-break LOCA.

Q9) If the PORV and block valve are not essential to mitigate the consequences of accidents, why does the staff require these components to be upgraded?

A) These modifications will reduce the number of challenges to the emergency core cooling system and the safety valves during operation. The repeated unnecessary challenges to these systems is undesirable.

As discussed in our Safety Evaluation for TMI-1 restart, NUREG-0680, the NRC has required and Metropolitan Edison has committed to make changes in PORV setpoint, power supply requirements and valve position indication before restart. See pages C2-11, C2-12, C8-10 and C8-11 to C8-14.

The availability of emergency power to the PORV will reduce the number of challenges to the safety valves.

The availability of emergency power to the block valve, changes in setpoint and valve position indication will provide reasonable assurance that a stuck open PORV will be an unlikely event which, if it occurs, will be detected by the operator so that the block valve will be closed. These modifications will reduce the number of challenges to the Emergency Core Coolant System.

WALTON L. JENSEN, JR.

PROFESSIONAL QUALIFICATIONS

I am a Senior Nuclear Engineer in the Reactor Systems Branch of the Nuclear Regulatory Commission. In this position I am responsible for the technical analysis and evaluation of the public health and safety aspects of reactor systems.

From June 1979 to December 1979, I was assigned to the Bulletins and Orders Task Force of the Nuclear Regulatory Commission. I participated in the preparation of NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants."

From 1972 to 1976, I was assigned to the Containment Systems Branch of the NRC/AEC, and from 1976 to 1979, I was assigned to the Analysis Branch of the NRC. In these positions I was responsible for the development and evaluation of computer programs and techniques to calculate the reactor system and containment system response to postulated loss-of-coolant accidents.

From 1967 to 1972, I was employed by the Babcock and Wilcox Company at Lynchburg, Virginia. There I was lead engineer for the development of loss-of-coolant computer programs and the qualification of these programs by comparison with experimental data.

From 1963 to 1967, I was employed by the Atomic Energy Commission in the Division of Reactor Licensing. I assisted in the safety reviews of large power reactors, and I led the reviews of several small research reactors.

I received an M.S. degree in Nuclear Engineering at the Catholic University of America in 1968 and a B.S. degree in Nuclear Engineering at Mississippi State University in 1963.

I am a graduate of the Oak Ridge School for Reactor Technology, 1963-1964.

I am a member of the American Nuclear Society.

I am the author of three scientific papers dealing with the response of B&W reactors to Loss-of-Coolant Accidents and have authored one scientific paper dealing with containment analysis.

professional qualifications, the title of the document being "NRC Staff Testimony of John J. Zudans Relative to Reactor Coolant Pressure Boundary Compliance with GDC 1, 14, 15, and 30 (UCS Contention Six)"?

A (Witness Zudans) Yes.

Q Do you also have before you a copy of a one page document entitled "NRC Staff Testimony of John J. Zudans Relative to Board Question Regarding UCS Contention Six"?

A (Witness Zudans) No, I do not.

Q Do you recollect having prepared such a document?

A (Witness Zudans) Yes.

Q Do you now have a copy?

A (Witness Zudans) Now I do.

Q Do you have any corrections or modifications you wish to make to this testimony?

A (Witness Zudans) Yes, I do. On the response to UCS contention six I have a couple of typographical errors that need to be corrected.

The first one is on page 5. In parentheses there is for letters: PROV should be changed to PORV on the third line.

MR. CUTCHIN: These appear in the reporter's copy, Mr. Chairman.

CHAIRMAN SMITH: All right. Page 5. What is the correction, now?

MR. CUTCHIN: I believe Mr. Zudans says the correction is in the third line; in parentheses it should be PORV rather than PROV.

BY MR. CUTCHIN:

Q The other corrections, Mr. Zudans?

A (Witness Zudans) On page 6, subparagraph c near the middle of the page, the last line should read C8-8 and C8-9, and subparagraph d on the second line there, it should read C1-15. That would complete it, then.

Q That involves the insertion of the letter C before the page numbers as they appear there; is that correct?

A (Witness Zudans) Correct.

Q As modified, do you adopt these documents as your testimony in this proceeding?

A (Witness Zudans) I do.

Q Are they true and correct to the best of your knowledge and belief?

A (Witness Zudans) They are.

MR. CUTCHIN: Mr. Chairman, I would ask that the documents identified along with the copy of the outline which accompanies the testimony in response to the contention be received into evidence. I am sorry.

I ask that these documents be received into evidence and bound into the transcript along with the outline which accompanies the documents entitled "Testimony in Response to UCS Contention Six."

CHAIRMAN SMITH: So received.

(The testimony of John J. Zudans follows.)

Page in
#5

8824

OUTLINE

This testimony of John J. Zudans contains the NRC Staff's response to UCS Contention 6.

The purpose of this testimony is to demonstrate that, contrary to the assertions made in the contention, additional qualification testing of reactor coolant system relief and safety valves is not required to provide reasonable assurance of no undue risk to public health and safety.

Conclusions to be drawn from this testimony:

-- Except for verification testing of their ability to withstand loadings resulting from two-phase and solid-fluid flow, reactor coolant pressure boundary safety and relief valves meet the Staff's current interpretation of the requirements of GDC 1, 14, 15 and 30.

-- Such verification testing is presently scheduled to be completed by July, 1981.

-- Analyses of the conseqeunces of a stuck open PORV predict that no fuel damage will occur.

-- Improvements in design and emergency procedures to be completed prior to restart will decrease the likelihood of PORV failure.

-- The recent transient at Crystal River provided evidence that the safety valves will perform properly under two-phase flow and solid-fluid flow conditions.

-- Operation of TMI-1 prior to completion of the verification testing will not endanger public health and safety.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of METROPOLITAN EDISON COMPANY, ET AL (Three Mile Island Nuclear Station Unit 1)	))))))	Docket No. 50-289

NRC STAFF TESTIMONY OF JOHN J. ZUDANS
RELATIVE TO REACTOR COOLANT
PRESSURE BOUNDARY COMPLIANCE WITH
GDC 1, 14, 15 & 30

(UCS CONTENTION 6)

Q.1. Please state your name and position with the NRC.

A. My name is John J. Zudans. I am an employee of the Nuclear Regulatory Commission assigned to the Equipment Qualification Branch, Division of Engineering, Office of Nuclear Reactor Regulation. I am a senior Mechanical Engineer assigned to the Seismic and Dynamic Load Qualification Section.

Q.2. Have you prepared a statement of professional qualifications?

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

Q.3. Please state the nature of the responsibilities you have had with respect to the Three Mile Island Nuclear Station, Unit 1.

A. Soon after the accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979 I was asked to evaluate the Residual Heat Removal Pumps at TMI-2 which are similar to those at TMI-1 for possible use for long term decay heat removal.

Q.4. Please state the purpose of this testimony.

A. The purpose of this testimony is to address UCS Contention #6.

USC Contention 6 reads as follows:

> "Reactor coolant system relief and safety valves form part of the reactor coolant system pressure boundary. Appropriate qualification testing has not been done to verify the capability of these valves to function during normal, transient, and accident conditions. In the absence of such testing, verification compliance with GDC 1, 14, 15 and 30 cannot be found and the public health and safety is endangered."

Q.5. What are the requirements of General Design Criteria (GDC) 1, 14, 15, and 30?

A. General Design Criteria 1 (GDC 1) as stated in the Code of Federal Regulations Part 50 Appendix A requires that "structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit."

GDC 14 requires that "the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture."

GDC 15 requires that "the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

GDC 30 requires that "components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical and that means shall be provided for detecting and to the extent practical, identifying the location of the source of reactor coolant leakage."

Q.6. What are the requirements which the Reactor Coolant Pressure Boundary including safety and relief valves (SRV) must meet to comply with the requirements of GDC 1, 14, 15, and 30?

A. The current staff position with respect to the requirements which must be met to comply with GDC 1, 14, 15, and 30 require that applicants assess their RCPB including safety and relief valves to the following standards:

1) Standard Review Plan (SRP) 3.9.2, "Dynamic Testing and Analyses of Systems, Components, and Equipment,"

2) SRP 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures,"
3) Regulatory Guide 1.67, "Installation of Overpressure Protection Devices,"
4) Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water Cooled Power Reactors,"
5) Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems,"
6) The appropriate sections of Appendix B to 10 CFR 50.

Q.7. In what respect is the Staff's position with respect to the requirements of GDC 1, 14, 15, and 30 regarding RCPB, including safety and relief valves, not met?

A. The staff's position requires that the safety and relief valves function as expected during design transient and accident conditions. The extent to which the current staff interpretation of the requirements of GDC 1, 14, 15, and 30, relative to the reactor coolant system safety and relief valves, are not yet verified is that the tests performed to date did not cover loadings which result from two-phase flow or solid fluid flow.

The reactor coolant system safety valves were originally designed and tested for operation on saturated steam in accordance with the applicable edition and addenda of Section III of the ASME Boiler and Pressure Vessel Code. Additionally, the safety valves have been designed to be functional after exposure to loads resulting from the maximum hypothetical earthquake for the TMI-1 site. As required by Article 9 of the Code, the safety valve relieving capacity has been provided so that the pressure limitation specified in the Code will be maintained under all of the system transients or accidents postulated to occur. The power operated relief valve (~~PROV~~ PORV) is a pilot operated valve and does not replace a code required safety valve nor does it contribute to the Code required relieving capacity for the reactor coolant system. However, the PORV was designed to the same ASME Code requirements as the safety valves as it relates to pressure boundary integrity.

Q.8. What is being done to demonstrate that the safety and relief valves at TMI-1 can withstand the loadings resulting from these flow conditions?

A. A test program has been initiated by the Electric Power Research Institute (EPRI) which will address safety and relief valve operability. Metropolitan Edison Company (MET-ED) in the TMI-1 Restart Report has committed to participating in this test program and has as one of its objectives to satisfy the long-term requirement on SRV testing as set forth in Section 2.1.2 of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations." In the staff's SER on TMI-1 restart (NUREG-0680), the staff requires that MET-ED justify the EPRI test program is applicable to the TMI-1 SRVs. Should this program demonstrate that these valves are not qualified for the above stated loadings the staff will require the licensee to take corrective actions.

Q.9. When will the test program be completed?

A. Present schedules indicate that this testing will be completed by July 1, 1981.

Q.10. Would the health and safety of the public be endangered should TMI-1 be allowed to restart prior to completion of the EPRI test program?

A. An analysis of a stuck open PORV has been performed (see NRC testimony by W. Jensen in response to UCS Contention #8) and the results showed that no fuel damage is predicted to occur. In addition, the following measures have or will be implemented by the licensee prior to restart to lessen the severity of a stuck open PORV:

(a) if the PORV should fail open, sensors which will be installed prior to restart at the PORV discharge will allow the operator to determine if the PORV is open or shut (see TMI-1 Restart SER NUREG-0680 pages C8-11 to C8-13);

(b) TMI-1 Small Break LOCA Procedures require the PORV block valve to be closed early in a LOCA;

(c) the PORV and PORV block valve are all powered from emergency busses as part of the originally approved TMI-1 design and therefore meet short term lessons learned Item 2.1.1 (see TMI-1 Restart SER NUREG-0680, pages C8-8 to C8-9);

(d) small break LOCA emergency procedures have been upgraded at TMI-1 and have been approved by the NRC (as discussed on page C1-15 of NUREG-0680).

Furthermore, as stated on Page 2-1 of NUREG-0565, "with the increase in PORV lift setpoint, the reduction in the setpoint of the high pressure reactor trip and the addition of the anticipatory reactor trips, lifting of the PORV is not likely to occur for the loss of feedwater and turbine trip transients." Thus these valves will be challenged considerably less.

This has been verified by operating experience since there have been 20 transients (as of 6/80) which would have, with the old setpoints, opened the PORV and did not with the new setpoints. The lessening of challenges to the PORVs provides reasonable assurance that PORV failures will be greatly lessened.

With regard to the safety valves, there is presently no evidence that these valves will not operate properly during the anticipated transients which produce two phase flow and solid fluid flow. In fact the transient which occurred at the Crystal River Nuclear Unit on February 26, 1980 provides evidence that the safety valves would perform their intended functions under these load conditions. The Crystal River facility has a B&W nuclear steam system and components similar to those at TMI-1.

Based on the above considerations, operation of TMI-1 prior to completion of the EPRI test program would not endanger the health and safety of the public.

PROFESSIONAL QUALIFICATIONS

OF

JOHN J. ZUDANS

My name is John J. Zudans. I am currently employed by the U.S. Nuclear Regulatory Commission as a Senior Mechanical Engineer, Equipment Qualification Branch, Division of Engineering, Office of Nuclear Reactor Regulation, (NRR). Prior to the NRR reorganization I was a member of the Engineering Branch, Division of Operating Reactors, NRR.

My duties and responsibilities include the review and evaluation of structural mechanical aspects as related to safety issues involving equipment qualification in nuclear reactor facilities being licensed or operating. I am specifically involved with mechanical and environmental qualification of pumps and valves. In this capacity I am responsible for evaluating purge and vent valve operability for all operating reactors, deep draft pump operability, and I am also involved in reviews of various active safety related components such as relief valves, block valves and their associated equipment. I am a graduate of Villanova University with a Bachelor of Science Degree (1970) in mechanical engineering. I am also attending the University of Maryland towards a masters degree in mechanical engineering.

Prior to my appointment with the NRC, I was employed by Stone & Webster Engineering Corp., Cherry Hill, N.J. (1974-1976) and Ingersoll-Rand Co., Phillipsburg, N.J. (1972-1974).

My duties as a Principal Engineer at Stone & Webster included the design and analyses of containment structures and attachements thereto. While employed at Ingersoll-Rand Co., my duties included the design, analyses and testing of pumps used in the U.S. Navy nuclear submarine program. Operability and reliability of these components was a key requirement in performance of my duties.

Professional Societies

American Society of Mechanical Engineer

Member of the ASME Committee on Operation and Maintenance of Nuclear Power Plants-WG on Inservice Testing of Pumps and Valves.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of METROPOLITAN EDISON COMPANY, ET AL (Three Mile Island Nuclear Station Unit 1)	))))))	Docket No. 50-289

NRC STAFF TESTIMONY OF JOHN J. ZUDANS
RELATIVE TO BOARD QUESTION REGARDING UCS CONTENTION #6

QUESTION

"The board wants more than just a schedule for testing of reactor coolant system safety and relief valves, as is required pursuant to NUREG-0578. Is there reasonable assurance that the tests will be successful, e.g., that there is good evidence that these valves will indeed perform in an accident environment?"

RESPONSE

The answer to this board question is contained in my response to UCS Contention #6.

MR. CUTCHIN: The witnesses are available for cross examination.

CHAIRMAN SMITH: So received.

(The document previously marked USC Exhbit 5 for identification was received in evidence.)

MR. CUTCHIN: The witnesses are available for cross exmination.

DR. JORDAN: I don't know whether mine is a clarifying question or a substantive one, for Mr. Jensen. I thought it might be well to get it out of the way early.

BOARD EXAMINATION

BY DR. JORDAN:

Q On page 4, in response to Question 7, you say: "The releases to the public," the second sentence, "The releases to the public for the large-break LOCA were calculated to be a thyroid dose of 0.26 rem and a whole-body dose of 0.0085 rem at the edge of the exclusion area, and a thyroid dose of 0.07 rem and a whole-body dose of 0.0075 rem at the low-population zone boundary."

Those do not come even close to the numbers that I find in looking at the SER at the operating license stage. Do you have a copy of the SER with you? Can you show me how those numbers --

A (WITNESS JENSEN) I have one back on the table.

(Pause.)

The confusion is between the design basis accident and the large-break LOCA. The design basis accident was for the plant for the purpose of calculating the doses, the maximum doses to the public. That was done assuming there was a large amount of release of radioactive material from the core.

Q So therefore, you did not use the design basis figures?

A (WITNESS JENSEN) No.

Q What figures are these?

A (WITNESS JENSEN) Let me find that. This comes from section 14.2.2, .3, .7. It is titled "Environmental Analysis of a Loss-of-Coolant Accidents."

Q This is the environmental analysis. All right.

A (WITNESS JENSEN) These doses are based on the expected releases from a large-break loss-of-coolant accident rather than the design basis accident where a much larger amount of release of radioactive material was assumed.

Q What are the circumstances, then, that you assumed for your figures or that was assumed in the SER in order to get the environmental doses? How are they different? Was it the wind velocities, the value of the mixing chi over q, or what?

A (WITNESS JENSEN) Primarily, the core damage that would have to occur, I believe.

Q I see. In other words, the values for the design basis LOCA assume the 100-percent release of gaseous fission products? The releases you took were those calculated according to 50.46, then?

A (WITNESS JENSEN) This assumes that the activity associated with the gas in all the fuel rod was assumed to be released --

Q All of the gas?

A (WITNESS JENSEN) The noble gas activity -- some of the noble gases were released, I guess. Basically, what I was trying to show here was that for this large-break LOCA event, which is using the values of the activity released from the core, assuming there was some core damage -- which they did here, but still less than design basis accident -- that these releases to the public were still much less than the guidelines of 10 CFR Part 100. There was some core damage, but not to the extent that was assumed for the design basis accident.

Q I understand now why -- where you got your figures from. I guess I don't understand the rationale for not assuming the design basis LOCA.

A (WITNESS JENSEN) I was trying to say, "Here is a loss-of-coolant accident with some core damage. These values of dose were calculated. However, for this stuck-open PORV case, there would be no core damage. So the doses expected to the public would be less."

This is certainly -- neither one of these are the design basis event for siting the plant.

Q I understand what you're saying. I was a little surprised, because this was the accident at TMI-2, and in that case the releases were very much larger than the ones you are assuming so far as the source term is concerned.

A (WITNESS JENSEN) I don't know what they were at TMI-2, but, of course, the core was damaged, because the high-pressure injection water was shut off prematurely.

Well, here I have assumed that the high-pressure injection system would operate and cool the core and keep it covered so there would be no core damage.

Q Thank you.

(Board conferring.)

CHAIRMAN SMITH: Ms. Weiss, I don't have cross-examination plan for Mr. Jensen.

MS. WEISS: I am not sure if I have one either. I am riffling through my papers to see if I have anything. I wasn't expecting to get this far.

CHAIRMAN SMITH: I haven't lost it. My immediate concern is have I lost something.

MS. WEISS: No.

CHAIRMAN SMITH: Mr. Dornsife?

MR. DORNSIFE: If I could ask you, with your last questions, the numbers that Mr. Jensen used, I just happen to have Table 14-46 of the FSAR.

DR. JORDAN: Yes. I was looking at Table 15. I understand.

MR. DORNSIFE: It is based on 1 percent failed fuel, reactor coolant activity, with just the gas activity being released, which they call the "realistic" analysis.

CROSS EXAMINATION

BY MR. DORNSIFE:

Q Are you generally familiar with the upgrading that has been performed on both the PORV and the block valve?

A (WITNESS JENSEN) Somewhat. To the extent that it is written in the NRC safety evaluation report.

Q In your judgment, how much has the reliability been improved with the upgrading compared to if both components would have been made safety grade? Can you give me a qualitative discussion about that, what your opinion is? In other words, where it would not meet the safety-grade qualifications and how far away is that, qualitatively, from being fully safety grade?

A (WITNESS JENSEN) Well, I understand that the primary difference between the PORV and the block valve and being fully safety grade is in the single-failure requirements that would be on the safety-grade system. I suppose more valves would have to be added.

Q In that vein, how close does having the availability or having the ability for the operator to switch to either diesel in the control room improve the reliability compared to if there were two valves?

I realize you don't have a numerical quantitative -- I am looking for a qualitative answer. Is that a lot of reliability compared to if it were fully safety grade, a little? I am looking more at failure rates of components now. That is primarily -- the electrical power supply seems fairly reliable. It is the component failure now that may be a problem. That's the type of thing I am looking for.

MS. WEISS: I would like it to be established, before the witness answers, that he has any information about the failure rates of the components. It is certainly not within the scope of his testimony. He may but --

MR. DORNSIFE: I wasn't looking for a quantitative answer --

MS. WEISS: I am not sure -- his qualitative opinion might be useful if it were based on some knowledge, but I don't believe it would be useful if it were sheer speculation.

CHAIRMAN SMITH: Would you address Ms. Weiss' concerns in your answer, too?

WITNESS JENSEN: I have looked at the failure rates of PORVs in the past history of B&W plant operation. I believe the numbers were 9 failures out of some 300 challenges. A number of these failures were done -- occurred during plant startup and testing and did not occur at power.

BY MR. DORNSIFE:

Q I think that specific was directed toward the power supplies of the block valves, not the PORV. Are block valves of that nature typically pretty reliable components? Do they have a high failure rate? Would that be a factor, an overwhelming factor?

A (WITNESS JENSEN) As far as adding the emergency power to the block valve, it would make the system more reliable in terms of the ability to close and isolate. If that is what you are talking about -- the ability to close and isolate -- adding the emergency power to the PORV I don't believe would increase its ability to close, because it closes under loss of power.

As far as improving the ability of the valves to open on demand, I suppose adding emergency power to the PORV would assist there.

Q The power supply for the PORV, was it always from the battery, do you know?

A (WITNESS JENSEN) No, I don't.

Q You don't know if it is was really upgraded?

A (WITNESS JENSEN) No.

MR. DORNSIFE: I have no further questions.

(Pause.)

DR. JORDAN: I may have a few questions, but I will have to look up something first.

Mr. Baxter, do you have any questions?

MR. BAXTER: No, I don't.

(Pause.)

FURTHER BOARD EXAMINATION

BY DR. JORDAN:

Q I want to refer you to NUREG-0737, under section II-K3.2. On page III-140. Are you gentlemen familiar with this requirement in the review plan? Do you know whether the licensee is submitting the reports required on the schedule shown in there?

A (WITNESS JENSEN) I don't know whether it will be submitted on schedule or not. I don't know what the schedule is.

Q It says, for example, under "Implementation," second paragraph: "All applicants for operating license should submit documentation four months prior to the expected issuance of the staff's safety evaluation report or four months prior to the listed implementation date, whichever is later." The implementation date is given as January 1, 1981. So, presumably, the licensee has submitted these documents.

Have you seen and reviewed those documents, or do you know anyone who has?

(Pause.)

A (WITNESS JENSEN) I don't remember seeing this particular report. I don't know whether it has been submitted or not.

A (WITNESS ZUDANS) I have not see such a report either.

Q Then I won't ask you any questions concerning the staff's evaluation of that report or how you are proceeding, if you haven't seen it.

(Pause.)

Q Mr. Zudans, on page 3 of your testimony, first full paragraph, it says, quote: '"GDC-14 requires that the reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." Quote.

Do you know that the PORV's do have a low probability, and what standards or criteria do you use for making the judgment of low probability

A (WITNESS ZUDANS) I know that the pressure integrity for the safety valve and the PORV have met the standards that were in effect at the time that the plant was licensed. And that was the ASME Section 3 requirements for pressure integrity.

Q Your quotation says that they shall be tested so as to have an extremely low probability of abnormal leakage. How do you -- and you do have some figures I think you have quoted on the frequency of failures of PORV's.

My question is, does that frequency of failures meet the requirements of GDC-14 for low probability, or are there any requirements at all for probability?

It says, "to have an extremely low probability of abnormal leakage." Hasn't the staff, on a Standard Review Plan perhaps, made an interpretation of what is meant by GDC-14 when applied to PORV's, and haven't they given some guidance on what is meant by the low probability?

A (WITNESS ZUDANS) No, I do not have any guidance on what is meant to be low probability.

Q Are you familiar with the Standard Review Plan on PORV's?

A (WITNESS ZUDANS) Standard Review Plan, I am familiar with Standard Review Plan 393, which discusses operability assurance programs for pumps and valves. In that, we are required to evaluate the pressure integrity of the valve and also the operability of the valve.

Q These are valves that form part of the coolant pressure boundary ?

A (WITNESS ZUDANS) Yes.

Q And there is no guidance on what is meant by "low probability of abnormal leakage" in the Standard Review

Plan?

A (WITNESS ZUDANS) In the passive mode, where the valve is not required to operate, you must meet ASME Code Section 3 rules.

Q Does that give a probability?

A (WITNESS ZUDANS) No. The ASME Code does not give any probability.

Q Is there a NUREG -- I mean --

MS. WEISS: Reg Guide?

DR. JORDAN: Reg Guide. Thank you.

BY DR. JORDAN: (Resuming)

Q Is there a Reg Guide that assists in the interpretation of this GDC-14 with respect to valves? Does it address the problem of probability?

A (WITNESS ZUDANS) Not that I know of.

Q Do you believe that the PORV's have a low probability of abnormal leakage?

A (WITNESS ZUDANS) Yes.

Q On page 4 you mentioned, in your answer to Question 7, that the tests performed to date did not cover loadings which resulted from two-phased flow or solid fluid flow. Is this the section that the staff has reservations yet -- is this the area in which the staff has reservations concerning the adequacy of the testing program?

A (WITNESS ZUDANS) Yes.

Q Does the staff believe that the testing program is going adequately in this respect? And if so, what is the basis for believing that the program will lead to confidence that the valves will be able to handle two-phase flow or solid flow?

A (WITNESS ZUDANS) The comments which were prepared for revision 1, July '80 EPRI program, the comments which were prepared by the staff for that program include one paragraph which requires the two-phase flow and solid fluid flow to be part of the test program. I believe the staff will require that kind of testing before any judgment can be made on the test.

Q Is the staff sufficiently confident in the outcome that they believe that restart should be permitted before the tests are finished?

A (WITNESS ZUDANS) Yes, we are.

Q Why?

A (WITNESS ZUDANS) Because of the measures that have been taken by the Licensee to lessen the effect of a stuck-open PORV. These measures are listed in my answer to Question No. 10 on page 6: the installation of sensors a, b, c and d.

Q I see, okay. You started to mention them briefly. Go ahead.

A (WITNESS ZUDANS) The installation of the sensors, which will be installed prior to the restart, that will allow the operator to know the position of the valves. The small break LOCA procedures, which require the PORV to be closed early in the transient; the fact that the PORV's and block valves are powered from emergency power buses; and small break LOCA procedures have been upgraded and approved by the staff; and the fact that the valve, in the succeeding paragraph, that the valves will be challenged considerably less. We believe that is a very favorable aspect of the program.

Q Do you think it might be a good idea for the nuclear plants to operate with the block valve closed, to reduce the challenges from the PORV?

A (WITNESS ZUDANS) I would preface my answer by saying that I am not qualified to answ[illegible] system questions. But I never have believed that if you have a valve that is there for a purpose, such as this one, of pressure regulation, that you should block it off if it is shown to be operable.

Q We did hear the reason the valve was put there was to ride through certain loss of load transients. That seems now to have disappeared. Is it worthwhile, in view of the possibility of the failure and the subsequent LOCA -- on balance, should the block valves be left closed, the remaining reason being it is kind of a backup for the safety valve?

Have you considered these alternatives?

A (WITNESS ZUDANS) I have not really considered this in great detail.

Q On page 7, the first paragraph, you say: "There has been verified by operating experience -- "This has been verified by operating experience, since there have been 20 transients as of 6/80, which would have, with the old set points, opened the PORV and did not with the new set points." Quote.

Did the reactor protection trip in each of those cases?

A (WITNESS ZUDANS) I don't know.

Q I had a question with respect to the next paragraph, but it has already been answered.

If you remember, there were some questions of the Licensee's panel with respect to some of the requirements of the NUREG-0758, Section 2.1.2, particularly with respect to the circuitry that goes along with the PORV, the fact that it is not single failure-proof.

And in view of the fact that these circuits can indeed result in a failure of the PORV, I then also question you as to whether they should be upgraded to safety grade since it does affect the pressure boundary?

A (WITNESS ZUDANS) I believe that some of the circuitry will be upgraded as a result of the I&E Bulletin 79-01B in terms of the environmental qualifications of the circuitry. As far as whether it should be safety grade, I cannot offer an opinion.

Q There will be an upgrading of the circuitry involved?

A (WITNESS ZUDANS) Yes, I believe there will be.

Q I think that's all the questions I had for Mr. Zudans. I will see if I had any more for Mr. Jensen.

MR. BAXTER: Could I ask one follow-up of Mr. Zudans while you are looking at your notes.

DR. JORDAN: Yes, please do. It would be a good time.

CROSS-EXAMINATION ON BOARD EXAMINATION

BY MR. BAXTER:

Q Dr. Jordan asked you about whether it was safe to restart TMI-1 pending the completion of the EPRI test program. To your knowledge, has the Commission or the staff ordered any pressurized water reactors shut down pending the completion of that program, that test program?

A (WITNESS ZUDANS) No.

MR. DORNSIFE: I would like to ask a follow up while we are waiting.

BY MR. DORNSIFE:

Q Could you tell us what is so different about two-phase and solid flow that makes it a unique situation as far as the valve testing? Is it strictly loading on the valve or are there other considerations involved?

A (WITNESS ZUDANS) I don't believe that the loading is the primary interest. The primary interest with two-phase flow is the capacity of the valve, and that is why the staff is interested in two-phase flow testing.

Q In other words, the two-phase flow -- certainly, solid flow, you probably get more capacity, more pounds of fluid out of the system than steam flow; is that not correct?

A (WITNESS ZUDANS) Yes.

Q The concern is that possibly with slug or two-phase flow there may be discontinuity, you may get less than steam flow?

A (WITNESS ZUDANS) I believe that with solid flow we are concerned about the loading on the valve. You might have higher loadings on the valve due to the solid flow than you would in steam or two-phase flow.

Q There are two different concerns, loadings for solid flow and capacity for two-phase flow?

A (WITNESS ZUDANS) That's correct.

FURTHER BOARD EXAMINATION

BY DR. JORDAN:

Q I believe Mr. Jones quoted the figures on the capacity of the valves. I presume or I suspect that those were figures --

I presume or I suspect that those were figures for steam released in pounds per hour. Do you think that was -- would that be your understanding?

A (WITNESS ZUDANS) That's correct.

Q Is it significantly less for, say, two-phase flow?

A (WITNESS ZUDANS) That's what I think we want to find out.

Q You don't know yet really?

A (WITNESS ZUDANS) Yes, sir, we don't know yet.

Q

BY MR. DORNSIFE:

Q Is there reason to suspect that there would be a large difference, a large reduction? Are there other tests that have been done with two-phase flow that lead you to believe that would be the case? Or nobody has verified it yet? Is there any reason that you think that might be the case?

A (WITNESS ZUDANS) I think that no one has really verified it. Maybe Walt can help me with the two-phase flows a little bit more.

A (WITNESS JENSEN) I think, in general, for a flow through pipes, I have looked at a number of experiments, and the flow is greater for two-phase flow than it is for steam.

BY DR. JORDAN:

Q In pounds per hour?

A (WITNESS JENSEN) Yes. It is still greater for solid water.

Q This must come into your calculations for a small-break LOCA.

A (WITNESS JENSEN) Yes, it does. And there are critical-flow tables that are used that are based on experimental data.

Q I see. Good.

DR. JORDAN: I think that's all the questions I have.

(Board conferring.)

CHAIRMAN SMITH: What do you want to do now?

MS. WEISS: I can do some of this.

CROSS EXAMINATION

BY MS. WEISS:

Q For the purposes of this testimony have you looked at all at the capability of the PORV, Mr. Jensen? I am referring to the amount of water that can flow with the various phases, water, steam, in two-phase?

A (WITNESS JENSEN) Yes, I have evaluated that.

Q Can you describe the extent of your evaluation?

A (WITNESS JENSEN) I don't remember the pounds per hour off the top of my head. There are tables that are based on experiments that have been done for critical flow.

They give the flow rate in pounds per hour per square foot as a function of the fluid that enters the valve.

Q I want to make sure I understand what your assumptions are in your answer to Question 7 with respect to the doses from a stuck-open PORV. Dr. Jordan asked you about that. I believe your answer was that those doses assume no core damage; is that correct?

A (WITNESS JENSEN) The analysis I was referring to from the FSAR had some core damage. The activity in the gap of the fuel rods was released, which would -- which would occur if the cladding were damaged or burst in some manner.

DR. JORDAN: Would this be under the assumptions of 50.46 Appendix K?

A (WITNESS JENSEN) I don't know.

Q Do you know how those doses would change if you assumed the amount of core damage that had occurred during the TMI-2 accident?

A (WITNESS JENSEN) They would be increased, but I don't know how much.

Q Do you have an idea how much orders of magnitude?

A (WITNESS JENSEN) No, I really don't know.

Q No idea at all? You stated that you have taken a look at the failure rate of PORVs and that you discovered 9 failures out of 300 challenges. Can you discuss that some more, what sorts of failures and what the causes were?

A (WITNESS JENSEN) This is written up in NUREG-0565. I haven't looked at that in some time. There were binding stems, I believe, and loss of -- let's see, what was it -- binding stems -- that comes to mind, the stems, different components would bind up the valve. They would bind up inside the valve.

Q Do you know if any of those failures were due to loss of power, loss of --

A (WITNESS JENSEN) I don't know, but I understand the valves were closed on loss of power, at least the one at TMI-1.

Q But with respect to these others, you don't know how many of which that is also true?

A (WITNESS JENSEN) I don't remember.

Q Mr. Zudans, you stated that you believe the PORV has a low probability of abnormal leakage. I believe that was the final question to you. I just wanted to make sure that I asked the question with respect to the exact language in GDC-14. I believe that is extremely low probability of abnormal leakage. Do you believe the PORV has an extremely low probability of abnormal leakage?

A (WITNESS ZUDANS) I think if you consider the pressure integrity of that valve, then it has a very extremely low probability of leakage, yes.

Q When you say, "If you consider the integrity of the pressure boundary," any opening -

A (WITNESS ZUDANS) I did.'t say that.

Q Any opening of the PORV would constitute a breach of the reactor coolant pressure boundary; wouldn't it?

A (WITNESS ZUDANS) Opening would be a normal function of the valve, as long as it closes again.

Q Considering those instances in which the valve has failed to reseat, do you still believe that the PORV has an abnormal -- has an extremely low probability of abnormal leakage?

A (WITNESS ZUDANS) No.

Q I was also interested in your answers to Dr. Jordan about whether the staff is confident about the outcome of the EPRI test program or the valve testing program.

And his question in connection with that about where the staff finds reasonable assurance that the plant is safe enough to operate until those tests are done, I understand your answer to be that you find reasonable assurance based on measures that you have taken to detect a stuck-open PORV and not based on any prejudgment of the results of the EPRI program. Is that correct?

A (WITNESS ZUDANS) My answer was that I believe the safety of the public will not be endangered because of the measures that the TMI staff has taken to first identify the source of the leakage by the sensors, the procedures, the lessening of the challenges to the safety systems.

I do not like to prejudge a test program. However, there are indications from other incidents, such as Crystal River, that show some of the -- some potential results could occur during the pretesting.

Q What incidents other than Crystal River?

A (WITNESS ZUDANS) None that have been documented. However, if you have transients, you would have the type of loads that we are talking about.

Q Is it your professional opinion that what you observe from the Crystal River event and any other indications that you have would allow you to state with confidence that the testing program will show that the valves are capable of relieving two-phase and solid water flow without a high leakage rate?

A (WITNESS ZUDANS) I think the testing program will definitely show that the valves will allow the relieving capability. The amount of leakage, I cannot make a determination on now.

Q The testing program goes both the -- it goes to all the valves, I guess, the PORV, its block valve, and the safety valve. You referenced measures that the staff has taken to enhance the ability of the operator to detect a stuck-open PORV. Considering Mr. Jones' testimony that challenges to safety valves have been increased, where do you find reasonable assurance that measures have been taken to protect against safety valve failures?

MR. BAXTER: I don't recall any such testimony by Mr. Jones, Mr. Chairman. I object to the characterization.

CHAIRMAN SMITH: Do you object to a short-cut? Mr. Jones is sitting here.

MS. WEISS: That's fine. I assume I heard it wrong. Was that not what you said?

WITNESS JONES: I do not believe I stated that the frequency of relief valve -- of safety valve challenges has been increased.

CHAIRMAN SMITH: You will have an opportunity to address that again when you get the transcript tomorrow.

MS. WEISS: I think the question is also a valid question, even if the challenge rate to the safety valve remains the same. In other words, you have taken -- the staff and the licensee have taken certain steps to protect it against PORV failure. But the safety valves also haven't been tested, and what specific measures have you taken to protect against safety valve failure?

A (WITNESS ZUDANS) I don't think that the safety valves are going to be challenged any more now than they were previously.

Q Take that part out of the question. I don't think it matters whether they are going to be challenged. More or less assume that they are going to be challenged at the same rate. The point is that the valve has not been qualified to operate for two-phase or solid water operation. Where do you find reasonable assurance that it is safe to operate the plant under those conditions? What measures have you taken to protect against failure of the safety valves?

A (WITNESS ZUDANS) I have not been able to find any occurrences where the safety valve did not perform its function, and that is of over-pressure protection. So I really can't address your question very well.

Q Wouldn't the answer to my question be that the staff has taken no measures to protect against safety valve failure?

A (WITNESS ZUDANS) We have no belief at this time that the valves will not operate.

Q Why are you doing a test?

A (WITNESS ZUDANS) The tests are confirmatory tests.

Q What evidence do you have that these are going to confirm what you think they are going to confirm?

A (WITNESS ZUDANS) Like I said, we had Crystal River, which was one, and there has been no evidence that the valve will not perform its function.

Q That's interesting. I find that statement pops up again and again in NRC documents. You are doing confirmatory tests, and you don't have any evidence one way or the other. Other than the Crystal River event, one event, you have no evidence to show that the valve either will work or won't work under two-phase or solid water flow. Isn't that essentially accurate?

MR. CUTCHIN: We need a definition of "will work or won't work." Is the failure that is being postulated one of failure to open or failure to close?

CHAIRMAN SMITH: Ms. Weiss, what is your --

MS. WEISS: I am trying to decide whether it makes a difference. Let's say: failure to -- answer both ways: failure to open and failure to close.

WITNESS ZUDANS: There is no evidence that I have read that the valve will fail to open. And my own personal looking has not found any evidence where the valve did not close, either.

BY MS. WEISS:

Q Was it tested?

A (WITNESS ZUDANS) Yes, the valve has been tested for saturated steam.

Q They have never been tested for two-phase or solid water flow; is that correct?

A (WITNESS ZUDANS) Not with documentation, no.

Q Other than the Crystal River event, there is no evidence that they will function as intended for solid water and two-phase flow; is that correct?

A (WITNESS ZUDANS) That's correct.

MS. WEISS: I have no further questions at this time. I do have some more, but I don't feel competent to go forward with them.

CHAIRMAN SMITH: All right. Anything further before we adjourn for this evening?

MR. BAXTER: I have one, Mr. Chairman.

CHAIRMAN SMITH: All right.

CROSS EXAMINATION

BY MR. BAXTER:

Q I thought you said in response to Ms. Weiss' question about general design criterion 14, that if you considered the PORV opening and failing to reclose, you couldn't say that that represented an abnormally -- an extremely low probability of abnormal leakage. Did you consider in that answer the PORV block valve and its ability to isolate?

A (WITNESS ZUDANS) No, I did not consider the PORV block valve.

Q Is it part of the reactor coolant system pressure boundaries?

A (WITNESS ZUDANS) It becomes part of the reactor coolant pressure boundary when you close it.

MR. BAXTER: Thank you. That's all.

MR. DORNSIFE: I have one follow-up.

CROSS EXAMINATION

BY MR. DORNSIFE:

Q When I ask you what is different about two-phase and solid flow as far as loadings on solid flows, isn't it relatively simple without testing to determine what the loadings are on a valve and design the valve structure and supports to take that into account?

A (WITNESS ZUDANS) I would say under steady-state conditions, yes.

Q Really, two-phase flow is the only unknown as far as a possible problem area; is that correct?

A (WITNESS ZUDANS) Yes. Except that we know there are tremendous nonlinearities when it comes to qualifying any valve, any component such as this. Where there are gaps, there are always nonlinearities. We would like to have the tests to confirm the results.

Q Mr. Jensen, didn't you testify also that the experiments that have been done on two-phase flow indicate that it is greater typically than steam flow?

A (WITNESS JENSEN) Yes.

Q There is no reason at this point to think that it would be less, is that not correct, based on those experiments?

A (WITNESS JENSEN) We haven' tested a PORV, of course, but there is data for pipes and flows through valves and orifices, but not PORVs. This data shows that the mass flow rate for all the other data I have seen, the mass flow rate has gone up.

Q It would be the same phenomena as the flow through the relief valve; is that not correct?

A (WITNESS JENSEN) The magnitude might change.

Q As far as the phenomenon of flow through an orifice, you would expect the same kind of results?

A (WITNESS JENSEN) I would think so.

MR. DORNSIFE: Thank you.

CHAIRMAN SMITH: Anything further this evening?

MR. CUTCHIN: No, Mr. Chairman.

CHAIRMAN SMITH: This panel is not to be excused. That is our understanding.

MS. WEISS: I have talked to Mr. Pollard at lunchtime at the lunch break today. And unless he is Lazarus, I don't think he is going to be able to make it in tomorrow. He sounded very bad.

CHAIRMAN SMITH: Let's consider the possibility that he will not. What will we do in that event? Are there any recommendations?

MR. CUTCHIN: With respect to Mr. Jensen, there is no real problem. But with respect to Mr. Zudans, there is no plan to bring him back on any other contentions. And, of course, to have him stay around tomorrow or to have him come back next week or some unknown time is indeed somewhat burdensome. But it would be nice if we could get some indication. There wasn't -- in fact, there was no rebuttal testimony of Mr. Zudans. I would have thought that the bulk of the questions could have already been focused on before today. I have difficulty conceiving of a great deal that could arise out of what has been covered today that Mr. Pollard would home in on.

CHAIRMAN SMITH: You are not prepared to release Mr. Zudans, however?

MS. WEISS: No.

MR. BAXTER: I would hope Ms. Weiss would be able to consult with Mr. Pollard this afternoon yet and come in and finish the examination. Our rebuttal also is only 20-25 minutes, part of which was Mr. Jordan questioning. It wasn't extensive.

MS. WEISS: It may be possible, but it may not. And I will make every effort to do that.

CHAIRMAN SMITH: Is it okay with you, Mr. Zudans, if you defer your departure until at least tomorrow morning, and we will see what happens?

WITNESS ZUDANS: Yes.

CHAIRMAN SMITH: Let's make plans for -- what are we going to do tomorrow, then?

MR. CUTCHIN: We start off with the argument. And after that I would have to focus on the schedule here. I don't have my next witnesses. I don't know whether the Licensee does. It doesn't seem likely that we would get through another issue tomorrow, but I could be surprised.

MR. BAXTER: I would have hoped that after the argument on Dr. Beyea's testimony, we could return our panel to the stand for cross examination on rebuttal.

MS. WEISS: I don't think that's going to be possible. The man is sick, Mr. Baxter. I am not going to be able to spend four hours of time tonight going over the rebuttal. He's sick in bed.

MR. BAXTER: Yes, ma'am. I was just suggesting that you at least check to see --

MS. WEISS: I will certainly check.

MR. BAXTER: -- to see if on 20 or 25 minute rebuttal testimony you can't do it.

MS. WEISS: I will certainly check. But I thought it would be a courtesy to let you all know that I don't think it will be possible. And maybe there is no need for everybody to appear here at 9:00 a.m. tomorrow morning, or 8:30.

MR. CUTCHIN: We could go forward with the argument that was planned, and then we will just have to see how far we go, and when we run out of gas, I guess we have got no choice.

CHAIRMAN SMITH: All right.

MR. CUTCHIN: I do not have my witness for the next issue, nor did I plan to bring him up tonight.

CHAIRMAN SMITH: That's fine.

Ms. Bradford, you may be the entire program for tomorrow morning.

MR. TROWBRIDGE: At 8:30, Mr. Chairman?

CHAIRMAN SMITH: Yes.

MS. BRADFORD: If you would rather that do that this evening, that would be all right with me.

CHAIRMAN SMITH: Are you prepared to do it this evening?

MS. BRADFORD: Yes, sir. I was able to get some papers from Mr. Sholly.

CHAIRMAN SMITH: It doesn't matter now.

MS. BRADFORD: If people have to come tomorrow anyway.

CHAIRMAN SMITH: There is nothing to be gained. So if you prefer tomorrow, we will keep it at 8:30 tomorrow.

MS. BRADFORD: That's fine. Thank you, sir.

CHAIRMAN SMITH: That will give us a chance to prepare for it, too.

So we will adjourn until 8:30 tomorrow.

(Whereupon, at 4:59 p.m., the hearing was adjourned, to reconvene at 8:30 a.m. Friday, December 19, 1980.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

____

in the matter of: METROPOLITAN EDISON COMPANY (TMI UNIT 1)

Date of Proceeding: December 18, 1980

Docket Number: 50-289 (Restart)

Place of Proceeding: Harrisburg, Pa.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Barbara L. Miller

Official Reporter (Typed)

Barbara L. Whitlock

Official Reporter (Signature)