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TASK FORCE ON

THREE MILE ISLAND ACCIDENT

Washington, D. C.

TUESDAY, MAY 15, 1979

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U. S. House of Representatives,

Committee on Interior and Insular Affairs,

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The committee met at 9:50 a.m. in room 1334, Longworth House Office Building, the Honorable James Weaver, presiding.

Present: Representatives Weaver, Carr, Markey, Vento, Marriott, Edwards, and Cheney.

Staff present: Messrs. Myers, Burnam, Terrell, Reis, Scoville, and Chakoff.

Also present: Messrs. McMillan, Deddens, Nelson, Ellis, Bergson, Edwards, Benitez, Edgar, and Stang.

Mr. Weaver. All right, the Task Force will be in session.

We're very pleased that members of the firm of Babcock & Wilcox, the engineering firm that designed the Three Mile Island Unit 2, as well as other pressurized light water reactors in the United States, are here to give us their responses to the various questions that have been raised by the Three Mile Island Unit 2 accident. And we would like to -- we have a series of questions here prepared by staff, and that's the format that I intend to follow, unless you have some remarks you'd like to make at the outset. If you do, please go ahead.

Identify yourselves for the record, and we want to welcome you and thank you for coming. And why don't you start by identifying yourself and telling us whether you have remarks you'd like to say.

Mr. McMillan. Thank you, Mr. Chairman.

My name is John McMillan. I'm Vice President of the Nuclear Power Generation Division of the Babcock & Wilcox Company. Our Division has the responsibility for the design, development, manufacture and support of the startup and the operation of the commercial nuclear plants, which includes a number of contracts, including the Three Mile Island 2 nuclear steam system and the nuclear fuel for that plant.

I have with me Mr. James C. Deddens, who is Manager of Project Management. He was my deputy and was in charge of the

Lynchburg operations at the time. On March 28th, when the incident occurred, I was out of town and he was acting on my behalf, and I thought it would be appropriate for him to be here, since I understand you have some questions about what might have happened in the early hours of the incident.

Mr. Weaver. You mentioned Lynchburg. Would you tell us what Lynchburg is?

Mr. McMillan. Yes. Lynchburg, Virginia, is the headquarters of our Nuclear Power Generation Division, and all of the engineering and project management service personnel in the B&W Company are located in the Lynchburg facility.

Mr. Weaver. What ratio of business that B&W does is in nuclear design?

Mr. McMillan. I'm sorry, sir, I didn't hear the question.

Mr. Weaver. You do other things than design nuclear plants?

Mr. McMillan. Yes, sir.

Mr. Weaver. What ratio is nuclear? Half of your business?

Mr. McMillan. Well, let me express that in the following terms. The Babcock & Wilcox Company, of course, is a wholly owned subsidiary of J. Ray McDermott. Let me speak specifically to the Babcock & Wilcox business.

Our business is heavily involved in the steam generating business. About 50 percent of the business of Babcock & Wilcox is supplying equipment to the utility industry for steam generation. That can either be nuclear equipment or the more conventional fossil boilers.

The backlog of the B&W Company runs on the order of $4 billion, and approximately half of that backlog right now is represented in all phases of our nuclear business. The backlog would stand to emphasize the nuclear more because of the long-term nature of the nuclear business, so that our business would not be 50-50 on a current sales basis. We've never identified what that specific breakdown is.

Mr. Weaver. The backlog of $4 billion is actually the sales by B&W as opposed to the cost of the plants themselves that you're working on?

Mr. McMillan. That's sales by B&W, yes, sir.

Mr. Weaver. What total sales did B&W receive from the TMI-2, do you know what?

Mr. McMillan. I don't have a number. I would specifically be glad to obtain that.

Mr. Weaver. Would you?

(Information to be furnished.)

Mr. McMillan. You did raise an important point in terms of the total cost of the Three Mile Island plant. Our scope of supply represents approximately something in the range of 10 to 12 percent of the total plant costs.

Mr. Weaver. Roughly $100 million, then.

Mr. McMillan. Well, of course, I don't believe this plant cost a billion dollars at the time it was built.

Mr. Weaver. I'm not sure of that. I don't know.

Mr. McMillan. If you were to start today and build a unit, I think you would be looking at a billion dollar plant, and our scope could be in the range of $100 million, I guess, is a fair representation.

Mr. Weaver. Very good.

Did you have a statement?

Mr. McMillan. Yes, sir. I would like the opportunity, with your permission, to go through a summary of the events that occurred on the morning of March 28th as we view it and identify what we consider to be some of the crucial factors in that sequence of events.

Mr. Weaver. I think that's very good. We welcome that.

How long will this take?

Mr. McMillan. I would estimate about 15 minutes.

Mr. Weaver. Oh, fine. Please go ahead.

Mr. McMillan. I have some overhead transparencies, and I believe copies were made of these.

(Slide.)

This is a diagram of the secondary system at Three Mile Island 2, somewhat simplified. You'll see the steam generators from the nuclear steam system shown on the right-hand side of the diagram. Normally, steam is transmitted from those steam generators to the turbine generator, and from the turbine generator and the condenser, where the steam is returned to the water state, through condensate pumps, condensate polishing equipment, which maintains the quality of the water, through condensate booster pumps, low-pressure feedwater heaters, main feedwater pumps, high-pressure feedwater heaters, through the feedwater control valves, and back in to each of the steam generators.

In the event of an emergency where, for some reason, main feedwater is lost, there are auxiliary feedwater pumps. There are three of these, two of which are motor-driven, one of which is steam-driven. These take suction either from the large condensate storage tank source of water or from the main feed system, and discharge through control valves and back valves into the steam generators in the upper region of the steam generator, in order to give the maximum force for natural circulation under emergency conditions.

At 4:00 o'clock in the morning of March 28th, the operators were working with the condensate polishing equipment, and I don't know at this point precisely what happened there, but they had some difficulty, which ended up in blocking the feedwater flow coming from the condenser to the condensate booster pump. They were unable to get this bypass valve open in the early seconds of this transient.

The condensate booster pumps, having lost suction pressure, turn off automatically, and that's what they did. That reduces the suction pressure on the feedwater pumps and they turned off automatically, which interrupted all feedwater flow to the steam generators.

Under those circumstances, the equipment is designed so that the auxiliary feedwater pumps come on automatically, which they all did, and the turbine trips. Stop valves in the turbines close automatically. The auxiliary feedwater pumps came on, the control valves functioned properly, and the operators are reported to have looked at the pressure downstream of these pumps and established that immediately they had pressure, those pumps were running.

They noticed, however, some time later, that they were not getting any change in the water level in the steam generator; the water level was in fact decreasing. And this puzzled them somewhat. And about eight minutes in the transient they discovered that these valves, which are shown open here, which would be their normal operating position, for some reason were closed. And so they were not getting auxiliary feedwater to the steam generator. There was no feedwater supply to the steam generator at all, and that certainly is one of the key events I'll mention later in the sequence.

(Slide.)

Let's turn now to the primary system or the reactor coolant system, as it's sometimes called. You can see the steam generator shown schematically here, the second one over here.

Let me draw your attention to the reactor vessel. The reactor core is contained in the reactor vessel, and high-pressure water comes into the reactor vessel, goes through the core, is discharged through the high-temperature hot leg, as we call it, of the reactor coolant system, goes into the top of the steam generator and comes down through the steam generator tubes, discharges through two outlets in the bottom of the steam generator, goes through reactor coolant pumps, which circulate the water back to the reactor; a similar loop on the other side of the reactor vessel.

On the A loop of the reactor vessel there is a pressurizer, and the free surface in the pressurizer between the water and the reactor coolant system, and steam in the top of the pressurizer; and this maintains the reactor coolant system pressure during operation.

At the top of the pressurizer there are two safety relief valves --we call them code valves -- which maintain the pressure within the design limits for the reactor coolant system, and one pilot-operated relief valve, whose set pressure is below the safety valves, and is put in there for relieving under-pressure transients and avoiding the use of the safety valves unless you get into a very extreme condition.

Upstream of that pilot-operated relief valve is an isolation or block valve which allows you to shut off that valve if, for some reason, it should not reseat properly, without violating the safety or code requirements, which are still maintained by the safety valves themselves.

Each of these valves discharges into a reactor coolant quench tank or drain tank inside the reactor building. And then, on this drain tank is a pressure relief valve and a rupture disk, which protects the integrity of the drain tank, relieving to the reactor building itself.

Provided as part of the safety system are high-pressure injection pumps, which pump water into the reactor coolant system at the four locations indicated here. During normal operation we let down a fraction of the fluid through a decay heat makeup and purification system. It comes off the high-temperature line, goes down through the makeup system and comes back, normally, through this makeup line shown here. In addition to that, for emergency conditions there is a letdown system, and this is for low-pressure coolant, a letdown system that comes off the bottom of the steam generator, goes to letdown coolers, and then is brought back into the reactor coolant system, directly into the reactor vessel for emergency low-pressure injection.

There also are core flood tanks, two core flood tanks, which are pressurized with nitrogen and are provided for flooding the reactor vessel when the pressure drops to a very low level in the case of a major loss of coolant accident.

Now, to trace the sequence of events, what happened when they interrupted feedwater flow to the secondary side of the steam generator and they shut off the turbines so they weren't drawing any energy out of the steam generators, we were putting energy into the reactor system and the reactor itself, the temperature of the reactor coolant system heated up, the volume of the coolant increased.

That caused a pressurizer level increase, compressed the steam in the top of the pressurizer, and up to the point where the pilot-operated relief valve opened, and it opened at the pressure that it was designed to open. That happened at about eight seconds into the transient.

The pressure continued to rise following that and it rose to the point where the reactor protective system automatically shut down the reactor, at a pressure somewhat above the operating pressure of the pilot-operated valve, but below the safety valve settings. That reactor protective system did in fact work, and it shut the reactor down at 12 seconds into the transient.

That terminted the rise in pressure, and then the reactor coolant system started to decrease in temperature, and we then had a case where we were -- in fact, pressure was falling, and this valve should have reseated. It did not. It stuck open. And that certainly is another significant item in the sequence of events. It stuck open and was not diagnosed by the operators for about two and a quarter hours into the accident, although it should have reseated in the first, say, 15 seconds of the transient.

Pressure continued to decrease in the reactor coolant system because we were still blowing steam out of the pressurizer and it decreased until the pressure reached about 1600 pounds in the reactor coolant system. And at that point, the reactor safety system calls for automatic startup of the high-pressure injection, and in fact they did start up and they started pumping water into the reactor coolant system, as designed.

The operator noticed shortly thereafter that his level in the pressurizer started to go back and he was reaching the top of the pressurizer. And apparently -- we don't know what the operator was thinking precisely, but apparently he elected to cut back on the high-pressure injection flow in order to keep from going solid, getting full-water condition and blowing water out the relief valves.

He recognized he had not yet identified the fact that our pilot-operated relief valve was still open. So he cut back on the flow of high-pressure injection.

The pressure continued to drop. It dropped down in the range of 1300 pounds, and at that point we were reaching a saturation pressure of the hot leg, the hot temperature of that reactor coolant coming out of the reactor. And what happened then was, we started to flash or form steam in the reactor coolant system, which was swept through the reactor coolant system by the operating pumps.

This continued for some period of time, and at 73 minutes into the accident the operators noticed they were getting erratic behavior of the coolant pumps, high vibration on the pumps. They were worried about the performance of those pumps. And again, I don't know exactly what line of thought they went through, but the conclusion they reached was to secure the two pumps in the B loop at 73 minutes.

Continuing into the operation, and pressure still at a low level and being held at that level by the flashing of the water in the high-temperature portion of the reactor coolant system.

At 100 minutes into the accident they turned off the other set of reactor coolant pumps, which ceased all forced circulation through the reactor core. Our analyses indicate that in that situation they began to boil water in the reactor core; and somewhere in the range of two hours they uncovered the top of the reactor core, and in the range of time from two to four hours boiled away and uncovered the core.

Evidence is that the cladding temperature rose to a high level, in the range of 2 to 3,000 degrees Fahrenheit, there was a substantial oxidation of the zirconium and reaction with the steam and water, and it generated a large amount of hydrogen, and the zirconium oxidized, the hydrogen being the source of the famous bubble in the reactor system and also a source of hydrogen which subsequently got out into the reactor building through the relief valve.

The reactor stayed in this condition for some period of time, and then finally, after about 15 hours of operation, the operators, at the recommendation of a number of people, cranked up the high-pressure injection to full flow, repressurized in the reactor coolant system, got one of the reactor circulating pumps in the A loop back into operation, and stabilized at about 1,000 pounds pressure and about 280 degrees. And we stayed at that condition for a number of days following the early first 15 or 16 hours of the incident.

Now, I ought to mention while I have this diagram on the chart or on the screen, that there are some indicators on this quench tank. There is a level indicator, there's a pressure indicator. These are alarmed in the reactor control room, so that if they exceed their design levels an alarm goes off that has to be acknowledged by the reactor operator.

There are thermocouples on the discharge piping coming out of these relief valves which, through the computer, can be called up to determine whether or not there is hot fluid in this line or not.

And I'm not sure -- I can check my sequence. At some point in the sequence, the pressure in this tank got to the point where not only did the relief valve operate, but subsequently the rupture disk in that tank broke and ventilated this tank to the reactor building.

Now, that is a very quick summary of the sequence of events. Let me turn to what we consider to be the significant factors in the accident.

(Slide.)

And we have identified six here. These are the same six factors which the NRC has developed and has discussed in some of their proceedings.

Certainly, the delayed auxiliary feedwater was a significant factor. Auxiliary feedwater block valves should have been open. They were not. They stayed closed for about eight minutes. That produced behavior in the reactor coolant system which the operators were not familiar with from their previous experience or from training, and inhibited their ability to get the accident under control.

The second significant factor was the pilot-operated pressurizer relief valve not reseating. That was a straight mechanical failure. Probably of greater significance was the lack of recognition of that fact until two and a quarter hours into the event.

Mr. Weaver. Mr. McMillan, we have had that described to us as the power-operated pressure relief valve.

Mr. McMillan. It's sometimes called power-operated; it's sometimes called pilot-operated.

Mr. Weaver. You mean "pilot operator"?

Mr. McMillan. The reason we call it the pilot-operated relief valve is that the valve is designed such that to actuate, to open the valve, you must move a pilot stem, which vents the steam to the side of the valve, which actually causes the valve to open. And then, when you reposition the pilot, the valve closes. So that it's a pilot-operated valve, rather than, in the typical safety valve, you've just got the disk with a spring behind it and the pressure gets high enough to lift that spring; the pressure itself is the thing that operates the code safety valve.

So this is a pilot-operated valve, where in fact you have to move the pilot stem in order to vent the pressure to the right part of the valve.

Mr. Weaver. Now, the solenoid was de-energized; therefore they thought that that valve was closed. But in effect, the solenoid was de-energized, but the valve did not close.

Mr. McMillan. Right.

The pilot I talked about, a spindle that vents the pressure to the actual valve itself is actuated by a solenoid. When you energize the solenoid, it moves the pilot down; and when you deenergize the solenoid, the pilot is supposed to return to its position, which would call for the closing of the valve.

Clearly, in this case, for some reason, the valve didn't close. We don't know why. At this point, we won't know until we get in there and see it.

You are correct that the indicator that the operator has tells him whether or not there is power to that solenoid. And the presumption is that when there's power in the solenoid, the valve is open. When the power is off, the solenoid -- the valve is closed.

But that indicator does not actually measure the position of valve.

Mr. Weaver. Why doesn't the operator have a light or a gauge or whatever, a control to tell him that the valve is actually closed?

Mr. McMillan. Well, that could be provided, and one of the things that we have said in our earlier testimony to the Advisory Committee on Reactor Safeguards is that we will, in fact, develop an offer, an indicator which will tell whether the valve itself is open or shut.

Mr. Weaver. Why didn't you do that originally?

Mr. McMillan. I have no answer for that.

We felt that the important thing was for him to know whether there was power to that valve or not, and we had provied back-up information to him, which allowed him to determine whether the valve was open or shut.

And I mentioned the readings of the pressure, temperature, adn volume in the quench tank, and the temperature on the tailpipe, or the discharge piping on the other valve, as means of indicating or measuring whether or not the valve was fully closed, or whether it was just partially closed

Mr. Cheney. On those --I don't mean to take you away from your presentation; I want to get back to it -- but on those alternative indicators that PORV was open, we had some testimony to the effect that the tailpipe temperature did not rise significantly during that key period of time before they discovered it --

Mr. McMillan. My information indicates that operator did, in fact, ask the computer what the temperature was on that tailpipe, and he got a reading somewhere around 280 degrees, in that range.

Mr. Cheney. -- whereas, he said his expectation would have been somewhere around 600 degrees, as I recall.

Mr. McMillan. The maximum temperature that reads out on the computer is 280 degrees. The thermocouple probably was in fact indicating higher, but the printout, I am told, is limited to 280 degrees.

Normally, if there were no steam flow in that line, he would normally expect to see a temperature which would be close to the reactor building temperature in that region, which would, say, be 100 to 120 degrees.

Mr. Cheney. Except he knew the PORV had opened early on in the transient.

Mr. McMillan. Yes, sir, and then there would be some time for that temperature to come back down as the pipe cooled off.

Mr. Cheney. Is there a manual that specifies the readings you expect on that tailpipe under recurring circumstances?

Mr. McMillan. I can't answer that question. I don't know.

Mr. Carr. What about your simulator up in Lynchburg in your computer program; do you have the training scenarios? Are they trained to look at that thermocouple and try to interpret meanings from it?

Mr. McMillan. One of the training exercise that we do conduct is the identification of either a simmering or an open relief valve. More frequently, this information is used to tell whether you have a weeping or a simmering type situation with the power-operated relief valve.

And that particular transient is quite familiar.

Mr. Cheney. Where are those indicators located in the control room?

Mr. McMillan. The temperature indicator on the discharge piping comes through the computer, as part of the computer display.

Mr. Cheney. In other words, there isn't any one place on the control panel where it's laid out? You have to query the computer.

Mr. McMillan. The pressure, the volume, and the conditions in the quench tank are located on the quench tank panel, which is around behind one of the large vertical panels. All of that, including the alarm and the acknowledgement of the alarm, are located on that panel.

Mr. Cheney. So it's on the back side. It's not out front?

Mr. McMillan. That's right.

Now, in order to acknowledge that alarm when it went off for showing high pressure in the quench tank, somebody had to walk around behind that tank and push the acknowledge button.

Mr. Weaver. Mr. McMillan, there was a similar occurrence at Davis-Besse in 1977, where a relief valve, a power-operated relief valve, stuck open for 20 minutes. What was your response to that? Did you make any changes or inform anybody as to this possibility occurring in other plants?

Mr. McMillan. Yes, sir, the event at Davis-Besse that you referred to did result in the sticking open of the pilot-operated relief valve. The pilot-operated relief valve at Davis-Besse we purchased from Crosby. That's a different supplier than the power-operated relief valve.

Mr. Weaver. You got this from Dresser.

Mr. McMillan. This is a Dresser one.

After the incident at Davis-Besse, we did go and inspect the valve and had our suppier assess the condition of the valve. And what we found there was that a relay in the electrical circuitry controlling that valve had been left out in the electrical equipment.

The purpose of that relay was, once the valve had opened, to allow the pressure to drop to the point -- enough so that when it reseated, it would just not operate open again. If you don't have some dead band in there for the valve, then it just sits there -- open, close; open, close -- in just a cycle. In fact, that's what happened. That valve just sat there in its cycle until it finally jammed itself.

Mr. Weaver. At Davis-Besse?

Mr. McMillan. At Davis-Besse.

The Davis-Besse valve, because of the absence of this relay in the electrical circuitry, just sat there and open, shut; open, shut; open, shut, until it jammed open.

What we did at that point, obviously, was to make sure that that relay was added, as it should have been in the original design, to the electrical circuitry. And we went took that valve and inspected it and determined that there was some clearances that ought to be opened up; and, in fact, opened up those clearances, made a modification of the valves, checked it out again for performance, and reinstalled it at Davis-Besse.

Mr. Weaver. Are you saying that Crosby didn't fail to put this in, or you didn't design it in?

Mr. McMillan. It was in the design. It was not in the portion of the equipment that Crosby supplied. Crosby supplied the valve. It was in the control equipment which fed the electrical signal to the solenoid on the valve.

Mr. Weaver. Whose fault that this signal -- or, what did you call it?

Mr. McMillan. It's a relay.

Mr. Weaver. Whose fault was it that the relay was not there at Davis-Besse?

Mr. McMillan. I believe -- and I'd like to check this, but I believe that that equipment was supplied by Bechtel, the architect/engineer. I believe that electrical equipment was supplied by Bechtel.

Mr. Weaver. What I'm really trying to establish here was was it just simply an oversight, that you didn't think it was needed? Or did somebody actually not put in something that was called for?

Mr. McMillan. The relay that was not put in, that was called for.

Now, because the Davis-Besse valve was a Crosby valve, and because the problem was in the electrical circuitry and the relay, the actions that we took at that point were unique to the Davis-Besse design and were not applicable to any of the earlier designs with the Dresser valve. So there was no corrective action called for in terms of the other operating units.

Mr. Cheney. Mr. McMillan, I believe it was March of 1978 a problem with a stuck open PORV at TMI Number 2.

Mr. McMillan. We've had the Dresser valve stick open, first of all at Oconee. And what we did in that case, again, was to look at the valve, inspect it, determine what was the cause of that problem -- this is the Dresser valve. There were modifications in that valve to change some of the clearances in the valve itself, and there were modifications to the maintenance procedures for the inspection and maintenance of those valves.

Those modifications and those changes in procedure were transmitted to GPU and were incorporated in the valves at Three Mile Island.

I believe you're correct, that there was na earlier occasion where we had a stock-open valve at Three Mile Island subsequent to those modifications that were recommended from the Oconee incident.

Mr. Cheney. This is March 29th of '78. Do you know what steps were taken to correct that situation then?

Mr. McMillan. I do not know that.

Let me find out, and I'd be glad to get that record for

Mr. Weaver. Then we've had other failures of this particular valve, the power-operated relief valve, the power-opearted relief valve -- Rancho Seco in June 1978, where we had valve leakage.

Mr. McMillan. I believe, if my information is correct, we've had four occasions on which we've had a stuck-open, power-operated relief valve: the one at Oconee, that I mentioned, the one at Davis-Besse that you addressed earlier.

Mr. Weaver. Arkansas one.

Mr. McMillan. The conditions I was addressing specifically was where the valve stuck wide open. We've had some where we've had weepage, of what we call "simmering," where there might have been a failure for the valve to completely reseat. And frankly, that's the reason why --

(Slide.)

Mr. McMillan. -- that's the reason why we provided an isolationvalve,or a block valve, upstream over the pilot-operated valve, in order that an operator can isolate the pilot-operated relief valve in a circumstance where it does not fully reseat.

Mr. Weaver. The reacon you have this relief valve it has given a great deal of trouble; is that a correct statement?

Mr. McMillan. As I said, we've had four occasions that I'm aware of in which the valve has stuck fully open out of some 150 occasions on which the valve has been called to operate.

Mr. Weaver. The reason, I undersand, that you put the relief valve in the first place is that the Code valves, the other two are of a different nature, and that once they break open, that they don't really close tightly, and therefore you have a real long-range problem. Whereas, the power-operated relief valve can open and close and get you out of transient in a hurry; is that also correct?

Mr. McMillan. The purpose of the pilot-operated relief valve is to provide a means of relieving pressure from the reactor coolant system and a pressure below the acutating pressure for the safety.

Mr. Wesver. It's not required by code, while the safety of the code valves are.

And really, as I see it -- I'm trying to check my own thinking -- as I see it, you put the relief valve, the power-operated relief valve, in there and make the system more energy efficient, because you're going to run into longer down time if those safety code valves open. Is that the reason you put it in?

Mr. McMillan. I would say the purpose is to improve the availability of the system.

Mr. Weaver. Right, the availability.

Mr. McMillan. Because the code valves, not only these on the pressurizer, but the valves on the secondary

The one thing I want that operator to do, in hindsight, is, even though the solenoid is deenergized, go and push that button again, just to make darn sure that that valve is closed, which they don't do until about 2-1/2 hours into the accident. But had the operators really been aware that this relief valve is a serious problem -- having a serious problem, you'd think they'd have done that. Why wasn't that made apparent to them? What do you think?

Mr. McMillen. Well, let me respond in this way. Certainly in retrospect -- and I was going to get to this a little bit later in my presentation -- as we look back at the incident and we look at the actions that we took in the days following the incident, one of the things that we did was to modify our simulator so that it could fully reproduce the conditions of the Three Mile Island incident; and then offer a training program for all the operators, in which we would simulate the events that occurred at Three Mile Island, one; secondly, we d simulate events associated with a loss of feedwater; thirdly, we'd simulate events associated with the sticking open of the pilot-operated relieve valve; and finally, and I think most important, we conduct training on how to identify a situation where you have saturated conditions in the reactor coolant system -- formation of steam in the reactor coolant system.

What are the symptions that you oberved? And then what do you do to get yourself out of that situation?

And that certainly highlighted and brought deal of attention to the recognition of a stuck-open, pilot-operated relief valve.

I would say that again this is a transient, or a fault, an equipment fault, which is included in the training program. And you just have to say, having had the occurrence at Three Mile Island, it wasn't given enough emphasis.

Now, I would say that we certainly have had some difficulty with this valve, and we have made modifications. We've inspected the valve. We have recommended changes in the maintenance procedures for these valves, but it's not unusal with safety valves or relief valves of this sort to have problems with the valves simmering or failing to fully reseat.

Mr. Marriott. Didn't you indicate earlier that notwithstanding the failure of a valve, that the operator should have read other indicators that would have corrected the situation, so you're indicating then that this is an error on the part of the individual rather than the valve, in terms of avoiding the situation.

Mr. McMillan. Let me review that again.

When this valve sticks open, there are a number of pieces of evidence that the operator had at his disposal. He has pressure in the reactor coolant system. His pressure kept going down in the reactor coolant system. In the normal functioning of this equipment, once he got below the reseat pressure on this valve, that pressure should have leveled off at a much higher level than it actually did and continued, as we said, for a number of hours, to decrease. So he had reactor coolant pressure as an indicator. He had an alarm, that somebody had to acknowledge, on the pressure level in the quench tank.

He had reactor building pressure once that quench tank rupture disc opened up, the reactor building pressure started to go up, so we had a reactor building pressure indication that something was abnormal, was unusual. And we've already talked about the temperature indication on the tailpipe of the discharge of the pressurizer valve. And he did, in fact, call that up, and for some reason discounted the information.

So, it's puzzle to me that in the period -- in the first 2-1/4 hours that that the operators -- it's not just one, but the operators did not acknowledge or recognize that this fault condition existed.

Mr. Weaver. They explained this by saying that these temperatures gradually subside or water is still flowing in, that there's a lag time in the times that these things finally dissipate. And therefore, they thought the reactions were normal the things subsiding, because they said they couldn't make those judgments.

Mr. McMillan. You know, I accept that rationale on the part of the operators. I understand that's what they were thinking.

Mr. Edwards. Let me ask you something on that: You indicate that this problem with the valve is occurring, what, about 3 percent of the time, which is, when you're dealing with something that important, is a fairly high percentage of failure. To what extent were they able to make a better judgment or act a little better?

You are saying that they did have some training in dealing with this problem? It was my understanding there wasn't much training. Is this problem taken care of on the computer simulations? Now, I'm not talking about after Three Mile Island, I'm talking about before Three Mile Island. When you'd had this record of this problem with the valve, to what extent were they getting training when they came to your training sessions? To what extent were they getting training to deal with that situation?

Can we reasonably expect them to have been able to look at it, understand what was happening, and react differently that they did?

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Mr. McMillan. Even before the Three Mile Island incident itself, the condition of a stuck-open or simmering relief valve in the pressurizer was one of the equipment fault conditions which was a part of the training of the simulator training of the operators who were being licensed for the operation of those units.

And my answer to your question is yes, I think they should have been able to recognize and acknowledge and correct the situation.

And, in fact, the other incidents where this has occurred, that those operators did respond, did recognize the situation in a much shorter time-frame than what was seen.

Mr. Weaver. Well, Mr. McMillan, you've described the steps you took after the Davis-Besse incident in '77, in which you had a simmering valve. And you described that you changed the components and went with other manufacturers than Crosby and Dresser.

However, what happened in Three Mile Island was not exactly that; it was a new wrinkle, so to speak, in the problems with this relief valve.

So I have to ask myself in trying to evaluate all of this, do we have to have an accident as serious as Three Mile Island in order to really go into all the problems of each piece of equipment?

I mean, that's an enormously expensive learning process.

Can we, in effect, really, when we know we've got a problem with the relief valve like this, go in and resolve it before we have to face a TMI incident?

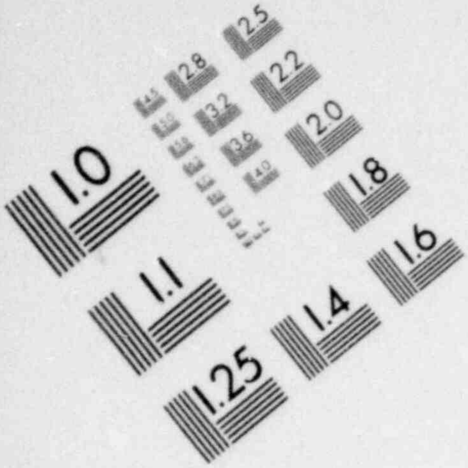
Mr. McMillan. Mr. Chairman, the events that I described as our response to Davis-Besse, we felt were fully responsible for correcting that problem and assuring the probability of its occurring again was minimal.

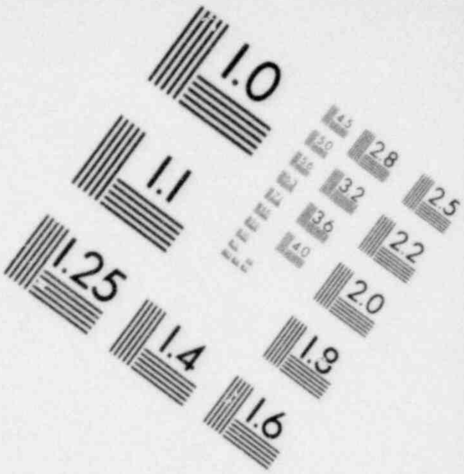
We felt the same thing at the time that we had the earlier problem with the dresser valve at Oconee. The actions that we took, the investigation we conducted, and the modifications in both the equipment and procedures were responsive to the evidence that we had at that time about the reliability of that valve. And that we had taken the appropriate action not only at Oconee, but at the other units which incorporated that valve.

We don't know at this point what happened to the valve at Three Mile Island. Anything that we said here to explain what happened at Three Mile Island would be sheer speculation. And until we have a chance to get in there and look at that equipment and try to establish what it was, in fact, that occurred to this plarticular piece of equipment at Three Mile Island, I have no way of responding.

Mr. Weaver. Can I ask one final question and let you get down with this testimony?

Did you send an instructor specifically to say when you





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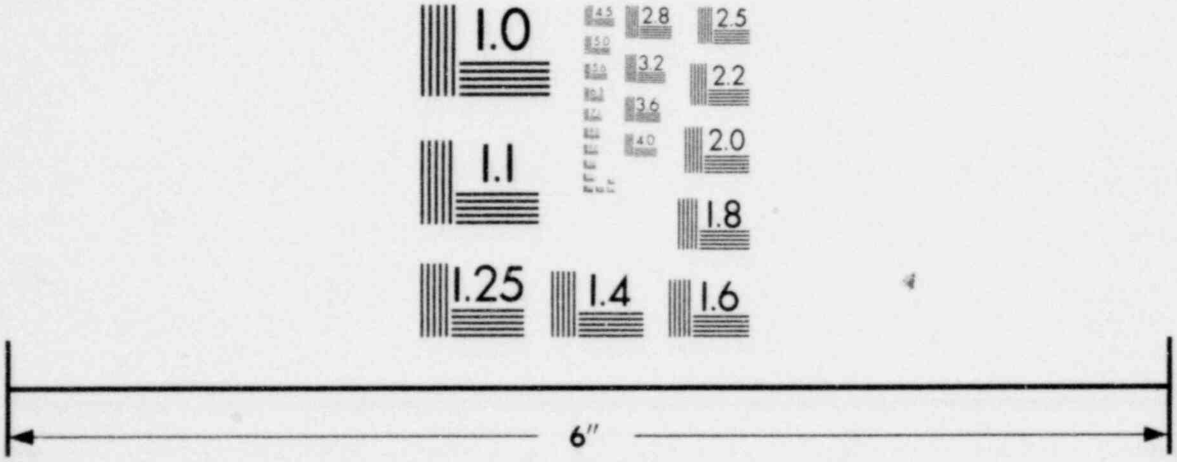

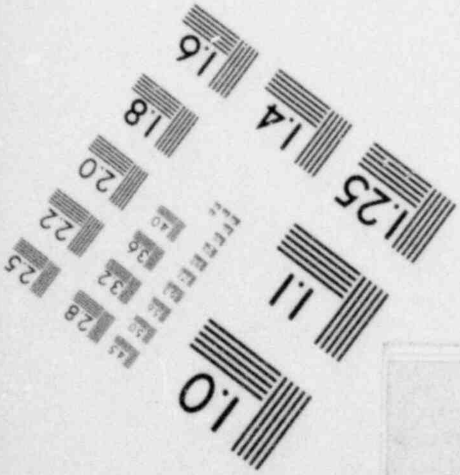

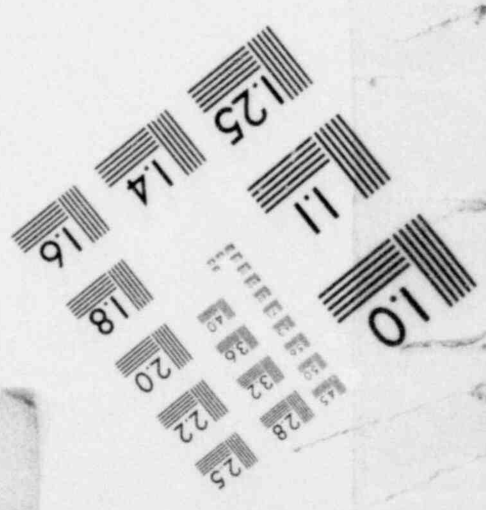

have trouble, you're in doubt, close that valve?

Mr. McMillan. I don't know that we said that.

Mr. Weaver. You have a B&W man on site.

Mr. McMillan. We did have a man at that time, yes. We still do, as a matter of fact.

Mr. Marriott. Just one more question, if I may. Who's responsible for the training of operators, plant operators? Do they have to be certified? What does the NRC have to say? What does the plant itself say? What does your company have to say about plant operators and their training and certification?

What's the scenario?

Mr. McMillan. Let me just take a few minutes to address that question.

Clearly, the selection of the operator candidaes is the responsibility of the operating utility, the licensee. The operating utility also is responsible for developing the training program for these operators. But they have guidelines that they follow that are industry guidelines and which are approved by the Nuclear Regulatory Commission.

The training of operators is different from one utility to the next, as to who provides that training and who's responsible for the various segments of it.

But generally, in the starting of a new unit, there's first of all, sort of academic training. What is radiation?

How does the reactor operate? What are the principles of reactor operation?

There's some math and physics associated with that and that's classroom training.

Secondly, there's a segment of training which involves familiarization with the specific equipment that this operator is going to be called on to operate in detail. All of the sysems, what do they do, what's the design basis, what are the various components, how do they function together?

Following that, there's an operations segment of the training which involves two phases. One is going to an operating unit and observing what happens and how is an operating unit managed? What are the operators called upon to do in routine operation, start-up, shutdown, and so forth?

Then also, a segment of that is simulator training, where they go to a simulator and they spend time going through a whole sequence of normal and abnormal or fault conditions and responding to those and getting the training in how to handle off-normal conditions.

Mr. Marriott. Do they just do that once?

Mr. McMillan. They do that in the initial phases of the training and then they go to the site and they spend about a year at the site. And during that phase, typically, what the operators do is they are converting the guidelines for operation into specific operating procedures. They're helping to run the tests on the equipment, get familiar with the equipment, where is it, what does it look like, how do you run it -- physically, what are its characteristics? And that's about a year's program.

Usually, after that they come back and take a quick refresher course on the simulator before they take their NRC license exam.

Now after the simulator training, we conduct -- I guess you'd call it a mock exam or a pre-test, in which we expose the operators to a test which is of the same nature that they're going to get when they go up for their license examination.

And having concluded the simulator training and all the prior training, we indicate to the licensee, the utility, operators A through Z have been through the training. We've certified they've completed that training and have demonstrated the ability to pass the trial examination.

Mr. Marriott. At this point, we're about two years?

Mr. McMillan. Well, yes, about a year and a half, roughly. Then the NRC comes in and it conducts its formal examination, where it tests the operator and his knowledge of the systems and the equipment and where the equipment is and how it operates and what its function is, and whether or not he knows the operating procedures, both normal and emergency.

They, then, say, he either passes or he fails that exam.

If he passes, they license him and give him either a reactor operator license or a senior reactor operator license, depending upon what his application is for.

And at that point, the operator is licensed to go ahead and conduct the control room functions that are required of the operators.

So the utility selects the operators. The utility in conjunction with others defines the training program. Various portions of that training program are conducted by various organizations. We do a lot of simulator training in support of those programs.

And then the NRC examines and certifies and licenses the operators as competent to perform.

Mr. Marriott. Do they take any refresher courses after they receive their operators or senior operators license?

Mr. McMillan. Yes. After, I believe it's two years, they're required to take a refresher course and come back and go through an updating or upgrading and refresher of some of the simulator transients that reflect emergency, as well as normal conditions.

Some utilities elect to do that more frequently. I believe the regulation is it has to be done at no less than a two-year interval.

Mr. Marriott. Just one other question. The people who are on duty, the pilots on duty in this plant at the time of the accident, do you know what their status was? Were they senior operators, operators -- how many years have they been on duty?

Do you have a breakdown on that?

Mr. McMillan. I don't have that in detail. They had both reactor operators and senior operators. But I think that information really ought to be supplied by the utility.

I just don't know.

Mr. Edwards. Bob, could I follow up on his for just one second?

I just wanted to ask what is the failure rate on these exams after you train these people? Do very many people get washed out?

Mr. McMillan. I don't know what our experience is overall. Typically, out of a class of 20, you might have one or two that fail. Maybe 10 percent.

Mr. Deddens. I would say about 10 percent.

Mr. Marriott. How many of them pass high enough to be senior operators?

Mr. McMillan. I'm sorry. I didn't understand your question.

Mr. Marriott. How many pass high enough to be classified as senior operators?

Mr. McMillan. Usually they apply either for a senior operator or a reactor operator license. It's not a case of the ones qualifying the highest get to be seniors and the ones who don't get to be juniors.

They take a course with the objective at the end of it of either qualifying as a reactor operator, or a course aimed at qualifying as a senior reactor operator.

And the exam, then, is based on whether or not they've applied for senior or reactor card.

On the training part, one of the operators made the comment that the experience of training for the exam, or training for the license, was really training for an exam, not training to operate a nuclear power plant.

Do you have any comment about that?

Mr. McMillan. No, I have no comment on that.

Mr. Carr. Why would you in the thermocouple in the tailpipe, why would you put a maximum indication of 280 degrees when the temperature in the pressurizer is so much higher? Why wouldn't you gauge for the entire range of temperatures?

Mr. McMillan. As I indicated earlier, the primary purpose of that thermocouple is to allow the operator to determine whether the valve is simmering, is weeping.

And what you're looking for there is the difference between the normal temperature of the piping in the building of, say, 100 to 120 degrees and some level significantly higher than that which would indicate that there's some steam flowing through that line.

And you're really looking for the difference between 120 degrees and, say, 200, 250 degrees.

If you see that kind of a difference, then you know that you're getting some seepage through that line, whether it's a full flow or just a weeping.

And that's the principal reason why we do that.

Mr. Carr. What would be the cost differential on the thermocouple?

Mr. McMillan. No difference.

Mr. Carr. So very little or a negligible cost could provide the operator with an A and B case.

Mr. McMillan. Cost is not a factor in that determination. We could and, obviously, in retrospect, will modify that so that that thermocouple reads it for range.

As I say, the reason that it was put in with the limit initially was because the primary purpose of that is to demonstrate the simmering or weeping types.

Mr. Edwards. I wonder if I can get you to comment on something you said that you'd rather not comment on.

I'm also intrigued by the earlier testimony by the operator who said that his training was more a matter of cramming for an exam rather than creatively dealing with situations, and that his training had been inadequate.

It seems to me that that's a fairly major situation here. In the whole Three Mile Island incident, the quality of the training that these people had is a pretty significant factor.

And I really would like to hear what your position is as to whether or not they're trained adequately to deal with these kinds of situations that come up.

How would you answer it when somebody says, you know, we were just taught to cram for the exam, so we'll have a high rate of passing, I suppose.

What is your comment on that?

Mr. Weaver. Let me respond to that this way. I don't want to speculate on what the operator said or why he said it, or whether there was any validity to it.

I think the important thing is that the training that we conduct is aimed at providing the operator with the skills and the knowledge to responsibly manage the operation of a nuclear plant. The test or the examination that is administered at the end of that program has the purpose of measuring whether or not that operator has assimilated a sufficient level of knowledge in the training program to qualify him as a licensed operator.

But our program is not aimed at getting him a license. Our program is aimed at providing the kind of background and perspective and knowledge that's required to responsibly operate a nuclear power plant.

Mr. Edwards. But what kind of an NRC exam is given? Is it primarily a memory exam, or is it operating with simulated crises? What is — you know, and I ask that just out of lack of knowledge -- what kind of an exam does NRC give?

Mr. McMillan. Well -- could you answer that, Jim? Jim Deddens used to be in charge of our service organization and was involved in some of the early structuring and planning for the training of operators.

He's really more competent to answer that.

Mr. Deddens. The exam, as John McMillan indicated, is administered on two levels, depending upon whether the candidate is a candidate for senior reactor operator or reactor operator.

In either case, they are given a one-day written examination. And that's a comprehensive exam that covers all aspects of the plant as it's designed and is intended to be operated. And it also tests their basic knowledge of the principles of reactor operation.

In the case of a senior reactor operator candidate, he is given an additional one-day written examination.

So the reactor operator, it's one day. For the senior reactor operator, it's two days.

That additional examination for the senior reactor operator concentrates on knowledge that he would be expected to have as a supervisor. The senior reactor operator generally is a supervisory individual, as compared to a reactor operator who manipulates the controls.

The additional part of that examination is an oral examination conducted in the simulator in which an NRC examiner puts each operating candidate through a series of operations, and that's at the discretion of the NRC examiner.

He may ask him to take the plant on a start-up and a shut-down. He may want to see the operator's response to, say, a loss of feedwater transient.

That's at the discretion of the NRC operator, and that's an oral exam conducted in the simulator in Lynchburg.

Mr. Cheney. I wonder if, following up on that again, would the people, then, who are in charge at Three Mile Island on the day the operator and senior operator, on the day of the accident of the transient went through your training program --

Mr. McMillan. That's correct. We provided the simulator training.

Mr. Cheney. Can you tell us whether or not during the course of that training they were given any guidance as to what emergency procedures to follow when you've got the combination of the loss of feedwater, low auxiliary feedwater and a stuck open PORV valve?

Mr. McMillan. I think I can answer that. Normally, they are fully drilled in the emergency procedures. The emergency procedures are available for loss of feedwater and other emergency procedures are available for loss of reactor coolant.

And another one would be available for the various other elements.

I don't believe in the training program that was conducted at that time that they had the integrated effect of all of those faults being thrown together at the same time.

Mr. Cheney. What would be the reason?

Mr. McMillan. Would you concur in that?

Mr. Deddens. I would concur.

Mr. Cheney. Could you verify that, or possibly go back and see what the training program was at that point that those particular operators went through?

Mr. McMillan. I feel very confident that we did not give them the integrated sequence of events.

Mr. Cheney. What's the reason?

Mr. McMillan. We can check our records and establish that they were individually trained in the individual philosophy, pressure as relief valves simmering --

Mr. Cheney. Would it be fair for us to conclude then that the operators had not been trained to deal with this type of transient?

Mr. McMillan. I think it's fair to conclude that the combination of the loss of feedwater, the delayed operation of the auxiliary feedwater, and the pilot-operated relief valve sticking open, that combination of sequences was not a part of their training. I think that's a fair assessment before the incident, it has been — that training has been made available for all the operators of B&W subsequent to the incident.

Mr. Chaney. But you would agree that this wouldn't have happened before the incident.

Mr. Weaver. In terms of the operators at Three Mile Island, yes.

Mr. Reis has a question concerning the temperature reading on the thermocouple.

Mr. Reis. We're going back again to the outlet temperature the discharge temperature behind the pressure relief valve.

Now according to the NRC's report, the implications of this, there are a number of parameters to evaluate whether or not that pilot-operated relief valve was open.

One is the position of the valve, which is not measured anywhere. It's only the command that is measured, quench tank level and pressure, which is behind the control panel way out of the way.

The only other way is the discharge pipe temperature detection system.

Now, they three times requested the temperature of that. They got 285 degrees, plus or minus a degree or two all three times. The operators have indicated, a number of other people who have talked to us have indicated that that doesn't help them any because he knew the thing was closed tight, 100

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percent shut. The temperature would be approximately that temperature anyway just because of the latent heat, since the thermocouple resides right on the metal pipe.

So that pressure or temperature detection is essentially useless in their minds because it doesn't tell them whether or not it's simmering, whether it's opened and closed, or whether it's full fledged open.

So why, given the fact that there is no position indication, would you not have previously have either raised the thermocouple to read above 285 degrees, taken it off the pipe system so that it would not continue to read high, no matter whether the thing had been closed, or take some other action so that that parameter was actually usable.

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Mr. McMillan. Let me say again that the primary purpose of that thermocouple was to determine whether or not we were getting a simmering or weeping of those valves.

Mr. Reis. But according to their testimony, even if you had a simmering and then it had fallen shut, this thing would not help you because the latent heat from the metal on which it resides would still be causing it to be elevated. So they couldn't have told if it had opened and simmered and then shut.

Mr. McMillan. Well, I haven't made the calculations nor seen the calculations of the rate at which that temperature decays in the post-discharge condition. I think that's something that we ought to check and get back to you on.

Mr. Weaver. Do you have anyone here who could respond to that?

Mr. McMillan. Do you know of any calculations we've done? You blow the relief valve and discharge goes up. How fast does it decay?

Mr. Deddens. I don't have any knowledge of any specific calculations that we have made, but it would be a fairly easy calculation to make.

Mr. McMillan. Mr. Chairman, if you like, we can get that information for you and submit it to the committee.

Mr. Weaver. I would appreciate that very much.

(Information be furnished.)

Mr. McMillan. Before we leave that issue, I would like to again indicate that this is not a situation where we're relying on a single source of information; that is, the temperature on the tailpipe. That's one of the pieces of evidence that's available. During the transient the reactor coolant system pressure was continuing to go down, down, down, down. The reactor building pressure was going up. He had an alarm which he had to acknowledge on the pressure and level in the quench tanks. So there are other sources of information which were available to him, and he was not depending specifically on this one piece of data to help him recognize the condition.

Mr. Weaver. I understand that. They have answers to every one of those. But I agree with you that that's a fundamental question here, and it's puzzling.

Mr. Cheney.

Mr. Cheney. You cited earlier in connection with the same thing the increase in pressures in the reactor building, the containment building, as being one indicator that might have told the operator that he had a problem with the PORV.

Mr. McMillan. He had a problem in which he was getting an abnormal increase in reactor building pressure, so he's dumping energy into the reactor building from some source.

Mr. Cheney. But doesn't that same reactor building pressure, that same parameter, serve as the trigger for the containment isolation of the systems at TMI?

Mr. McMillan. Yes, sir. When the pressure gets to four psi, the reactor building isolation system comes into operation.

Mr. Cheney. And it did not go into operation, then?

Mr. McMillan. It did go into operation when the pressure reached four psi, which was several hours into the incident. I can check that in my sequence. But even before the isolation, four psi, there was a gradual buildup of reactor building pressure.

Mr. Cheney. I guess the thing I am suggesting is that if the parameter did not increase sufficiently to trigger containment isolation, is it realistic to expect that the increase was sufficient to trigger the operator to say, "My PORV is open"?

Mr. McMillan. I don't know how to answer that. It's abnormal to have that kind of pressure level in the reactor building, and, certainly, it would be a cause for the operator to wonder why is the reactor building pressure going up.

Mr. Cheney. Again, I come back to the basic point: Is it fair for us to look to the operator, who, first of all, hasn't even been trained to deal with this combination of circumstances. He hasn't even anticipated it will occur. To assume that the increase of containment pressure which is not sufficient to isolate the containment is nonetheless supposedly sufficient to alert him to the fact that the PORV is stuck open.

Mr. McMillan. I think the combination of information that was available to him should have been an adequate basis for him to determine at a much earlier time than he did the fact that that pilot-operated relief valve was open.

Mr. Cheney. Maybe we can go on to another point here.

Mr. Weaver. We really should let him go through --

Mr. Reis. Would that be sufficient to tell an engineer what was going on or sufficient to tell an operator what was going on? That's a very important difference.

Mr. McMillan. Yes. Both.

Mr. Weaver. Do you have a related question?

Mr. Myers. Could these phenomena that were being observed be due to something other than the stuck-open pressure relief valve? I mean, in their minds. Presumably, it was, in their minds.

Mr. McMillan. Reactor building pressure increase could come from other sources. It could have a leak in a secondary side of the steam generator and be dumping steam into that reactor building from the secondary side of the generator. And that could cause pressure to go up.

Again, you can't rely on any single piece of information. You have got to look at the available data and put that together to determine where the source of the problem is.

I think the significant item here was that the primary system pressure, the reactor coolant pressure, was down, down, down, down, down, indicating abnormal conditions in the reactor coolant system, and that in combination with the other data that he had should direct his attention at the reactor coolant system. The quench tank pressure and volume comes from the safety valves and the reactor pressurizer. That's the primary source of that information.

Mr. Myers. That information was not readily available to him.

Mr. McMillan. Except that it was alarmed. He had to go around the panel and acknowledge that alarm.

Mr. Cheney. But we've heard testimony -- if I may, on that point -- that there were as many as a hundred alarms going off at that time, early in the morning, that there were numerous alarms coming into the system.

Mr. McMillan. I am sure there were numerous alarms in this sequence.

Mr. Cheny. This wouldn't necessarily stand out more than any other in the computer printout that produced written information on paper that you could look at and identify the problem was running several hours behind because of the backlog being stored up because of the number of alarms that were coming in.

Mr. Carr. The tractor was chewing it up, too.

Mr. Reis. An additional point is that they indicated -- they noticed the alarm and turned and looked at the command, the solenoid indicator, and that it in fact operated, and they were then presented with a problem. We had a furious signal that they had to figure out which one was spurious. In the meantime there were worried, of course, about other things.

But I think the point that the operator has made was well made: They had spurious signals, and this was not the only instance, and they had to make a rapid decision as to which one was spurious and they didn't have enough information. So it's not as if they did not see that alarm and react to it.

Mr. McMillan. I think the point is well made, and, certainly, in retrospect, as we have already initiated action to accomplish, it would be better to have a clear indication of whether that valve was opened or closed, whether or not there's only a power supply to the solenoid.

Mr. Myers. I think at the time also, that they thought that possibly one of the safety valves had opened. If that had happened, then the safety valve was stuck open and not necessarily the same kind of cases that they were getting. I think that may --

Mr. McMillan. You have information I don't have.

Mr. Weaver. By the way, that brings a point up: What if the safety valve had opened and you said it doesn't receive, then we can't even correct the situation, can't close the safety valve. Or is there another valve downstream?

Mr. McMillan. You cannot close the safety valve. If the safey valves opens and stays open, you then depend upon your safety system, the high-pressure injection system, the core flood tanks in the longer term.

Mr. Weaver. So we're in the same situation exactly as occurred. In other words, they don't have -- they can't close the valve and get the system back.

Mr. McMillan. They're blowing reactor fluid out of the reactor vessel. You've got a loss-of-coolant accident associated with that safety valve, and the safety systems, the high-pressure, low-pressure, and core-flood systems, are designed to accommodate and take care of that condition.

Mr. Weaver. But, I mean, my question is: Aren't we faced with the situation, suppose they had closed the power-operated relief valve and done it on time, but there had been a jump to 2500 psi, the safety valves had opened; wouldn't we have had the same sequence of events then?

Mr. McMillan. Let me say, if the valve closes, as designed, and pressure goes back up, the pilot-operated relief valve will open again. If he closed the block valve and for some reason the reactor coolant system pressure went up and then opened the safety valve and the safety valve stuck open, we then are in a situation where you are blowing the steam out of the safety valve, and you again are dependent upon the reactor protective systems and the high-pressure injection and low-pressure and core-flood systems to maintain core cooling.

Mr. Weaver. But, of course, they did some things they probably wouldn't have done had they known the valve was open, whereas if they had known the safety valve was open, they wouldn't have done those things, possibly. Anyway, I realize that's speculation.

Mr. DeCamp, the president of GP, told me when we were at TMI, when I asked him that question when we were beginning this investigation, he said, "Well, with everything going on in that control room" -- and this is comparable to what Mr. Cheney said -- "with everything going on in the control room we couldn't expect the operator to be cognizant and analyze everything." That was the quote, that he told me informally. What's your reaction to that?

Mr. McMillan. Mr. DeCamp and I don't agree on that.

Mr. Weaver. Won't you proceed, please? We've interrupted you for some time. Sorry.

(Slide.)

Mr. McMillan. I think we have pretty well addressed the second significant item.

Mr. Edwards. Mr. McMillan, how far away is the thermocouple from where the discharge would be?

Mr. McMillan. I don't have any idea.

Mr. Edwards. It is a significant distance?

Mr. McMillan. I don't know where it is. Let's make a note. I will get you an answer for that.

(Information to be furnished.)

Mr. McMillan. A third significant item was the securing or the shutoff of the high-pressure injection flow prematurely. As you recall, I indicated that the high-pressure injection came on at approximately 1600 pounds, as designed. And some of the pumps were shut down in subsequent minutes of the transient. And our calculations indicate that had those pumps been left on and allowed to perform the high-pressure injection function, that there would not have been core damage, there would not have been release of radioactivity at the level that was observed at the Three Mile Island incident.

Mr. Cheney. On this particular point, the thing that we've heard as we went through this inquiry, repeatedly, is that the pressurizer level is the key thing that the operator looked at in terms of adjusting the flow of the high-pressure injection pumps. Isn't that what he's trying to do?

Mr. McMillan. Could I just defer the answer to that until we come to item No. 5?

Mr. Cheney. I do want to get back to this.

Mr. McMillan. Let me address that at that time.

The fourth significant factor is related to containment isolation. We've talked a little bit about that here. The containment isolation design at Three Mile Island was one in which the isolation occurs at a four-pounds-per-square-inch reactor building pressure. In other designs there are other events which trigger containment isolation. The significance pressurizer indication. And we've done a fair amount of investigation and satisfied ourselves that the level indication as displayed on the control panel was indicating essentially the amount of water that was in the pressurizer, that there was not an erroneous instrumentation reading, and that in fact he had an indication of how much water was in that pressurizer.

The point here is that he interpreted that, apparently -- we're doing a lot of reading between the lines here -- but apparently he interpreted that indication as meaning he had a full reactor system when in fact what he had was a system where he had the temperature of the water in the hot leg of the reactor coolant system at the saturation pressure, and flashing and creating steam in the reactor coolant system, even though he had indications of full pressurizer all during the time that he was continuing to blow steam out of the pilot-operated relief valves.

But I think the significant factor here is the inappropriate emphasis that was placed on pressurizer level alone at a time when he had other indications available to him that should have triggered his concern about what conditions existed in the reactor coolant system.

Now, your question.

Mr. Cheney. Is this the first time, to your knowledge, that you ever had this kind of thing happen, where you had the pressurizer level not reflect the coolant level in the reactor?

Mr. McMillan. No. We've had some other instances where this has been a concern, where we've had instrumentation or electrical failures, electrical problems.

Mr. Cheney. I guess the question is -- coming back again to the basic training program -- would the operators that were in charge of Three Mile Island on the day of the accident and then trained in your facility -- I guess at Lynchburg -- trained to deal with specifically this kind of situation where the pressurizer level did not in fact reflect what was happening inside the reactor?

Mr. McMillan. I think that again the training program is aimed at preparing the operator to manage the plant and availing himself of all the information that's available to him.

Mr. Cheney. Could we find out again by going back to the tests perhaps in the basic program in terms of emergency procedures whether or not he was trained in reading that pressurizer level as indicating something other than the reactor coolant pump?

Mr. McMillan. Let us see if we can find that.

Mr. Cheney. It seems to me that's a key point. Has this ever been questioned before, whether or not the pressurizer level was an accurate reflection of the reactor coolant level, in a general sense, in terms of the basic design of the reactor?

Mr. McMillan. I think a lot of people have asked that question: What does the pressurizer level indicate to you? Clearly, it indicates to you that if you have sub-cooled conditions in the reactor system, that you have a solid system and one that you can depend upon, the level in the pressurizer as indicating the level of the coolant in the reactor system. If you have a situation where you have saturated conditions in the reactor coolant system, then the pressurizer level is not necessarily an indication -- it is not an indication alone of the condition you have in the system, and that is the situation prevailing here; again, the pressure being a very significant factor in evaluating the condition of the fluid in the reactor coolant system.

Mr. Cheney. I would just like to come back to that basic question: whether or not those operators at TMI on that day were trained to read that pressurizer level as anything other than indicating reactor level, coolant level in the reactor?

Mr. McMillan. Let us trace that through and see if we can find what evidence --

Mr. Cheney. Do you think that's some flaw in the design of the reactor?

Mr. McMillan. No, sir.

Mr. Cheney. Why not have a coolant level indicator from within the reactor itself?

Mr. McMillan. That's been suggested by a number of people, post-TMI.

Mr. Weaver. Mr. McMillan, how do you measure the level of water in the reactor vessel?

Mr. McMillan. You do not measure the level of water in the reactor vessel. There is no mechanism in this design or any of the other reactors that I am familiar with for measuring the actual water level in the reactor vessel. What you're depending upon there is measurement of the water level in the pressurizer in conjunction with sub-cooled conditions in the reactor coolant system.

Mr. Weaver. But isn't that an important thing to know? I mean, why don't you have something that tells us the level of the water?

Mr. McMillan. Well, I think --

Mr. Cheney. I guess I would like to follow up on this.

Mr. McMillan. I am sorry. I wanted to respond to the chairman's question.

The equipment is designed such that in the combination of the level and the pressure indication that the operator has the capability of determining whether the pressurizer level indication he has is a valid indication of the level in the reactor vessel. Again, in retrospect, in looking back at the accident, the Three Mile Island accident, and one of the first things we did in response to the accident, was to bring to the attention of the operators the importance of determining when they had sub-cooled conditions in the reactor coolant system and comparing the temperature in the hot leg of the reactor coolant system with the saturation temperature corresponding to the pressure coolant system, when you are sub-cooled, the pressurizer level is a valid indicator. When you are not sub-cooled, then you do not have a valid indication.

Mr. Weaver. Let Henry follow up. I think you had a follow-up.

Mr. Myers. Do you believe the operators were trained to recognize that pressure was in fact indicating that the system was sub-cooled or not sub-cooled?

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Mr. McMillan. The operators were trained in the loss of coolant accident.

Mr. Meyers. Did they have steam tables to indicate that the system was indeed subcooled?

Mr. McMillan. That's a different question. Let me answer the first one. The operators were trained in how to handle a loss of coolant accident, and in that loss of coolant accident condition you do get into a situation where you've got saturated temperature conditions in the reactor coolant system. So they had been trained on how to address that kind of a situation.

Now, the second question you asked, I believe, is, if I interpret it properly, did the operator have at his disposal, displayed to, either right in front of him or in a steam table someplace, a method of comparing what his hot temperature was relative to saturation temperature. The answer to that question to the best of my knowledge is no, he did not.

Mr. Meyers. Also, at this time he did not realize that he had a loss of coolant accident.

Mr. McMillan. I think that's correct, yes.

Mr. Meyers. Did he have a reason, then, to think that the system was or was not subcooled?

Mr. McMillan. He had a reactor coolant pressure which was way down, around 1200 or 1100 pounds, and that's an abnormal condition that should have triggered his attention immediately to the probability that he had a saturated reactor coolant system.

Mr. Cheney. I guess I wonder: Hasn't it been brought to your attention before that there was potentially exactly this kind of a problem with your design? I'm referring specifically to a TVA study that was done, with reference made to the letter of April 1978 to Babcock & Wilcox, Lynchburg, Virginia, Attention Mr. James McFarland. It talks about emergency core cooling systems, small break LOCA analysis, the possibility or concern that the pressurizer level is not a correct indicator of water level for the reactor core because of the loop seal in the pressurizer.

It may be possible to have a full pressurizer while the core is partially uncovered, and this could lead to direct operator action.

Mr. McMillan. Your question was, was I aware of that. I was not made aware of the Michelson report personally until about a month ago.

Mr. Cheney. Who at Babcock & Wilcox would have been aware of the Michelson report?

Mr. McMillan. The Michelson report was transmitted to us, I am told by my associates, in a letter from TVA along about April 1978, I believe.

Mr. Cheney. It was dated April 27th, '78.

Mr. McMillan. That was responding to a discussion that we had on the telephone with TVA in December of 1978. They asked us for some additional information, which we documented in a letter to them of January 1979. They responded to that letter, indicating that they had a couple of clarifications that they were interested in our responding to, and that that would essentially resolve the concerns we had. In terms --

Mr. Cheney. Did you ever respond to those concerns?

Mr. McMillan. Well, we have, in the period -- let me go back and answer that, taking it one step back. When this whole concern about the small break at the TVA in the TVA design was brought to our attention in early 1978, April '78, we looked at that problem and we felt that the condition that was raised there in terms of the small break and the ability to cool to address the concerns that Mr. Michelson raised had been bounded by calculations that we had made.

We did not respond immediately to that because we were not at that stage of the licensing phase of the TVA application. When we did get back to Mr. Michelson and TVA in December, we felt that we had substantially satisfied their concerns, again expressing our judgment that we had bounded those calculations.

More recently, since the Three Mile Island accident, we have in fact gone back and done the specific calculations at the very small break size, the .01 square feet which is the area of concern to Mr. Michelson as expressed. We have documented those calculations in submittals to the NRC. The

NRC has reviewed those calculations and have reviewed their assessment of those calculations with the Advisory Committee on Reactor Safeguards, I believe it was last week, and have expressed their satisfaction that the calculations that we had done resolved the concerns that Mr. Michelson had in his report on the TVA, in fact confirming the judgment that we had expressed at the time that the issue was initially brought up in 1978.

Mr. Cheney. Are you saying, then, that Michelson was wrong?

Mr. McMillan. No. Michelson -- if you read Michelson's report, Michelson raises some questions. He doesn't state that this will happen or that will happen. He just says, I'm concerned about certain aspects of the small break analysis and I think additional analysis ought to be made. And we tried to persuade him, by the engineering judgments and the calculations that we had made, that we had in fact covered that situation. And our subsequent calculations, when we went back and actually did the detailed calculations on a complex computer code to simulate that condition, confirmed the judgment that we had expressed at the outset.

Mr. Cheney. Do you think that judgment today is correct?

Mr. McMillan. With respect to the small break analysis and the ability to maintain core cooling in a small break condition, our judgment was correct, yes.

Mr. Cheney. Go ahead, Henry.

Mr. Myers. Well, Michelson was saying the full pressurizer may convince the operators to trip the high-pressure injection pump, accounting for a subsequent loss of level. Although this response appears desirable, the full pressurizer may not always be a good indication of high water level in the reactor coolant system.

Wasn't Michelson saying, okay, if you leave the high-pressure injection system on, that will take care of the small break. But here he was saying in his paper that the full pressurizer might convince the operator to trip the high-pressure injection pump, and that is what the operator seems to have done. And if perhaps the operator had been told, warned of this condition, he would not have tripped the pump.

Mr. McMillan. I think that was a valid concern as expressed by Mr. Michelson, and certainly one of the things, you'll recall, that has been done in the immediate period following TMI was to issue a bulletin in conjunction with the Nuclear Regulatory Commission instructing the operator when his high-pressure injection system comes on, leave it on until you have satisfied one of two conditions, either that you have subcooled conditions in the reactor coolant system, 50 degrees below T, or you have reached the point where you have your low-pressure injection system on, and you stay there at that point of operation.

Mr. Myers. But no instruction was ever issued to the B&W utilities presenting them this conclusion of Michelson, the question that the full pressurizer might not be indicative of a full pressure vessel, therefore if there was any question they should leave on the high-pressure injection.

Mr. McMillan. Not to my knowledge.

Mr. Vento. One of the problems with that, with the operators, appeared to be -- I think I'm at the right point, where they began having problems with the pumps, the vibration. They were concerned about the equipment in terms of building a solid system here. That's when they proceeded to turn them off, did they not?

I mean, even if they had been instructed to leave them on, isn't there a contradiction there in terms of the problem?

Mr. McMillan. You're talking about different pumps. If they had left the high-pressure injection pumps on, they would have maintained a subcooled condition in the reactor coolant system. The concerns that they subsequently had about the reactor coolant pump would not have come up.

Mr. Cheney. I don't mean to interrupt you, Bruce, but I'd just like to focus in for a second more on this question of how the process works, so that somebody like TVA, for example, sits down and does a fairly complicated analysis which I would say is prophetic in terms of indicating exactly ultimately what did happen at TMI.

What's the normal procedure, then, for Babcock & Wilcox and the NRC for looking at that and saying, yeah, that guy has highlighted a problem; we need to fix it. How would that ordinarily be responded to?

Mr. McMillan. Let me take specifically the TVA letter and indicate how that kind of thing would be addressed. This concern was raised by TVA in the course of the design phase of the project.

Mr. Cheney. How does TVA get into this? Do they buy the reactor?

Mr. McMillan. We have a contract with TVA to supply two nuclear steam systems at their Bellefonte unit down in Alabama. And we were in the process of designing this equipment, and in reviewing the design and preparing licensing material for that unit as it approached the operating license phase, this analysis was done by Mr. Michelson, apparently. And he, as I said, raised some concerns there.

He said, I don't know, I've done what he called fairly simple calculations, and compared to the computer type of calculations they are fairly simplified, and on the basis of these simple calculations I have some concerns.

In the normal course of a design program of this sort, we would respond to those concerns in what would be considered a timely fashion for the schedule of that particular contract. That's why there was some lag between the April 1978 letter and the December 1978 response, because we were not in a phase of the TVA contract at that time which would cause us to have to address specifically that answer.

Having responded as we did, we felt that the concerns that he raised with respect to the small break analysis had been bounded by the analysis that we had done.

Mr. Cheney. What does that mean, "bounded by"?

Mr. McMillan. You don't do a calculation on every conceivable break size and every conceivable sequence of events. What you try to do is to look at a number of worst cases, and if you look at those worst cases and you can handle the consequences of those cases, then the lesser cases which you have not looked at, you can satisfy yourself that you can handle those as well. So --

Mr. Cheney. It seems like a logical procedure.

Mr. McMillan. You would draw an envelope around all the cases and you look at the ones at the boundary, and you say, as long as we can handle those boundary ones we ought to be able to handle the other ones. And this is what we were saying to Michelson, and this is what's been confirmed in the subsequent --

Mr. Cheney. Again, I don't quite understand. I'm a layman, so I need a little help with the technical aspects of it.

He makes a suggestion that a particular thing, under a certain set of circumstances, could get you into difficulty, and he's fairly precise about what those are.

Mr. McMillan. You're addressing specifically, now, the pressurizer level?

Mr. Cheney. The pressurizer level that he especially was concerned about: "A full pressurizer may convince the operator to trip the HPI pump and watch for subsequent loss of level. Although this response appears desirable, the full pressurizer may not always be a good indication of high water level in the reactor coolant system."

Very precise from my standpoint. He's talking about a small break LOCA analysis. Then you receive that from him and you sit down and do an analysis that basically says he's wrong. Is that correct?

Mr. McMillan. Mr. Michelson raised about a dozen concerns in that letter, and the major concerns that we were addressing by our analyses were: Do you keep the core covered and therefore cooled, without damaging the core during the sequence of events that he had identified?

With respect specifically to the issue of pressurizer level and his concern there, I think he would say to you today we haven't resolved that concern.

Mr. Cheney. I think that would be a fair conclusion.

Mr. McMillan. In today's environment, I think we would have to say -- and as I indicated, we have already given instructions to the operator that he cannot depend solely on the pressurizer level, and that he should not cut off his or shut down his high-pressure injection pumps until he is satisfied that the condition of demonstrated subcooled condition in the reactor coolant system or has depressurized to the point where he has stable cooling with the low-pressure injection system. And our instructions today incorporate that, those modifications.

Mr. Cheney. Okay. Again, I don't mean to dominate all the time.

Mr. Weaver. Please. Your question is excellent.

Mr. Cheney. I guess I'm coming back to that basic fundamental notion of notification from TVA, the customer, somebody who's going to buy a reactor. He raises a specific problem, and you respond, then, in December. And now you've given a whole different set of directions to operators since the Three Mile Island accident.

Getting back to the basic [illegible]tion of who handled the response to TVA's concerns, and would you agree that that response was not adequate?

Mr. McMillan. As it applies specifically to the concern about the pressurizer level being a misleading -- could be a misleading indicator, I have to say that in retrospect on Three Mile Island, we did not satisfy that condition. Now, we've not carried on the dialogue with TVA to the resolution of their concerns on this issue. This is an ongoing contract relationship with them, leading up to the commercial operation of the Bellefonte unit.

Mr. Cheney. Would the Nuclear Regulatory Commission have been aware of this exchange between TVA and you?

Mr. McMillan. I would say in the normal course of events they probably would not have been, unless it was brought to their attention by the licensee, and I know of no reason why the licensee would do that until he was satisfied that his concerns had been resolved.

Mr. Weaver. Thanks very much. Excellent line of questioning.

Are you opposed, Mr. McMillan, to water level measurement of the reactor vessel?

Mr. McMillan. No, sir. In fact we've said to the ACRS that we were investigating ways in which that can be accomplished.

Mr. Weaver. Why hasn't this been done before? Is there some insurmountable problem or difficult to surmount problem?

Mr. McMillan. It's a difficult problem, yes. But I'm not at this point saying it's insurmountable. I think we've got the technology, which we really can go after. We can figure out a way to do it, to accomplish that.

Mr. Weaver. Doesn't that -- it just strikes me that if we can't measure the water, the most important single thing from the public, whom I represent, the public here, in the sense that the public wants to know if those rods are covered. And I'd want to know whether they were. And so I'd say the first thing I was going to do, if I were going to design a nuclear plant, is to make sure that we knew what the water level was.

Why hasn't this been done in the years we've had development of nuclear plants? Please edify me. I'm puzzled.

Mr. McMillan. I think the sequence of design events here developed along the line that in any -- let's say in most transients, you design the safety system to perform the function of maintaining water in the reactor coolant system at a level which will cool the fuel and, in the most extreme case, keep the temperature of the cladding below the 2300 degrees Fahrenheit that was required by the NRC regulations. And the safety systems are designed, developed, checked in terms of the analytical check, to demonstrate that you can accomplish those objectives.

And for most transients the combination of the pressure in the reactor coolant system and the level of the pressurizer is an adequate indication of the level that you have on water in the reactor coolant system.

Again, in retrospect, it would be very nice to have a reactor vessel water level indication for the very abnormal conditions that we saw develop at Three Mile Island.

Mr. Carr. What is the difficulty? You say it's a difficult task to get that water level. Why? That wouldn't occur to me. Why would that be so difficult?

Mr. McMillan. Well, you just have got to develop a technique by which you can measure the level, bring a line out the top, bring a line out the bottom, put a reference leg in to try to measure the pressure drop. But again, the interpretation of that can be misleading, particularly where you have a mixture of steam and water.

What you're measuring is a difference in the density of the water, and interpreting that requires some sophistication. We're looking at other ways: Can you do it acoustically, by transmission, by radiation transmission methods or something like that.

Mr. Edwards. Let me pursue a little bit on this, because I'm trying to get a grasp here of just what your company is doing about some of this.

As I listen to it -- I'm a layman also, but as I listen to it, you have no water level indicator. There was no indicator of whether the relief valve was open or closed until the NRC told you, as I understand it, to put one in the control valve.

When Michelson pointed out some of these problems, there was at least some reticence to address all of the things he brought up as quickly as possible.

You know, what changes are you initiating? Is anything coming through as you go along, saying to you, we need to initiate some changes in the panel or the way we're doing things?

You know, I see, it seems to me, maybe unfairly, that your company is not really reacting as quickly as it should on your own initiative to make some changes here that seem to be called here.

Mr. McMillan. I'd hoped to get to that in my presentation. I wanted to address the response of the company in the days after the Three Mile Island incident.

Mr. Edwards. But what I'm interested in is before the Three Mile. The Michelson report was before Three Mile Island. The lack of water indicator has been on your control panel before Three Mile Island. The lack of a relief valve indicator was on the panel before Three Mile Island.

You know, what I'm saying is, do we have to wait until after every accident before your company says, well, let's make changes.

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Mr. McMillan. I would just like to make sure that I'm clear with you with respect to a couple of the questions you raised.

First of all, with respect to the indicator of the position of the pilot-operated relief valve, we made the decision in the post-accident situation that it would be more appropriate to have some more positive method of identifying the position of that pilot-operated relief valve.

That's something which we said that we voluntarily undertook. It's not something that we were told to do by the NRC or by anybody else.

With respect to the reactor vessel water level indication, that's not a feature in our design. It's not a feature in any domestic reactor design. And so that's not a case of our having done something or not having done something that was not characteristic of the industry in general.

And as I say, it was felt that the indications that were available were adequate in conjunction with the safety systems to protect the core.

Now we make modifications in our design as it goes through the design phase and construction phase and after units go into operation to reflect lessons learned or experiences developed that are developed not only in the operation of our units, but in the operation of other units.

Mr. Edwards. But you have to understand that where you may understand some things, those of us who are laymen may have a different reaction. It may be because we don't know enough about it.

I must say that when I was at TMI and I learned that there was no indication of whether the relief valve was open or closed, you know, I thought, now maybe I'm not an engineer, but maybe I would have thought that you should have one.

You know, the first analogy that came to my mind was being in an airplane where the pilot lowered the landing gear and he had no light to tell him whether the landing gear was locked in place or not.

It just seemed to me that it did not require a great deal of expertise to know, you know, that this is something that ought to be on the control panel.

Mr. Vento. If the gentleman will yield to me, it seems that Mr. McMillan, from what you've said, that you're telling us that it isn't just a simple matter of putting a gauge on the reactor vessel in order to make the determination as to actually whether or not this was covered by water, had enough cooling capacity on it.

In other words, you're saying that you don't know if this is solvable based upon what we know about this accident and based upon the way that your reactors work.

In other words, you're saying that you're investigating — this is what I heard you say — you're investigating various other methods — radioactive material, acoustical methods, in order to determine the level.

So it isn't just a matter of putting on some known technology to understand whether that reactor was covered or not covered.

You said also that in response to some of the questions by our colleague from Wyoming, that very few reactor designs actually have that in place.

In other words, a reliable system of determining whether or not the reactor vessel, that the core is actually covered with water.

Is that accurate or not? Am I repeating what you said?

Mr. McMillan. That's what I said, yes.

Mr. Vento. So you're saying, in other words, if this were to happen again tomorrow at any reactor, the same problem is theoretically possible based upon the scenario, whether it's design error, whether it's operator failure.

You know, I'm not trying to assign blame in this particular sense, but saying it's entirely possible that this could happen again.

Mr. McMillan. That isn't what I said. What I said was that there are not provisions made in any domestic reactors, to my knowledge, for measuring actual water level in the reactor vessel.

The second thing I said was that it isn't just a simple matter of going in and adding a gauge, that it takes some degree of sophistication to accomplish that and we are looking at various means of accomplishing that, a meaningful water level indication in the reactor vessel.

Those are the comments.

Mr. Vento. Is the conclusion that one might draw from that -- in other words, you're saying that you don't have an available reliable technology right now to put a gauge on that reactor vessel.

Is that right?

Mr. McMillan. I didn't say we didn't have a reliable technology. I said that measuring the water reactor, measuring the reactor vessel water level is not a simple matter of just going in and putting a gauge on. It takes some degree of sophistication to work out a means of getting an accurate measure of the water level.

I'm not saying it's impossible. I'm just saying that we've got to put some effort into finding an appropriate way of doing that.

Mr. Vento. The relief valve that was in place in this, it was pointed out to us that many designs do not have relief valves such as this in them. Babcock & Wilcox, apparently, is one that does.

Mr. McMillan. I don't believe that that's an accurate statement.

Mr. Vento. You don't believe that's accurate?

Mr. McMillan. I think most pressurized water reactors have pilot-operated relief valves.

Mr. Vento. They have safety valves. But in other words, is this true that it has an extra valve on it?

Mr. McMillan. This has a valve which is over and above the valve, the safety valves required by the code. And we're not unique in that respect.

Mr. Vento. You're not unique.

Have you — it was pointed out that these valves in terms of reactor years, I think, it turned out that they open up about 4 times a years.

Have you had any indications, problems with those valves in the past?

Mr. McMillan. You're speaking now of the pilot-operated relief valve?

Mr. Vento. Yes, I am.

Mr. McMillan. We discussed earlier the frequency of the operation of those. The statistics I have indicate that we have had about 150 occasions no which that valve has been called upon to operate. And I believe that we have had four cases in which it is actually stuck open.

Mr. Carr. Mr. McMillan, when we were at Three Mile Island, they had, or you had, or somebody had, placed a new panel which I guess had the temperature readings. Do you know the new panel that I'm talking about?

Mr. Cheney. For the reactor core temperatures?

Mr. Carr. Yes, the reactor core temperature.

Mr. McMillan. I haven't seen that new panel.

Mr. Carr. In a slight variation on a theme here, we're concerned about water level in the rector. But what we're really concerned about is the temperature of those fuel rods.

And so, even going toward a water level on the reactor, you're not really going right directly to the flaps or the gears that Mr. Edwards was talking about.

Now that -- I mean, what youre mainly concerned about is that those things don't get too darn hot, right, and that they have a certain code limitation on the cladding and temperature indication. You might have a full reactor, but it might not be circulating properly.

There may be some kind of a blockage. Maybe a filter system didn't work or something, and you're not getting the kind of cooling that you need.

So really, maybe we ought to be talking about water level gauges.

But it's been demonstrated because, apparently, they got it on the plant right now, that you can somehow or other measure the temperature of those fuel rods.

Mr. McMillan. Let me respond to that in this way. We provide in our reactors as a part of the in-core instrumention a thermocouple which reads the outlet temperature of the fuel assemblies in which the in-core detectors are located.

There are 177 fuel assemblies in Three Mile Island type reactors. And in 52 of those assemblies, we have in-core detectors and in 52 of those detectors, we read the temperature of the water coming out of the top of the fuel assembly.

And it's those thermocouples that have been displayed in a special panel at Three Mile Island in recent days, recent weeks.

Previously, those thermocouples read out through the computer and were printed out as the temperature of the water coming out of the fuel assemblies on those fuel assemblies. That does not measure fuel cladding temperature.

In each of the fuel assemblies, there are some 200 fuel rods. There are 177 fuel assemblies and each fuel assembly has about 200 fuel rods in it. And we don't make any effort to try to measure the temperature of the cladding in each of those.

Mr. Carr. I see. But there wouldn't be really -- would there be any real technical difficulties if there was a requirement, say, within the next year that all plants had to have an individual thermocouple attached to the fuel cladding?

Mr. McMillan. That would be a technical nightmare.

Mr. Carr. Why would that be?

Mr. McMillan. For two reasons. First of all, you've got to decide where to put the thermocouple along the 12-foot length of the core in order to measure the temperature, and that could be different, depending on the power distribution on the length of the fuel rod.

Secondly, you've got to find a means of attaching that thermocouple to the cladding so that you get a meaningful reading.

And thirdly, you've got to find some way to get those tens of thousands of thermocouple leads out of the reactor cooling system to some place where you can read them.

And that would make a very difficult, a very difficult job.

Mr. Carr. What are they reading now? They're reading water temperature at the top of each fuel assembly?

Mr. McMillan. They're reading the water temperture at the discharge or top end of the fuel assembly in which they're located. And there are 52 of those out of 177 fuel assemblies

Mr. Weaver. And 600 degrees is the highest they measure. Is that correct?

Mr. McMillan. No, sir. They measured quite a bit higher than that during this accident.

Mr. Carr. So there you have 52 things that somehow or other have to get outside the reactor vessel.

Mr. McMillan. Which are not connected to the cladding, remember.

Mr. Carr. Right, I understand that.

Mr. McMillan. Their position in the reactor coolant —

Mr. Vento. If the gentleman will yield, what they read were average temperatures of those, was it not? They were not reading the direct temperature of any one of those 52. They were reading average temperatures, were they not?

Isn't what they read on the panel, doesn't it come out as the average temperature?

Mr. McMillan. That's a different instrument. There are other resistance thermometers in the reactor coolant system which read the reactor coolant temperature. And then by averaging the hot and the cold temperature, you get what is called TM, or average temperatures.

Those are entirely different. They're not in any way related to these thermocouples in the reactor core.

Mr. Carr. But you do have at least 52 thermocouples in the reactor core, and they have leads that come out. And so, presumably, you could go from, say, 52 to 177 without a major overhaul of technology, right?

Mr. McMillan. That would require putting 177 lines of the sort that we have in the reactor vessel now.

Mr. Carr. Right. You'd be increasing by a factor of about 3. But that's not an insurmountable thing.

Mr. McMillan. The next step is a factor of 200. That I would say is insurmountable.

Mr. Weaver. All right. Were you finished, Mr. McMillan, with the remarks you wished to volunteer?

Mr. McMillan. I would like to just say I think a few more words, Mr. Chairman, if I could, about the response of B&W in the hours, days, and weeks after the incident.

We were notified of the occurrence early in the morning of March 28th, and I'm told just before 8:00 in the morning.

We assembled a group of our technical managers, our people to assess the information that we had available at that time. In about a day's time, we had set up an around-the-clock manning of a communications and command center in Lynchburg, in order to respond to the requirements of the operator and to help him evaluate contingency plans and contingency procedures for whatever might transpire in the hours ahead.

And we maintained that command center on a 7-day a week, 24-hour a day operation for about a month. And we continued to have a communications center there where people can call in from the site and get information, or we can get people out of bed, or whatever necessary to try and answer or respond to questions.

On the day of the accident, we dispatched people to the site to help assess the situation there. I went to the site personally to head up our representation there.

Mr. Weaver. When were you informed? What hour of the day at Lynchburg?

Mr. McMillan. As I said, I was out of town at this time. Jim Deddens was there in Lynchburg. Maybe you ought to respond to that question.

Mr. Deddens. We were first informed about 7:45 in the morning, when one of the managers in the service end of our business was called by the B&W representative in residence at Three Mile Island. That individual, the gentleman in residence at Three Mile Island had been called by Metropolitan Edison personnel about 6:00 in the morning. He arrived at the plant site about 7:00 in the morning and called our Lynchburg office about 7:45 that morning.

Mr. Weaver. Thank you.

Mr. McMillan. As I say, we had a team at the site. I personally was there for approximately a month in support of the operations at the site. Our primary priority at that point was to support GPU in getting the unit into a long-term cooling condition.

We did a lot of work in developing procedures and plans, most of which were subsequently adopted by the utility.

We did a lot of work on natural circulation cooling so we could cool the reactor system without depending on the reactor coolant pumps and so on.

We immediately notified -- by "immediately," I mean on the Friday and Saturday. We called our other operating utilities and asked them to make sure that they checked their auxiliary feedwater system to make sure the auxiliary feedwater system was in a condition to perform if it was called upon to operate.

We issued an advisory on Sunday to the utilities with the same information in it. We offered to the other operating utilities to come to Lynchburg and sit down and we would tell them as much as we knew about the incident and the events associated with it, help them with a comparative analysis of their system with the Three Mile Island system so they could see where there were similarites or differences.

That meeting took place on the Tuesday following the event.

We also offered at that point to provide supplementary training for their operators and we modified our simulator so it could fully simulate the Three Mile Island conditions, offered a one-day training program for the operators of our other operating units.

I'm not sure what the number is now. As of last week, we had about 130 operators that had been through that training. In fact, now it's up more in the range of 150 -- to acquaint them with the Three Mile Island incident, and more importantly, how to identify conditions of that sort by whatever technique they might find themselves in a saturated reactor coolant system situation.

We also worked with our operating utilities in preparing license information in response to Nuclear Regulatory Commission requests, and we're involved with them in the support of their analyses and the discussions that were held with the Nuclear Regulatory Commission on the continued operation of their units.

We subsequently also issued advisories in parallel with the Regulatory Commission on the importance of keeping the high pressure injection system in operation once it comes in, and in the resetting of the pressure set points for the pilot-operated relief valve and the reactor pressure trip.

As those units are now set up, as the pressure goes up, it will first reach the reactor scram set point on the reactor protective system and the reactor will shut itself down.

It pressure continues to rise beyond that, then the pilot-operated valve will actually -- our analyses indicate that that will be a very infrequent occurrence with the present revised set points for the reactor scram and the pilot-operated relief valves.

We are now involved in support of GPU in assessment of the recovery requiredments and what is it going to take to get that unit back into operation, having essentially gotten into the long-term safe shutdown and cooling situation.

And we are involved in looking at various modifications to our equipment which might be indicated from the lessons learned at Three Mile Island.

These include the reactor vessel water level indicator. It includes a more positive indication of whether or not the pilot-operated relief valve is open or not.

We're agreed to design an auxiliary feedwater system control which is independent of the integrated control system for the entire plant. We have agreed to develop and have already implemented on the operating units a scram of the reactor for either a turbine trip or a loss of feedwater flow. And we are looking beyond that at other modifications that might be appropriate in the equipment, including such things as a Vent on the reactor vessel. So that if you ever got into a condition where you had non-condensable gas in the reactor, you might be able to vent it off by a technique other than the one that was used at Three Mile Island, which took, as you recall, several days to accomplish the purging of that non-condensable gas from the reactor coolant system.

Mr. Weaver. Mr. McMillan, would you describe -- you were discussing now what your plant is to do with the TMI 2. Would you describe the sequence of events required to open the reactor vessel?

Mr. McMillan. We're going through a fairly detailed assessment of that right now. And it would be probably premature to give you more than a general concept of what might be involved.

Mr. Weaver. We're not going to know for, what, six months or a year?

Mr. McMillan. This will be worked out in a matter of a few months.

There are two key factors associated with getting back into the reactor building: One is that there are several hundred thousand gallons of water presently in the bottom of the reactor building. This --

Mr. Weaver. The containment vessel?

Mr. McMillan. In the containment vessel.

This water has contamination of radioactive materials in it, and one of the key steps before we would be in a position to try to get into the reactor building will be the removal and the processing of that radioactive water so that we can go into a dry reactor or containment building.

Mr. Weaver. Is that happening now?

Mr. McMillan. It is not started yet. What is happening now at the site is that they are preocessing and cleaning up some low-level radioactive water which they have in their waste storage tanks. And they're using a portable demineralizer system to try and clean that water up and make room in the waste tanks so that they can transfer some of the water from the reactor building to the waste storage tanks.

As of last week -- I haven't checked this week -- but as of last week they had not started that transfer operation.

The second thing that needs to be done is: There is still some gaseous activity in the containment building, primarily krypton, and that gaseous radioactivity will have to be disposed of by some means, and then, having removed the radioactive water and the radioactive gas, at that point we can begin to assess the steps for actually getting into the reactor building.

Mr. Weaver. You haven't really gotten into that?

Mr. McMillan. Not in detail. One of the things we'll have to determine is: What is the radiation level in there? So, the radiation monitors that are in the reactor building have stopped functioning, and so we're going to have to get other monitor in there to find out what the radiation level is so that we can determine the accessibility to the reactor.

Mr. Weaver. Is it possible the rods are so mangled they could never come out?

Mr. McMillan. In the reactor core itself? We simply don't know what the condition is in the reactor core. We have done as much analytical or diagnostic work as we can with the available instrumentation, and it's very clear that there's been substantial damage to the reactor core, particularly in the upper regions of the reactor core. And until we get in there and we see what does it actually look like, I wouldn't want to speculate.

Mr. Reis. Is there a possibility that the reactor core is in a condition that it will take a very long time for you to be able to reduce pressure or be able to reduce pressure without increasing the temperature to a prohibitive degree; in other words, that you will not be able to force coolant through certain areas of the core unless you are under very high pressure?

Mr. McMillan. No. I think the heat-generation levels of the decay heat or the radioactive decay are at a low enough level now so that as long as you keep water in the core that the core will remain cool.

Mr. Weaver. Mr. McMillan, we skipped over the fourth point of your significant factors -- containment isolation. And Mr. Cheney would like to ask a few questions.

I intend, once we adjourn here today, that's it. We've got the Alaska bill on the floor, and we're going to have to be on the floor for it. But I feel I have some more questions, and Mr. Cheney does, and some other members may, so I want to continue, if it's all right with you, Mr. McMillan, as long as we can, and when we're done, we're done. Is that all right?

Mr. McMillan. We're here at your pleasure.

Mr. Weaver. Thank you very much. I appreciate that.

Mr. Cheney has some questions on containment isolation.

Mr. Cheney. Thank you, Mr. Chairman.

I was just curious. This was, as you mentioned, one of the key areas. To your knowledge, didn't the fact that water was drained out of the sump pump, out of the sump in the reactor building, the auxiliary building, that was the key event, was it not, the release of radioactive material to the atmosphere?

Mr. McMillan. That was one of the principal sources of radiation. I think it's been determined there have been other sources, such as some of the water that is continuously let down from the reactor coolant system goes into a makeup tank that has a free surface in it and some radioactive gases got into that tank and found its way into the waste gas system, an indication that some of that leaked out into the auxiliary building also. But the major source was the transfer of that water.

Mr. Cheney. I wonder -- as I understand it, at TMI there is one parameter that determines whether or not containment is isolated; that's pressure in the reactor building itself. Is that correct?

Mr. McMillan. I am trying to refresh my memory. The first one to actuate containment isolation is at four pounds per square inch.

Mr. Cheney. I guess the question is, then: Are all of your reactors designed the same way, to the same specifications?

Mr. McMillan. No. The isolation conditions or ground rules are different on different reactors.

Mr. Cheney. Why is that? Isn't there some standard that applies?

Mr. McMillan. No. There is no specific standard. There is a degree of flexibility left in the design of the plant for actuation of the isolation.

Mr. Cheney. Are you saying the NRC doesn't have any standard for determining when you isolate the containment vessel?

Mr. McMillan. There is no standard that says all reactor vessels must be isolated if X happens. There is some degree of flexibility, depending on this particular design.

Mr. Cheney. That's something we should perhaps pursue with NRC.

Mr. McMillan. All reactor building containment isolation design bases are reviewed and approved by the NRC.

Mr. Cheney. It's sort of on a case-by-case thing. There is no standard approach.

Mr. McMillan. That certainly was correct at the time these units were designed.

Mr. Cheney. I understand, for example, Arkansas 1 -- and is it Oconee Units 1, 2, and 3 -- the reactor sump and the reactor coolant drain tank discharge lines are normally closed and require operator actions for the transfer of fluids out of the containment vessel?

Mr. McMillan. I don't know what the situation is at Arkansas. I do know that Oconee, they have provision for automatic sump transfer going from the reactor building into the auxiliary building. Their operating requirements are such that the only time they use that automatic transfer is when they're shut down. Before they go back into operation, they eliminate that automatic transfer provision, so that in the event they were to spill water in the reactor containment building at Oconee it would not be automatically transferred during the period when the reactor is in operation.

Mr. Cheney. Why would there be different design specifications for those reactors, say, TMI on the one hand, and the Oconee units on the other?

Mr. McMillan. Well, let me take a step back. This is not a system that we designed, so there is no basis on which we could provide a standard that all the designs ought to be this way.

Mr. Cheney. Who does design it, then?

Mr. McMillan. The balance of plant, which we call anything outside of the nuclear steam system, is designed either by the utility, in the case of Duke, or several of the utilities, or by their agent, the architect engineer. In the case of Three Mile Island, the balance of plant was designed by Burns & Rowe. So at each reactor, balance of plant, the things that support the structures, the whole secondary system, the auxiliary feedwater system, the main feedwater system, all the electrical systems, all of that is designed by an architect engineer or by the utility.

Each of those has his own unique requirements or desires, and there is enough flexibility within the NRC guidelines so that they can design it to their own specific requirements. But they have to be approved by, on a case-by-case basis, have to be approved by the NRC.

Mr. Cheney. Babcock & Wilcox, then, would not be involved in that question of when the containment is isolated or what actuates isolation?

Mr. McMillan. We are involved to this extent: There is certain equipment and certain lines going through the reactor building wall that need to be kept open in an isolation case, and we would identify what those are. We usually work with the architect engineer or the utility in the development of the containment isolation philosophy so that we can assure ourselves that those lines that are required are in fact available.

Mr. Cheney. Then whose responsibility is it to see to it that certain criteria are established? I guess it's the NRC, if we're determining when the containment vessel in fact goes up. It seems to me one of the problems of Three Mile Island, frankly, was that after we had the transient we had the feedwater problem, and then we got into the problem of damage to the core through that whole process the PORV blowing off and so forth, that if there had been no removal of that water from the reactor sump in the auxiliary building, we would have minimized even more than we already did; we would significantly reduce the amount of radiation that has reached the atmosphere.

I guess what I am looking for is guidance on where we ought to look for trying to ascertain how to improve that system.

Mr. McMillan. I think the whole issue of containment isolation philosophy appropriately needs to be reviewed and reassessed, and I think we will move in the direction through the regulations of isolating the reactor building or selected lines in the reactor building. In any event, where you have an emergency core cooling system actuation -- and many units are designed that way -- it's not designed that way at Three Mile Island.

Mr. Cheney. Okay.

Mr. Myers. Mr. Weaver has gone to vote.

Mr. Terrell. Mr. McMillan, could I ask you a question while they're discussing the vote?

Mr. McMillan. Yes.

Mr. Terrell. Why were the operators 15 minutes into the event, the reactor coolant's drain tank blew at 190 psi, when I understand the set point was 200? Why would he not have known at that point in time that that would certainly require some sort of a tremendous energy force to bring that about? Why wouldn't he have known about that 15 minutes into the event, that PORV was causing that; or could there have been another source for the energy that brought that about?

Mr. McMillan. The only two sources of energy that could have driven the pressure up in that quench tank or containment drain tank are either the pilot-operated relief valve or the code safety valves in the pressurizer. Once that rupture disk broke -- and the rupture disk was designed at a pressure to break before you reach the design pressure of the tank to protect the integrity of the tank -- once that rupture disk broke and the pressure tank dropped, he would have to say, "I wonder where all that came from?" It either has to come out of the pilot-operated relief valve or the code safety valve.

Mr. Terrell. Aren't the code safety valves set at a much higher pressure?

Mr. McMillan. The code safety valves are set at 2500 pounds pressure. And, of course, at no time during this transient did the pressure get to that level.

Mr. Terrell. That should have been a very clear indicator 15 minutes into the event, something like two hours -- what -- 2.3 hours into the event before they even realized it.

Mr. McMillan. That should have been an indicator. As I said earlier, the pressure and volume alarm, the pressure and level alarm on that quench tank should have been an indicator; the loss of pressure in the reactor coolant system was an indicator. Once that tank burst, the reactor building pressure should have been an indicator.

There are a lot of symptoms that would indicate that the power-operated relief valve was open.

Mr. Terrell. And/or the safety relief valve was open. In which case they didn't have a pressure that would indicate that that took place because it never got that high. But there's no other source that would have generated the kind of energy to blow that disk?

Mr. McMillan. No.

Mr. Vento. Apparently, one of the problems is that some sump pump went on at some point, and that was, apparently, the first indication that the operator had that came to his attention that there was some excess water in the reactor building. He didn't know where it was coming from, but at this particular area or at this particular facility, apparently, they go on quite frequently to remove condensation that builds up in that building and that flows down. I don't know what the other immediate problems are, but it's not an unusual event for that to occur.

Is it possible, you know, based on the type information we have, that the operator did follow a procedure as was established -- but I guess it's impossible to predict exactly what the scenario was, but according to the information that they presented, they were following a procedure that they thought was appropriate.

Mr. McMillan. I really can't comment on the sump. The frequency with which the sump pump is called into operation.

Mr. Vento. Apparently, there is no direct information that that was called up by someone else. Someone phoned in that information or communicated it to them. He did not have direct information for him on that activity. But you suggest he did have it on the quench tank. There is an alarm system on that quench tank that indicates that it's filling up.

The other one deals with the thermocouple in the vent pipe in terms of what its temperature was and what it should have been. Is there a procedure regarding training operators on that temperature of that tailpipe that they should follow? Do you know that they were within or without the parameters? Do you know what the parameters are for that?

Mr. McMillan. I don't know. I said earlier that's something we'll have to run a check on.

And we also will run a quick calculation, having once discharged, how fast is that temperature --

Mr. Vento. These are the same questions we asked of the operators last week, trying to get some feel. It was their judgment, you know, quite expectedly, that they were following the procedure as it was set out in terms of what they were doing, that there was not the direct readout in terms of pressure variant that would have led them to believe that it's not the direct -- of course, I point out the average temperature problems issue. Is there any way that, for instance, even today, other than the chemical samples, with regard to the breakdown, which I think is the type of analysis that you're doing right now, when you indicate that you're doing some analysis, is that the type of analysis that you're doing in the breakdown of the fuel material? It's pretty much a chemical analysis at this point?

Mr. McMillan. What we have in the way of evidence right now are reactor coolant samples which we have withdrawn and analyzed for the various radioactive constituents, in order to determine what is the level of radioactivity in the reactor coolant system. We do not yet have similar samples from the bottom of the containment vessel. We have not gotten those samples.

Mr. Vento. They called out, of course, readings, I guess, in two instances, where the amount of boron in the water was low, considering the amount of injection that had gone on in terms of the amount of water. I think they said 400 parts per million, you know, which is, apparently, low. I mean, the operators -- I think I am relating it correctly -- they at least at two points -- I may have got the figures wrong -- but they had called out at two points readings on the amount of boron in the water. Apparently, that was not a definitive reading as to reaction in the coolant or of the fuel material as to what its status was.

Right now, in other words, there apparently is no means to make a determination as to the integrity of the fuel that is really satisfactory concerning -- in other words, if they had had that information, obviously, they would have realized that something was happening to the fuel. And apparently -- in other words, we're talking about temperature, we're talking about the amount of boron in the water, with how it's being absorbed.

And so these are factors that they just didn't have, apparently, definitive information on, at this point.

Mr. McMillan. I am not at all familiar with your comments about the boron. There is no way, except by chemical sampling, to determine the boron level.

Mr. Vento. I have to leave. You will have to excuse me. Thank you.

Mr. Weaver. I am sure that the engineering business is comparable to Congress, where you're called on various different things, Mr. McMillan, but we usually have about 10 going on at once, and we try to concentrate as best we can.

I have a series of what I hope are just simple factual questions. I have already asked you when Lynchburg was informed. Your response was: at 7:45.

The second question in this series is: What advice did B&W offer the people who were dealing with the accident at TMI over the course of the first day?

Mr. Deddens. Well, shortly after we were informed at 7:45 in the morning, word was relayed to various other individuals in our office, and we convened a team of people -- managerial and technical people -- to review what information we did have available to us at that time. And that meeting took place, oh, perhaps at 8:30 or so the morning of the accident, on March 28.

And I would say, at that point it was more in the nature of assessing what information we did have and establishing a list of additional information that we would need in order to be able to reach a conclusion on the reactor coolant system so that we could provide recommendations on what steps to take next. At that 8:30 to 9:00 meeting that's pretty much what was done.

In addition to that, we dispatched three experienced engineers to Three Mile Island by charter aircraft, these individuals being ones who had experience in analyzing loss-of-coolant transients on the reactors and individuals who had experience in being able to interpret plant data and relay that back to us at Lynchburg so that we could make proper judgments.

Mr. Weaver. Is that unusual? I mean, you've had other transients at other plants that you designed or supplied. Is this something you've done a number of times before, dispatch the engineers? Or is this an unusual thing that you did?

Mr. Deddens. It varies with the plants that have been through loss-of-feedwater transients. The loss-of-feedwater transient is not particularly unusual, as a nuclear plant transient.

Mr. Weaver. I'm asking, the dispatching of engineers and your meetings, was that an unusual thing?

Mr. Deddens. Okay. Those engineers went to the site, and the next point in our sequence --

Mr. Weaver. Excuse me. I was wondering if that is an unusual thing for B&W to do?

Mr. Deddens. To send engineers?

Mr. Weaver. Have you done this five or ten times before, dispatched three engineers, had the meetings to discuss the transient?

Mr. McMillan. My answer would be yes, we have had situations where unusual events occur in the operation. And frequently, for example, we have loose parts monitors on our equipment and if they get an unusual noise we will frequently dispatch two or three engineers with special detection equipment to go help evaluate. What is that noise? What does it mean?

Mr. Weaver. And they have the meetings as described in Lynchburg? So it's happened before. So far this is not a highly unusual thing, as far as Lynchburg is concerned?

Mr. Deddens. You're referring to Three Mile Island?

Mr. Weaver. Yes, sir.

Mr. Deddens. I think there was an element to this one that made it different, and that is that we knew early in the morning of March 28th -- that is, at the time of the first phone call, that there was some radioactivity in the reactor building.

Mr. Weaver. I see.

Mr. Deddens. And that established a difference between this particular transient and those that we had experienced before.

Mr. Weaver. Proceed, please.

Mr. Deddens. Okay. I think the next major occurrence as far as our activities at Lynchburg were concerned was to set up a room in Lynchburg, in our office building in Lynchburg, in which we could establish communications with the control room. Initially we did not get a direct telephone line established to the control room, but we were in contact with a second individual who is in residence at Three Mile Island, who was at his home, and he in turn was in periodic contact with the control room, with Mr. Rogers, who was the individual who was called in at 7:00 o'clock in the morning.

Mr. Weaver. What time was this that you did this, this communications center?

Mr. Deddens. We did have that communication link established, and what we received was periodic updates on the status of the plant, and were able to get supplied to us information that we required in order to reach some kind of judgment on what the next step should be. Those concentrated mostly on the condition of the reactor coolant system: temperatures, pressures, any flow established in the reactor coolant system, how was heat being removed from the reactor coolant system, which developed information on one steam generator being used and another steam generator being isolated.

Mr. Weaver. What time was this that you set up this communications?

Mr. Deddens. These communications took place mostly in the late morning, between, say, about 10:30 to 11:00, over the lunch hour, and until early afternoon.

Mr. Weaver. Now this was unusual, wasn't it? You hadn't done that before, right?

Mr. Deddens. The whole transient, the whole sequence of events was unusual.

Mr. Weaver. The response of B&W to it, things that you hadn't done in previous transients, is what I'm trying to establish here; is that correct?

Mr. Deddens. Yes.

Mr. Weaver. Thank you.

Mr. Deddens. I would say by about 1:00 to 1:30 in the afternoon we had a fairly clear picture of the reactor coolant system condition, which indicated to us that the reactor coolant system was in a superheated condition and that there was steam in the various parts of the loop, which then led to an evaluation and a recommendation on our part that the high-pressure injection be reinitiated. At that point we knew that the high-pressure injection pumps had been stopped.

Our principal recommendation was to get water back into the reactor coolant system to increase its pressure and drive the temperature down, in order to quench the steam accumulated in the reactor coolant system and reestablish cooling to the core. And that was the major thrust of our recommendation at that point.

Mr. Myers. The NUREG-0560 has a chart of times when various pumps were on. It indicates that the high-pressure injection pumps were on, to some extent, for almost the whole period.

Do you have any indication of the extent to which the volume of water that they were pumping in?

Mr. Deddens. They secured early, as Mr. McMillan indicated, when there was a depressurization of the reactor coolant system. The high-pressure injection system came on. All the pumps started.

But then, within a few minutes, the operators shut down all but one of those pumps, and they did maintain some flow into the reactor coolant system through one of the high-pressure injection pumps or, as it's referred to in that mode of operation, the makeup pump.

Mr. Myers. 1-A was on for the first four hours, it seems. 1-C and 1-B -- well, 1-B was on for four hours through 17 hours, and 1-C was on intermittently from -- well, was on continuously from four hours to nine hours, and went on and off from there on.

But do you know how much volume was being injected during that time?

Mr. Deddens. That would be about the capacity of one pump, which is approximately 150 gallons per minute.

Mr. Myers. Is there a record of exactly how much or reasonably -- is there a reasonable estimate of how much water was pumped in per hour, as a function of time?

Mr. Deddens. I don't have the figures in my mind, but there should be a record.

Mr. Myers. Have you seen that record?

Mr. Deddens. There should be a record of that.

Mr. Myers. Have you seen it?

Mr. Deddens. Not as a continuous display.

Mr. Myers. I mean, do you know, if you say -- I mean, do you know where to get the record of how much water was pumped in?

Mr. McMillan. That's one of the mysteries. It is just unknown.

Mr. Myers. Is that data that is recorded, the volume of water being pumped in by the high-pressure injection? That's data that doesn't exist?

Mr. McMillan. That's right. You have to depend upon the memory of the operator as to what kind of throttling he may have been doing.

Mr. Myers. At this point is the operator going to remember?

Mr. McMillan. No. I think you were at the ACRS meeting where I discussed the situation and indicated one of the frustrations in doing a complete analytical assessment of the conditions that took place during the accident is the lack of detailed knowledge of what the high-pressure injection flow was as a function of time. And you can do -- the best job you can do is try to assess that from the fact pumps were on or off and some assessment of what the operator recalls as the situation.

But that is one piece of information that's missing to allow for the really detailed analysis of the transient.

Mr. Weaver. At what point -- we're going to have to proceed. At what point did B&W conclude that significant fuel damage had been done?

Now, we've heard the amount of one-half a percent damage to the cladding or whatever, and that's not what I'm talking about. Significant fuel damage had occurred; at what point did you conclude that?

Mr. Deddens. I don't know how you define "significant," but there were some early indications, obviously, from the discharge of radioactivity into the reactor building and the contamination which existed in the water that had been pumped from the reactor building to the auxiliary systems building. Now, that told us that there was fuel damage and leakage of fission products from the cladding. I don't believe that we were able to put together a detailed analysis to get a good picture on the full extent of the damage for a couple of days after.

Mr. Weaver. Weren't there indications, critical indications that serious damage had occurred? I realize we know all this by hindsight now. But weren't there at the time serious indications: the amount of radicactivity?

Mr. Deddens. Certainly the amount of radioactivity.

Mr. Weaver. The measurements of hydrogen, et cetera.

Mr. Deddens. The presence of a high degree of radioactivity and the presence of hydrogen in the reactor coolant system, the presence of the famous bubble, would indicate that there had been a significant amount of metal-water reaction taking place due to overheating of the core, and that information was available within -- the hydrogen bubble, most of that occurred on the 29th.

Mr. Weaver. But didn't you know on Wednesday the amount of hydrogen? There was the hydrogen burn on Wednesday, 1:50 p.m.

Mr. Deddens. That information was not immediately available to us on the 28th. Most of the information available to us on the 28th was related to the reactor coolant system temperatures and pressures, and the data being relayed to us was being used to assess the condition of the reactor coolant system, so that we could get it to a safe and long-term stable depressurized condition.

Mr. Weaver. When were you told that a pressure spike had occurred in the containment, that this would have occurred, of course, at 1:58 p.m. Wednesday afternoon?

Mr. Deddens. Who was told?

Mr. Waaver. When?

Mr. Deddens. I did not become aware of it personally until, I believe -- I don't really recollect when I became -- when I first became aware of it.

Mr. Weaver. Wednesday?

Mr. Deddens. I would say some time Wednesday evening or perhaps early Thursday morning.

Mr. Weaver. How did you read this, the pressure spike? What was your reaction?

Mr. Deddens. My reaction to it was that it possibly -- it was not an explosion; that it could have been one of two things: a transient on the instrument which indicates pressure in the reactor building, an electrical transient which could cause momentary spike.

Mr. Weaver. You're talking about a malfunction of the action reading, then?

Mr. Deddens. That's something that could not be ruled out at that time.

Mr. Weaver. One of the things that's perturbed me throughout our investigation is that they misread the gauge on the PORV and didn't -- I mean, they never questioned that, that it might be, whatever you called it, a spurious signal. They didn't question that that could be a spurious signal. But everybody says that the spike, we thought that was a spurious signal.

I just don't understand this.

Henry?

Mr. Myers. Were you aware Wednesday or Thursday that the spike was accompanied by a turning on of the containment sprays?

Mr. Deddens. Accompanied by what?

Mr. Myers. The contaiment sprays turning on.

Mr. Weaver. The turning on of the sprays in the containment.

Mr. Deddens. Yes.

Mr. Myers. You were aware that those two happened. Were you aware that the air intakes for some of the pumps inside the containment showed high temperature at about the same time?

Mr. Deddens. We were aware of that information in turn, as it came in.

Our situation was in Lynchburg that we were getting information relayed to us over the telephone. We did not -- we were not -- we did not have an operating presence at the power plant in the control room. That was Met Ed's responsibility. But we were getting information relayed to us as we could, primarily over the telephone, verbally, and by way of telecopy.

Mr. Myers. Do you remember approximately when you learned that the containment sprays had been turned on?

Mr. Deddens. I personally was not aware of that until, I would say, some time on the 29th.

Mr. Myers. Were any of your colleagues in Lynchburg aware that the containment sprays had been turned on at about the time that the pressure pulse was observed on Wednesday?

Mr. Deddens. I cannot answer that specifically.

Mr. Myers. You have not discussed this with any of your colleagues?

Mr. Deddens. Oh, yes.

Mr. Myers. Have you asked them when they were aware that there was a pressure pulse in conjunction with containment sprays turning on, in conjunction with hot air being detected at air intakes of the reactor coolant pumps?

Mr. Deddens. I have not asked that specific question.

Mr. Myers. Do you think, given all these things happened in conjunction, that it is reasonable to infer that this was some sort of electrical transient?

Mr. Deddens. I think my remark on that was perhaps misinterpreted. When I -- and the context of my answer to that question was, when I personally first became aware of the pressure spike, it entered my mind that it could be one of two things: either a spurious signal, an electrical transient, which does occur frequently in power plants, or a real pressure spike that did in fact occur in the reactor room.

Having the specific information on the pressure spike, a very sharp, narrow spike on a chart, you could not rule out, with that one piece of information, you could not rule out either one. That was the answer.

Mr. Myers. But would that not suggest to you that you look to see whether it was indeed a pressure spike? One indication of it being a pressure spike would be that the containment sprays had also turned on.

Mr. Deddens. Yes, and that was done subsequently.

Mr. Myers. Did that occur to you when the pressure spike was first mentioned to you? Did you ask, well, did the containment sprays turn on also?

Mr. Deddens. I don't recollect asking that particular question.

Mr. Weaver. Wouldn't you consider that a very significant occurrence? It's hard for me to judge, of course, because I don't know these plants.

Mr. Deddens. At the time, our attention was primarily focused on getting the reactor to a safe and stable shutdown condition. The pressure spike was something that occurred at one instant in time. It was a sharp spike in which the pressure went up to the range of 28 pounds and then right back down again.

It was not a condition where the pressure went up and stayed for a long period of time. So it didn't at that point represent the top priority item in our sequence. The top priority was being given to getting the reactor down to a safe and stable condition.

Mr. Reis. When you did first learn of the pressure spike and/or when you then later heard about the pressure sprays and/or the intake temperatures that Mr. Myers mentioned, what did you think of as the possible causes of that spike, other than the possibility of a transient of some sort, an electrical transient?

What sort of conclusions did you come to?

Mr. Deddens. Well, it could have been a burning. It was not an explosion.

Mr. Reis. If it was a burning, it would have been a burning of?

Mr. Reis. What gas would you, as an engineer trained in this plant --

Mr. Weaver. The answer is hydrogen. And what would that mean?

Mr. Deddens. Well, that would mean that there was some discharge of hydrogen from the reactor coolant system into the reactor building atmosphere.

Mr. Weaver. Significant to get to 28 psi, correct? Doesn't it take quite a bit of hydrogen in the containment to do that?

Mr. Deddens. Not necessarily. It could have been a localized concentration of hydrogen which, if it were dispersed throughout the entire containment building atmosphere, would be a very small concentration. But it could have been localized in a certain small volume such that its concentration was high enough to burn. And if it were then ignited, it would in fact burn and create a very short-term interval pressure spike, which could then be detected by instrumentation.

Mr. Weaver. Even that much hydrogen, what significance would that have in your analysis as to what was going on in the reactor vessel?

Mr. Deddens. There's no way to relate the pressure spike with any particular concentration of hydrogen in the reactor building.

Mr. Weaver. I'm sorry. My understanding is this -- and I want to stand corrected, and I wish you would -- that the way hydrogen created here is that the zirconium combines with oxygen and frees hydrogen. Therefore, it's a function -- the amount of hydrogen is a function of how much zirconium oxidation has occurred.

Mr. Deddens. Yes.

Mr. Weaver. Therefore, it's a measure of how much damage to the cladding has occurred; is that a fact?

Mr. Deddens. It's an indication of damage to the core and it can be an accurate indication, providing you know what the concentration, the amount of that hydrogen gas that has been released, really is. All we had was a pressure spike and an estimate chart that said that something created a spike up to 28 pounds.

But there's no way that that particular piece of data could have been related to the concentration of hydrogen in the reactor building and therefore related back to damage to the core.

Mr. Weaver. At 2:00 o'clock Thursday afternoon, the NRC Commissioners testified to this Subcommittee that there were no more real problems, everything was taken care of, there had been an extremely small amount of damage to the fuel rods, but an almost insignificant amount.

Is that the kind of testimony that could occur from the information that we have now about the amount of hydrogen, et cetera? Could you testify in that regard?

Mr. Deddens. Today?

Mr. Weaver. No, on Thursday, the 29th of March, knowing about this pressure spike, knowing about the amount of hydrogen, et cetera, could you testify that everything was fine, that there was a very insignificant amount of damage to the fuel rods?

Mr. Deddens. I think I certainly could not have testified that everything was fine, because it wasn't.

Mr. Weaver. I don't mean fine, but under control and taken care of. 8

Mr. Deddens. On the 28th, we did not at that point know the extent of damage to the reactor core. I could not -- don't believe on the 28th I could personally have testified on the extent of damage. I certainly could have testified that there was damage to the reactor core, there was an indication to that effect.

Mr. Weaver. Do you realize what you're saying is that we don't know what's going on inside that reactor vessel? Do you realize the kind of indictment that is of our ability to control these things? That's what you're saying. You're saying this enormous amount of damage that occurred, we didn't know it, we didn't know if for two or three days. What you're saying is we don't know what's going on inside that reactor vessel, isn't it?

Mr. Deddens. No.

Mr. Weaver. No? You just told me.

Mr. Deddens. That's not what I said. I don't believe that's what I said.

Mr. Weaver. Please clear it up, then. You just said that what this involved, even with the pressure spike, even with the amount of hydrogen, even with the radioactivity, that you wouldn't have testified that there was significant damage in that reactor vessel. You just said that.

And now I'm saying, well then, therefore, you didn't know what was going on inside that reactor vessel. And I'm just simply saying that's a terrible indictment. That scares me.

Mr. Deddens. What I thought I said -- and perhaps I didn't make myself clear -- is that on the 29th I could not have testified to the full extent of the damage that had occurred to the reactor core. But I certainly could testify that damage had occurred.

Mr. Terrell. Mr. Deddens, let me ask you this. Without some indication of the quantity or the volume of hydrogen -- and all you had was an indication of the spike, a very short pulse, something like one second -- and all the other parameters that were going on, you would know on March 29th, based on that, that there had been some damage to the fuel. But without some of those other parameters, without some fairly reasonable readout of the volume of hydrogen, you wouldn't really know how much or to what extent the fuel had been damaged; is that correct? Or anyone with any kind of background?

Mr. Deddens. Well, there were other data available to us. And when it became obvious and apparent to us that there was hydrogen trapped in the reactor coolant system, the famous bubble, we devised a series of tests to be able to measure about how much hydrogen was trapped inside the reactor vessel.

Mr. Terrell. Was that done prior to the 29th?

Mr. Deddens. That was mostly on the 29th.

Again, April -- March 28th, the day of March 28th was concerned with reestablishing adequate core cooling, to get the steam binding that existed out of the reactor coolant system and to enable the operators to proceed to an orderly and safe long-term depressurization and cooldown of the reactor core, to get it into a safer condition than it was then in.

Mr. Terrell. I understand. What I was getting at is, did the NRC -- could they have had better information, numbers or whatever, measurements, where they could have come up here on March the 29th in the morning, the Commissioners, and given us any indication within a reasonable bound of fuel damage, how significant?

Would they have known that? Would they have had any information that you didn't have, in order to come up here and say that, we estimate the percentage is floating around between a half and one percent? Would they have known enough on information, perhaps, they had that you didn't have, that would have inferred that there would have been a lot greater damage to the fuel? Or would it be just supposition that they would be saying that?

Mr. Deddens. Some of the best information that was used to assess the extent of core damage were the in-core instrumentation in the form of thermocouples and the self-powered in-core instruments. The thermocouples provided data on the core outlet temperatures and essentially a map across the top of the core, which gave outlet core temperature distributions from one location to another.

Not every fuel assembly has a thermocouple, but there are a sufficient number of them in there to give a pretty good map of the profile across the outlet of the core. That information was available.

The in-core instruments had failed due to overheating during the transient. But they did operate for a sufficient period of time to be able to give information. That information is displayed, it's logged into the computer. It takes some time to get that out or to analyze it and draw conclusions from it, and that process took a day or two, to be able to take the information and analyze it and to draw conclusions which you could then use to be able to make a better judgment on the extent of the core damage.

Mr. Terrell. But would that information have been available by Thursday morning?

Mr. Deddens. I think it could have been available by Thursday morning.

Mr. Terrell. Do you know if it was?

Mr. Deddens. To the extent it was known to the NRC, I can't answer.

Mr. Terrell. But you didn't have it? B&W didn't have that informatin?

Mr. Deddens. We began to get thermocouple data on the 28th, and in-core instrument data -- my recollection is that we began to get the bulk of that on the 29th.

Mr. Terrell. Some time in the afternoon? Morning? When?

Mr. Deddens. Without going back and checking the records, I couldn't say.

(Discussion off the record.)

Mr. Weaver. On the record.

My understanding is that minority staff asks whether you know of the 2,000-degree reading taken directly from the wires from the thermocouple, as opposed to the computer. That's what I thought the question was you responded to. Would you respond again, then, please?

Were you aware of the 2,000-degree reading from the thermocouples on Wednesday? What time was it Wednesday?

Mr. Myers. I understood that it was something like 7:00 in the morning.

Mr. Weaver. In the morning on Wednesday?

Mr. Myers. They went down. They saw the core thermocouples off-scale and they went down. They took a direct reading and one of those indicated 2es. Were you aware of that?

Mr. Deddens. No, sir, I was no that particular reading. That would have had to haveading where it was taken at the job site by specialnt.

Mr. Weaver. Exactly.

Mr. Deddens. By disconnecting le leads into the computer.

Mr. Myers. Were you aware of aadings, measurements?

Mr. Deddens. Yes, I was, at a ; not on the 28th.

Mr. Myers. You were aware of ms made after the 28th, or were you aware of measude on the 28th at some later date?

Mr. Deddens. I was aware of me later, that measurements had been made at the ply the technique which you described.

Mr. Weaver. Had you been notif0, when the call came in at 7:45 Wednesday mornid, there have been thermocouple measurements made ,000-degree reading in the core, what would yourhave been then?

Mr. Deddens. That certainly woeen information that would shed a different light ontion at hand. It would be speculation on my part amy reaction would have been, but it certainly would have been more significant information.

Mr. Weaver. It would have had to have been significant damage?

Mr. Deddens. Yes, it would have been indicative of significant core overheating.

Mr. Weaver. Your actions would have been different advice, different, if you had that information?

Mr. Deddens. I don't think so. I think, again, our concern was with cooling the core, getting the core down to a safe and stable condition. Conditions that had existed prior to that time as the result of an accident would not have changed the need to get the core down to a safe and stable condition.

Mr. Weaver. Thank you very much.

Mr. McMillan, during your early testimony you mentioned -- actually, I think it's your point six significant factor -- the delay in auxiliary feedwater. You felt -- most of our other testimony has downplayed this, more or less, as just a transient that had occurred.

But you said -- you gave it more significance, I thought, than other people who testified. Do you consider this a really critical -- obviously -- no, excuse me, forgive me. I'm off.

We're not talking about the condensate pump that started the whole thing. By the way, do you agree that the condensate pump did start the whole thing, its tripping?

Mr. McMillan. The evidence I have is that suction pressure on the condensate booster pump was lost and it shut down, and that in turn --

Mr. Weaver. Tripped downline, yes. And we don't know yet what caused that.

Mr. McMillan. I don't know what caused that.

Mr. Weaver. I'm sorry, forgive me. What I meant to get was the closed valve on the auxiliary feedwater. You felt that was more significant, I think, than others who have testified.

Would you explain that, what the significance was to you?

Mr. McMillan. Let me say at the outset that the failure to get auxiliary feedwater into the steam generator for eight minutes of itself would not have been a major factor, that this plant could certainly recover from that without any difficulty.

I think what it did was, it created an unusual response in the reactor cooling system which the operators had not previously seen in a typical loss of feedwater condition. So I think it was a confusing factor, but not one that was critical in terms of the ultimate outcome of the incident.

I would characterize the most significant feature as the shutdown of the high-pressure injection system.

Mr. Weaver. That's, by the way, the conclusion I reached on the auxiliary feedwater, is that it posed a confusion that actually affected the actions one way or another throughout the rest of the transient, but did not in itself cause a significant event.

Mr. Terrell. Mr. Weaver, could I interrupt a moment on that 2,000-degree reading? This is Mr. Chakoff. He's working on the minority staff. He's a nuclear engineer. I wonder if he could ask a question relative to that 2,000-degree thing.

Mr. Weaver. By all means. If you'll permit us, we're going back to the 2,000-degree reading.

Mr. Chakoff. I'm curious. What's the design range for those thermocouples that you described that are in the outlet of these fuel bundles?

Mr. McMillan. I don't know. We ought to get you an answer to that.

Mr. Chakoff. For example, when I say design range, I mean, is the range from 400 to 650 or 700 degrees, with an accuracy over that range of, let's say, three percent or one percent, something like that?

Mr. McMillan. We ought to get you an answer. They will read well above the 600-degree range with, you know, reasonable -

Mr. Chakoff. Will it read to 1,000 degrees and would you trust the reading that you got?

Mr. McMillan. In terms of a general indication, I think that we would feel that that would be --

Mr. Chakoff. Would you trust an indication that you got at 1500 degrees?

Mr. McMillan. How about letting me find out what the design range is and get you an answer on that.

Mr. Chakoff. All right.

(Information to be furnished.)

Mr. Chakoff. The point that I was interested in is, if these pieces of instrumentation had been taken above their design capability and had in some way been distorted by the accident, would you in fact after that trust the reading that you got from such an instrument? That was my question, and that's why I was curious about the particular design range of the instrument as to assure the meaningfulness of the data which you got from it after it had been exercised above a limit beyond which it had been intended to. That was my concern.

Mr. McMillan. We ought to find out for you what that design range is. I think we have testified previously that they were off-scale in terms of what the computer readout would permit. Our feeling has been, and subsequently confirmed, that had the computer been allowed at read out at a higher temperature, we would have gotten, again, a general indication as to what the situation was.

But we ought to get a specific answer for you.

Mr. Weaver. That has always puzzled me, why the computer wasn't allowed to read out all the things.

Mr. McMillan. Let me see if I can't respond to that. These thermocouples were put in for the purpose of monitoring the temperature distribution at the outlet of the reactor during normal operation. It was being used as a confirmatory reading to compare with the in-core detectors and try to get some confirmation of the distribution of the power across the reactor in its azimuthal character during a normal operation. And in normal operation, those temperatures never get over 650 degrees. So, for the purpose for which they were intended, a 680-degree cutoff was a perfectly logical situation.

Mr. Weaver. I understand.

Mr. McMillan. We are very happy we had those thermocouples. Certainly we would want to have more, a wider range of readout in the future.

Mr. Weaver. Would the pressurizer have voided if the valves had been open?

Mr. McMillan. I'm sorry, could I try that again?

Mr. Myers. If the auxiliary feedwater valves had been open initially, would the system have cooled, leading to a voiding of the pressurizer initially?

Mr. McMillan. Would we have drained the pressurizer? No. Normally, in a loss of feedwater where the auxiliary equipment, the auxiliary feedwater comes on, there is a drop in pressurizer level. And one of the first things the operators are trained to do -- and it's almost a reaction, involuntary reaction -- when they get loss of feedwater, they immediately trip on one of the makeup pumps, so they get additional makeup for the reactor coolant system.

With that kind of action, the pressurizer level drops, and then it starts to come back in.

Mr. Myers. So that, to the extent that things would have been better or worse had those valves been open, they probably would be better. To some extent they would have been better. You can't see having those valves open initially and lead to something that was worse?

Mr. McMillan. I don't think so.

Mr. Weaver. We're just going to take a couple more minutes, if you don't mind bearing with us.

I have always been concerned, relative to nuclear energy, about the effects of radioactivity damage to the physical structures of the equipment. I think Dr. Weinberg of Oak Ridge has been in the paper concerned with this. But we can't go into that today.

What I am concerned here is that the radioactivity in containment, the rising water levels in the containment, could possibly have caused more equipment failures than it did. Would you elaborate on that possibility of eventually this Three Mile Island becoming worse than it actually got because of equipment failures caused by radioactivity, rising water? What if a pump had failed or the power had failed?

Mr. McMillan. Is your question to the effect that because we had water in the bottom of the reactor vessel or the containment vessel, and because we had high radiation levels, did anything happen that aggravated the incident?

Mr. Weaver. That and what could have happened. Could we have ever faced a worse accident because of this situation? I mean, were we hanging on the edge of the cliff, so to speak, because of the rising water levels and the high radioactivity in the containment, through equipment failure?

Mr. McMillan. Maybe we can think this through together. The water level in the bottom of the containment vessel as such does not create a problem in terms of the operation of the reactor coolant system until it gets up to the level where you begin to get some of the instrumentation transmitters.

Mr. Weaver. How high is that?

Mr. McMillan. The lowest ones are about two and a half feet off the floor and there are several at about the three and a half foot level.

Mr. Weaver. I thought we got higher than that with the water.

Mr. McMillan. We are. Those transmitters, we feel confident at this point, are under water.

Mr aver. I see. Are they functioning?

Mr. McMillan. Some of them functioned for some period of time, days after we believed they were covered with water. They are designed in order to operate in a very high humidity post-accident building condition. They're very tightly sealed and continue to function.

The most -- I'd say the most sensitive of those in terms of the recovery and long-term cooling configuration for the reactor system are the pressurizer level instrumentation. There are three independent channels of pressurizer level transmitters. With time, since the accident we have lost two of those, and a third one seems to come on and go off. It looks, from all evidence, to be approaching the point of also losing that one permanently.

In fact, we may have -- as I said, I haven't been in touch with the site since last week.

I don't see that in the framework of your concern about was there something that happened there that made the accident worse.

Mr. Weaver. Well, I'm really supposing now and asking hypothetical questions. Suppose all three of those pressurizer gauges had gone out on the first day. What would we have been faced with then?

Mr. McMillan. When I mentioned earlier that one of the things that we were doing in Lynchburg in support of the recovery operation, we were developing contingency plans. We were asking ourselves questions like, what happens if that reactor coolant pump stops? What does the operator do?

Mr. Weav r. Exactly. What would? That's exactly what I'm asking.

Mr. McMillan. The first thing he was to do was to try to get another one started. And if that didn't work, he'd try the other two. If those failed, then we had a contingency plan in which he would actuate the high-pressure injection pumps and go solid on the reactor coolant system and control -- throttle the makeup so that he always maintained the water.

Mr. Weaver. Going solid wouldn't have hurt anything at that juncture?

Mr. McMillan. I don't think so.

Mr. Weaver. So your response to this is, if the measuring gear just had failed, we'd just get all the water you possibly can in there?

Mr. McMillan. Our contingency for that condition was and continues to be -- or continued to be for, say, a matter of a week or two -- if you lose all your pressurizer instrumentation, then you should go solid in the reactor coolant system and maintain your pressure, and make sure you've always got water in the reactor coolant system. All you've got to do is make sure you've got water in the system at this point.

Mr. Weaver. Now, we've had some people say that going solid is a very bad thing. Does that mean just in normal operation, or could something very serious have happened to the plant if you went solid?

Mr. McMillan. I don't think any thing serious can happen to the plant when you go solid. It's an undesirable mode of operation for any normal operation. You would not want, in the normal operation of the unit, to go solid. But in the situation in which we are presently at Three Mile Island, with the low levels of decay heat that we're dealing with, we don't see a problem with going solid.

In fact, they have gone solid just to check out the procedure and make sure that it can be implemented effectively.

Mr. Weaver. What other things could have occurred with this water rising and radioactivity?

Mr. McMillan. The first concern would be in instrumentation. I think we've talked about that.

The next concern would be when it flooded to the level where you began to get, where you flooded pumps and valves. They might fail to operate. In fact, the indications are that we have flooded the sump pumps. They no longer are available for future operation because they're under water.

I think the most vital instrumentation we need to be concerned about there for the long term is probabli the decay heat letdown valve. That is the source of flow of water from the hot leg into the decay heat removal system, sometimes called the RHR system. That, I'd say, today is a much less critical issue than it might have been two weeks ago, because we are using other means of getting the decay heat out.

Mr. Weaver. What would have happened two or three weeks ago?

Mr. McMillan. If we had gotten into a situation where we felt we had to use that decay heat system two or three weeks ago, the water level was low enough so that we were confident that that valve would function.

Mr. Weaver. What if it didn't, though?

Mr. McMillan. Then you would have to go through a cooling system that would involve pumping high-pressure water into the reactor coolant system, drawing it out through the reactor building sump, and recirculating it through the reactor system. That involves drawing the water out of the bottom of the containment vessel and circulating it through the auxiliary building and going back in. It's not a desirable mode of operation, but it is a practical one.

Mr. Weaver. And that's foolproof, is it? Could something happen to that?

I'm trying to establish how close we were.

Mr. McMillan. I guess my overall response would be, were we on the edge of the cliff ready to fall off, is no, I don't believe we were. The water in the bottom of the reactor vessel was a concern for us in terms of the longevity of the instrumentation. But I don't believe that at any point we felt we were on the edge of a really serious problem.

Mr. Weaver. What would happen if both outside and diesel power failed in one of these plants?

Mr. McMillan. You're postulating a situation where you lose all off-site power?

Mr. Weaver. You have two sources of power, off-site and diesel.

Mr. McMillan. And the installed diesel generators.

Mr. Weaver. That's what I'm talking about, the diesel generators.

Mr. McMillan. Of course, that means you have no electrical power in the unit at all from any source.

Mr. Weaver. Exactly.

Mr. McMillan. That means you have no instrumentation to monitor what the condition of your equipment is, and you have no power with which to drive pumps.

Mr. Weaver. I'm asking this and I'm finding out what that would mean.

Mr. McMillan. And to manipulate valves. So you're in a situation where your reactor coolant system is sitting there and you're just boiling water off, and the water, the pressure would increase in the reactor coolant system until you reached the 2500-pound, the set point of the pressurizer, and those valves would open and start to relieve the steam from the generator of the reactor coolant system.

Mr. Weaver. You've got a scram long since.

Mr. McMillan. Oh, long since.

I'm sorry. Maybe I misunderstood you. I thought you were asking what would happen today if we lost --

Mr. Weaver. I'm talking about operating Three Mile Island or any other plant, if it lost both sources of power.

Mr. McMillan. While operating at full power, and then you lost all off-site power, and you lost the diesel generators. One of the things, one alternative or possibility at that point is, you've got your turbine generator, the plant turbine generator running, so that you've got a source of electrical power from that turbine generator.

Mr. Weaver. But wouldn't you scram and the turbine would trip?

Mr. McMillan. Yes, you're right, because you've lost power to the reactor.

(Pause.)

Mr. McMillan. We're conferring.

Mr. Weaver. Fine.

(Discussion off the record.)

Mr. McMillan. What we were debating is, in the situation where you lost off-site power, would you be able to retain the turbine generator in operation and use the site turbine generator as a source of electrical power. And I believe the answer to that is yes. If you lost the site turbine generator and the off-site power and the diesel generators, then the only source of electricity you've got in the plant are the battery-backed vital busses for your instrumentation, which could allow you to monitor the condition of the plant by the battery-backed system.

But in that circumstance you would boil off the reactor.

Mr. Weaver. Boil off the water? You see how much I know about this. I have to ask these questions. I don't know.

You've got one source of off-site power on most plants. This is the main power from whichever utility you've got.

Mr. McMillan. No, sir. Normally there are multiple off-site sources of power. Usually they come in from different geographical directions.

Mr. Weaver. Really? So there are several coming into the plant.

Mr. McMillan. Several.

Mr. Weaver. And how many diesel generators?

Mr. McMillan. At least two.

Mr. Weaver. Do they operate separate systems or is one a backup to the other?

Mr. McMillan. The two diesel generators --

Mr. Weaver. Does one diesel operate one part of the system and the other another part of the system, or is one a backup to the other?

Mr. McMillan. One is usually sufficient to do the job, and the other is a standby.

Mr. Weaver. I have no further questions.

Mr. Myers. I just have one more question. This is going back to the pressure pulses.

There was this other. It did occur at the same time that they opened the pressurizer block valve. Were you aware of that?

Mr. Deddens. I don't have all the time sequences fixed in my mind.

Mr. Myers. They were standing there. They were operating the block valve, watching the strip chart. They opened the valve, the pressure went up, containment sprays went on. But you were not aware of all those things happening at once?

Mr. Deddens. No.

Mr. Weaver. One thing. Always something that occurs to you -- that's why we try to pass out a bill as fast as we can, before something gets into the mind of somebody. But the question is asked -- I was reading this report, this staff report from the NRC, and something struck me, and that is that the Babcock & Wilcox plants have had three transients per reactor-year. And it wasn't that so much, that you've had a slightly greater transient than other plants, albeit that's something we're going to have to come to grips with.

But what really struck me was that the Westinghouse had 1.84 transients per reactor year. This is, I guess, in the last year. I can't remember. I think what it was, that Combustion Engineering had 1.83 transients per reactor-year. When I saw that 1.84 and 1.83 I said, you know -- I mean, it's amazing how in line they were, one one-hundredth of a point difference.

Are these predictable, these transients? They almost seem to be.

Mr. McMillan. Let me say this. The fact that they're 1.83 and 1.84, I would say is highly fortuitous or just by chance, and I don't believe, at least at this point, we are in a position to answer the question of why is that different from the roughly three that might reflect the B&W experience.

I would like to say this, and that is that the feedwater system, the initiating source of all the transients that you're mentioning, is a system that is designed and supplied by the utility or his engineer. So the frequency of upset conditions or loss of feedwater or transients in the secondary system is not really related to the design of the nuclear steam system. And so I wouldn't want you to draw any correlation or to reach a conclusion that because of the statistics there, there is something unique to the B&W system versus the GE or Westinghouse or Combustion.

The question of whether they're predictable or not, I guess they're predictable to this extent: that in feedwater systems in steam plant operation, that kind of frequency of loss of feedwater seems to be typical of the kind of operation that steam plants are exposed to, and they frequently are initiated by protective devices built into the system to keep from damaging equipment, because usually the condensate pumps or the booster pumps or the feedwater pumps and so on --

Mr. Weaver. Is there much difference? You build coal plants. Is there --

Mr. McMillan. Yes, sir.

Mr. Weaver. Is there much difference between the secondary system in TMI and the secondary system in a coal plant, the steam system?

Mr. McMillan. I just have to say, in general they are very similar. There is a couple of significant differences. Your boiler pressure -- that's the pressure at which you deliver steam from a boiler -- and the temperature at which the steam is delivered from a boiler are higher than the temperatures and pressures.

For example, typically a boiler might deliver 2400-pound pressure and 1,000 degrees Fahrenheit, frequently with some reheat. The steam comes back and gets reheated to 1,000 degrees. Whereas with a nuclear plant, of course, it's delivered at 1,000 pounds pressure and at temperatures in the 500 to 600-degree range.

So that the temperature and pressure range of the steam coming out and the pressure of the feedwater coming in are different in a boiling unit than they would in a nuclear unit.

Mr. Weaver. Give me those figures again, because I thought I heard the same thing for coal as nuclear, 1,000 pounds. I thought you said coal delivered at 1,000 pounds.

Mr. McMillan. In nuclear, steam is delivered at 1,000 pounds. A coal-fired plant would deliver at either 2400 or some of the units are supercritical, they get up over 3,000 pressure.

Mr. Weaver. I see. And the temperature in the two is 600?

Mr. McMillan. In the nuclear unit it would be 500 to 600 degrees, and in a boiler it would be 1,000 degrees.

Mr. Weaver. 1,000 degrees. Okay, I got you.

Now, to let you know what politicians are doing when they're out on the hustings, when I make a speech I often use this analogy, and I'd like you to criticize it, please, because I never like to say something that I think is wrong. I say with a coal plant -- and I am no more friendly to coal than I am to nuclear. I am just looking for the best way we can. I'm not trying to criticize anything over another. I don't have any coal plants in my district. That's what I'm really saying.

I say, if something goes wrong with a coal plant, take a monkey wrench out there and fix it. If something goes wrong in a nuclear plant, you can't do that.

Would you comment on that?

Mr. McMillan. I think there's no question that the element of radiation associated with the reactor coolant system in a nuclear plant complicates the maintenance or repair functions that would be required as it applies to the reactor portion of the plant. I don't think it makes much difference for the secondary system. If you've got a feedwater problem in a nuclear plant, you could go right down and stand next to that pump and work on it and maintain it with a monkey wrench, the same way you could maintain a feedwater pump on a coal --

Mr. Weaver. I know. But frankly, I don't think it's even a matter of degree. I think it's a matter of the most profound category, the difference, in that if you'd had a condensate pump go out on a coal plant you wouldn't have had this problem you had at TMI, nothing even faintly resembling it. Isn't that correct.

Mr. McMillan. I would rather address the question this way, because I don't know the answer to the question about what happens in a boiler when you lose a feedpump. It's a very traumatic transient, and not something --

Mr. Weaver. In a coal plant?

Mr. McMillan. In a coal-fired plant.

Mr. Weaver. Oh, it is? You see, I didn't know this.

Mr. McMillan. You've got to kill the fire and you've got hot slag on the walls of the furnace, and you've got to get water in there and keep that furnace cool.

Mr. Weaver. It's costly but not dangerous; would that be fair?

Mr. McMillan. When I said traumatic, I meant it's a very severe transient. It's not something that the operator handles just like that. He's got to be very careful in the way it recovers in a fossil unit.

Mr. Weaver. But you couldn't lose your plant, could you?

Mr. McMillan. If he makes a mistake in the operation or he interferes with the cooling, normal cooling process in a boiler, what can happen is you can burn through the wall of the furnace, melt the wall of the furnace and blow water and steam around.

The repair of that wall segment is an easier job than the kind of recovery operation we're faced with at Three Mile Island and the difference there is the ingredient of radiation. You can walk in with a welding torch and cut out a section of the wall and put a new one in and weld it up and hydro-test it and inspect it and so forth, with direct hands-on operations, without any having to wait for any decay of radiation.

Mr. Weaver. You'd only be down a couple months, three months?

Mr. McMillan. Two or three months, probably.

Mr. Weaver. Mr. McMillan, I want to thank you very much. I really appreciate your taking the time and being so courteous with us. Thank you very much.

Mr. McMillan. Could I just extent to the Committee an invitation, Mr. Chairman? I know these are very complicated matters and I know that it helps, from my own personal experience, to see some of these operations take place. And we certainly would be very pleased to have the Committee or any portion of the Committee or the staff come to Lynchburg and visit our simulator facility there. We can show you the facility, run through the transient as it occurred at Three Mile Island, show you some of the training operations that we have available for our operators.

And I think you would find it helpful in terms of trying to get some visual picture of what kind of an operation is involved in running these nuclear plants. We'd be delighted for you to come and visit us.

Mr. Weaver. We thank you very much, because we want very much to do that. I appreciate the opportunity. We'll try to schedule it.

Thanks again.

(Whereupon, at 1:20 p.m., the hearing was adjourned.)