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Transcript of Proceedings

UNITED STATES OF AMERICA

PRESIDENT'S COMMISSION ON THE ACCIDENT AT
THREE MILE ISLAND

DEPOSITION OF: DANIEL M. STERNBERG

Bethesda, Maryland

July 30, 1979

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PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND

DEPOSITION OF: DANIEL M. STERNBERG

Room 6211
~~Madison~~ National Bank
7735 Old Georgetown Road
Bethesda, Maryland

Maryland

July 30, 1979
11:00 o'clock a.m.

APPEARANCES:

On Behalf of NRC:

MARIAN E. MOE, ESQUIRE
Office of General Counsel, NRC
Washington, D. C.

On Behalf of the Commission:

GARY M. SIDELL, ESQUIRE
Associate Chief Counsel
2100 M Street, N. W.
Washington, D. C.

I N D E X

WITNESSES:	DIRECT	CROSS	REDIRECT	RECROSS
Daniel M. Sternberg				

E X H I B I T S

P R O C E E D I N G S

Whereupon,

DANIEL M. STERNBERG

having been first duly sworn, was called as a witness herein and was examined and testified as follows:

BY MR. SIDELL:

Q. Would you please state your name for the record, Mr. Sternberg?

A. Daniel M. Sternberg.

Q. Your current occupation?

A. Section Chief in the Reactor Operations and Nuclear Support Branch, U. S. Nuclear Regulatory Commission's Region V office, Walnut Creek, California.

Q. In March, 1978, what was your position?

A. I was acting as the Section Chief for the pressurized water reactor section of the Reactor Operations and Nuclear Support Branch in Region I of the NRC.

Q. How long were you in that position?

A. Approximately five months.

Q. To about November, 1978?

A. No; it was probably from February to July of 1978.

Q. Are you acquainted with an individual by the name of D. Haverkamp?

A. Yes. Don Haverkamp was the project inspector for the Three Mile Island site in the section that I was the

supervisor of.

Q At what particular time?

A During the whole period of time that I was acting Section Chief.

Q Do you know if he was still in that section when you stopped being acting Section Chief?

A Yes, he was.

Q Have you ever had your deposition taken before?

A No. Not in this context.

Q At all?

A I have given testimony in public hearings for licensing things.

Q Let me briefly explain what we are going to be doing.

Your testimony is under oath, it is sworn, even though we are sitting in a conference room at the NRC Building in Bethesda. Your testimony has the same effect as though it were given in a Court before a Judge or jury, so that it is imperative that you be as precise and accurate in your responses to my questions as you can be.

If you don't understand a question, ask me for clarification and I will try to explain what I am looking for. At the end of the deposition the reporter will transcribe your testimony and we will present a copy of it to you for correction and for you to sign.

If you find places that you feel it necessary to make corrections or changes and we consider those changes to be of a substantial nature, we are entitled to comment on those changes and our comments may adversely affect your credibility, so again the emphasis is to be as precise and accurate as you can be now.

Since the reporter is taking the testimony down, I would ask that you answer audibly all the questions, since it is not possible to take down nods of the head or gestures. I also ask that you wait until I finish a question before you respond, even if you know where the question is going, and I also will make every effort to try and refrain myself from asking the next question until you finish your response.

Do you have any questions?

A. No.

Q. The resume that you provided to me -- is this an accurate description of your professional background?

A. Yes, it is.

Q. This resume was made for the purpose of this deposition, is that correct?

A. Yes, it was.

MR. SIDELL: Let's have this marked as Exhibit 1, please.

(The document referred to was marked for identification as Exhibit 1.)

BY MR. SIDELL:

Q. Mr. Sternberg, according to Exhibit 1 to this deposition, your resume, there is a listing of 1978 to the present which specifies your current position. There is no specification that deals with your position in Region I, is there?

A. You mean relative to the acting Section Chief?

Q. Yes.

A. Yes. That is because there is no official paperwork written changing your job description when you are appointed as an acting Section Chief so my position at all times according to the official forms was that of a reactor inspector.

However, I had been relieved of inspection responsibility and assigned to act as the Section Chief because of the promotion of Bert Davis, who had been the Section Chief.

Q. So with that caveat, this resume fully states your prior employment to 1964?

A. Yes, it does.

Q. But on the resume itself there is no indication that you were an acting Section Chief?

A. That is correct. There were other periods of time where I may have acted in the same capacity.

Q. For purposes of your deposition we are primarily concerned with your acting Section Chief.

A. Okay.

Q. On March 29, 1978, did you happen to become aware of an incident at Three Mile Island unit two?

A. Yes, I did.

Q. How did that come about?

A. There was probably a telephone call from the licensee or from Don Haverkamp if he were on the site, I don't recall at this point, but we became aware that there had been a reportable event involving a safety injection, perturbation of reactor coolant system chemistry due to the injection of sodium hydroxide, and we initiated an inspection or an investigation of the event to determine what had happened, what were the causes and what the licensee was doing about it.

Q. What did you find out to be the causes of the accident?

A. Well, the main thing we were concerned with at that point was reactor coolant system chemistry and what I found was that there had been an inadvertent or undesired safety injection and that had been produced by a depressurization of the reactor coolant system.

That in turn had been caused by an opening of the PORV which had been produced by a loss of vital bus which had been in turn produced by the failure of an electrical inverter.

Q. In layman's terms is that essentially a situation where the electrical system was set up incorrectly?

In other words, the PORV in order to remain closed had to have a continuous source of energy?

A. Indirectly, that is true. The bi-stable which controlled the PORV was designed to de-energize on increasing pressure. The increasing pressure was the parameter that would normally open the valve or require it to open.

That meant that anything which de-energizes that bi-stable, a loss of power or failure of the control relay would give a signal to open the PORV, and that was the as built design for that component, so to say whether it was right or wrong, I don't know at that point.

Q. Well, did you subsequently come to a conclusion as to whether or not that was a proper method to install a PORV system?

A. Well, I had two concerns about the PORV at that point. One was that it would fail open or a loss of power to that bus and the other one was that it was unenunciated.

Q. Who do you mean by "unenunciated'?

A. That was not a form of indication on the control

panel directly telling the operator that that valve was being commanded open.

Q There was no direct position actuation switch on the PORV?

A Not only that, there wasn't any other kind of indication saying that it was electrically being commanded open.

Q So there was no indication at all?

A No direct indication.

Q Was there an indirect indication?

A The parameters that would be disbursed by the valve being opened, the pressurizer level --

Q So there were no direct methods of indication that there was a problem with the PORV?

A That it was open.

Q There were no indirect methods determining that the PORV was open in terms of a light or a warning signal tied directly to the PORV?

A That is correct.

Q In other words there were no indirect methods of actuation where an energy signal was sent to the solenoid that would show up in a control room?

A That is correct.

Q What are the indirect parameters that the operator would have had available to determine whether there was a

problem with the PORV?

A. Decreasing reactor coolant system pressure, a change in pressurizer level, the --

Q. Pressurizer level indication?

A. Yes.

Q. Which direction would that go, up or down?

A. I would assume it would go down, because I would be losing reactor coolant system inventory through the PORV.

Q. Do you know of a case where it would go up?

A. Subsequent events have given sight as to which way pressurizer level would behave.

Q. Which subsequent events?

A. The failure of the actuator on March 29. Also the Michelson Report.

Q. Do you recall when you read the Michelson Report?

A. In April, 1979.

Q. After the accident?

A. After the accident.

Q. Have you also had an opportunity to read what is referred to as the NOVAK or Israel memorandum?

A. No, I have not.

Q. To date you have not seen that?

A. No.

Q. Please continue.

A. Okay.

The other parameters that would indicate an open pilot operated relief valve would be the quench tank or drain tank temperature level and pressure.

Q. Is that the temperature on the tail pipe?

A. The tail pipe temperature also, depending on what the previous temperature was, it may or may not respond.

Q. Do you know what at the time was considered to be a normal operating temperature at the tail pipe for Three Mile Island?

A. No, I do not.

Q. Are you aware of it now?

A. I am aware from reading that the normal temperature was below 130 degrees and that at times it read up to 190 degrees.

Q. Would you consider a reading of 190 degrees to be abnormal?

A. It would indicative of a problem that ought to be understood.

Q. What would the cut off point be in terms of temperature increase above 130 degrees to indicate that the operator should start checking some systems to see if there was a problem?

A. I don't think there is an absolute number. I think whenever you find a reading that isn't what you expect it to be you investigate the cause. In this case you might assume

that it would be weeping PORV or a safety valve and then you might look at things like the level in the quench tank to see how frequently it had to be pumped to get an indication of how much flow existed through the possible leaking valve.

Q. Well, would 150 degrees be sufficient to arouse an operator's curiosity as to whether there was a problem in the system?

A. I don't have any absolute numbers. The reason that you don't know is each particular valve has a given ambient condition that exists in each thermocouple and you would have to know what was a normal condition and whether or not it was deviating from the normal, so that absolute numbers I wouldn't be familiar with.

Q. If an operator was instructed that the normal range of temperature was let's say 120 to 130 degrees and he noticed a reading of 150 to 160, would that be sufficient to cause him to inquire as to what the result of that 20 to 30 degree temperature rise was for?

A. Yes, but I have a feeling that what they would do though is look at the other parameters that would give an indication as to what is the cause of that elevated temperature.

Once you master that, then you have to decide what action to take.

Q. Well, one of the other parameters you mentioned was the level indication on the pressurizer, correct?

A. Yes.

Q. You said that you would expect the level indication to decrease if the PORV failed open, correct?

A. If it were open, yes.

Q. Well, in light of Three Mile Island in March of 1979, we know that the pressurizer level indication can also go up as well as down with the PORV being open, correct?

A. That is correct.

Q. So that if the operator noticed 150 to 160 degree range in temperatures from the tail pipe and also noticed the level indication of the pressurizer going up rather than down, would that be sufficient to create some concern on the part of the operator?

A. I am confused now whether you are talking about the accident condition at Three Mile Island or a plant that is operating with everything else apparently normal.

Q. Everything apparently normal, but knowing as we do that the pressurizer level indication can go in two directions.

A. Well, I think it is important that 150 degrees would be indicative of a small weep or a small leak and you would expect temperatures much higher than that ~~or~~ on an actual opening of the power operator relief valve.

Q. How much higher?

A. I assume in excess of 212 degrees, because the tail

pipe is venting to atmospheric pressure and therefore I would expect that you would get temperatures in excess of saturation temperature for atmospheric conditions.

Namely, you know, in excess of 210 or 220 degrees.

Q. So if an operator had readings of 220 or 230 degrees, in your opinion would that be virtually conclusive evidence he had a problem with the PORV or another small break above the pressurizer?

A. It would indicate that the valve had probably recently passed a steam flow. The situation is, though, that once that valve opens you would expect that tail pipe temperature to hang up for some extended period of time depending on what the ambient temperatures and cooling was in the vicinity of the valve.

Q. If the temperatures were on the order of 210 or 220 degrees and it was merely one release of steam and the valve did properly close, what length of time are we talking about before you would recognize that or notice that with the drop in the temperatures?

A. A couple of hours.

Q. That long?

A. From my experience in boiling water reactors and high tail pipe temperatures, yes.

Q. How is a boiling water reactor's tail pipe temperature similar?

A. They have power operator relief valves on boiling water reactors with tail pipe instrumentation. I don't know whether it is a strap on or a weld thermocouple, but the idea of looking at tail pipe temperatures, it is a good case of the initiation of the event, but it doesn't tell you a lot about the course of the events.

Q. In addition to the tail pipe temperature and the pressurizer level indication I believe you also mentioned pressure in the tail pipe system as falling if the PORV was open?

A. Yes.

Q. How would that manifest itself to the operator?

A. There are low pressure alarms, pressurizer heaters that would come on as pressure decreased. If the pressure continued to decrease there would be a safety injection initiation.

Q. Sixteen hundred pounds decreasing?

A. At this plant, yes.

Q. I believe you also mentioned the level indication on the quench tank would be indicative of a problem with the PORV.

Do you know where the indication for the quench tank is in Three Mile Island in the control room of unit two?

A. I have learned subsequent to the accident that it is behind the front panel. It is not a front panel parameter.

Q. Is the location of the indicator for the quench tank level in a conveniently located place so that in an emergency the operator could instantaneously determine what was going on in the quench tank?

A. I don't really know the answer to that, convenient or not. Given the fact that there was the accident and --

Q. Let's specifically deal with the indication level first of all.

If you were an operator and you had the options of checking the tail pipe temperatures, secondly the pressurizer level indication, thirdly the pressure in the primary system, and fourthly the indication level of the quench tank, is there anything else you could check to determine if there was a PORV problem?

A. Well, there is a pressure in the quench tank.

Q. Where is that indication located?

A. As I understand it is also on the back panel near the quench tank level and temperature indicators.

Q. So all the quench tank indicators are on the back of a panel, is that your understanding?

A. Yes.

Q. At Three Mile Island unit two?

A. Unit two, yes.

Q. Do you know the physical location of the other indicators that we have just mentioned?

A. No. I have never been in Three Mile Island unit two's control room, so I am not familiar with the lay out there.

Q. Would it be reasonable to presume that all the other indicators are facing the operator?

A. For example, pressurizer level --

Q. Pressurizer level, pressurizer heaters, primary system pressure.

A. Yes.

Q. Tail pipe temperature?

A. I don't know about tail pipe temperature.

Q. That may be off in a different area?

A. It might well be, yes. It may be on a back panel.

Q. How would you classify the temperature and pressure and level indication on the quench tank in terms of their importance in providing relevant information to an operator during an emergency?

A. I would not classify them as significant.

Q. Would they be less significant or more significant than tail pipe temperature, for instance?

A. Probably more significant in that tail pipe temperature you would not expect to be meaningful if you had an expected operation of the PORV. For example, in the Three Mile Island event the PORV did open. It was known since you

have a high pressure SCRAM that clearly the PORV had opened at some point.

Once it opened, as you can expect your tail pipe temperature will be elevated and remain elevated for some extended period of time at which point if indeed your concern was how to detect the failed open PORV, the tail pipe temperature would not be a particularly good indicator.

From my reading of the sequence of events to date, I do not understand the temperature readings that tended to indicate that the tail pipe temperature was coming down during the time that the PORV was open.

I believe the first time the operator requested the PORV tail pipe temperature he got a number around 285 degrees, then 283 and sometime later, 223 -- an hour or so later it was down to 223 and it was open.

Q 223 is still above your general range of temperatures providing concern for the operator, is it not?

A Yes, but we are talking about the trend of the data. ~~The~~ That was trending down.

Q Trending down, but nevertheless still in an area or range that should have been sufficient to awaken the operator's interest on why it was still at that high level?

A I think the fact that the operator requested the temperature printout indicated the operator was interested in what was going on. To me the significant thing, you would

have to compare the rate of decrease of tail pipe temperature to some other given event in the past.

For example, when the PORV had opened and did anybody plot the rate of temperature decrease so you could get a normal rate of temperature decrease, and even that is not too meaningful in ~~that~~ that you don't know how long it was open in the past compared to this event.

Q We have established that quench tank indication level or temperature or pressure are more significant than tail pipe temperature.

What about relative to pressurizer indication level? Which is more important in your opinion?

A I would think pressurizer level.

Q Even having read the Michelson Report?

A The interesting thing about the Michelson Report is that that particular phenomena would not have occurred in the March, 1978 event at Three Mile Island in that it was a relatively clear core, a little bit if any decay heat inventory.

The potential for flashing and voiding in a major cooling system did not exist during this event, so that the course of the event in 1978 would be totally different than the course of the event in 1979.

Q Would this be due to the fact that the operating license was issued four or five weeks before the accident at

TMI-II?

A. Yes, and the fact that they were still undergoing low power seismic testing during that time.

Q. Do you know the range of power operation they were at?

A. I believe it was less than one percent, essentially zero power.

Q. Do you have any opinions as to a threshold level of production where you would run into problems similar to those mentioned by Mr. Michelson in his report -- in other words, is nine percent above or below a problem level?

A. I think that the essential parameter is not exactly the power level you are operating at but the power history. For example, if you would just come with a clear core up to nine percent power, I don't think there is enough decay heat to produce that kind of voiding.

The fact is you could have significant decay heat and only be at nine percent power and have the problem, which is now understood was the root cause of the Three Mile Island event.

Q. It is not only a question of the current level of production of power, but also from which direction the plant came?

A. What we refer to as decay heat history or inventory.

Q. So that a plant just starting up and working its

way toward nine percent would have a much less probability of running into problems raised by Mr. Michelson than a plant coming down from nine percent -- coming down from something above nine percent?

A. Well, more importantly, it is what the power history for the facility was and what percent rate of power you might refer to the decay heat at the time you would get a SCRAM and a low pressure condition.

Q. What about quench tank indication levels and temperature and pressure relative to reactor pressure?

Which would be more important in terms of indicating a possible PORV problem there?

A. I think the quench tank.

Q. So the four indicators, we have established that pressurizer level indication is relatively more important than quench tank indicators, correct?

A. Could you say that again, please.

Q. We have established that pressurizer level indication is more important in terms of informing the operator of a possible PORV problem than is the temperature or pressure level indication on the quench tank?

A. I hesitate to generalize.

Q. More useful if not more important.

A. I have a feeling that under some conditions one would be much more useful than the other, but I can't feel

comfortable in generalizing.

Q In any event, the temperature and pressure level indication on the quench tank is, in your opinion, a rather significant or useful piece of information for the operator to have during a potential problem with the reactor, is it not?

A It is one of the many pieces of information which he can refer to, yes.

Q You have not become aware that this information is on the back of a control room indication panel?

A Yes.

Q In a less than convenient location relative to the other instrument displays at TMI-II?

A I hesitate to use the word "convenient". There is only so much panel space available.

Q Well, for instance, in an emergency, based on what you have indicated about the quench tank readings and their importance in assisting the operator in determining if there is a problem and what the problem might be, the operator would have to turn around behind the back of the control panel to get the data for this particular indication, whereas for reactor pressure, that is on the face of the panel.

Tail pipe temperature is on the face of the panel. Pressurizer level indication is on the face of the panel. Would it not seem as though an operator trying to resolve a

problem wher ime is a factor would find it less convenient to check an indication on the back of a panel than one staring him in the face?

A. He very well might. There is another aspect of that. That gets involved with the number of operators in the control room and the NRC's requirement that operators do not go around to the back panels if they are the only operator in the control room.

Q. There is a NRC requirement that prohibits --

A. Going around to the back if they are the only operator in the control room.

Q. Do you know what that requirement is?

A. No. There is some guidance in the IE manual. I think it is 10 CFR50.34, but I am not sure.

(Pause in proceedings.)

THE WITNESS: An operator or senior operator licensed pursuant to part 55 of this chapter shall be present at the controls at all times during the operation of the facility.

BY MR. SIDELL:

Q What was the CFR designation of the section?

A 10-CFR-50.54-K.

Q Is there any NRC interpretation as to what "present at the controls of a reactor" means? In other words, does that mean in front of the control panel rather than behind it?

A Yes. One that I am sure of is IE chapter 71710 B, although I don't have it here. There may have well been some interpretation that under emergency conditions does permit the operator to leave, but under routine conditions nobody -- the panel cannot be without an operator in front of it.

So, the question about the back panel presentation -- it is one aspect that the operator may if he doesn't feel an emergency condition exists, if he were there by himself might not go around to a back panel unless he felt there was a compelling reason.

Q In view of the importance that you have established that the quench tank indicators have in determining whether or not an emergency exists it would be to say the least rather

difficult for the operator who is required to be in front of the control pane' to look at important indicators behind the panel to determine if there is an emergency, is that not correct.

A Yes.

Q Catch 22, in other words?

A Well, if a real emergency exists, I believe that the guidance is rather clear.

Q Are you aware of whether or not the NRC has any part in approving control ~~program~~ room design?

A I am not aware.

Q One way or the other?

A No.

Q Region one didn't have any involvement in the TMI-2 control room design, to the best of your knowledge?

A Not at all. Can we go off the record for a second?

(Discussion off the record.)

MR. SIDELL: On the record.

BY MR. SIDELL:

Q Mr. Sternberg, we just had a brief discussion off the record of some prior longstanding concerns you have had with control room design. Can you indicate for the record what those concerns were?

A During the period when I was a reactor inspector

S:amapad2:3

I inspected primarily boiling water reactors. As a result of inspecting certain incidents which had occurred at boiling water reactors, problems with control room panel layouts, which I believe contributed to those incidents had arisen and I had written a memo in probably 1976 or 1977 requesting that the design adequacies and the human factors analysis be initialed for control room panel layouts of boiling water reactors.

Q Can you recall whether that memo was sent to Region one or headquarters?

A To headquarters from Region one.

Q From your recollection can you remember the time, approximately?

A No, I can't. I believe it followed an inadvertent critical incident from Millstone I.

Q Do you recall to whom you sent the memo in headquarters?

A No.

Q Do you remember whether there was any follow up by headquarters to your concern?

A I don't believe there was.

Q Would you have been aware of follow ups if they existed?

A If it was after the time when the action item tracking system was in effect, there would have been a follow

up memo. I don't recall.

Q You previously indicated that you became aware in one form or another of the TMI-2 incident in March of 1978. What steps did you take in determining the precise problem and resolving that problem?

A Well, again, it was really the follow up on what the licensee was doing. Our primary concern was the possibility of potential corrosion problems or radiological problems from the injection of the sodium hydroxide. We knew that the licensee had gotten in contact with B and W.

Q How did you know that?

A Through conversations on the telephone with licensee.

Q You spoke with someone at TMI-2?

A Yes.

Q Do you recall who that was?

A I am not sure. I think it was Jim Seelinger.

Q Who was he at the time, if you recall?

A I think he had the capacity of technical engineer or plant engineer.

Q And he indicated to you that the utility had been in touch with B and W?

A Right. I believe there had been one or maybe two previous injections of sodium hydroxide similar in character to this same thing through inadvertant safety injection, and

we had ~~explored~~ expressed our concerns at that time with them and they were in the process of cleaning up the reactor cooling system. We had also expressed our concerns over the design that would inject sodium hydroxide into the reactor coolant system which was clearly not necessary.

Q Was it not becoming necessary to have sodium hydroxide injected into the system and determination made during the emergency or the transient or looking back?

A It was never intended to go into the reactor cooling system. It was designed for reactor building container spray. It turns out that the way that that chemical attitude gets into the proper pump, the potential for injecting it into the reactor cooling system existed at that time.

As I understand it was subsequently changed so that everytime there is a safety injection you will not inject sodium hydroxide into the reactor cooling system.

Q What corrective measures if any did the utility take after the March 1978 incident?

A Well, again, the concerns which I relayed to them had to do with the sodium hydroxide injections, the failure of the PORV in the ~~opposition~~ open position, the lack of any indication on the PORV, and there may have been something else related to why the business was lost with a single inverter failure.

I don't recall at this point, but as a result the

licensee rearranged the electrical configuration of the high pressure bi-stable. The added indication, electrical indication, a light on the control panel to show that the valve was being commanded open.

Q Not an actual position of the valve?

A Not an actual position of the valve.

Q Just to show that a signal was sent to close the valve?

A More importantly that it was open. In other words, whenever the solenoid was energized on the PORV an indicating light would illuminate on the control panel and relative to the sodium hydroxide, the initiating signal that would open the sodium hydroxide additional valve was changed from a safety injection to another paramater which more -- which related more to the condition that would require the sodium hydroxide, either high containment ~~container~~ pressure or something along those lines, higher than which initiates safety inject injection.

Q The change in the initiating criteria for sodium --

A Hydroxide addition.

Q Do you know at the time whether they considered changing the containment isolation factors?

A I don't think that ever came up, containment isolation.

Q Should that have come up if they were considering the sodium hydroxide --

A Not really. I don't think there is a direct connection at all.

Q Was it your opinion at the time of the March 1978 accident that the problem with the PORV at TMI-2 could be considered a generic problem to B and W reactors?

A Which aspect of it?

Q For instance, having no indication.

A Yes. I believe that the potential did exist that it was probably part of the design supplied by B and W and therefore it was potentially generic.

Q So as far as you know, all other B and W reactors are built in the same way as affects the PORV and the energy source?

A Well, no. I don't know whether or not it was similar.

Q At the time -- in March 1978?

A Yes, I did not know.

Q Do you know now?

A I suspect they are not the same. I, subsequent to your interview with me, made some calls around and found out at least one plant that is energized to open.

Q Which is contrary to the original set up at TMI-2?

A Yes.

Q Which plant is that?

A Arkansas Nuclear 1.

Q Well, that leaves about six other reactors.

A Yes.

Q Is it your feeling that they would have PORV's energized on the basis of the way TMI-2 originally was or the way Arkansas was?

A I really don't know at this point.

Q It is entirely possible they could have the same set up as TMI-2?

A It is probable.

Q Did you follow up in any way in terms of your concerns about the generic applicability of the ~~probabili~~ problem with PORV?

A Yes.

Q How?

A The event occurred on March 29, 1978. We informed the headquarters the next day by a preliminary notification, and at the same time I wrote that I wrote a memo to IE headquarters requesting the design adequacy of a PORV failing in the ~~opposition~~ open position.

We looked at Three Mile Island and all ~~over~~ other B and W plants. I believe that was sent out the day after on March 31, 1978.

Q Let me show you a memorandum addressed to Seyfrit, Assistant Director Technical Programs, IE, dated March 31, 1978, by Daniel M. Sternberg, Acting Chief, Reactor project

section number 1 and ask if you have seen this before?

A Yes, I have.

Q Is this the memo you just made reference to?

A Yes, it is.

MR. SIDELL: Let's mark this Deposition Exhibit 2 to the deposition, please?

(Whereupon, the document referred to was marked as Sternberg Deposition Exhibit 2, for identification.)

BY MR. SIDELL:

Q Mr. Sternberg, what is the second page attached to what has now been marked as Exhibit 2 to this deposition?

A That is an action item control form, which is a part of a system that I & E uses to insure that correspondence which is sent into headquarters or from headquarters requiring action actually gets its action accomplished and it is a tracking system.

Q So, this tells what the problem is and who sent it and who received it and what was done, is that correct?

A Yes.

Q Is there an indication on the action item control form as to someone receiving your concerns?

A Yes. The action office has a person assigned.

Q And who is that person?

A Roger Woodruff.

Q That is indicated on the action item control form which is attached to exhibit 2 of this deposition. It appears as though he completed or he sought to complete action by 5-8-78 and there is a separate section termed action taken with an action date of 5-3-78 and can you see the closeout action?

A Yes. It was a memo to Brunner, who was my boss, my branch chief at the time while I was acting.

Q Let me show you a memo for E.J. Brunner, Chief Reactor Operations and Nuclear Support Branch, Region one operations, dated May 3, 1978 from Karl V. Seyfrit, Assistant Director for Technical Programs, Division of Reactors operations Inspection. Have you seen that previously?

A Yes.

Q Is this the memo referred to as the action taken by headquarters of Bethesda of the NRC?

A Yes, it is.

Q And in this memorandum what does it appear the result of any I & E headquarters analysis was?

A That they reviewed it and discovered that this type of failure had been studied in the FSAR for Three Mile Island-2.

Q Is there an occasion in the May 3rd -- strike that.

MR. SIDELL: Let's have this marked as Exhibit 3.

MS:ama:11:p2

(Whereupon, the document referred to was marked as Sternberg Deposition Exhibit 3, for identification.)

BY MR. SIDELL:

Q On exhibit 3, to this deposition is there an indication that Mr. Seyfrit's office actually reviewed your concerns in your March 31, 1978 memo, Exhibit 2 to this deposition?

A I believe they did. There is a statement, their request is based on failure of the valve in the ~~opposition~~ open position, which was certainly my concern.

Q What was stated in Exhibit 2, your memo to Mr. Seyfrit?

A Yes, in that I stated that the blowdown in the March 29, 1978 ~~has~~ had been caused by the pressurizer relief valve opening on a loss of electrical power.

Q So, they could have written Exhibit 3, the May 3, 1978 memo, merely on the basis of the memo provided to them?

A Well, somebody clearly researched it at a point where they discovered it was discussed at a point in the FSAR.

Q And they concluded that official review is not warranted, is that correct?

A Yes.

Q Which closed out the concern you raised in your March 31, 1978 memorandum? Is that correct?

MS:ama:12:p2

A Yes.

Q Did you feel that was a proper conclusion?

A Yes. Now I will elaborate.

Q Please do.

A My view of the role of the inspector and the regional management is to bring to IE and NRR management's attention conditions which are discovered in the field during routine inspections during investigations of incidents or accidents, and I feel that the role of IE headquarters is basically to bring a bold perspective on these findings and to take appropriate action when they deem necessary.

I was aware also that NRR knew about the failure of the valve in the ~~opposition~~ open position because I had discussed it with the licensing project manager at the time of the event back in March 1978.

Q Who was that?

A I believe it was Harley Silver.

Q Were these conversations over the phone or --

A Yes, over the telephone.

Q Was there any written memoranda of these conversations?

A We were working rather closely during this period of time in that the plant had just been licensed and there had been a safety injection system to this before the licensing and we had looked at it in depth then and I was

staying very closely in contact with the licensing project manager then just to make sure that he was aware of things that were going on at the plant directly.

Although, there are other mechanisms for him getting that information, the most timely is a telephone call.

Q So, the final resolutions of the problems was for the licensee, at TMI-2 or Metropolitan Edison in this case to install a light which indicated that a signal was sent to the ~~cellunoid~~ solenoid for the PORV, but not an actual position indicator for the PORV, is that correct?

A That is correct.

1S:ama: pad3:1

Q And headquarters of I & E in Bethesda found no generic concerns with this problem?

A Yes. I think, getting back to my statement that I thought the response from headquarters was acceptable, I think in part it was colored by the fact that the licensing at Three Mile Island II had what I considered to be corrective actions which in my mind improved safety and tended to reduce the probability of an similar occurrence at that facility.

Q Dealing with the PORV, are you referring to the installation of the warning light for the fact that a signal was sent to the solenoid?

A That and the reconfiguration of the by-stable so it would energize to open the valve rather than deenergize to open the valve.

Q Were these specifics of the solution known by I & E headquarters when they reviewed your memorandum?

A I don't know. I am trying to remember whether or not that was discussed in inspection report. It may be that it was discussed by the licensee in his follow up licensee report of this event.

Q In other words, the licensee would have been under a duty to inform the NRC of the precise problem as well as its solution?

A That is correct. An official form called a

S:ama:2:p3

licensee event report is submitted. I don't have that in front of me, but one was submitted for this event.

Q Would that have been because this would have been a violation of their technical specs?

A It is interesting. I do not recall what particular aspect of this whole event in March of 1978 made the occurrence reportable. I think, interestingly, that what it was was the reactor coolant system chloride exceeded the technical spec limit. I believe that may have been the only aspect of the event which was reportable by the current technical specs.

Q Based on only one aspect of an event being reportable is the licensee required to report everything that goes on during the event?

A No.

Q So, it is entirely possible that if the reportable part of the March 1978 event at TMI-II did not deal with the PORV problem, that never would have come to light?

A Well, it might not have come to light --

Q In a LER?

A Yes, in a LER. During our records review and log review, clearly some portion of it would have certainly surfaced. Additionally, the licensee does inform the regional offices of significant events.

For example, I would think that prior to licensing

the safety injection which had occurred it would not have been reportable in any way by the current reporting requirements, but we were certainly aware of it and worked very closely in reviewing the events then that preceeded this March 28 or March 29, 1978 event.

Q Do you know whether at the time of the March 1978 problem at TMI-II the PORV was considered to be safety related or not?

A I reviewed it at the time, and I concluded from the review I had done that it was not safety related and I think I indicated that in the March 31 memo, that the release valve does not appear to be a safety related component. Paragraph two.

Q And if you know, was that the reason why the PORV would not have been the basis for a LER?

A Yes. If it were not safety related -- in fact, the PORV did not malfunction in the March 29, 1978 event. It performed exactly as you would expect it to perform. When it got a signal to open it opened, and when that signal went away it closed. The PORV performed normally.

As a matter of fact, that whole control system performed as designed, namely when the bi-stable deenergized, it opened the valve, which is what you would expect.

Q But with the basic problem that although the mechanical parts operated as they should have, there was

insufficient indication to allow the human component, in this case the operator, to know what to do when?

A Well, in the March 1978 event -- as I recall, my impression at the time was that the operators were not aware that the reactor coolant system was depressurizing and that there attentions were being called to other things that were being produced by the loss of the inverter on that bus.

I think there attentions were towards getting the bus reenergized.

Q Are you aware of what the first indication was that advised or indicated to the operators they had a problem with the PORV?

A As I recall, based on telephone discussions and I don't know whether it was with the licensee or our inspector the first indication that they had anything involving the coolant system was the initiation of the safety injection at 1600 pounds decreasing.

Q So, the first time as far as you can recall the operator knew there was a problem with the PORV failing open or at least open without any indication of that being the case, was ~~what~~ that the ECCS system came on at 1600 PSI?

A That is my recollection.

Q And that is a relatively serious event, is it not?

A What aspect are you trying to emphasize the seriousness of?

Q That is almost the last safety factor that comes

MS:ama:5:p3

into play if there is a problem with the reactor?

A Well, --

Q Alternatively you miss quite a few things up to the point of the 1600 PSI.

A That is what I am saying. I do not recall what other paramaters responded. As I remember, there were other instruments being supplied by the inverter that had failed, in other words pressurizer level and pressurizer pressure, which may have been two paramaters that would have given the operator some indication.

They may not have responded or may have responded and thought it was part of the spurious problem. I think not all of the operators were aware what instrument they may lose or might give anomalous indications for the loss of that inverter.

Q So, as a result of the problem with the electrical set up and the PORV, they also lost pressurizer level indication? Did you also say they lost reactor pressure indication?

A I think pressurizer pressure.

Q Which is another mechanism an operator can use to determine what is going on in the core?

A Yes.

Q So they were essentially left with tail pipe temperature and reactor pressure and quench tank indications to

1S:ama:6:p3

determine what was going on?

A And I don't know whether or not they were supplied by that same inverter. I do not know what was supplied by it.

Q Can you recall whether or not the operator did in fact have reliable quench tank indications readings?

A It never came up. The valve as I recall was open less than five minutes until they got the inverter reenergized and as soon as they did the reactor coolant system pressure started to respond to the high pressure injection pumps and everything turned around from there.

Q So, it is conceivable that the most important indicator they had during this transient was the quench tank in terms of level and temperature and pressure indications?

A Well, yes, if you look at the problem as being the PORV being on. But as I said before, our concern at the time was not the PORV but the injection of sodium hydroxide.

Q Let me show you an exceprt from a report, a periodic report which deals with the March 9, 1978 problem at TMI and ask you if you have ever seen this before.

A Yes, I have.

Q What is this report from, if you know?

A It is from a publication that the NRC subscribes to called Nuclear Power Experiences, which is a private publication that indexes and cross references reports from licensees and other sources to discuss various problems which

have occurred in nuclear power facilities.

Q Is this report available to anyone other than the NRC?

A I do not know. I assume that other people subscribe to it. I don't know whether somebody from the general public could get a copy of it.

Q In terms of other operating reactors, would it be likely they would subscribe to this publication?

A Some may, but there are other mechanisms whereas licensees can exchange event reports. I think the Edison Institute has a system where each licensees reports are routed to other licensees.

Q How does the description of the March 1978 TMI-II problem in this publication refer to -- for instance, that required by the NRC?

A I don't have the licensee report in front of me, but it appears to be consistent with my recollection of what was in the LER from Three Mile Island - II.

Q Does it appear to be more information, less information or about the same, if you can recall?

A Well, it probably has more information than the licensee event report, which is a form apparently used for computer input. As I remember, Three Mile Island II also would attach another letter to their LER submittal which was several pages long and would probably go into great

detail or comparable detail to that which is here in front of me now.

Q But this excerpt from the periodical you mentioned provides sufficient detail to alert any other reactors who might subscribe to this service, is that correct?

A Yes.

MR. SIDELL: Let's have this marked as Exhibit 4.

(Whereupon, the document referred to was marked as Sternberg Deposition Exhibit 4, for identification.)

BY MR. SIDELL:

Q Do you recall the date of this specific excerpt, Mr. Sternberg?

A I think it is on the bottom of the page there, April '78 .

Q I think it might be August '78.

A Yes, August '78.

Q So that before the accident we are now investigating at TMI-II in March of 1979, several months before, the public being defined as other operating reactors had readily available information about the specific problem at TMI-II in March of 1978?

A Well, the public certainly did in that the licensee report and the correspondence in our inspection reports following the event are all publicly recorded and used in the public document room.

MS:ama:9:pad3

Q When I use the term public it is of a more limited nature, defined in this context as being other operating reactors who may subscribe to it.

A Or the LER exchange program. I am not familiar with the details of that -- I also know that B&W was aware of the event in that I knew they were at the least aware of the sodium hydroxide injection initiation and therefore I felt that the NSSS was aware of this.

I don't know to what extent they may have investigated the electrical aspect of it.

Q Who are you referring to when you say NSSS?

A Well, the company that supplied the nuclear reactor and the steam system -- Babcock and Wilcox.

Q Were they specifically aware, to you knowledge, of the precise problem with the PORV?

A I have no idea. That is what I said. I know they were involved in reviewing the event and I would assume, but I don't know. The initiating event.

Q It would be likely for them to be knowledgeable about the problem of the PORV, since they were reviewing the entire event?

A I would expect it to certainly be likely, yes.

Q It would certainly be in their interest to find out about all the problems, whether or not reportable to the NRC, ~~NRC~~, would it not?

A Yes.

Q Are you aware of any analysis or follow up by B&W in this regard?

A I know they performed an analysis of the chemical aspect of the event.

Q So you did not see a notice from B&W to other operating reactors indicating suggestions to change the electronic system of the PORV?

A No, I am not aware of it.

Q. If there was such a notice, would you have been aware of it in your position as acting Section Chief in Region I?

A. I don't think so. I don't think NRC at that point in time would have been made aware of it necessarily. That was before 10 CFR Part 21 reporting requirements were required, and it might have come out on something similar to what GE has in the surface information letter.

Q. So it would have been possible for a vendor such as B&W to remedy what may have been generic problems to all of its plants for matters that were not deemed safety related and therefore would not have shown up in LER's, is that correct?

A. Yes.

Q. They would not necessarily have notified the NRC of any of these changes or corrections?

A. That is correct.

Q. As far as you know, this was the accepted NRC procedure at the time?

A. Yes.

Q. Do you know whether that NRC procedure has been changed since the accident in March of 1979 at TMI-II?

A. I think not, if you put on the caveat that it is not safety related.

Q. Are you aware of any change in definition of what is or is not safety related as a result of the accident in

TMI-II of this year?

A. There is no change that I am aware of. There are studies going on, but there is no official change that I am aware of.

Q. You are referring to the Lessons Learned Study?

A. Among others.

Q. Does I&E have its own separate study?

A. Yes.

Q. Your background is primarily, if not exclusively in electrical engineering, is that correct, Mr. Sternberg?

A. My college degree is in electrical engineering. However, my Navy training in nuclear power covered all areas of nuclear thermal hydraulics, chemistry, core physics, materials, accident analysis.

Q. When was it if you recall that you first saw the Michelson Report?

A. In April of 1979.

Q. Was that a typed version or handwritten?

A. Yes, typed.

Q. Did you read the entire report?

A. Yes, I did.

Q. How many times?

A. Once.

Q. Did you understand it?

A. Certain portions I was comfortable with, other

portions I was impressed with but didn't understand.

Q. Which portions were you comfortable with, if you can recall?

A. Well, I was reading it in light of the Three Mile Island accident and I gathered that the possibility for the pressurizer lever to increase in steam space brakes in the pressurizer was discussed there.

Q. In view of the fact that you read the report after the accident at Three Mile Island and knowing some of the essential facts that occurred during the accident, would you conclude that the Michelson Report or at least those sections you feel comfortable with was an accurate conclusion of what went on at Three Mile Island in March of 1979?

A. I don't really feel qualified. It seemed like the Michelson Report predicted a potential and that similar events occurred. Whether or not the report explains accurately what went on I don't know.

There could be several explanations for what was observed. That may not necessarily be a correct one, but it certainly seemed to fit.

Q. Well, would you say that the Michelson Report had it been reviewed in light of its significance before the accident would have been successful in preventing the accident?

Based on all the facts you now know that occurred

at Three Mile Island?

A. In my understanding of Three Mile Island, based on preliminary information is there were a large number of factors. It is my own personal belief about events at nuclear power plants that typically there are many things which have to line up before an event can occur with any consequences, anyone of which if they didn't occur may very well have stopped it.

The contribution of increasing pressurizer level to the decision of the operators to stop safety injection I think contributed to the significance, but I don't think it was the sole contributor to the significance of the accident.

Q. Do you know what the operator based his decision on when he terminated the system?

A. From reports I have read and interviews I have heard it was the information of taking the reactor coolant system solid.

Q. What was the reason, if you know, that the operator had that information?

A. That that was part of the standard training, is that you don't want to take the reactor system solid.

Q. How would he know the system was going solid?

A. The best indication would be the pressurizer level.

Q. That based on the pressurizer level indication, which was either increasing or off scale high, the operator

then terminated the HPI system, is that correct?

A. Yes.

Q. Have you received any information on the March, 1979 accident at Three Mile Island whether or not the operator referred to reactor temperature or pressure prior to termination of the HPI system?

A. I don't know. I don't recall seeing anything on that.

Q. Are you aware of what the general method of operation was at TMI preceeding the accident of this year in terms of operator reliance on pressurizer level indication?

A. Other than what I have heard that that was the primary indicator. I am not familiar in detail with anything.

Q. How have you heard that the operator primarily relied on pressurizer level indication?

Were these conversatio[illegible] Region I when you were there?

A. No, definitely not. It was all after the accident in terms of the investigation, interviews with the operators.

Q. Did you speak with the operators?

A. No. I heard transcripts of the interviews with the operators. I was working in NRC's incident response center during the accident and one of the things I contributed to was getting transcripts made of the first interviews with the operators.

Q. What day was that?

A. I was in the response center from Friday, March 30 through Tuesday, April 11.

Q. The conversations you were responsible for transcribing occurred when?

A. I believe they were interviews made by Region I investigators with the operators on watch on April third and fourth at the site.

Q. No earlier than that?

A. No. I believe they were the first interviews with those operators, at least by, you know, people tasked ultimately with the investigation of the accident rather than coping with the event?

Q. Are you aware of any operator training licensing supervised by the NRC?

A. No, there is none supervised. The NRC reviews the operator training and licensing.

Q. Where does the operator develop or obtain official training and licensing procedures?

A. You confuse me. You used the word licensing and procedures and training.

Q. Licensing and procedures or training and procedures.

A. The licensee, Metropolitan Edison Company here conducts its own training program, which is a program that is designed to get its operators licensed by the NRC. They also

then subsequently participate in that training and requalification program, which is reviewed and approved by the NRC and then periodically inspected by the NRC inspectors.

Q But in the first instance the utility obtains the training procedures from the vendor, B&W in this case?

A Well, they generate their own procedures. They may use B&W generic procedures or hire a company to come in and write procedures for them. All of those things have been done. I am sure they do work with other utilities that have the same power plant.

Q What involvement does the NRC have in, for instance, insuring that an operator of a reactor has been successfully trained or examined?

A What assurances does the NRC have?

Q Does the NRC in any way get involved in determining that the operator is in fact qualified to run the plant?

A NRC examiners go to the site and conduct written examinations, practical examinations, walk-through examinations, oral discussions, and based on those examinations, if the person is deemed to be qualified, a NRC license to that individual to operate that specific power plant is issued.

Q Do you know whether there was any -- there were any operators at TMI-II in March of 1978 who had not previously successfully passed their relicensing exams?

A Well, I wouldn't think at that point it would be

relicensing. That was the first time, and I don't know.

Q. Do you have any information as to whether or not any operators at TMI-II have failed to pass their relicensing or licensing?

A. No. I was trying to remember whether or not I knew that. Some utilities I know have never had anybody who have gone up for a failed examination.

Q. Has TMI --

A. I can't recall whether TMI has.

Q. Are you aware of the testing procedure of the 11 and 12 valves at TMI-II?

A. No, I am not.

Q. When was the last date of your responsibilities in Region I, if you can recall?

A. In the acting capacity, as acting Section Chief or when did I leave Region I?

A. Both, acting and when you left Region I.

A. I will answer the easy question first. I left Region I around August 25, 1978. Probably I stopped acting in late July or August, and I don't recall.

Q. Shortly before you left?

A. Yes.

Q. Do you know low long Mr. Haverkamp was responsible for inspecting TMI-II?

A. Well, he was already the inspector when I took over

the section, and as I understand it, he was still the inspector at the time of the accident in March of 1979.

Q So essentially he was the only inspector by the NRC?

A For the operational things, yes. Well, he was the assigned principal inspector.

Q Did he have inspectors who reported to him?

A Not in a line capacity, but in a technical capacity, yes. The whole range of specialists -- I can renumerate them, but they all inspected at that facility. In other words, the IE inspection program looks across all aspects of plant operations and the principal inspector's role is primarily oriented around the operations of the facility but it is also concerned with the coordination that insures all other aspects of a plant are looked at in a timely fashion.

Q How are you defining "timely"?

A In accordance with the schedule which has been prescribed in the IE Inspection Program Manual.

Q Well, would the precise length of time necessary to be considered a timely review vary with the significance of the component involved?

A Well, it is not specifically oriented around components. It is oriented around functional areas. For example, quality assurance, health physics, training, calibration and the frequency that each of those particular

functional areas is reviewed is prescribed in the IE manual, and the frequency is generally related to the direct safety significance of that functional area.

Q What about the primary system itself? If there was a problem with the primary system, what would be considered a timely review of correcting the problem, or would it have to be limited to a specific part of the primary system?

A Well, the program which I have described to you doesn't relate to your question directly. If you are talking about a specific identified problem which may be reported in the Licensee Event Report, we have inspection requirements for looking at Licensee Event Reports and closing them out to see that the correct action is taken.

Other general problems with reactor coolant systems may be reviewed and things like an inspection where we review when examinations the licensee has performed of primary system components, so there is not direct correlation between our inspection program and how long it would take to look at a given problem.

Q For instance, if there was a problem concerning the accuracy of the pressurizer level indication, would 14 months for the resolution of the problem be considered timely?

A If it was reportable it would be looked at sooner. For example, the pressurizer level may have been required to read 30 inches and during calibration it was found not to read

30 inches and that is outside of acceptable tolerance, that report may be looked at within a couple of months.

If it had resulted in an event of any kind of significance, it may be that an inspector was sent to the site that day or within a few days or some intense conversation on the telephone may have taken place to gather more information about it.

So the 14 months, I am not sure what you are getting at there.

Q If you had an event and there was a problem concerning the accuracy of the pressurizer level indication, would 14 months have been a timely review and resolution of the problem given its importance in the primary system?

A Well, I guess I am getting back to the work accuracy here. I am thinking perhaps in a narrower term than you were. Accuracy says when a pressurizer level is at a given point, what does it indicate, and it has tech spec requirements on that and I would look at that.

That would be looked at much more quickly than 14 months. It would require immediate correction by the licensee.

Q So 14 months would be an untimely review and correction of the problem?

A Relative to accuracy level against technical specifications, yes.

Q That is the only function for the pressurizer level

indication, is it not?

A. Are you getting back to the Michelson Report?

Q. No, unless I should be.

A. As I understand it, the Michelson then -- the level in the pressurizer was fine. It is just that the pressurizer level didn't reflect the coolant inventory.

Q. No. I have a different 14 months.

Are you familiar with someone by the name of James Cresswell?

A. Yes.

Q. How do you know him?

A. I met him at the response incident during the Three Mile Island accident. He came in with the beginning of the IE investigation team. He and Bob Martin and Tim Martin started out their investigation at the Incident Response Center.

Q. Do you remember what day that was?

A. Perhaps some time in the first week in April.

Q. Did Mr. Cresswell during the time you were working together in the Incident Response Center indicate to you any problems he may have come across that were similar to TMI-II within his area of responsibility?

A. He had a general discussion with me about some transience of Davis-Besse with apparently an anomalous pressurizer level and indication, yes.

Q Did he give you any more specifics about what the problem was, what the date of the occurence was?

A No. We discussed some of the details, but I am not familiar.

Q Well, for instance, did he indicate after a few days when people in the Incident Response Center had pretty much the relevant facts that were going on at TMI-II, I had the same problem with Davis-Besse a year and a half ago?

A I believe there was some discussion along that line, yes.

Q What did he relate to you about that?

A I think we just had a general discussion that indeed there was a problem with flashing and voiding in the reactor coolant system, that he thinks that Davis-Besse had had the same problem some period of time before that.

Q Did you come to learn why there were not those problems at Davis-Besse?

A No. I mean, perhaps, we had some discussions of power levels in the decay heat inventory, but everything is blending sort of.

Q Let me ask you, if a reactor were going up to nine percent power, had no prior history with decay heat, could that have explained the lack of a TMI-II type problem at Davis-Besse as Mr. Cresswell explained it to you?

A I am not sure what TMI-II type problem you are

talking about, the increasing pressurizer level, an off scale high -- I don't remember if Davis-Besse had that problem, but it would explain it, because that as I understand it the mechanism, namely that the temperature and pressure relationships are such that you get voiding in the core which is displacing water from the pressurizer and if you don't have that temperature relationship you won't see that phenomena.

Q. Phenomena specifically referred to as off scale high --

A. Or increasing level in a pressurizer with a decreasing average coolant temperature, the divergent temperature average.

Q. The significance of the pressurizer level indication going off scale high is that the operator referring or relying on that information terminates the HPI system, which in the case of TMI-II and presumably Davis-Besse was terminated prematurely, is that correct?

A. Yes.

Q. I believe you previously indicated, but let me just make sure, that you had conversations with Harley Silver, the project manager, concerning the March, 1978 TMI-I problems?

A. Yes.

Q. Did you specifically inform him of problems dealing with the PORV?

A. I assume I did. I remember at the time being very concerned with the fact that the valve failed to open without enunciation, concerned to the point that I wrote the memo on the same day basically as the event occurred, and I feel confident that I would have discussed that with Harley at the time.

Q. Did he give you any indication whether that was a problem from Bethesda's view point or it wasn't a problem or --

A. As I said, I don't recall specifically saying anything to Harley, so I certainly don't recollect any response.

Q. In view of the importance that you attach to the PORV problem by writing a memo suggesting headquarters review, either that day or the subsequent day, would you consider it more likely than not that you would have mentioned the PORV problems to Harley Silver?

A. Yes. Let me discuss another aspect of the IE modus operandi. Several occasions in the past I had written memos of the same general nature and included a request that this issue be transferred to licensing for review.

I had been informed and I do not recall by whom in this was preempting the prerogatives of headquarters to do the review themselves and that such a request should not be included in the future memos.

Q. Who informed you of that?

A. I don't recall.

Q. Was it someone in Region I?

A. Well, if it were, it was at the direction of someone in IE headquarters, but it was -- and I recognized what it was that was being said, that in effect it was IE headquarters mission to make that decision, what should be transferred to licensing, that it was not really a field prerogative to decide, so I specifically included that request from this memo.

Q. Well, in other words, if I understand you correctly, if you find a problem in the field during an inspection or investigation of a plant, and the inspector or the investigator feels that this is a problem of significant size that should be considered by licensing in an attempt to resolve it across the board if it is generic, before you can get to the point of resolution you have to go through the entire I&E structure?

A. Yes.

Q. Someone at the top of I&E in headquarters must make a similar determination that, yes, in fact this is a serious generic problem that should be corrected by licensing, is that correct?

A. Or look at by licensing, yes.

Q. After that procedure is followed, does it then go

directly from the top of I&E down to the bottom of licensing where the problem is resolved or does it also have to go down the licensing chain of command?

A As I recall, there are compatible levels with IE and NRR and the agreement to transfer responsibility is effected at that level and whatever internal mechanism NRR uses to get it to a reviewer, I don't know.

Q You are not familiar with the NRR mechanism?

A No.

Q Based on your experience with the I&E mechanism, what are the potentials for successfully raising a potential generic safety concern at the field level and getting it through all the various stages of I&E up to the top where a decision can be made to send it to NRR?

A I think it depends on the significance of the particular item that is being addressed and perhaps to some extent on the way it is written.

The memo as I look at it now, you know, perhaps didn't have enough red flags waving. I emphasize in there what I consider to be the potential generic aspects, the fact that it was safety grade and things like that.

I have a feeling -- my own personal feeling was that there was enough there if somebody was sensitive to my concerns that there were enough flags there, but I have a feeling if I was truely concerned with getting that thing

looked at I could have either gone back with a follow up memo or have written it differently to demand something.

Q Well, is it you opinion that based on your experience with I&E as you attempt to bring a safety, a generic matter up the ladder in I&E, at each individual step it is possible that the matter will be terminated without resolution?

A Yes, that is correct.

Q Is it more likely than not that substantial safety concerns of regional people or inspectors or investigators will not get to the top of the ladder because of the way in which the procedure is established?

A I don't think it is more likely than not. If it is truely substantial, I think it will and has clearly been in the past transferred over. I think -- you know, again I will get back to what I think headquarter's mission is, is to bring a perspective and a judgment into play.

The fact is there are a lot of concerns which surface from the field which may be valid concerns, but which don't necessarily represent a significant threat to the health and safety of the public to warrant an allocation of resources.

Q Well, you felt sufficiently concerned in your March 31, 1978 memo to Mr. Seyfrit requesting generic expedited review by B&W of their PORV problems that -- well,

first of all to write a memo, and secondly to attempt to climb the I&E ladder to get a resolution, which was not the resolution, evidently you eventually got.

A. As I said earlier, though, part of my acceptance of the responsibility which I got was the fact that as far as I was concerned Metropolitan Edison had gone a long way towards reducing the probability of the same event, which I was concerned with in the March 29, 1978 event from occurring.

(Recess.)

Q Mr. Sternberg, before we took a short break you indicated that on most occasions you accepted I&E headquarter's perspective on matters that field representatives may raise in terms of resolving possible generic or significant problems.

A Yes.

Q And that you specifically accepted their judgement on the March 29, 1978 TMI-II problem dealing with the PORV and the fact that there was no indication light as to whether the valve was open or closed.

A Yes.

Q Do you know whether the same individuals who resolved your concern in May of 1978 were also the same people who were in a position to resolve similar concerns in March of 1979?

A Well, Karl Seyfrit wasn't. He had been promoted as the Director of Region four in Dallas, Fort Worth.

Q What about Roger Woodruff, since he appears to be the individual most directly responsibile for reviewing your concerns?

A I believe Roger is still in the same basic capacity.

Q So, based on the events that happened at TMI-II in March of 1979, had someone in I&E headquarters given a

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little more attention to your concerns, there might not have been a TMI-II in March of 1979, is that correct?

A I don't think there is any direct connection.

Q Well, there was no adequate indication as to the actual position of the PORV at TMI-II was there?

A No, but the thing which was responsive or would have been responsive to my concerns relative to the valve position indication would at least -- at least what satisfied me in that I understood that direct indication was not available because of the construction of the valve, would have been an electrical indication, which would not have necessarily functioned during the condition at Three Mile Island-II in March of 1979.

Q But we don't know whether had I&E headquarters reviewed the problem more thoroughly than they apparently did, they would have seen the neccessity for an actual position indication on the PORV a year before the TMI-II accident we are investigating. In other words, they looked at the minimum rather than the maximum of precautionary measures necessary?

A Yes.

Q To your knowledge, were individuals responsible for review by I&E headquarters aware of the B&W change in the PORV by the installation of an indirect warning light?

A A, I don't know whether it was B&W that made

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the change. I think it was Metropolitan Edison who made the change.

Q You think that was based on a recommendation by -- B&W?

A I don't know what role if any B&W played in that change. They may or may not be aware of it through the licensee report mechanism, which they did have access to, as I indicated earlier.

Q So, it is entirely possible, based on the face of Exhibit 3 to this deposition, the responses to you from Karl Seyfrit terminating any further review of the PORV problem, that they were unaware of the installation of an indirect position actuation light?

A I don't know. It is possible. I just don't know.

Q And yet they still concluded that the matter was not sufficient to require further review?

A That is based on the discussion in the FSAR, which indicated the size of the PORV which you could then compare to the analyzed range of pipe break accidents and they also indicated that in the event that the valve did ~~fail~~ fail open, there was a power operator block valve upstream to isolate it.

Q Are you aware of whether or not I&E headquarters has changed their procedures at all as the result of TMI-II

in March of 1979?

A There have been many changes.

Q Their procedures for reviewing the types of concerns you expressed in your March 31, 1978 memo?

A No, I am not aware of any.

Q Are you aware of any proposed in the lessons learned or I&E reports or investigations?

A Let's go off the record for a second.

(Discussion off the record.)

MR. SIDELL: Back on the record.

BY MR. SIDELL:

Q I believe the question was whether or not you are aware of any changes that are proposed to I&E in reviewing the types of concerns you expressed in your March 1979 memo?

A Yes. I am serving on a task force to review what kind of changes I&E should make and I have drafted such a proposed change.

Q What does that change include?

A Establishment of permanent groups composed of NRR, I&E, the NSSS, architect engineers to review events. Another system to dissemanate field events to all concerned people through a mechanism similar to the I&E bulletin.

A simplified what I call a reappeal of a response to get it one level above, although I suspect that that mechanism was always available to me.

Q Are you referring to the open door policy?

A Yes. The technical dissenting view or the minority opinion, or whatever -- something where without any prejudice, a simple thing saying let's look at it again by another personnel would be available.

Q Is the essense of your proposal to revamp the I&E system to establish more direct communications with other branches within the NRC without having essentially to go all the way up the I&E ladder to get all the way down to the bottom of the NRR ladder?

A It is to assure that the information becomes available to everybody who might be sensitive to the significance of it and provide an earlier recognition of the potential significance.

Also, to bring about a national perspective and reliable mechanism for determining the generic aspects. For example, a problem may occur in Region one and Region four and they each individually may not recognize either the similarities of the event or the possibility for a synergistic effect that you could bring about at a national level.

Q So, essentially things that are both compartmentalized on an internal basis in headquarters as well as compartmentalized between various Regions as you see them?

A Well, it is not so much that they are

compartmentalized, but for example the information through the grey book and through morning reports is made available from one region to another. However, if you had -- I am proposing a standing group whose job is to review events, I have a feeling that they would perhaps dig more deeply into the possibility that there may be an underlying cause for some events or the possibility that two events independently which have occurred, if they occurred simultaneously could provide a condition which hasn't been considered previously.

Q You previously mentioned that looking back you may not have had enough red flags waiving in your March 31, 1978 memo to signal the, what you perceived as significant, generic problems with the PORV.

After your reading of the Michelson report, did you find that there were enough red flags waiving in that report to signify significance?

Q I don't recall, but I was not left with an overwhelming impression that it was the cry of a desperate man. No, I didn't sense that. That it was a well written technical report which hypothesized certain conditions may occur, and I don't even remember if there was any action requested or suggested at that point.

Q Do you know who that report was written for?

A No, I don't .

Q Nevertheless, you were aware in relatively clear terms of Mr. Michelson's concerns dealing with pressurizer

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level indication problems and there inaccuracies?

A Yes. Well, the potential for misleading them, yes.

Q As well as the fact that operators apparently primarily rely on that method of indication to determine what is going on in the core?

A Yes.

MR. SIDELL: Do you have anything?

MS. MOE: No, not at this time.

MR. SIDELL: Mr. Sternberg, rather than adjourn the deposition we will recess it so that in the event we have further questions you will still be available. We will make every effort to avoid that. We will recess the deposition now, and when it is transcribed, it will be presented to you for review, correction if necessary, and signature. -Thank you, very much.

(Whereupon, at 12:51 the deposition was recessed to reconvene if necessary on a later date.)

I have read the foregoing pages, 1 through and they are a true and accurate record of my testimony therein recorded.

Daniel M. Sternberg 8/14/79

DANIEL M. STERNBERG

Subscribed and sworn to before me this
this _____day of _____, 1979

_____

Notary Public

My Commission Expires:_____

REPORTER'S CERTIFICATE

DOCKET NUMBER:

CASE TITLE: DEPOSITION OF DANIEL M. STERNBERG

HEARING DATE: July 30, 1979

LOCATION: Bethesda, Maryland

I hereby certify that the proceedings and evidence herein are contained fully and accurately in the notes taken by me at the hearing in the above case before the PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND and that this is a true and correct transcript of the same.

Date: July 31, 1979

Marcia Stein

Official Reporter
Acme Reporting Company, Inc.
1411 K Street, N.W.
Washington, D.C. 20005