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UNITED STATES OF AMERICA

PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND

DEPOSITION OF: HOWARD K. SHAPAR

Room 9205
7735 Old Georgetown Road
Bethesday, Maryland

August 3, 1979
10:30 o'clock, a.m.

APPEARANCES:

On Behalf of the Commission:

GARY SIDELL, ESQ.
Assistant Chief Counsel
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Washington, D. C. 20037

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I N D E X

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PROCEEDINGS

Whereupon,

HOWARD K. SHAPAR

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

DIRECT EXAMINATION

BY MR. SIDELL:

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

A. Howard K. Shapar, S-H-A-P-A-R.

Q. And your current position at the NRC?

A. Executive Legal Director.

Q. I would note for the record that you are not currently represented by counsel. Do you know whether or not you were supposed to be represented?

A. I do not.

Q. Do you desire to be represented?

A. No. Well, let me revise that.

Q. Okay.

A. If it will ~~expedite~~ EXPEDITE the President's Commission's considerations I would not hold it up to have counsel here.

Q. And by profession you are an attorney?

A. I am.

Q. And you have no hesitation about continuing with the deposition without the assistance of another attorney representing your interest?

A. None.

Q. For the record, again, I am sure this is perfunctory but let me ask you if you ever have had your deposition taken?

A. I can't recall that I have.

Q. Let me briefly, then, explain what we will be doing, which I am sure is well familiar to you.

Your testimony is, of course, sworn under oath and even though we are sitting in your office in one of the buildings of the NRC in Bethesda, Maryland, a relatively informal atmosphere, your testimony has the same effect as if you are in a court of law before a judge or jury.

Therefore, the need arises for you to be as precise and accurate to me as you can. Should you have any confusion or misunderstanding about my questions, please ask for clarification and I will try to explain what I am looking for.

Since the reporter is taking the testimony down, it is necessary that you wait until I finish my question completely, even though you may know where the question is leading, before you begin to respond. I will try and restrain myself from asking my next question until you have completed your answer.

Furthermore, in view of the fact that the reporter is transcribing the testimony, it is helpful that you avoid giving nods of the head, or gestures, in response to my questions but rather answer it.

At the conclusion of the deposition, your testimony will be transcribed by the reporter, presented to you for your review, corrections, if any, and your signature. Should you find any necessity for correction or changes in the transcript, you, of course, will be entitled to make those changes.

However, you should be advised, in view of the fact that we might consider some of the changes to be of a substantial nature, we can comment on those changes and that, in turn, may adversely effect your credibility. Again, the necessity to, in the first instance, be as precise as possible in your responses to the questions.

Do you have any questions concerning what I have just mentioned?

A. No.

Q. Can you give me a brief description of your responsibilities in your current position at the NRC?

A. Yes. As I indicated before, I am Executive Legal Director and my office is responsible for providing legal advice to the Executive Director of Operations and all elements of the staff that report to him.

Q. Is that, in fact, the entire staff of the NRC, with the exception of the Commissioners reporting to the Executive Director?

A. With the exceptions of those elements of the staff that report directly to the Commission, such as the Office of General Counsel OPE.

Q. What does OPE stand for? Do you happen to have an organizational chart or telephone book handy?

A. Yes. (Pause) Office of Policy Evaluation. Other such offices would be the Office of Inspection and Auditor, Office of the Secretary, Office of Public Affairs, and Office of Congressional Affairs; although I do, on occasion, upon request, give advice to some of those offices.

Q. Do you provide legal advice to the Office of Nuclear Reactor ~~[illegible]~~ REGULATION, NRR?

A. I do.

Q. Are you responsible for determining, in the ultimate instance, whether or not to bring any actions under Part 21 of the NRC regulations?

A. No.

Q. Who is responsible for that, if you know?

A. The Office of Inspection and Enforcement, though the Office of Inspection and Enforcement consults with my legal staff prior to taking such action.

Q. Are those consultations reviewed by you?

A. Mostly not, but on occasion yes.

Q. What specific instances, or occasions, would you find it necessary to review consultations with I&E and your staff?

A. One situation would be where the staff, my staff, felt it desirable to seek my advice, probably because it presented a novel question or a matter first of first impression and that sort of thing.

Q. Would you conclude that your staff might seek your advice concerning Part 21 problems if there were a substantial safety question involved as opposed to possibly a technical violation of Part 21?

A. I don't think it would rest on the significance of the safety question. It would be more associated with the difficulty of the legal question.

Q. In other words, whether or not it was a difficult case to prove?

A. Not necessarily a difficult case to prove but more in the nature of a difficult question of law that had not been resolved before.

The matter of proof would not be the type of thing that would usually be brought to my attention.

Q. Would you be consulted as to whether or not to seek revocation of a plant's license as a possible penalty?

A. The utility license?

Q. Yes.

A. Yes, probably yes.

Q. In all instances or that was a possible result?

A. We have never revoked a utility license. It would be a highly novel situation in the sense that there is no previous precedent for it. I think the likelihood is that I would be consulted.

Q. You just mentioned that there is no precedent for revoking a utility license since none have ever been consulted. Have you concluded whether or not under the statutory enabling authority provided for the NRC and the NRC's regulations if you could seek such a result?

A. Yes. We can, in my opinion.

Q. Would--

A. Section 186 of the Act, I believe, is to the effect that among other grounds the Commission can revoke a license on any grounds that would have entitled it to deny the original application for the license.

Q. Would you be consulted on matters where a maximum fine is considered?

A. Not necessarily.

Q. What is the maximum fine available at the current time for Part 21 violations?

A. (Pause) "Not to exceed $5,000 for each violation provided that in no event shall the total penalty payable by any person exceed $25,000 for all violations by such person occuring within any period of 30 consecutive days. If any violation is a continuing one, each day of such violation shall constitute a separate violation for the purpose of computing the applicable civil penalty."

I should add that the Commission has, is seeking legislation to increase the maximum fines that could be imposed.

Q. In order to respond to my earlier question as to the maximum fine, you made reference to specific regulations, is that correct, in reference to the book?

A. I referred to the statute.

Q. And that was the enabling statute for the NRC?

A. The Atomic Energy Act of 1954, as amended.

Q. Do you know what the maximum that is sought by the Commission is going to be?

A. For TMI?

Q. No, for changing the statute.

A. No. I would have to look that up. I don't recall offhand.

Q. Can you recall whether or not it is less than $100,000 as a maximum?

A. I could find out for you immediately by making a phone call.

Q. Okay, if you would, please.

(A short recess was taken)

Q. In your position as Executive Legal Director Mr. Shaper, would you be aware of the number of civil penalty actions previously brought by the NRC?

A. No. One of the divisions reporting to me is the Division of Enforcement and Rule-Making, and the head of that division is James Murray. He would be in the position to give you factual data of that type.

Q. Would it be possible to get some figures from Mr. Murray as to the number of prior civil penalties successfully brought, those brought in the first instance without satisfactory resolution, and the same information for the AEC?

A. He could get it for you, as could the Division of Enforcement and Inspection. Either source would be available.

Q. Is that something that is readily available?

A. I don't know how accessible it is, but it could certainly be obtainable.

Q. Could we get in touch with Mr. Murray and ask him to provide that information?

A. Surely. Do you want to contact him directly or would you like me to do it?

Q. If you might give him a call now?

A. He is on leave but I could talk to his assistant.

Q. All right.

A. Could you repeat the question, please?

Q. Sure. All information concerning civil penalty actions brought under Part 21 of the NRC regulations with the ultimate resolution in terms of the maximum or the total fine assessed; all civil penalty actions brought by the NRC without satisfactory resolution. In other words, all actions instituted.

A. Just Part 21?

Q. Yes.

(There was a discussion off the record)

Q. During an off the record discussion, Mr. Shaper, with Mr. James Lieberman, someone on your legal staff, he has agreed to provide the information I requested of you concerning Part 21 and Part 206; or is it 206?

A. I think it is Section 206. Let me check. (Pause) 234 of the Atomic Energy Act of 1954, as amended, civil monetary penalties for violation of license requirements.

Q. With the conjunction of Section 234 and Part 21, would that cover all the civil actions the NRC would bring against reactor operators for violations of regulations?

A. Well, Part 21 is pursuant not to the Atomic Energy Act but pursuant to the Energy Reorganization Act. The sections we have been talking about relates to civil monetary penalties.

We have other enforcement authority. Authority to revoke a license or to suspend a license or to impose license conditions, to amend a license so that the spectrum of enforcement authority ranges at the low end from a notice of violation to the other end of the spectrum, which is a revocation of a license. In between is a civil penalty.

One of the premises for our civil penalty authority was to have a more complete arsenal of enforcement actions, something betwen a notice of violation and an outright revocation of the license.

Q. I believe, as you have indicated previously, there has, to your knowledge, never been a revocation of a utility license?

A. That is right--

Q. Has there ever been--

A. --to the best of my recollection.

Q. Has there ever been, to your knowledge, a case in where--

A. We are talking now about utilities?

Q. Yes.

A. Okay.

Q. Has there ever been, in your knowledge, a case where a utility's license was suspended for a period of time?

A. Yes, there have been suspensions of licenses.

Q. Do you recall what the grounds for those suspensions were?

A. It was on health and safety grounds.

Q. Do you recall when they were?

A. Well, the most recent--one of the most recent suspensions ~~have~~ HAS been in connection with TMI, also in connection with the seismic problems that occurred a few months ago.

Q. That is referring to the five plants for questions of construction integrity that had their licenses evidently suspended?

A. Yes.

Q. Were there any instances preceeding the accident at TMI 2?

A. I think that there were, but I can't be more precise without going back and checking the records.

Q. Would you be able to provide that information to us?

A. Yes. What you would like is all instances where the NRC or the AEC suspended a license for a nuclear power plant?

Q. Correct.

A. Okay.

Q. And as I believe you have previously indicated, Section 234 of the Atomic Energy Act and Part 21 of the Reorganization Act.

A. No. Part 2' is part of our regulations.

Q. Pursuant to the Reorganization Act?

A. Pursuant to the Reorganization Act.

Q. Covering the spectrum of the NRC's capability of requesting monetary fines for violations and regulations, is that correct?

A. Yes.

Q. You have provided me with a biographical outline which specifies your educational background and professional memberships, and I have requested that you provide us also with information dealing with your professional employment, which you have agreed to provide to us.

As the biographical outline contains the information of your educational and professional associations, is that information true and accurate to the best of your knowledge?

A. It is.

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

BY MR. SIDELL:

Q. Were you consulted for a legal opinion concerning the suspension of the operating license for Metropolitan Edison at TMI 2 this year?

A. I probably was. I just can't fix at what time or place or who consulted me, but I must have been involved.

Q. That was a matter that would fall within one of the categories you mentioned earlier as to novelty or prime importance?

A. Indeed it would.

Q. Were you also consulted concerning the suspension of licenses for the five plants dealing with--

A. Yes.

Q. --structural integrity?

A. Yes.

Q. Who consults you in these matters? I&E exclusively?

A. No. It could occur in a rather wide variety of circumstances.

Q. Well, with TMI 2.

A. It could occur as a result of an attorney on my staff seeking my advice without a direct contact on my part with the initiating operating division.

In so far as utilities would be concerned, there would be two offices mainly involved: NRR and I&E. In that kind of a situation I could get involved essentially one of three ways: by the attorney who has been working with either I&E or NRR, or NRR itself coming directly to me or I&E coming directly to me.

Now, in most occasions it would be the attorney who worked on a day to day basis with one of those two divisions who would feel that he wanted advice and would seek it from me.

Q. With reference to both I&E and NRR, would the person most likely to contact you concerning a license suspension or revocation, if one were to arise, be the directors of those offices?

A. I would say most likely the director or the deputy director.

Q. Are there established formal proceedures for NRR and I&E to contact you concerning license suspension or license revocation?

A. Not that I am aware of.

Q. It is merely done on an informal ad hoc basis?

A. Yes, but I think that Murray could give you more detail on that than I because the initial contact would be with him, and he may have worked out some particular arrangements with I&E.

Most likely, the NRR contact would not be through Murray. Murray provides services only to I&E in the enforcement area.

In so far as NRR is concerned, they would go to my hearing division and the head of the hearing division is Edward Christenbury.

So, the way it works essentially is that Murray's division would be providing legal services on a day to day basis to I&E. The hearing division would be provided legal services on a day to day basis to NRC. To the extent that either the chief counsel for the hearing division or the chief counsel for Enforcement and Rule-making hearings had a problem that he thought was novel or different enough to warrant seeking my advice, he would seek my advice.

Q. Would it be fair to conclude that the hearing division would provide information concerning generic problems to Mr. Christenbury, whereas I&E would provide information to Mr. Murray dealing with plants specific items or problems?

A. No. I don't think I would put it that way. I think that both of them would be dealing with both generic matters and plant specific ones.

It is really a question of who has the lead on the action, whether it is NRR or I&E. If I&E had the lead, then it would go to Murray. If NRR had the lead, then it would go to Christenbury.

Q. So, these are usually cases involving both I&E and NRR and it is merely a question as to the originating office being the one to contact which particular division in your office?

A. Yes.

Q. So, by the time the problem gets to your office, there has been substantial communication and analysis performed by both I&E and NRR?

A. Well, that depends on how the crisis arose. If it is a genuine crisis, I would say IT IS LIKELY THAT THERE HAS BEEN COMMUNICATION - HOW SUBSTANTIAL, I DON'T KNOW ~~[illegible]~~

Q. In a situation where, for example, an I&E inspector in the field comes across a specific problem in a plant, and that report goes up the ladder in I&E, over to NRR and there is a generic problem recognized by NRR and its subsequent resolution to the point where they feel that there are matters that should be brought to your offices attention, those matters would involve some formal technical resolution by both I&E and NRR preceeding their contact with your office, is that correct?

A. Well, my attorneys work WITH THEIR CLIENTS very closely on a day to day basis and they may very well have been involved at an earlier stage. So, I don't think that you really portrayed a usual situation necessarily.

If the problem is identified as a serious problem, the attorneys may be involved from the very beginning. It isn't a question of the attorney sitting back and waiting for the problem to have a definitive solution in so far as NRR or I&E are concerned.

The attorneys are frequently involved in much earlier stages and work with I&E and NRR during the attempted resolution of the problem.

Q. What kinds of involvement would your attorneys perform?

A. Well, certainly they would be involved in the legal aspects of it. Beyond that, to the extent that they were requested to provide policy advice. That would depend to a large extent on the personal relationship between the attorney and the client, and, of course, it varies between the attorneys and the clients.

Beyond that, of course, the line between what is legal on the one side and technical and policy on the other is not always clear. So, to some extent it does depend on the kind~~s~~ of relationship that HAS evolved between the attorney and the client.

Q. Would you have had occasion to, once a plant specific violation has been determined, institute a civil action against that particular plant while continuing to pursue any generic implications that problem has?

A. That could very well be the case.

Q. Is that a usual situation?

A. Well, I think it is fair to say that when a specific problem arises, if it has generic implications, that the responsible operating office would be looking at the generic matter.

From a legal standpoint, of course, we are perfectly free to go ahead and deal with the specific matter while the generic matter is being looked at in a longer range time framework.

Q. Who resolves whether or not an issue is generic as opposed to plant specific?

A. The operating divisions.

Q. That would be NRR?

A. Yes. Now, you realize, of course, that the Commission gets involved in these matters on occasion as well, and staff is staff and Commission is Commission. The Commission frequently gets briefed on problems that have generic implications.

Q. Who briefs the Commission concerning wide ranging or generic problems?

A. It depends on the nature of the problem, whether it is an I&E focus or NRR focus. I would say in the main it is NRR.

Q. Would that be because NRR has the technical expertise to determine, in fact, that a generic problem exists?

A. Well, ~~it~~ THERE is considerable competence as well in I&E but the way the process works NRR is very heavily involved in the generic aspects of safety problems relating to nuclear power reactors.

Q. Do you know whether, as a rule, I&E makes generic determinations?

A. Oh, I guess it depends on what you mean by generic determinations in terms of what the requisite degree of safety is and what means are needed to achieve that objective or standard.

That is basically an NRR determination. One of the tools they have to use, of course, is the information provided them by I&E.

Q. So, generally, I&E is the first line of investigation coming up with a particular problem, forwarding that, in turn, to NRR, which presumably has the wide-ranging view capability to determine that it may or may not be a generic problem?

A. Yes. It relates--that is essentially right, I believe. It is essentially related to the functions of the two offices.

NRR is the licensing office. It is the one that reviews the applications, it is the one that proposes rules, safety rules. They are generated mainly in NRR, but there is a very close interaction between NRR and I&E. When, for example, NRR proposes a generic resolution of the safety problem there is close interaction and input from I&E, and from Standards as well.

Now, we haven't talked about the Standards Division, but we are talking about generic problems, and the Standards Division is also heavily involved. As a matter of fact, most of the rules changes, or many of the rules changes relating to safety standards and safety requirements, are actually prepared by the Standards Division and going up to the Commission.

So, although you are talking about safety, you are talking about generic problems. I would say that all three divisions are involved. I would say the main ~~lead~~ LEAD on the technical aspects of it is in NRR.

Q. Would it be fair to conclude that the Safety Division proposes a rule change, or a new rule, once NRR has determined a generic safety problem to exist?

In other words, the Safety Division is a continuation on the spectrum?

A. I don't know what you mean by the Safety Division. Do you mean Standards?

Q. Yes, Standards.

A. Now, could you repeat the question?

Q. Certainly. I&E, in the first instance, finds a particular problem as a generalization. That particular problem is referred to NRR for potential generic considerations.

On the assumption NRR makes a generic finding of a plant specific problem provided by I&E, would it be fair to conclude that the Standards Division, as the next step on the continuim, proposes to change a rule or institues a new rule to deal with the problem?

A. I would think that the initiative may very well come from NRR and the formulation of the staff paper going to the Commission MAY BE DONE IN STANDARDS. It might very well be done by Standards.

Q. But Standards would not be involved in the first instance with I&E in determining whether or not a particular problem in a specific plant involved a violation of one of the NRC's regulations?

A. I would think not. I would think that NRR would be heavily involved in ~~support of~~ SUCH a situation.

Q. Standards are more concerned with more wide-ranging resolutions of problems?

A. Well, their responsibility, of course, relates not just to utilities and nuclear power reactors, but to all the material ~~reliance~~ LICENSEES as well. They would get involved, for example, in situations about what the release rates should be and that sort of thing — more gneral regulations like Part 20. ~~NRR~~ NRC's Part 20 would be a good example of the ~~sort~~ TYPE of activities that Standards would be involved in ~~as the extent of impact on reactors~~. Then, there is a very close relationship — a give and take between Standards and NRR.

Q. Is there any input from Standards to assist NRR in resolving whether or not a problem is, in fact, generic, and not NRR?

A. I would guess the main actor in the process of determining whether a problem is generic in so far as power reactors is concerned, ~~would rest with~~ IS NRR, with consultation with other divisions, particularly Standards, and to some extent with I&E.

Q. So, once that generic resolution is made by NRR, the ball essentially goes to the Standards Division?

A. Well, again it is in reaction that--

Q. At least in terms of operating reactors, not materials or other problems of that nature?

A. The paper ~~might be prepared by~~ going to the Commission, might be prepared by Standards, but if it is a generic resolution of a safety problem relating to reactors, then I would assume in most instances that the basic determination, at least at the staff level, is by NRR. I am talking about hardware and that type of thing in the main.

Q. Within a nuclear reactor?

A. Yes.

Q. Let me show you a letter dated March 29, 1979, entitled: "Subject: Board notification", which is signed by Howard K. Shacar, which, I believe to be two pages, and ask you if you have ever seen this previously, without, of course, the notations made on the copy?

A. (Pause) Yes. I recognize this, remember it very well.

Q. Is that a letter that you sent to the several people listed on the top?

A. Yes. It is a memorandum, or a note, rather.

MR. SIDELL: Let's have this marked as Exhibit 2.

(The document referred to was marked for identification as Exhibit 2)

BY MR. SIDELL:

Q. Exhibit 2 to this deposition apparently was distributed to E. Christenbury--is that Edward Christenbury, who is the head of your Hearings Division?

A. Yes.

Q. J. Scinto, who might that be?

A. Deputy in that division.

Q. G. Cunningham?

A. The head of one of the hearing sections, one of our four hearing sections.

Q. E. Reis?

A. Head of another hearing section.

Q. J. Tourtellotte?

A. Tourtellotte, head of another hearing section.

Q. and, S. Treby?

A. Treby.

Q. Treby.

A. Head of another hearing section. There are four sections.

Q. What prompted this two page board notification memorandum that you distributed to the previously named six individuals, that you can recall?

A. Can I see it?

Q. Sure.

A. (Pause) As the first paragraph of Exhibit 2 indicates, "It has come to my attention today that an NRR request of March 6, 1979, recommending the transmission of information to Licensing Boards regarding a reactor inspectors concerns about B&W plants was not sent to these Boards until today"-- the date of my note, which was March 29, 1979,-- "after the delay was brought to our attention by Commissioner Bradford's office."

Q. So, you learned of the request for board notification from Commissioner Bradford, is that correct?

A. Or his office.

Q. Which might include his technical assistants?

A. It might.

Q. Do you recall--

A. I don't recall how I learned from Commissioner Bradford's office about this, but as the note indicates, that was my belief at the time and I have no reason to doubt it.

Q. Is this the usual method by which you receive information of requests for hearing boards, from a Commissioner's office?

A. No. I can't recall any other occasion.

Q. So, you would classify this as rather unique?

A. Yes, indeed, which I think explains, perhaps, the directness of that memorandum in part.

Q. Would you characterize the content of Exhibit 2 as a rather frank and candid statement of your offices policy with respect to requests for board notification?

A. Yes, indeed, and I think the policy speaks for itself.

Q. Well, for the record, what is that policy?

A. That there be prompt review of any proposed transmittal of information to a board, that it not be held up, and under no circumstances should information be withheld from a Board that any of the operating divisions wish to send to a Board.

In fact, I made that explicit when the procedures were being developed. As a matter of fact, I should add in the interests of completeness, that I have recommended changes in the procedures so that the information would go directly from the Operating Division to the Hearing Boards and not to go through this office. The transmittal WOULD go directly from ~~the~~ NRR, or NMSS or any other division. NMSS is the OFFICE OF Nuclear Material Safety and Safeguards.

Q. In addition to NRR and I&E, what were the operating offices you just mentioned?

A. I would think the operating offices. In terms of a hearing matter would be either NMSS or NRR. On occasion it might be Research, on occasion it might I&E.

Q. But the bulk of the work in terms of the Hearing Boards comes from either NMSS or NRR?

A. The bulk of it is NRR, very sure, because there are very few hearings associated with materials licenses.

Q. When were those proceedures established?

A. I can't recall offhand. I would guess within the last year or two, something like that. Lots of paper-- there is lots of paper around relating to them.

Q. The paper was widely distributed to your staff?

A. Yes.

Q. Also to the staff of NRR?

A. I--

Q. If you know.

A. You would have to check with NRR.

Q. Did you learn from Commissioner Bradford's office, when they contacted you about this request for board notification, how his office came to be informed of the problem?

A. No, I did not.

Q. Would this--

A. I am not sure that Commissioner Bradford's office came to me directly or whether I could have heard about it indirectly.

Q. If you heard about it indirectly, who would that have been from, if you can recall?

A. I can not recall.

Q. Would this type of matter, hearing about a request for board notification, have been the result of the open door policy of NRR?

A. I am searching my recollection. It might have had something to do, and I am not--my recollection on this is not very good. It might have had something to do with the Vepco case involving some seismic problems at that site.

I think it also had a relationship with the open door policy, as well. I am not completely clear in my recollection on those points.

Q. But your recollection today is that the subject contained in your March 29, 1979 note, Exhibit 2 to this deposition, at least in part dealt with seismic problems at a Vepco utility?

A. No. No, I think you asked me what the premise or genesis was for the board notification procedure. I understand that to be your question and I thought my answer was that.

Again, with my recollection not being very clear, that it might have had something to do with a problem in trans--mitting information in a Vepco case about a year or two ago.

and also possibly some interaction with the open door policy.

Q. What I was primarily looking for is whether or not you had any information that the subject matter of your March 29, 1979, memorandum, Exhibit 2, was originated through Commissioner Bradford's office by someone exercising the open door policy?

A. I have no information on that. I should add, in view of your question, that what promoted the directness and forthrightness of that memorandum, was the delay in my office's dealing with the transmittal of the information to the Board.

Q. Do you recall the date the request was first received?

A. I do not, but my general recollection is that the amount of time it had been in this office was much too long and transgressed the guidelines that I had set down.

Q. Well, can you recall the length of time the request had been in your office?

A. I can not.

Q. What are the--

A. That would be ascertainable, though, from the attorneys involved in reviewing the transmittal request.

Q. What length of time is acceptable once a request for board notification has been received by your office?

A. I would have to check this out, but my recollection is that it is to move through this office within three working days. I am not sure about that, but that is the best of my recollection at this time.

Q. Is that a formal requirement of your office?

A. I believe that there is a piece of paper that I have issued that reflects that.

Q. Well, we would be interested in getting a copy of that requirement if you could provide us with one.

A. I will seek it and provide it.

Q. Let me show you a memorandum dated January 19, 1979, from James G. Keppler, who is Director of Region 3, for N.C. Moseley, Director of Division of Reactor Operations Inspection, IE; and H.D. Thornburg, Director, Division of Reactor Construction Inspection, IE, which contains a memorandum for J.F. Streeter, who is Chief of Nuclear Support Section from J.S. Creswell, Reactor Inspector in Region 3, which is dated January 8, 1979.

There is also a letter from Lowell Roe to Robert W. Reid, Chief of Operating Reactors Branch No. 4 of the NRC in Washington, which contains a reporter analysis performed apparently by Toledo Edison, whom Mr. Roe represents, as well as several excerpts from Mr. Creswell's January 8, 1979 memorandum, and ask you whether or not you have ever seen this before?

A. (Pause) I have no recollection of ever having seen that before.

Q. Have you read the cover letter, or memorandum, from Mr. Keppler to Mr. Mosely and Thornburg that I have just showed you, or have you just merely--

A. Merely just cast my eye over it. It is not the type of document I would usually see.

Q. Would this be the type of document that would initiate a request for board notification coming to your office?

A. I would have to look at it more carefully.

Q. Please do.

A. (Pause) I notice that the memorandum is from Keppler to Moseley and Thornburg. I would think in the usual course of events it would be the determination of Moseley or Thornburg, and or the Director of I&E to make the decision to notify the Board, in which case at that point in time it would come to the attorneys.

Q. Do the directors of I&E and NRR have discretion as whether or not to notify Licensing Boards based on an inspector's request for such a notification?

A. I think that rests on the precise words of the existing proceedures for Board notification. I think that the key probably has been the use of the words--and I am speaking from memory--relevant and material.

The question is in view of the massive information that is available on a day to day basis what is relevant and material to an on-going hearing in a licensing case. I think that is the main problem in implementing the Boards notification procedures.

Q. When were you contacted by Commissioner Bradford's office?

A. If I was. Remember, I said that I can't recall whether or not I got this first hand from Commissioner Bradford's office or was informed about it by some third party.

Q. If you were informed about it by some third party, that would be a highly exceptional situation, would it not?

A. The fact that a Commissioner's office was aware, was asking a question about Board notification? That is the only example of it that I can recall, as with the case we are dealing with here.

Q. So, it would have been even more exceptional if you had learned about the problem through a third party rather than the Commissioner's office?

A. Well, the circumstances are exceptional. So, I don't know whether it is more exceptional to hear about it from a third party or directly from the Commissioner's office.

Q. You refer, in your March 29, 1979, memo, Exhibit 2 to this deposition, of information regarding B&W plants based on an inspector's concerns. Do you have any information as to what those concerns dealt with?

A. I do not.

Q. Do you have any information as to who that inspector was?

A. I do not.

Q. What region he was from?

A. I do not. Again, my central, if not exclusive, concern that prompted the writing of that memorandum was the delay in the review time attributed to my office.

I did not get into the details or the substance of the communication. My focus at that time was on the failure to have PERFORMED ~~provided~~ a more timely review on the transmittal.

Q. And the delay you are referring to would have exceeded three days, three working days?

A. To the best of my recollection, yes, or else I would not have been concerned in the manner I was.

Q. Do you have any recollection at this point as to whether or not it was more than one month?

A. I do not.

Q. Well, based on the candor contained in your March 29, 1979, memorandum to your staff, would you conclude that the period of time of the delay involved would have been on the order of one to two months?

A. I just can not recall.

Q. Do you have any notes or other information that would refresh your recollection as to the length of the delay?

A. I think, as I recall, that memorandum--doesn't the last paragraph request an inventory by the close of the day, March 30? (Pause) Yes. The paragraph reads: "I wish an immediate inventory to be made to identify any other NRR or NMSS recommendations for informing Boards which have not yet been acted on and to report in my office by the close of business today on the results of the inventory. I will also expect that for those recommendations located on which action has not been taken, the Board notification letters from the Office be sent no later than the close of business tomorrow, March 30, 1979."

My recollection is that that deadline was met.

Q. And that request was made to your staff?

A. That is correct.

Q. So, they had already in their hands requests for Board notification as opposed to them going out and actively seeking from I&E, NRR, NMSS any requests for Board notifications?

A. I would think so. See, the way the process worked under existing procedures is, or then existing procedures, is that it goes through my office for review and help on the question of relevancy and materiality.

That, I recollect, was the premise for review by this office and as I indicated before, I have proposed a change in procedures that would cut this office out completely and have the operating offices make the transmittal directly to the Board without review of materiality and relevancy by this office. But, under my proposal, the operating offices could, as they saw fit, consult with my office about the transmittal.

Q. Let me show you a letter dated January 9, 1979, from J.S. Creswell to J.F. Streeter, who are both individuals in Region 3, on the subject of conveying new information to Licensing Boards--Davis-Besse Units 2 and 3 and Midland Units 1 and 2, with a letter dated March 1, 1979, from Dudley Thompson, Executive Director for Operations Support, IE for Domenic B. Vassallo, Assistant Director for Light Water Reactors, NRR. This seems to be an inter-divisional memorandum.

I ask you whether you have seen this information before, in response to your March 29, 1979, request?

A. I have no recollection of having seen this. It is not the kind of document that I would ordinarily see.

I must add, though, that I recall having dug a little bit into the background of the delay that prompted my memo and it is quite possible that I may have seen those documents as a result of trying to ascertain why there was such a delay on the part of my office in making the transmittal.

Q. What did you--

A. In that case, I would have examined it, not so much for the substance of it but to try and explain, at least to myself, why the delay took so long.

So, I may have seen it in such an inquiry, but I have no recollection of having seen the documents.

Q. What did you determine as a result of your investigation concerning the delay in your office?

A. I determined that the procedure should be revised and that my office should be cut out of the process completely except to the extent that the operating divisions wished to seek on their own volition legal advice about materiality and relevancy on an individual case basis.

If they felt that it was relevant or material, then there should be no review in this office whatever, that it should move immediately. I proposed such procedures, and they are in writing and if you haven't seen them I can make them available to you.

Q. We would request, then, a copy of those new procedures.

A. They are not in effect as yet.

Q. Do you have any question as to whether or not they will, indeed, become effective?

A. Yes. There is some resistance.

Q. By whom?

A. By NRR and by NMSS.

Q. What, if you know, is that resistance?

A. The feeling that the documents should continue to go through this office. The matter is close to resolution, but I can not say that it has been resolved as of this date. I am taking a strong position that it ought not to go through this office, but that this office would would remain available for advice when requested.

Q. Well, evidently NRR and NMSS are taking a commiserately strong position in maintaining their insistance that should matters do go through your office for merely relevancy or materiality determinations, is that erroneous on my part or--

A. I think that it is better stated as being more of a reluctance on their part to have a transmitall effected without some review by this office.

However, it may be that at least on the part of NRR, that they will go along with my position. There has been a rather continuing discussion about this and it may be, but I can't be sure that NRR is willing to go along with the new procedures.

Ed Christenbury, of my staff, has been discussing this fairly recently with Mr. Denton.

Q. Well, if you were to be successful in implementing your new **procedures**, is not the net effect of those new **procedures** to cut off three days of review, three working days of review in your office before the matter gets to a Licensing Board?

A. I think that is a fair statement, except the transmittal time for the documents to reach my office and then for my office to do transmittal. So, it is three working days plus some indefinable transmission time.

Q. Are we talking about another matter of a day or a month?

A. Another day or something like that, a relatively short period of time. It would be a relatively short period of time if the present system works: namely, that there are no more lapses on transmittal.

Q. Were you able to determine where the particular lapse in the system was concerning your March 29, 1979, memo, Exhibit 20 to this deposition?

A. I have a recollection of it, yes. I am not completely sure, but my recollection is that it is on the part of one of the attorney's in ~~NRR~~ MY HEARING DIVISION.

Q. Who did not forward the information, or the request, for the Licensing Board notification to your office when he should have?

A. If I understand your question, the reason for the delay in transmitting it to the Board was the fact that the of the attorney's in the Hearing Section had not found the opportunity to perform the review that was required.

Q. Well, is this attorney in your office or in NRR?

A. All attorneys for the staff are in my office, and ~~it~~ He is in the Hearing Division of OELD.

Q. He deals exclusively with NRR matters?

A. Yes.

Q. So, the matter had been sent to your office one to two months before you were made aware of the problem or the request for the notice--

A. Well,--

Q. --sent to the Licensing Board?

A. I am not sure of the time frame at all, as I indicated before.

Q. Sometime far in excess of the three day working rule?

A. Some time in excess. How much in excess I don't know or I have no recollection of at this time.

Q. Well, if the matter was sent to your office--

A. To my office? You mean this office as a whole?

Q. Yes.

A. You don't mean in my particular office.

Q. No.

A. Okay.

Q. To the offices who report to you and are concerned with forwarding matters of concern to Licensing Boards upon request.

If the matter had been forwarded to that particular office four working days before you had been advised of the matter, would you have written the March 29, 1979, memo?

A. You mean if the time had been very--

Q. Close to three working days?

A. Probably not.

Q. So, it is safe to assume we are talking about a relatively substantial period of time prior to your notification, your office's notification?

A. At least enough in excess of three days to have prompted that kind of response on my part, but the time can be ascertained, I am sure, by you.

Q. Well, I have several documents that appear to respond to your March 29, 1979, memorandum. The first is dated March 29, 1979, concerning Board notification of Davis-Besse, Erie, Greene County, Midland 1 and 2, Pebble Springs, Three Mile Island 2, which is a letter addressed to "Ladies and Gentlemen", from Joseph F. Scinto, Deputy Director, Hearing Division.

It includes a listing of addresses for Toledo Edison, Ohio Edison, Power Authority of the State of New York, Consumers Power Company, Portland General Electric Company, and Metropolitan Edison, which presumably are the owning utilities of the reactors previously stated.

Let me show you this document and ask you if that is, in fact, a response to your March 29, 1979, memorandum?

A. It appears to be, and if it coincides with the inventory that was submitted in response to my memorandum, it undoubtedly was.

Q. Let me show you a March 6, 1979, memorandum for Edward S. Christenbury, Hearing Division Director and Chief Counsel, OELD, from D.B. Vassallo, Assistant Director for Light Water Reactors, NRR, concerning Board notification--reactor inspector concerns about B&W plants.

This memorandum states that it is awaiting I&E's discussion and evaluation of matters raised by a reactor inspector requesting Board notification concerning Davis-Besse Units 1 and 2 and Midland Units 1 and 2.

I ask you if you have ever seen this memorandum?

A. I have no recollection of havin g seen it, but, again, it is possible I may have in connection with my review of the matters that prompted my memorandum that we discussed before.

Q. In the March 29, 1979, response by Joseph F. Scinto, concerning notification of Hearing Boards, there is a reference that concerns a memorandum that relates to "Certain concerns raised by reactor inspector in Region 3 concerning the Davis-Besse and Midland Units."

Does it appear to you that the March 6, 1979, memorandum from Mr. Vassallo to Mr. Christenbury is the response to Mr. Scinto's memorandum?

A. It would appear to be, but I can't be sure.

MR. SIDELL: Let's mark this as Exhibit 3, the March 29, 1979, memorandum by Joseph F. Scinto concerning the Board's notification of Davis-Besse, Erie, Greene County, Midland 1 and 2, Pebble Springs, and Three Mile Island 2.

(The document referred to was marked for identification as Exhibit 3)

MR. SIDELL: And as Exhibit 4, dated March 6, 1979, a memorandum for Edward S. Christenbury from D.B. Vassallo, Subject: Board notification Reactor Inspector concerns regarding B&W plants, which makes specific reference to Davis-Besse Units 2 and 3 and Midland Units 1 and 2.

(The document referred to was marked for identification as Exhibit 4)

BY MR. SIDELL:

Q. Referring to Exhibit 4 of this deposition, the March 6, 1979, memorandum for Mr. Christenbury. He is the Chief of the Hearing Division within your office?

A. Yes.

Q. So, your office was informed of the regional concern in Region 3 of Davis-Besse B&W potential generic problems as of March 6, is that correct?

A. (Pause) Yes. It appears so.

Q. And apparently Mr. Vassallo of NRR received the information he provided in Exhibit 4 by virtue of a March 1, 1979, memorandum for him from Dudley Thompson, who is Executive Director for Operations Support in I&E.

Let me show you this memorandum and ask you if we are still dealing with the same Davis-Besse Units 2 and 3 that eventually found their way to your office?

A. It would appear so. As I recall, when I looked into the--as I recall, when I looked into the reason for the delay in the transmittal of that information to the Board, I think I recall having seen something from Mr. Christenbury's predecessor, Mr. Grossman, on this matter, saying that in his opinion it should be sent to the Board.

I am not sure about that, but I have a vague recollection of it. I think it would relate to the matters that you have been discussing with me here.

Q. I would note for the record, that Exhibit 5, the March 1, 1979, memorandum for Domenic B. Vassallo from Dudley Thompson, encloses a memo from N.C. Moseley to D. Thompson dated February 28, 1979, as well as a memo from J.S. Creswell to J. F. Streeter dated January 8, 1979, which has some enclosures.

As one of those enclosures of February 2?, 1979, memorandum for Dudley Thomoson from Norman C. Moseley, Subject: Notification of Licensing Boards, and which refers to Region 3 concerns requesting Licensing Boards to consider matters dealing with Babcock and Wilcox, which request the matters to be forwarded to the Licensing Boards even though there has been a negative determination by, apparently, Mr. Moseley's office, but that the originator of the concerns continues to believe the information should still be submitted to the Licensing Board.

Let me show you this February 2?, 1979, memorandum from Norman Moseley to Dudley Thomoson and ask you if it appears this forwards the concerns dealing with Davis-Besse and Midland that we are talking about?

A. It appears to, yes. Of course, it is an internal I&E document from someone in I&E to somebody else in I&E.

Q. Both Dudley Thomoson and Norman C. Moseley are individuals in I&E?

A. I believe so. It must be so indicated on the--

Q. Yes?

A. It says so.

Q. It is mentioned on the February 2?, 1979, memorandum?

A. Yes.

Q. Would this be the normal procedure, within I&E to forward a concern through your office to a Licensing Board?

To have it go through I&E first before it gets to your office?

A. You mean from the field?

Q. Yes.

A. I think, yes, that it would go from the region to the I&E headquarters and FROM I&E headquarters ~~to~~ TO NRR headquarters, WHICH would make the determination as to transmitting it through my office.

MR. SIDELL: Let's mark this as Exhibit 6, February 28, 1979, memorandum for Dudley Thompson from Norman C. Moseley, "Subject: Notification of Licensing Boards", which has an an enclosure also including the Creswell to Streeter, January 8, 1979, memorandum.

(The document referred to was marked for identification as Exhibit 6)

BY MR. SIDELL:

Q. Let me show you a document dated March 12, 1979, which is a memorandum for Domenic B. Vassallo, in NRR, from Dudley Thompson, in I&E, "Subject: Information for Board notification, Davis-Besse 1 and 2 and Midland 1 and 2, referring to a March 1, 1979, memorandum dealing with the same matters; and a March 7, 1979, memorandum for Dudley Thompson from Norman Moseley, both within I&E, "Subject: Board notification of Licensing Boards," referring to

February 27, 1979, memo, concerning the same matters.

I ask you if you have ever previously seen these, or whether these deal with the chain of events concerning notification of the Licensing Boards?

A. They would appear to be so related. I have no recollection of having seen them, though I might have in connection with my inquiry to why this office took so long in notifying the Board.

I note, however, that, again, it is correspondence between I&E and NRR in one case and between I&E and I&E on the other.

Q. And would such inter-divisional notification be consistent with getting the matter requesting Board notification of Licensing Boards through your office to those Licensing Boards?

A. I am not sure I follow that question.

Q. You referred to the fact that the March 12 and March 7 memorandum deal with notification and among NRR and I&E. My question is whether or not such internal, or inter-divisional, or inter-office notification is consistent with pushing the matter raised by a field inspector up through your office to a Licensing Board, if you know.

A. I don't know. I think the thing that stands out in my mind is that one office would have the lead in deciding whether or not the Board should be notified and these memoranda appear, to me, to be related to the process of whether or not a Board should be notified or not.

Now, once the lead office decides it wants to notify the Board, it seems to me that the way the system is supposed to work, is that that office, after it has made its mind up, transmits it my lawyers, who determine whether or not the material is material and relevant.

I have to add that my internal procedures provide ~~showed~~ that in the event that an office wants to notify the Board, under no circumstances is the information to be witheld. It is to be transmitted to the Board.

MR. SIDELL: Let's mark these memoranda as Exhibit 7 and 8. The March 7, 1979, memorandum for Dudley Thompson, Executive Officer for Operation Support, from Norman C. Moseley, Division of Reactor Operations Inspection, also I&E, "Subject Notification of Licensing Boards", as Exhibit Number 7.

(The document referred to was marked for identification as Exhibit 7)

MR. SIDELL: Let's mark as Exhibit 8 the March 12, 1979, memorandum for Domenic B. Vassallo from Dudley Thompson, Mr. Vassallo being Assistant Director for Light Water Reactors in NRR. Dudley Thompson is, again, the Executive Officer for Operations Support within I&E.

"Subject: Information for Board Notification, Davis-Besse 1 and 2 and Midland 1 and 2.

(The document referred to was marked for identification as Exhibit 8)

MR. SIDELL: Let's substitute as the previously marked Exhibit 3, dated March 29, 1979, a letter to "Ladies and Gentlemen" from Joseph F. Scinto, Deputy Director Hearing Division. The same document in addition to enclosures that are stated in the memorandum itself, which includes a March 6, 1979 memorandum for Edward S. Christenbury, previously marked as Exhibit 4.

March 1, 1979, memorandum for Domenic Vassallo, previously marked as Exhibit 5: the February 29, 1979, memorandum for Dudley Thompson, previsouly marked as Exhibit 6, as well as the January 8, 1979, memorandum for J. F. Streeter, Chief, Nuclear Support Section 1 from J.S. Creswell, Reactor Inspector, "Subject: Conveying new information to Licensing Boards, Davis-Besse Units 2 and 3 and Midland Units 1 and 2, which is a three page memorandum, which has attached to it a December 22, 1979, letter from Lowell E. Roe, to the Director of Nuclear Regulations, Mr. Robert Reid, which has appended to that a nine page analysis entitled " Additional Safety Evaluation of transient resulting from inability of operator to control steam generator level at 35 inches, which has appended to that a June 10, 1977, letter addressed to Mr. T.D. Murray, Station Superintendant, Davis-Besse Nuclear Power Station from F.R. Faist, Site Operations Manager of Babcock and Wilcox.

It also has appended to it an August 9, 1977, letter to Mr. T.D. Murray, previously referred to, Mr. Ivan D. Green, Site Operations Manager, also of Babcock and Wilcox, which has attached to that a sequence of events at SMUD, Sacremento Metropolitan Utility District, which is a three page chronology of events, Revision 1 dated 5-25-79, which includes three graphs or charts, which brings us to a March 29, 1979, memorandum for Domenic B. Vassallo, from Dudley Thomoson: Domenic Vassallo being Assistant Director for Ligth Water Reactors in NRR, Dudley Thomoson being Executive Officer for Operations Support, IE, and the reference 00 following his name on this memorandum. "Subject, Board notification, Davis-Besse Units 2 and 3 and Midland Units 1 and 2 with references to one memo, Thomoson to Vassallo dated March 3, 1979, and two, memo Thomoson to Vassallo dated March 12, 1979.

As enclosures to this memorandum, there is a memo from Mosely to Thomoson dated March 2, 1979, which it has enclosures and memo from Mosely to Thomoson dated March 29, 1979.

Let me ask you if you have ever seen this March 29, 1979, memo from Mr. Vassallo to Mr. Thomoson, or if that refers to the concern about Davis-Besse Units 2 and 3 and Midland Units 1 and 2, dealing with notification of the Licensing Board?

A. I have no recollection of ever having seen it. It would appear to so relate.

MR. SIDELL: Let's mark th e March 29, 1979, Thomoson to Vassallo memorandum as Exhibit 9.

(The document referred to was marked for identification as Exhibit 9)

BY MR. SIDELL:

Q. Let me ask you if you have previously seen what appears to relate to the same concerns we are discussing on March 29, 1979, memorandum for Dudley Thomoson, Executive Officer for Operations Support, IE, Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE, "Subject: notification of Licensing Board," referring to the TMI 2 incident of March 28, 1979, of this year.

A. No recollection of ever having seen it. It appears to be related.

MR. SIDELL: Let's mark this as Exhibit next in order.

(The document referred to was
marked for identification as
Exhibit 10)

BY MR. SIDELL:

Q. Finally, let me refer to a March 29, 1979, memorandum for Dudley Thompson, Executive Officer for Operations Support, IE from Norman C. Moseley, Director, Division of Reactor Operations Inspection, also of IE, "Subject: Notification of Licening Board", which states as follows: "On February 28, 1979, six items concerning Babcock and Wilcox designed nuclear plants were sent to you for forwarding to the appropriate licensing Boards. At that time only a preliminary evaluation had been done. We have completed our evalcation of each of the items and that information is enclosed. This additional information should be forwarded to the Licensing Boards." Thns is signed by Norman C. Moseley, which includes an enclosure of the evaluation of concerns raised by the January 8, 1979, memo from Creswell to Streeter.

Let me show you, Mr. ShapAr, the March 29, 1979, memo I have just referred to and aks you whether or not you have ever seen that. That refers to the matters we are now discussing with forwarding the matter to the Licensing Board?

A. (Pause) It appears to so refer. I have no his January 7, 1979, memorandum dealing with possible generic B&W problems originating at Davis-Besse?

A. I am really not in a position to answer that. I just don't know.

Q. Well, by considering the documents contained as part of Exhibit 12, would it appear that the concerns raised by Mr. Creswell finally found their way to the Licensing Board through several I&E offices, and in turn, several NRR offices, and in turn, from that to your office?

A. It would so appear.

Q. Let me refer to what has been marked as Exhibit 7 to this deposition, which is a memorandum from Norman Moseley to Dudley Thompson, which states in the first paragraph "In that memo, we committed to providing a written discussion and evaluation of each item within seven days.

Before we can complete the discussion and evaluation, additional information is needed from Region 3. Region 3 will be unable to provide the information until March 12, 1979. We will provide the complete write-up to you by March 16, 1979."

Would it appear from these memoranda, Mr. Shapar, that there was some confusion whithin I&E as to whether or not to refer the matter to the Licensing Board, or to come to any resolution about the possible problems raised by Mr. Creswell?

A. There would appear to be some difficulty, yes.

Q. Substantial difficulty?

A. Substantial difficulty, I would think.

Q. So it would appear, would it not, that Mr. Creswell's concerns originally relayed to NRC headquarters in his January 8, 1979, memorandum, requesting Licensing Board notification, that it took approximately two and a half months until March 29, 1979, memorandum, Exhibit 2 to this deposition, to produce that notification?

A. It would so appear.

Q. Would you consider that a standard length of time for notification of Licensing Boards based on a regional inspector's complaints?

A. I am not in a position to answer that because I don't see the Board notification usually. So, I am simply not in a position to answer that. I would think not, but I couldn't cite you any factual basis for that based on any personal observations.

Q. Referring to what has been marked as Exhibit 4 to this deposition, the March 6, 1979, memorandum for Edward Christenbury from D.B. Vassallo, concerning Board notification, reactor inspector's concerns regarding B&W plants.

Would this memorandum appear to be the first notification to your office of Inspector Creswell's concerns?

A. It would appear to be, but I can't be sure that I&E or NRR did not informally consult with members of my staff prior to the date of this memorandum.

Q. So, at least the concerns raised by Inspector Creswell's dealing with the Davis-Besse and Midland Units with possible generic consequences to other B&E plants, was in your office since March 6, 1979?

A. It would appear so, if not before.

Q. So, we have a time period of some 23 calendar days preceeding your March 29, 1979, memorandum, Exhibit 2 to this deposition, before there was any notification of Licensing Board as sought by Mr. Creswell?

A. It would appear so from these records.

Q. Do you have any reason to doubt the accuracy--

A. None.

Q. --of any of these documents?

A. None.

MR. SIDELL: Let's mark as Exhibit 13 the January 19, 1979, memorandum for Norman Moseley, Director of Division of Reactor Operations Inspection, IE, and H.D. Thornburg, Director, Division of Reactor Construction Inspection, also IE, from James G. Keppler, Director of Region 3, concerning the "Subject: recommendation for notification of Licensing Boards and request for technical assistance", which is a three page memorandum, including a three page memorandum from J.S. Creswell, Reactor Inspector for J.F. Streeter, Chief, Nuclear Support Section 1, dated January 9, 1979, which includes a similar analysis as included in Exhibit 12, provided by Toledo Edison.

(The document referred to was marked for identification as Exhibit 13)

BY MR. SIDELL:

A. I ask you if it appears that the notice from Mr. Keppler was the mechanism by which Mr. Creswell's concerns about Davis-Besse and Midland got to headquarters from Region 3?

A. (Pause) It would appear so.

Q. So, to attempt to construct the sequence of events with the memorandum dealing with Mr. Creswell's concerns of Davis-Besse and Midland, we have, first, Mr. Creswell's January 9, 1979, memo to his immediate supervisor, Mr. Streeter, in Region 3, which produces a memorandum from Mr. Keppler, who is the Director of Region 3, to both Mr. Moseley and Mr. Thornburg, individuals in headquarters I&E in Bethesda, which in turn produces a memo on February 29, 1979, from Mr. Moseley to Mr. Thompson concerning notification of Licensing Boards, which is Exhibit 6 to this deposition, the last paragraph which states "We will provide a written discussion and evaluation of each item within seven days of the date of this memorandum", which appears to be signed by an E.L. Jordan for Norman Moseley.

Do you know whether or not E.L. Jordan is Edward Jordan, who at the time, was Assistant Director for Mr. Moseley?

A. I do not.

Q. Do you know Mr. Jordan?

A. I do not.

Q. Based on Exhibit 6 and the information contained in it, it would appear, would it not, that Mr. Moseley or someone on his staff would have completed this analysis of Mr. Creswell's concerns by March 7, 1979, last paragraph?

A. Yes, it would appear so.

Q. And in Exhibit 5, the March 1, 1979, memo, we see that Mr. Vassallo has received from Mr. Thompson Mr. Moseley's forwarding of Mr. Creswell's concern, which requested to be informed of what is going on with the problem, does it not?

A. It would so appear.

Q. And on March 6 we have a memorandum from Mr. Vassallo to Mr. Christenbury, of your office, forwarding Mr. Creswell's concerns, who in turn forwarded Mr. Keppler's concerns, who in turn forwarded Mr. Streeter's concerns.

And, therefore, finally forwarding Mr. Creswell's original concerns about Davis-Besse requesting notification of Licensing Boards, which in the last paragraph of Exhibit 4. "When we received the IE written evaluations, we will review them to determine whether additional review should be provided by DSS. In any event, we will follow this up with additional information for the Board in the near future".

Does it appear from Exhibit 4, Mr. Shaoar, that we are moving the matter along?

A. (Pause) Yes.

Q. In Exhibit 7 to this deposition, we have Mr. Moseley sending Mr. Thompson a memorandum saying that Region 3 needs more time and therefore we will need more time to evaluate the concerns originally promoted by Mr. Creswell, is that correct?

A. (Pause) Yes.

Q. ANd, it appears from Exhibit 8 to this deposition, of March 12, 1979 memorandum from Mr. Thompson to Mr. Vassallo, we are going back down the chain from IE to NRR, providing notice to NRR that "we have been informed by the enclosed memorandum that delays in getting certain information have caused us to change our submittal date to 3-17-79".

Would it appear, then, from Exhibit 8, Mr. Shaoar, that the concerns about Davis-Besse 1 and 2 and Midland 1 and 2 are preceding back down the ladder of command?

A. Well, the ladder of command may not be the right terminology. It shows that it is going from IE back to NRR in the person of Vassallo.

Q. Which is the person, or who is the person who previously provided the information to Mr. Thomoson?

A. Indeed.

Q. And by virtue of Exhibit 11, dated March 28, 1979, from Mr. Moseley to Mr. Thomoson concerning the same subject, there is now discussion and evaluation provided by someone within IE concerning Mr. Creswell's concerns, and therefore appears to resolve further IE involvement in notification of the Licensing Board, is that correct?

A. It would appear so.

Q. March 28, 1979, is also the date of the accident at TMI 2, is it not?

A. I believe so.

Q. The same date that Mr. Moseley is providing Mr. Thomoson with an evaluation of Mr. Creswell's concerns.

Let me refer you to Exhibit 11, the March 28, 1979, memo from Mr. Moseley to Mr. Thomoson, to the sixth page of the exhibit, which is a discussion and evaluation of Mr. Creswell's third of six concerns which dealt with a pressurizer level indications going off scale at Davis-Besse on November 29, 1977 due to a loss of off site power, wherein the following statement is made, "The events at Davis-Besse which resulted in loss of pressurizer level indication has been reviewed by NRR, and the conclusion was reached that no unreviewed safety questions existed."

Let me ask you whether or not it appears from that discussion and evaluation that the basic reason for providing notice to the Licensing Board was based exclusively on Mr. Creswell's concerns and not on evaluation by NRR of his concerns at Davis-Besse?

A. I would have to review the documents in greater detail to give any--

Q. Do you want to take a few minutes--

A. --any reasonable answer to that question.

Q. Do you want to take a few minutes and look at that section, at least?

A. Only if you will repeat the question.

Q. [illegible] r or not it appears that the review--strike that. The notification to the Licensing Board is based exclusively on Mr. Creswell's persistance in maintaining his concern s of possible safety problems at Davis-Besse rather than NRR's conclusions to the same effect?

A. (Pause) I still don't feel I can give a responsible answer to that without going back and reviewing documents relating to Mr. Creswell's concern with which I am not familiar, to be able to reach an affirmative answer that it is based exclusively on Creswell's concerns.

Q. Well, based on your reading of Exhibit [illegible], and the reference I have indicated, would it appear as though

NRR would pursue a request for a notification of a Licensing Board dealing with the loss of pressurizer level indication which occurred at Davis-Besse on November 29, 1977?

A. Again, I don't feel competent to answer that question. I am not familiar with these documents and I would feel very unsure of myself in giving a direct answer to that question.

I think if I took the time to read those documents carefully I could reach a conclusion, but I haven't.

Q. Well, has it been your experience that NRR, for instance, would pursue notification of Licensing Board, if they had previously concluded that there was no unreviewed safety questions?

A. I have practically zero experience of being personally involved in Board notification. They don't come through my office and I don't see them. I just don't know.

I know how the system is supposed to work, and that there is a lead office. This particular circumstance seemed to involve an origin outside of NRR, which I guess is not usually the case. How the interaction works between IE and NRR I just have no experience with it.

Now, my attorneys would, the people who are actually reviewing the transmittals, but I don't think I have seen more than one or two transmittals in my work experience here.

Q. Well, let me ask you if Licensing Boards are not concerned with reviewing potential safety concerns?

A. Yes, but several facts have to be added to that. This Board notification system is rather unusual. There is a proceeding going on and there are litigated issues. There are issues in contention and the usual course in any Federal District Court that I know of, or state court, or most other administrative agencies, is the parties decide what evidence they want to put on and if they think it is relevant to their case and will help they will put it on. If they don't they don't.

This Board notification system is over and beyond that, and irrespective of any strict construction of what is being litigated; in essence, anything that is relevant or material should be transmitted.

So, the answer to your question is yes. It is complicated further by the fact that the Boards have sua sponte authority to raise issues. I think that is directly relevant to your question.

Now, they are not supposed to do it unless it is a major safety issue, as I recall, the guidance FROM the Appeal Board AND the Commission, and that sort of thing.

Q. What is the definition, if one exists, of a major safety issue?

A. I am not sure that one exists that would have been of any meaningful help, but I merely say this by way of background.

I think it kind of furthers the thought that you were expressing of whether the Boards have a safety function to perform. They certainly do, and I am pointing THIS out not only with respect to the issues that are placed squarely in contention ~~with~~ BY the parties, but they have this over-riding sua sponte authority to raise issues on their own.

I would think there is some quite direct connection between the ~~adequacy~~ EFFICACY of the Board notification procedure and the sua sponte authority of the Hearing Boards as well as the Appeal Board.

Q. Well, it appears, based on the fact that Mr. Creswell's concerns raised origanally in its January 8, 1979 memo to his immediate supervisor, Mr. Streeter, got to the Hearing Board, does it not?

A. It does.

Q. So, there appears to be a procedure where by a regional inspector can raise concerns before a Hearing Board or Licensing Board, even though it clearly is not related with the utilities involved and would not be an active party to an on-going proceeding as long as his concerns deal with a possible safety question?

A. I guess there is a standard of relevant and material as part of the established notification procedures. I guess one could ask the question how would an inspector in the field necessarily know, with respect to a proceeding WITH which he is not intimately involved, whether or not the information is relevant or material.

Q. Do you happen to have an answer to that question?

A. No, I don't. My OWN personal inclination is: when in doubt--I have tried to establish THIS IN my own procedures— when in doubt, send it.

Q. Which appears to be precisely what Mr. Creswell did?

A. Yes.

Q. Is it not?

A. Well, he may have--I don't know. He may have thought that it, the generic importance of it is so important that it was material. I wouldn't discount that either. There may not have been a material--I don't know whether he focused on relevancy and materiality, but I won't discount the possibility that he might have.

Q. But in any event, the problem that he perceived was of a significantly high level to evidently quite persistently pursue his concerns to the Licensing Board?

A. Yes.

Q. Through two divisions of NRC as well as your office?

A. Well,--

Q. Once the item got on that track?

A. Well, Creswell isn't the one WHO sent it to our office, was HE?

Q. He was the one who originally raised the problem.

A. Yes.

Q. That started the process.

A. I think your question IS: was he persistent enough to pursue it with two offices and then you added a third. I don't know if his persistance was directed at our office rather than his own office AND ~~to~~ NRR.

I think one of the things you would want to do would be to try and marry the established procedures with the curcumstances that took place. Where, if any place, did the procedures break down. I suppose THERE IS also the question, is the procedure good enough.

Q. Well, would you venture a conclusion based on the exhibits presented in this deposition as to whether or not the procedure is good enough?

A. Well, I would say that if the information meets the substantive standards for information to be transmitted it got ~~their~~ THERE too late. Not too late, but it didn't get there fast enougn, is a better way of putting it.

Q. Well, do you have any--

A. I have to qualify that by saying I don't know ~~that~~ the degree of difficulty that was reasonably needed or involved to make the determination that it ought to be transmitted.

I couldn't make that distinction without reviewing the documents in detail and looking at the issues in the cases.

Q. Well, if all along the way people in headquarters, in both IE as well as NRR, made determinations that there was no safety question and therefore they saw no reason to pursue notification of Licensing Boards, but Creswell, himself, persisted.

Evidently, there is a regulation manual Chapter 1530, I believe, which allows an originator of a concern, who persists in concluding that a safety question after there has been a review of his concern to be entitled to raise the matter before a Licensing Board.

It appears the procedure exists, does it not, for a disatisfied inspector in a region to have a resolution on a question regardless of any contrary resolution by everyone else in the system?

A. Well, without looking at the precise words of the procedure, I would think that it is encumbant on the office of which that person is a member to notify the Board that there is a technical opinion within that organization that raises ~~that~~ a safety question.

I think there are two problems here, there are two questions that need to be answered. Number one, is there a genuine safety question. Number two, is it material and relevant to the issues in the case or to the proceeding.

Q. Are you familiar at all with the events that occurred at Davis-Besse on November 29, 1977?

A. I am not.

Q. Well, let me represent to you that among other problems there was loss of pressurizer level indication off scale low due to the loss of off-site power, which Mr. Creswell concluded might be generic to other B&W plants and might require further review.

I am sure you are aware of some of the specifics that occurred at TMI 2 of this year?

A. Right.

Q. And the fact that pressurizer level indication went off scale, although in this instance on the high side. Again, coming to the general conclusion that pressurizer level indication, as recognized by others previous to the accident, is not always an accurate or clear indicator of core inventory.

In view of the concern on the part of the Licensing Board for relevant and material information dealing with safety related matters, and based on Mr. Creswell's concerns of a possible generic B&W problem with pressurizer level indication, would it not appear as though Mr. Creswell had a relevant and material concern about pressurizer level indication in its operation at B&W reactors?

A. It would appear so, but my qualifications for making that response are distinctly impeachable.

Q. Well, I believe earlier you stated that whether or not Mr. Creswell's concerns got to the Licensing Board, if there were a safety problem of some significance on time, and it was that they had not, but before you said they had not I believe you indicated it had not prevented a problem by not getting to the Licensing Board sooner that it did.

You corrected that to say that it merely had gotten to the Board in a substantial period of time.

A. That it could have reached the Board faster.

Q. And, if it could reach the Board faster, knowing what what we do know about Mr. Creswell's concerns about the accident at TMI 2, would you conclude it might have been possible to prevent the accident at TMI 2 if there had been speedier resolution along the chain of events.

A. I can't answer that.

Q. As a result of the delays in your office, at least of pushing Mr. Creswell's concerns through to the Licensing Board, I believe you indicated you have modified your procedures in this office?

A. Yes, and I will get you a copy of the modified procedures.

Q. Would I be correct in concluding that as an essential result of those modifications, you are trying to speed up the process by which questions can get to Licensing

Board?

A. Yes, mainly by removing this office from the chain. It would seem on the theory that if an office believes that it has information which ought to be transmitted to the Board, I would not rely on strict legal principles of materiality and relevancy to withhold the information from the Board. I would send it over there.

Q. Essentially to get the information to the Board and let them determine if they want to do anything about it?

A. That is correct.

Q. In an attempt thereby to eliminate the confusion that appears to have resulted from getting Mr. Creswell's concerns through both IE and NRR?

A. No. No. Even if this office had not been involved at all. It is perfectly apparent to me from that correspondence that most of that time would still have been involved unless something else is changed, mainly to eliminate any potential delay attributable to my office.

I would err on the side of transmitting irrelevant and immaterial stuff to the Board, if I have to err.

Q. Do you know whether or not any procedural changes that have been proposed as a result of TMI 2, and this particular situation that would speed up the process of getting inspector's concerns to the Licensing Board?

A. I don't know of any.

Q. Are you in a position to make such suggestions for other divisions in NRC?

A. I suppose I am in a position to make any suggestion, but I would think that the operating offices would take the lead in such a thing.

Q. Well, in view of the fact that we evidently have two operating offices involved in the particular problem, IE as well as NRR, and the problem involving them, eventually got to your office, your office has a bit better perspective in seeing the back and forth between the two, does it not?

A. Yes, but I am afraid the attorney's are looking at it from the narrow perspective from the way the procedures are run out.

If you are looking ~~out~~ for a broader perspective in the fact that more than one office is involved, I think the lead should come from the Executive Director For Operations ~~of the Office~~.

Q. That would be Mr. Gossick?

A. Yes.

Q. Do you know whether or not he is aware of this substantial change of memorandum dealing with Mr. Creswell's concerns?

A. I don't know, but I do know he is aware because I apprised him of my unhappiness with the continuing role of my office in reviewing the materiality and relevancy considerations on the transmittal chain.

Q. I believe you indicated previously you are unaware of any proposed change in procedures as they might affect NRR and IE.

A. Well, only the revision that I have suggested ~~which~~ is that they make the transmittal directly. It certainly affects them as well as affecting me.

Q. But on its face, it does not eliminate the potential for a similar situation to the one we are discussing from happening again, does it?

A. No, but I think there are two problems here. I think you have to decide what goes to the Board. You can't say that everything goes to the Board, and that is part of the problem.

It was part of the problem in developing the original procedures. I mean, there are mountains of stuff generated every day. I think it would defeat the system, because you would inundate the Board with unscreened information.

Q. Is that the reason for the relevancy and materiality requirement?

A. I think so, I think so. So, what you have to do is recognize the genuine need for the Board to get the information. That is the easy point.

It is good policy in my opinion but what goes? If you give them every bit of information generated around here, they won't know how to discriminate, and the staff will be developing analyses on every bit of information.

As a matter of fact, the NRR has been criticized in a recent Appeal Boards decision for sending over stuff without any analysis. Well, there is a tension here because if they send the stuff immediately they get criticized because they haven't had time to develop an analysis.

Q. And if they take sufficient time to require them to perform adequate analysis, they may be caught in a situation such as we are discussing?

A. Exactly.

Q. Catch 22?

A. Catch 22.

Q. Would you feel that having been informed, at least during this deposition, of the chain of memoranda originating with the January 9, 1979 Creswell concerns through your March 29, 1979, memorandum to your staff, you would be in a position to propose suggestions to either IE or NRR or the Executive Director for streamlining matters such as this in the future?

A. Of course, I have made my proposal to streamline it to some extent.

Q. Which involves your office alone?

A. Which involves my office alone and would provide for direct transmittal. I guess I am not in a position to make suggestions at this time, because I am trying to transmit this experience.

What I don't have a feel for is how difficult a decision this was made to make from a technical standpoint and to whether or not the Board ought to see that. I just don't know that.

Under any system you can have difficult questions WHICH take time to resolve, but I don't really have a feel for all these memoranda that ~~was~~ WERE floating ABOUT. Were they really needed or was it buck passing, to make my words rather blunt. I can't tell you without having a feel for how genuine the basic substantive issue was.

Q. Are you aware of something referred to as the Michaelson memorandum?

A. I have heard about it. I don't know what is in it and I haven't seen it.

Q. Do you know the date of it?

A. No. I understand he was a consultant at TVA.

Q. And to the ACRS?

A. Yes.

Q. Have you also heard of something referred to as the Novak memoranda?

A. No.

Q. Would it appear from your perspective that based on the chain of memoranda taking approximately two and a half months, there should be some substantial organizational changes within either IE and NRR?

A. Again, a lot would depend on how difficult the substantive issue was. I have a gut feeling that no matter how difficult it was, it took too long for the decision making process to run its course, but that is just a gut feeling without knowing how difficult the substantive issue was.

It would seem to me that if my gut feeling is right, that the process took too long, no matter how difficult, that I am not in a position to say that organizational changes are necessarily the solution, although they might be.

Another problem is to take the bull by the horns the way I did and say that you will get it to the Board by a certain defined date no matter how ~~much~~ MANY offices are involved, and you can be that blunt.

It merely says that if you are not sure you send. That is the net result of it, but you may pay a price for that. I don't know how many difficult ones there are. It may mean giving the Board a lot of information, some of which they are going to have to spend their time going through needlessly because the staff hasn't done its job of screening it for them.

So, you have really got a policy choice there. Who do you want to do the screening? Are the Boards really equipped to do the screening then they are basically to resolve the controverted matters, or do you want to develop some kind of standard for the staff that will provide a legitimate and reasonable basis for doing a proper job of screening. Then, if they don't do their job then you know how to take care of that, but I am not sure that organizational changes are the answer. It may be a tougher attitude towards the time frame work.

Q. You, in your immediately preceding answer, indicated that you might have to pay a price for sending too many questions to Licensing Boards.

A. You are making them to the extent that there hasn't been screening. By definition you are sending them stuff that shouldn't go if you had time to screen it properly and thereby you are diverting them from focusing on genuine safety issues to take the time to screen, and some of which will screen out.

How, who is best equipped to do that? That is something about which reasonable men could differ.

Q. Well, on the basis that Mr. Creswell's concerns about loss of pressurizer lever indication in a B&W reactor were construed to be valid safety concerns of a rather significant proportion. Who paid the price in a delay of relaying his concerns to the Licensing Board?

A. I think the system paid the price, and the public paid the price because to the extent that valid safety concerns were witheld from the Board and the process isn't working properly. Therefore, you are not getting the best decision, which is the one thing the public is entitled to have.

Q. That being a proper and prompt correction of an existing safety defect?

A. Now, wait a minute. The Board is not going to be dealing with generic questions. It is going to be dealing with its decision in the individual case.

You are talking about initial licensing, about whether or not a license should be issued. My point was more limited. My point is before they decide that the applicant ought to be authorized to start to build, or to start to operate, that if the decision can be impeached because it is not based on the best safety information, then that cuts against safety, which I think is the ultimate sin in this business.

Q. If, then, the Licensing Boards are not in a position to resolve generic safety questions, but they find one based on a plant specific problem such as may have existed at Davis-Besse, with loss of pressurizer level indication, is there any formalized proceedure for the Licensing Board to information any one else in the NRC of a possible generic problem?

A. I don't know of any formal proceedure, but the Board in the past, have pointed out matters that might be regarded ad dictum, and they could write letters to the Commission or to the staff.

I guess, you have to take into account what the role of the Board is and perhaps what it ought to be. They are there to essentially resolve controverted matters in a given case as to whether a given reactor ought to be allowed to be constructed or operated.

There are other elements of the overall organization that are supposed to deal with generic matters. I mean, THE BOM can't do everything.

Q. Well, is NRR supposed to resolve generic matters if it finds them?

A. Yes.

Q. Does it appear from the chain of memorandum originating with Mr. Creswell that NRR properly concluded they could have resolved a possible generic problem with pressurizer level indication loss?

A. I can't answer that question. I am simply not familiar with the substantive technical details on that correspondence.

Q. Well, if NRR were to conclude that there was a generic problem raised by Mr. Creswell originating at Davis-Besse, would it not have been reasonable for them to recommend a Licensing Board review rather than to not recommend one?

A. Well, their decision was whether or not this information was relevant and material under existing standards, proceedures to be transmitted to the Board in the context of this particular case, which I think is a different question from how NRR would approach a generic question.

Q. Well, NRR in its review concluded--

A. This information was material and relevant enough to go to the Board.

Q. Was that the conclusion they actually made or did they forward it on the basis of a manual chapter where the originator's insistence was the force that got the matter to the Licensing Board?

A. I am not sure of the answer to that question, but I assume they sent it to the Board in conformance with existing proceedures for Board notification. I am assuming that, which speaks for themselves.

MR. SIDELL: At this time I have run out of questions and documents. What we have been doing in the past is to rather than adjourn the deposition, merely to recess them in case, in the hopefully unlikely event we come up with more questions and we can merely continue on.

MR. SHAPER: Sure.

MR. SIDELL: We will do that in this case. I doubt whether or not we will find it necessary to reconvene this deposition. I would tell you that we have not done that to date with the number of depositions we have taken. We plan to do it in a very finite number of situations.

Therefore, it's most unlikely it will happen in your case, but the possibility is not completely eliminated.

MR. SHAPER: At any time.

MR. SIDELL: After almost three hours of testimony, do you have anything else at this point that you would like to correct or change?

MR. SHAPER: I can't think of anything.

MR. SIDELL: As I stated originally, we will provide you with a copy of your deposition to review and change if you feel the necessisty, but I am primarily concerned at this point with more substantive matters than typographical matters or things of that nature.

MR. SHAPER: Sure.

MR. SIDELL: All right. Thank you very much for your time.

(The deposition was recessed at 1:15 p.m.)

Page 53 is missing.
HKS
EXHIBIT #2 IS NOT LEGIBLE.
HKS

I have read the foregoing pages,* 1 through 80, and they are a true and accurate record of my testimony therein recorded.

Howard K. Shaper
HOWARD K. SHAPER

Subscribed and sworn to before me this 16th day of August 1979.

Marilyn Jollerstere
Notary Public

My Commission expires: July 1, 1982

REPORTER'S CERTIFICATE

DOCKET NUMBER:

CASE TITLE: Accident at Three Mile Island

HEARING DATE: August 3, 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: August 6, 1979

Suzanne Conrad

Official Reporter
Acme Reporting Company, Inc.
1411 K Street, N.W. Suite 600
Washington, D.C. 20005

Shaper Exhibit #1
8-3-79 sc

BIOGRAPHICAL OUTLINE

HOWARD K. SHAPAR, B.A., Amherst College, J.D., Yale Law School; Executive Legal Director, U.S. Nuclear Regulatory Commission; member of bars of State of New Mexico, Court of Appeals for District of Columbia Circuit, and U.S. Supreme Court; past president, Los Alamos County (New Mexico) bar association; vice-president, International Nuclear Law Association, past chairman, atomic energy law committee, World Peace Through Law Center; past chairman, Committee on International Uses of Atomic Energy (International Law Section), American Bar Association; past vice-chairman, Committee on Energy (Administrative Law Section), American Bar Association; past chairman, atomic energy law committee, Federal Bar Association; author of numerous articles in the field of nuclear law.

8-3-79 .5c

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[illegible]
[illegible]
S. Tracy

[illegible]

[illegible] Bradford's office.

[illegible]

[illegible]

[illegible] Chief Hearing Counsel's office.

#1

50-320

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 29, 1979

BOARD NOTIFICATION

Re:	Davis Besse	Docket Nos. 50-500, 50-501
	Erie	Docket Nos. STN 50-580, STN 50-581
	Greene County	Docket No. 50-549
	Midland 1 & 2	Docket No. 50-329 OL, 50-330 OL
	Pebble Springs	Docket Nos. 50-514, 50-515
	Three Mile Island 2	Docket No. 50-320

Ladies and Gentlemen:

Enclosed for the information of the Boards is a recent memorandum relating to certain concerns raised by a reactor inspector in Region III concerning the Davis Besse and Midland units. We are informing the Boards with respect to Davis Besse 2 and 3 and Midland 1 and 2. We are also providing information to the Boards in connection with Erie, Greene County, Pebble Springs, and Three Mile Island 2 since those facilities have similar Babcock & Wilcox reactor units.

Sincerely,

Joseph F. Scinto
Deputy Director, Hearing Division

Enclosure
As Stated

Distribution: (see attached list)

7908110377 [redacted]

Shaper Exhibit #3
8-3-79 sc

Distribution:

Copies of a "Board Notification" letter dated March 29, 1979, signed by Joseph F. Scinto have been served on the following persons. Those whose addresses are at the U.S. Nuclear Regulatory Commission have been served by the NRC internal mail system and others have been served by deposit in the U.S. Mail. One copy has been served on each person even though his or her name appears on more than one service list. In addition to copies served on Atomic Safety and Licensing Board and Atomic Safety and Licensing Appeal Board members identified on the service list, 5 copies of the cover letter for each captioned proceeding and 5 copies in total of the attachment have been provided to the Atomic Safety and Licensing Board Panel, and 1 copy of both cover letter and attachment has been provided to the Atomic Safety and Licensing Appeal Board Panel.

In the Matter of

THE TOLEDO EDISON COMPANY, *et al.*

(Davis-Besse Nuclear Power Station, Units 2 and 3)

Docket Nos. 50-500
50-501

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Office of the Secretary
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In the Matter of	)	
	)	
OHIO EDISON COMPANY, _et al_.	)	Docket Nos. STN 50-580
	)	STN 50-581
(Erie Nuclear Plant, Units 1 and 2)	)	
	)	

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In the Matter of POWER AUTHORITY OF THE STATE OF NEW YORK (Greene County Nuclear Power Plant)	))) Docket No. 50-549)))

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Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing
Appeal Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of)
)
CONSUMERS POWER COMPANY) Docket Nos. 50-329 O.L.
) 50-330 O.L.
(Midland Plant, Units 1 and 2))

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In the Matter of PORTLAND GENERAL ELECTRIC COMPANY (Pebble Springs Nuclear Plant, Units 1 and 2)	)))))))	Docket Nos. 50-514 50-515

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Washington, D.C. 20555

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of

METROPOLITAN EDISON COMPANY, et al.) Docket No. 50-320

(Three Mile Island Nuclear Station, Unit No. 2)

Alan S. Rosenthal, Esq., Chairman
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Atomic Safety and Licensing
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U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Snaper Exhibit #4
8-3-79 Sr

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 6, 1979

MEMORANDUM FOR: Edward S. Christenbury, Hearing Division Director and Chief Counsel, OELD

FROM: D. B. Vassallo, Assistant Director for Light Water Reactors, Division of Project Management, NRR

SUBJECT: BOARD NOTIFICATION - REACTOR INSPECTOR CONCERNS REGARDING B&W PLANTS (BN-79-10)

The enclosed memorandum from I&E provides information originated by a Reactor Inspector as Board Notification material. Although I&E concluded that the information was not relevant and material the originator still believes that Boards should be informed.

Since we have not yet received I&E's written discussion and evaluation of these matters we have not reviewed the material in any detail. Regardless, however, in accordance with established procedures the information should be provided to appropriate Boards based on the originator's concerns.

The originator recommends that the Davis Besse 2 & 3 and Midland 1 & 2 Boards be informed.

In neither case is the SER Supplement issued but we have no objection to providing the information. In addition, since the concerns appear to apply to B&W plants, we recommend that you also provide the information to the Erie, Greene County, Pebble Springs and TMI-2 Boards.

When we receive the I&E written evaluations we will review them to determine whether additional review should be provided by DSS. In any event, we will follow this up with additional information for the Boards in the near future.

D B Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

Enclosure:
As stated

cc: See attached sheet

~~7905110383~~ 7905160185 P

cc: H. Denton
E. Case
D. Eisenhut
J. Davis
R. Boyd
V. Stello
R. DeYoung
L. Nichols
B. Grimes
J. Stolz
R. Baer
O. Parr
S. Varga
IE (7)
E. Jordan
D. Thompson

Shafer Knight -
8-3-79 SC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 01 1979

MEMORANDUM FOR: Domenic B. Vassallo, Assistant Director for Light Water Reactors, NRR

FROM: Dudley Thompson, Executive Officer for Operations Support, IE

SUBJECT: INFORMATION FOR BOARD NOTIFICATION - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2

The enclosed information is being forwarded for Board Notification. Your contact on this matter for any additional information is E. L. Jordan, ext. 28180.

Please note that the 2/28/79 cover memorandum, Moseley to Thompson, states that the originator, after being informed of IE Headquarters evaluation, still believes the information should be sent forward to the boards.

We request to be informed of your disposition on this matter.

Dudley Thompson
Dudley Thompson
Executive Officer for
Operations Support, IE

Enclosures:
1. Memo NCMoseley to DThompson dtd 2/28/79
2. Memo JSCreswell to JFStreeter dtd 1/8/79 w/enclosures

cc: N. C. Moseley, ROI w/o encls
E. L. Jordan, ROI w/o encls
J. F. Streeter, RII w/o encls
J. S. Creswell, RII w/o encls
G. C. Gower, XOOS w/encls
IE Files w/encls

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SHAREC Ex. -6
8-3-79 SC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 2 8 1979

MEMORANDUM FOR: ~~Dudley~~ Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS (AITS F30468H2)

Enclosed are six items sent in by Region III for forwarding to sitting Licensing Boards for cases involving Babcock and Wilcox as the Nuclear Steam System Supplier. Our preliminary evaluation indicates these items do not appear to be new issues or to put a different light on the issues and therefore, in our opinion, do not meet the intended criteria for Board notification.

The originator was informed, via telephone, of this determination on February 27, 1979. His position was that our evaluation did not provide any information that he did not already have and his concern was whether or not these items had been considered and resolved on a generic basis for all B&W plants. Because of this he still believed the items should be sent to the Licensing Boards. IE Manual Chapter 1530 requires that if, after a negative determination, the originator continues to believe that the information should be submitted to the Board(s), the information will be submitted. We therefore request the enclosed items be sent to the appropriate Licensing Boards.

We will provide a written discussion the evaluation of each item within seven (7) days of the date of this memorandum.

for Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

Enclosure:
Memorandum Creswell to Streeter
dated January 8, 1979

cc w/o encl:
S. E. Bryan
E. L. Jordan
J. Kirkpatrick
J. C. Stone
G. C. Gower
R. F. Heishman, RIII

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7904210475

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

(5)

MAR 7 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

On February 28, 1979, six items involving Babcock and Wilcox plants were sent to you for forwarding to the appropriate licensing boards. In that memo, we committed to providing a written discussion and evaluation of each item within seven days.

Before we can complete the discussion and evaluation, additional information is needed from Region III. Region III will be unable to provide the information until March 12, 1979. We will provide the complete write-up to you by March 16, 1979.

Norman C. Moseley

Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

cc: R. F. Heishman, RIII
S. E. Bryan
E. L. Jordan
D. Kirkpatrick
J. C. Stone
~~G. S.~~ Gower

CONTACT: J. C. Stone
(x28019)

SHAPER Exhibit #7
8-3-79 sc

(6)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 1 4 1979

MEMORANDUM FOR: Domenic B. Vassallo, Assistant Director for Light Water Reactors, NRR

FROM: Dudley Thompson, Executive Officer for Operations Support, IE

SUBJECT: INFORMATION FOR BOARD NOTIFICATION - DAVIS BESSE 1 & 2 AND MIDLAND 1 & 2

By memorandum dated 3/1/79 we provided information for Board notification on the subject plants and indicated that a written discussion and evaluation would follow in seven days. We have been informed by the enclosed memorandum that delays in getting certain information have caused us to change our submittal date to 3/17/79.

Dudley Thompson

Dudley Thompson
Executive Officer for
Operations Support, IE

Enclosure:
Memo NCMoseley to DThompson
dtd 3/7/79

cc w/o enclosure:
J. C. Stone, ROI
R. F. Heishman, RIII

SHAPER Exhibit #8
8-3-79 5-

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

(9)

MAR 29 1979

MEMORANDUM FOR: Domenic B. Vassallo, Assisant Director for Light Water Reactors, NRR

FROM: Dudley Thompson, XOOS

SUBJECT: INFORMATION FOR BOARD NOTIFICATION - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2
REFERENCES: 1. Memo: Thompson to Vassallo dtd 3/1/79
2. Memo: Thompson to Vassallo dtd 3/12/79

As noted in the above referenced submittals additional information in the form of staff discussion and evaluation would be forthcoming on the captioned board notification. Enclosed is the additional information for submittal to the appropriate Boards.

Dudley Thompson, Executive Officer
for Operations Support
Office of Inspection and Enforcement

Enclosures:
1. Memo: Moseley to Thompson dtd 3/28/79 w/encls
2. Memo: Moseley to Thompson dtd 3/29/79

cc: N. C. Moseley, IE, w/o encl
S. E. Bryan, IE, w/o encl
J. F. Streeter, RIII, w/encl
J. S. Creswell, RIII, w/encl
G. C. Gower, IE, w/encl
IE Files w/encl

CONTACT: G. C. Gower, IE
49-27246

SHAPER Exhibit #9
8-3-79 SC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

(8)

March 29, 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

In light of the transient experienced at Three Mile Island on March 28, 1979, we will review our preliminary evaluation of Item 3 contained in my March 28, 1979 memorandum to you. It is possible that the additional information will cause our assessment to change.

Norman C. Moseley

Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

cc: S. E. Bryan
E. L. Jordan
R. F. Heishman, RIII
J. C. Stone
D. C. Kirkpatrick
G. C. Gower
V. D. Thomas

SHAPER EXHIBIT #10
8-3-79 SC

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

(7)

MAR 2 8 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

On February 28, 1979, six items concerning Babcock and Wilcox designed nuclear plants were sent to you for forwarding to the appropriate licensing boards. At that time only a preliminary evaluation had been done. We have completed our evaluation of each of the items and that information is enclosed. This additional information should be forwarded to the licensing boards.

Norman C. Moseley

Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

Enclosure:
Evaluations of Concerns

cc: S. E. Bryan
E. L. Jordan
R. F. Heishman, RIII
J. C. Stone
D. Kirkpatrick
~~G. C.~~ Gower
V. D. Thomas

SHAPER Exhibit #11
8-3-79 SC

CONTACT: J. C. Stone
(x28019)

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

1. During a recent inspection at Davis-Besse Unit 1 information has been attained which indicates that at certain conditions of reactor coolant viscosity (as a function of temperature) core lifting may occur. The licensee informed the inspector that this issue involves other B&W facilities. The Davis-Besse FSAR states in Section 4.4.2.7:

> The hydraulic force on the fuel assembly receiving the most flow is shown as a function of system flow in Figure 4-39. Additional forces acting on the fuel assembly are the assembly weight and a hold down spring force, which resulted in a net downward force at all times during normal station operation.

The licensee states that there is a 500°F interlock for the starting of the fourth reactor coolant pump. However, no Technical Specification requires that the pump be started at or above this temperature. A concern regarding this matter would be if assemblies moved upward into a position such that control rod movement would be hindered.

DISCUSSION AND EVALUATION

The potential for core lifting in B&W plants is a concern which has been previously reviewed by NRR. The concern was first raised in connection with the Oconee 2 and 3 reactors, where the primary coolant flow rates were found to be in excess of the design flow rates. For example, the Unit 2 flow rate was found to be 111.5% of the design flow rate. Since this was very near the predicted core lift flow rate of 111.9%, an analysis was done by B&W to determine what effect core lifting would have on the previous safety analysis for these plants. This analysis (dated May 2, 1975) indicated that the potential for core lifting did not result in an unreviewed safety question. A subsequent review of this B&W analysis by NRR also concluded that an unsafe condition did not exist (letter from R. A. Purple to Duke Power, dated 9/24/75). It should be noted that the potential vertical displacement of the core is limited to a very small distance by the upper core support structure. Core lifting at power would result in a slight reduction in reactivity since the rising fuel would tend to engage the withdrawn control rods to a slightly greater extent than it would in the bottomed condition. The amount of this change in reactivity is, of course, available for reinsertion should the fuel settle back to its original position. The potential reactivity increase caused by the settling of the 16 centrally located control rod assembly elements (assumed to have been subject to lifting in the Oconee 2 reactor) was calculated to be 0.1% Δ K/K. This value is insufficient to have much effect on the accident and transient safety analyses.

An additional concern was the potential for damage to the fuel assembly and fittings which might be caused by fretting due to repetitive fuel movement. Consequently, Duke Power was requested by NRR to make certain examinations of the Oconee 2 fuel during the first refueling to confirm that fuel element motion was not occurring. The results of this examination (letter from W. O. Parker to R. C. Rusche dated 7/21/76) showed that no fuel lifting or other type of motion had occurred during the first cycle of operation.

After the core lift concern was identified, B&W developed newer types of fuel holddown springs which provide more margin against core lifting than the previous springs did. It is our understanding that the newer types of springs have been installed in all B&W reactors.

For these reasons, we believe that there is presently little likelihood that core lifting will occur during normal power operation. At lower temperatures, there is an increased flow induced lifting force on the fuel due to the higher viscosity of the reactor coolant. Consequently, we view the restriction against 4 pump operation below 500°F as a prudent precaution against fuel fretting. However, since the potential for core lifting has little safety significance and because critical operation below 500°F is not permitted, we have no basis to recommend including this restriction in the Technical Specifications.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

Inspection Report 50-346/78-06, paragraph 4, reported reactivity - power oscillations in the Davis-Besse core. These oscillations have also occurred at Oconee and are attributed to steam generator level oscillations. B&W report BAW-10027 states in A9.2:

> The OTSG Laboratory model test results indicated that periodic oscillations in steam pressure, steam flow, and steam generator primary outlet temperatures could occur under certain conditions.
>
> It was shown that the oscillations were of the type associated with the relationships between feedwater heating chamber pressure drop and tube nest pressure drop, which are eliminated or reduced to levels of no consequence (no feedback to reactor system) by adjustment of the tube nest inlet resistance. As a result of the tests, an adjustable orifice has been installed in the downcomer section of the steam generators to provide for adjustment of the tube nest inlet resistance and to provide the means for elimination of oscillations if they should develop during the operating lifetime of the generators. The initial orifice setting is chosen conservatively to minimize the need for further adjustment during the startup test program.

We also note that the effect on the incore detector system for monitoring core parameters during the oscillations is not clear.

DISCUSSION AND EVALUATION

Power Oscillations of the order of 1.5% of full power have been observed at all of the Oconee plants and are considered normal. In 1977 the power oscillations experienced by the Oconee 3 reactor increased to a maximum of 7.5% of full power. At that time the problem was reviewed by NRR with the conclusion that there was no significant safety consideration at that value (Note to B. C. Buckley from S. D. MacKay, dated January 27, 1978). It should be noted that the 7.5% power oscillations cause about a 1°F oscillation in core average temperature due to the short period of the oscillations. The important core safety parameters, which are, the departure from nucleate boiling ratio and the average maximum linear heat generation rate are affected very little by oscillations of this amplitude. The primary cause of the power oscillations is believed to be a fluctuation of the secondary water level in the steam generators. This can be minimized by increasing the flow resistance

in the downcomer region of the steam generators. The corrective effort at Oconee 3 was complicated by the fact that the orifice plate provided for this purpose could not be fully closed.

However, the oscillations at other B&W plants have been kept to about 1.5% of full power by appropriate adjustment of the downcomer flow resistance. For these reasons, the power oscillations at B&W plants are not considered to be a significant safety concern.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

4. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% power with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Enclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

DISCUSSION AND EVALUATION

The event at Davis Besse which resulted in loss of pressurizer level indication has been reviewed by NRR and the conclusion was reached that no unreviewed safety question existed.

The pressurizer, together with the reactor coolant makeup system, is designed to maintain the primary system pressure and water level within their operational limits only during normal operating conditions. Cooldown transients, such as loss of offsite power and loss of feedwater, sometimes result in primary pressure and volume changes that are beyond the ability of this system to control. The analyses of and experience with such transients show, however, that they can be sustained without compromising the safety of the reactor. The principal concern caused by such transients is that they might cause voiding in the primary coolant system that would lead to loss of ability to adequately cool the reactor core. The safety evaluation of the loss of offsite power transient shows that, though level indication is lost, some water remains in the pressurizer and the pressure does not decrease below about 1600 psi. In order for voiding to occur, the pressure must decrease below the saturation pressure corresponding to the system temperature. 1600 psi is the saturation pressure corresponding to 605°F, which is also the maximum allowable core outlet temperature. Voiding in the primary system (excepting the pressurizer) is precluded in this case, since pressure does not decrease to saturation.

The safety analysis for more severe cooldown transients, such as the loss of feedwater event, indicates that the water volume could decrease to less than the system volume exclusive of the pressurizer. During such an event, the emptying of the pressurizer would be followed by a pressure reduction below the saturation point and the formation of small voids throughout much of the primary system. This would not result in the loss of core cooling because the voids would be dispersed over a large volume and forced flow would prevent them from coalescing sufficiently to prevent core cooling. The high pressure coolant injection pumps are started automatically when the primary pressure decreases below 1600 psi. Therefore, any pressure reduction which is sufficient to allow voiding will also result in water injection which will rapidly restore the primary water to normal levels.

For these reasons, we believe that the inability of the pressurizer and normal coolant makeup system to control some transients does not provide a basis for requiring more capacity in these systems.

General Design Criterion 13 of Appendix A to 10 CFR 50 requires instrumentation to monitor variables over their anticipated ranges for "anticipated operational occurrences". Such occurrences are specifically defined to include loss of all offsite power. The fact that T cold goes off scale at 520°F is not considered to be a deviation from this requirement because this indicator is backed up by wide range temperature indication that extends to a low limit of 50°F. Neither do we consider the makeup flow monitoring to deviate since the amount of makeup flow in excess of 160 gpm does not appear to be a significant factor in the course of these occurrences.

The loss of pressurizer water level indication could be considered to deviate from GDC 13, because this level indication provides the principal means of determining the primary coolant inventory. However, provision of a level indication that would cover all anticipated occurrences may not be practical. As discussed above, the loss of feedwater event can lead to a momentary condition wherein no meaningful level exists, because the entire primary system contains a steam water mixture.

It should be noted that the introduction to Appendix A (last paragraph) recognizes that fulfillment of some of the criteria may not always be appropriate. This introduction also states that departures from the Criteria must be identified and justified. The discussion of GDC 13 in the Davis Besse FSAR lists the water level instrumentation, but does not mention the possibility of loss of water level indication during transients. This apparent omission in the safety analysis will be subjected to further review.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

4. A memo from B&W regarding control rod drive system trip breaker maintenance is attached as Enclosure 2. This memo should be evaluated in terms of shutdown margin maintenance and ATWS considerations particularly in light of large positive moderator coefficients allowable with B&W facilities.

DISCUSSION AND EVALUATION

Our investigation of the above circuit breaker problems has revealed that eight failures of reactor scram circuit breakers to trip during test have been reported from Babcock & Wilcox (B&W) type operating facilities since 1975. In each case, the faulty circuit breaker was identified as a GE type AK-2 series (i.e., AK-2A-15, 24, or 50). The causes for failure were attributed to either binding within the linkage mechanism of the undervoltage trip device (UV) and trip shaft assembly or an out-of-adjustment condition in the same linkage mechanism. B&W and GE determined that the binding and the out-of-adjustment conditions resulted from inadequate preventive maintenance programs at the affected operating facilities.

In addition to the breaker problems experienced at the B&W facilities, three circuit breakers of the aforementioned GE type failed in similar fashion at the Oyster Creek operating facility on November 26, 30, and December 2, 1978. As in each case above, cleaning and relubricating of the UV/trip shaft assembly within the circuit breaker was required to correct the problem. It is significant to note that during the November 30, 1978 event, both redundant service water pump circuit breakers failed to trip as required during the loss of off-site power test. These failures in turn created a potential overload condition on the emergency busses during the sequential bus loading by each diesel generator.

However, both diesel generators successfully picked up their required bus loads without experiencing a unit shutdown from an overload condition. With respect to the generic implications and safety significance of this issue, both B&W and GE are in the process of issuing alert letters to their customers. These letters are scheduled for issuance by late March and will describe the causes for failure and provide recommendations to resolve the problem.

Based on our study findings and on information obtained in discussions about the breaker problem with the knowledgeable people from B&W, GE and Region II, we plan to issue an IE Circular covering the matter. The thrust of the Circular will be directed toward the need for adequate preventive maintenance programs at all operating facilities. Specific recommendations from GE to resolve the above breaker problem will also be mentioned in the Circular.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

5. Inspection and Enforcement Report 50-346/78-17, paragraph 6 refers to inspection findings regarding the capability of the incore detector system to determine worst case thermal conditions. The reactor can be operated per the Technical Specifications with the center incore string out of service. If the peak power locations is in the center of the core (this has been the case at Davis-Besse), factors are not applied to conservatively monitor values such as F_Q and F delta H.

DISCUSSION AND EVALUATION

We do not believe that there is a valid basis for requiring the center string of incore detectors to be always operable in B&W reactors.

The power distributions for various plant conditions, throughout the fuel cycle, are calculated prior to the operation of the reactor. The power distribution is verified at the beginning of operation, and periodically thereafter, by comparison with the available incore detectors. The power in fuel assemblies that lack detectors (including those with failed detectors) is derived by using the known power distribution to determine the power ratios between such an assembly and nearby assemblies that have detectors. These ratios can then be multiplied by the power in the measured assemblies to derive the power level in any specific unmeasured assembly. The central assembly is not fundamentally different than any other assembly in this respect. Although this assembly is the highest powered assembly in the Davis Besse reactor at the beginning of the fuel cycle, this is not the case at all reactors. Nor does the central assembly have the highest power, in the Davis Besse reactor, at the end of the first fuel cycle. Since there is some variation between the calculated power distributions and the actual ones, an appropriate margin is assumed for this variation in establishing the allowable power peaking factors.

Fixed incore detectors must function in an extremely harsh environment and are subject to high failure rates. In order to ensure that an adequate number will survive the fuel cycle, many more detectors are installed than are necessary for the power distributions determinations. To require the central string to be always operable would likely result in unnecessary power restrictions. Neither the standard Technical Specifications (STS) for B&W plants nor the STS for CE plants (which also have fixed incore detectors) require the central detectors to be operable.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

6. Enclosure 3 describes an event that occurred at a B&W facility which resulted in a severe thermal transient and extreme difficulty in controlling the plant. The aforementioned facilities should be reviewed in light of this information for possible safety implications.

DISCUSSION AND EVALUATION

Following the cooldown transient at Rancho Seco, NRR evaluated the event and concluded that no structural damage had occurred to the primary coolant system which would preclude future operation of Rancho Seco. However, in their safety evaluations they concluded that positive steps should be taken to preclude similar transients and that the generic implications of this event should be reviewed. In addition, IE initiated a Transfer of Lead Responsibility, Serial No. IE-ROI 78-04, dated April 25, 1978, recommending that:

1. NRR perform a generic review of the non-nuclear instrumentation power supplies for other B&W units, if design changes to the non-nuclear instrumentation (NNI) power supplies are required at Rancho Seco.

2. NRR evaluate the susceptibility of B&W plants to other initiating events or failures which could cause similar significant cooldown transients.

This event is currently being evaluated by NRR.

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

Shaper Exhibit #1
8-3-79 sc

BIOGRAPHICAL OUTLINE

HOWARD K. SHAPAR, B.A., Amherst College, J.D., Yale Law School; Executive Legal Director, U.S. Nuclear Regulatory Commission; member of bars of State of New Mexico, Court of Appeals for District of Columbia Circuit, and U.S. Supreme Court; past president, Los Alamos County (New Mexico) bar association; vice-president, International Nuclear Law Association, past chairman, atomic energy law committee, World Peace Through Law Center; past chairman, Committee on International Uses of Atomic Energy (International Law Section), American Bar Association; past vice-chairman, Committee on Energy (Administrative Law Section), American Bar Association; past chairman, atomic energy law committee, Federal Bar Association; author of numerous articles in the field of nuclear law.

8-3-79 sc

[illegible] for this and a copy of ... [illegible] ..., 1979.

I have scheduled a meeting for 10:00 a.m. [illegible] ... 14, 1979 to further discuss this matter. [illegible]

[illegible]

Attachment:

#4

50-320

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 29, 1979

BOARD NOTIFICATION

Re:	Davis Besse	Docket Nos. 50-500, 50-501
	Erie	Docket Nos. STN 50-580, STN 50-581
	Greene County	Docket No. 50-549
	Midland 1 & 2	Docket No. 50-329 OL, 50-330 OL
	Pebble Springs	Docket Nos. 50-514, 50-515
	Three Mile Island 2	Docket No. 50-320

Ladies and Gentlemen:

Enclosed for the information of the Boards is a recent memorandum relating to certain concerns raised by a reactor inspector in Region III concerning the Davis Besse and Midland units. We are informing the Boards with respect to Davis Besse 2 and 3 and Midland 1 and 2. We are also providing information to the Boards in connection with Erie, Greene County, Pebble Springs, and Three Mile Island 2 since those facilities have similar Babcock & Wilcox reactor units.

Sincerely,

Joseph F. Scinto
Deputy Director, Hearing Division

Enclosure
As Stated

Distribution: (see attached list)

~~7905110377~~ 7905160163

P

Shapar Exhibit #3
8-3-79 -sc

Distribution:

Copies of a "Board Notification" letter dated March 29, 1979, signed by Joseph F. Scinto have been served on the following persons. Those whose addresses are at the U.S. Nuclear Regulatory Commission have been served by the NRC internal mail system and others have been served by deposit in the U.S. Mail. One copy has been served on each person even though his or her name appears on more than one service list. In addition to copies served on Atomic Safety and Licensing Board and Atomic Safety and Licensing Appeal Board members identified on the service list, 5 copies of the cover letter for each captioned proceeding and 5 copies in total of the attachment have been provided to the Atomic Safety and Licensing Board Panel, and 1 copy of both cover letter and attachment has been provided to the Atomic Safety and Licensing Appeal Board Panel.

In the Matter of THE TOLEDO EDISON COMPANY, *et al.* (Davis-Besse Nuclear Power Station, Units 2 and 3)	)))))))	Docket Nos. 50-500 50-501

Alan S. Rosenthal, Esq., Chairman
Atomic Safety and Licensing
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U.S. Nuclear Regulatory Commission
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Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of	)	
	)	
OHIO EDISON COMPANY, *et al*.	)	Docket Nos. STN 50-580
	)	STN 50-581
(Erie Nuclear Plant, Units 1 and 2)	)	
	)	

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In the Matter of POWER AUTHORITY OF THE STATE OF NEW YORK (Greene County Nuclear Power Plant)	)))))))	Docket No. 50-549

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Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of

CONSUMERS POWER COMPANY

(Midland Plant, Units 1 and 2)

Docket Nos. 50-329 O.L.
50-330 O.L.

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In the Matter of PORTLAND GENERAL ELECTRIC COMPANY (Pebble Springs Nuclear Plant, Units 1 and 2)	Docket Nos. 50-514 50-515

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Washington, D.C. 20555

In the Matter of METROPOLITAN EDISON COMPANY, et al. (Three Mile Island Nuclear Station, Unit No. 2)	))))))	Docket No. 50-320

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Washington, D.C. 20555

SHAPER Exhibit #4
8-3-79 SR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 6, 1979

MEMORANDUM FOR: Edward S. Christanbury, Hearing Division Director and Chief Counsel, OELD

FROM: D. B. Vassallo, Assistant Director for Light Water Reactors, Division of Project Management, NRR

SUBJECT: BOARD NOTIFICATION - REACTOR INSPECTOR CONCERNS REGARDING B&W PLANTS (BN-79-10)

The enclosed memorandum from I&E provides information originated by a Reactor Inspector as Board Notification material. Although I&E concluded that the information was not relevant and material the originator still believes that Boards should be informed.

Since we have not yet received I&E's written discussion and evaluation of these matters we have not reviewed the material in any detail. Regardless, however, in accordance with established procedures the information should be provided to appropriate Boards based on the originator's concerns.

The originator recommends that the Davis Besse 2 & 3 and Midland 1 & 2 Boards be informed.

In neither case is the SER Supplement issued but we have no objection to providing the information. In addition, since the concerns appear to apply to B&W plants, we recommend that you also provide the information to the Erie, Greene County, Pebble Springs and TMI-2 Boards.

When we receive the I&E written evaluations we will review them to determine whether additional review should be provided by DSS. In any event, we will follow this up with additional information for the Boards in the near future.

D B Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

Enclosure:
As stated

cc: See attached sheet

~~7905170583~~ 7905160185 P

cc: H. Denton
E. Case
D. Eisenhut
J. Davis
R. Boyd
V. Stello
R. DeYoung
L. Nichols
B. Grimes
J. Stolz
R. Baer
D. Parr
S. Varga
IE (7)
E. Jordan
D. Thompson

S/nHPZR CRAIOI -
8-3-79 SC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 0 1 1979

MEMORANDUM FOR: Domenic B. Vassallo, Assistant Director for Light Water Reactors, NRR

FROM: Dudley Thompson, Executive Officer for Operations Support, IE

SUBJECT: INFORMATION FOR BOARD NOTIFICATION - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2

The enclosed information is being forwarded for Board Notification. Your contact on this matter for any additional information is E. L. Jordan, ext. 28180.

Please note that the 2/28/79 cover memorandum, Moseley to Thompson, states that the originator, after being informed of IE Headquarters evaluation, still believes the information should be sent forward to the boards.

We request to be informed of your disposition on this matter.

Dudley Thompson
Executive Officer for
Operations Support, IE

Enclosures:
1. Memo NCMoseley to DThompson dtd 2/28/79
2. Memo JSCreswell to JPStreeter dtd 1/8/79 w/enclosures

cc: N. C. Moseley, ROI w/o encls
E. L. Jordan, ROI w/o encls
J. P. Streeter, RII w/o encls
J. S. Creswell, RII w/o encls
G. C. Gower, XOOS w/encls
IE Files w/encls

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7904260038 CF

DHHREC CX. -6
8-3-79 SC

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 28 1979

MEMORANDUM FOR: ~~Dudley~~ Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS (AITS F30468H2)

Enclosed are six items sent in by Region III for forwarding to sitting Licensing Boards for cases involving Babcock and Wilcox as the Nuclear Steam System Supplier. Our preliminary evaluation indicates these items do not appear to be new issues or to put a different light on the issues and therefore, in our opinion, do not meet the intended criteria for Board notification.

The originator was informed, via telephone, of this determination on February 27, 1979. His position was that our evaluation did not provide any information that he did not already have and his concern was whether or not these items had been considered and resolved on a generic basis for all B&W plants. Because of this he still believed the items should be sent to the Licensing Boards. IE Manual Chapter 1530 requires that if, after a negative determination, the originator continues to believe that the information should be submitted to the Board(s), the information will be submitted. We therefore request the enclosed items be sent to the appropriate Licensing Boards.

We will provide a written discussion and evaluation of each item within seven (7) days of the date of this memorandum.

for Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

Enclosure:
Memorandum Creswell to Streeter
dated January 8, 1979

cc w/o encl:
S. E. Bryan
E. L. Jordan
J. Kirkpatrick
J. C. Stone
G. C. Gower
R. F. Heishman, RIII

10 ((

7904210475

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

(5)

MAR 7 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

On February 28, 1979, six items involving Babcock and Wilcox plants were sent to you for forwarding to the appropriate licensing boards. In that memo, we committed to providing a written discussion and evaluation of each item within seven days.

Before we can complete the discussion and evaluation, additional information is needed from Region III. Region III will be unable to provide the information until March 12, 1979. We will provide the complete write-up to you by March 16, 1979.

Norman C. Moseley

Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

cc: R. F. Heishman, RIII
S. E. Bryan
E. L. Jordan
D. Kirkpatrick
J. C. Stone
~~L. G. Gower~~

CONTACT: J. C. Stone
(x28019)

SHAPER Exhibit #7
8-3-79 SC

(6)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 14 1979

MEMORANDUM FOR: Domenic B. Vassallo, Assistant Director for Light Water Reactors, NRR

FROM: Dudley Thompson, Executive Officer for Operations Support, IE

SUBJECT: INFORMATION FOR BOARD NOTIFICATION - DAVIS BESSE 1 & 2 AND MIDLAND 1 & 2

By memorandum dated 3/1/79 we provided information for Board notification on the subject plants and indicated that a written discussion and evaluation would follow in seven days. We have been informed by the enclosed memorandum that delays in getting certain information have caused us to change our submittal date to 3/17/79.

Dudley Thompson

Dudley Thompson
Executive Officer for
Operations Support, IE

Enclosure:
Memo NCMoseley to DThompson
dtd 3/7/79

cc w/o enclosure:
J. C. Stone, ROI
R. F. Heishman, RIII

SHAPER Exhibit #8
8-3-79 S-

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

(9)

MAR 29 1979

MEMORANDUM FOR: Domenic B. Vassallo, Assisant Director for Light Water Reactors, NRR

FROM: Dudley Thompson, XOOS

SUBJECT: INFORMATION FOR BOARD NOTIFICATION - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2
REFERENCES: 1. Memo: Thompson to Vassallo dtd 3/1/79
2. Memo: Thompson to Vassallo dtd 3/12/79

As noted in the above referenced submittals additional information in the form of staff discussion and evaluation would be forthcoming on the captioned board notification. Enclosed is the additional information for submittal to the appropriate Boards.

Dudley Thompson

Dudley Thompson, Executive Officer
for Operations Support
Office of Inspection and Enforcement

Enclosures:
1. Memo: Moseley to Thompson dtd 3/28/79 w/encls
2. Memo: Moseley to Thompson dtd 3/29/79

cc: N. C. Moseley, IE, w/o encl
S. E. Bryan, IE, w/o encl
J. F. Streeter, RIII, w/encl
J. S. Creswell, RIII, w/encl
G. C. Gower, IE, w/encl
IE Files w/encl

SHAPER Exhibit #9
8-3-79 SC

CONTACT: G. C. Gower, IE
49-27246

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

8

March 29, 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

In light of the transient experienced at Three Mile Island on March 28, 1979, we will review our preliminary evaluation of Item 3 contained in my March 28, 1979 memorandum to you. It is possible that the additional information will cause our assessment to change.

Norman C. Moseley

Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

cc: S. E. Bryan
E. L. Jordan
R. F. Heishman, RIII
J. C. Stone
D. C. Kirkpatrick
G. C. Gower
V. D. Thomas

Shaper Exhibit #10
8-3-79 SC

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

(7)

MAR 2 8 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

On February 28, 1979, six items concerning Babcock and Wilcox designed nuclear plants were sent to you for forwarding to the appropriate licensing boards. At that time only a preliminary evaluation had been done. We have completed our evaluation of each of the items and that information is enclosed. This additional information should be forwarded to the licensing boards.

Norman C. Moseley

Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

Enclosure:
Evaluations of Concerns

cc: S. E. Bryan
E. L. Jordan
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SHAPER Exhibit #11
8-3-79 SC

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EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

4. During a recent inspection at Davis-Besse Unit 1 information has been attained which indicates that at certain conditions of reactor coolant viscosity (as a function of temperature) core lifting may occur. The licensee informed the inspector that this issue involves other B&W facilities. The Davis-Besse FSAR states in Section 4.4.2.7:

> The hydraulic force on the fuel assembly receiving the most flow is shown as a function of system flow in Figure 4-39. Additional forces acting on the fuel assembly are the assembly weight and a hold down spring force, which resulted in a net downward force at all times during normal station operation.

The licensee states that there is a 500°F interlock for the starting of the fourth reactor coolant pump. However, no Technical Specification requires that the pump be started at or above this temperature. A concern regarding this matter would be if assemblies moved upward into a position such that control rod movement would be hindered.

DISCUSSION AND EVALUATION

The potential for core lifting in B&W plants is a concern which has been previously reviewed by NRR. The concern was first raised in connection with the Oconee 2 and 3 reactors, where the primary coolant flow rates were found to be in excess of the design flow rates. For example, the Unit 2 flow rate was found to be 111.5% of the design flow rate. Since this was very near the predicted core lift flow rate of 111.9%, an analysis was done by B&W to determine what effect core lifting would have on the previous safety analysis for these plants. This analysis (dated May 2, 1975) indicated that the potential for core lifting did not result in an unreviewed safety question. A subsequent review of this B&W analysis by NRR also concluded that an unsafe condition did not exist (letter from R. A. Purple to Duke Power, dated 9/24/75). It should be noted that the potential vertical displacement of the core is limited to a very small distance by the upper core support structure. Core lifting at power would result in a slight reduction in reactivity since the rising fuel would tend to engage the withdrawn control rods to a slightly greater extent than it would in the bottomed condition. The amount of this change in reactivity is, of course, available for reinsertion should the fuel settle back to its original position. The potential reactivity increase caused by the settling of the 16 centrally located control rod assembly elements (assumed to have been subject to lifting in the Oconee 2 reactor) was calculated to be 0.1% Δ K/K. This value is insufficient to have much effect on the accident and transient safety analyses.

An additional concern was the potential for damage to the fuel assembly and fittings which might be caused by fretting due to repetitive fuel movement. Consequently, Duke Power was requested by NRR to make certain examinations of the Oconee 2 fuel during the first refueling to confirm that fuel element motion was not occurring. The results of this examination (letter from W. O. Parker to R. C. Rusche dated 7/21/76) showed that no fuel lifting or other type of motion had occurred during the first cycle of operation.

After the core lift concern was identified, B&W developed newer types of fuel holddown springs which provide more margin against core lifting than the previous springs did. It is our understanding that the newer types of springs have been installed in all B&W reactors.

For these reasons, we believe that there is presently little likelihood that core lifting will occur during normal power operation. At lower temperatures, there is an increased flow induced lifting force on the fuel due to the higher viscosity of the reactor coolant. Consequently, we view the restriction against 4 pump operation below 500°F as a prudent precaution against fuel fretting. However, since the potential for core lifting has little safety significance and because critical operation below 500°F is not permitted, we have no basis to recommend including this restriction in the Technical Specifications.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

2. Inspection Report 50-346/78-06, paragraph 4, reported reactivity - power oscillations in the Davis-Besse core. These oscillations have also occurred at Oconee and are attributed to steam generator level oscillations. B&W report BAW-10027 states in A9.2:

> The OTSG Laboratory model test results indicated that periodic oscillations in steam pressure, steam flow, and steam generator primary outlet temperatures could occur under certain conditions.
>
> It was shown that the oscillations were of the type associated with the relationships between feedwater heating chamber pressure drop and tube nest pressure drop, which are eliminated or reduced to levels of no consequence (no feedback to reactor system) by adjustment of the tube nest inlet resistance. As a result of the tests, an adjustable orifice has been installed in the downcomer section of the steam generators to provide for adjustment of the tube nest inlet resistance and to provide the means for elimination of oscillations if they should develop during the operating lifetime of the generators. The initial orifice setting is chosen conservatively to minimize the need for further adjustment during the startup test program.

We also note that the effect on the incore detector system for monitoring core parameters during the oscillations is not clear.

DISCUSSION AND EVALUATION

Power Oscillations of the order of 1.5% of full power have been observed at all of the Oconee plants and are considered normal. In 1977 the power oscillations experienced by the Oconee 3 reactor increased to a maximum of 7.5% of full power. At that time the problem was reviewed by NRR with the conclusion that there was no significant safety consideration at that value (Note to B. C. Buckley from S. D. MacKay, dated January 27, 1978). It should be noted that the 7.5% power oscillations cause about a 1°F oscillation in core average temperature due to the short period of the oscillations. The important core safety parameters, which are, the departure from nucleate boiling ratio and the average maximum linear heat generation rate are affected very little by oscillations of this amplitude. The primary cause of the power oscillations is believed to be a fluctuation of the secondary water level in the steam generators. This can be minimized by increasing the flow resistance

in the downcomer region of the steam generators. The corrective effort at Oconee 3 was complicated by the fact that the orifice plate provided for this purpose could not be fully closed.

However, the oscillations at other B&W plants have been kept to about 1.5% of full power by appropriate adjustment of the downcomer flow resistance. For these reasons, the power oscillations at B&W plants are not considered to be a significant safety concern.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

3. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% power with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Inclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

DISCUSSION AND EVALUATION

The event at Davis Besse which resulted in loss of pressurizer level indication has been reviewed by NRR and the conclusion was reached that no unreviewed safety question existed.

The pressurizer, together with the reactor coolant makeup system, is designed to maintain the primary system pressure and water level within their operational limits only during normal operating conditions. Cooldown transients, such as loss of offsite power and loss of feedwater, sometimes result in primary pressure and volume changes that are beyond the ability of this system to control. The analyses of and experience with such transients show, however, that they can be sustained without compromising the safety of the reactor. The principal concern caused by such transients is that they might cause voiding in the primary coolant system that would lead to loss of ability to adequately cool the reactor core. The safety evaluation of the loss of offsite power transient shows that, though level indication is lost, some water remains in the pressurizer and the pressure does not decrease below about 1600 psi. In order for voiding to occur, the pressure must decrease below the saturation pressure corresponding to the system temperature. 1600 psi is the saturation pressure corresponding to 605°F, which is also the maximum allowable core outlet temperature. Voiding in the primary system (excepting the pressurizer) is precluded in this case, since pressure does not decrease to saturation.

The safety analysis for more severe cooldown transients, such as the loss of feedwater event, indicates that the water volume could decrease to less than the system volume exclusive of the pressurizer. During such an event, the emptying of the pressurizer would be followed by a pressure reduction below the saturation point and the formation of small voids throughout much of the primary system. This would not result in the loss of core cooling because the voids would be dispersed over a large volume and forced flow would prevent them from coalescing sufficiently to prevent core cooling. The high pressure coolant injection pumps are started automatically when the primary pressure decreases below 1600 psi. Therefore, any pressure reduction which is sufficient to allow voiding will also result in water injection which will rapidly restore the primary water to normal levels.

For these reasons, we believe that the inability of the pressurizer and normal coolant makeup system to control some transients does not provide a basis for requiring more capacity in these systems.

General Design Criterion 13 of Appendix A to 10 CFR 50 requires instrumentation to monitor variables over their anticipated ranges for "anticipated operational occurrences". Such occurrences are specifically defined to include loss of all offsite power. The fact that T cold goes off scale at 520°F is not considered to be a deviation from this requirement because this indicator is backed up by wide range temperature indication that extends to a low limit of 50°F Neither do we consider the makeup flow monitoring to deviate since he amount of makeup flow in excess of 160 gpm does not appear to be significant factor in the course of these occurrences.

The loss of pressurizer water level indication could be considered to deviate from GDC 13, because this level indication provides the principal means of determining the primary coolant inventory. However, provision of a level indication that would cover all anticipated occurrences may not be practical. As discussed above, the loss of feedwater event can lead to a momentary condition wherein no meaningful level exists, because the entire primary system contains a steam water mixture.

It should be noted that the introduction to Appendix A (last paragraph) recognizes that fulfillment of some of the criteria may not always be appropriate. This introduction also states that departures from the Criteria must be identified and justified. The discussion of GDC 13 in the Davis Besse FSAR lists the water level instrumentation, but does not mention the possibility of loss of water level indication during transients. This apparent omission in the safety analysis will be subjected to further review.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

4. A memo from B&W regarding control rod drive system trip breaker maintenance is attached as Enclosure 2. This memo should be evaluated in terms of shutdown margin maintenance and ATWS considerations particularly in light of large positive moderator coefficients allowable with B&W facilities.

DISCUSSION AND EVALUATION

Our investigation of the above circuit breaker problems has revealed that eight failures of reactor scram circuit breakers to trip during test have been reported from Babcock & Wilcox (B&W) type operating facilities since 1975. In each case, the faulty circuit breaker was identified as a GE type AK-2 series (i.e., AK-2A-15, 24, or 30). The causes for failure were attributed to either binding within the linkage mechanism of the undervoltage trip device (UV) and trip shaft assembly or an out-of-adjustment condition in the same linkage mechanism. B&W and GE determined that the binding and the out-of-adjustment conditions resulted from inadequate preventive maintenance programs at the affected operating facilities.

In addition to the breaker problems experienced at the B&W facilities, three circuit breakers of the aforementioned GE type failed in similar fashion at the Oyster Creek operating facility on November 26, 30, and December 2, 1978. As in each case above, cleaning and relubricating of the UV/trip shaft assembly within the circuit breaker was required to correct the problem. It is significant to note that during the November 30, 1978 event, both redundant service water pump circuit breakers failed to trip as required during the loss of off-site power test. These failures in turn created a potential overload condition on the emergency busses during the sequential bus loading by each diesel generator.

However, both diesel generators successfully picked up their required bus loads without experiencing a unit shutdown from an overload condition. With respect to the generic implications and safety significance of this issue, both B&W and GE are in the process of issuing alert letters to their customers. These letters are scheduled for issuance by late March and will describe the causes for failure and provide recommendations to resolve the problem.

Based on our study findings and on information obtained in discussions about the breaker problem with the knowledgeable people from B&W, GE and Region II, we plan to issue an IE Circular covering the matter. The thrust of the Circular will be directed toward the need for adequate preventive maintenance programs at all operating facilities. Specific recommendations from GE to resolve the above breaker problem will also be mentioned in the Circular.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 3, 1979, FROM J.S. CRESWELL TO J.F. STREETER

5. Inspection and Enforcement Report 50-346/78-17, paragraph 6 refers to inspection findings regarding the capability of the incore detector system to determine worst case thermal conditions. The reactor can be operated per the Technical Specifications with the center incore string out of service. If the peak power locations is in the center of the core (this has been the case at Davis-Besse), factors are not applied to conservatively monitor values such as F_Q and F delta H.

DISCUSSION AND EVALUATION

We do not believe that there is a valid basis for requiring the center string of incore detectors to be always operable in B&W reactors.

The power distributions for various plant conditions, throughout the fuel cycle, are calculated prior to the operation of the reactor. The power distribution is verified at the beginning of operation, and periodically thereafter, by comparison with the available incore detectors. The power in fuel assemblies that lack detectors (including those with failed detectors) is derived by using the known power distribution to determine the power ratios between such an assembly and nearby assemblies that have detectors. These ratios can then be multiplied by the power in the measured assemblies to derive the power level in any specific unmeasured assembly. The central assembly is not fundamentally different than any other assembly in this respect. Although this assembly is the highest powered assembly in the Davis Besse reactor at the beginning of the fuel cycle, this is not the case at all reactors. Nor does the central assembly have the highest power, in the Davis Besse reactor, at the end of the first fuel cycle. Since there is some variation between the calculated power distributions and the actual ones, an appropriate margin is assumed for this variation in establishing the allowable power peaking factors.

Fixed incore detectors must function in an extremely harsh environment and are subject to high failure rates. In order to ensure that an adequate number will survive the fuel cycle, many more detectors are installed than are necessary for the power distributions determinations. To require the central string to be always operable would likely result in unnecessary power restrictions. Neither the standard Technical Specifications (STS) for B&W plants nor the STS for CE plants (which also have fixed incore detectors) require the central detectors to be operable.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

6. Enclosure 3 describes an event that occurred at a B&W facility which resulted in a severe thermal transient and extreme difficulty in controlling the plant. The aforementioned facilities should be reviewed in light of this information for possible safety implications.

DISCUSSION AND EVALUATION

Following the cooldown transient at Rancho Seco, NRR evaluated the event and concluded that no structural damage had occurred to the primary coolant system which would preclude future operation of Rancho Seco. However, in their safety evaluations they concluded that positive steps should be taken to preclude similar transients and that the generic implications of this event should be reviewed. In addition, IE initiated a Transfer of Lead Responsibility, Serial No. IE-ROI 78-04, dated April 25, 1978, recommending that:

1. NRR perform a generic review of the non-nuclear instrumentation power supplies for other B&W units, if design changes to the non-nuclear instrumentation (NNI) power supplies are required at Rancho Seco.

2. NRR evaluate the susceptibility of B&W plants to other initiating events or failures which could cause similar significant cooldown transients.

This event is currently being evaluated by NRR.

SHAPER Exhibit #12
8-3-79 SC

(12)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 29, 1979

BOARD NOTIFICATION

Re:	Davis Besse	Docket Nos. 50-500, 50-501
	Erie	Docket Nos. STN 50-580, STN 50-581
	Greene County	Docket No. 50-549
	Midland 1 & 2	Docket No. 50-329 OL, 50-330 OL
	Pebble Springs	Docket Nos. 50-514, 50-515
	Three Mile Island 2	Docket No. 50-320

Ladies and Gentlemen:

Enclosed for the information of the Boards is a recent memorandum relating to certain concerns raised by a reactor inspector in Region III concerning the Davis Besse and Midland units. We are informing the Boards with respect to Davis Besse 2 and 3 and Midland 1 and 2. We are also providing information to the Boards in connection with Erie, Greene County, Pebble Springs, and Three Mile Island 2 since those facilities have similar Babcock & Wilcox reactor units.

Sincerely,

Joseph F. Scinto
Deputy Director, Hearing Division

Enclosure
As Stated

Distribution: (see attached list)

Distribution:

Copies of a "Board Notification" letter dated March 29, 1979, signed by Joseph F. Scinto have been served on the following persons. Those whose addresses are at the U.S. Nuclear Regulatory Commission have been served by the NRC internal mail system and others have been served by deposit in the U.S. Mail. One copy has been served on each person even though his or her name appears on more than one service list. In addition to copies served on Atomic Safety and Licensing Board and Atomic Safety and Licensing Appeal Board members identified on the service list, 5 copies of the cover letter for each captioned proceeding and 5 copies in total of the attachment have been provided to the Atomic Safety and Licensing Board Panel, and 1 copy of both cover letter and attachment has been provided to the Atomic Safety and Licensing Appeal Board Panel.

In the Matter of

THE TOLEDO EDISON COMPANY, *et al.*

(Davis-Besse Nuclear Power Station, Units 2 and 3)

Docket Nos. 50-500, 50-501

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In the Matter of OHIO EDISON COMPANY, *et al*. (Erie Nuclear Plant, Units 1 and 2)	)))))) Docket Nos. STN 50-580 STN 50-581

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In the Matter of POWER AUTHORITY OF THE STATE OF NEW YORK (Greene County Nuclear Power Plant)	))) Docket No. 50-549)))

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In the Matter of CONSUMERS POWER COMPANY (Midland Plant, Units 1 and 2)	Docket Nos. 50-329 O.L. 50-330 O.L.

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In the Matter of PORTLAND GENERAL ELECTRIC COMPANY (Pebble Springs Nuclear Plant, Units 1 and 2)	Docket Nos. 50-514 50-515

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Washington, D.C. 20555

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY, *et al.*	)	Docket No. 50-320
	)	
(Three Mile Island Nuclear Station, Unit No. 2)	)	

Alan S. Rosenthal, Esq., Chairman
Atomic Safety and Licensing
Appeal Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. W. Reed Johnson, Member
Atomic Safety and Licensing
Appeal Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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Atomic Safety and Licensing
  Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing
  Appeal Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docketing and Service Section
Office of the Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 6, 1979

MEMORANDUM FOR: Edward S. Christenbury, Hearing Division Director and Chief Counsel, OELD

FROM: D. B. Vassallo, Assistant Director for Light Water Reactors, Division of Project Management, NRR

SUBJECT: BOARD NOTIFICATION - REACTOR INSPECTOR CONCERNS REGARDING B&W PLANTS (BN-79-10)

The enclosed memorandum from I&E provides information originated by a Reactor Inspector as Board Notification material. Although I&E concluded that the information was not relevant and material the originator still believes that Boards should be informed.

Since we have not yet received I&E's written discussion and evaluation of these matters we have not reviewed the material in any detail. Regardless, however, in accordance with established procedures the information should be provided to appropriate Boards based on the originator's concerns.

The originator recommends that the Davis Besse 2 & 3 and Midland 1 & 2 Boards be informed.

In neither case is the SER Supplement issued but we have no objection to providing the information. In addition, since the concerns appear to apply to B&W plants, we recommend that you also provide the information to the Erie, Greene County, Pebble Springs and TMI-2 Boards.

When we receive the I&E written evaluations we will review them to determine whether additional review should be provided by DSS. In any event, we will follow this up with additional information for the Boards in the near future.

D B Vassallo

D. B. Vassallo, Assistant Director
for Light Water Reactors
Division of Project Management

Enclosure:
As stated

cc: See attached sheet

cc: H. Denton
E. Case
D. Eisenhut
J. Davis
R. Boyd
V. Stello
R. DeYoung
L. Nichols
B. Grimes
J. Stolz
R. Baer
O. Parr
S. Varga
IE (7)
E. Jordan
D. Thompson

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 01 1979

MEMORANDUM FOR: Domenic B. Vassallo, Assistant Director for Light Water Reactors, NRR

FROM: Dudley Thompson, Executive Officer for Operations Support, IE

SUBJECT: INFORMATION FOR BOARD NOTIFICATION - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2

The enclosed information is being forwarded for Board Notification. Your contact on this matter for any additional information is E. L. Jordan, ext. 28180.

Please note that the 2/23/79 cover memorandum, Moseley to Thompson, states that the originator, after being informed of IE Headquarters evaluation, still believes the information should be sent forward to the boards.

We request to be informed of your disposition on this matter.

Dudley Thompson
Executive Officer for
Operations Support, IE

Enclosures:
1. Memo NCMoseley to DThompson dtd 2/23/79
2. Memo JSCreswell to JFStreeter dtd 1/8/79 w/enclosures

cc: N. C. Moseley, ROI w/o encls
E. L. Jordan, ROI w/o encls
J. F. Streeter, RII w/o encls
J. S. Creswell, RII w/o encls
G. C. Gower, XOOS w/encls
IE Files w/encls

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 2 8 1979

MEMORANDUM FOR: ~~Dudley~~ Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS (AITS F30486H2)

Enclosed are six items sent in by Region III for forwarding to sitting Licensing Boards for cases involving Babcock and Wilcox as the Nuclear Steam System Supplier. Our preliminary evaluation indicates these items do not appear to be new issues or to put a different light on the issues and therefore, in our opinion, do not meet the intended criteria for Board notification.

The originator was informed, via telephone, of this determination on February 27, 1979. His position was that our evaluation did not provide any information that he did not already have and his concern was whether or not these items had been considered and resolved on a generic basis for all B&W plants. Because of this he still believed the items should be sent to the Licensing Boards. IE Manual Chapter 1530 requires that if, after a negative determination, the originator continues to believe that the information should be submitted to the Board(s), the information will be submitted. We therefore request the enclosed items be sent to the appropriate Licensing Boards.

We will provide a written discussion and evaluation of each item within seven (7) days of the date of this memorandum.

for Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

Enclosure:
Memorandum Creswell to Streeter
dated January 8, 1979

cc w/o encl:
S. E. Bryan
E. L. Jordan
J. Kirkpatrick
J. C. Stone
B. C. Gower
R. F. Heishman, RIII

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

January 8, 1979

Docket No. 50-500/501
50-329/330

MEMORANDUM FOR: J. F. Streeter, Chief, Nuclear Support Section 1

FROM: J. S. Creswell, Reactor Inspector

SUBJECT: CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2

During the course of my inspections at Davis-Besse, certain issues have come to my attention which I am submitting for consideration for forwarding to the Atomic Safety and Licensing Board which has proceedings pending for the aforementioned facilities. This submittal is made pursuant to Regional Procedure 1300A (November 16, 1978), step 2 and information supplied to me per step 1. The issues for consideration are:

1. During a recent inspection at Davis-Besse Unit 1 information has been attained which indicates that at certain conditions of reactor coolant viscosity (as a function of temperature) core lifting may occur. The licensee informed the inspector that this issue involves other B&W facilities. The Davis-Besse FSAR states in Section 4.4.2.7:

   > The hydraulic force on the fuel assembly receiving the most flow is shown as a function of system flow in Figure 4-39. Additional forces acting on the fuel assembly are the assembly weight and a hold down spring force, which resulted in a net downward force at all times during normal station operation.

   The licensee states that there is a 500°F interlock for the starting of the fourth reactor coolant pump. However, no Technical Specification requires that the pump be started at or above this temperature. A concern regarding this matter would be if assemblies moved upward into a position such that control rod movement would be hindered.

2. Inspection Report 50-346/78-06, paragraph 4, reported reactivity - power oscillations in the Davis-Besse core. These oscillations have also occurred at Oconee and are attributed to steam generator level oscillations. B&W report BAW-10027 states in A9.2:

90

The OTSG laboratory model test results indicated that periodic oscillations in steam pressure, steam flow, and steam generator primary outlet temperatures could occur under certain conditions.

It was shown that the oscillations were of the type associated with the relationships between feedwater heating chamber pressure drop and tube nest pressure drop, which are eliminated or reduced to levels of no consequence (no feedback to reactor system) by adjustment of the tube nest inlet resistance. As a result of the tests, an adjustable orifice has been installed in the downcomer section of the steam generators to provide for adjustment of the tube nest inlet resistance and to provide the means for elimination of oscillations if they should develop during the operating lifetime of the generators. The initial orifice setting is chosen conservatively to minimize the need for further adjustment during the startup test program.

We also note that the effect on the incore detector system for monitoring core parameters during the oscillations is not clear.

3. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% power with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Enclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

4. A memo from B&W regarding control rod drive system trip breaker maintenance is attached as Enclosure 2. This memo should be evaluated in terms of shutdown margin maintenance and ATWS considerations particularly in light of large positive moderator coefficients allowable with B&W facilities.

5. Inspection and Enforcement Report 50-346/78-17, paragraph 6 refers to inspection findings regarding the capability of the incore detector system to determine worst case thermal conditions. The reactor can be operated per the Technical Specifications with the center incore string out of service. If the peak power locations is in the center of the core (this has been the case at Davis-Besse), factors are not applied to conservatively monitor values such as Fq and F delta H.

6. Enclosure 3 describes an event that occurred at a B&W facility which resulted in a severe thermal transient and extreme difficulty in controlling the plant. The aforementioned facilities should be reviewed in light of this information for possible safety implications.

J. S. Creswell
Reactor Inspector

Enclosures: As stated

cc w/o enclosures:
G. Fiorelli
R. C. Knop
T. N. Tambling

Preliminary copy

TOLEDO EDISON

LOWELL E. ROE
Vice President
Facilities Development
(419) 259-5242

Docket No. 50-346

License No. NPF-3

Serial No. 473

December 22, 1978

Director of Nuclear Reactor Regulation
Attention: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Reid:

In response to the December 20, 1978, telephone conversation between your Mr. Guy Vissing and our Mr. E. C. Novak, and the December 20, 1978 telephone conversation between NRC Region III personnel (G. Fiorelli, R. Knop, T. Tambling and J. Streeter) and our Mr. E. C. Novak, attached is an additional safety evaluation supporting continued operation of Davis-Besse Nuclear Power Station Unit 1. This additional safety evaluation supplements the analysis we provided to you by our letter dated December 11, 1978, Serial No. 471. The attached safety evaluation analyzes the transient resulting from the operator not controlling steam generator level at 35 inches in accordance with current operating procedures.

Yours very truly,

Lowell E Roe

LER:CRD

Enclosure

bf a/7

Docket No. 50-346
License No. NPF-3
Serial No. 475
December 22, 1978

Additional Safety Evaluation
of Transient Resulting from
Inability of Operator to Control
Steam Generator Level at 35 Inches.

I. INTRODUCTION

The Davis-Besse Unit 1 Steam and Feedwater Line Rupture Control System (SFRCS) design objectives are to prevent the release of high energy steam, to automatically start auxiliary feedwater (AFW), and to provide adequate AFW, via essential steam generator level control, to remove decay heat during anticipated and design basis events when AFW is required. Table 1 correlates the station variables and accident conditions for which AFW actuation is required. For all actuation signals, the SFRCS initiates and controls AFW addition automatically to maintain a 120" level (96" indicated on the startup range instrumentation) in the steam generators.

The recent natural circulation test at Davis-Besse 1 (TP800.04) demonstrated that a 35-inch (indicated) steam generator level of AFW provides adequate natural circulation for decay heat removal.

The auto essential SG level control setpoint of 120-inches (96-inch-indicated) is thus in excess of minimum SG level requirements.

Operating procedures requiring manual control of steam generator level at 35-inches on the startup range level indicators following non-LOCA events were developed and used at Davis-Besse Unit 1 pending installation of permanent design changes to the SFRCS. Margin in maintenance of indicated pressurizer level and assurance of adequate natural circulation capability will exist through operator intervention during conditions where AFW is required.

Inability of the operator to comply with the present operating procedures will possibly result in a momentary loss of pressurizer level and/or level indication under certain conditions, but will not produce consequences which are non-reversible or detrimental to safe operation of Davis-Besse Unit 1.

II. DISCUSSION

The following section is divided into three segments: Relationship with Events Presented in the Davis-Besse Unit 1 FSAR, Loss of Offsite Power, and Loss of Feedwater.

A. Relationship with events presented in the FSAR

Addition of auxiliary feedwater at rates considerably greater than the decay heat generation rate will result in overcooling of the reactor coolant, contraction and a reduction of pressurizer level. This sequence of events is typical of several transients presented in the FSAR which have been submitted to the NRC and approved as a part of the licensing process. Overcooling transients can be caused by a variety of circumstances, failures, combinations of operating equipment, and improper operator interactions. From a practical viewpoint each single discoverable possible transient cannot be analyzed and presented as a part of the FSAR analysis, but a broad variety of transients have been selected. This specific transient fits within that broad category. Each of the FSAR transients has been demonstrated to produce acceptable results.

Overcooling transients resulting from a variety of causes are described in Section 15.2.10 "Excessive Heat Removal due to Feedwater Malfunctions". This section describes a transient resulting from excessive main feedwater addition, which is similar to the specific transient of increased level addition by auxiliary feedwater. Further information is presented in response to questions 15.2.15 and 15.2.16.

The steam line break (see FSAR sections 15.4.4, 15.4.8, 15.4.1) is the most severe overcooling transient, in that the reactor coolant system is decreased 50°F in average core temperature over a 30 second time period.

This is compared with the cooldown in question, which takes a much longer time to achieve a similar temperature drop and system conditions. During the steam line break, RC system pressure is reduced from 2200 psia to about 900 psi as system temperature is driven toward equilibrium with the unaffected (pressurized) steam generator attaining saturation temperature of about 530°F. The pressurizer is near empty at about 20 seconds and thereafter loses its influence on the system, thus permitting the upper elevations of the reactor coolant loop to approach saturation as cooldown continues toward 530°F. High pressure injection (HPI) pumps are actuated on low RC pressure such that pressurizer level will be restored. As shown in Figures 15.4.4-1 and 15.4.4-2 of the Davis-Besse Unit 1 FSAR, the rapid cooldown of RCS after reactor trip is limited by the pressure maintained in the pressurized steam generator in much the same fashion as anticipated for events such as the event of concern. As the RCS approaches saturation, core cooling is not impeded. Minimum DNBR>1.3 occurs just before reactor trip and subsequently increases with substantial margin throughout the remainder of the cooldown.

The close relationship of the auxiliary feedwater level increase as an overcooling transient with these similar overcooling transients allows us to draw the conclusion that no unreviewed safety question exists. To show a comparison to the detailed analyses reported in the FSAR, we have performed conservative bounding analyses of two representative cases.

I. Loss of Feedwater and Loss of Offsite Power

We have analyzed two transients resulting from auxiliary feedwater addition and establishment of SG level above the operating procedure 35" limit. The two transients examined are a loss of offsite power (reactor coolant pumps stop, makeup stops, main feedwater stops) and a loss of feedwater (reactor coolant pumps continue, makeup continues).

Of these two transients the loss of feedwater results in the greater volumetric coolant contraction, because the forced coolant flow (RC Pumps operating) causes a faster rate of heat rejection to the steam generator.

1. Loss of Offsite Power

Preliminary calculations for a reactor trip following a loss of offsite power show that the pressurizer loses indication but does not empty. The assumptions used to derive this result included full runout auxiliary feedwater flow (~1400 gpm) resulting in a fill time to 120" of about 4-minutes. No net mass change to the primary coolant (no makeup, no letdown) was considered, even though the makeup controls would respond to decreasing pressurizer level by increasing the net input to above 100 gpm. At the termination of the transient the pressurizer level is slightly above the outlet into the surge line. Reactor coolant pressure reaches about 1600 psi and high pressure injection may be automatically initiated.

Although the net makeup was not considered, it would in fact cause the pressurizer to refill to the normal level. At the same time compression of the steam would cause a partial repressurization of the system ensuring that the coolant remains subcooled. This transient presents no safety concerns.

2. Loss of Feedwater

This transient has a greater reactor coolant contraction than the loss of offsite power case, resulting in emptying of the pressurizer. Consequently it will be described in greater detail.

A brief summary of the events is:

• Reactor trip	Time = 0
• Makeup control valve opens wide admitting full makeup to reactor coolant system	Time = 0+
• AFW initiated	Time ≈ 40 sec
• Pressurizer empties; RC system pressure slightly greater than 1300 psi	Time ≈ 1 min
• HPI initiated by SFAS; makeup isolated	Time ≈ 1+ min
• Steam generator level = 10 ft; voids exist in reactor coolant	Time ≈ 4 min
• HPI inflow replaces volume occupied by voids; pressurizer level begins to be restored	Time ≈ 7-8 min

The major concerns that evolve from this transient are the disposition of the steam voids and the approach to DNB. Both of the concerns are ameliorated by the reactor coolant pumps.

Steam voids will not collect in reactor coolant piping and no flow blockage will occur because of dispersal and mixing by the forced flow. DNB acceptance criterion limit will be met because the power output of the core is at the decay heat level and all reactor pumps are operating, maintaining core heat removal. We conclude that no safety problem exists.

TABLE 1: STEAM AND FEEDWATER LINE RUPTURE CONTROL SYSTEM (SFRCS) ACTUATION PARAMETERS

Actuation Parameter Station Variables	Setpoint	Accident
1. Low Steam Line Pressure	$<$ 591.6 psig[1,2]	Steam Line Break Feedwater Line Break
2. Low SG Level	$\leq$ 17 inches[1]	Loss of FW
3. SG Pressure Minus Main Feedwater Line Pressure	$>$ 197.6 psi[1]	FWLB, LOMFW
4. Loss of All RC Pumps[3]		Loss of Off-Site Power

NOTES:

1. When actuated, SFRCS closes both main steam isolation valves, closes both main FW control and stop valves, initiates AFW and controls AFW to maintain a 120 inch level in the SGs.

2. Alignment of AFW to a pressurized SG is provided for steam and feedwater line breaks.

3. AFW initiation but steam and feedwater line isolation does not occur.

III. Bounding Analysis of Loss of Feedwater Event With Failure of Operator to Control Feedwater Level at 35"

Introduction:

The following bounding analysis conservatively predicts the events occurring within the primary reactor coolant system and reactor following a loss of main feedwater from 100% power for the Davis-Besse Unit 1. Auxiliary feedwater control has been assumed at 10 feet within both steam generators.

Results:

Because of the conservative, bounding, nature of this calculation, the overcooling of the primary system due to auxiliary feedwater injection causes a contraction of coolant volume sufficient to create steam within the primary system. The steam is shown to be uniformly distributed within the RCS and the void fraction is 4%. The reactor coolant pumps maintain full capability. The DNB ratio is shown to exceed 1.3 and no return to criticality potential exists. Thus, during the course of the incident, no core problems develop. Further, following the time of maximum contraction, the system recovers to full pressure, pressurizer function is regained and the reactor coolant returns to a subcooled water configuration without operator action.

Analysis:

The following assumptions have been made to assure the bounding nature of the results:

Reactor Power:

100% until boiling stops in the steam generators; 0% after that time. This assumption is conservative as core heat would compensate for the cooling caused by the auxiliary feedwater.

Initial Coolant Inventories Water:

RCS = 11290 ft^3

Pressurizer = 864 ft^3

These assumptions are nominal operating values.

Initial Temperatures:

The whole system is taken to be at $T_{average}$ = 582°F.

This assumption is a reasonable average.

Initial System Mass: ~ 500,000 lbm

The mass is figured from the temperature and volumes above.

Makeup System:

No credit is taken for additional makeup flow which will occur as the pressurizer loses level. (In all likelihood, the makeup system will contribute approximately 200 ft^3 extra liquid volume).

Local Power (kw/ft): 13.4 kw/ft

This value is taken as the maximum allowed by Technical Specifications.

Secondary Side Volume At 10 Foot Level

711 ft^3 per generator, actual volume.

Auxiliary Feedwater Flow:

166.5 ft^3/min. per generator actual value.

Auxiliary Feedwater Enthalpy:

8 Btu/lbm lower bound for maximum cooling.

With the initiating event, loss of main feedwater, the reactor coolant system pressure will start to rise. Reactor trip will occur on high RCS pressure. Following trip, the RCS pressure will fall because core power has been reduced and boiling of residual main feedwater or auxiliary feedwater is occurring in the steam generators. These events are almost identical to those which occur in a main feed line break and are analyzed in detail in Section 15.2.8 of the FSAR.

In short order, the system will return to its initial configuration but, because the auxiliary feedwater heat absorption rate exceeds the decay heat generation rate, the RCS continues to depressurize. During this phase, residual main feedwater and injected auxiliary feedwater will be boiled and vented through the steam generator safety relief valves. The primary system average temperature will fall to the saturation temperature of water at the safety valve set pressure. At this time, primary and secondary conditions are expected to be approximately as follows:

	Primary	Secondary
Pressure	1800 psia	980 psia
Temperature	542 F	542 F
Mass	502144 lbm	0 lbm
Liquid Volume in Press.	400 ft^3	N.A.
Time Into Transient	$\simeq$ 2 min.	$\simeq$ 2 min.

It is conservative to assume complete boiling of the secondary side water and complete equilibrium between primary and secondary sides, as these assumptions lead to the maximum follow on injection of auxiliary feedwater and therefore, maximum contraction. RCS pressure is held up by the steam bubble in the pressurizer.

The time has been estimated by calculating the necessary energy loss by the primary system from its initial conditions, the mass of auxiliary feedwater required to remove this energy and then dividing by the auxiliary feedwater flow rate.

$$\text{time} \simeq \frac{(586 - 542)\ 502344}{(1194-8)\ \ 333\ \ 62} \simeq 54 \text{ sec.}$$

Six seconds was used to estimate the initial pressurization portion of the transient.

In performing the remainder of the evaluation 10 feet of cooled (40 F) auxiliary feedwater is placed in each steam generator and the thermal equilibrium condition calculated. Because after a 10 foot level is obtained this auxiliary feedwater flow stops, this condition represents the maximum contraction possible. The state variables resulting are:

	Primary	Secondary
Pressure	560 psia	560 psia
Temperature	478 F	478 F
Enthalpy of Water	462 Btu/lbm	462 Btu/lbm
Specific Volume	.020 ft^3/lbm	.020 ft^3/lbm

From the specific volume, the primary liquid volume can be calculated:

$Vol = MV_f = 10052\ ft^3$

As 10052 is smaller than the RCS minus pressurizer volume, the remaining volume must be filled with steam.

$V_{st} = 10426 - 10052 = 374\ ft^3 \cong 400\ ft^3$

400 ft^3 corresponds to a system void fraction of 3.8% $\simeq$ 4%, and as will be shown later, is of no consequence as far as core heating or system performance is concerned. This steam volume is larger than actually expected for two reasons: 1) some temperature difference would always exist between the primary and secondary systems, and 2) the effect of core decay heat has been ignored. Both of these would increase the primary side liquid temperature, thus increasing its volume and reducing the steam volume.

Following this state of maximum contraction, no further heat is removed from the RCS via the secondary side until the RCS rises in temperature due to decay heating; this will expand the liquid volume, compress the steam and repressurize the RCS. As no mass can be lost from the secondary system prior to achieving 980 psia the first reheating stage will end at a primary system pressure, temperature, and liquid volume of 980 psia, 541 F, 10832 ft^3. Subtracting 10426 from 10832 shows that about 400 ft^3 of fluid has been forced back into the pressurizer. Pressurizer function would then be restored (if not directly, then, by either the makeup or HPI system), the RCS subcooled and the transient ended.

Several questions exist about the transient:

I. How is the 400 ft^3 dispersed within the primary system and can that volume collect in one location? From the auxiliary feedwater flow rate, over 4 minutes are required to fill the generators. As the pressurizer has 400 ft^3 in it at 980 psia and the RCS has 400 ft^3 in it at maximum contraction, approximately 2 minutes are used to eject steam from the pressurizer to the RCS. Because this steam will be superheated when it enters the RCS it will first desuperheat and then condense at a rate governed by its expanding pressure compared to the contraction of the liquid coolant. In the time of 2 minutes the reactor coolant will have made about 8 complete circles of the primary system and the steam can be considered well mixed. As the flow velocity in the RCS will remain normal, about 25 ft/sec, steam water separation will tend not to occur. Some limited steam accumulation may occur in the upper head of the reactor vessel as in that specific location of the RCS, velocity is low.

II. How well will the pumps work? Experiments performed on steam carry over capability show that for void fractions up to 10% no loss of pump capability-is observed. This is documented in Figure 5-47 of BAW-10104, "B&W's ECCS Evaluation Report With Specific Application to 177 FA Class Plants With Lower Loop Arrangement." Actually pump capability increases for the first 5% of void introduced into the system.

III. Will any return to power be encountered because of the low RCS temperature? A return to power can occur for a non-borated core at 490F. This temperature includes the assumption of the most reactive rod stuck out of the core; if that rod were taken as inserted the critical temperature would fall to at or below 400F. Although no credit was taken for HPI in calculating the RC steam volume below 1600 psia, the HPI will be injecting borated water and, therefore, preventing any return to power condition. If the primary system were to stabilize at 1600 psia and thus prevent the HPI from providing boron the RCS temperature would be at least 511F and, therefore, no return to power would be expected.

IV. Will DNB be encountered in the core? The maximum contraction condition is again:

P = 960 psia
T = 478F
a = 4%,

and occurs at least 5 minutes after power shutdown (trip occurs very early within 10 seconds of main feedwater loss). At this time, the decay heat rate is less than 3.2% using ANS + 20% (the LOCA evaluation curve). As low pressure and high void and high power are conservative bounds a DNB evaluation was performed at:

P = 500 psia
T = corresponding saturated value
α = 8%
power = 10%
W = full volumetric flow.

The resultant DNBR was >15 in the hottest channel with maximum design conditions assumed and well within acceptable values.

V. Will any steam remain trapped in the primary system? Some may be trapped for a short period of time in the upper head of the reactor vessel but this will be of no consequence and will eventually be condensed by thermal conduction through the interfacing water.

<u>Conclusion</u>

The maximum contraction of the RCS water has been calculated taking no credit for mitigating systems (makeup flow, HPI) and no credit for decay heating. No adverse consequences of the transient have been shown and, therefore, this transient poses no concerns to the safe operation of the plant.

IV. <u>CONCLUSIONS</u>

For SFRCS actuation and fill of the steam generators to the auto-essential level control point of 120" without operator action:

- No unreviewed safety question exists
- The loss of offsite power transient will not cause the pressurizer to drain although a loss of pressurizer indicated level will occur.
- The loss of feedwater transient may result in pressurizer emptying, however acceptance criteria for DNB will be met. Steam bubbles which exist in the reactor coolant for a short time will be collapsed by HPI injection. Pressurizer refilling by HPI will occur.
- No return to power will result in the long term.

Babcock & Wilcox

Power Generation Group
P.O. Box 1260, Lynchburg, Va. 24505
Telephone: (804) 384-5111

June 12, 1978

SOM #382 620-0014
12343 T2.2.1
SIP #14/259

Mr. T. D. Murray, Station Superintendent
Davis-Besse Nuclear Power Station
5501 North State Route #2
Oak Harbor, Ohio 43449

Subject: CRDCS Trip Breaker Maintenance

Dear Terry:

In the past, some of our plants have experienced problems with CRDCS Trip Breakers. The problems have been traced to lack of preventive maintenance. B&W suggests that a planned, carefully executed, maintenance program be established using the maintenance program outlined in the Diamond Power CRDC System Vendor Manual. Particular attention should be directed to proper cycling, cleaning, and lubrication of the breakers.

B&W further recommends that this program be scheduled at a minimum frequency of every refueling cycle and more frequently for plants during startup, when the equipment is subject to adverse environmental conditions.

Our concern is that if proper maintenance is not accomplished, additional failures will occur resulting in an NRC demand for diverse qualified trip breakers. Also, we need to prevent all failures we can to reduce the number of lost capacity days.

If we can be of further assistance, please advise.

Yours truly,

for
F. R. Faist
Site Operations Manager

FRF:DDG:mlf
cc: W. E. Spangler
R. C. Luken
R. L. Pittman
D. A. Lee
J. S. Grant, TECo
L. C. Novak, TECo
C. R. Domeck, TECo
J. C. Evans, TECo
B. R. Beyer, TECo
J. D. Lenardson, TECo
R. E. Blanchong, TECo
J. C. Buck, TECo

J— B&W

fil

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505

Telephone: (804) 384-5111

August 9, 1978

SOM #-03 / 12322 / SIP #14/295

620-001- / T3.3.1

Mr. T. D. Murray, Station Superintendent
Davis-Besse Nuclear Power Station
5501 North State Route #2
Oak Harbor, Ohio 43449

Subject: SMUD Rapid Cooldown Transient

Dear Terry:

On March 20, 1978, Rancho Seco experienced a severe thermal transient initiated by the loss of electrical power to a substantial portion of the Non-Nuclear Instrumentation (NNI). The loss of power directly caused the loss of Control Room indication of many plant parameters, the loss of input of these parameters to the plant computer, and erroneous input signals (midrange, zero, or otherwise incorrect) to the Integrated Control System (ICS).

The plant response was not the usual transient in that the ICS responded to the erroneous input signals rather than actual plant conditions, and resulted in a Reactor Protection System (RPS) trip on high pressure. Subsequent to the Reactor Trip, the erroneous signals to the ICS contributed to the rapid cooldown of the RCS. Plant operators had extreme difficulty in determining the true status of some of the plant parameters and in controlling the plant because of the erroneous indications in the Control Room.

An investigation of the events following this loss of power points out a need for a close look at operator training and emergency operating procedures for any loss of NNI power (or portion thereof). The following recommendations are made to assist your staff in a review of training and procedures to assure proper operator action for events of this nature.

1. Operators should be trained to recognize a loss of power to all or a majority of their NNI (e.g. indicators fail to mid-range, automatic or manual transfer to alternate instrument strings brings no response). The loss of power is emphasized here rather than the failure of any one instrument or control signal which are adequately covered in current simulator training courses.

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2. Given that the operator can determine that electrical power has been lost to all or part of the NNI, he should know the location of the power supply breakers, and have a procedure available to quickly regain power.

3. If the fault cannot be cleared (i.e. the breakers to the power supplies reopen), the operator should have a list of alternate instrumentation available to him, and he should be thoroughly trained in its use. Examples are:

   a. ESFAS panels
   b. RPS panels
   c. ECI (Essential Controls and Instrumentation)
   d. SRCI (Safety Related Controls and Instrumentation)
   e. Remote shutdown panels
   f. Local gages
   g. Plant computer

4. Recognizing that no procedure can cover all possible combinations of NNI failures, the operator's response should be keyed to certain variables. If the operator realizes that he has an instrumentation problem (as opposed to a LOCA or steam line break, for example), he can limit the transient by controlling a few critical variables:

   a. Pressurizer Level (via HPI or normal Makeup Pumps)
   b. RCS pressure (via Pressurizer heaters, spray, E/M relief valves, etc.)
   c. Steam Generator Level (via feed flow, feedwater valves, etc.)
   d. Steam Generator pressure (via turbine bypass system)

   The pressurizer Level and RCS pressure assure that the Reactor Coolant System is filled; the Steam Generator Level and pressure assure adequate decay heat removal.

Attachments 1 and 2 are provided to give a brief description of the events following this loss of NNI power at Rancho Seco. As can be seen by this transient, prompt precise operator action and the ability to recognize a loss of NNI power are critical factors in limiting the severity of a transient such as this.

If you have any questions or comments, please advise.

Yours truly,

Ivan D. Green
Site Operations Manager

IDG:JTS:alf
encl.
cc: See attached sheet.

ATTACHMENT 4

SEQUENCE OF EVENTS - SMUD 04:23 to 05:34 - MARCH 20, 1978

(Revision 1, 5/25/78)

T	EVENT
:3:35	- Lost NNI power supply cabinets 5, 6, & 7
	- This caused a loss of valid signals to the ICS. BTU limits ran back feedwater, resulting in a partial loss of feedwater (actual Rx power was 72%).
	- Probable opening of "B" turbine bypass valves to the condenser (timing uncertain).
:3:44	- Reactor trip on high pressure, turbine trip on interlock.
	- Pressurizer code relief setting was known to be low (approximately 2225 psig). The electromatic relief was isolated due to previous leakage problems. The data indicates primary pressure went ≈2400 psig ⇒ code relief valve lifted.
	- ICS closes main control and start-up feed valves and drive main feed pumps to minimum speed following trip.
	- Decay heat and RC pumps energy removal accomplished through generators by inventory boil off and the addition of main feedwater.
:6:25	- Pressurizer code relief valve reseats at approximately 2200 psig.
	- Operator starts HPI pump "B".
:8:23	- Operator stops HPI pump "B".
:0.	- OTSG "B" pressure reaches 435 psig set-point of Steam Line Failure logic.
	- OTSG "B" goes dry.

- Operator increases speed of a MFP and feeds "A" OTSG. This starts RCS on pressure and temperature decrease.

34:45 - RC pressure ≈1900 psi

36 - SFAS actuation at 1600 psig

This starts HPI, LPI and initiates emergency feed. The emergency FW pump is started and the bypass emergency FW valves are opened to full open position. The system makes no automatic attempt to control steam generator water level.

40 - RC pressure at 1475 psig. It starts to recover from this point due to HPI. T_{ave} = 523°F.

:43:56 - "A" HPI pump secured.

:46:09 - LPI secured.

:54 - "A" HPI initiated. From this point on, the operator started and stopped HPI pumps as necessary to maintain pressurizer level.

:50 - Steam Line Failure Logic closes ICS-controlled start-up feed valves to each OTSG when the corresponding OTSG pressure falls below 435 psig.

:51:25 - Secured RCP-D (T_{ave} ≈435°F)
This reduced #RCP's to three

:57:27 - OTSG "A" water level - 599.7"

Speculate that ≈2 ft. of tubes are not flooded (at top) due to steam line arrangement.

1:00 - Hourly computer log print-out
Steam temp. 380°F (OTSG "B")
Steam pressure 171 psig (OTSG "B")

Assuming T_{ave} = T_{sat} => T_{ave} = 380°F

13:47

- OTSG "B" level - 599.1"

14

- Power restored to NNI cabinets 5,6,&7

T_{ave} = 235°F

RCS Pressure ≈2000 psig

Both OTSG full level ranges pegged high

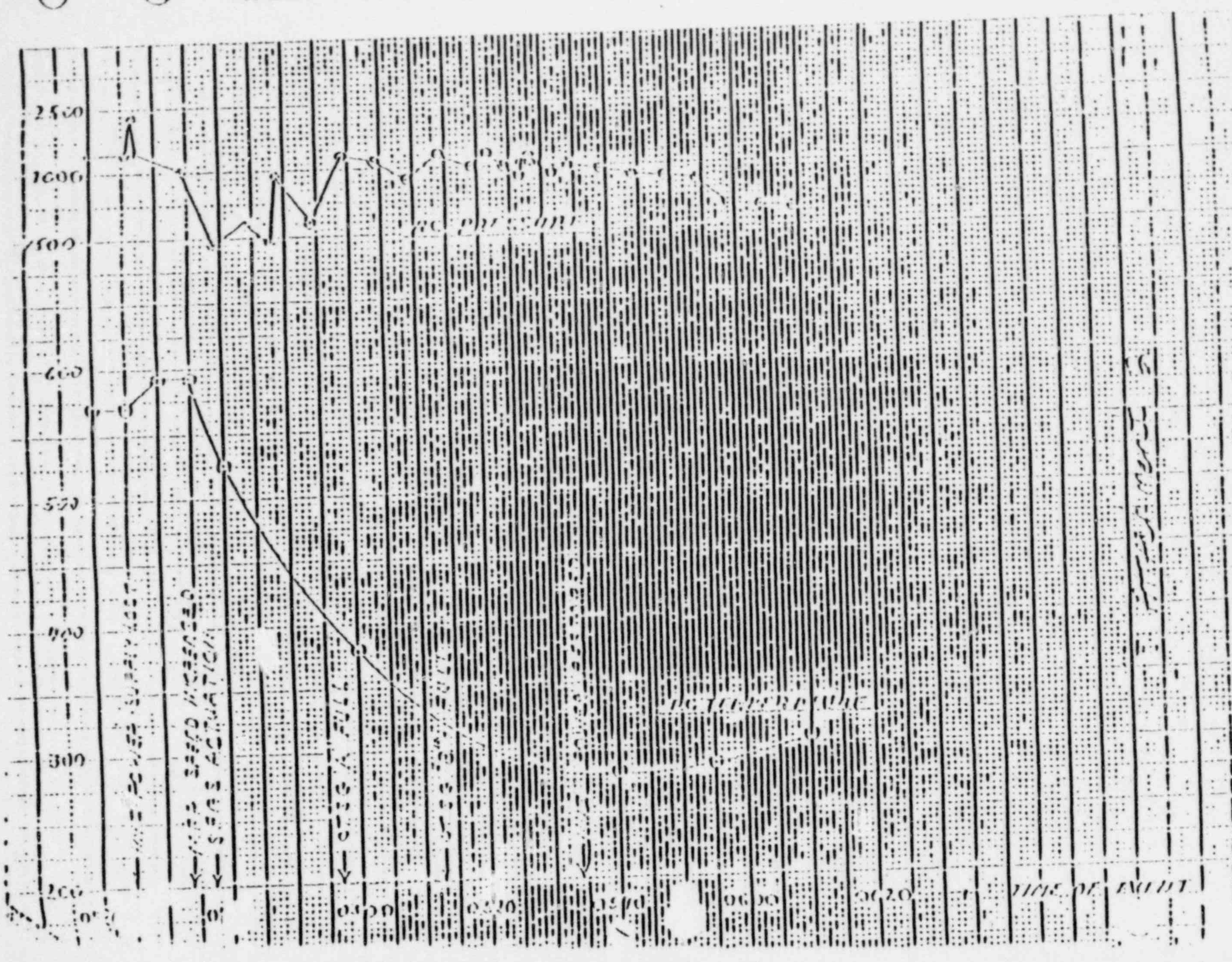
Operator begins to reduce RC pressure using pressurizer spray.

ICS closes turbine bypass valves to condenser.

Operator stops emergency FW flow.

Operator stops main FW pumps.

46 1470



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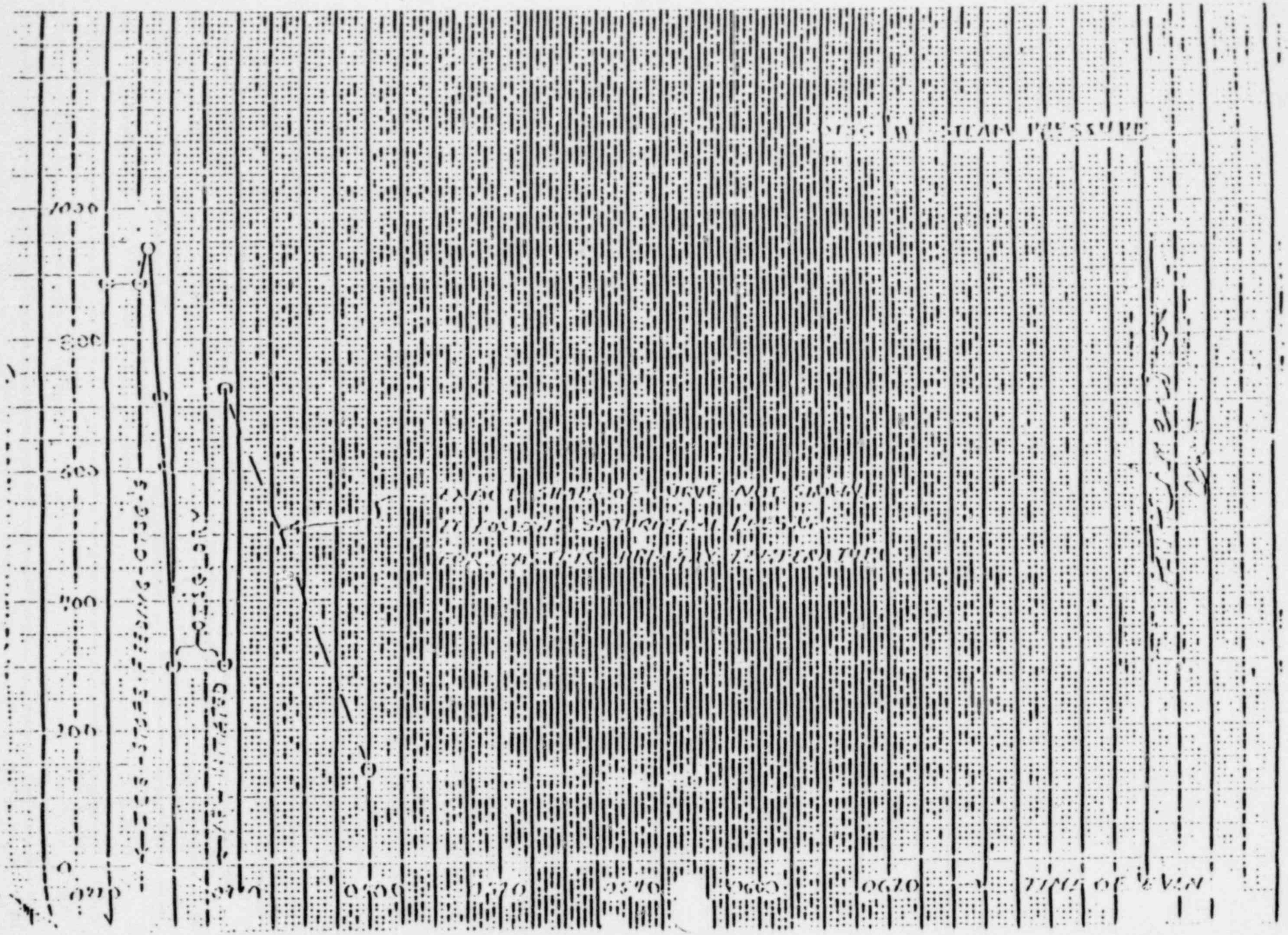
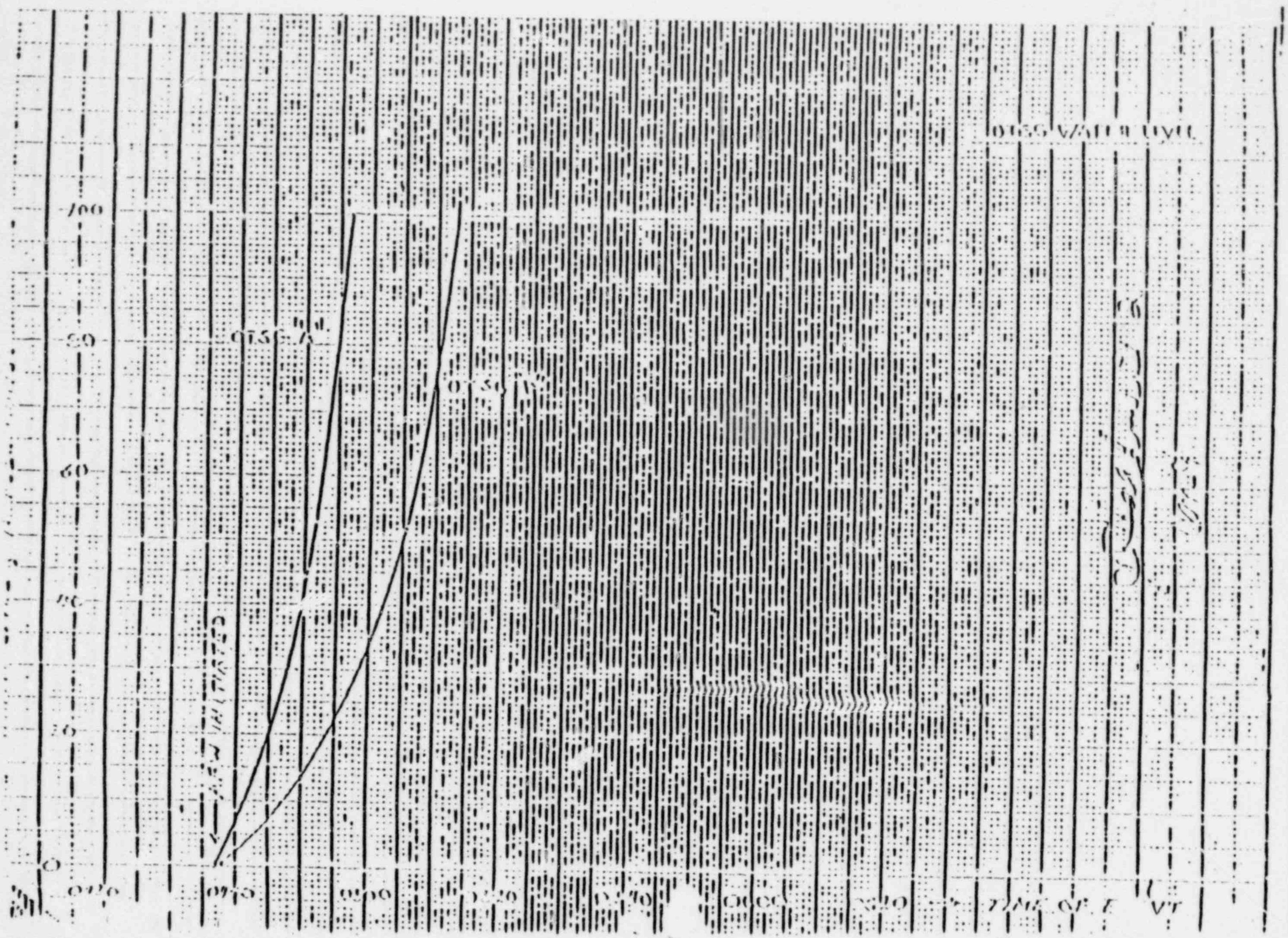


STEAM PRESSURE
1000
800
600
400
200
0
EXACT SHAPE OF CURVE NOT KNOWN
0620
TIME OF

46 1470



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 29 1979

MEMORANDUM FOR: Domenic B. Vassallo, Assisant Director for Light Water Reactors, NRR

FROM: Dudley Thompson, XOOS

SUBJECT: INFORMATION FOR BOARD NOTIFICATION - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2

REFERENCES: 1. Memo: Thomson to Vassallo dtd 3/1/79
2. Memo: Thomson to Vassallo dtd 3/12/79

As noted in the above referenced submittals additional information in the form of staff discussion and evaluation would be forthcoming on the captioned board notification. Enclosed is the additional information for submittal to the appropriate Boards.

Dudley Thompson, Executive Officer
for Operations Support
Office of Inspection and Enforcement

Enclosures:
1. Memo: Moseley to Thompson dtd 3/28/79 w/encls
2. Memo: Moseley to Thompson dtd 3/29/79

cc: N. C. Moseley, IE, w/o encl
S. E. Bryan, IE, w/o encl
J. F. Streeter, RIII, w/encl
J. S. Creswell, RIII, w/encl
G. C. Gower, IE, w/encl
IE Files w/encl

CONTACT: G. C. Gower, IE
49-27246

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 29, 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

In light of the transient experienced at Three Mile Island on March 28, 1979, we will review our preliminary evaluation of Item 3 contained in my March 28, 1979 memorandum to you. It is possible that the additional information will cause our assessment to change.

Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

cc: S. E. Bryan
E. L. Jordan
R. F. Heishman, RIII
J. C. Stone
D. C. Kirkpatrick
G. C. Gower
V. D. Thomas

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 2 8 1979

MEMORANDUM FOR: Dudley Thompson, Executive Officer for Operations Support, IE

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, IE

SUBJECT: NOTIFICATION OF LICENSING BOARDS

On February 23, 1979, six items concerning Babcock and Wilcox designed nuclear plants were sent to you for forwarding to the appropriate licensing boards. At that time only a preliminary evaluation had been done. We have completed o evaluation of each of the items and that information is encl d. This additional information should be forwarded to the licensing boards.

Norman C. Moseley
Director
Division of Reactor
Operations Inspection, IE

Enclosure:
Evaluations of Concerns

cc: S. E. Bryan
E. L. Jordan
R. F. Heishman, RIII
J. C. Stone
D. Kirkpatrick
~~L. C.~~ Gower
V. D. Thomas

CONTACT: J. C. Stone
(x28019)

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

1. During a recent inspection at Davis-Besse Unit 1 information has been attained which indicates that at certain conditions of reactor coolant viscosity (as a function of temperature) core lifting may occur. The licensee informed the inspector that this issue involves other B&W facilities. The Davis-Besse FSAR states in Section 4.4.2.7:

   > The hydraulic force on the fuel assembly receiving the most flow is shown as a function of system flow in Figure 4-39. Additional forces acting on the fuel assembly are the assembly weight and a hold down spring force, which resulted in a net downward force at all times during normal station operation.

   The licensee states that there is a 500°F interlock for the starting of the fourth reactor coolant pump. However, no Technical Specification requires that the pump be started at or above this temperature. A concern regarding this matter would be if assemblies moved upward into a position such that control rod movement would be hindered.

DISCUSSION AND EVALUATION

The potential for core lifting in B&W plants is a concern which has been previously reviewed by NRR. The concern was first raised in connection with the Oconee 2 and 3 reactors, where the primary coolant flow rates were found to be in excess of the design flow rates. For example, the Unit 2 flow rate was found to be 111.5% of the design flow rate. Since this was very near the predicted core lift flow rate of 111.9%, an analysis was done by B&W to determine what affect core lifting would have on the previous safety analysis for these plants. This analysis (dated May 2, 1975) indicated that the potential for core lifting did not result in an unreviewed safety question. A subsequent review of this B&W analysis by NRR also concluded that an unsafe condition did not exist (letter from R. A. Purple to Duke Power, dated 9/24/75). It should be noted that the potential vertical displacement of the core is limited to a very small distance by the upper core support structure. Core lifting at power would result in a slight reduction in reactivity since the rising fuel would tend to engage the withdrawn control rods to a slightly greater extent than it would in the bottomed condition. The amount of this change in reactivity is, of course, available for reinsertion should the fuel settle back to its original position. The potential reactivity increase caused by the settling of the 16 centrally located control rod assembly elements (assumed to have been subject to lifting in the Oconee 2 reactor) was calculated to be 0.1% Δ K/K. This value is insufficient to have much effect on the accident and transient safety analyses.

An additional concern was the potential for damage to the fuel assembly and fittings which might be caused by fretting due to repetitive fuel movement. Consequently, Duke Power was requested by NRR to make certain examinations of the Oconee 2 fuel during the first refueling to confirm that fuel element motion was not occurring. The results of this examination (Letter from W. O. Parker to R. C. Rusche dated 7/21/76) showed that no fuel lifting or other type of motion had occurred during the first cycle of operation.

After the core lift concern was identified, B&W developed newer types of fuel holddown springs which provide more margin against core lifting than the previous springs did. It is our understanding that the newer types of springs have been installed in all B&W reactors.

For these reasons, we believe that there is presently little likelihood that core lifting will occur during normal power operation. At lower temperatures, there is an increased flow induced lifting force on the fuel due to the higher viscosity of the reactor coolant. Consequently, we view the restriction against 4 pump operation below 500°F as a prudent precaution against fuel fretting. However, since the potential for core lifting has little safety significance and because critical operation below 500°F is not permitted, we have no basis to recommend including this restriction in the Technical Specifications.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

2. Inspection Report 50-346/78-06, paragraph 4, reported reactivity - power oscillations in the Davis-Besse core. These oscillations have also occurred at Oconee and are attributed to steam generator level oscillations. B&W report BAW-10027 states in A9.2:

> The OTSG laboratory model test results indicated that periodic oscillations in steam pressure, steam flow, and steam generator primary outlet temperatures could occur under certain conditions.
>
> It was shown that the oscillations were of the type associated with the relationships between feedwater heating chamber pressure drop and tube nest pressure drop, which are eliminated or reduced to levels of no consequence (no feedback to reactor system) by adjustment of the tube nest inlet resistance. As a result of the tests, an adjustable orifice has been installed in the downcomer section of the steam generators to provide for adjustment of the tube nest inlet resistance and to provide the means for elimination of oscillations if they should develop during the operating lifetime of the generators. The initial orifice setting is chosen conservatively to minimize the need for further adjustment during the startup test program.

We also note that the effect on the incore detector system for monitoring core parameters during the oscillations is not clear.

DISCUSSION AND EVALUATION

Power Oscillations of the order of 1.5% of full power have been observed at all of the Oconee plants and are considered normal. In 1977 the power oscillations experienced by the Oconee 3 reactor increased to a maximum of 7.5% of full power. At that time the problem was reviewed by NRR with the conclusion that there was no significant safety consideration at that value (Note to B. C. Buckley from S. D. MacKay, dated January 27, 1978). It should be noted that the 7.5% power oscillations cause about a 1°F oscillation in core average temperature due to the short period of the oscillations. The important core safety parameters, which are, the departure from nucleate boiling ratio and the average maximum linear heat generation rate are affected very little by oscillations of this amplitude. The primary cause of the power oscillations is believed to be a fluctuation of the secondary water level in the steam generators. This can be minimized by increasing the flow resistance

in the downcomer region of the steam generators. The corrective effort at Oconee 3 was complicated by the fact that the orifice plate provided for this purpose could not be fully closed.

However, the oscillations at other B&W plants have been kept to about 1.5% of full power by appropriate adjustment of the downcomer flow resistance. For these reasons, the power oscillations at B&W plants are not considered to be a significant safety concern.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

3. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% power with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Enclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

DISCUSSION AND EVALUATION

The event at Davis Besse which resulted in loss of pressurizer level indication has been reviewed by NRR and the conclusion was reached that no unreviewed safety question existed.

The pressurizer, together with the reactor coolant makeup system, is designed to maintain the primary system pressure and water level within their operational limits only during normal operating conditions. Cooldown transients, such as loss of offsite power and loss of feedwater, sometimes result in primary pressure and volume changes that are beyond the ability of this system to control. The analyses of and experience with such transients show, however, that they can be sustained without compromising the safety of the reactor. The principal concern caused by such transients is that they might cause voiding in the primary coolant system that would lead to loss of ability to adequately cool the reactor core. The safety evaluation of the loss of offsite power transient shows that, though level indication is lost, some water remains in the pressurizer and the pressure does not decrease below about 1600 psi. In order for voiding to occur, the pressure must decrease below the saturation pressure corresponding to the system temperature. 1600 psi is the saturation pressure corresponding to 605°F, which is also the maximum allowable core outlet temperature. Voiding in the primary system (excepting the pressurizer) is precluded in this case, since pressure does not decrease to saturation.

The safety analysis of more severe cooldown transients, such as the loss of feedwater event, indicates that the water volume could decrease to less than the system volume exclusive of the pressurizer. During such an event, the emptying of the pressurizer would be followed by a pressure reduction below the saturation point and the formation of small voids throughout much of the primary system. This would not result in the loss of core cooling because the voids would be dispersed over a large volume and forced flow would prevent them from coalescing sufficiently to prevent core cooling. The high pressure coolant injection pumps are started automatically when the primary pressure decreases below 1600 psi. Therefore, any pressure reduction which is sufficient to allow voiding will also result in water injection which will rapidly restore the primary water to normal levels.

For these reasons, we believe that the inability of the pressurizer and normal coolant makeup system to control some transients does not provide a basis for requiring more capacity in these systems.

General Design Criterion 13 of Appendix A to 10 CFR 50 requires instrumentation to monitor variables over their anticipated ranges for "anticipated operational occurrences". Such occurrences are specifically defined to include loss of all offsite power. The fact that T cold goes off scale at 520°F is not considered to be a deviation from this requirement because this indicator is backed up by wide range temperature indication that extends to a low limit of 50°F. Neither do we consider the makeup flow monitoring to deviate since the amount of makeup flow in excess of 160 gpm does not appear to be a significant factor in the course of these occurrences.

The loss of pressurizer water level indication could be considered to deviate from GDC 13, because this level indication provides the principal means of determining the primary coolant inventory. However, provision of a level indication that would cover all anticipated occurrences may not be practical. As discussed above, the loss of feedwater event can lead to a momentary condition wherein no meaningful level exists, because the entire primary system contains a steam water mixture.

It should be noted that the introduction to Appendix A (last paragraph) recognizes that fulfillment of some of the criteria may not always be appropriate. This introduction also states that departures from the Criteria must be identified and justified. The discussion of GDC 13 in the Davis Besse FSAR lists the water level instrumentation, but does not mention the possibility of loss of water level indication during transients. This apparent omission in the safety analysis will be subjected to further review.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

4. A memo from B&W regarding control rod drive system trip breaker maintenance is attached as Enclosure 2. This memo should be evaluated in terms of shutdown margin maintenance and ATWS considerations particularly in light of large positive moderator coefficients allowable with B&W facilities.

DISCUSSION AND EVALUATION

Our investigation of the above circuit breaker problems has revealed that eight failures of reactor scram circuit breakers to trip during test have been reported from Babcock & Wilcox (B&W) type operating facilities since 1975. In each case, the faulty circuit breaker was identified as a GE type AK-2 series (i.e., AK-2A-15, 24, or 50). The causes for failure were attributed to either binding within the linkage mechanism of the undervoltage trip device (UV) and trip shaft assembly or an out-of-adjustment condition in the same linkage mechanism. B&W and GE determined that the binding and the out-of-adjustment conditions resulted from inadequate preventive maintenance programs at the affected operating facilities.

In addition to the breaker problems experienced at the B&W facilities, three circuit breakers of the aforementioned GE type failed in similar fashion at the Oyster Creek operating facility on November 26, 30, and December 2, 1978. As in each case above, cleaning and relubricating of the UV/trip shaft assembly within the circuit breaker was required to correct the problem. It is significant to note that during the November 30, 1978 event, both redundant service water pump circuit breakers failed to trip as required during the loss of off-site power test. These failures in turn created a potential overload condition on the emergency busses during the sequential bus loading by each diesel generator.

However, both diesel generators successfully picked up their required bus loads without experiencing a unit shutdown from an overload condition. With respect to the generic implications and safety significance of this issue, both B&W and GE are in the process of issuing alert letters to their customers. These letters are scheduled for issuance by late March and will describe the causes for failure and provide recommendations to resolve the problem.

Based on our study findings and on information obtained in discussions about the breaker problem with the knowledgeable people from B&W, GE and Region II, we plan to issue an IE Circular covering the matter. The thrust of the Circular will be directed toward the need for adequate preventive maintenance programs at all operating facilities. Specific recommendations from GE to resolve the above breaker problem will also be mentioned in the Circular.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

5. Inspection and Enforcement Report 50-346/78-17, paragraph 6 refers to inspection findings regarding the capability of the incore detector system to determine worst case thermal conditions. The reactor can be operated per the Technical Specifications with the center incore string out of service. If the peak power locations is in the center of the core (this has been the case at Davis-Besse), factors are not applied to conservatively monitor values such as F_Q and F delta H.

DISCUSSION AND EVALUATION

We do not believe that there is a valid basis for requiring the center string of incore detectors to be always operable in B&W reactors.

The power distributions for various plant conditions, throughout the fuel cycle, are calculated prior to the operation of the reactor. The power distribution is verified at the beginning of operation, and periodically thereafter, by comparison with the available incore detectors. The power in fuel assemblies that lack detectors (including those with failed detectors) is derived by using the known power distribution to determine the power ratios between such an assembly and nearby assemblies that have detectors. These ratios can then be multiplied by the power in the measured assemblies to derive the power level in any specific unmeasured assembly. The central assembly is not fundamentally different than any other assembly in this respect. Although this assembly is the highest powered assembly in the Davis Besse reactor at the beginning of the fuel cycle, this is not the case at all reactors. Nor does the central assembly have the highest power, in the Davis Besse reactor, at the end of the first fuel cycle. Since there is some variation between the calculated power distributions and the actual ones, an appropriate margin is assumed for this variation in establishing the allowable power peaking factors.

Fixed incore detectors must function in an extremely harsh environment and are subject to high failure rates. In order to ensure that an adequate number will survive the fuel cycle, many more detectors are installed than are necessary for the power distributions determinations. To require the central string to be always operable would likely result in unnecessary power restrictions. Neither the standard Technical Specifications (STS) for B&W plants nor the STS for CE plants (which also have fixed incore detectors) require the central detectors to be operable.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

6. Enclosure 3 describes an event that occurred at a B&W facility which resulted in a severe thermal transient and extreme difficulty in controlling the plant. The aforementioned facilities should be reviewed in light of this information for possible safety implications.

DISCUSSION AND EVALUATION

Following the cooldown transient at Rancho Seco, NRR evaluated the event and concluded that no structural damage had occurred to the primary coolant system which would preclude future operation of Rancho Seco. However, in their safety evaluations they concluded that positive steps should be taken to preclude similar transients and that the generic implications of this event should be reviewed. In addition, IE initiated a Transfer of Lead Responsibility, Serial No. IE-ROI 78-04, dated April 25, 1978, recommending that:

1. NRR perform a generic review of the non-nuclear instrumentation power supplies for other B&W units, if design changes to the non-nuclear instrumentation (NNI) power supplies are required at Rancho Seco.

2. NRR evaluate the susceptibility of B&W plants to other initiating events or failures which could cause similar significant cooldown transients.

This event is currently being evaluated by NRR.

5.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

January 19, 1979

MEMORANDUM FOR: N. C. Moseley, Director, Division of Reactor Operations Inspection, IE

H. D. Thornburg, Director, Division of Reactor Construction Inspection, IE

FROM: James G. Keppler, Director, RIII

SUBJECT: RECOMMENDATION FOR NOTIFICATION OF LICENSING BOARDS AND REQUEST FOR TECHNICAL ASSISTANCE (AITS F30468H2)

The enclosed inspector memorandum dated January 8, 1979, with enclosures, identifies several potential problems which are being or will be pursued at Davis-Besse 1 which appear to be generic to B&W plants. In addition to the items identified in the memorandum, an issue (described in enclosed Action Item AITS F30385H2) concerning GDC 17 which was recently resolved at Davis-Besse 1 could possibly be common to other plants under review by NRR (e.g., Davis-Besse 2 and 3). The GDC 17 item and some of the other items may only be generic to B&W plants having Bechtel as the architect-engineer. We are aware that some of the items have been previously identified and dispositioned at other plants.

In accordance with the inspector's recommendation, RIII supervision has reviewed the materiality and relevancy of these matters to all pending cases before Boards involving B&W as the NSSS supplier. Based on information we have on those cases (Davis-Besse 2 and 3, Midland 1 and 2, Greene County, Three Mile Island 2), guidance given in MC 1530, and a liberal interpretation of the MC 1530 words "...any new information that could reasonably be regarded as putting a new or different light upon an issue before the Board or as raising a new issue", RIII believes NRC policy dictates that the information be forwarded to all sitting Boards for cases involving B&W as the NSSS supplier. To our knowledge, none of the information relates to specific issues under consideration in the pending hearings. RIII does not know the significance of the information as it may affect current staff positions.

SHAPER EXHIBIT #13
8-3-79 SC

Although RIII believes that NRC policy as described in MC 1530 dictates the transmittal of this information to sitting Boards, RIII questions the appropriateness of doing so. It would seem that a more effective and less premature way of handling this information would be for NRR to review and disposition the information during the development of the SER and SER Supplement relating to OL issuance for the affected plants. In the case of Three Mile Island 2 and other operating plants where the SER and SER Supplement have already been issued, the information could be evaluated for application to those plants as an NRR generic review task.

For your information, listed below is the status of reviews at Davis-Besse 1 of the items in the inspector memorandum:

Item 1 - During a recent inspection the licensee was requested to provide information to reconcile the apparent inconsistency between the FSAR statement on fuel assembly net holddown force and the administrative requirement to place restrictions on starting the fourth reactor coolant pump. This information will be available February 1979.

Item 2 - We have been following the licensee's efforts to determine the magnitude of the power oscillations. To date the maximum oscillations have been approximately 1.5% and do not appear to present a safety problem.

Item 3 - The pressurizer level question is presently the subject of communications between NRR and the licensee. We have not addressed the possibility that Tcold and makeup instrumentation do not meet GDC 17.

Item 4 - To our knowledge, this problem has not developed at DB 1. We plan to inspect this item in February 1979.

Item 5 - In response to an item of noncompliance, the licensee is developing criteria for detector substitution when the reactor is operated with incore strings out of service.

Item 6 - To our knowledge, this problem has not developed at DB 1. We plan to inspect this item in February 1979.

RIII will use the results of any technical reviews conducted which relate to items in the inspector memorandum to disposition the items as they relate to Davis-Besse 1. By copy of this letter, the Assistant Directors for Technical Programs and Field Coordination are requested to provide RIII with answers to the following questions:

Assistant Director for Technical Programs

1. Has NRR generically determined that the B&W core lift problem is not an unreviewed safety question?

2. Has NRR generically determined that the B&W power oscillation problem is not an unreviewed safety question?

3. Does the failure of Tcold and makeup instrumentation to follow the transient constitute a GDC 13 problem?

Assistant Director for Field Coordination

1. Is there a need to develop standard B&W technical specifications for continued plant operations with failed incore detector strings?

2. Is there a need to develop standard B&W technical specifications for restrictions on starting a fourth reactor coolant pump below certain temperatures?

For your convenience, the items in the inspector memorandum have been retyped on separate pages. If you need additional information please contact J. S. Creswell (387-9311) or J. F. Streeter (387-9228) of my staff.

James G. Keppler
James G. Keppler
Director

Enclosures:
1. Memorandum from J. S. Creswell to J. F. Streeter, dtd, 1/8/79
2. Retyped excerpts (6) from the 1/8/79 memorandum
3. Memorandum from J. F. Streeter to R. W. Woodruff, dtd, 6/9/78

cc: w/enclosures
E. L. Jordan, IE
S. E. Bryan, IE
J. S. Creswell, RIII

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

January 8, 1979

Docket No. 50-500/501
50-329/330

MEMORANDUM FOR: J. F. Streeter, Chief, Nuclear Support Section 1

FROM: J. S. Creswell, Reactor Inspector

SUBJECT: CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2

During the course of my inspections at Davis-Besse, certain issues have come to my attention which I am submitting for consideration for forwarding to the Atomic Safety and Licensing Board which has proceedings pending for the aforementioned facilities. This submittal is made pursuant to Regional Procedure 1530A (November 16, 1978), step 3 and information supplied to me per step 1. The issues for consideration are:

1. During a recent inspection at Davis-Besse Unit 1 information has been attained which indicates that at certain conditions of reactor coolant viscosity (as a function of temperature) core lifting may occur. The licensee informed the inspector that this issue involves other B&W facilities. The Davis-Besse FSAR states in Section 4.4.2.7:

   > The hydraulic force on the fuel assembly receiving the most flow is shown as a function of system flow in Figure 4-39. Additional forces acting on the fuel assembly are the assembly weight and a hold down spring force, which resulted in a net downward force at all times during normal station operation.

   The licensee states that there is a 500°F interlock for the starting of the fourth reactor coolant pump. However, no Technical Specification requires that the pump be started at or above this temperature. A concern regarding this matter would be if assemblies moved upward into a position such that control rod movement would be hindered.

2. Inspection Report 50-346/78-06, paragraph 4, reported reactivity - power oscillations in the Davis-Besse core. These oscillations have also occurred at Oconee and are attributed to steam generator level oscillations. B&W report BAW-10027 states in A9.2:

The OTSG laboratory model test results indicated that periodic oscillations in steam pressure, steam flow, and steam generator primary outlet temperatures could occur under certain conditions.

It was shown that the oscillations were of the type associated with the relationships between feedwater heating chamber pressure drop and tube nest pressure drop, which are eliminated or reduced to levels of no consequence (no feedback to reactor system) by adjustment of the tube nest inlet resistance. As a result of the tests, an adjustable orifice has been installed in the downcomer section of the steam generators to provide for adjustment of the tube nest inlet resistance and to provide the means for elimination of oscillations if they should develop during the operating lifetime of the generators. The initial orifice setting is chosen conservatively to minimize the need for further adjustment during the startup test program.

We also note that the effect on the incore detector system for monitoring core parameters during the oscillations is not clear.

3. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% power with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Enclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

   Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

4. A memo from B&W regarding control rod drive system trip breaker maintenance is attached as Enclosure 2. This memo should be evaluated in terms of shutdown margin maintenance and ATWS considerations particularly in light of large positive moderator coefficients allowable with B&W facilities.

5. Inspection and Enforcement Report 50-346/78-17, paragraph 6 refers to inspection findings regarding the capability of the incore detector system to determine worst case thermal conditions. The reactor can be operated per the Technical Specifications with the center incore string out of service. If the peak power locations is in the center of the core (this has been the case at Davis-Besse), factors are not applied to conservatively monitor values such as F_Q and F delta H.

6. Enclosure 3 describes an event that occurred at a B&W facility which resulted in a severe thermal transient and extreme difficulty in controlling the plant. The aforementioned facilities should be reviewed in light of this information for possible safety implications.

J. S. Creswell

J. S. Creswell
Reactor Inspector

Enclosures: As stated

cc w/o enclosures:
G. Fiorelli
R. C. Knop
T. N. Tambling

Preliminary copy

TOLEDO EDISON

LOWELL E. ROE
Vice President
Facilities Development
(419) 259-5242

Docket No. 50-346

License No. NPF-3

Serial No. 475

December 22, 1978

Director of Nuclear Reactor Regulation
Attention: Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Reid:

In response to the December 20, 1978, telephone conversation between your Mr. Guy Vissing and our Mr. E. C. Novak, and the December 20, 1978 telephone conversation between NRC Region III personnel (G. Fiorelli, R. Knop, T. Tambling and J. Streeter) and our Mr. E. C. Novak, attached is an additional safety evaluation supporting continued operation of Davis-Besse Nuclear Power Station Unit 1. This additional safety evaluation supplements the analysis we provided to you by our letter dated December 11, 1978, Serial No. 471. The attached safety evaluation analyzes the transient resulting from the operator not controlling steam generator level at 35 inches in accordance with current operating procedures.

Yours very truly,

Lowell E. Roe

LER:CRD

Enclosure

bj 4/7

Docket No. 50-346
License No. NPF-3
Serial No. 475
December 22, 1978

Additional Safety Evaluation
of Transient Resulting from
Inability of Operator to Control
Steam Generator Level at 35 Inches

I. INTRODUCTION

The Davis-Besse Unit 1 Steam and Feedwater Line Rupture Control System (SFRCS) design objectives are to prevent the release of high energy steam, to automatically start auxiliary feedwater (AFW), and to provide adequate AFW, via essential steam generator level control, to remove decay heat during anticipated and design basis events when AFW is required. Table 1 correlates the station variables and accident conditions for which AFW actuation is required. For all actuation signals, the SFRCS initiates and controls AFW addition automatically to maintain a 120" level (96" indicated on the startup range instrumentation) in the steam generators.

The recent natural circulation test at Davis-Besse 1 (TP800.04) demonstrated that a 35-inch (indicated) steam generator level of AFW provides adequate natural circulation for decay heat removal.

The auto essential SG level control setpoint of 120-inches (96-inch-indicated) is thus in excess of minimum SG level requirements.

Operating procedures requiring manual control of steam generator level at 35-inches on the startup range level indicators following non-LOCA events were developed and used at Davis-Besse Unit 1 pending installation of permanent design changes to the SFRCS. Margin in maintenance of indicated pressurizer level and assurance of adequate natural circulation capability will exist through operator intervention during conditions where AFW is required.

Inability of the operator to comply with the present operating procedures will possibly result in a momentary loss of pressurizer level and/or level indication under certain conditions, but will not produce consequences which are non-reversible or detrimental to safe operation of Davis-Besse Unit 1.

II. DISCUSSION

The following section is divided into three segments: Relationship with Events Presented in the Davis-Besse Unit 1 FSAR, Loss of Offsite Power, and Loss of Feedwater.

A. Relationship with events presented in the FSAR

Addition of auxiliary feedwater at rates considerably greater than the decay heat generation rate will result in overcooling of the reactor coolant, contraction and a reduction of pressurizer level. This sequence of events is typical of several transients presented in the FSAR which have been submitted to the NRC and approved as a part of the licensing process. Overcooling transients can be caused by a variety of circumstances, failures, combinations of operating equipment, and improper operator interactions. From a practical viewpoint each single discoverable possible transient cannot be analyzed and presented as a part of the FSAR analysis, but a broad variety of transients have been selected. This specific transient fits within that broad category. Each of the FSAR transients has been demonstrated to produce acceptable results.

Overcooling transients resulting from a variety of causes are described in Section 15.2.10 "Excessive Heat Removal due to Feedwater Malfunctions". This section describes a transient resulting from excessive main feedwater addition, which is similar to the specific transient of increased level addition by auxiliary feedwater. Further information is presented in response to questions 15.2.15 and 15.2.16.

The steam line break (see FSAR sections 15.4.4, 15.4.8, 15.4.1) is the most severe overcooling transient, in that the reactor coolant system is decreased 50°F in average core temperature over a 30 second time period.

This is compared with the cooldown in question, which takes a much longer time to achieve a similar temperature drop and system conditions. During the steam line break, RC system pressure is reduced from 2200 psia to about 900 psi as system temperature is driven toward equilibrium with the unaffected (pressurized) steam generator attaining saturation temperature of about 530°F. The pressurizer is near empty at about 20 seconds and thereafter loses its influence on the system, thus permitting the upper elevations of the reactor coolant loop to approach saturation as cooldown continues toward 530°F. High pressure injection (HPI) pumps are actuated on low RC pressure such that pressurizer level will be restored. As shown in Figures 15.4.4-1 and 15.4.4-2 of the Davis-Besse Unit 1 FSAR, the rapid cooldown of RCS after reactor trip is limited by the pressure maintained in the pressurized steam generator in much the same fashion as anticipated for events such as the event of concern. As the RCS approaches saturation, core cooling is not impeded. Minimum DNBR>1.3 occurs just before reactor trip and subsequently increases with substantial margin throughout the remainder of the cooldown.

The close relationship of the auxiliary feedwater level increase as an overcooling transient with these similar overcooling transients allows us to draw the conclusion that no unreviewed safety question exists. To show a comparison to the detailed analyses reported in the FSAR, we have performed conservative bounding analyses of two representative cases.

3. Loss of Feedwater and Loss of Offsite Power

We have analyzed two transients resulting from auxiliary feedwater addition and establishment of SG level above the operating procedure 35" limit. The two transients examined are a loss of offsite power (reactor coolant pumps stop, makeup stops, main feedwater stops) and a loss of feedwater (reactor coolant pumps continue, makeup continues).

Of these two transients the loss of feedwater results in the greater volumetric coolant contraction, because the forced coolant flow (RC Pumps operating) causes a faster rate of heat rejection to the steam generator.

1. Loss of Offsite Power

Preliminary calculations for a reactor trip following a loss of offsite power show that the pressurizer loses indication but does not empty. The assumptions used to derive this result included full runout auxiliary feedwater flow (~2400 gpm) resulting in a fill time to 120" of about 4 minutes. No net mass change to the primary coolant (no makeup, no letdown) was considered, even though the makeup controls would respond to decreasing pressurizer level by increasing the net input to above 200 gpm. At the termination of the transient the pressurizer level is slightly above the outlet into the surge line. Reactor coolant pressure reaches about 1600 psi and high pressure injection may be automatically initiated.

Although the net makeup was not considered, it would in fact cause the pressurizer to refill to the normal level. At the same time compression of the steam would cause a partial repressurization of the system ensuring that the coolant remains subcooled. This transient presents no safety concerns.

2. Loss of Feedwater

This transient has a greater reactor coolant contraction than the loss of offsite power case, resulting in emptying of the pressurizer. Consequently it will be described in greater detail.

A brief summary of the events is:

• Reactor trip	Time = 0
• Makeup control valve opens wide admitting full makeup to reactor coolant system	Time = 0^+
• AFW initiated	Time ≈ 40 sec
• Pressurizer empties; RC system pressure slightly greater than 1800 psi	Time ≈ 2 min
• HPI initiated by SFAS; makeup isolated	Time ≈ 2^+ min
• Steam generator level = 10 ft; voids exist in reactor coolant	Time ≈ 4 min
• HPI inflow replaces volume occupied by voids; pressurizer level begins to be restored	Time ≈ 7-8 min

The major concerns that evolve from this transient are the disposition of the steam voids and the approach to DNB. Both of the concerns are ameliorated by the reactor coolant pumps.

Steam voids will not collect in reactor coolant piping and no flow blockage will occur because of dispersal and mixing by the forced flow. DNB acceptance criterion limit will be met because the power output of the core is at the decay heat level and all reactor pumps are operating, maintaining core heat removal. We conclude that no safety problem exists.

TABLE 1: STEAM AND FEEDWATER LINE RUPTURE CONTROL SYSTEM (SFRCS) ACTUATION PARAMETERS

Actuation Parameter		Accident
Station Variables	Setpoint	
1. Low Steam Line Pressure	$<$ 591.6 psig[1,2]	Steam Line Break Feedwater Line Break
2. Low SG Level	$\leq$ 17 inches[1]	Loss of F/W
3. SG Pressure Minus Main Feedwater Line Pressu.	$>$ 197.6 psi[1]	FWLB, LOMFW
4. Loss of All RC Pumps[3]		Loss of Off-Site Power

NOTES:

1. When actuated, SFRCS closes both main steam isolation valves, closes both main FW control and stop valves, initiates AFW and controls AFW to maintain a 120 inch level in the SGs.

2. Alignment of AFW to a pressurized SG is provided for steam and feedwater line breaks.

3. AFW initiation but steam and feedwater line isolation does not occur.

III. Bounding Analysis of Loss of Feedwater Event With Failure of Operator to Control Feedwater Level at 35"

Introduction:

The following bounding analysis conservatively predicts the events occurring within the primary reactor coolant system and reactor following a loss of main feedwater from 100% power for the Davis-Besse Unit 1. Auxiliary feedwater control has been assumed at 10 feet within both steam generators.

Results:

Because of the conservative, bounding, nature of this calculation, the overcooling of the primary system due to auxiliary feedwater injection causes a contraction of coolant volume sufficient to create steam within the primary system. The steam is shown to be uniformly distributed within the RCS and the void fraction is 4%. The reactor coolant pumps maintain full capability. The DNB ratio is shown to exceed 2.0 and no return to criticality potential exists. Thus, during the course of the incident, no core problems develop. Further, following the time of maximum contraction, the system recovers to full pressure, pressurizer function is regained and the reactor coolant returns to a subcooled water configuration without operator action.

Analysis:

The following assumptions have been made to assure the bounding nature of the results:

Reactor Power:

100% until boiling stops in the steam generators; 0% after that time. This assumption is conservative as core heat would compensate for the cooling caused by the auxiliary feedwater.

Initial Coolant Inventories Water:

RCS = 11290 ft^3

Pressurizer = 864 ft^3

These assumptions are nominal operating values.

Initial Temperatures:

The whole system is taken to be at $T_{average} = 582^oF$.

This assumption is a reasonable average.

Initial System Mass: ~ 500,000 lbm

The mass is figured from the temperature and volumes above.

Makeup System:

No credit is taken for additional makeup flow which will occur as the pressurizer loses level. (In all likelihood, the makeup system will contribute approximately 200 ft^3 extra liquid volume).

Local Power (kw/ft): 18.4 kw/ft

This value is taken as the maximum allowed by Technical Specifications.

Secondary Side Volume At 10 Foot Level

711 ft^3 per generator, actual volume.

Auxiliary Feedwater Flow:

166.5 ft^3/min. per generator actual value.

Auxiliary Feedwater Enthalpy:

8 Btu/lbm lower bound for maximum cooling.

With the initiating event, loss of main feedwater, the reactor coolant system pressure will start to rise. Reactor trip will occur on high RCS pressure. Following trip, the RCS pressure will fall because core power has been reduced and boiling of residual main feedwater or auxiliary feedwater is occurring in the steam generators. These events are almost identical to those which occur in a main feed line break and are analyzed in detail in Section 15.2.8 of the FSAR.

In short order, the system will return to its initial configuration but, because the auxiliary feedwater heat absorption rate exceeds the decay heat generation rate, the RCS continues to depressurize. During this phase, residual main feedwater and injected auxiliary feedwater will be boiled and vented through the steam generator safety relief valves. The primary system average temperature will fall to the saturation temperature of water at the safety valve set pressure. At this time, primary and secondary conditions are expected to be approximately as follows:

	Primary	Secondary
Pressure	1800 psia	980 psia
Temperature	542 F	542 F
Mass	503344 lbm	0 lbm
Liquid Volume in Press.	400 ft^3	N.A.
Time into Transient	$\sim$ 2 min.	$\sim$ 2 min.

It is conservative to assume complete boiling of the secondary side water and complete equilibrium between primary and secondary sides, as these assumptions lead to the maximum follow on injection of auxiliary feedwater and therefore, maximum contraction. RCS pressure is held up by the steam bubble in the pressurizer.

The time has been estimated by calculating the necessary energy loss by the primary system from its initial conditions, the mass of auxiliary feedwater required to remove this energy and then dividing by the auxiliary feedwater flow rate.

$$\text{time} \simeq \frac{(586 - 542)\ 503344}{(1194-8)\ \ 353\ \ 62} \simeq 54 \text{ sec.}$$

Six seconds was used to estimate the initial pressurization portion of the transient.

In performing the remainder of the evaluation 10 feet of cooled (40 F) auxiliary feedwater is placed in each steam generator and the thermal equilibrium condition calculated. Because after a 10 foot level is obtained this auxiliary feedwater flow stops, this condition represents the maximum contraction possible. The state variables resulting are:

	Primary	Secondary
Pressure	560 psia	560 psia
Temperature	478 F	478 F
Enthalpy of Water	462 Btu/lbm	462 Btu/lbm
Specific Volume	.020 ft^3/lbm	.020 ft^3/lbm

From the specific volume, the primary liquid volume can be calculated:

$\text{Vol} = MV_f = 10052\ ft^3$

As 10052 is smaller than the RCS minus pressurizer volume, the remaining volume must be filled with steam.

$V_{st} = 10426 - 10552 = 374\ ft^3 \approxeq 400\ ft^3$

400 ft^3 corresponds to a system void fraction of 3.8% $\simeq$ 4%, and as will be shown later, is of no consequence as far as core heating or system performance is concerned. This steam volume is larger than actually expected for two reasons: 1) some temperature difference would always exist between the primary and secondary systems, and 2) the effect of core decay heat has been ignored. Both of these would increase the primary side liquid temperature, thus increasing its volume and reducing the steam volume.

Following this state of maximum contraction, no further heat is removed from the RCS via the secondary side until the RCS rises in temperature due to decay heating; this will expand the liquid volume, compress the steam a[illegible] pressurize the RCS. As no mass can be lost from the secondary system prior to achieving 980 psia the first reheating stage will end at a primary system pressure, temperature, and liquid volume of 980 psia, 542 F, 10832 ft^3. Subtracting 10426 from 10832 shows that about 400 ft^3 of fluid has been forced back into the pressurizer. Pressurizer function would that be restored (if not directly, then, by either the makeup or HPI system), the RCS subcooled and the transient ended.

Several questions exist about the transient:

I. How is the 400 ft^3 dispersed within the primary system and can that volume collect in one location? From the auxiliary feedwater flow rate, over 4 minutes are required to fill the generators. As the pressurizer has 400 ft^3 in it at 980 psia and the RCS has 400 ft^3 in it at maximum contraction, approximately 2 minutes are used to eject steam from the pressurizer to the RCS. Because this steam will be superheated when it enters the RCS it will first desuperheat and then condense at a rate governed by its expanding pressure compared to the contraction of the liquid coolant. In the time of 2 minutes the reactor coolant will have made about 8 complete circles of the primary system and the steam can be considered well mixed. As the flow velocity in the RCS will remain normal, about 25 ft/sec, steam water separation will tend not to occur. Some limited steam accumulation may occur in the upper head of the reactor vessel as in that specific location of the RCS, velocity is low.

II. How well will the pumps work? Experiments performed on steam carry over capability show that for void fractions up to 10% no loss of pump capability is observed. This is documented in Figure 5-47 of BAW-10104, "B&W's ECCS Evaluation Report With Specific Application to 177 FA Class Plants With Lower Loop Arrangement." Actually pump capability increases for the first 5% of void introduced into the system.

III. Will any return to power be encountered because of the low RCS temperature? A return to power can occur for a non-borated core at 490F. This temperature includes the assumption of the most reactive rod stuck out of the core; if that rod were taken as inserted the critical temperature would fall to at or below 400F. Although no credit was taken for HPI in calculating the RC steam volume below 1600 psia, the HPI will be injecting borated water and, therefore, preventing any return to power condition. If the primary system were to stabilize at 1600 psia and thus prevent the HPI from providing boron the RCS temperature would be at least 511F and, therefore, no return to power would be expected.

IV. Will DNB be encountered in the core? The maximum contraction condition is again:

P = 560 psia
T = 478F
a = 4%,

and occurs at least 5 minutes after power shutdown (trip occurs very early within 10 seconds of main feedwater loss). At this time, the decay heat rate is less than 3.2% using ANS + 20% (the LOCA evaluation curve). As low pressure and high void and high power are conservative bounds a DNB evaluation was performed at:

P = 500 psia
T = corresponding saturated value
a = 8%
power = 10%
W = full volumetric flow.

The resultant DNBR was >15 in the hottest channel with maximum design conditions assumed and well within acceptable values.

7. Will any steam remain trapped in the primary system? Some may be trapped for a short period of time in the upper head of the reactor vessel but this will be of no consequence and will eventually be condensed by thermal conduction through the interfacing water.

Conclusion

The maximum contraction of the RCS water has been calculated taking no credit for mitigating systems (makeup flow, HPI) and no credit for decay heating. No adverse consequences of the transient have been shown and, therefore, this transient poses no concerns to the safe operation of the plant.

IV. CONCLUSIONS

For SFRCS actuation and fill of the steam generators to the auto-essential level control point of 120" without operator action:

- No unreviewed safety question exists
- The loss of offsite power transient will not cause the pressurizer to drain although a loss of pressurizer indicated level will occur.
- The loss of feedwater transient may result in pressurize emptying, however acceptance criteria for DNB will be met. Steam bubbles which exist in the reactor coolant for a short time will be collapsed by HPI injection. Pressurizer refilling by HPI will occur.
- No return to power will result in the long term.

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505

Telephone: (804) 384-5111

June 12, 1978

SOM #382 620-0014
12343 T3.3.1
SIP #14/289

Mr. T. D. Murray, Station Superintendent
Davis-Besse Nuclear Power Station
5501 North State Route #2
Oak Harbor, Ohio 43449

Subject: CRDCS Trip Breaker Maintenance

Dear Terry:

In the past, some of our plants have experienced problems with CRDCS Trip Breakers. The problems have been traced to lack of preventive maintenance. B&W suggests that a planned, carefully executed, maintenance program be established using the maintenance program outlined in the Diamond Power CRDC System Vendor Manual. Particular attention should be directed to proper cycling, cleaning, and lubrication of the breakers.

B&W further recommends that this program be scheduled at a minimum frequency of every refueling cycle and more frequently for plants during startup, when the equipment is subject to adverse environmental conditions.

Our concern is that if proper maintenance is not accomplished, additional failures will occur resulting in an NRC demand for diverse qualified trip breakers. Also, we need to prevent all failures we can to reduce the number of lost capacity days.

If we can be of further assistance, please advise.

Yours truly,

for
F. R. Faist
Site Operations Manager

FRF:IDG:mlf
cc: W. E. Spangler
R. C. Luken
R. L. Pittman
D. A. Lee
J. S. Grant, TECo
E. C. Novak, TECo
C. R. Domeck, TECo
J. G. Evans, TECo
B. R. Beyer, TECo
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Jr
Ban B791

Babcock & Wilcox

Power Generation Group

P.O. Box 1260, Lynchburg, Va. 24505

Telephone: (804) 384-5111

August 9, 1978

SOM #403 620-0014
12322 T3.3.1
SIP #14/295

file

Mr. T. D. Murray, Station Superintendent
Davis-Besse Nuclear Power Station
5501 North State Route #2
Oak Harbor, Ohio 43449

Subject: SMUD Rapid Cooldown Transient

Dear Terry:

On March 20, 1978, Rancho Seco experienced a severe thermal transient initiated by the loss of electrical power to a substantial portion of the Non-Nuclear Instrumentation (NNI). The loss of power directly caused the loss of Control Room indication of many plant parameters, the loss of input of these parameters to the plant computer, and erroneous input signals (midrange, zero, or otherwise incorrect) to the Integrated Control System (ICS).

The plant response was not the usual transient in that the ICS responded to the erroneous input signals rather than actual plant conditions, and resulted in a Reactor Protection System (RPS) trip on high pressure. Subsequent to the Reactor Trip, the erroneous signals to the ICS contributed to the rapid cooldown of the RCS. Plant operators had extreme difficulty in determining the true status of some of the plant parameters and in controlling the plant because of the erroneous indications in the Control Room.

An investigation of the events following this loss of power points out a need for a close look at operator training and emergency operating procedures for any loss of NNI power (or portion thereof). The following recommendations are made to assist your staff in a review of training and procedures to assure proper operator action for events of this nature.

1. Operators should be trained to recognize a loss of power to all or a majority of their NNI (e.g. indicators fail to mid-range, automatic or manual transfer to alternate instrument strings brings no response). The loss of power is emphasized here rather than the failure of any one instrument or control signal which are adequately covered in current simulator training courses.

Babcock & Wilcox

2. Given that the operator can determine that electrical power has been lost to all or part of the NNI, he should know the location of the power supply breakers, and have a procedure available to quickly regain power.

3. If the fault cannot be cleared (i.e. the breakers to the power supplies reopen), the operator should have a list of alternate instrumentation available to him, and he should be thoroughly trained in its use. Examples are:

   a. ESFAS panels
   b. RPS panels
   c. ECI (Essential Controls and Instrumentation)
   d. SRCI (Safety Related Controls and Instrumentation)
   e. Remote shutdown panels
   f. Local gages
   g. Plant computer

4. Recognizing that no procedure can cover all possible combinations of NNI failures, the operator's response should be keyed to certain variables. If the operator realizes that he has an instrumentation problem (as opposed to a LOCA or steam line break, for example), he can limit the transient by controlling a few critical variables:

   a. Pressurizer level (via HPI or normal Makeup Pumps)
   b. RCS pressure (via Pressurizer heaters, spray, E/M relief valves, etc.)
   c. Steam Generator level (via feed flow, feedwater valves, etc.)
   d. Steam Generator pressure (via turbine bypass system)

   The pressurizer level and RCS pressure assure that the Reactor Coolant System is filled; the Steam Generator level and pressure assure adequate decay heat removal.

Attachments 1 and 2 are provided to give a brief description of the events following this loss of NNI power at Rancho Seco. As can be seen by this transient, prompt precise operator action and the ability to recognize a loss of NNI power are critical factors in limiting the severity of a transient such as this.

If you have any questions or comments, please advise.

Yours truly,

Ivan D. Green

Ivan D. Green
Site Operations Manager

IDG:TFS:alf
encl.
cc: See attached sheet.

ATTACHMENT 1

SEQUENCE OF EVENTS - SMUD 04:25 to 05:34 - MARCH 20, 1978

(Revision 1, 5/25/78)

EVENT

:35 - Lost NNI power supply cabinets 5, 6, & 7

- This caused a loss of valid signals to the ICS. BTU limits ran back feedwater, resulting in a partial loss of feedwater (actual Rx power was 72%).

- Probable opening of "B" turbine bypass valves to the condenser (timing uncertain).

5:44 - Reactor trip on high pressure, turbine trip on interlock.

- Pressurizer code relief setting was known to be low (approximately 2225 psig). The electromatic relief was isolated due to previous leakage problems. The data indicates primary pressure went ≈2400 psig => code relief valve lifted.

- ICS closes main control and start-up feed valves and drive main feed pumps to minimum speed following trip.

- Decay heat and RC pumps energy removal accomplished through generators by inventory boil off and the addition of main feedwater.

26:15 - Pressurizer code relief valve reseats at approximately 2100 psig.

- Operator starts HPI pump "B".

28 3 - Operator stops HPI pump "B".

30 - OTSG "B" pressure reaches 435 psig set-point of Steam Line Failure Logic.

- OTSG "B" goes dry.

- Operator increases speed of a MFP and feeds "A" OTSG. This starts RCS on pressure and temperature decrease.

-

4: - RC pressure ≈1900 psi

7:.. - SFAS actuation at 1600 psig

This starts HPI, LPI and initiates emergency feed. The emergency FW pump is started and the bypass emergency FW valves are opened to full open position. The system makes no automatic attempt to control steam generator water level.

0 - RC pressure at 1475 psig. It starts to recover from this point due to HPI. T_{ave} = 528°F.

3:56 - "A" HPI pump secured.

6:09 - LPI secured.

9:54 - "A" HPI initiated. From this point on, the operator started and stopped HPI pumps as necessary to maintain pressurizer level.

50 - Steam Line Failure Logic closes ICS-controlled start-up feed valves to each OTSG when the corresponding OTSG pressure falls below 435 psig.

51:25 - Secured RCP-D (T_{ave} ≈435°F)

This reduced #RCP's to three

57:27 - OTSG "A" water level - 599.7"

Speculate that ≈2 ft. of tubes are not flooded (at top) due to steam line arrangement.

30 - Hourly computer log print-out
Steam temp. 380°F (OTSG "B")
Steam pressure 171 psig (OTSG "B")

Assuming $T_{ave} = T_{sat} \Rightarrow T_{ave} = 380°F$

3:47 - OTSG "B" level - 599.1"

34 - Power restored to NNI cabinets 5,6,&7

$T_{ave} = 285^{\circ}F$

RCS Pressure =2000 psig

Both OTSG full level ranges pegged high

Operator begins to reduce RC pressure using pressurizer spray.

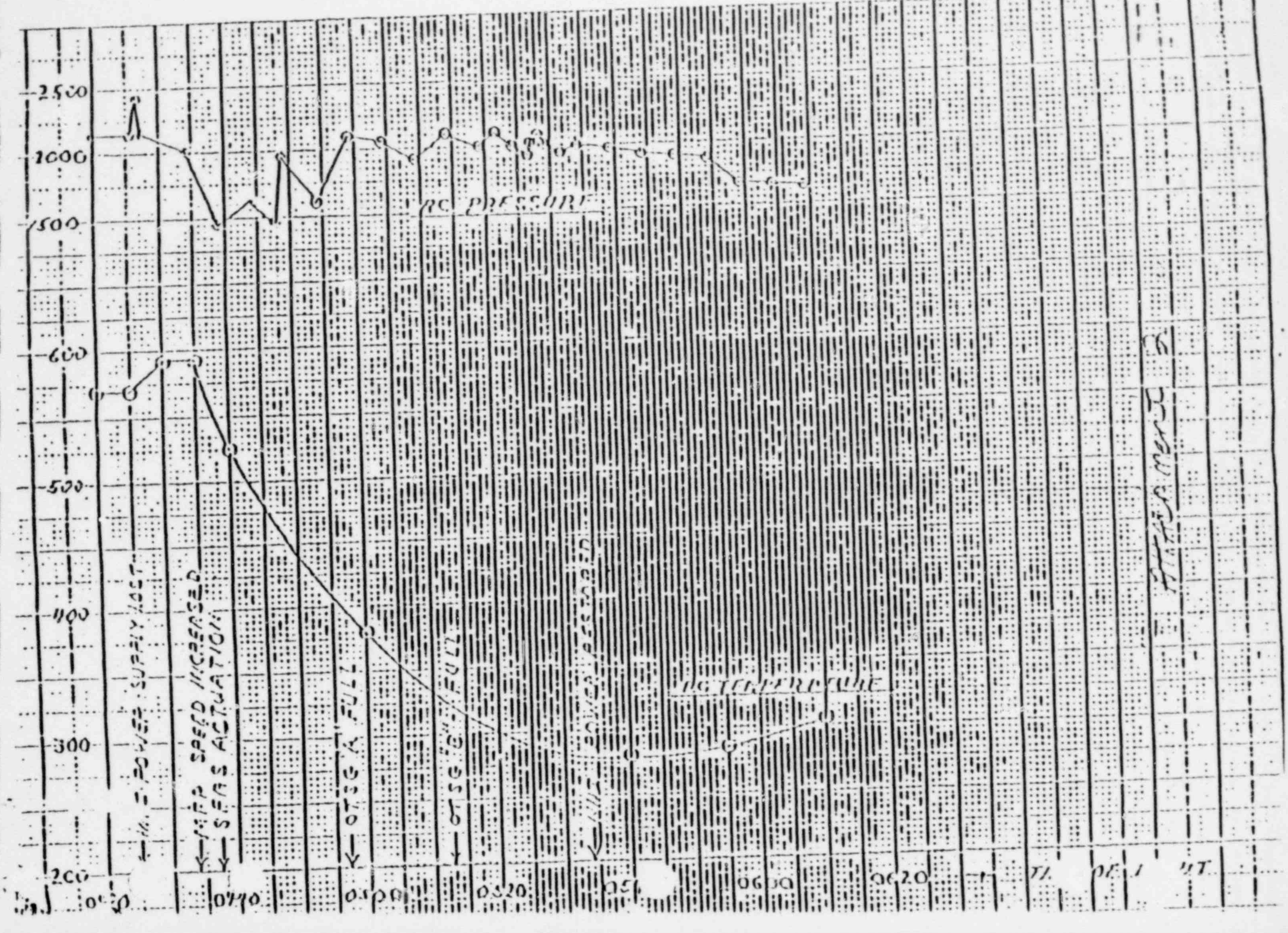
ICS closes turbine bypass valves to condenser.

Operator stops emergency FW flow.

Operator stops main FW pumps.

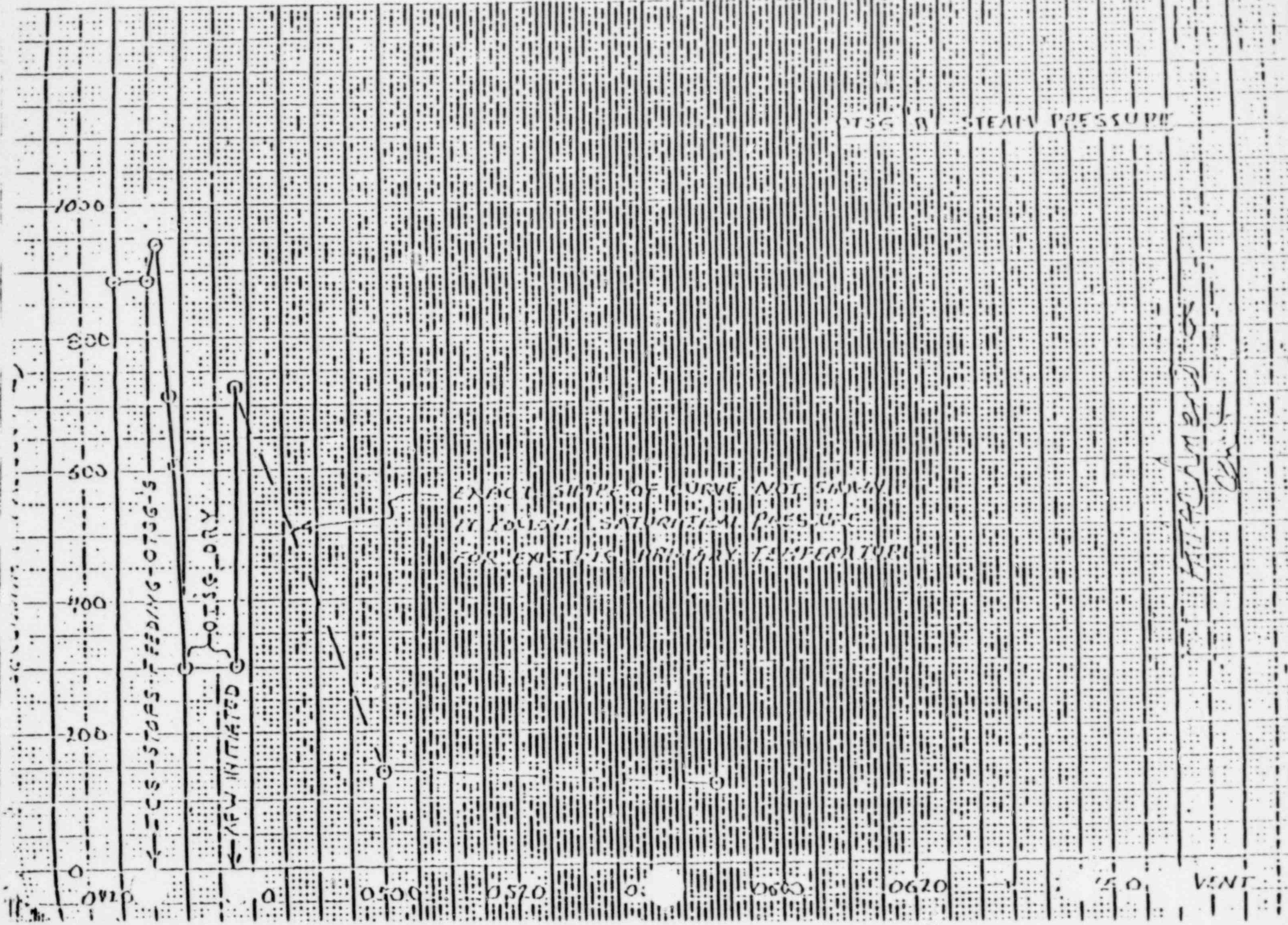
K-E 10 X 10 TO ½ INCH KEUFFEL & ESSER CO.

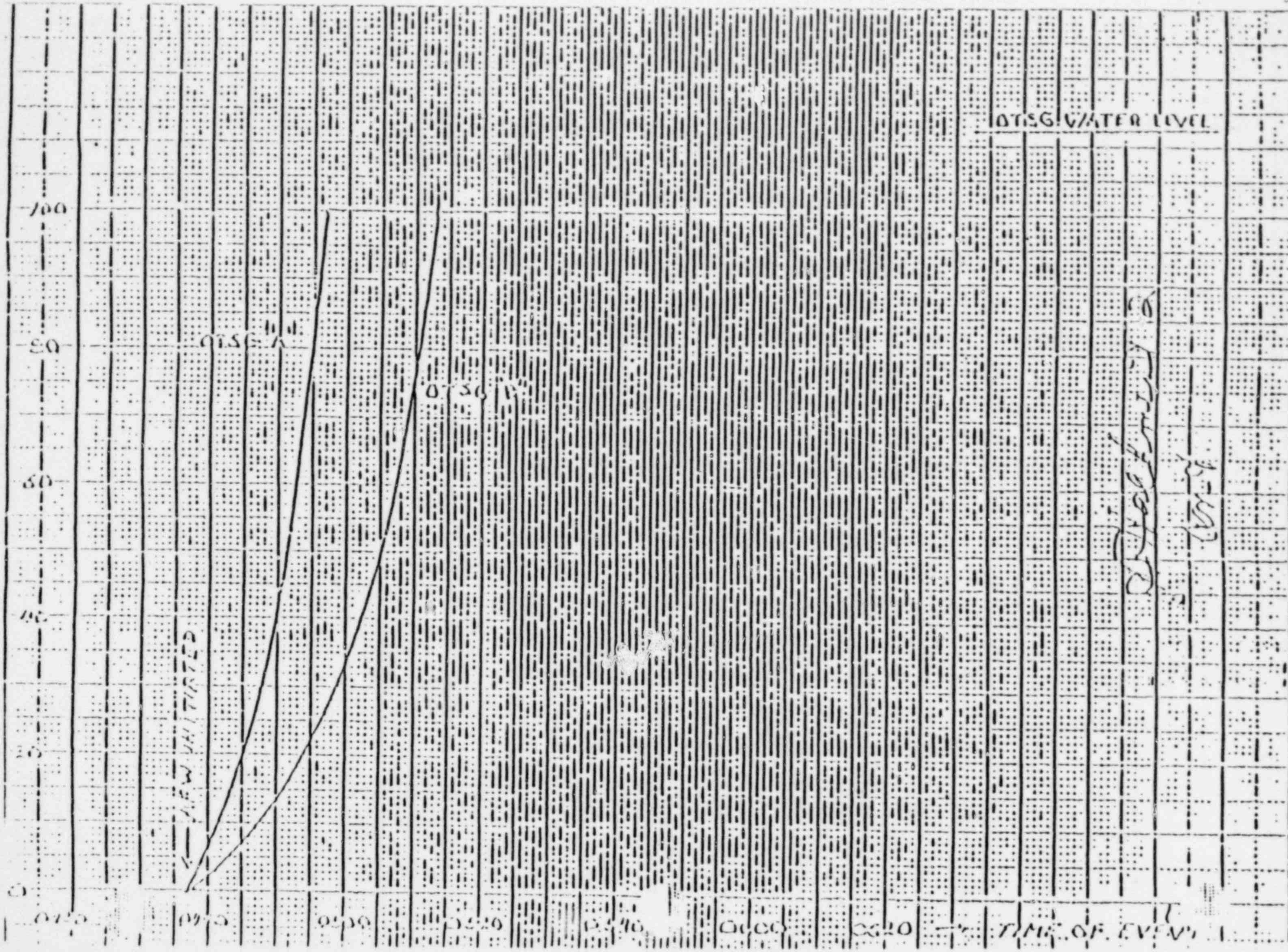
46 1470



K&E KEUFFEL & ESSER CO.

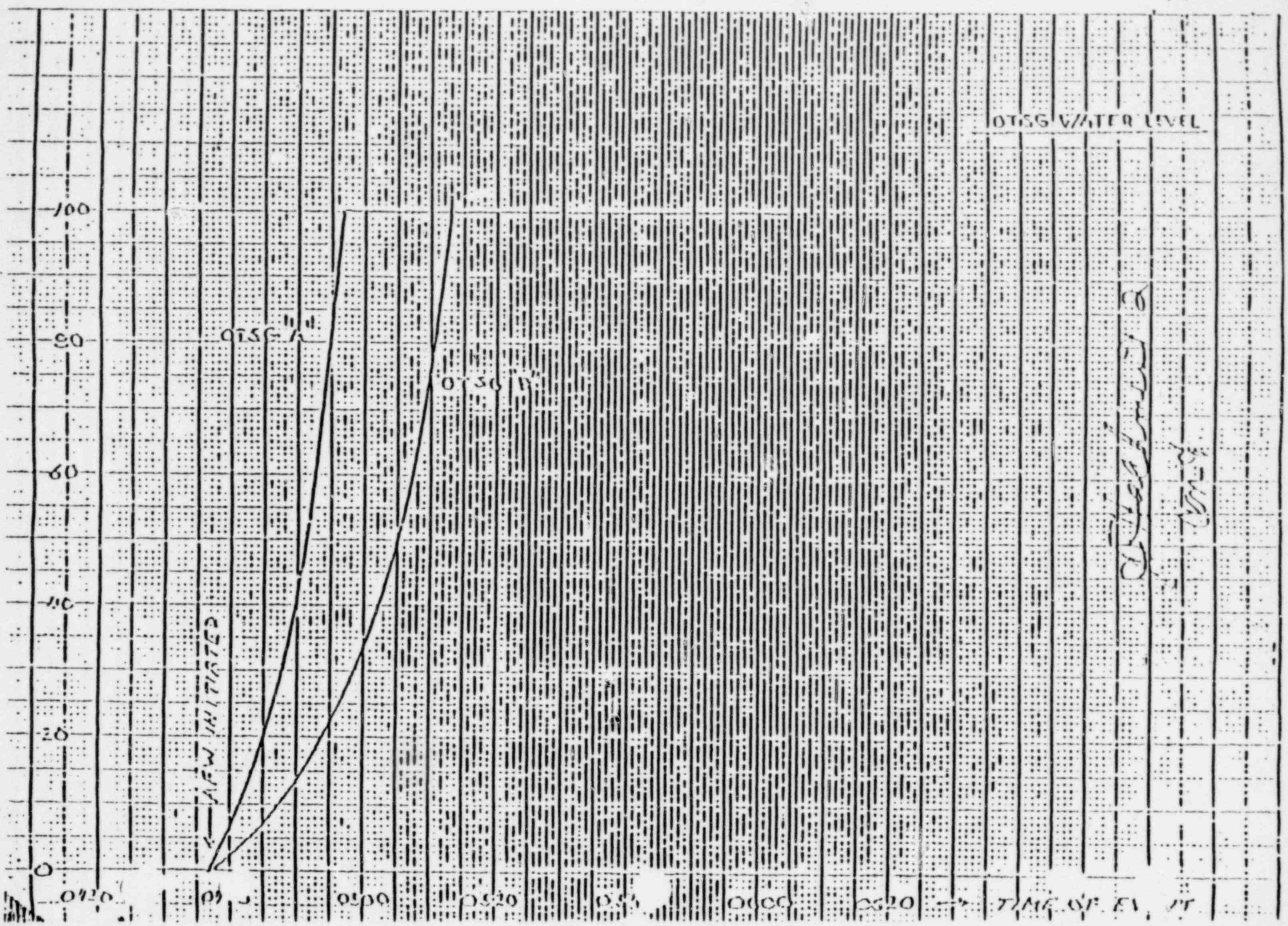
46 1470





OTSG WATER LEVEL
100
80
60
40
20
0
OTSG A
OTSG B
AFW INITIATED
TIME OF EVENT

K-E 10 X 10 TO ½ INCH 7 X 10 INCHES KEUFFEL & ESSER CO. 46 1470



EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

1. During a recent inspection at Davis-Besse Unit 1 information has been attained which indicates that at certain conditions of reactor coolant viscosity (as a function of temperature) core lifting may occur. The licensee informed the inspector that this issue involves other B&W facilities. The Davis-Besse FSAR states in Section 4.4.2.7:

> The hydraulic force on the fuel assembly receiving the most flow is shown as a function of system flow in Figure 4-39. Additional forces acting on the fuel assembly are the assembly weight and a hold down spring force, which resulted in a net downward force at all times during normal station operation.

The licensee states that there is a 500°F interlock for the starting of the fourth reactor coolant pump. However, no Technical Specification requires that the pump be started at or above this temperature. A concern regarding this matter would be if assemblies moved upward into a position such that control rod movement would be hindered.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

2. Inspection Report 50-346/78-06, paragraph 4, reported reactivity - power oscillations in the Davis-Besse core. These oscillations have also occurred at Oconee and are attributed to steam generator level oscillations. B&W report BAW-10027 states in A9.2:

> The OTSG Laboratory model test results indicated that periodic oscillations in steam pressure, steam flow, and steam generator primary outlet temperatures could occur under certain conditions.
>
> It was shown that the oscillations were of the type associated with the relationships between feedwater heating chamber pressure drop and tube nest pressure drop, which are eliminated or reduced to levels of no consequence (<u>no feedback to reactor system</u>) by adjustment of the tube nest inlet resistance. As a result of the tests, an adjustable orifice has been installed in the downcomer section of the steam generators to provide for adjustment of the tube nest inlet resistance and to provide the means for elimination of oscillations if they should develop during the operating lifetime of the generators. The initial orifice setting is chosen conservatively to minimize the need for further adjustment during the startup test program.

We also note that the effect on the incore detector system for monitoring core parameters during the oscillations is not clear.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

3. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% power with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Enclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

4. A memo from B&W regarding control rod drive system trip breaker maintenance is attached as Enclosure 2. This memo should be evaluated in terms of shutdown margin maintenance and ATWS considerations particularly in light of large positive moderator coefficients allowable with B&W facilities.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

5. Inspection and Enforcement Report 50-346/78-17, paragraph 6 refers to inspection findings regarding the capability of the incore detector system to determine worst case thermal conditions. The reactor can be operated per the Technical Specifications with the center incore string out of service. If the peak power locations is in the center of the core (this has been the case at Davis-Besse), factors are not applied to conservatively monitor values such as F_Q and F delta H.

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER

6. Enclosure 3 describes an event that occurred at a B&W facility which resulted in a severe thermal transient and extreme difficulty in controlling the plant. The aforementioned facilities should be reviewed in light of this information for possible safety implications.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

June 9, 1978

Docket No. 50-346

MEMORANDUM FOR: R. W. Woodruff, Acting Assistant Director for Technical Programs, Division of Reactor Operations Inspection, IE

THRU: G. Fiorelli, Chief, Reactor Operations and Nuclear Support Branch

FROM: J. F. Streeter, Chief, Nuclear Support Section #1

SUBJECT: DAVIS-BESSE UNIT 1 COMPLIANCE WITH THE REQUIREMENTS OF GENERAL DESIGN CRITERION 17 (AITS F30385H2)

General design criterion 17 requires, in part, that one of the offsite power circuits supplying a nuclear power station "...be designed to be available within a few seconds following a loss-of-coolant accident..." RIII understands from recent conversations with NRR that this GDC 17 requirement is interpreted as requiring an automatic transfer of station auxiliary power from the main generator to an offsite source in the event of a LOCA since manual switching could not reasonably be expected to be accomplished within a few seconds. The purpose of this memorandum is (1) to point out that Davis-Besse Unit 1 may not comply with this GDC 17 requirement during a loss of load condition, and (2) to recommend that this design matter be forwarded to NRR for review.

The Davis-Besse FSAR contains the following statements related to the automatic fast transfer of station auxiliary power from the main generator to offsite sources following loss of normal (main generator) power:

(a) Appendix 3D, page 3D-15, last paragraph "...In the event the main generator unit is lost, station auxiliaries will be transferred automatically by fast bus transfer schemes to the offsite power..."

(b) Section 8.3.1.1, page 8-6, second paragraph "...The system will have a fast transfer to the reserve power source following a turbine generator or reactor trip, without loss of auxiliary load."

These FSAR statements appear to indicate that the fast bus transfer action is unconditional and, therefore, will occur anytime main generator power is lost.

In addition, the following FSAR statement indicates that the emergency diesel generators will not be called upon to supply the auxiliary power unless both the main generator and offsite power are lost:

> Appendix 3D, page 3D-15, second paragraph "...Upon loss of the normal and reserve (offsite) power sources, the two 4160 volt essential buses are energized from their respective emergency diesel-generators.

In the event the plant experiences a loss of load (anticipated operational occurrence as described in FSAR Section 15.2.7) due to the main generator 345 KV output breakers opening, the plant is designed to run back to 15% power and maintain station auxiliary loads on the main generator. RIII has determined that if a LOCA occurs when the plant is in this condition the emergency diesel generators would be immediately called upon to supply auxiliary power without the fast transfer circuit attempting to obtain auxiliary power from the preferred (offsite) source. This is due to the design of the fast transfer circuit which requires the 345 KV breakers to have been closed immediately prior to the loss of the main generator. This is the area where the plant may not comply with the GDC requirement.

The licensee's position and basis of fast transfer design is that any event other than generator faults which would cause opening of the generator output breakers would also result in loss of all offsite power sources; therefore, having automatic transfer to this unavailable source would not make any sense. Under generator fault conditions the output breakers open and auxiliary loads are immediately transferred to offsite power. The licensee believes he complies with all requirements of GDC 17.

RIII understands from recent discussions with NRR that Davis-Besse was judged to conform to GDC 17 before OL issuance based on the above mentioned FSAR statements which appear to indicate that (1) the fast transfer scheme is unconditional, and (2) the emergency diesel generators are not called upon to supply auxiliary power unless all other power sources have been lost. Since this is not the case, NRR may not find the design acceptable. RIII recommends this matter be forwarded to NRR for review.

J. F. Streeter, Chief
Nuclear Support Section #1

cc: IE Files
Central Files
T. N. Tambling, RIII

CONTACT: J. Smith
387-9350

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 15, 1979

TO: ~~Howard Shapar~~

FROM: Richard S. Mallory, OGC RSM

Enclosed is a copy of the transcript of your deposition before the President's Commission on the Accident at Three Mile Island.

Please read through the transcript carefully and correct any errors (other than unimportant punctuation errors) in black pen on this copy. Correct any errors you can identify in the questions, as well as in your answers. This copy will not be retyped, but will be reproduced as you have marked it, so your corrections should be dark and legible. If you cross out words in the transcript, draw only a single line through them, so that they can still be easily read when the transcript is copied; do not obliterate them.

After you have corrected the transcript, please sign and date the certificate at the end, and type your name under your signature.

You may wish to make a copy of the transcript for yourself before returning the original to me. When you return the transcript, please indicate if you object to making your transcript available to the Commission or to the Commission's investigation of Three Mile Island. Because of Commissioner interest, we would appreciate receiving your corrected copy by c.o.b. Thursday, August 16, if possible.

The President's Commission on Three Mile Island will also be sending you a copy of your transcript with a request to make an "errata sheet" and sign a signature page. Please make up an "errata sheet" based on the copy of the transcript that you have retained and return the errata sheet and signature page to the President's Commission as requested in their letter. Please send me a copy of the errata sheet and signature page also.

If you have any questions or problems, do not hesitate to call me or the attorney who represented you at the deposition.

Enclosure: Transcript

I have no objection to making the attached transcript available to the Commission or to the Commission's investigation of TMI.

Howard K. Shapar
8/16/79

COPY

Transcript of Proceedings

UNITED STATES OF AMERICA

PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND

DEPOSITION OF: HOWARD K. SHAPER

Bethesda, Maryland

August 3, 1979

Acme Reporting Company

Official Reporters

1411 K Street, N.W.

Washington, D. C. 20005

(202) 623-4869

8001280514

CERTIFICATE

I certify that I have read this transcript * and corrected any errors in the transcription that I have been able to identify, except for unimportant punctuation errors.

Date: 8/16/79 [signature]

* Page 53 is missing.
HKS
Exhibit #2 is illegible.
HKS