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Transcript of Proceedings

UNITED STATES OF AMERICA

PRESIDENT'S COMMISSION ON THE ACCIDENT AT
THREE MILE ISLAND

DEPOSITION OF: LEON B. ENGLE

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Bethesda, Maryland

August 3, 1979

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PRESIDENT'S COMMISSION ON THE ACCIDENT AT THREE MILE ISLAND

DEPOSITION OF: LEON B. ENGLE

Room 6072
7735 Old Georgetown Road
Bethesda, Maryland

August 3, 1979
1:30 o'clock p.m.

APPEARANCES:

On Behalf of the Commission:

GARY M. SIDELL, ESQ.
Associate Chief Counsel
2100 M Street, N.W.
Washington, D.C. 20037

On Behalf of NRC:

SHELDON TRUBATCH, ESQ.
Office of the General Counsel
1717 H Street, N.W.
Washington, D.C.

I N D E X

P R O C E E D I N G S

Whereupon,

LEON B. ENGLE

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

EXAMINATION

BY MR. SIDELL:

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

A. Leon B. Engle.

Q. And your current position in the NRC.

A. At the present time I have been temporarily assigned Bulletin and Orders Task Group, TMI, and I am presently on loan to Mr. Volmer, Division of Operating Reactors, in regards to the TMI-I restart order.

Q. You have provided me with a copy of your resume which is four pages. Is this correct and a complete copy of your professional and educational background?

A. Yes.

Q. I notice on Page 4 on of your publications entitled "Reactor Power Excursion Studies," authored by you and a Donald Peterson as well as W. R. Stratton. Is that Bill Stratton formerly of the ACRS?

A. It is. Yes.

Q. You appear to have authored a number of articles with Mr. Stratton.

A That is correct. When I was employed in Los Alamos I worked for Dr. Stratton.

MR. SIDELL: Let's have this marked as Exhibit No.1 to the Engle deposition.

(The document referred to was marked for identification as Deposition Exhibit No. 1.)

MR. SIDELL: Mr. Engle, have you ever had your deposition taken before?

THE WITNESS: Never.

MR. SIDELL: Let me tell you briefly then what we are going to be doing. Your testimony is sworn, of course, and as such, even though we are in the Chairman's office in the NRC Building in Bethesda, Maryland, it has the same effect as though it were given in a court of law before either a judge or jury. Therefore, it is necessary that you be as precise and accurate in your responses to my questions as you can be.

If there is an occasion where you don't understand a question that I ask, please ask me for clarification, and I will try to explain to you what I am looking for in the way of a response.

Since the court reporter is taking this testimony down, it would be helpful if you would not answer questions with either a nod of the head or other gestures but rather audibly so that you responses may be accurately recorded.

Furthermore, I would ask that you wait until I finish a question before you begin your response, even though you may know where the question might be headed. I will also try and restrain myself from asking my next question before you finish your answers, since it is obviously much easier for the reporter to record one of us speaking at a time.

At the conclusion of the deposition your testimony will be transcribed and presented to you to review, to correct if you find any necessity for correction and to sign. If you do find corrections that are required in your opinion, you may make them, of course. Should we deem those changes to be of a substantial nature, we will be entitled to comment on those changes, and those comments may adversely affect your credibility. Therefore, the necessity again arises to be as precise and accurate in your responses now as you can be.

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

THE WITNESS: No.

BY MR. SIDELL:

Q. In 1978, approximately the middle of the year, what position did you hold with the NRC?

A. In the middle of 1978 I was Project Manager in the

~~White~~ Light Water Reactors Branch No. 1, Division of Project Management.

Q. And in that position were you responsible in one aspect or another for the operation of Davis-Besse Unit 1 in Ohio?

A. Yes, I was at that time.

Q. Can you tell me what your responsibilities included relative to Davis-Besse Unit 1 at that time?

A. At that time the primary concerns were providing amendments to either licensing conditions or technical specification requests that the licensee may have provided. At that time there was also, I believe, in April of '78 the problem coming up on burnable poison rod assemblies which at first happened at Crystal River. And being a generic problem the Toledo Edison Company, the licensee for Davis-Besse 1 was beginning to indicate that they would be taking the latch assemblies out on these burnable poison rod assemblies, and that that would require a major amendment. I was engaged in setting up the staff review for that item.

At that time I also had responsibility for the BSAR 205 Standard Plant. And at that time we were into what you would call Q-2's, which were positions going out relative to what the staff's review was at that time.

Q. Do you recall having conversations in late or middle 1978 with an individual by the name of James Crestwell?

A. Yes. I believe that relates to burnable poison rod assemblies. I think I first met Mr. Crestwell at a meeting held at the Phillips Building.

Q. That is the Phillips Building in Bethesda?

A. Bethesda. That is my first rememberance of Mr. Crestwell.

Q. Do you recall the day of that meeting?

A. No; I do not.

Q. Did there come a time after your initial meeting with Mr. Crestwell when you discussed matters relating -- other matters relating to Davis-Besse in the fall of 1978?

A. No. I do not remember directly talking to Mr. Crestwell. Most of my conversations with I&E Region III, which incidently was where Mr. Crestwell was located, was with the inspector on Davis-Besse 1, which was Tom Tamblin and also with Tom's chief which was Dick Knopp.

Q. Are you aware of how Mr. Crestwell fits into the organizational structure with Mr. Knopp and Mr. Tamblin?

A. It was my understanding that he had been assigned to that particular group inspection and that he also was making inspections out there at Davis-Besse as part of that group.

Q. And Mr. Crestwell's area of concern was more limited or finite then Mr. Tamblin's area of responsibility?

A. Well, I had overall responsibility from the standpoint of licensing on Davis-Besse 1, and my -- Tom had been on Davis-Besse for a long, long time, and my main interactions were with either Tom or his immediate superior, Dick Knopp.

Q. Do you recall speaking with Mr. Crestwell about the problem of the Davis-Besse operator blocking the safety actuation system before and turning off the HPI system, that occurred on December 29, 1977 at Davis-Besse? Strike that.

-- event occurring on September 24, 1977 at Davis-Besse?

A. I do not remember directly talking to Mr. Crestwell about that event.

Q. Did you speak with Mr. Tamblin about that event?

A. Oh, many, many times.

Q. What was the substance of your conversation with him?

A. That goes back to the very initial beginning of the event.

Q. The September 27, 1977 transient?

A. September 24, 1977.

Q. I mispoke. You are correct.

That transient, not the November, 1977 transient?

A. The event I am thinking of is the September 24, 1977 transient at Davis-Besse.

Q. And what was the substance of your conversations with Mr. Tamblin concerning that transient at Davis-Besse?

A. They covered many days after that event in determining and assessing all the matters that related to that event as far as equipment that would need to be prepared or the status of the plant. And I think the first time that I -- Mr. Tamblin was sent out as part of a team right after that event. I believe he was sent out on the 25th. The event occurred Saturday night, and as I remember he was sent out Sunday as part of a Team.

And then I had various conversations with him later on, the status of the plant relating to the various systems that had been involved in that incident.

Q. Did you discuss any problems with loss of pressurizer level indications?

A. Only generally as the event went. We discussed a lot of items. That is one I did not discuss in detail. Only as it related to the overall scenario of events that has happened.

Q. What did you discuss concerning pressurizer level indications?

A. I don't remember any item with specificity, just in relation to as events occurred, as the pressurizer level came up, as the operator may have seen it coming up and shut off the high pressure safety injection, actuation. Like the initial occurrence when the pressurizer level went up and the operator saw that level going up, and manually scrammed the reactor.

Q Had you before the September 24, 1977 Davis-Besse transient been aware of any other instances of B&W reactors where there was a loss of pressurizer level indication off scale high?

A No.

Q Would you consider that this particular instance of loss of pressurizer indication level high was rather exceptional?

A No; I did not at that time.

Q Why was that?

A I was interested in the overall performance of the plant, the total transient, and all matters related, like the auxiliary feed water system, the secondary system, the primary system, damage which had occurred to the event, such as stripping off insulation on the No. 2 steam generator, spurious trip in the steam feed water and rupture control system, the cavitation of the reactor coolant pumps.

I was concerned that the transient in regards to the design specifications of the equipment had not been exceeded by the transient. I was concerned in all of these events total picture as relating to getting that plant, which I knew they would sooner or later want to restart, and I wanted to assure myself that all of these things had been taken care of.

Q Prior to September of 1977 were you aware of what reliance an operator placed on pressurizer level indication?

A I knew that it was one of the parameters which he observed.

Q Would you say that you knew it was one of the more significant or substantial elements that he observed?

A Yes.

Q Would you also say that at time you knew the operator relied almost exclusively on pressurizer level indication to inform as to the status of core inventory?

A No.

Q Have you since that accident learned that to be case for operators of B&W reactors?

A Based on history from that date it has become very apparent that the pressurizer level was an extremely important indication to the operator.

Q Would you conclude that it is an equally important indicator in terms of accuracy of the information provided the operator?

A No. I think the pressurizer level may not be as important as other parameters.

Q Well, Mr. Engle, I believe my question was directed more to the accuracy of the pressurizer level indication as to its importance to the operator in running the system. If my question was unclear -- I am looking for the degree of accuracy you now believe the pressurizer level indication to have in B&W reactors.

A. It should be as accurate as can be feasibly made.

Q. It should be, but is it?

A. At the present time I would say based on past events, no.

Q. And which past events are you specifically referring to?

A. Three Mile Island.

Q. Any other situations?

A. No.

Q. Are you aware of loss of pressurizer level indication problems occurring at Rancho Seco?

A. No.

Q. At TMI-II in March, 1978?

A. ~~No.~~ Yes

Q. At Arkansas-I?

A. Yes. I recently have been project manager for Arkansas Nuclear-I unit-2. And in discussion with the licensee after Three Mile Island-II and discussing some of the simiarities with the Davis-Besse event in the first six minutes as compared to TMI-II, the licensee mentioned to me that they had had a problem in this regard.

Q. Did they tell you when the accident occurred?

A. I don't remember if they did or did not in that discussion.

Q. Can you recall if it were before the accident at Three Mile Island?

A. It was after the accident that I talked to them.

Q. Do you know whether or not they filed a Licensee Event Report on that problem?

A. No; I do not.

Q. Would you be in a position to be apprised of that fact if they had done so?

A. No; I would not, because I was not project manager for Unit-1. I was project manager for Unit-2. Unit-1 is a B&W plant; Unit-2 is a Combustion Engineering plant.

Q. Do you know if there is any cross-licensing between the two plants for operators?

A. In some areas there are probably areas where the operators are trained for both plant operations. But I could not state just what those areas are.

Q. Are you fairly certain that some operators are licensed to operate both units at Arkansas?

A. I would guess that they are.

Q. Why would you make that conclusion?

A. Because it is very possible that some of their senior operators may have had the necessary experience to qualify them for both B&W plants and Combustion Engineering plants.

Q. Are you familiar with the control rooms of both Arkansas units?

A. No; I am not.

Q. Do you know whether or not both control rooms are the same in terms of setup?

A. No; I am not.

Q. Are you familiar with the distinctions between the Combustion Engineering and B&W facilities?

A. As regards the nuclear steam supply system, I have a general knowledge of the differences in the two systems.

Q. Does the Combustion Engineering system have a once-through steam generator?

A. No; it does not.

Q. Does the B&W system have that feature?

A. Yes; it does.

Q. And as a result of the once-through steam generator, do you know the length of time it takes the B&W reactor to boil dry?

A. To the best of my recollection it is about 30 seconds.

Q. Do you know the length of time that it takes for a CE-Reactor to boil dry its steam generator?

A. Based on the best evidence to date I think it is about 20 minutes.

Q. And those are recirculating steam generators in a CE system?

A. Yes.

Q. Do you know of any other distinctions between the two steam plants, CE and B&W? For instance, does a B&W Reactor have a PORV?

A. Would you restate the question?

Q. Yes. Does a B&W Reactor have a PORV?

A. Yes; it does.

Q. Does a C&E Reactor have a PORV?

A. In most cases that it true. However, on Arkansas Nuclear-1 Unit-2 there is no PORV.

Q. Is there merely a code safety valve?

A. That is correct.

Q. Do you know how many there are?

A. I believe there are two.

Q. Do you know how many there are on the B&W Reactor?

A. Two.

Q. Do you know of any other distinctions between a B&W Reactor and the CE-Reactor at Arkansas?

A. One difference is in the B&W plants they have the vent valves in the core. Comb ustion Engineering plants do not have that valve.

Q. At all.

A. At all. The reactor coolant pumps and their actual elevations -- I can't remember the exact distances though --

elevation positions for reactor coolant pumps, the hot and cold leg, where the surge line for the pressurizer connects to the hot leg, I would imagine -- if I could look at the FSAR -- I would find a lot of little distinctions where there are differences.

Q. Does a specific CE-Reactor at Arkansas have a pressurizer level indication to indicate core inventory?

A. Would you repeat that question?

Q. Does the CE-Reactor at Arkansas have a pressurizer level indication to indicate core inventory?

MR. TRUBATCH: If we could break the question up -- I think it is the existence of the indicator first --

BY MR. SIDELL:

Q. Does the CE-Reactor at the Arkansas Plant have a pressurizer level indicator?

A. Yes.

Q. What is the purpose of it?

A. The primary purpose of the pressurizer level is to indicate the -- is to give an indication of reactor coolant volume.

Q. In the core.

A. In the core.

Q. That is the same function as the pressurizer level indicator places in the B&W reactor; is it not?

A. That is right.

Q. Is there any difference between the two reactors at Arkansas whether or not the operator relies on one or the other in different manners or they are approximately the same?

A. I am not that familiar with those operating procedures to answer that question.

Q. Are there alternative methods that the operator at Arkansas-1 and the CE-Reactor can determine what's going on in the core besides pressurizer level indication? For instance, temperature, pressure in the primary system.

A. Now, I am quoting from memory from the FSAR, yes, they have primary coolant pressure. They had Thot, Tcold for the primary system, and they also can integrate that to Taverage.

Q. Do you know what the normal operating temperature of the tail pipe on the CE-Reactor at Arkansas is?

(Pause.)

The average or normal operating temperature on the tail pipe at the CE-Reactor at Arkansas.

A. Tail pipe encompasses a lot of items. I need further specificity on what you mean by a tail pipe.

Q. The reactor temperature during normal operations.

A. The use of the nomenclature "tail pipe" I can only associate with the pipe downstream of the PORV. I do not relate it to any other item.

Q. That is what I am looking for, the temperature at that location during normal operating conditions.

A. I just do not remember that maximum temperature that would be on there.

Q. How about for the B&W Reactor at Arkansas?

A. I do not know at Arkansas.

Q. Would a range of 180 to 210 or 220 degrees be about right for normal operating temperatures?

A. That certainly would put it in a range compared to the opening of a valve under pressure, and I would think it would be more towards the 220 degrees than the 180.

Q. Would an operator at the CE or B&W Reactor at Arkansas be able to conclude whether or not the PORV was opened if he read a temperature at the tail pipe of approximately 260 degrees?

A. 260?

Q. Yes.

A. At 260 degrees he would either know that that valve was open or had been opened.

Q. Based on your prior response at a range of 180 to 220 degrees at the tail pipe would be normal operating temperature for either the B&W or the CE-Reactor at Arkansas. I gather you would conclude that a temperature of 604 degrees would be excessively abnormal. I would that be a fair conclusion?

MR. TRUBATCH: May we go off the record.

MR. SIDELL: Off the record.

(Off the record.)

MR. SIDELL: Back on the record.

THE WITNESS: Yes; that would be abnormal.

BY MR. SIDELL:

Q Would a temperature of approximately 604 degrees Fahrenheit at the tail pipe on either of the Arkansas reactors indicate that the PORV had failed opened?

A Yes; it would.

Q Rather than merely open as a temporary release of steam and then close.

A Yes. To get it up to 604 that would have meant that you had steam or steam water passing through there.

Q Which under normal operating conditions would not have?

A That is correct. That would put you back down around within the 180 to 220 regime.

Q What resulted from the inspection by several people whom I believe you indicated went to investigate the September 1977 Davis-Besse transient? Was there a report that was generated?

A Yes. There were several reports generated. It is my recollection that the I&E Report that came out on investigating that event was somewhere in November, 1977. I believe it was around November 22, 1977.

Q. Were they before the November transient at Davis-Besse?

A. Yes.

Q. Can you recall the conclusions of that report?

A. The conclusions of that report went into the investigations in the early days right after the event and ensuing inspections they made to insure that the equipment such as the reactor coolant pumps, spurious half-trip and the steam feed water rupture control system, the turbine governor on the No. 2 steam generator, and design basis specifications for equipment had not been exceeded in the transient.

Q. Can you recall whether or not there was as a result of the September transient the installation of a position actuation switch or indicator for the PORV?

A. I remember in the I&E Report -- I believe it was either the I&E Report or the licensee supplement to the LER where they stated that they were going to put position indicating in there so that they could have specificity on the determination of the POVR. But that really was related to the magnetic solenoid valve which would give an indication whether the stem on the release valve was open or closed.

Q. Would that merely mean that there would be an indication where the operator conclude that an electrical signal was sent to close the valve but not that in fact the valve was actually closed?

A. I believe that was the quoted design that would be put in there, that the signal was reading, that the solenoid valve was in the correct position.

Q. But not merely that the valve itself was where it should be?

A. I believe that's correct.

Q. Can you recall the manufacturer's design of the PORV at Davis-Besse?

A. It will come to me sooner or later. I remember that name. I cannot think of it at this moment.

Q. Would it be a Crosby valve?

A. That's it.

Q. Is that distinct from the type of manufacturer PORV's at other B&W plants?

A. I believe Davis-Besse was the only one that had that type of valve.

Q. Do you know why?

A. No; I do not.

Q. Are you familiar with the mechanical distinctions between the two types of PORV's?

A. Not at this point. I would have to look at the design.

Q. Do you know whether or not it is easier to put on indicators showing an actual position of a Crosby PORV than it is for a Dresser PORV?

A. I couldn't begin to answer that without looking at the design diagrams.

Q. Let me show you a packet of information which is addressed to Mr. James G. Keppler, K-E-P-P-L-E-R, from Toledo Edison -- dated of document 11-14-77 and date received is 11-25-77, which includes a November 14, 1977 letter to Mr. Keppler from Terry D. Murray, Station Superintendent, Davis-Besse Nuclear Power Station, entitled "Supplement to Recordable Occurrence NP-32-77-16 Concerning the September 24, 1977 occurrence," which is some 59 pages in length, and ask you if you have ever seen this document before. (Handing document.)

A. (Perusing document.) I have seen that document before.

Q. This document just referred to also appears to have your name written across the front page; does it not?

A. Yes, sir; it does.

Q. Is that your handwriting?

A. (Perusing document.) No; that is not my handwriting.

Q. Do you recall whether or not this copy of the report was distributed to you?

A. A copy went to me because I was the project manager on the case at that time, and therefore, I would receive a copy.

Q. Could this particular document that I have just shown you be your copy or a copy of the one that was sent to you?

A. No. I still have my copy.

Q. Could this then be a Xerox of the one you still have?

A. It could well be.

Q. And, therefore, the reason for the appearance of your name on the cover page?

A. It could very well be. As I remember I made some Xerox copies so people could see some of the information.

Q. Do you recall whether or not you made a copy or provided the original of this report to anyone yesterday?

A. No, not yesterday.

MR. SIDELL: Let's have this marked as Exhibit No. 2 to this deposition.

(The document referred to was marked for identification as Deposition Exhibit No. 2.)

BY MR. SIDELL:

Q. Do you recollect any conversations with Mr. James Crestwell of Region III in late fall or early winter of 1978 dealing with the inspection of the Davis-Besse September, 1977 transient?

A. I do not remember ever having talked to Mr. Crestwell other than that meeting that I spoke to earlier -- Well, I only remember speaking to Mr. Crestwell directly at what I believe was the April, 1978 meeting where we were discussing the burnable poison rod assembly.

MR. SIDELL: Off the record.

(Off the record.)

MR. SIDELL: Back on the record.

BY MR. SIDELL:

Q Mr. Engle, did you participate in the on-site inspection of Davis-Besse in September, 1977?

A I was out on a trip to Davis-Besse on September 30, 1977. The occurrence of that -- I was notified on the previous Wednesday by an Andy Szukiewicy ~~Sukabitch~~ (ph.) who was in the Instrumentation and Control Branch that DSS was planning to make a trip to Davis-Besse. And he asked me if I were going. I answered him, "I did not know about the trip."

I then called Jerry Mazetis of the Reactor Systems Branch and asked Jerry if he knew about this trip, and Jerry indicated he was going. I then notified my branch chief that this trip was going to transpire and I felt that I should go.

That happened about Wednesday noon, as I remember. I did not get any word until, I believe, it was Thursday noon that my management had decided that I would go on that trip. I remember that very clearly because the secretary had to work very fast to get my travel set up.

I did go to that meeting on September 30th.

Q. Can you remember whether there was Mr. Domeck ~~Domick~~ (ph.) representing Te-Co. at that meeting?

A. There was a Chuck Domeck ~~Domie~~ there. He had just, I think, recently made project manager for Davis-Besse-1.

Q. Can you remember whether Terry Harpster (ph.) from Region-3 was there?

A. Oh, yes. I remember Terry very well.

Q. And W. S. Little from Region-3?

A. I do not remember W. S. Little distinctly as being able to relate to him.

Q. How about E. C. Novak from Toledo Edison?

A. Yes, very well, because I had dealt with him during the licensing process on Davis-Besse.

Q. And Lowell Rowe (ph.)

A. I know Lowell Rowe very well because I had also dealt with Lowell on licensing matters.

Q. And Arthur McBridge from B&W?

A. I know there were B&W people there. I can't relate to McBride himself.

Q. Were there also people from Bechtel there?

A. There were also people from Bechtel there.

Q. In other words, a pretty substantial meeting.

A. I would say about -- oh, thinking of that room, 50 people there.

Q. And as one of the topics of conversation of this meeting at Davis-Besse on September 30, 1977, was there mention of the fact that the operator had turned off the HPI system shortly after it was initiated on the basis of his observation of increasing pressurizer indication level?

A. There have been so many scenarios on that, and I was involved in so many scenarios I cannot really state -- remember that it was mentioned there, but I believe it was because I personally tried to see that that meeting describe the overall scenario of the various input parameters related to it.

Q. And was there also mentioned at that meeting of the PORV failing open?

A. Absolutely.

Q. Do you remember the length of time into the transient that the operator recognized that the PORV was stuck open and as a result closed the block valve?

A. 20 minutes as scram -- manual scram.

Q. So, the operator scrammed the reactor, not that any automatic scram was instituted; is that correct?

A. That's right. That operator right after -- oh, approximately two minutes after the spurious signal in the steam feed water and rupture control system saw the pressurizer level rapidly building up, and he manually scrammed that reactor. And as I remember 20 minutes to closure of the block valve is based on T equals zero at the manual scram.

Q. "T" as time or temperature?

A. Time.

Q. Can you recall whether or not there was any report that was produced as a result of the September 30, 1977 meeting at Davis-Besse?

A. After that meeting on the 30th, where we got their reactimeter data, which was data that B&W -- which was a -- Reactimeter measures rapid intervals of data during the start-up of the B&W reactors and during ~~start-up~~ and testing. I received a lot of data. And on my way home Friday night I became very interested in this data and I spent the weekend, Saturday and Sunday, producing a poster-size graph which I felt would describe thé overall events. I worked both days on that.

This is the graph. I would be glad to get reproductions. I would like to keep it. It is a souvenir to me, but I would be glad to get copies of it. (Handing chart.)

I transcribed abscissas and coordinates from that reactimeter data and produced what I felt was significant curves relating to the incident.

On Monday morning I briefed my branch chief as to the meeting. And shortly thereafter Jerry Mazetis called me and told me that DSS was going to have a trip report meeting -- that was somewhere around 8:30.

Q. Do you recall the date of that meeting?

A. That would have been -- Let's see. We went out there Friday, the 30th, and October 1st and 2nd was Saturday and Sunday when I worked on that graph, and Monday would have been October 3rd, I believe.

Q. And that was the date of the meeting.

A. In the morning.

I told Jerry I had this graph, and it was large size, and it might help him in explaining to the DSS people the overall transient. He said, "Come on down." And I went there. He did use that to give the general overall picture. I got the feeling that it was a ~~the~~ DSS meeting and understood that Jerry was going to be producing the trip report.

Q. Is that a standard report after an accident such as this one?

T1S2

A. That depends on various matters. A trip report should be issued. During the course of that meeting, it was determined -- the meeting being described here -- that the Office of Enforcement and Inspection would maintain lead responsibility on investigating the accident. At that time I became very concerned that I wanted to follow through very, very closely on equipment that may have broken down, equipment that needed repairs, and felt that I had to interface very, very closely with I&E and to keep a close -- and also the licensee to keep a close status on how plant repairs were progressing, because after all I was the project manager, and I knew that sooner or later the licensee would want to bring the plant back up. I felt it was my responsibility to see that those repairs, based on I&E's inspection, had been completed prior to the time they brought the plant back up.

So, I waited to issue a trip report based until I felt that I&E and the inspectors who were out in the field actually at the plant, going through it in detail, had reached some definite conclusions.

Q. Did there come a time when the inspectors did make specific conclusions about the September 24, 1977 transient?

A. Yes. I think it was in that week that I am speaking of -- I think it was on October 6th, which would have been a Thursday -- I called -- it was either Tom Tamblin or Dick Knopp. I called them constantly. I was making a bother of myself.

But I called either Dick Knopp or Tom Tamblin and wanted to know based on their inspection what actions they were going to take prior to allowing the plant to come back to -- to operate. They indicated to me -- I believe by Friday, which would have been the 7th, that they were going to issue what is called an immediate action order specifying to the licensee what actions would be required.

Q. Do you know if they in fact did file that report?

A. I cannot remember actually seeing a copy of that immediate action letter, but I know that the licensee, in talking to them about things that were going on, that I&E had actually specified "There are things you are going to do. Like you got to make B&W analysis of the excursion in case D&BR had been exceeded, any fuel rupture, whether they were within their design basis accidents." For instance, "Check out that spurious signal on the steam feed water rupture control system. Check your reactor coolant pumps in case -- because of cavitation for reactor seals and impellers."

Of course, they had to get that insullation back on that steam -- No. 2 steam generator. There was another item that I became very concerned about, and that was the relay valve in the electromagnetic solenoid.

Q. Controlling the PORV.

A. Controlling the PORV.

It became apparent that that relay had been missing, and I became very, very concerned about that relay valve missing. I called both I&E and the licensee, the man you just mentioned, Mr. Lowell Rowe, and Eugene Novak. I was concerned that there had been a breakdown in procedures somewhere that "Why was that valve missing?"

Now, it must be understood that that particular item is non-safety grade. So, it doesn't go through our normal QA procedure. But the licensee had what was called a yellow check-off list which they make during pre-operational tests, which indicated that that valve had been working right.

So, at that time -- back in time that relay had been in there, but that relay was missing. And that is what caused that PORV to stick open.

I was concerned "Was it sabotage," or "Were there other relays of that type missing?"

Q. Would the operator have had to ever clean the relays in the PORV during use? In other words, would he have to take it apart and clean it, and put it back together again at all?

A. Would you state that question again?

Q. Would the operator ever have to take the PORV apart to clean the relay that was apparently missing on the date of September 24, 1977?

A. First, I'll answer your question this way. First of all, if you are speaking of an operator in the control room I don't think he would be in that area making -- cleaning relays.

Item 2, inasmuch as that was not a safety grade QA item to check, I do not know with any specificity what the licensee's check procedure may have been on that relay. But, nevertheless, the relay was missing, and that concerned me very much.

Q. Well, my prior question did not deal exclusively with the control room operator, but with anyone as part of Toledo Edison who may have been involved with cleaning the relay and taking it apart. Do you know if that is a situation that occurs, the relay is periodically cleaned or is the unit a sealed unit and is not subject to maintenance?

A. Since it is not a safety grade item, I don't know what the licensee's procedure would be in repair of that. But I would like to say I became very concerned about that relay valve, and I talked to Lowell Rowe, who is vice-president of construction, I believe, at Toledo Edison, and I talked to the I&E people.

I think based both on my concern and the licensee's concern, and I&E's concern, Toledo Edison made a through-plant search where other relays of that type were to assure themselves that they were in there. As I remember they also put some notices up on the wall asking anyone who knew anything about this relay missing to come forward and to tell them about it.

Q. Were you at all involved in the investigation of the electrical setup of the PORV at Davis-Besse on September 24th?

A. Not to any greater specificity than I have just been relating. If there had been a matter come up that I&E wanted us to specifically address, I&E would have issued a transfer of responsibility to DSS and then I would have been involved in setting up the time to get the reviewers to look at it. And there never was a transfer of lead responsibility.

Q. Do you know as of this date whether or not there was a problem with the electrical setup of the PORV at Davis-Besse?

A. Inasmuch as that relay was missing, there was an upset.

Q. Is that based on the missing component or the electrical wiring of the valve?

A. To my knowledge it is the missing relay which cut off the signal.

Q. So, at this date you are unaware of any problem that existed with the electrical wiring of the PORV at Davis-Besse in September of 1977; is that correct?

A. Well, to get current from one place to another you can say that a relay is part of the electrical system.

Q. Another problem exclusive of the relay?

A. No; I did not.

Q. And you currently are unaware of any problem involving the electrical system of the PORV at Davis-Besse excluding the relay at this point and time?

A. At this point and time.

Q. Do you recall who was at the Monday, October 3rd meeting?

A. I remember a lot of them, and they came in at different times. Of course, there was Jerry who I had mentioned.

Q. Jerry Mazetis.

Szukiewicz

A. I believe Andy ~~Sukabitch~~ was there. Jim Knight of the Mechanical Engineering Branch was there. I believe Jai Raj N. Rajan ~~Roge Rojan~~ (ph.) was there who had been one of the NRR people that went out to the site. Vince Leung, L-E-U-N-G, was there, and he is with the Auxiliary Systems Branch.

Q. How about Carl Seyfrit (ph.)?

A. Carl Seyfrit was there. He was with I&E.

Q. How about Roger Mattson?

A. Roger Matteson, Director of DSS, was there.

Q. Sandy Israel?

A. Sandy Isarel was -- Now, I am remembering faces at that meeting. I don't remember seeing Sandy -- I can't remember, but I would certainly think he had been there.

Q. Tom Novak?

A. Yes, Tom Novak was there.

Q. Denny Ross?

A. I don't specifically remember Denny Ross being there. I don't believe he was. I don't remember seeing him there. They were coming in and out.

Q. Were there any discussion during this meeting of the operator relaying on pressurizer level indication turning off the HPI system prematurely?

A. As I remember, no. That was not specifically brought up there. And as I related before, Jerry started out using this graph to go through the overall scenario. People were coming in interested in their particular area of expertise. A person would come in and say, "What do we have to do with the 'aux' feed system?" What do we have to do with this system. There were a lot of interruptions.

Q Were you there for the entire meeting?

A Yes.

Q So, if the matter of an operator relying on pressurizer level indication turning off the HPI system prematurely did come up, you would have known about it?

A Yes. Now, I would like to state that based on this graph where it indicates some of the action, they may well have said that the HPI's were turned of. at such and such a point.

Q Did you present this graph at this October 3rd meeting?

A I had made it up because of my interest. When I saw there was going to be this trip report I volunteered this to Jerry Mazetis to use, and he did use it.

Q Did you volunteer this to Mr. Mazetis during the meeting on October 3rd, Monday?

A Oh, yes. Before it began I said, "Jerry, this might help you. I made this up."

Q Did he during the meeting rely on this diagram?

A He did earlier -- I mean as much as he was trying to use that scope out the transient as a function of time.

Q There appears to be no indication on this diagram or graph of pressurizer level indication --

A Oh, yes.

Q Let me finish. -- going off scale high at the time the HPI pumps were turned off; is there? It appears the HPI pumps were turned off at approximately four -and-a-half minutes into the accident; is that correct?

A That is correct.

Q And at that time the pressurizer level indication appears to be at approximately 220 inches?

A That's correct.

Q And do you recall what the pressurizer level indication maximum reading was at Davis-Besse during this transient? Was it greater or less than 220 inches?

A Oh, it was greater.

Q So, according to this graph, pressurizer level is increasing but still on scale at the time the HPI pumps were turned off; is that correct?

A That is correct. And that is one of the reasons that the operator who saw his pressurizer level coming up to about normal operating, so he shut off his HPI pumps.

Q What would be the normal level of operation for pressurizer level indication?

A Without going to the FSAR, as best as I remember it would be about 220.

Q And at some time within about a half hour and after -- Well, as a matter of fact within three minutes after the operator turned off the HPI system, pressurizer level indication went off scale high; is that correct?

(Pause.)

At least it went up to a level of about 310 inches. Would that be off scale high?

A That would be very close to off scale.

MR. SIDELL: If we can get a copy of this graph, I would like it as Exhibit No. 3 to the deposition. Do you know how we can reproduce this? Does the NRC have some method to produce something of this size?

THE WITNESS: If I could use your request as part of the President's Commission, I could probably expedite them making me a copy. I would be glad to give this, but I went through a lot of work that weekend. It has been on my wall ever since.

MR. SIDELL: It certainly appears it. We can use a copy of this.

THE WITNESS: I will get it to you as quickly as possible.

MR. SIDELL: Do you know if it is possible to make a copy that has the same color or color differentiations as appear on the original?

THE WITNESS: I can try. I cannot say.

BY MR. SIDELL:

Q. And you produced this diagram based on the computer readings of the actual event at Davis-Besse in September?

A. Yes, from that Friday 30th meeting and the reactimeter data they gave me.

If I had to do it over again, I might put more curves on there or less. But those are the items that I was interested in at the time, and it was primarily the primary system.

Q. Well, I think this more than adequately deals with what we are concerned with.

MR. SIDELL: For the record, let me just state that this is titled "Davis-Besse Unit-1 Reactor Trip from about Nine Percent Full Power at 2130 Hours, September 24, 1977," a diagram which plots reactor core pressure, PSIG versus time of the transient with several characteristics including pressurizer level indication, cold temperature, hot temperature, reactor core pressure, saturation pressure, and various operator actions occurring at a time of something less than two minutes preceeding the transient to a time of 68 minutes into the transient.

At the first opportunity Mr. Engle has agreed to provide us with a copy of this diagram.

BY MR. SIDELL:

Q. Based on the operator's action in turning off the

HPI system by his reliance on the pressurizer level indication as depicted in this diagram, Exhibit No. 3 to this deposition, would you conclude that pressurizer level indication would be an important characteristic or parameter of the primary system on which the operator relied during the abnormal conditions?

A. It is one of several important indications that he should watch.

Q. Well, based on this diagram, it appears as though it is one of four characteristics: hot temperature, cold temperature, saturation pressure, and pressurizer level indication; is that correct?

A. That's right.

Q. Do you have any information in investigating this transient that allows you to conclude pressurizer level indication was relied upon more heavily than the other three parameters by the operator?

A. In that December 30th meeting out at Davis-Besse where I mentioned the people were --

Q. Excuse me. I believe you said December 30th. Did you mean to say October 30th at Davis-Besse?

A. I mean the September 30th meeting where the staff first went out.

We talked with the operators to some length. And I remember I fielded several questions to the operators

"What did you do," "Why did you do it?" At that time I remember the question was brought up "Why did you turn the high pressure injection off?" And his answer was "We relied very, very heavily on pressurizer level." However, I think it is only fair to state that in our questioning and the licensee's response the operator went on to realize that he was reaching P-sat condition, saturation conditions.

He was an operator who to me knew something about steam thermodynamics. He also from that graph at certain points tripped two of his reactor coolant pumps. And as I remember his reason for tripping those reactor coolant pumps was because he knew that he had reached P-sat conditions -- saturation conditions -- he had some steam beginning to form in there. And inasmuch as Davis-Besse had only been operating at nine percent power, and only had one effective full power day of operation -- They had just been coming up in pre-op tests -- the reactor coolant pumps were supplying as much heat -- Those pumps each produce about five megawatts of power.

In those conditions the man was concerned that he was going to trip two of those pumps to keep down heat generation, down there in the impeller blade area to cut down on possible, larger bubble entrainment.

I personally was quite impressed with- that man at that time.

Q. Do you recall his name?

A. I do not recall the man's name, and I should remember it.

Q. Do you recall whether or not Mr. Mazetis made a copy of your diagram and used it with his trip report?

A. No; I do not.

Q. Is it likely that you would have remembered if he had made a copy?

A. Yes; I think so.

Q. Do you recall when he returned your original diagram to you? Was it within a few days after this meeting, September 30th or a couple of months later?

A. I am not sure if I took it back with me from that meeting. I just don't remember. But I know I have kept it on my wall ever since except for lately.

Q. Why haven't you kept it on your wall lately?

A. Because after one day after TMI-II it disappeared from the wall and people were asking to see it.

Q. Do you know how it disappeared from your wall?

A. Well, it disappeared one night. I became interested -- because this is firmly embedded in my mind. The next day it was back there and then several people began asking me in the halls what happened to Davis-Besse with some specificity and I would tell them to come on down to the office and I will show you. "I've got my graph." And then people began borrowing it. They wanted it as a backup in the early hours of the event to try and see if there was any relation.

Q. Do you know if anyone who borrowed it took it to the Incident Response Center?

A. No; I don't. I don't know where it went. I know it was going all over everywhere, and my concern was both that it was being useful and also would I ever get it back.

Q. Is that what prompted you to write "Return to Leon Engle" in the upper righthand corner?

A. Yes.

Q. So, that was not included on the diagram when you originally prepared it?

A. That occurred after TMI.

Q. Can you recall the names of anyone who borrowed your diagram? Specifically, did Sandy Israel borrow it?

A. Well, the first man who borrowed it who I had a real conversation regarding some of the technicalities was a Stuart Rubin ~~Trebee~~ (ph.) in the Division of Operating Reactors. ~~I think his name was Stuart Trebee. I know it is Stuart. I know him when I see him. I guess I am not completel_ — absolutely sure of the name Trebee~~. But he works in DOR in the assistance group up there. He asked if he could have it because there were review groups set up that interspersed with DSS and all the groups at that time, and he said he wanted to use it because he felt it showed him some significant

I told him, "Go ahead and use it." I know from there it went all over the building because people in the hall would say "Hey, we saw your graph."

Q. Which building was this?
A. Phillips Building.
Q. The new Phillips or the old Phillips?
A. Both.
Q. Do you know Sandy Israel?
A. Yes; I know Sandy Israel.
Q. Do you know if he saw your graph?
A. I do not know if he saw my graph.
Q. What about Tom Novak?
A. I do not know if Tom saw it. I would think they might have during those discussions.
Q. After TMI-II?
A. After TMI-II.
Q. Do you know if Roger Matteson saw it?
A. Well, they all should have seen it back at that Monday morning meeting for the trip report.
Q. Well, I am specifically referring to some time shortly after the TMI-II accident of this year.
A. Inasmuch as I was not there, I cannot answer that.
Q. So, the only person you specifically know that your graph was viewed by was Stuart ~~Tresse~~ Rubin, you believe his name to be?

A. That is correct. I know it was going elsewhere.

Q. Did he indicate that to you?

A. Yes, because I asked him about it, too. "Where is my graph? Be sure I get it back."

There were several people that saw that graph before TMI-II that I had discussions with.

Q. Sandy Israel?

A. No.

Q. Tom Novak?

A. No.

Q. Jerry Mazetis?

A. No.

Q. Who did you discuss it with before TMI-II?

A. Well, some time during the period of January, 1978 and April of 1978 I had a conversation with a Mr. Jack Rowe (ph.) who is in the Safeguards Branch.

Q. What is his position or what does he do?

A. He is involved in industrial security.

Q. Is that something that involves security over fuel for reactors on there way to or from the reactor?

A. It is more involved in the security of a plant in the way of sabotage and those type of matters.

Q. Did Mr. Rowe believe there was sabotage involved at Davis-Besse in September, 1977?

A. No. But he was inte[illegible]sted in the scenario from one of the same standpoints, and that was that that relay was missing. And he asked me if he could reproduce this to emphasize on licensees how if some little item might be missing, what could happen. And I believe he sent that to Los Alamos because Los Alamos was involved in part of the safeguards review.

There were two other people that had interests in that graph prior to TMI-II. Another person was a Dominic Dijanni ~~Theodami~~ (ph.) in the Division of Operating Reactors. And I think at that time he was involved with the Mechanical Engineering Branch at DOR.

Dominic came down to my room and said he was investigating the LER.

Q On September 24, 1977 for Davis-Besse?

A That's right. And I pointed out this curve to the young man, and I think he took it for a short time. The amount of time I don't remember.

There was another man that I showed this curve to which was very short after we got back from that meeting. That was Robert McDermott (ph.) Bob -- I have been involved very closely with him because he is involved in start-up and preoperational tests. And, of course, Davis-Bessee was going through this period.

I showed this to Bob, and he used it in writing a report to his assistant director in QA, Donald Showholt ~~Skovall~~ (ph.), and my name was on that note for a copy attached.

Q. Do you know if Mr. Skovholt ~~Skovall~~ saw the diagram?

A. No; I do not know if Mr. Skovholt ~~Skovall~~ saw the diagram.

Q. But in Bob's note to Skovholt ~~Skovall~~ he indicated that he had talked to me and he went through the assessment of the incident. And his concluding paragraph said, "I am having further communications with the I&E inspector regarding the relay."

Q. Did he mention the specific I&E inspector's name?

A. No; he did not.

Q. Would that have been Tom Tamblin?

A. It would have either been Tom Tamblin or Terry Harpster because those men were most deeply involved in investigating that event at that time. Tom was -- Both Tom and Terry were at the very beginning and then were three or four other inspections they had out there prior to the time Davis-Besse came back on the line.

I remember that one or the other might be back in headquarters, because if I called one and couldn't get them I called another. I was calling both of them.

Q. At this September 30th meeting, Monday, I believe you previously stated you couldn't recall whether or not there was any mention of premature termination by the operator of the HPI system based on his observation of pressurizer level indication; is that correct?

A. I don't think the word "premature" was used at that time, but I highly imagine it was mentioned that the operator turned the HPI's off.

Q. So, there was discussion about operator termination of the HPI system based on his viewing of the pressurizer level indication?

A. I remember Jerry followed fairly closely these particular parameters on this graph, and I am sure that came out.

Q. Do you recall whether or not during the course of that meeting there was a statement by Mr. Seyfrit that I&E would follow up on investigating or resolving the questions raised by the September 24, 1977 transient?

A. Yes; I remember that.

Q. Mr. Seyfrit did indicate I&E would follow up?

A. Right, that lead responsibility was still with I&E.

Q. Do you know whether or not there was any follow up by Mr. Seyfrit or someone in I&E --

A. Oh, there was follow-up based on their inspections out there, there conclusions and report that I&E made. I think I mentioned that. I think October 22nd. And then the licensee supplement, which you have mentioned, indicates actions that they had taken. And I called Carl on several times because of my concern in getting these items all done. I talked to Carl several times and "How is the inspection going.

Q. That's Carl Seyfrit?

A. Yes.

Q. I also had a concern, and I think I must have talke to Carl Seyfrit, Jerry Klingler (ph.), Tom Tamblin, Terry Harpster, and also Dick Knopp, all of I&E, whether during their evaluation at any time they thought they might transfer lead responsibility to us on some matters. Now, this is of great concern to a project manager, because if that occurs then he must make the responsibility of getting all the troops together that will review that item, and they may be all busy on all other types of reviews and schedule, and you don't want to hold something up.

I was very concerned about that. That was my responsibility.

Q. Do you recall if there was any emphasis on the possible safety ramifications of an operator prematurely terminating the HPI system based on his observations of pressurizer level indication?

A. To the best of my knowledge, no.

Q. Was that a concern to you at the time?

A. At the time I don't believe it was as much a concern to me as the missing relay and seeing that all that equipment was checked out.

Q. Was there any investigation or a suggestion of an investigation that the pressurizer level indication should be checked out?

A. At that time, no.

Q. Did everyone presume the pressurizer level indication to be an accurate measure of core inventory at the time?

A. I didn't hear anything that would have indicated otherwise.

Q. There is no indication of core inventory on your diagram, which we will mark as Exhibit No. 3 to this deposition; is there, Mr. Engle?

A. There is no plot of core inventory specifically. However, one can certainly infer core inventory from some of those parameters.

Q. Temperature, pressure?

A. Yes. It is not a definite indication, but if you know the reactor coolant system from Thot, Tcold and pressure, you can come to a fairly good conclusion on what the overall inventory is in the core. And by inventory in the core that you are not late getting to core uncovery.

Q. So, at approximately seven minutes into the accident -- or from six minutes into the accident, according to your diagram, we have an increasing pressurizer level indication which appears on your diagram as a red line, but decreasing temperature both of the hot leg and the cold leg of the reactor; is that correct?

A. That's correct.

Q. Would that be the expected, normal results of those parameters to go in divergent directions?

A. Not to the extent that you see on that graph. However, there was another event going on at this time. Another transient that I haven't pictured there. And that involves the steam generators.

Way back when you get to the beginning -- what first caused the problem was that there was a spurious signal. By spurious, nobody knew where it came from. In the start-up steam water valve for the No. 2 steam generator, this spurious signal locked into the logic and closed that startup feed water valve. That immediately began to lower the water level in No. 2 steam generator. And at the point where the low level in the steam generator that locked in the other part of the half-trip and caused a full-trip of the main steam line isolation valves and started for Davis-Besse and the steam feed water and rupture control system the auxiliary feed water system pumps.

Now, because of that drop in the water level and closure of the main steam line isolation valves, that raised the temperature in the reactor coolant system and caused an expansion of the reactor coolant and that caused a rise in the presurizer level also and also in the reactor coolant pressure. And the reactor coolant pressure continued to reach that point where we get to where we discussed before that the POVR opens and we discussed.

Now, getting back to the No. 2 steam generator, when the auxiliary feed water system came on for the No. 2 steam generator -- The auxiliary feed water pump is a steam driven turbine feed water pump -- and it failed to come up to its rated speed of 3600 rpm. It only ~~rates~~ rated to about 2600 rpm because of a linkage bar fault in the governor, and at that point the 2600 rpm's did not provide sufficient head compared to the steam generator to allow water to go on into the No. 2 steam generator. So, during this event the No. 2 steam generator went dry.

Now, I didn't put any items of the secondary system on that graph at that time because I felt that that transient, although it certainly gets involved in the parameters that are on the graph -- I wanted to keep track of the reactor coolant system, the primary system. There has been much talk about what happened since, especially at TMI. I stand on this: I think the opening of the PORV was more significant than any of the malfunctions in the aux feed system for Davis-Besse, because the design basis accident and analysis for Davis-Besse was based partly on the assumption that it would operate within its specs on only one steam generator. And they had that No. 1 steam generator.

But whenever you come up with any graph it is the sum of many of these different transients.

Q. Can you recall any reports produced as a result of the September, '77 Davis-Besse transient that concerned changing operating procedures for turning off the HPI system?

A. No; I do not.

Q. Have you seen a copy of Jerry Mazetis' trip report on this transient?

A. No; I have not.

Q. So, as far as you know, one doesn't exist?

A. I have heard Jerry mention it.

Q. Would you normally in the course of procedures at the NRC receive a trip report if an accident occurred at a plant for which you were project manager?

A. Yes; I would.

Q. And you have not received a trip report concerning this transient at Davis-Besse?

A. No; I have not.

Q. And based on that would you assume that a trip report was or was not produced?

A. I don't know.

Q. So, if someone from Region-3 called you up and said, "Is there a trip report on this transient," even up until today, what would your response be to them?

A. I have not seen a trip report cross my desk.

Q. So, would you tell the inquiring individual that nothing existed by way of a trip report?

A. Yes.

Q. Because you would expect to be copied on --

A. I would see it -- be getting a copy of it.

T2S1

Q. Would you characterize premature operator termination of the HPI as an unresolved safety issue?

(Pause.)

Do you understand my question, Mr. Engle?

A. In what time period?

Q. If the operator observing pressurizer level indication turns off the HPI system before he properly should turn it off, would that be an unresolved safety problem in your view?

A. Based on TMI-II, yes. But it is being looked into now.

Q. In the lessons learned --

A. Lessons learned -- Bulletin and Orders Group. That is part of the questions that have gone out relating to operator procedure.

Q. Did you consider the premature HPI termination by the operator at Davis-Besse in September, 1977 based on his observation of pressurizer level indication to be an unresolved safety matter?

A. No; I did not at that time.

Q Do you know whether I&E considered it to be an unresolved safety matter at that time?

A I don't know.

Q If I&E considered this problem to be an unresolved safety question, would they have provided you with copies of any documents they would have generated in attempting to resolve the problem?

A If they thought that their manpower did not encompass the expertise to review that area, they would have sent a formal transfer of responsibility specifying that area where we should have evaluated.

Q And to your knowledge you have not seen such a transfer of responsibility?

A To my recollection there never was a transfer of responsibility on the Davis-Besse event.

Q Would you conclude on that basis then that I&E felt that they had sufficient technical staff to resolve the question?

A Yes.

Q Would the fact that you did not see any reports dealing with premature termination of the HPI system as an unresolved safety matter indicate to you that I&E did not consider it a safety problem?

A Would you repeat that question?

MR. SIDELL: Would you read it back, please.

(Whereupon, the pending question was read by the reporter.)

THE WITNESS: I don't know.

BY MR. SIDELL:

Q. You didn't conclude one way or the other by not seeing an I&E report whether or not they believed it to be a safety problem or not?

A. I just don't know.

MR. SIDELL: Off the record.

(Discussion off the record.)

MR. SIDELL: Back on the record.

BY MR. SIDELL:

Q. Am I correct in concluding that you previously stated, Mr. Engle, that at the Monday meeting on September 30th there was no indication that HPI termination by the operator was viewed as a safety concern when I&E accepted responsibility for resolving the matter?

A. Repeat.

Q. Did you previously state that there was no mention during the Monday, September 30th meeting that HPI termination occurring in Davis-Besse in September, '77 was viewed as a safety matter when I&E agreed to look into things?

MR. TRUBATCH: Was or was not viewed as a safety matter?

MR. SIDELL: Was not viewed as a safety matter.

THE WITNESS: I don't know. I don't know what was in their minds.

BY MR. SIDELL:

Q But at the September 30th meeting did you previously state termination of the HPI was not mentioned to be a safety matter -- safety problem?

A To the best of my knowledge it was not specified as a given safety problem.

Q Alternatively there was no emphasis by Mr. Novak, Mr. Israel, Mr. Mazetis or anyone at that meeting to I&E to consider in their investigation that premature termination of the HPI system was a safety concern that they should look into and resolve?

A Not to my knowledge. I would like to state that meeting was a little hectic. People coming in and asking questions. Some people may have been hearing one thing. They didn't all get there at the same time, and being a project manager that sort of went against my grain because I normally conduct meetings. But they were all --

In the time the meeting went on the most specificity was given at the very first when Jerry was describing this event.

Q Do you recall the length of the meeting, start to finish?

A I don't believe it was over an hour.

Q. Were people informed of the existence of the meeting by telephone call or was there a formal memoranda sent around to participants?

A. I don't believe so. However, I was calling a lot of people trying to get some input as regards certain items.

Q. It was essentially notification by phone?

A. Notification by phone. I had many, many telephone calls with I&E. Like I said, I was concerned whether there was going to be a transfer of lead responsibility. I was also concerned with what was going on. I was very concerned about that relay missing.

Q. But in order to get people to this Monday morning meeting most people to your knowledge were notified by telephone?

A. Oh, absolutely.

Q. And then you subsequently followed it up with I&E with further telephone calls?

A. That is correct.

Q. Who did you speak with in I&E, if you can recall?

A. It is a variety of individuals. Shortly after the event it was mostly Terry Harpster and Tom Tamblin and their boss, Dick Knopp. I also talked to Jerry Klingler ~~Clingler~~ (ph.) several times, I&E and headquarters. And as I remember I also talked to Carl Seyfrit several times.

Q. So, you spoke with people both in Region-3 and in I&E headquarters here in Bethesda?

A. Right.

Q. In your conversations with Tom Tamblin from Region-3, did he indicate to you that there was no safety concern from Region-3's perspective of turning off the HPI system prematurely?

(Pause.)

Do you understand the question, Mr. Engle?

A. I understand the question. I am just trying to give you the -- I do not remember whether shut off of the HPI's specifically whether I discussed that with Tom Tamblin. It seems to me our conversations were more related to their investigations of what had caused the transient and corrective actions that they were taking.

Q. So, it appeared to you that Region-3 was not paying an overly great amount of attention to the manual override of the HPI system?

A. No. That is not correct, because I was not overly sensitized to that item at that time, too. So, I can't say that I was -- had some -- I, myself, was not paying too much attention to that item.

Q. But would you conclude that Region-3 also was not devoting a great deal of time to that matter?

A. I was so involved at the time in these other things I can't answer that in that time interval. In hindsight

I would say yes.

Q. Region-3 was not spending a great deal of time investigating turning off the HPI system?

A. In hindsight, yes.

Q. In hindsight do you think they should have?

A. In hindsight we all should have.

Q. Are you aware of any inspection and investigation reports filed by Jim Crestwell concerning the September, '77 Davis-Besse transient?

A. Not specifically.

Q. Have you heard of them by word of mouth?

A. I have heard of them by word of mouth relating to that event, event further on, and then matters relating to -- I think it's pressurizer level after I transferred Davis-Bessie to DOR.

Q. Do you recall when you heard of Mr. Crestwell's report?

A. I think it was right at the very time that TMI was happening.

Q. Shortly before or after, if you can recall?

A. A very, very narrow band, very, very shortly before.

Q. Do you remember how you heard of Mr. Crestwell's report?

A. I heard it by word of mouth.

Q. Do you remember from whom?

A. No; I don't remember from whom, but I do remember being curious and getting a copy of that. There was so much -- That was in the TMI event. That's when I got it because there was so much going on then, and I did get it from somebody but there are so many people involved in what I was doing then, I just can't remember who I got it from.

Q. So, your best recollection is that your received copies of Mr. Crestwell's report after TMI occurred?

A. That is right.

Q. Were those reports dealing with the September, '77 transient or the November, '77 transient at Davis-Besse?

A. To my best recollection November of '77.

Q. Have you seen any reports authored by Mr. Crestwell concerning the September event?

A. No; I have not.

Q. Have you heard of any?

A. No; I have not.

Q. Are you familiar with something referred to as the Michaelson report?

A. I certainly am now.

Q. You were not preceeding TMI-II of this year?

A. No, sir.

Q. Have you read the report?

A. I have not yet, but I certainly intend to based on all my --

Q. Do you have a copy of it?

A. I do not.

Q. Do you know the essential thesis of the Michaelson report?

A. The basic thesis I believe is that in a certain size break there may well be transients that need further investigation as far as adequate core cooling. And I believe that includes the diameter break on the PORV valves -- I mean the diameter size that would have been for Davis-Besse and for TMI.

Q. Would you conclude that Mr. Michaelson's central thesis is that the design basis for LOCA's, at least in B&W reactors, did not adequately deal with very small break LOCA's?

A. My best answer to that is that it would demand that analysis be made in that range of breaks, which is now being done.

Q. Are you involved in that analysis?

A. Only very indirectly, and that is involving licensing matters of actions that had to be completed and based on the Bulletin and Orders Group's upwriting of safety evaluation based on licensees input.

Q. Have you heard of a memorandum by Mr. Novak -- Tom Novak?

A. Tom writes a lot of memorandums. What memorandum?

Q. Specifically dealing with the accuracy or ambiguity of pressurizer level indication and an operator's reliance on pressurizer level indication and turning off the HPI system.

A. I have never received a copy.

Q. Have you heard of it?

A. I have heard of it recently.

Q. After the accident?

A. After the accident.

Q. Not before?

A. Not before.

MR. SIDELL: At this point, Mr. Engle, I have no further questions. What we have been doing as our general procedure is rather than adjourn the deposition we will merely recess it in the event that we have further questions for you we can more easily continue the deposition.

I will advise you that we have not yet recalled any deponent to continue his deposition, although we expect to do that in a very small number of cases. I at this point and time would doubt that we would find that necessary in your case but merely to be consistent with our prior policies we will recess rather than adjourn the deposition. And we will await a copy of your diagram dealing with the September 24, 1977 Davis-Besse transient as Exhibit No. 3 to this deposition.

THE WITNESS: Can I get your name and where I can get this to you?

MR. SIDELL: You certainly may. And if counsel will so stipulate that upon receipt of a copy of Mr. Engle's diagram it will be included as an after-included exhibit as No. 3.

MR. TRUBATCH: We so stipulate.

MR. SIDELL: Off the record.

(Whereupon, at 4:00 o'clock p.m., the taking of the deposition was recessed.)

I have read the foregoing pages, 1 through 62, and they are a true and accurate record of my testimony therein recorded.

____
LEON B. ENGLE

Subscribed and sworn to before me

this ______________ day of __________, 1979

____
Notary Public

My Commission Expires: __________________________

REPORTER'S CERTIFICATE

DOCKET NUMBER:

CASE TITLE: Accident at Three Mile Island
deposition of Leon B. Engle

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 7, 1979

[signature]

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

Leon B. Engle
Licensing Project Manager
Light Water Reactor Group No. 1
Division of Project Management, Office of Nuclear Reactor Regulation

Telephone 492-8349
Room 143

..ERIENCE:

Nov. 1974 to present

DIVISION OF PROJECT MANAGEMENT, OFFICE OF NUCLEAR REACTOR REGULATION
(Nuclear Regulatory Commission - 1975 to present and Atomic Energy Commission - 1974)
Staff Member - Licensing Project Manager

Responsible for managing and coordinating staff review and interfacing with applicants and licensees for radiological safety reviews and for plant operation and preparation of Safety Evaluation Reports, Supplements, Operating Licenses (Amendments) and reporting to the Advisory Committee on Reactor Safeguards on the following nuclear power plants: (1) Davis-Besse Nuclear Power Plant, Unit 1 (Ohio), (2) Standard Balance of Plant Boppsar-BSAR-205 (Illinois), (3) Palo Verde Nuclear Power Station, Units 4 & 5 (Arizona), (4) Diablo Canyon, Units 1 & 2 (California), (5) Crystal River Nuclear Power Plant, Unit 3 (Florida), (6) Midland Nuclear Power Plant, Units 1 & 2 (Michigan), and (7) Greenwood Energy Center, Units 2 & 3 (Michigan). Also, managing Arkansas Nuclear One, Unit 2 operating licensing matters. Presently engaged in licensing matters regarding NRC Commissioner's Order regarding the restart of Three Mile Island, Unit 1.

May 1973 to Nov. 1974

DIVISION OF PROJECT MANAGEMENT, OFFICE OF NUCLEAR REACTOR REGULATION, ATOMIC ENERGY COMMISSION - on loan status from Los Alamos Scientific Laboratory
Staff Member - Licensing Project Manager

Responsible for managing radiological safety review, interfacing between applicant and review staff, and preparing Safety Evaluation Report for the Greenwood Energy Center, Units 2 & 3 (Michigan).

June 1971 to May 1973

Group N-2, LOS ALAMOS SCIENTIFIC LABORATORY
Staff Member - Physicist

Analysis of LMFBR hypothetical accidents for code and model development; Equation-of-state studies and testing of the calculational method; Application of code and methods to existing and proposed reactor designs and to conceptual experiments appropriate to reactor safety; Basic physics includes the use of a coupled neutronics-hydrodynamic code, the use of neutron-transport theory, heat transfer, and dynamic forces; Analysis of experimental and theoretical results associated with energy release for critical assemblies; Progress reports to RDT, 1971, 1972 and 1973; The use and knowledge of computers (CDC 6600 and IBM 7090).

Jan. 1967 to June 1971

Group N-2, LOS ALAMOS SCIENTIFIC LABORATORY
Staff Member - Physicist

Calculation of energy release associated with reactor accidents as input to safety analysis; Evaluation of experimental results from critical assemblies for theoretical code development; Fast reactor parametric studies; Testing and using transport, kinetic, and dynamic reactor models; Liaison with consultants to N-2 on reactor accident calculations; Studies as related to Rover reactors for design and start-up analysis; Fortran coding and use of computers; Familiar with AEC regulations: radiation, safety and environment.

Jan. 1969 to
Jan. 1971

STATE OF NEW MEXICO HEALTH AND SOCIAL SERVICES BOARD
Member (Jan. 1969–Jan. 1971); Chairman (Oct. 1969–Oct. 1970)

Policy-making and regulatory jurisdiction in environment, public health, and social services; Board member most involved with drafting and steering toward adoption (Jan. 1970), the Ambient Air Standards and Air Quality Control Reg. for State of New Mexico; Mathematical evaluation of Four Corners power plant complex; Board representative at NAPCA seminar for regional implementation plans, Kansas City, Mo. (Jan. 1970); New Mexico representative at final consultation on Four Corners Interstate Air Quality Region (Nov. 1970); Hearing officer (Board) for air pollution reg.; Interchange with Federal and State Agencies (N.M., Colo., Utah, Ariz., Texas, etc.) on air pollution, and with industry, conservationists, and elected officials regarding environmental problems; Personal and group meetings with Medical Societies, health institutions, Indians, Welfare and Minority Groups; Acquainted with Clean Air Act (42 U.S.C. 1857 et seq.) and air quality criteria (particulates, SO_x, NO_x, etc.); Assessment of efficiency of electrostatic precipitators in power generation utilizing low-sulfur, high-ash content coals.

Jan. 1959 to
Jan. 1967

GROUP N-2, LOS ALAMOS SCIENTIFIC LABORATORY
Staff Member - Physicist

Neutron Transport and Kinetic calculations for safety analysis on proposed nuclear power plants; Parametric studies for Rover reactors for design and startup analysis; neutronic assessment of Phoebus 2 nuclear rocket reactor; computer programming and calculations for successive mockup loadings used in experimental configurations for criticality determinations; developed computer programs for data analysis of experiments on delayed gammas from fast neutron fission and operated and controlled critical assembly used in experiment.

April 1955 to
Jan. 1959

Group N-2 LOS ALAMOS SCIENTIFIC LABORATORY
Staff Member - Physicist

Calculation and evaluation of six-group fast neutron cross sections for reactor studies; Diffusion theory studies as related to critical assemblies; Operation of critical assemblies; Computation and assessment of reactivity contributions of various elements in critical assemblies.

July 1951 to
April 1955

Group W-1, LOS ALAMOS SCIENTIFIC LABORATORY
Research Assistant, Physicist

Weapon design and evaluation; Radioactive monitoring, safety surveillance, statistical analysis, quality control; Development of procedures, training, and supervision of military and civilian personnel as related to weapons assembly; Supervised t[illegible] in monitoring and quality control; Supervised military and civilian crews (4-5 persons) for six-month intervals.

Nov. 1950 to July 1951 — HQS. 2754TH EXPERIMENTAL WING, HOLLOMAN AFB, NEW MEXICO
Physicist

Evaluated data reduction methods to be used by radar beacon missile tracking system; Considerable effort required to effect necessary communication between representatives of Air Force, Civil Service, and private contractors regarding systems problems. Involved travel and working in various climatic conditions--desert and mountains.

EDUCATION:

1946-1950 — B.S., Physics and Mathematics: Colorado College, Colorado Springs, Colorado.

1955-1958 — University of New Mexico Extension Courses, Los Alamos Scientific Laboratory; Nuclear Reactor Technology and Engineering; Statistical Analysis; Matrix Theory; and Electrodynamics.

Jan. 1970 — Seminar for Regional Implementation Plans, National Air Pollution Control Administration, Kansas City, Mo.

Nov. 1970 — Air Pollution Control Technology, National Air Pollution Control Administration, University of Texas, Austin.

July, 1973 — Nuclear Safety, Massachusetts Institute of Technology, Cambridge, Massachusetts.

Nov. 1974 — Technical Writing, sponsored by the Atomic Energy Commission, Bethesda, Maryland.

Sept. 1977 — Pressurized Water Reactor Fundamentals (CE Systems 80 Design), U.S. N.R.C., Bethesda, Maryland.

Oct. 1978 — Boiling Water Reactor Systems Fundamentals, N.R.C., Bethesda, Maryland

Jan. 1979 - May. 1979 — Environmental Impact of Energy; Nuclear / Fossil, Catholic University, Washington, D.C.

PROFESSIONAL REFERENCES:

Dr. Gordon E. Hansen
Dr. William R. Stratton
Dr. H. C. Paxton

Los Alamos Scientific Laboratory, Los Alamos, N. M.

Mr. John Stolz

Light Water Reactors Branch No. 1, DPM, NRR

MILITARY SERVICE:

1943-1946 — U. S. NAVY, CM2C; Navigation, Meteorology.

PUBLICATIONS:

"Reactor Division of Technology and Space Nuclear Systems Office Progress Reports," W. R. Stratton, L. B. Engle, and D. M. Peterson, (1971 and 1972)

"Application of the PAD Code to LMFBR Power Transient Studies," L. B. Engle, W. R. Stratton, and D. M. Peterson, ANS Trans. 15, No. 2, 820 (Nov. 1972)

"The Pajarito Dynamics Code with Application to Reactor Experiments," W. R. Stratton, D. M. Peterson, and L. B. Engle, ANS Trans. 15, No. 2, 819-820 (Nov. 1972)

"Reactor Power Excursion Studies," W. R. Stratton, L. B. Engle, and Donald M. Peterson, International Conference on Engineering of Fast Reactors for Safe and Reliable Operation, Karlsruhe, Germany (Oct. 1972)

"Large Accidents," W. R. Stratton, L. B. Engle, and D. M. Peterson, ANS Trans. 13, No. 2, 721-722 (Nov. 1970)

"Pewee 2 Control Vane Reactivity Accident," W. R. Stratton, D. M. Peterson, and L. B. Engle, N-2 Progress Report (March 1970)

"Reactor Explosion Calculations," W. R. Stratton, L. B. Engle, and D. M. Peterson, N-2-8451 (Jan. 1970)

"Critical Dimensions of Homogeneous Spheres Containing ^{235}U, ^{238}U, and Carbon for Various C/^{235}U Ratios and ^{235}U Enrichments," L. B. Engle and W. R. Stratton, LA-3883-MS (Dec. 1967)

"Delayed Gammas from Fast-Neutron Fission of Th^{232}, U^{233}, U235, U^{238}, and Pu^{239}," P. C. Fisher and L. B. Engle, Phys. Rev. 134, 13796-516 (May 1964)

"Energy and Time Dependence of Delayed Gammas from Fission," P. C. Fisher and L. B. Engle, LAMS-2642 (Jan. 1962)

"Reactivity Contributions of Various Materials in Topsy, Godiva, and Jezebel," L. B. Engle, G. E. Hansen, and H. C. Paxton, Nucl. Sci. Eng. 8, 543-569 (Dec. 1960)

NRC FORM 195 U.S. NUCLEAR REGULATORY COMMISSION DOCKET NUMBER 50-346

FILE NUMBER INCIDENT REPORT

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

TO: Mr. James G. Keppler

FROM: Toledo Edison
Toledo, Ohio 43652
Terry D. Murray

DATE OF DOCUMENT 11/14/77

DATE RECEIVED 11/25/77

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DESCRIPTION

ENCLOSURE Consists of Licensee Event Report (RO 50-346/NP-32-77-16) on 09/24/77 concerning a series of event occuring which resulted in depressurization of the primary system from a normal operating pressure of 2150 psi to 900 psi in approximately 8 minutes, and the release of approximately 11,000 gallons of water in the form of steam within the contirment through the pressurizer quench tank rupture disc...

PLANT NAME: DAVIS-BESSE UNIT # 1
jcm 11/25/77

1p

59p

NOTE: IF PERSONNEL EXPOSURE IS INVOLVED SEND DIRECTLY TO KREGER/J. COLLINS

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November 14, 1977

L77-380

FILE: RR.2 (NP-32-77-16)

Docket No. 50-346
License No. NPF-3

Mr. James G. Keppler
Regional Director, Region III
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

Supplement to Reportable Occurrence NP-32-77-16
Davis-Besse Nuclear Power Station Unit 1
Date of Occurrence: September 24, 1977

Enclosed find three copies of Licensee Event Report NP-32-77-16 Supplement, which is being submitted in accordance with Technical Specification 6.9 to provide additional information of the subject occurrence.

Please note this report also satisfies the special 90 day report requirement of Technical Specification 6.9.2 for the Emergency Core Cooling Actuation on September 24, 1977.

Yours truly,

Terry D Murray

Terry D. Murray
Station Superintendent
Davis-Besse Nuclear Power Station

TDM/JRL/ljk

Enclosures

cc: Dr. Ernst Volgenau, Director
Office of Inspection and Enforcement
Encl: 40 copies Supplement

Mr. William G. McDonald, Director
Office of Management
Information and Program Control
Encl: 3 copies Supplement

1. SUMMARY

On September 24, 1977, a series of events occurred at the Davis-Besse Unit 1 which resulted in depressurization of the primary system from a normal operating pressure of 2150 psi to 900 psi in approximately 8 minutes, and the release of approximately 11,000 gallons of water in the form of steam within the containment through the pressurizer quench tank rupture disc.

On the afternoon of Saturday, September 24, 1977, the main turbine was shut down to repair a leak in a pressure sensing connection on a steam line from the turbine governing valves to the turbine inlet. The reactor was being held critical at approximately 9% thermal power.

At 2134 hours, a spurious half trip occurred in the Steam Feedwater Rupture Control System (SFRCS). This caused the startup feedwater valve on the No. 2 steam generator (which is the normal feed path at this power level) to close. Closure of this valve resulted in a low No. 2 steam generator level, which then resulted in a normal full trip of the SFRCS for this condition and initiation of the SFRCS. SFRCS initiation closes both main steam isolation valves and initiates feedwater flow to both steam generators from their individual steam-driven auxiliary feedpumps.

The half trip and resulting full trip of the SFRCS caused a reduction in heat removal from the primary system and a corresponding temperature/pressure rise in the primary system. The pressure rise in the primary system caused the pressurizer power relief valve to lift. This valve then rapidly oscillated closed-to-open approximately nine times and remained in the full open position.

The temperature rise in the primary system caused an increase in the pressurizer level, and the operator manually tripped the reactor on high pressurizer level approximately two minutes after the half trip on the SFRCS occurred.

The pressurizer power relief valve, in the full open position, rapidly reduced the primary system pressure, and a Safety Features Actuation Sys (SFAS) trip occurred at the 1600 psi setpoint of the primary system. The power relief valve discharge goes to the pressurizer quench tank, which became overloaded and overpressurized, and approximately 4½ minutes after reactor trip the rupture disc in this tank relieved due to overpressure, venting the steam into the containment. Approximately 20 minutes after reactor trip, the operators diagnosed the reason for the primary system depressurization as being the power relief valve, and from the control room closed the motorized block valve ahead of the power relief valve, terminating the loss of primary coolant into the containment.

Subsequent operator action using makeup pumps and high pressure injection pumps stabilized the primary system pressure and pressurizer level and a controlled shutdown to cold shutdown conditions followed.

The major physical damage from the incident was to the reflective metal insulation on the lower part of the No. 2 steam generator, which received the jet of steam coming from the pressurizer quench tank. A ventilating duct in the area of the quench tank was dimpled and required straightening. Twenty-three panels of reflective metal insulation required replacement. Entry into the containment was made at 0550 Sunday, September 25, 1977, for cleanup operations.

Another event occurred in the course of this incident that did not contribute materially to the above events, but did result in the No. 2 steam generator going dry. This was the failure of the No. 2 auxiliary feedpump to come up to full speed following the SFRCS trip. This feedpump came up to approximately 2600 rpm and stayed at this level with no flow to the steam generator until approximately 12 minutes after reactor trip, when the operators placed its control in manual and brought it up to full speed (commencing feedwater flow to the steam generator).

The depressurization of the primary system resulted in steam formation in the primary system, but evaluation has shown there was no appreciable boiling in the core. The pressure/temperature transients in the primary system components including the steam generator, reactor coolant pumps and fuel were severe, but analysis and subsequent pump testing has shown that these transients are within the design allowables and that no detrimental effects are to be expected on the primary system, pumps or fuel.

System/component maloperation or failure occurred in three areas: SFRCS (half-trip initiation), pressurizer power relief valve (oscillation and failing in the open position) and auxiliary feedpump (failure to come up to full speed). The causes of these maloperation/failures have been investigated and corrective action taken to prevent recurrence. Additional system/equipment modifications have been completed or initiated, and additional training has been initiated to strengthen the systems intelligence available to the operators and facilitate operator action.

At no time during the sequence of events was there any jeopardy to the health and safety of the public or plant operators, and there was no release of radioactivity to the environment. Activity levels within the containment at no time impeded containment access.

All safety systems performed their design functions in the proper manner. Operator action was timely and proper throughout the sequence of events.

2. EVENT DESCRIPTION

At the time this incident occurred, the reactimeter data logging system was in service which recorded at high speed a number of system parameters that would not have been available on such a time base through normal station instrumentation and records. This information, together with the computer alarm logging, has permitted a very detailed plotting of the transient conditions in the primary and secondary systems keyed to the system, component and operator actions. This data is plotted on four Figures in Exhibit B. Figure 1 is an 11-minute plot of primary system parameters from one (1) minute prior to event initiation (SFRCS half trip). Figure 2 is a 130-minute plot of three primary system parameters. Figures 3 and 4 are 95-minute plots of pressure and temperature for steam generators No. 1 and No. 2 respectively.

The event started at time 21:34:20 (T = 0) on September 24, 1977. The plant was in Mode 1 with Power (MWT) = 263. The turbine had been shutdown earlier in the evening to repair a leak in the main steam line at an instrument connection between the turbine stop valves and the high pressure turbine. At this time a half trip of the Steam and Feedwater Rupture Control System (SFRCS) was initiated by an unknown cause. The trip closed the startup feedwater valve to No. 2 steam generator and stopped all feedwater to this generator (at this low power level the main feedwater block valve is closed, isolating the main feedwater control valve). The low level alarm was reached in No. 2 steam generator at T = 24 sec. Before the operator could identify and correct the problem, this low level in No. 2 steam generator correctly produced a full trip of the SFRCS. This trip closed the main steam isolation valves and feedwater isolation valves in both steam generators (T = 58 sec.). SFRCS initiation also started both auxiliary feedwater pumps. The number one pump performed as intended, however, number two auxiliary feedwater pump only came up to 2600 rpm, insufficient to feed its steam generator (No. 2).

The loss of feedwater, first to one and then both steam generators, caused an increase in reactor coolant temperature, which resulted in an increase in pressurizer level and reactor coolant system pressure. At 2255 PSIG the pressurizer electromatic relief valve received an open signal. During the next 40 seconds, it received open and close signals, cycled close-to-open nine times and then remained open. This provided a continuous vent path from the pressurizer to the quench tank. When pressurizer level rose to 290", the operator manually tripped the reactor (T = 1 min. 47 sec.). Energy escaping through the electromatic relief valve and main steam relief valves caused a rapid cooldown and depressurization of the reactor coolant system. Reactor coolant system pressure dropped to 1600 PSIG (T = 2 min. 51 sec.) initiating the Safety Features Actuation System (SFAS). This started the high pressure injection pumps and closed certain containment isolation valves.

With the electromatic relief valve still open, the quench tank rupture disc ruptured (T = 6 min.), relieving steam into the containment.

When the reactor coolant system pressure decayed to approximately 1500 psig full high pressure injection flow was established and started to raise pressurizer level. At T = 6 min. 12 sec. the operator stopped the high pressure injection pumps. (The operators had been heavily involved before this time in regaining seal injection flow to the reactor coolant pumps which had been stopped by the SFAS actuation. By T = 5 min. 20 sec. the appropriate SFAS signals had been overridden and normal flows restored to the seals of the pumps). Reactor coolant system pressure continued to decrease until saturation pressure was reached and steam began to form in the reactor coolant system (approximately T = 8 min). This caused an insurge of water into the pressurizer and the pressurizer level went off scale at 320 inches. During this level increase the operator, seeing average reactor coolant system temperature and pressurizer level increasing, stopped one reactor coolant pump in each loop (T = 9 min.) to reduce the heat input into the reactor coolant system.

Due to decreasing pressure in No. 2 steam generator, the SFRCS system gave a low pressure block permit signal at T = 14 min. 13 sec. This alerted the operator to the low level and feed condition of No. 2 steam generator. He blocked the low pressure trip (T = 15 min. 18 sec.), took manual control of the speed of No. 2 auxiliary feedwater pump, which commenced full feedwater flow to No. 2 steam generator (T = 16 min.). The operator saw the rapid addition of cold feedwater into No. 2 steam generator was dropping the reactor coolant system temperature and reduced the feedwater addition to this generator.

At approximately T = 21 min., it was determined that the power relief valve was remaining open and the block valve was closed, isolating the power relief valve on the pressurizer and stopping the venting of the reactor coolant system to the quench tank. At T = 31 min., pressurizer level came back on scale. At T = 41 min. the operator started a second makeup pump to try and stop the pressurizer level decrease. This additional cold water started the reactor coolant system on a slow decreasing temperature transient. At T = 43 min., pressurizer level reached the low level interlock and cut off the pressurizer heaters. At T = 49 min. the operator started a high pressure injection pump to try and stop the decreasing pressurizer level.

OUT HPI BACK ON

The level and pressure in No. 2 steam generator again decreased to the point where the SFRCS gave a low pressure block permit signal. The operator again blocked the trip and, through manual speed control of its auxiliary feedwater pump, restored level and pressure in No. 2 steam generator (T = 51 min.)

With pressurizer ... :ll .. .s .. to recovering, the operator stopped the high ... re injection pump (T = 53 min. 24 sec.). At T = 57 min. he r... reactor coolant makeup flow to normal. This stopped the slow ... ng reactor c..lant temperature transient which started at ... in. All pla.t parameters were now fully under control and ... ant was brought to a steady state condition, and a normal pla... own started.

3. SYSTEM-EQUIPMENT ... ON

A. General

There were t... ems/components where maloperation or failure occu... the event. These are:

1. Steam Fee... ... Control System - SFRCS (half-trip initiatio...

2. Power Rel... ... (oscillat... and failing in the open position)

3. Auxiliary ... ailure ... come up to full speed)

The SFRCS is a ... y system designed to provide feedwater to the steam gene... for removal of decay heat from the primary system under a... of hypothesized plant operating conditions.

These hypothes... ditions in... de loss of normal feedwater flow, steam li... and feed... r line breaks. The components of this ... include sens... systems, logic and initiation systems, main ... isolation val..., steam turbine-driven auxiliary feed... ps, feed... isolation valves, auxiliary steam and feed ... supply valves and cross connect valves. A description of ... system is co... ed in Exhibit C of this report.

A half trip of ... initiat... his event by closing the startup feedwa... to the ... steam generator, which resulted in a ... p due to l... team generator level. This spurious or i... half tri... d possible reasons for it occurring, ar... ed in more ... ail below.

The pressuriz... relief va... a 2½" pilot-actuated relief valve ... to the ... the pressurizer with a motor-operat... ion or bloc... e located in the line immediately ... a relief ... The purpose of this power relief ... to provid... ns of relieving pressurizer pres... requiring ... tion of the spring-loaded ASME Code ... s.

During this event, the power-operated relief valve opened, oscillated closed-to-open and then failed to close and remained in the open position. Operator action from the control room closed the isolation valve ahead of the power relief valve about 20 minutes after reactor trip.

The reasons for the oscillations and the failure of the power relief valve to close are discussed in more detail below.

The steam turbine-driven auxiliary feedwater pumps are a part of the SFRCS. Upon initiation of the SFRCS, the auxiliary steam supply valve to the feedwater pump turbine opened as called for. The No. 2 auxiliary feedwater pump turbine came up to 2600 rpm and remained at this speed rather than continuing up to 3600 rpm, which is the design speed. Operator action at 14 minutes after reactor trip brought this pump up to design speed by placing the control (in the control room) in manual. Failure of this pump to come up to speed did not materially contribute to this event, but did result in the No. 2 steam generator boiling dry, which added to the transient condition in the primary system.

The reasons for this feedwater pump turbine to come up to speed are discussed in detail below.

B. SFRCS

The initiating event was a Steam and Feedwater Rupture Control System (SFRCS) Channel 2 momentary one-half trip from an unknown cause that went back to normal before the station computer could record the source. This one-half trip caused the following events:

1. The startup feedwater control valve (SP7A) on steam generator No. 2 closed. This caused a loss of feedwater incident on steam generator No. 2.

2. A one-half trip on Channel 2 sealed in on both main steam line isolation valves (MSIV). This one-half trip deenergized at least one solenoid valve on each MSIV, and resulted in a "Mn Stm Iso 1 (2) Trbl" alarm on the station computer for both MSIV's.

   This momentary one-half trip could have been caused by a spurious contact opening or a loose connection in a wire in a SFRCS input signal from a steam generator low pressure switch, a steam generator low level bistable or a main feedwater high pressure differential switch. The momentary one-half trip could also have been caused by trouble internal to the SFRCS cabinets. All possible causes were investigated. As a result of this investigation, it was determined that an input buffer card had failed.

C. Auxiliary Feedpump Turbine Governor

The auxiliary feedpump No. 2 failed to accelerate to the normal speed of 3600 rpm. The steam isolation valve opened properly and the pump came up to about 2600 rpm. The governor, a Woodward Type PG-PL with a speed changer motor driving the manual speed setting knob, was calling for a higher speed (the speed changer motor was turning in the "increase" direction). As required, the governor was left in accordance with procedures with the speed adjustment at the "full speed" position when the pump was shutdown. When the pump was called on to auto-start, with steam generator level below setpoint, the speed changer motor continued to drive, through a slip clutch, in the "increase" direction. However, the speed setting mechanism was already at its mechanical high speed stop applying a binding torque to the "T" bar, a portion of the "feed back" linkage, not allowing it to drop down and allow the piston rod to move down in the increase speed direction. The undesired binding in the feedback linkage gave the governor a false signal that the turbine was at the desired speed. Once the torque was removed, by operator remote manual action, from the "T" bar, the "T" bar dropped down and the auxiliary feed pump turbine proceeded to the high speed stop (3600 rpm).

D. Pressurizer Power Relief Valve

When the reactor coolant system pressure reached the setpoint for the power relief valve, 2255 psig, the valve opened properly. However, there is a seal-in relay which then keeps the valve open until pressure is reduced to a lower "reset" pressure (2205 psig). This seal-in relay that controls the closing of the valve was missing from the circuit. Without the relay, the valve reclosed as soon as pressure decreased below the "open" setpoint. The result was open-close cycles as pressure went above and below the "open" setpoint pressure instead of one or two longer blows to relieve the high pressure down to the "reset" pressure.

After approximately nine open-close cycles the power relief valve remained in the open position. When the valve was disassembled it was found that the pilot valve was stuck in the open position causing the main valve to stay open. The pilot valve was stuck in the open position due to unknown foreign material binding the stem in the guide area of the pilot valve nozzle.

4. SYSTEM TRANSIENTS AND ANALYSIS

A. Transients

During this rapid depressurization event (see section 2 above and Exhibit 3, Figures 7-1 through 7-4), the reactor coolant system pressure dropped from about 2300 psig to about 930 psig in 7½ minutes and gradually recovered to 1800 psig in two hours (see Figure 4-1). During this 7½ minutes the reactor coolant outlet temperature dropped at varying rates from about 580 F to about 533 F. Approximately 30 minutes after this initial temperature change, a second slower and smaller temperature change from 540 F to 505 F occurred over a 21-minute period. Following this second temperature decrease, the temperature gradually increased over a 2-hour period to 528 F. The reactor coolant inlet temperature changes and durations were similar to those of the reactor coolant outlet temperature (see Figure 4-2).

The secondary side pressure in steam generator No. 1 reached a maximum of 1050 psig and decreased to about 860 psig within 15 minutes (see Figure 4-3). The secondary side pressure in steam generator No. 2 reached a maximum of 980 psig, decreased to 610 psig in 14 minutes, and returned to 860 psig in 2 minutes. Twenty minutes later the pressure in steam generator No. 2 again decreased to 610 psig and gradually recovered over a 2-hour period (see Figure 4-4).

B. Analysis of the Reactor Coolant System

B&W has completed its evaluation of the September 24 transients and has found no harmful short or long-term effects on the reactor coolant system components. For this evaluation it was conservatively assumed that the total temperature decrease occurred at the initial rate. This results in a 49° F decrease in the reactor coolant outlet temperature over a 6-minute period.

The design specification for Davis-Besse Unit 1 required the evaluation of 40 cycles of a rapid depressurization event, which included a decrease in the reactor coolant pressure from 2200 psig to 800 psig, a change in the reactor coolant system average temperature from 563 F to 500 F in 15 minutes, and a decrease in secondary system pressure from 1050 psig to 640 psig.

The major difference between the actual transient and the design transient is the rate of the temperature change in the reactor coolant system. The actual rate of temperature change was twice the rate of the design transient, but the total temperature change was only 78% of that of the design transient. The net result is that the fatigue usage of this one rapid depressurization is about the same as that predicted for one cycle of the design transient.

As a more direct comparison, the transient event identified was analyzed for the reactor vessel shell and compared to the design transient. The results were that the range in thermal radial gradient stress for the actual transient was 5400 psi, and the range of thermal radial gradient stress for the design transient was 6600 psi. This comparison would be representative of other thicknesses throughout the reactor coolant system pressure boundary.

The conclusions of the analysis are:

(1) Stresses in the pressure boundary did not exceed those already calculated on a design basis. This is verified by the actual pressure not exceeding 2500 psig and the thermal transient being less severe than a combination of design transients for a rapid depressurization and a reactor trip.

(2) Fatigue life of the reactor coolant components is not affected if one cycle of the reactor trip design transient and two cycles of a rapid depressurization design transient are considered to be used for this transient. Two cycles of the rapid depressurization transient are necessary because the HPI system was actuated twice during the event and two cycles are necessary to reflect the thermal transient in the high pressure injection nozzle.

The effect of the entire event on the fatigue life of the steam generators can be accounted for by using one cycle of the design transient for rapid depressurization and one cycle of the design transient for loss of feedwater to one generator.

(3) The effect of the change in water level on the pressurizer has a very minor effect on the pressurizer shell stresses. The pressurizer has been previously analyzed for the thermal effect of water-steam interface, and the change of level does not affect that analysis.

(4) No significant thermal shock should occur to the heaters, because the heaters were deactivated due to a low water level sensor and not reactivated until the level recovered.

(5) No dynamic effects were caused by the rapid pressure decrease. No specific analysis was done, but a dynamic response of the shells would require a large pressure change in the order of milliseconds, and the actual change was on the scale of minutes.

The reduced feedwater flow to steam generator No. 2 was not sufficient to maintain a water level during the first five minutes of the event and this steam generator boiled dry. The primary concern with a dry generator is the tube to shell temperature difference. In this event a water level was established before the system cooldown was started, and acceptable tube to shell temperature differences were maintained. This condition is similar to the loss of feedwater design transient, followed by restart of a dry pressurized generator using the auxiliary feedwater system.

The burst rupture disc on the pressurizer quench tank resulted in a stream of steam and water impinging on steam generator No. 2. This stream removed a section of insulation 10' high and 20' wide from the lower shell of the generator and impinged directly on the generator shell. The temperature of the impinging water was assumed to be 212° F. A conservative evaluation of the rapid temperature change in this local region of the vessel shell was performed. The results of this evaluation indicate that this one event used less than 1% of the total fatigue life of the vessel. The predicted fatigue usage factor for the 40-year design life of the vessel in this area was less than 0.10. This jet impingement did not significantly reduce the fatigue life of the steam generator.

The reactor coolant pumps (RCP) experienced the following conditions during the September 24 transient.

All four RC pumps were subjected to the following:

0:00	Reactor trip
1:10	SFAS trip
1:12	Seal return valves shut for 1:16
1:13	Seal injection valves shut for 1:52 All four pumps operated for 1:15 with no seal injection and no seal return flow during the RCS de-pressurization
2:28	Seal return valves open
3:05	Seal injection valves open
6:00	Steam formation Pressure oscillating near P_{SAT} for 30 to 45 minutes
36:07	Total seal injection flow low alarm

Pump 1-1:

7:04	Pump tripped
7:45	Shaft stopped
36:07	About one minute of low seal injection flow (near 2 gpm) Flow imbalance starved seal injection
36:30	Seal return valve shut
1:12:55	Standpipe level high
1:17:07	Standpipe level normal

Pump 2-2:

4:20	High vibration
7:04	Pump tripped
36:07	[illegible]st seal injection for about one minute
36:22	[illegible]l return valve shut for about 40 seconds

Checkout of t[illegible]tor coolant pumps was initiated to assess whether maint[illegible]and/or repair was required as a result of the transient[illegible]

Operational c[illegible]re required to demonstrate that no significant damage h[illegible]red to the pump bearings, shaft and seals. The first se[illegible]tests were performed in Mode 5 due to operational [illegible]s. Later operational checks were performed in Mode 3. [illegible]s to be operated individually for a duration not [illegible]ed ten (10) minutes, providing all defined parameters re[illegible]in estab[illegible]ed limits.

The operation[illegible]ance was as follows:

1. Lift pump[illegible]ed and [illegible] shafts rotated by hand. Torque v[illegible] not to e[illegible]d 200 ft-lbs. A stethoscope was provi[illegible]of any [illegible]ual mechanical noises in seal housing [illegible] (This was satisfactorily completed on 10/3/77).

2. Mode 5 tes[illegible] [illegible]25 psig.

   2.1 Instr[illegible]ter Required[illegible]

      a. [illegible] lower cav[illegible] pressures - all four pumps.

      b. B[illegible]rizontal vibr[illegible]ion probes - all four pumps.

      c. S[illegible] pressure or s[illegible]tion pressure.

      d. V[illegible]be on [illegible]mp.

      e. S[illegible]leakage [illegible]llected and measured during t[illegible]

   2.2 Compu[illegible]

      Print[illegible] special su[illegible] trend for running RCP every[illegible]s.

   2.3 The[illegible] limits we[illegible]t to be exceeded:

      a. [illegible]ration - [illegible] peak to peak.

b. Total standpipe leakage (upper seal leakage) plus seal return should not exceed 0.6 gpm. If, during the test this limit is exceeded, the possibility exists of an open seal. In no case will total seal leakage be allowed to exceed 1.5 gpm. If this limit is exceeded, maintenance will be required before further pump operation.

c. All other normal plant limits and precautions prevail.

2.4 Sequence of Operation:

a. Secure standpipe flush.

b. Establish seal injection in accordance with plant operating procedure.

c. Measure and record standpipe leakage and return flow. Confirm that total leakage limits are not exceeded.

d. Assure communication between control room and personnel stationed at RCP standpipe leakage drain line.

e. Countdown from 10 to 0

Start strip chart recorders at high speed;
Start Reactor Coolant Pump 2-2 in accordance with plant operating procedure.
After approximately 11 sec., reduce strip chart speed.

f. Run pump for two (2) minutes unless any above limits are exceeded.

g. Data assessment by B&W and Byron-Jackson representatives.

h. Following assessment of data, pump may be run for an additional five (5) minutes to allow for venting procedure requirements.

i. Follow above sequence on 2-1, 1-2 and 1-1.

j. Assessment of this data will determine whether any maintenance is required before high pressure operation is allowed.

3. Similar tests were repeated with system pressure at greater than 1300 psig before a final determination on the condition of the pumps was made.

A[illegible] four reactor coolant pumps were run on 10/5/77 with the following results:

<u>RCP 2-2 10/5/77 Run (2 min.)</u>:

Steam pressure 225 psig
2nd Seal cavity pressure 165 psig
3rd Seal cavity pressure 123.9 psig
Horizontal vibration 5 - 7.5 mils
Vertical vibration .25 mils

3rd Seal leakage plus seal return flow < .4gpm

After the 2-minute run, the pump was run for 10 minutes for system venting. About 30 seconds before the pump was shutdown, there was a step increase in vertical vibration to 2.5 mils. The pump was run again on 10/6/77 for 10 minutes to check out this phenemenon. The vertical vibration was again .25 mils until about 5 seconds before shutdown, when it increased to 2.5 mils. To allow a longer run time, 2-1 and 2-2 pumps were run together for 10 minutes, then 2-2 was run alone for 10 minutes. The vertical vibration stayed at .25 mils for the entire run. This was monitored during pump runs during plant heat up. It should be noted that the step increase in vertical vibration was later assessed to be spurious instrument noise as a result of a loose connector on an instrument line. After the connector was tightened, vertical vibration remained less than .25 mils peak-to-peak amplitude.

<u>RCP 2-1</u>.

Steam pressure 225 psig
2nd Seal cavity pressure 132 psig
3rd Seal cavity pressure 70 psig
Horizontal vibration 5 - 7.5 mils

3rd Seal leakage plus return flow < .4 gpm

<u>RCP 1-2</u>

System pressure 225 psig
2nd Seal cavity pressure 40.29 psig
3rd Seal cavity pressure 81.3 psig
Horizontal vibration 5 - 7.5 mils

3rd Seal leakage plus return flow < .4 gpm

<u>RCP 1-1</u>

System pressure 225 psig
2nd Seal cavity pressure 77.98 psig
3rd Seal cavity pressure 89.27 psig
Horizontal vibration 5 - 7.5 mils

3rd Seal leakage plus return flow < .4 gpm

The apparent discrepancy on seal cavity pressures on 1-1 and 1-2 was checked on 10/6/77 by installing pressure gauges at the pressure transmitters. The gauges read as follows:

1-1:

2nd Seal Cavity Pressure	- 184 psig
3rd Seal Cavity Pressure	- 111 psig

1-2:

2nd Seal Cavity Pressure	- 184 psig
3rd Saal Cavity Pressure	- 112 psig

The readings indicate the seals are staging properly.

Based on the above performance, B&W saw no concern which would justify maintenance at the time.

By 10/13/77 all four reactor coolant pumps had been run at a system pressure greater than 1300 psig.

RC Pumps 2-1 and 2-2 have continued to run from the initial cold pump starts. Below is a typical line of data from each pump.

<u>RCP 2-1</u>

System Pressure	- 1650 psig
2nd Seal Cavity Pressure	- 1034 psig
3rd Seal Cavity Pressure	- 500 psig
Horizontal Vibration	- 3 mils

<u>RCP 2-2</u>

System Pressure	- 1650 psig
2nd Seal Cavity Pressure	- 1075 psig
3rd Seal Cavity Pressure	- 588 psig
Horizontal Vibration	- 3.5 mils

<u>RCP 1-1</u>

Steam Pressure	- 1650 psig
2nd Seal Cavity Pressure	- 1056 psig
3rd Seal Cavity Pressure	- 540 psig
Horizontal Vibration	- 4 mils

<u>RCP 1-2</u>

System Pressure	- 1650 psig
2nd Seal Cavity Pressure	- 920 psig
3rd Seal Cavity Pressure	- 520 psig
Horizontal Vibration	- 3 mils

Based on the above data, B&W felt that all four pumps were in good operating condition and require nothing more at this time than periodic monitoring.

B&W has reviewed the results of the operational checks and has concluded that no detectable damage has occurred to the pump components. B&W considers the reactor coolant pumps to be serviceable for sustained full operational conditions with no requirements for maintenance.

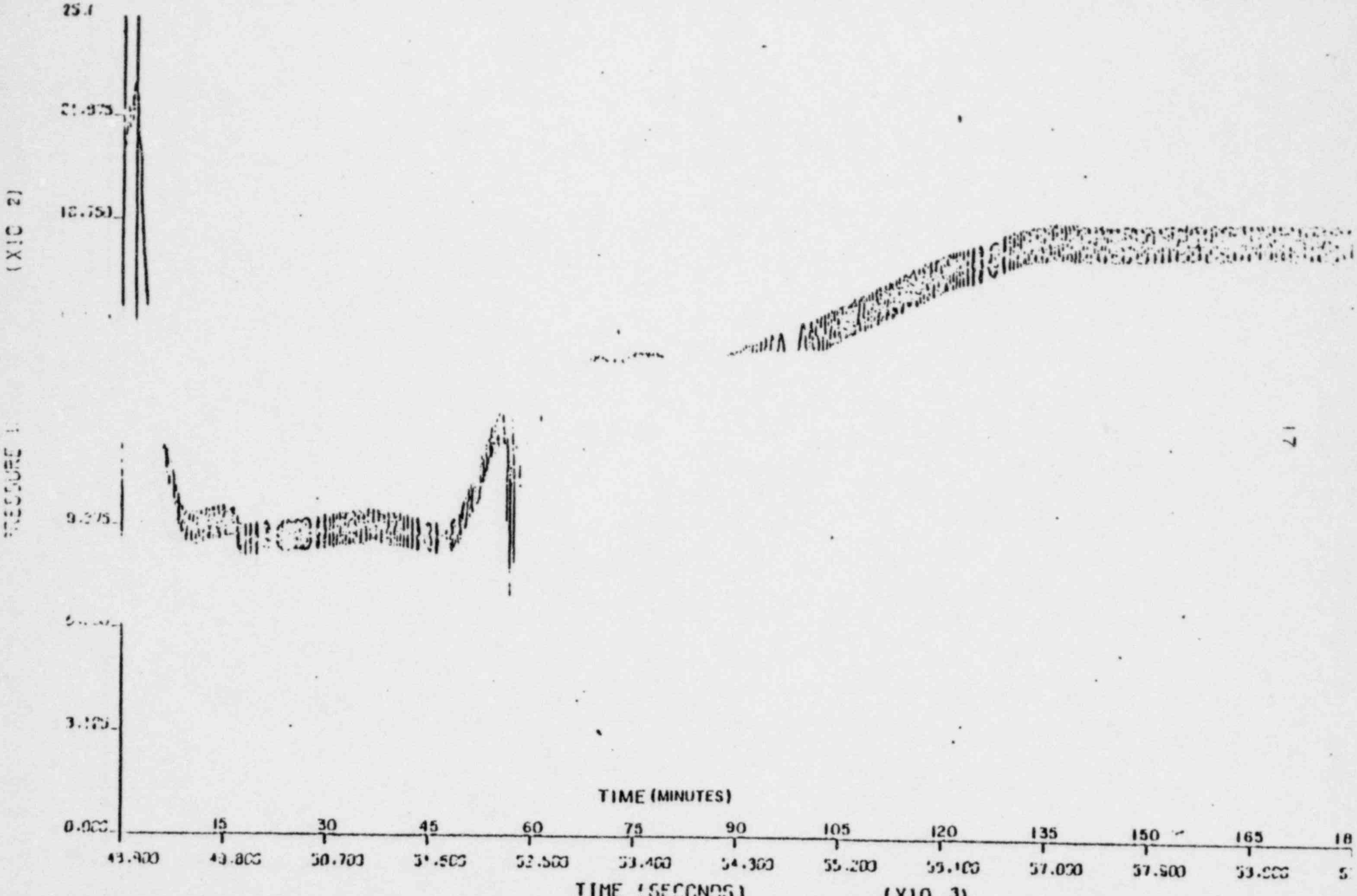
A more detailed analysis was completed to assess the core thermal conditions during the September 24 depressurization event at Davis-Besse Unit 1. Core conditions were analyzed to (1) determine if steam was produced in the core, (2) determine the maximum internal fuel rod pressure during the transient, and (3) determine if maximum lift force exceeded the limit.

Figure 4-5 shows transient thermal conditions as monitored by the reactimeter. The system pressure is measured at the pressure tap, which is approximately 65 feet above the top of the core. The RC pressure at the top of the core is approximately 50 psi higher than the measured pressure because of unrecoverable and elevation pressure losses. As shown in Figure 4-6, the predicted core coolant temperature is slightly higher than the minimum saturation temperature (based upon measured pressure); however, there is some uncertainty in both the measurement and the prediction. Therefore, it is possible that some vapor bubble formation (steam bubbles in water) could have occurred within the core. An examination of the reactimeter data (Figure 4-7) indicates that the RCS pressure level was near the saturation pressure for less than one hour and that during this time period the pressure oscillated with a variation of $\pm$ 50 psi. Therefore, the maximum time period during which the core could have been subjected to bubbly flow was less than one hour. If bubbles were formed during this period, the formation would be in the liquid as well as on the surface, as opposed to formation from a hot surface. With the temperatures, time duration, and type of formation, no significant effect on the components would be predicted.

Prior to the depressurization event the reactor had been operating at 15% power for approximately one week. Immediately prior to reactor trip the power level was 9% of rated power. The core burnup was 1 EFPD, therefore no significant fission gas production had occurred and none was released. During the 60-minute time period in which the indicated RCS pressure was estimated to vary from 900 to 1000 psia at the top of the core, the average coolant temperature was less than 540° F and no significant heat generation occurred in the fuel. An initial evaluation had predicted tensile stresses in the cladding based upon a maximum pressure differential across the cladding of 200 to 300 psi. This evaluation had been based upon a BOL TAFY analysis with an arbitrary safety factor added to ensure that actual conditions would be bounded by the prediction. A more recent analysis, again using TAFY, has resulted in a predicted maximum internal fuel rod pressure of 1000 psia. This analysis considered as-built fuel properties and hot, near zero power conditions at a coolant average temperature of 540° F. On the basis of this analysis it is concluded that the fuel rod cladding was not subjected to any significant level of tensile stress during the subject depressurization event.

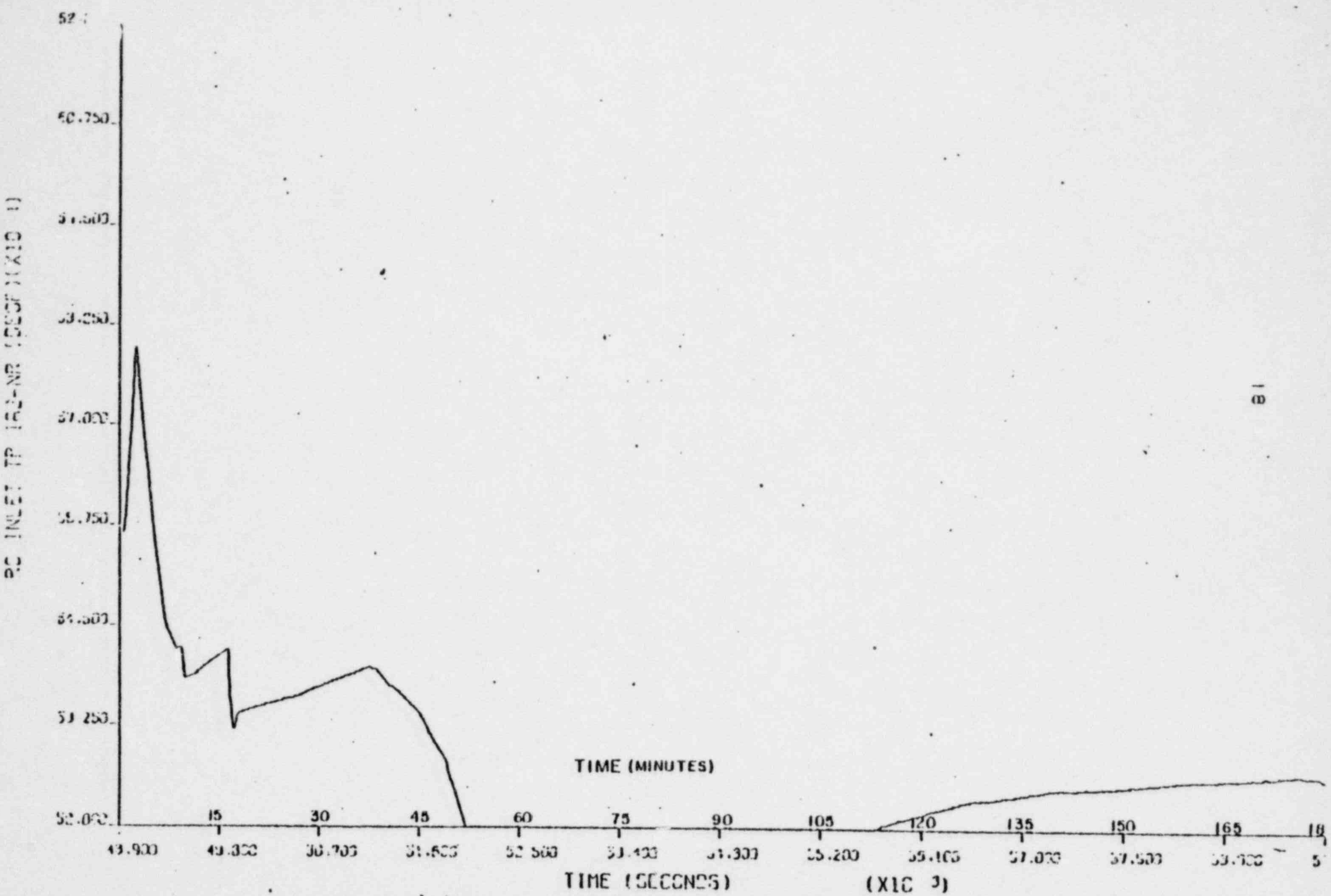
Because the cladding was not subjected to a large, long term tensile stress, no significant long term effects on the cladding resulted. The tensile stresses which could have occurred would have little effect on the cladding due to the small stress level and the short duration of the tensile stress.

Assuming a coolant temperature of 537 F and 150 X 10^6 lb/min system flow (per Figures 4-8 and 4-9), the net lift force will be less than 375 lb. The maximum allowable lift force is 472 lb. Therefore, we conclude that fuel assembly lift-off did not occur.



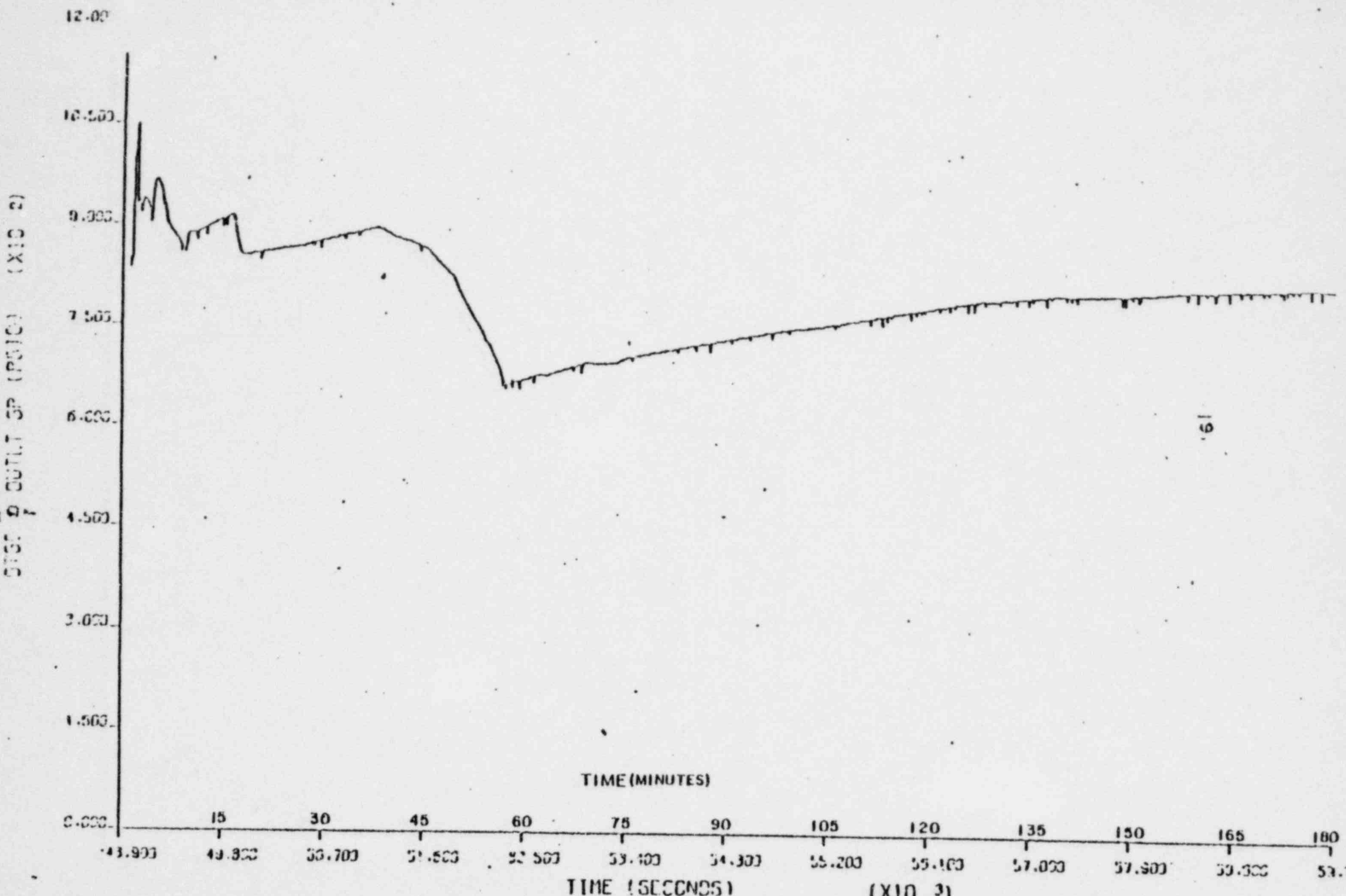
REACTIMETER PLOT TSN=71

FIGURE 4-1



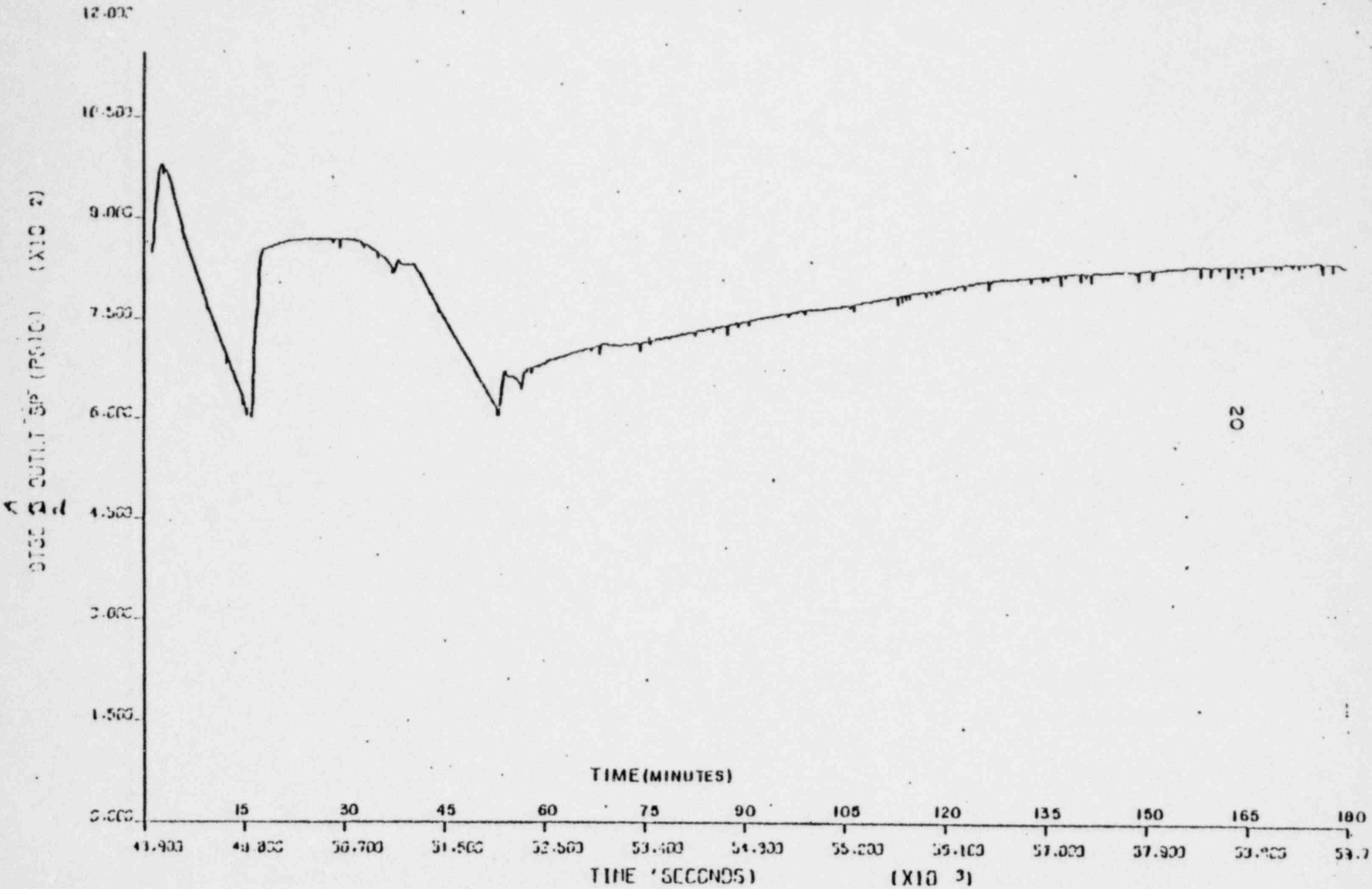
REACTIMETER PLOT TSN=71

FIGURE 4-2



REACTIMETER PLOT TSN=71

FIGURE 4-3



REACTIMETER PLOT TSN=71

FIGURE 4-4

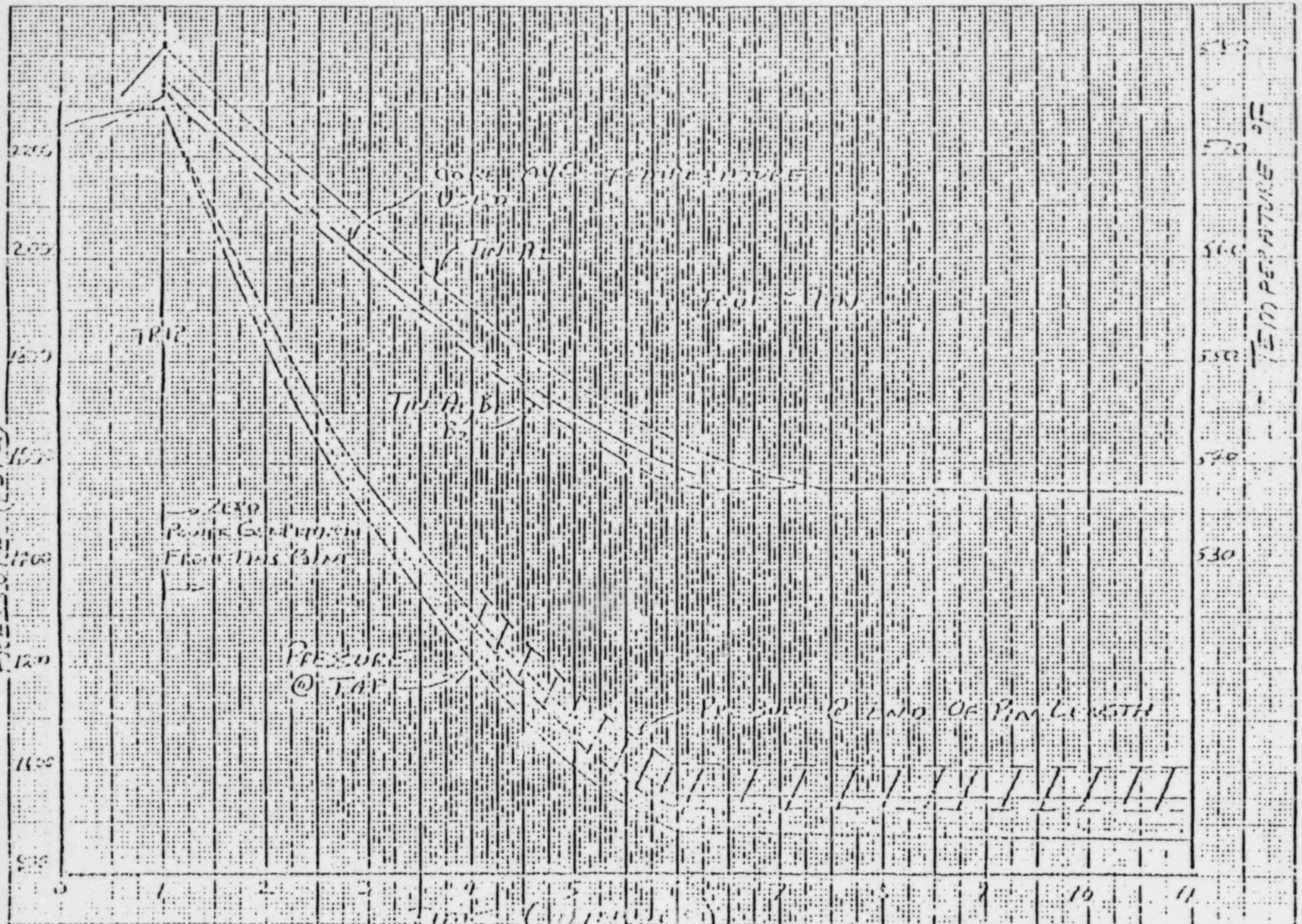
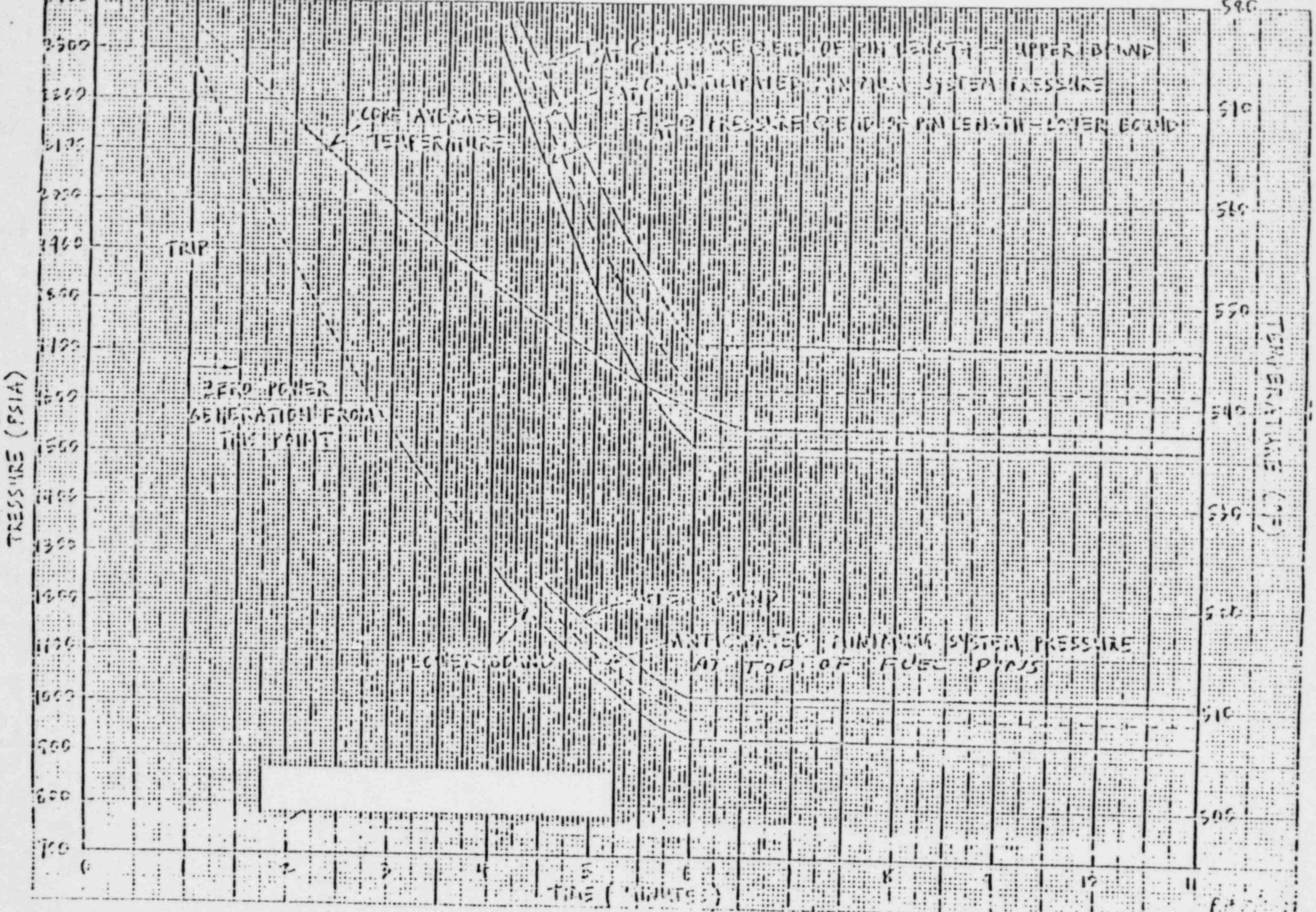
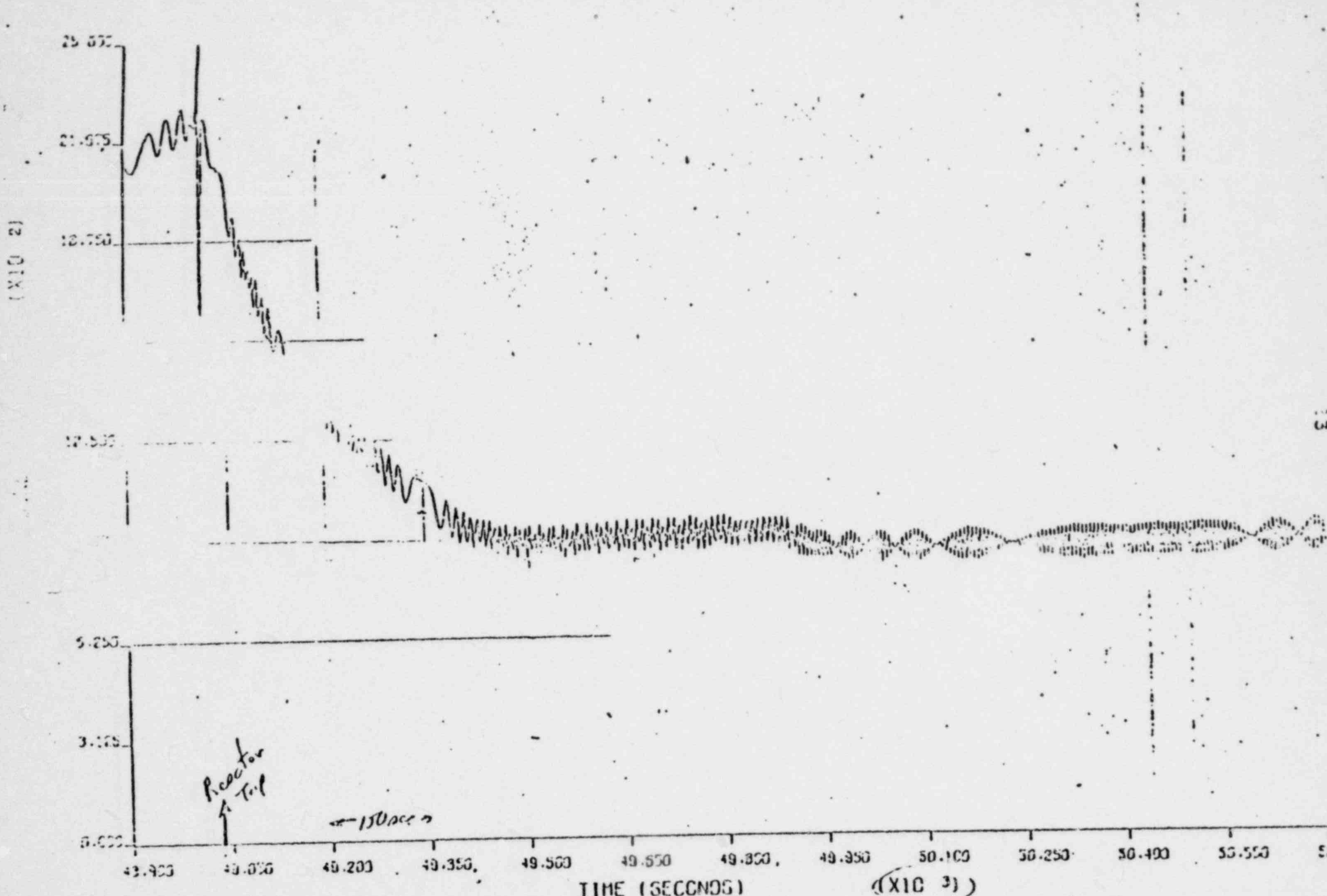

PRESSURE (PSIA)
TEMPERATURE °F
TRIP
PRESSURE @ TAP
530
540
0
1
2
10
11

FIGURE 4-5



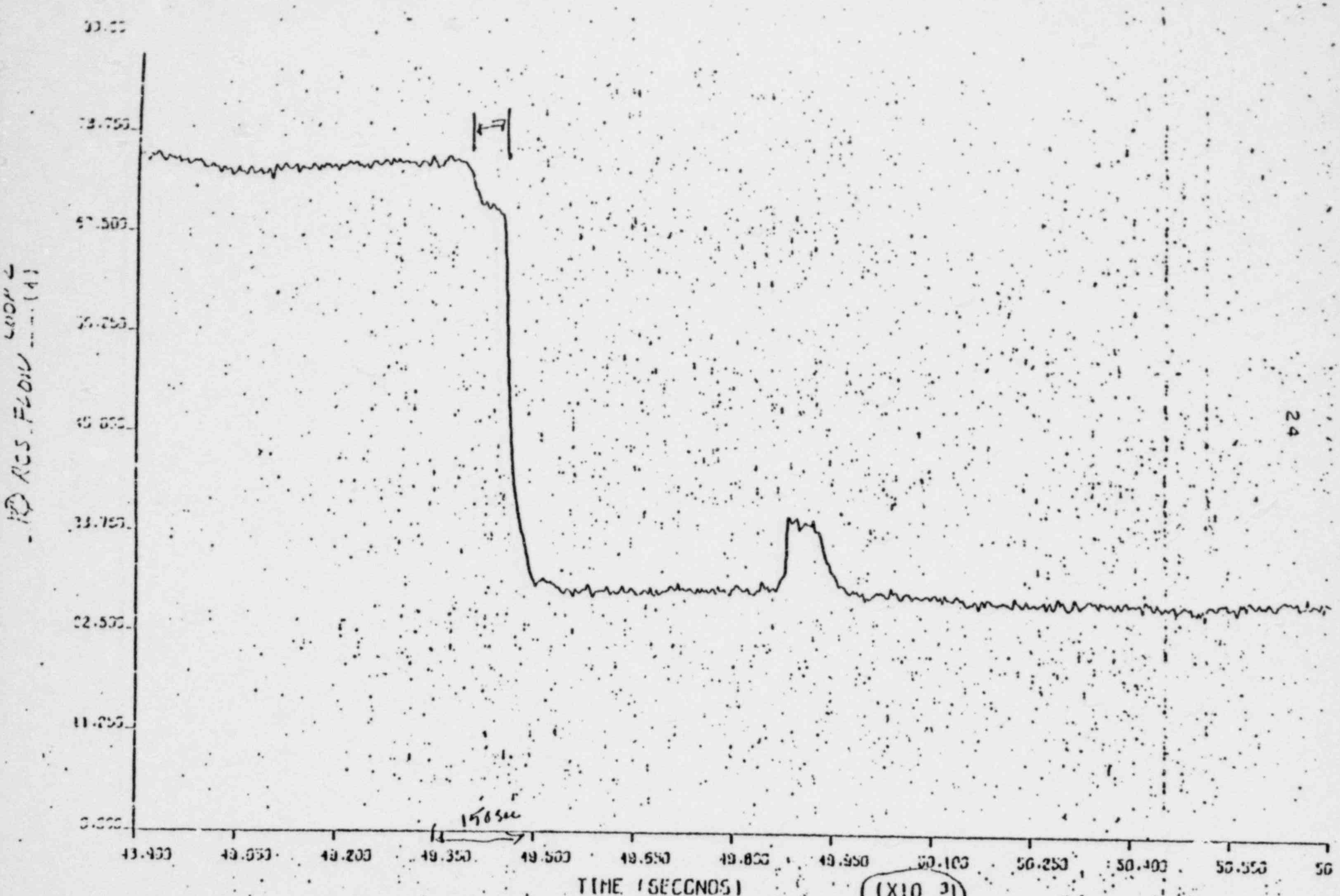
TRIP
CORE AVERAGE TEMPERATURE
ZERO POWER GENERATION FROM TRIP POINT
UPPER BOUND
ANTICIPATED MINIMUM SYSTEM PRESSURE AT TOP OF FUEL PINS
LOWER BOUND
PRESSURE (PSIA)
TEMPERATURE (°F)

FIGURE 4-6



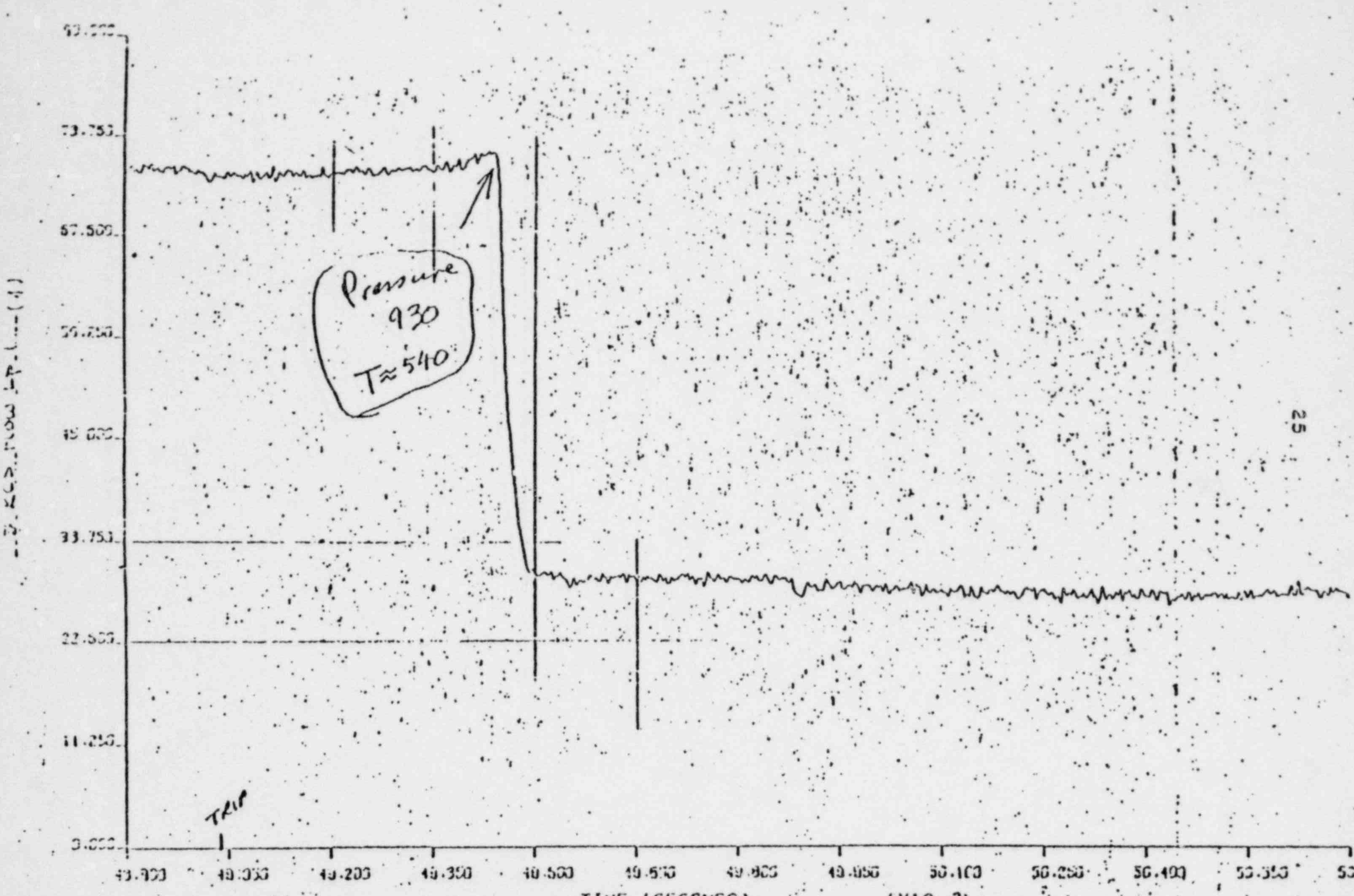
REACTIMETER PLOT TSN=71

FIGURE 4-7



REACTIMETER PLOT TSN=71

FIGURE 4-8



REACTIMETER PLOT TSN=71

FIGURE 4-9

5. EQUIPMENT DAMAGE, CLEANUP AND REPAIR

A. Entry and Cleanup

Prior to entering containment, air samples were collected at RE5030 (containment air monitor) for radioactive noble gases, particulates, iodines, and tritium; no airborne radioactive materials were detected. When containment was first entered at 0550 on September 25, 1977, to determine the levels of contamination, dirt was found on the walkways on elevation 565' and 585' in the east side of containment, and on 545' elevation the floor was completely covered with dirt which was washed down during the period when steam was being released from the quench tank and condensing on containment structures. The dirt was contaminated with activation products of Cr-51, W-187, Co-58, Zn-97, and Na-24 which were present in the reactor coolant system. Smears of the dirt indicated levels up to 40,000 dpm/100cm^2.

Decontamination was accomplished by shoveling gross amounts of dirt into drums, and vacuum sweeping the remainder. The level of contamination in walkways was reduced to meet clean area limits. Air samples collected during the decontamination work verified that contamination did not become airborne.

The outer surface of steam generator 1-2 was inspected in the region where the metal reflective insulation was blown off. Boric [illegible]s were observed on the outer surface of steam genera[illegible]-2; however, these minute quantities do not present any concern since the temperature of these surfaces are on the order of 500° F.

B. Equipment Damage

The pressurizer quench tank rupture disc ruptured from high pressure in the quench tank. The steam from the pressurizer quench tank vent damaged metal reflective insulation on the lower part of No. 2 steam generator. A ventilating duct above the quench tank was bent, and a ventilation louver had to be replaced. Several pressurizer heater cables were dampened from the moisture, causing low insulation resistance, and had to be dried out. Four cables were also found shorted to ground, but it is not known if the failures were a direct result of the incident. Two light fixtures and a combustion detector sensor in the quench tank area were also damaged.

Twenty-three (23) panels of reflective insulation were deformed, loosened or detached from the lower exterior of the steam generator. The panels, fabricated from thin stainless steel sheets with air spaces between them, are approximately 36" x 30" x 4".

The panels are formed to the contour of the steam generator and attached to the exterior on a frame to support the weight. Buckles and clips fasten the panels together. Panels blown frow the steam generator fell to the floor, piping and ventilation duct in the immediate vicinity. Some panels were repaired and reused; others had to be replaced. The damaged panels were intact but were bent.

C. Repairs

All damaged equipment was repaired or replaced. Instrumentation and equipment in the area was checked or tested for possible damage from the steam and water.

Twenty-three (23) panels of reflective insulation were replaced. The other effected panels were straightened, repositioned and reinstalled on the steam generator.

All essential and automatically-controlled pressurizer heaters were returned to service. The wet pressurizer heater cables were baked, heated or air dried to restore insulation resistance to vendor recommended values. Only two of the four cables shorted to ground were replaced with spares. The other two are on order.

A new rupture disc was installed on the pressurizer quench tank.

The deformed ventilation duct was straightened and a new louver was installed in the duct.

The damaged light fixture and combustion detector were replaced.

6. SYSTEMS/EQUIPMENT MODIFICATION AND TESTING

A. SFRCS

The Davis-Besse Instrument and Control (I&C) Group has tested logic channels 2 and 4 (channel 2) of the SFRCS, since it was indicated that the closure of SP7A (start-up feedwater valves) led to the sequence of events on 9/24/77. Logic channels 2 and 4 are the only SFRCS channels that actuate SP7A.

On 9/26/77, Maintenance Work Order (MWO) IC-622-77 was written to check the main steam line pressure switches PS 3687A through PS 3687H. A calibration check was completed on 9/27/77. All pressure switches actuated within ± 2 psig of the 612 psig setpoints. I&C personnel had nothing to report from the visual inspection.

On 9/27/77, MWO IC-636-77 was written to investigate the remaining inputs to the SFRCS. Pressure differential switches 2686C, 2686D, 2685A and 2685B were tested per ST 5031.14, Section 6.3. The setpoint of the pressure differential switches tested ranged from 176 psig to 187 psig, the setpoint being 177 ± 20 psig.

The steam generator level inputs to the SFRCS were tested per ST 5031.14, Section 6.4. Again, logic channels 2 and 4 were tested. All bistables tripped at the desired setpoints. The desired trip setting is .509 $\pm$.013 volts and the range of voltages for the bistables tested were from .5054 volts to .509 volts. In addition, the level transmitter calibration was checked per ST 5031.16. I&C tested for any non-linearities between transmitter input and output, especially at the lower ranges. LT-SP9A8, LT-SP9A9, LT-SP9B6, and LT-SP9B7 were well within the acceptable limits as specified by ST 5031.16 and no non-linearities were observed.

The inputs to the SFRCS from the loss of 4 reactor coolant pumps were not tested since this input actuates auxiliary feedwater only. This input does not affect the feedwater valves or main steam isolation valves.

In addition to testing all the input devices, I&C checked C5792. This is the cabinet for logic channels 2 and 4. All inputs and outputs were normal for existing plant conditions. I&C checked mechanical connections on the input and output buffers, and induced mechanical vibration on the input buffers, output buffers, main logic panels, and output relays without any system effect. The main logic panels were heated slightly with a heat lamp and slowly cooled to check for thermal variations, but this had no effect on the system.

On September 29, I&C cc   d th'ir check of SFRCS terminations. The following are the res.l   that check:

1. Screws on TB37 (y   blue) were tightened 1/2 of a turn. This is an input to th   on 15 volt logic supply for C5792.

2. In C5721 (Feedwat   one loose screw was found on 21TB11 Terminal 17 (left   f TB). This screw required little movement to thoroughly tig.   This is a main steam pressure switch input to logic channel [illegible]

3. In C5721, 21TB27 [illegible]   se screws. Terminal 17 (right side of Terminal Board) h   tightened 1 full turn. This is a main steam pressure sw   put to logic channel 3.

   Terminal 18 (righ   Terminal B rd) had to be tightened 1/2 a turn. This is   ure differential switch input to the SFRCS. Logic channel 3 T   left si   TB) had to be tightened slightly. This is   steam press e switch input to the SFRCS.

On September 30, HIS 4[illegible]   C were tightened to their mounting. These are stacked switc   switch u   themselves were secure, but the entire package   n the m   ing. This switch unit being loose would proba   affe   system operation. Temporary jumpers were installed   vent an inadvertent main steam isolation valve closure during ST   heckout.

On October 6, 1977, the   erator l   l instrumentation was checked out. I&C was specifica.   king for noi   spikes that could have caused an erroneous tri   analog inp   and outputs only had a 20 MV (typical) AC noise. DC   appeared   an".

On October 8, 1977, eig   nnel chart   orders were patched into the system for continuc   toring. The   corders were connected per the attached sheets. T   stem was then checked out for operability. Pressure differential a   nerato   level trips were tested. Since the SFRCS was blo   low s   pressure, pressure switch trips were initiated and   to th   was verified by a voltmeter reading. These pressure   inputs w   further tested during the SFRCS monthly, ST 5031.   ion 6.2, a   later date. Connecting the recorders has indicated   fect on system operability.

LOGIC CHANNEL 1 SFRCS TEST CONNECTIONS

INPUT	CONNECT COMMON TO BUFFER	CONNECT COMMON TO TEST POINT	CONNECT SIGNAL LEAD TO BUFFER	CONNECT SIGNAL LEAD TO TEST POINT	RECORDER	CHANNEL
'S '9A	1-1	TP4	1-1	TP5	9.1.48	1
'S3689B	2-1	TP2	1-1	TP7	9.1.48	2
'S3689C	2-2	TP2	1-1	TP9	9.1.46	9
'S3689D	1-2	TP4	1-2	TP5	9.1.48	4
'D2686A	1-3	TP4	1-3	TP7	9.1.48	5
'D2685C	2-3	TP2	1-3	TP9	9.1.48	6
.T SP9B8	1-4	TP4	1-4	TP5	9.1.46	7
.T SP9A6	2-4	TP2	1-4	TP7	9.1.46	8
.5 V. Power Supply Output	1-5	TP2	1-5	TP10	9.1.48	3
'681	2-7	TP2	2-7	TP10	9.1.46	10
'b.	2-6	TP2	2-7	TP16	9.1.46	11
.886	1-6	TP2	1-6	TP10	9.1.49	12

LOGIC CHANNEL 2 SFRCS TEST CONNECTIONS

INPUT	CONNECT COMMON TO BUFFER	CONNECT COMMON TO TEST POINT	CONNECT SIGNAL LEAD TO BUFFER	CONNECT SIGNAL LEAD TO TEST POINT	RECORDER	CHANNEL
S[illegible]A	1-1	TP4	1-1	TP5	9.1.44	1
S3687B	2-1	TP2	1-1	TP7	9.1.44	2
S3687C	2-2	TP2	1-1	TP9	9.1.41	9
S3687D	1-2	TP4	1-2	TP5	9.1.44	4
DS2685A	1-3	TP4	1-3	TP7	9.1.44	5
DS2686C	2-3	TP2	1-3	TP9	9.1.44	6
T SP9B6	1-4	TP4	1-4	TP5	9.1.41	7
T SP9A8	2-4	TP2	1-4	TP7	9.1.41	8
5 V. Power upply Output	1-5	TP2	1-5	TP10	9.1.44	3
[illegible]	2-7	TP2	2-7	TP10	9.1.41	10
672	2-6	TP2	2-7	TP16	9.1.41	11
396	1-6	TP2	1-6	TP10	9.1.41	12

LOGIC CHANNEL 3 SFRCS TEST CONNECTIONS

NPUT	CONNECT COMMON TO BUFFER	CONNECT COMMON TO TEST POINT	CONNECT SIGNAL LEAD TO BUFFER	CONNECT SIGNAL LEAD TO TEST POINT	RECORDER	CHANNEL
S_ 9E	1-10	TP4	1-10	TP5	9.1.47	13
S3689F	2-10	TP2	1-10	TP7	9.1.47	14
S3689G	2-11	TP2	1-10	TP9	9.1.45	21
S3689H	1-11	TP4	1-11	TP5	9.1.47	16
DS2686B	1-12	TP4	1-12	TP7	9.1.47	17
DS2685D	2-12	TP2	1-12	TP9	9.1.47	18
T SP9B9	1-14	TP4	1-13	TP5	9.1.45	19
.T SP9A7	2-13	TP2	1-13	TP7	9.1.45	20
5 V. Power Supply Output	1-14	TP2	1-14	TP10	9.1.47	15
68'	2-16	TP2	2-16	TP10	9.1.45	22
'671	2-16	TP2	2-16	TP16	9.1.45	23
.386	1-15	TP2	1-15	TP10	9.1.45	24

LOGIC CHANNEL 4 SFRCS TEST CONNECTIONS

INPUT	CONNECT COMMON TO BUFFER	CONNECT COMMON TO TEST POINT	CONNECT SIGNAL LEAD TO BUFFER	CONNECT SIGNAL LEAD TO TEST POINT	RECORDER	CHANNEL
PS. 7E	1-10	TP4	1-10	TP5	9.1.43	13
PS3687F	2-10	TP2	1-10	TP7	9.1.43	14
PS3687G	2-11	TP2	1-10	TP9	060404	21
PS3687H	1-11	TP4	1-11	TP5	9.1.43	16
PDS2685B	1-12	TP4	1-12	TP7	9.1.43	17
PDS2686D	2-12	TP2	1-12	TP9	9.1.43	18
LT SP9B7	1-13	TP4	1-13	TP5	060404	19
LT SP9A9	2-13	TP2	1-13	TP7	060404	20
15 V. Power Supply Output	1-14	TP2	1-14	TP10	9.1.43	15
P[illegible]	2-16	TP2	2-16	TP10	060404	22
P672	2-15	TP2	2-16	TP16	060404	23
[illegible]396	1-15	TP2	1-15	TP10	060404	24

On 10/23/77, the SFRCS again tripped from a spurious signal. The Startup Feedwater Valve on steam generator No. 2 went closed. This ultimately resulted in a valid Steam Generator low level trip input to the SFRCS and the system functioned as intended.

This was the first spurious trip received since the chart recorders had been connected to the SFRCS. All information on the charts could be explained except for a problem on SFRCS logic Channel 4 computer alarm, P680. This particular channel on the recorder was intermittently failing, giving spurious trip indications. Of the 48 total chart recorder channels, this was the only one that had failed.

I&C Technicians "checked out" the bad recorder channel for operation. They found that the channel was sensitive to any mechanical vibration, it did respond to a given input, and that the pens were slightly misaligned. From all of the information gathered it was concluded that the indication on the bad recorder channel was an input from the SFRCS.

The logic point under question then was the computer point ("P680" Low Main Steam Pressure Trip). Examining other charts indicated no change in the input to SFRCS logic Channel 4. Thus it was concluded the problem was internal to the system. In examining the logic control diagram, it was determined 3 IC "chips", 2 input buffers and associated wiring could have caused the fault. I&C personnel replaced all of the above equipment, with the exception of the interconnecting wiring. The wiring and buffer connections were visually inspected, and no faults were observed. A functional logic test was performed and the system responded satisfactorily.

Power Engineering had contacted Consolidated Controls Corporation, the manufacturer, and their representative was on site the morning of 10/26/77. The manufacturer also recommended changing the same equipment that TECo I&C personnel had changed.

The manufacturer performed a response time check on both input buffers in question. The response time test showed no defects. TECo I&C personnel continued to monitor one of the two input buffers in a test set. Failure of one input buffer did occur on the test set, which indicates that this was the cause of the half trip.

The manufacturer's representative also took a look at the logic system with an oscilloscope. He was looking for any erratic, noisy points, but everything tested appeared to be trouble free. The two input buffers will remain with TECo for further test and evaluation, while the 3 IC chips were returned to the manufacturer for evaluation.

The manufacturer's representative on 10/27/77 compiled a list of additional points they want monitored. TECo I&C personnel are assisting to connect up the recorders.

After the 10/23/77 even [illegible] study was also conducted to see if any single 120 VAC or 125 [illegible] fault induced voltage dip could have caused the one-half trip on [illegible]'s and closed the SG-2 SU control valve. This study revealed th[illegible] single fault of these power supplies could have caused this probl[illegible].

The following changes [illegible] made to the design of the SFRCS since the September 24, 1977 [illegible]:

1. Annunciator [illegible] have been added where computer alarms presently ex[illegible]

   a. St[illegible]tor Level [illegible]f/Full Trip for both Channels 1 [illegible]

   b. Ma[illegible]ter DP Half/Full Trip for both Channels 1 & [illegible]

   c. Los[illegible]tor Co[illegible] Pump Trip

2. A new annunc[illegible] and computer alarm has been added for a SFRCS Full Tr[illegible]

3. The resetting [illegible]CS rel[illegible] alarm will be delayed long enough to all[illegible] [illegible] the event.

These changes will be m[illegible] soon as possible.

B. Auxiliary Feedpump Turbine Governor

Before describing the modifications made to the auxiliary feedpump turbine (AFPT) governor, the governor action which resulted in the binding will be described. Figure 6-1 is a drawing of the Woodward Governor PG-PL speed setting mechanism, showing the governor in the bound up condition. The sequence of events creating this condition is as follows:

1. When the Bodine motor was at a minimum speed setting, the speed setting shaft nut was fully to the left. The link raised the collar, contacting the base speed setting nut, raising it and the "T"-bar to an idle condition. The pivot bearing would be contacting the floating lever.

2. Because the governor is not rotating, the speed setting servo remains in a fixed position at idle (as shown). It cannot move until oil pressure is available.

3. The thumbscrew is contacting the low speed stop pin.

4. As the Bodine speed setting motor is rotated toward high speed, the following events occur:

   4.1 The speed setting shaft nut moves towards the high speed stop pin.

   4.2 The link allows the collar to move downward.

   4.3 The collar moving downward, allows the base speed setting nut and "T"-bar assembly to move downward.

   4.4 The floating lever is fixed at the speed setting servo piston end.

   4.5 The low speed stop pin end of the link pushes down on the thumbscrew, which pushes down on the speed setting pilot valve until the dashpot land contacts the dashpot plug.

   4.6 Because the floating lever is now fixed on both ends it stops moving.

   4.7 The "T"-bar continues downward, following the collar. The pivot bearing leaves the floating lever. The "T"-bar continues downward until the retainer screw contacts the floating lever.

   4.8 The collar separates from the base speed setting nut and continues downward until the stop pin in the speed shaft contacts the stop pin in the speed setting shaft nut.

4.9 Because the Bodine motor continues to rotate the manual speed setting knob, slipping the clutch, a torque is placed on the speed setting shaft nut, link and collar. This torque against the "T"-bar causes friction that locks the "T"-bar in place.

5. When the turbine is started, the speed setting servo piston moves downward with increasing oil flow, increasing the speed setting of the governor. When the floating level contacts the pivot bearing, the speed setting pilot valve begins to raise.

6. When the pilot valve control land covers the metering port, the speed setting servo piston stops moving.

7. Because the torque is still present on the speed setting shaft, the "T"-bar is bound up, and the governor is at 2200-2600 rpm.

8. When the Bodine speed setting motor is backed off from the stop, the "T"-bar falls down to its high speed stop, dropping the pivot bearing. The pilot valve moves downward, increasing oil flow to the speed setting servo until the high speed condition is reached.

9. Any changes in speed setting shaft position are now normally followed by the "T"-bar, pivot bearing, pilot valve, and speed setting servo piston.

When the AFPT governors arrived at the Woodward Governor Company factory, one of the governors was placed on the test stand. While observing the operation of the speed setting linkage, it became evident that a simple link from the speed setting pilot valve (plunger) to the floating lever would allow removal of the bellows, coupling spring, low speed pin, "C" link and dashpot plug in the speed setting pilot valve sleeve (see Figure 2). This would allow the speed setting pilot valve to overtravel when the motor was set in a high speed condition with the speed setting servo at the minimum position (see Figure 6-3).

The required parts were manufactured, the unneeded parts removed and the governors were reassembled. The governors were tested at the Woodward factory and the tests confirmed that the modifications did remove all possibility of the undesired binding of the governors. Surveillance testing at the station has also confirmed that the auxiliary feedpump turbine governors function properly.

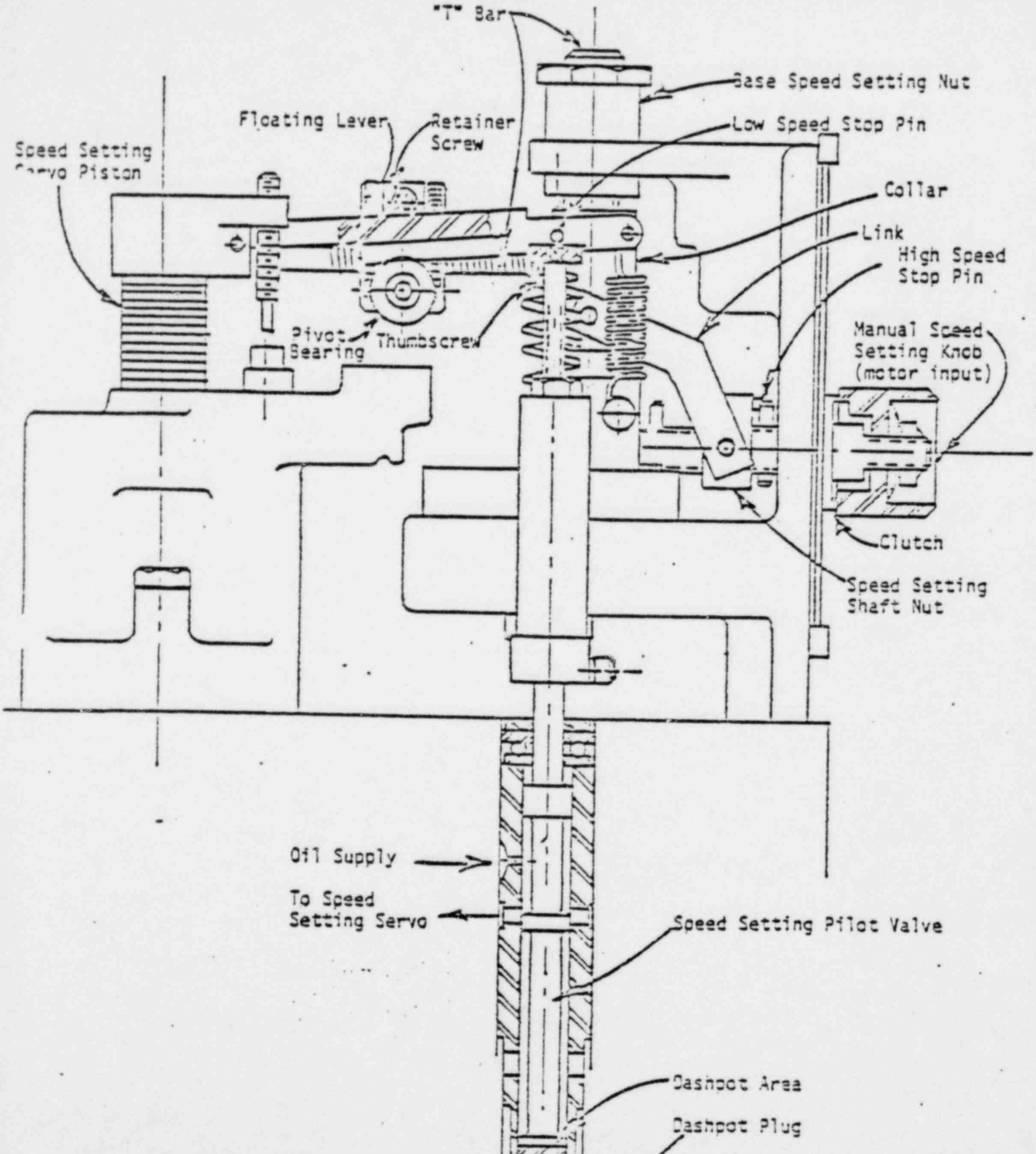

"T" Bar
Base Speed Setting Nut
Floating Lever
Retainer Screw
Low Speed Stop Pin
Speed Setting Servo Piston
Collar
Link
High Speed Stop Pin
Pivot Bearing
Thumbscrew
Manual Speed Setting Knob (motor input)
Clutch
Speed Setting Shaft Nut
Oil Supply
To Speed Setting Servo
Speed Setting Pilot Valve
Dashpot Area
Dashpot Plug

FIGURE 6 - 1

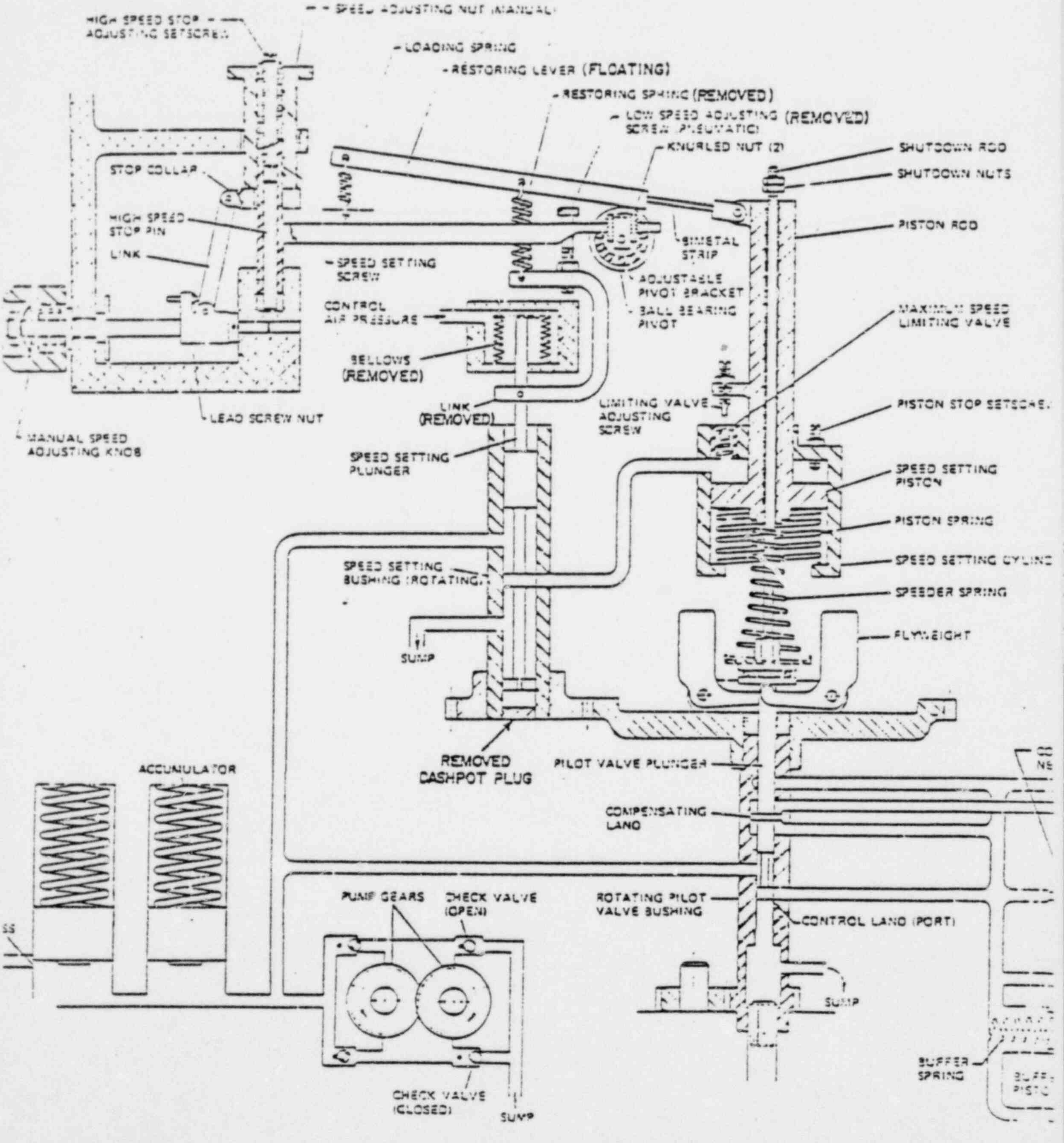
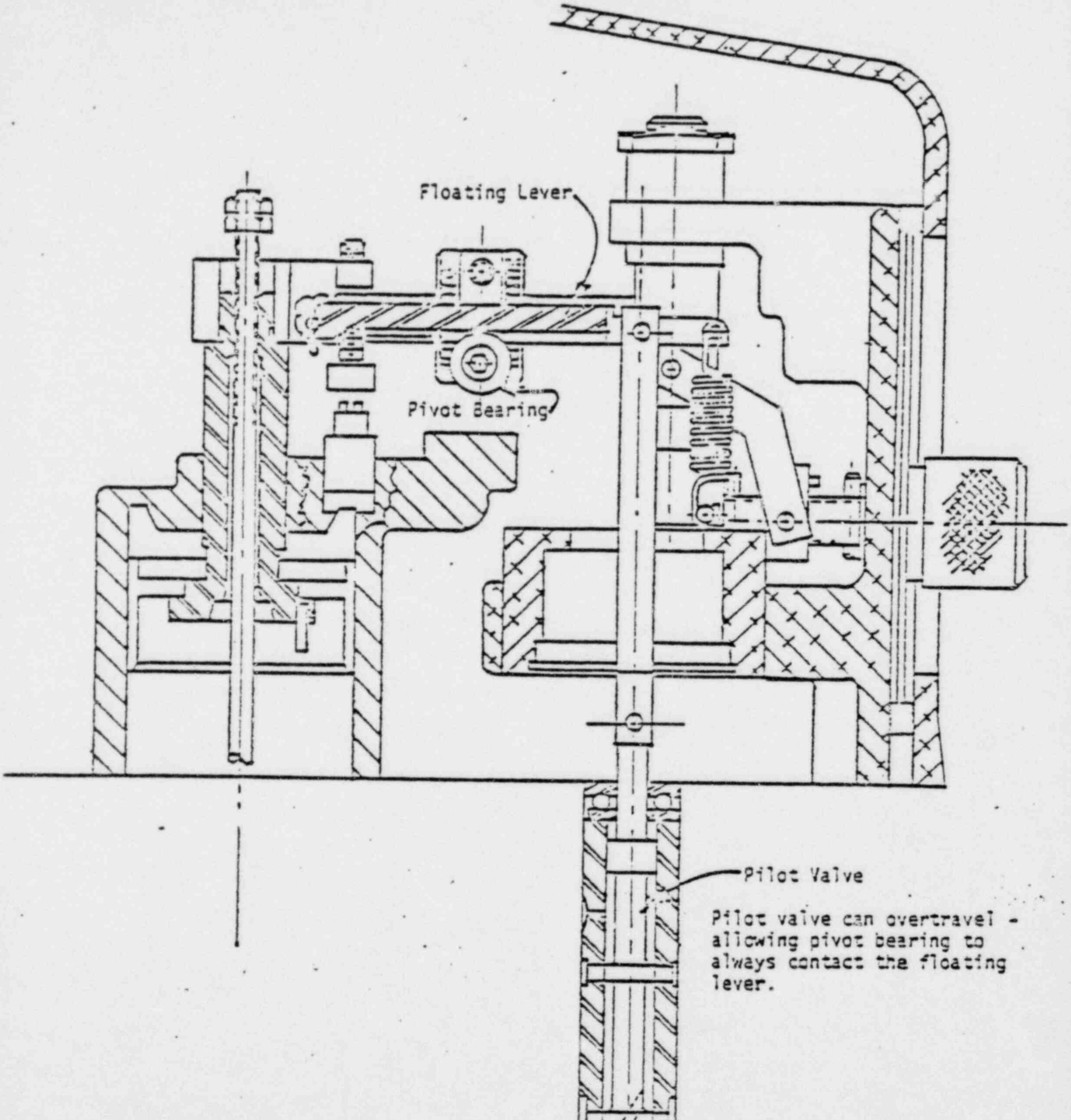

SPEED ADJUSTING NUT (MANUAL)
HIGH SPEED STOP ADJUSTING SETSCREW
LOADING SPRING
RESTORING LEVER (FLOATING)
RESTORING SPRING (REMOVED)
LOW SPEED ADJUSTING (REMOVED) SCREW (PNEUMATIC)
KNURLED NUT (2)
SHUTDOWN ROD
SHUTDOWN NUTS
STOP COLLAR
HIGH SPEED STOP PIN
LINK
SPEED SETTING SCREW
BIMETAL STRIP
PISTON ROD
ADJUSTABLE PIVOT BRACKET
BALL BEARING PIVOT
CONTROL AIR PRESSURE
BELLOWS (REMOVED)
MAXIMUM SPEED LIMITING VALVE
LINK (REMOVED)
LEAD SCREW NUT
MANUAL SPEED ADJUSTING KNOB
LIMITING VALVE ADJUSTING SCREW
PISTON STOP SETSCREW
SPEED SETTING PLUNGER
SPEED SETTING PISTON
PISTON SPRING
SPEED SETTING BUSHING (ROTATING)
SPEED SETTING CYLINDER
SPEEDER SPRING
SUMP
FLYWEIGHT
REMOVED DASHPOT PLUG
PILOT VALVE PLUNGER
ACCUMULATOR
COMPENSATING LAND
PUMP GEARS
CHECK VALVE (OPEN)
ROTATING PILOT VALVE BUSHING
CONTROL LAND (PORT)
SUMP
CHECK VALVE (CLOSED)
SUMP
BUFFER SPRING

FIGURE 6-2



Pilot valve can overtravel - allowing pivot bearing to always contact the floating lever.

FIGURE 6-3

C. Pressur... ...er Relief Valve

On Septe... 1977, the ...ve was completely disassembled. The main ... was found to be clean. The seats on the nozzle and main ...isc were lap...d. The pilot valve was found stuck in ... position ... it was thought that the pilot stem was ... the pilot stem was replaced and the nozzle guide are... cleaned up to remove the marks from the galling of the fo... material. The valve was reassembled and on October 1... the valve ... stroked six (6) times with a pressur... ...essure of approximately 600 psi. During this testing t... valve agai... stuck and the isolation valve had to be...

The val... ...n disassem...d and under closer observation it was f... ...he pilot ...lve stem was moving too far (3/8" vs ... ...esired). It was also found that the clearances between t... ...ed and ... ...zzle guide were too small (.0005" vs ... minimum of .001"). The clearances were opened up ... stroke of the pilot was shortened by adjustmen... ...lenoid posit... . The valve was tested again successfu... ...ing it ... ...lve (12) times on October 15, 1977, at ... ...r press... of approximately 900 psi and one time ... ...of 2... ...si.

D. Relay/Fuse ... Checks

Because of ... ...ing relay ... the pressurizer electromatic relief val... ...rol circuit, ... extensive review program of checking a... ...r relay cab... ...s was performed. All relay cabinets ... were i... ...cted for missing plug-in relays and ...s. A detailed review of drawings was made to determi... ...rvice of ... missing item and its effect on plant c... ... The one ...ditional relay and ten fuses found miss... ...re replaced. There were no essential functions affected by ... ...ional ... ...g relay and fuses. The missing fu... ...ay were ... generator iso phase bus control, a... ...icati... ...lay cabinets power supply and heater ... ...uits; ... feed pump turbine lube oil tank level ... ...on; and ... ...r coolant pump component cooling wa... ...turn valve co... ...ol.

Neither th... relay no... fuses were controlled under the static... ...r and lifted ... ...e control procedure. This indicates ... and rel... ...e removed by unknown persons a... ...ithout and ... ...g.

E. <u>Other Actions</u>

Following this incident a training program was developed and presented. This program was approximately eight (8) hours of instruction and discussion covering the events of this incident, including a detailed coverage of the transient and the actions taken by the operators, and a refresher training session covering the operation of the steam and feedwater rupture control system.

The training was presented to all in the operating shift crews, the management and staff level engineers and the QA/QC staff.

7. EXHIBITS

A. Event Chronology

B. Event Variables Plots

C. SFRCS Description

D. 10 CFR Part 21 Letter on Auxiliary Feedpum Turbine Governor

E. Historical Log

'7A Event Chronology

21:34:20 Startup Feedwater Valve to OTSG #2 went closed on a "½ trip" of the Steam and Feedwater Rupture Control System (SFRCS).

21:35:18 Received a complete SFRCS trip due to low level in OTSG #2.

21:35:23 Main Steam Isolation Valves went closed.

21:35:26-49 Pressurizer Power Relief Valve cycled 9 times before sticking open.

21:36:04 Auxiliary Feed Pump (AFP) #1 was feeding #1 Steam Generator (SG). AFP #2 did not come up to full speed (3600 rpm), and the discharge pressure was not sufficient to feed #2 SG.

21:36:07 Operator tripped the reactor.

21:37:17 Safety Features Actuation System Incident Levels 1 and 2 were initiated due to reactor coolant system pressure less than 1600 psi.

21:37:33 High Pressure Injection (HPI) Pump 1-2 was on and had normal flow.

21:37:49 HPI Pump 1-1 was on and had normal flow.

21:38:13 Re-established Reactor Coolant Makeup flow.

21:40:22 Containment Normal Sump Pump came on indicating the Quench Tank Rupture Disk had blown.

21:40:36 HPI Pumps were shutdown.

21:43:16 Auxiliary Boiler System was started and at normal conditions.

21:43:41 Tripped Reactor Coolant Pumps (RCP's) 1-1 and 2-2.

21:44:05 Re-established Reactor Coolant Letdown flow.

21:49:57 Put AFP #2 in hand and ran it up to speed (3600 rpm) and then lowered the speed.

21:58:00 Closed block valve to Pressurizer Power Relief Valve.

22:15:22 Started second Reactor Coolant Makeup Pump.

22:22:57 Started #2 HPI Pump.

22:27:24 Brought #2 Main Feed Pump back on with Auxiliary Boiler steam.

22:27:44 Shutdown #2 HPI Pump.

22:33:23 Shutdown #1 Reactor Coolant Makeup Pump.

22:43:54 Shutdown #1 and #2 AFP's.

EXHIBIT - B

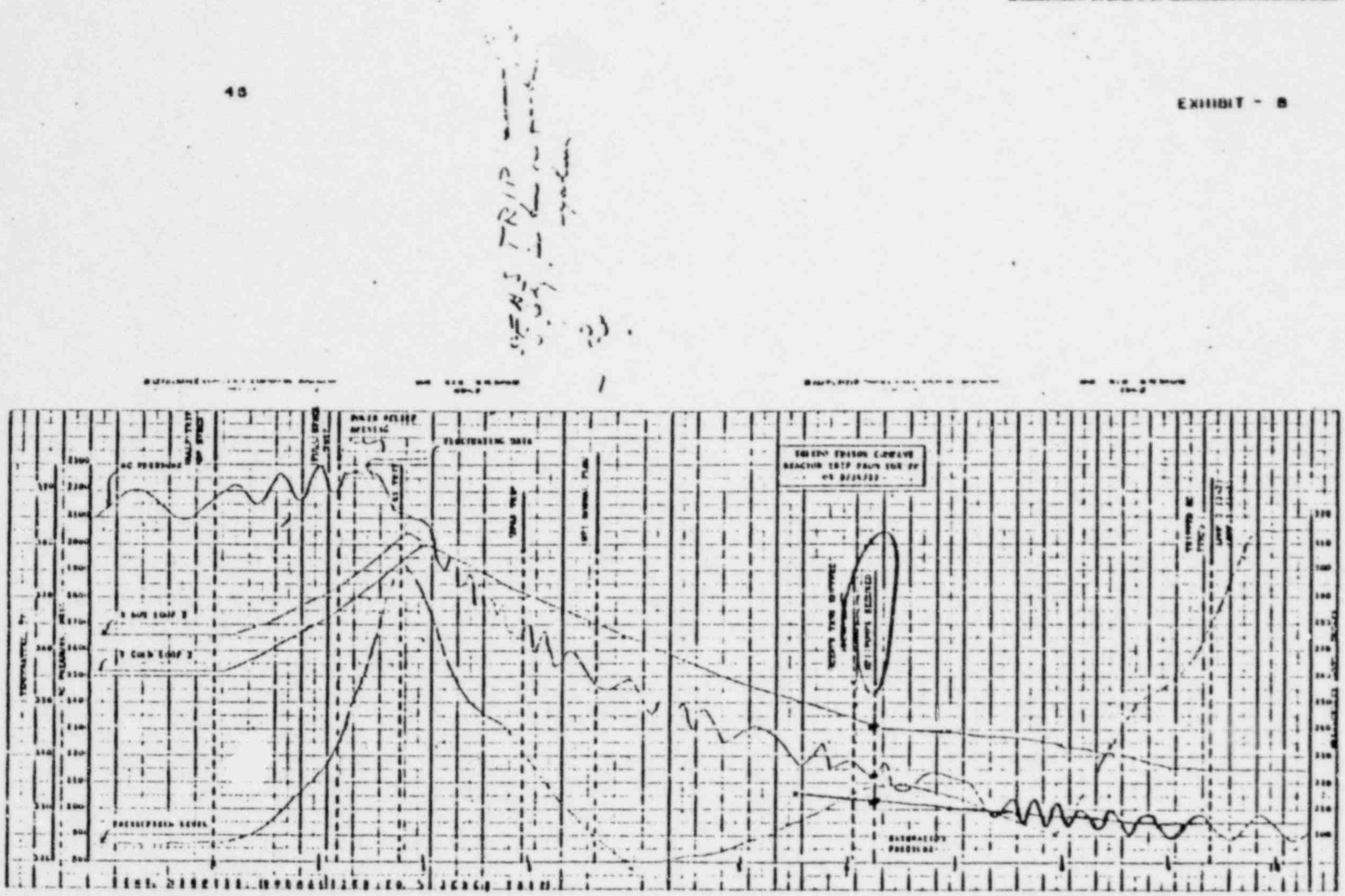


FIGURE 7-1

$P = 1125$ psig $T = 545°F$

$T_{SAT} = 560°F$ $P_{SAT} = 1026$ psig

∴ @ TIME OF SECURING HPI, RCS TEMP. WAS 15°F SUBCOOLED.

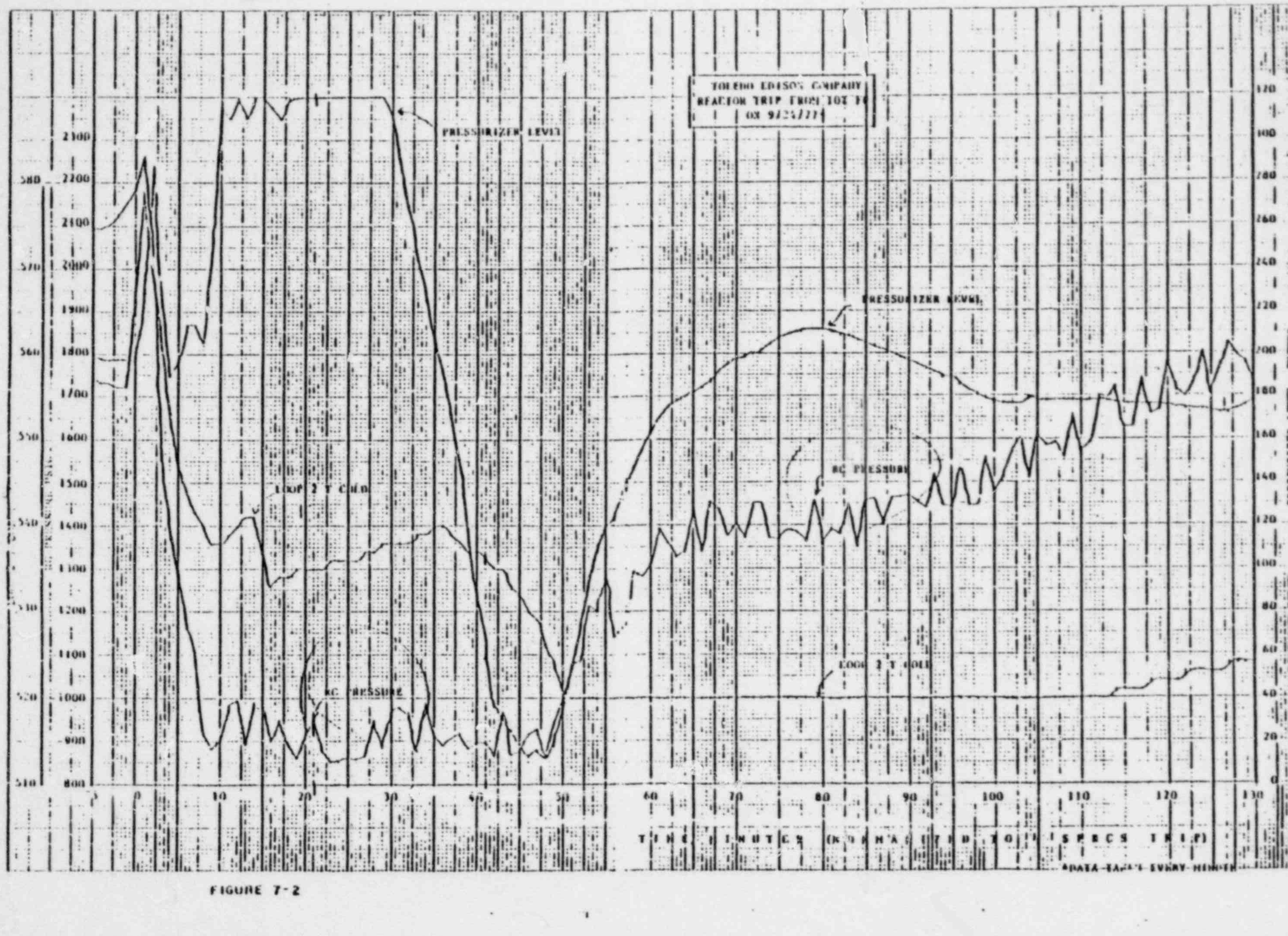

TOLEDO EDISON COMPANY
PRESSURIZER LEVEL
LOOP 2 T COLD
RC PRESSURE
PRESSURIZER LEVEL
RC PRESSURE
LOOP 2 T COLD

FIGURE 7-2

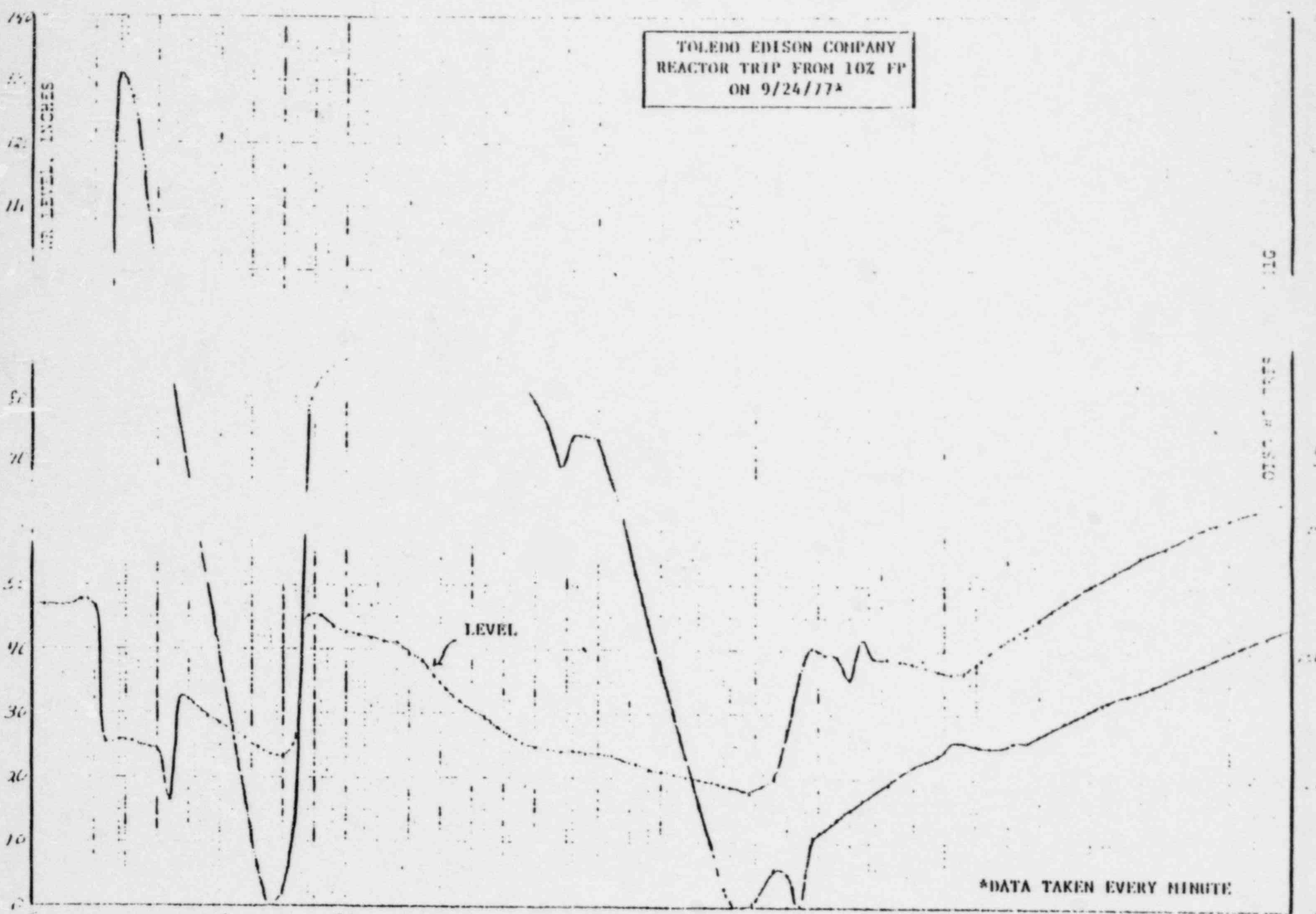
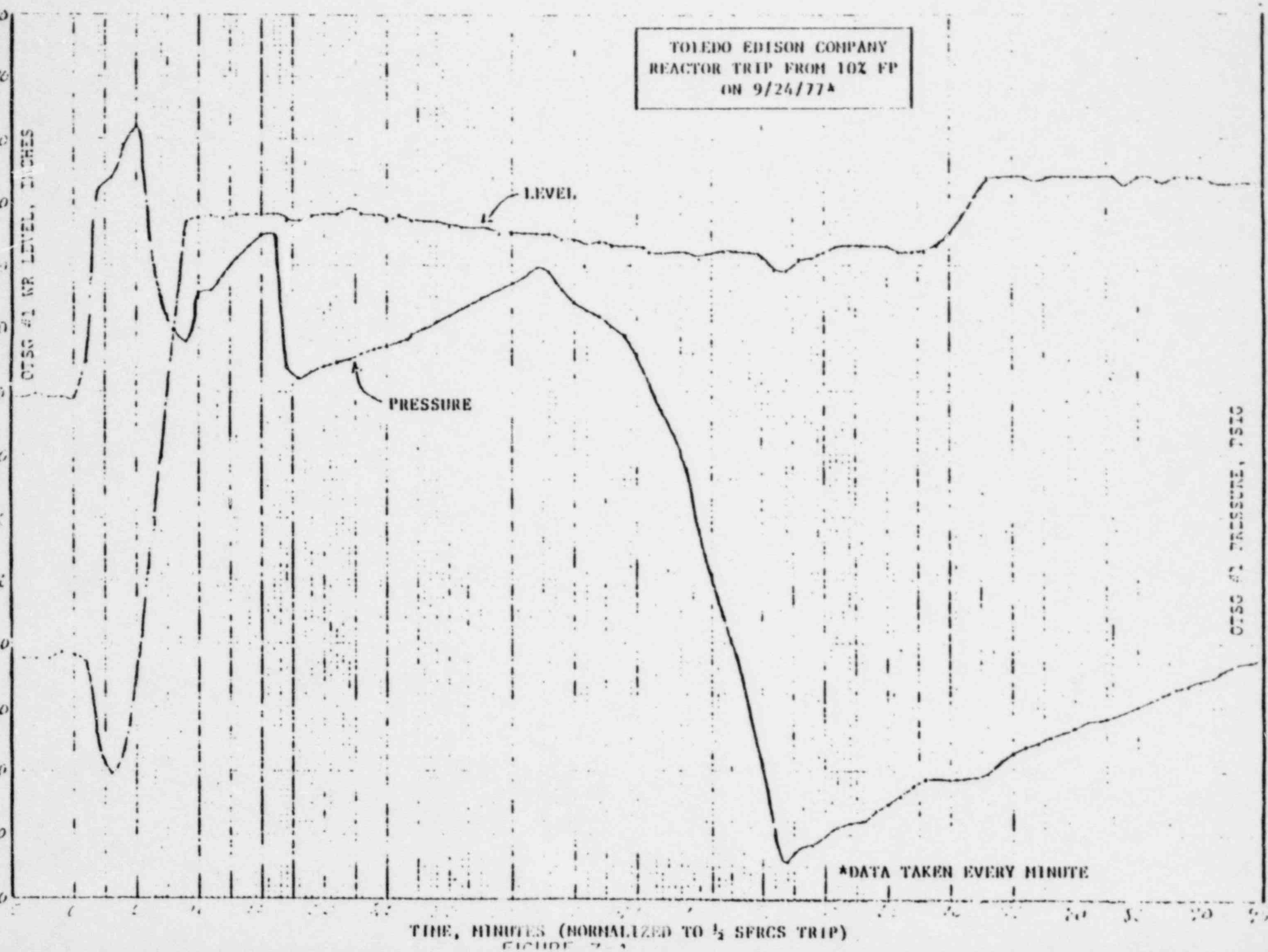

TOLEDO EDISON COMPANY
REACTOR TRIP FROM 10% FP
ON 9/24/77*
LEVEL
*DATA TAKEN EVERY MINUTE
TIME, MINUTES (NORMALIZED TO ½ SFRCS TRIP)

FIGURE 7-3



TOLEDO EDISON COMPANY
REACTOR TRIP FROM 10% FP
ON 9/24/77*
LEVEL
PRESSURE
OTSG #1 WR LEVEL, INCHES
OTSG #1 PRESSURE, PSIG
*DATA TAKEN EVERY MINUTE
TIME, MINUTES (NORMALIZED TO ½ SFRCS TRIP)

FIGURE 7-

C. System Description

Steam and Feedwater Rupture Control System

1. General

The steam and feedwater rupture control system (SFRCS) is an automatic system designed to protect against the following incidents:

a. Main steam line rupture, either upstream or downstream of main steam isolation valve (MSIV). This condition, if allowed to proceed, could rapidly blow down both steam generators, resulting in a rapid RCS cool down and therefore a rapid reactivity insertion under certain core conditions.

b. Main feedwater line rupture. If on the steam generator side of the feedwater check valve, this is approximately the same accident as the steam line rupture; on the feedwater side of the feedwater check valve this results in a total loss of feedwater.

c. Loss of all feedwater. This (as well as the above incidents) could result in boiling both steam generators dry. If this happens, there would be no steam available for running auxiliary feedwater pumps to remove decay heat.

d. Loss of 4 reactor coolant pumps (RCP). This results in loss of reactor coolant flow and therefore auxiliary feedwater is needed to establish reactor coolant natural circulation flow.

The SFRCS, upon indication of conditions a, b and c above will isolate both steam generators (close the main feedwater valves and main steam line valves and trip the turbine) and start the auxiliary feedwater system. Auxiliary feedwater is initiated to keep steam available for the auxiliary feed pump turbines and to remove decay heat from the reactor coolant system. Once this is accomplished, the operator will have time to begin a cool down in an orderly manner.

2. Design Criteria

The design criteria for the SFRCS and the auxiliary feedwater system are as follows:

a. The system must perform its safety function after a single active failure has occurred. This means that the single failure of any power supply, pump, turbine, instrument or control system logic channel will not prevent the system from removing decay heat from the reactor coolant system.

b. A main steam line break upstream of the MSIV or a main feedwater break downstream of the main feedwater isolation valve will disable one steam generator. After this event both auxiliary feed pumps and turbines will be aligned to the remaining intact steam generator. This remaining steam generator has adequate capacity to remove the decay heat from the reactor coolant system.

3. Functional Description (Refer to Enclosures 1 and 2)

The SFRCS is divided for redundancy, diversity, and testability into four logic channels. Logic channels 1 and 3 form channel 1, and logic channels 2 and 4 form channel 2. In one cabinet one logic channel has an AC power supply, the other a DC supply:

Logic Channel	Cabinet	Power Supply
1	C5762A	Y1 (120V AC)
2	C5792	Y2 (120V AC)
3	C5762A	D1P (125V DC)
4	C5792	D2P (125V DC)

Each logic channel receives the following inputs which will cause it to trip:

a. Six pressure switches, two on each main steam line set at 600 psig decreasing and one on each main steam line set at 650 psig decreasing.

b. Two main feedwater pressure differential switches, one from each main feedwater line (see Enclosure 1 for sensing points) set at 177 psid steam generator pressure higher than main feedwater line pressure.

c. Two level transmitters with bistables, one on each steam generator set at 17" decreasing level on the startup range.

d. A contact from RPS pump power sensing circuit; contact opens on loss of all four RCP's.

The SFRCS cabinets consist basically of an AC and a DC power supply, input buffers, logic modules, and output relays. The output relays de-energize to actuate their associated equipment. They also turn out a light on the cabinet when in the tripped state.

Each input to SFRCS has a test switch and light so that a trip of that input can be initiated for testing purposes.

The outputs from the SFRCS are contacts from the output relays. These contacts are in the control circuits for the SFRCS actuated equipment. Most components require two SFRCS logic channels to trip to actuate. See Enclosure 2 for a listing of actuated equipment.

There is a block feature associated with the low steam pressure trip. To prevent the system from actuating on cooldown, each logic channel has a "block" pushbutton on C5721 and on the SFRCS cabinet. When steam pressure goes below 650 psig a block permissive light is received on C5721 along with annunciator and computer alarms. When the block button is pushed, the channel will not trip on low steam pressure and a "MN STM LOW PRESS TRIP BLKD" light is actuated on C5721 as well as annunciator and computer alarms. On a heatup the block signal is automatically removed when the steam generator pressure exceeds 650 psig.

There is another block which is utilized on cooldown. If the decay heat system suction valves from the reactor coolant system (DH11 and 12) are open, this block will prevent the opening of the steam inlet valves to the auxiliary feed pump turbines. This prevents the SFRCS from starting the auxiliary feed pumps when all reactor coolant pumps are secured on shutdown. This "block" is automatically removed when the decay heat system is shut down on startup.

4. System Logic

a. The response of the actuated components depends on the type of trip: (refer to Enclosure 2)

1. On low steam pressure on one main steam line, both steam generators are isolated. In addition, both auxiliary feed pumps are aligned to the steam generator which is above 600 psig.

If both steam generators go below 600 psig, both steam generators are isolated and no auxiliary feedwater is initiated.

If any other trip (such as low steam generator level) accompanies a low steam pressure trip, the valves will align per low steam pressure trip logic.

2. On high feedwater pressure differential or low steam generator level on one steam generator, both steam generators are isolated and each auxiliary feedwater pump is aligned to feed its respective steam generator (1 to 1 and 2 to 2).

3. On loss of all four reactor coolant pumps, each auxiliary feedwater pump is aligned to its respective steam generator. The steam generators are not isolated.

4. On all of the above events, the turbine is tripped by the SFRCS.

b. The auxiliary feedwater pump governor control switch in the control room bus has 3 positions:

Auto-Essential (SFRCS)
ICS
Manual

In the auto-essential position, the auxiliary feedwater pump is in auto-essential level control. In the ICS position, the auxiliary feedwater pump is on level control from the ICS; via the Hand-Auto station. In manual, the auxiliary feedwater pump is controlled by the operator with the Raise-Lower switch.

c. The SFRCS starting of the auxiliary feedwater pumps will automatically reset once the trip condition on the input is removed. None of the valves, however, will return to their original position until operated individually from the control room or a new trip condition occurs.

5. System Operation

In order to understand the operation of the SFRCS system, it is best to follow the various system actions under several accident conditions. The following cases will be considered:

a. Steam Line Rupture
b. Feedwater Line Rupture
c. Loss of Feedwater Pumps
d. Loss of Four Reactor Coolant Pumps

Enclosures 1 and 2 should be used as an aid to understanding the description. All discussions assume 100% FP operation at start. Some non-SFRCS actions are considered to aid in understanding the transient.

(1) Steam Line Rupture - Assume steam line 1 shears downstream of MSIV. Steam pressure will rapidly drop. When either steam generator reaches 600 psig, all four logic channels will trip, isolating both steam generators. (See Enclosure 2 for specific valves.) The MSIV takes five seconds to shut, the main feedwater isolation valve 15 seconds. Both steam lines will probably drop below 600 psig, therefore, auxiliary feedwater will not start until one steam generator recovers to above 600 psig. Auxiliary feedwater pumps will align as described in Section 3 above to feed the steam generator that first recovers to 600 psig, with both auxiliary feed pumps. The SFRCS will trip the turbine. The reactor will trip on low pressure.

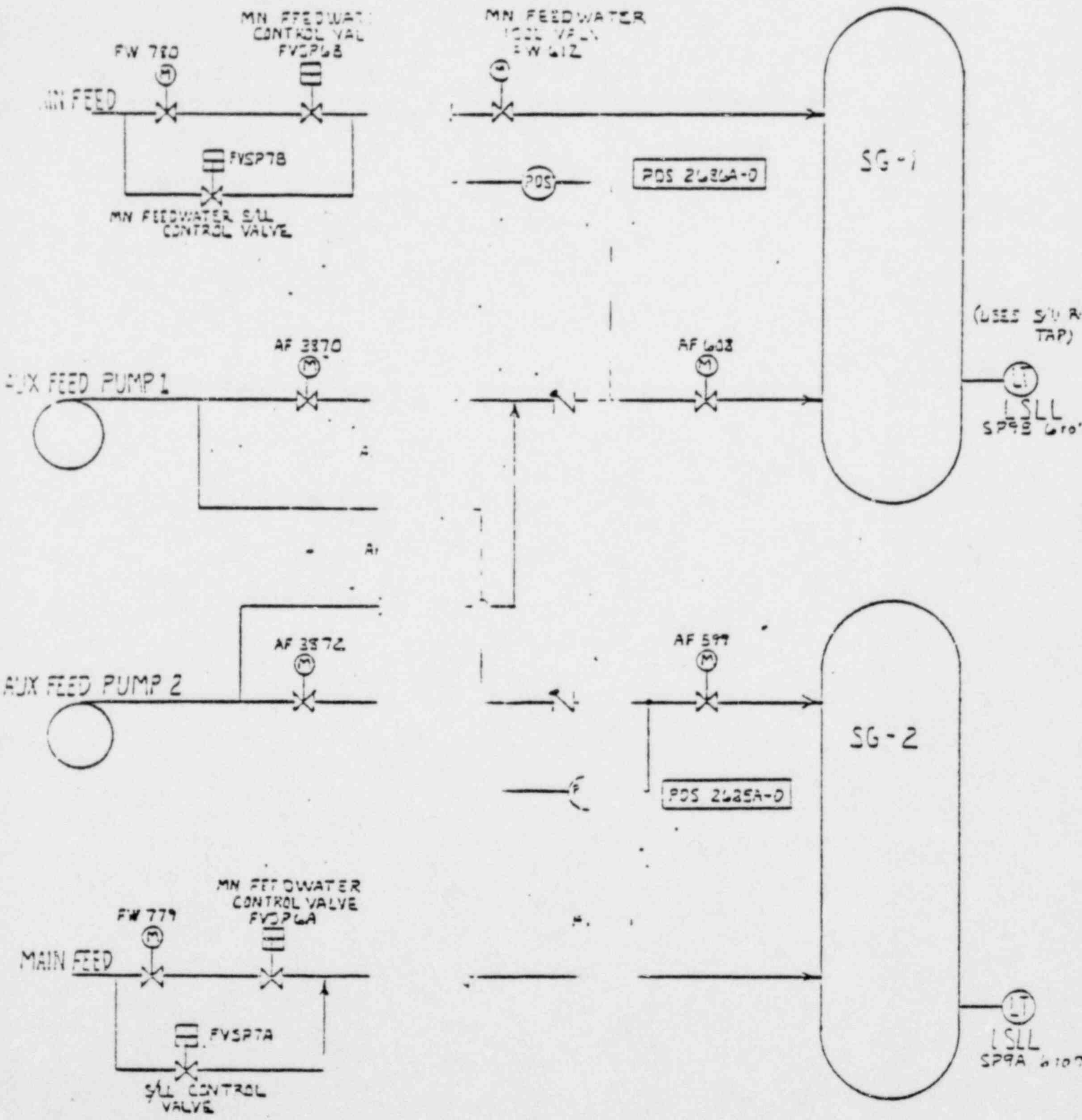
When both steam generators are above 600 psig, the trip condition automatically clears and the atmospheric vent valves may be used for pressure control cooldown if required and provided no other trips are present.

(2) Feedwater Rupture Line - Assume feedwater line 1 shears upstream of the feedwater line check valve. Feedwater pressure will rapidly drop. When either feedwater heater drops to 177 psig less than steam generator pressure, the SFRCS will isolate both steam generators and align the auxiliary feed pumps to their respective steam generator (1 to 1; 2 to 2). The reactor will trip on high pressure and the SFRCS will trip the turbine.

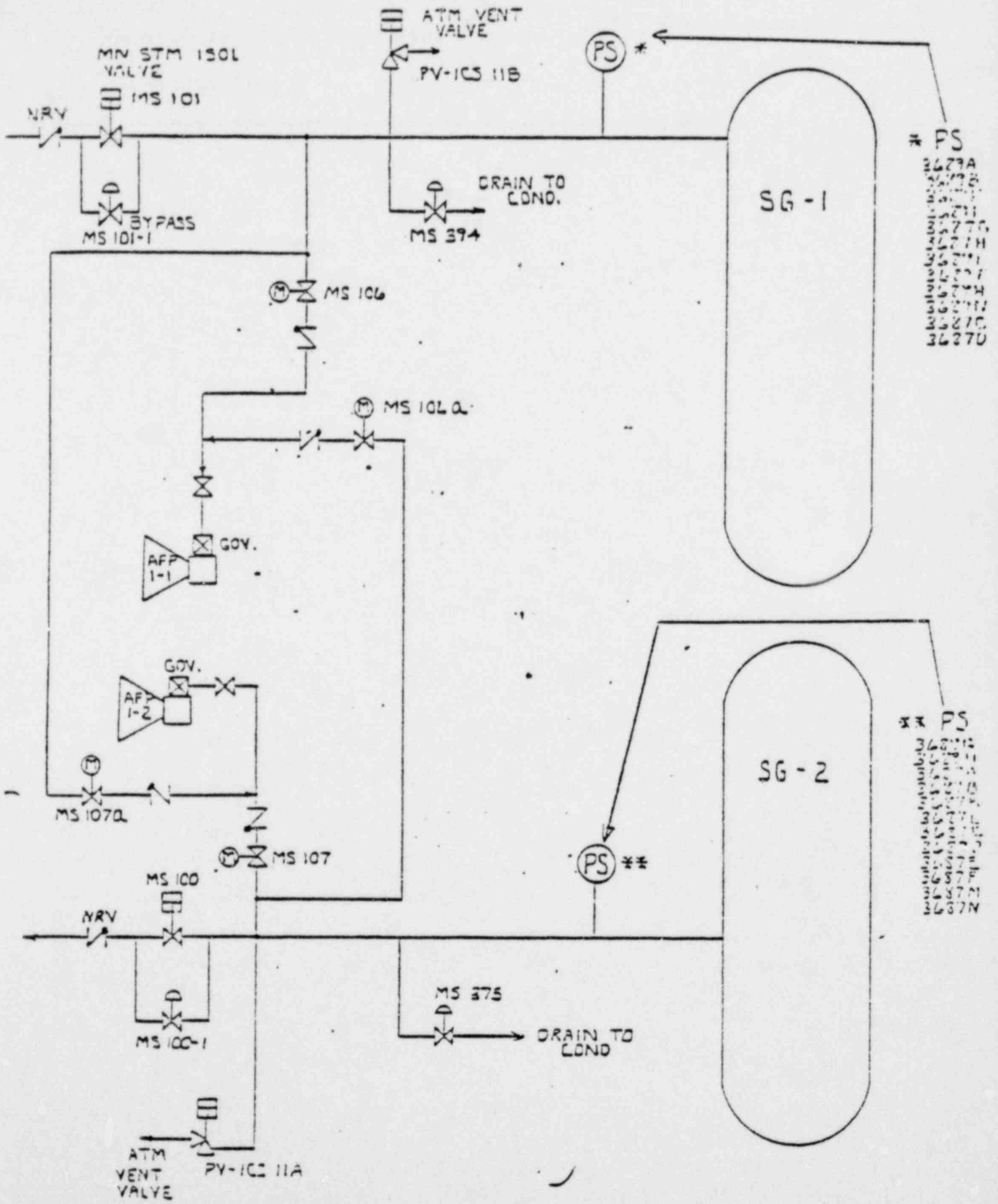
(3) Loss of Four Reactor Coolant Pumps - If all four reactor coolant pumps trip, the turbine will be tripped by the SFRCS and the reactor protection system will trip the reactor. The SFRCS will initiate auxiliary feedwater. The steam generators will not be isolated.

ENCLOSURE 1 SFRC[illegible] [illegible] EQUIPMENT (FEEDWATER)

(FOR STEAM VALVES SEE NEXT PAGE)



ENCLOSURE 1 SFRCS ACTUATED EQUIPMENT (STEAM)



ENCLOSURE 2

STEAM-FEEDWATER RUPTURE CONTROL SYSTEM ACTUATION

CHANNEL 1 (C 5767A)	MS 101	MS 100	MS101-1 NOTE 3		MS 394 NOTE 3		ICS 11B NOTE 3		FW 612		FW 780		SP 7B NOTE 3	
CHANNEL 2 (C 5792)	MS 101	MS 100		MS100-1 NOTE 3		MS 375 NOTE 3		ICS 11A NOTE 3		FW 601		FW-799		SP 7A NOTE 3
LOW PRESSURE MAIN Steam Line 1 (<600#)	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT
LOW PRESSURE MAIN Steam Line 2 (<600#)	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT
STM. - FW ΔP SG 1 HIGH (>177 PSID)	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT
STM. - FW ΔP SG 2 HIGH (>177 PSID)	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT
LOW LEVEL SG 1 (<17" SUR)	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT
LOW LEVEL SG 2 (<17" SUR)	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT	SHUT
LOSS OF 4 RC PUMPS														

CHANNEL 1 (C 5762A)	SP 6A		MS 106		MS 106A		AF3870		AF 3869		AF 608		MAIN TURBINE
CHANNEL 2 (C 5792)		SP 6B		MS 107		MS107A		AF3872		AF3871		AF 599	
LOW PRESSURE MAIN STEAM LINE 1 (<600#)	SHUT	SHUT	SHUT	OPEN NOTE 1	OPEN NOTE 1	SHUT	SHUT	OPEN NOTE 1	OPEN NOTE 1	SHUT	SHUT	OPEN	TRIP
LOW PRESSURE MAIN STEAM LINE 2 (<600#)	SHUT	SHUT	OPEN NOTE 1	SHUT	SHUT	OPEN NOTE 1	OPEN NOTE 1	SHUT	SHUT	OPEN NOTE 1	OPEN	SHUT	TRIP
STM-FW ΔP SG 1 HIGH >177 PSID	SHUT	SHUT	OPEN	OPEN	SHUT	SHUT	OPEN	OPEN	SHUT	SHUT	OPEN	OPEN	TRIP
STM-FW ΔP SG 2 HIGH >177 PSID	SHUT	SHUT	OPEN	OPEN	SHUT	SHUT	OPEN	OPEN	SHUT	SHUT	OPEN	OPEN	TRIP
LOW LEVEL SG 1 (<17" SUR)	SHUT	SHUT	OPEN	OPEN	SHUT	SHUT	OPEN	OPEN	SHUT	SHUT	OPEN	OPEN	TRIP
LOW LEVEL SG 2 (<17" SUR)	SHUT	SHUT	OPEN	OPEN	SHUT	SHUT	OPEN	OPEN	SHUT	SHUT	OPEN	OPEN	TRIP
LOSS OF 4 RC PUMPS			OPEN NOTE 2	OPEN NOTE 2	SHUT	SHUT	OPEN	OPEN	SHUT	SHUT	OPEN	OPEN	TRIP

NOTES:

1. If both main steam lines are <600#, these valves shut.
2. These valves will not open if DH 11 and DH 12 (DH Suction from RCS) are open.
3. These valves are closed on a ½ channel trip.

EXHIBIT D

File: 0017,0486, M-36

October 11, 1977

TOLEDO EDISON

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

Serial No. 391

Docket No. 50-346

Mr. James G. Keppler
Regional Director, Region III
Office of Inspection & Enforcement
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

This letter supersedes my letter to you on this subject dated October 5, 1977.

In accordance with 10 CFR Part 21.21(b), this is a report of a defect in a component installed in the Davis-Besse Nuclear Power Station Unit No. 1. The component involved is the governor on the auxiliary feed pumps.

The auxiliary feed pumps were supplied by Byron Jackson Pump Division. The steam driven pump turbine was supplied by Terry Corporation to Byron Jackson. In turn, the turbine governor was supplied to Terry Corporation by Woodward Governor Company. The turbine governor is identified as a type PG-PL, which has a servomotor control employing a Bodine Electric Company motor.

The defect involves a potential for the governor to bind under certain conditions and preventing the turbine from coming up to design speed. The operating procedures for this equipment called for the governor to be placed in the high speed stop position prior to shutting down the turbine. Investigation has shown that with the Bodine servomotor driving against the high speed stop, a misalignment force is applied to the T-bar of the governor linkage. This misalignment force creates a potential for the governor to bind at a speed position less than design speed upon a turbine startup. This misalignment force does not always cause the governor to bind and this misalignment force can be removed by driving the Bodine servomotor away from the high speed stop.

The safety hazard which could be created is the potential for both auxiliary feed pumps to fail to come up to design speed upon startup. This could result in a substantial loss of auxiliary feedwater flow to the steam generators when such flow was required. This in turn could cause significant reactor coolant system pressure/temperature transients, and significant boiling in the reactor coolant system if substantial decay heat were present in the reactor core.

The evaluation and identification of this defect was provided to me on September 30, 1977, and was discussed with Mr. T. Harpster of your office on September 30, 1977.

There are two identical auxiliary feed pumps with the turbine governors, described above, installed in the Davis-Besse Nuclear Power Station Unit No. 1.

The corrective action taken was to modify the governor including the removal of portions of the pneumatic speed-setting mechanism to assure that the governor will properly respond to speed demand signals. The pneumatic speed-setting mechanism was never an integral part of the functioning of the governor, because the governor employed servomotor control. This modification was accomplished at the Woodward Governor Company facilities. Subsequent testing at these facilities has proved the proper functioning of the governor. The modifications were completed prior to the current unit startup. The governors have been tested for proper functioning on auxiliary steam, and the surveillance test will be completed during Mode 3 of the current startup.

Yours very truly,

Lowell E. Roe

Lowell E. Roe
Vice President
Facilities Development

db b/9-10

bcc:
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E. Historical Log of Station Operations (September 24 - October 28)

Sept. 24	Reactor critical at 15% power, generator on the line at 110-140 mw, performing controls tuning
	1700 - Discovered steam leak on steam lead between No. 2 Turbine Control Valve and high pressure turbine
	1830 - Turbine-generator taken off the line to repair steam leak. Reactor critical at about 9% power.
	2135 - Received Steam and Feedwater Rupture Control System Actuation, resulting in Reactor Trip, and Safety Features Actuation
	2345 - Plant stable at 1800 psig, T_{ave} ÷ 525°F
Sept. 25	0415 - Started Plant Cooldown
	0645 - Completed initial survey of Containment
Sept. 26	Cleanup and repairs begun
Sept. 30	Completed repair and replacement of mirror insulation on No. 2 Steam Generator
Oct. 3	Auxiliary Feedpump Governors removed and sent to Woodward Governor Factory
	Quench Tank Rupture Disc replaced
Oct. 5	Vented Reactor Coolant System and run Reactor Coolant Pumps to get data to evaluate status of pumps and seals.
Oct. 6	Started Feedwater Cleanup in preparation for Reactor Coolant System heatup
Oct. 7	1830 - Received NRC approval to proceed with plant startup
Oct. 8	1530 - Checkout of Auxiliary Feedpumps (using Auxiliary Steam) completed
Oct. 11	Attempted to test pressurizer power relief valve. Unsuccessful due to electrical circuit problems.
Oct. 12	Pressurizer power relief valve control circuit working, stroked valve and it stuck open again

Oct. 13		in working on Power Relief Increased RCS pressure to complete testing or Coolant Pumps.
Oct. 15		repairs to ower Relief Valve and tested
Oct. 16		testing of Auxiliary Feedpumps. Governors modified by oodward to prevent sticking
		eactor critical
		ed the tu ne
Oct. 17		erator syn ronized
		erator off the line for overspeed tests
		nced Rea or Shutdown for Unit Power tdown Test We are now back in Power ation Se nce
Oct. 18		ted Un wer Shutdown Test
		enced Rea or Startup
	1	erator Syn ronized
Oct. 19	1	erator of e to repair steam leak n No. ntrol Valve and HP Turbine
Oct. 20		nerator syn onized
Oct. 23	1	half trip of FRCS caused low steam generator level, resul in full SFRCS trip, Reactor nd SFA
Oct. 27	2	or Crit.
Oct. 28	1	erator Syn nized

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: August 20, 1979

Leon B. Engle

Leon B. Engle