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SUBCOMMITTEE ON REACTOR OPERATIONS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON REACTOR OPERATIONS

Nuclear Regulatory Commission
Room 1046
1717 H Street, N.W.
Washington, D. C.

Wednesday, February 12, 1986

The meeting of the ACRS subcommittee convened at 1:30 p.m., Mr. Jesse C. Ebersole, chairman, presiding.

ACRS MEMBERS PRESENT:

MR. JESSE C. EBERSOLE
DR. WILLIAM KERR
MR. CARLYLE MICHELSON
DR. DADE W. MOELLER
MR. GLENN A. REED
MR. DAVID A. WARD

DR. IVAN CATTON, Consultant

MR. HERMAN ALDERMAN, ACRS Staff Member

PUBLIC NOTICE BY THE

UNITED STATES NUCLEAR REGULATORY COMMISSIONERS'

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

WEDNESDAY, FEBRUARY 12, 1986

The contents of this stenographic transcript of the proceedings of the United States Nuclear Regulatory Commission's Advisory Committee on Reactor Safeguards (ACRS), as reported herein, is an uncorrected record of the discussions recorded at the meeting held on the above date.

No member of the ACRS Staff and no participant at this meeting accepts any responsibility for errors or inaccuracies of statement or data contained in this transcript.

P R O C E E D I N G S

MR. EBERSOLE: The meeting will come to order.

This is a meeting of the ACRS Subcommittee on Reactor Operations. I am Jesse Ebersole, Chairman of the Subcommittee on Reactor Operations. The other members are Bill Kerr, Carlyle Michelson, Dave Moeller, Mr. Glen Reed, and we have Mr. Ivan Catton as a consultant.

The Subcommittee will discuss recent operating events. We will be briefed on the 50-54(f) Improvement Program on Fermi and Technical Specification Improvement Program.

The rules for participation have been announced as part of the notice of the meeting published in the Federal Register on January 21, 1986. It is requested that each speaker first identify himself or herself and speak with sufficient clarity and volume so that he or she can be readily heard.

We have received no written comments or requests for time to make oral statements from members of the public.

I will now ask if any Subcommittee members or consultants have any comments or solicitations to make before we get into the meat of this meeting.

MR. MOELLER: I have one comment. Just to ask, Carlyle Michelson had brought to my attention this draft report or report that the AEOD is preparing on the loss of

the ventilation system in the control room at the McGuire plant, I believe it was. We could take it up as a joint effort.

MR. EBERSOLE: Perhaps next time.

MR. MOELLER: It will depend on when they finish their report.

MR. MICHELSON: I wasn't going to get into it until the end of the meeting, but there is this question of, IE does a fine job of bringing to us these current events as they come up but we don't hear briefings on AEOD reports which deal with combinations often of current events and final analysis in the longer-range sense. Which subcommittee is going to listen to the results of those kind of studies as opposed to current events?

They are also operating history, just from a different viewpoint. We don't normally bring those up here because this is the operating events.

MR. EBERSOLE: I suggested that we take this up as a part of our presentation to the full committee.

MR. MICHELSON: That was the question, should this subcommittee do it?

MR. EBERSOLE: Let's introduce that as a copy tomorrow.

Well, we have just gotten through a full morning which I think ellipses most operation events of recent

times. But I see an interesting number here. Let me ask Mr. Dennis Allison to get us going right away.

MR. ALLISON: I am Dennis Allison. I am the acting branch chief for the events analysis branch in IE. Next to me is Ron Hernan, principal spokesman for NRR. Before we start the first event, I would like to say, I understand that the team that has been looking at the earthquake at Perry will arrive a little later, about 3:30. So there toward the end of the presentation, we won't be able to move them up until they get here. I think that is all we had to start off with. The first presentation is on McGuire, startup with the degraded HPSI system. It is not an air system but a HPSI system.

MR. GIITTER: My name is Joe Giitter. I am in the events analysis branch of the Office of Inspection and Enforcement. At the last Subcommittee meeting, you heard about a single failure of a nonsafety-related system, that resulted in a loss of instrument air and a challenge to the safety systems at both McGuire units. This afternoon, I am going to discuss a set of undesirable conditions that existed the following day when unit 1 attempted to start up.

The problem was failure to repair volume control tank isolation valve motor operators prior to startup which would have prevented VCT isolation on a safety injection signal. The safety significance of this is that HPSI may

not have functioned as required on a real demand.

As you recall, at 0640 on November 2, a rupture in a flexible pipe at the discharge of an instrument air compressor resulted in a loss of instrument air to both units 1 and 2, a trip of both units 1 and 2 and a safety injection in unit 1. Ordinarily on a safety injection, you would have isolation -- first you would have opening of these two valves here and then isolation of the volume control tank as suction is swapped from the VCT to the refueling storage tank. These valves will go closed as required.

However, it was later discovered that the valve motor operators had burned out. The licensee has two theories that they are looking at as to why these motor operators may have burned out. One is that there was a hammering problem.

In a hammering problem, as a valve closes, the torque builds up which causes a torque switch to open. When the torque switch opens, power is removed from the valve motor which relaxes torque. When the torque is relaxed, that torque switch is allowed to reset. If a close demand signal is still present, once that torque switch resets, the valve will start to close again even though it is closed. It will try to close against a seat. Torque will build up, the torque switch will open, and the

motor operator will stop -- will stop trying to close the valve, torque will relax, the same thing will happen over and over again and the valve will be hammered closed. This can eventually result in overheating of the valve motor.

Another theory is that on the McGuire system, there is a miniflow line from the centrifugal charging pumps that is not isolated on a safety injection signal. So what happened during this event is that the VCT actually went solid and the safety relief lifted on the VCT, pressure setpoint about 75 pounds. This increases the static head that these -- that this valve here sees. It was believed that there may have been some leakage past this valve here and when the operators, seeing that the VCT was full, tried to open remote this valve, remote manually, that they encountered high differential pressures and that caused the motor operator to burn out here and subsequently high differential pressure caused this motor operator to burn out because this motor operator here wouldn't see the DP that this one would.

It doesn't seem to be fully explained yet. The licensee is still investigating. But those are two of the theories right now.

MR. EBERSOLE: This event is still open as to why these valves failed?

MR. GIITTER: That is correct. The licensee is

still --

MR. EBERSOLE: It would have been unlikely that the torque switch would have been, the lockout or the return -- when the torque switch closes or opens to stop the motor, doesn't that result in an interlock which it cannot subsequently attempt to close again until it opens again?

MR. GIITTER: There are various types of antihammering devices on different types of valves. My understanding is that this particular type of valve had an old mechanical latch system that prevented that torque switch from resetting if the limit switch had sensed that valve being closed. Why that wouldn't have worked, I don't know.

MR. EBERSOLE: Was it both --

MR. GIITTER: Both of these.

MR. EBERSOLE: So we have an even validation of the single failure theory.

MR. GIITTER: It seems that a single failure did cause the failure of both of these motor operators.

MR. EBERSOLE: Well, it is nice to add another one to the list. Like Salem. Carry on.

MR. GIITTER: Prior to startup, the operators manually opened the valves, but did not repair the motor operators. So this put them in a situation that if they

were to have a safety injection signal, these valves would not have automatically closed. And you would then be in a situation where you were taking suction from the VCT and RWST. Unit startup was commenced at about 6:00 on the 3rd and the unit was in mode 2 for about six hours. The startup was pending repair of the severed instrument line on the secondary side of the steam generator. Because they couldn't get that repaired, they decided to go into cold shutdown.

MR. EBERSOLE: So what would have happened if you had taken common suction on the VCT as well as the RWST. Would you get nitrogen?

MR. GIITTER: I will get to that.

There are two concerns. One concern, I think this is a lesser one, that is because of the difference in volume -- this is the volume of the VCT is on the order of several thousand gallons whereas the volume of the RWST is on the order of several hundred thousand gallons. But there is a concern, if you were to draw some suction from the VCT, that the boron injected into the vessel would be lower than that assumed in the safety analysis. That is a concern more from reactivity perspective than anything else.

I think the more major concern is could the VCT, the hydrogen and the nitrogen, if this VCT level was low enough, could this hydrogen and nitrogen get entrained into

the charging pump suction path and ultimately result in gas binding of the charging pumps. I think that was a question you had.

MR. EBERSOLE: That is what happened at Palo Verde.

MR. GIITTER: Yes. We asked the licensee about this. And first -- well, I will get to that.

The NRC op center was notified at 1218 on January 14, 1986. This event happened on November 3rd. The only way we found out about this event is that the resident inspector, investigating the previous loss of instrument air event, went back through the shift supervisor's log and saw that they had started up with these valve motor operators being inoperable. That is the only way we found out about it.

The day after the resident found that out, the licensee called this in to the op center and notified us. Apparently what happened is the licensee looked at their tech specs and because they did not consider these valves to be in an ECCS flow path, didn't think there was any problem from a tech spec of starting up. And I believe they were planning to repair these valves once in the process of starting up or once they had actually gone to power.

MR. EBERSOLE: Why didn't they consider them in

the ECCS set? One basis would be if they are closed, they are not in the ECCS set. If they are open, they are.

MR. GIITTER: Right.

MR. EBERSOLE: Do they discriminate like that? That is the mode or the position of the valve in order to determine whether this is within or without the ECCS set?

MR. GIITTER: I can't tell you what the licensee's logic was. I don't think that we agree with it.

MR. EBERSOLE: Certainly if they are open, then the ECCS function is degraded?

MR. GIITTER: That is right.

MR. EBERSOLE: So they are in fact in the ECCS set if they are shut, are they not?

MR. GIITTER: I would have to agree with that.

MR. EBERSOLE: I am asking in the general context, do the licensees do that discriminatory classification to determine whether a valve in a given position is or is not in the ECCS set?

MR. GIITTER: They have some means of making that determination, but I don't know what the logic is.

MR. EBERSOLE: It sounds a little vague to me.

MR. GIITTER: That was the official response we got.

MR. EBERSOLE: We got these things on every plant. Are they all in the ECCS set?

MR. GIITTER: I am sorry.

MR. ALLISON: I don't think we know the answer to that today. We will have to --

MR. EBERSOLE: I am talking about the generic aspect. Carry on.

MR. GIITTER: Okay. They did repair the valve motor operator when they went into cold shutdown. They reviewed the design and determined that this -- at first, when it was believed that the hammering problem may have been a contributor, they looked at other valves and determined that hammering wasn't a problem in any other valve motor operators in the plant.

They performed a test -- let me pull this up a little bit -- when they were in operation, they performed a test. They started the charging pumps and they left these valves open and they sent an operator down there to see how long it would take him to go down and close these valves manually. The first time they conducted tests, it required 19.4 minutes for the operator from the time he left the control room or did whatever he had to do to actually closing the valves, it took 19.4 minutes. The licensee concluded at that point in time that the VCT level was down at around 20 percent. And according to their analysis, original analysis, at about the 20 percent level, initially you have a higher pressure here and the suction is

originally going to be taken from the VCT because you have 30 pounds cover gas pressure. But once the VCT level drops to 20 percent, the pressure decays because the gas volume is increasing. And at that point, the primary source of suction is the RWST. So they concluded that at 19.4 minutes that it would take the operator to get down there, that the VCT level would be at about 20 percent.

Under those conditions, they were assuming the VCT initial level was at 100 percent, which it was for this particular event because of the failure to -- the design in not isolating the miniflow lines.

They subsequently did two other tests. Both of those tests, they sent two different operators down. In one case it took them 15 minutes. The other case took them 18 minutes. They concluded that there would be no gas binding of the charging pumps sooner than that 18.5 minutes.

Now, there is one question that comes to mind is, in a real safety injection instance, is it likely, is it realistic that an operator could get down there in that amount of time, if they really did have a small break LOCA, for example. Is it realistic that they could get down there in that amount of time and that they would close those isolation valves.

There is an enforcement conference scheduled for February 28. I believe that these issues will be brought

up at that point in time.

MR. EBERSOLE: The fact that it is two valves there suggests that they needed redundancy and in fact it is almost evident that they had to be considered as safety grade elements of the design, just because there are two of them. Were they in the tech specs, to be periodically examined and tested. The safety grade list?

MR. GIITTER: I believe they are.

MR. EBERSOLE: You told me that the operator didn't even know what classification these valves were in.

MR. GIITTER: They looked at tech specs and they didn't see any problems. This is my understanding, from what the licensee told me, they didn't see any tech specs that addressed having these valves --

MR. EBERSOLE: -- function as designed. And they were not able to discern from the tech specs or the physical needs that they were critical to the safety function.

MR. GIITTER: They made the determination that they could have these valves inoperable, prior to startup.

MR. EBERSOLE: They did.

MR. GIITTER: They started up with those valves inoperable. It may have been an oversight on their part.

MR. EBERSOLE: What are you going to do about that, as far as I am concerned, a gross misinterpretation

of the reliability of the HPC?

MR. KERR: This was inoperable but closed? They assumed that they were inoperable but closed?

MR. MICHELSON: Open.

MR. GIITTER: In order to start up, you need to open these valves up.

MR. MICHELSON: You need to normally operate off the volume control tank.

MR. GIITTER: You need to open these valves up. They manually opened the valves so that if the valves received a safety injection signal, because the motor operators were inoperable, they would have been incapable of isolating the VCT on a safety injection and you would get suction from the VCT and the RWST.

MR. REED: Whether or not there is a loophole in the tech specs in saying these have to be operable. I think there is a question of good judgment here that comes into play. I really think the licensee is ignoring the safety-related aspects of this system and those valves in particular. I would certainly, if I was the staff, pursue how this decision got made.

MR. GIITTER: We are currently pursuing that right now. As I mentioned, there is an enforcement meeting on the 28th of this month to discuss this.

MR. EBERSOLE: I hope you will take that pursuit

into regions more distant from just this one. If they are that lax in interpreting the critical needs of the safety system, you may find it all over the place. This might not be an isolated case. This sounds like a gross misinterpretation of a critical functional need.

MR. ALLISON: I don't believe this is a typical kind of thing.

MR. REED: I have a little concern here, as I have watched resident inspectors function, I am surprised the resident inspector didn't catch this the very next morning and have his citation book going.

MR. EBERSOLE: That what I was about to say.

MR. REED: Normally they review the logs the very first thing in the morning, looking at all the incidents through the preceding day and over the nightfall. This thing just stands out.

MR. GIITTER: The resident was the one that did notice it.

MR. REED: But it took him how many days?

MR. GIITTER: I guess he was investigating the loss of instrument air. I really don't know what the situation was there. But it is possible that he could have been tied up with the loss of instrument air at both units.

MR. EBERSOLE: They knew that they were inoperable?

MR. GIITTER: That is my understanding.

MR. MOELLER: Is it part of the resident inspector's duty to go each morning and look at the log? I mean, is there a system of fines for a resident inspector?

MR. ALLISON: Looking at the logs every morning is part of their job.

MR. MOELLER: Does anyone ask, you know, how he missed it or she missed it?

MR. ALLISON: We will ask that and find out.

MR. EBERSOLE: Are the operators commonly asked about the nature of need of these valves in an operational examination?

MR. GIITTER: I can't answer that question. I can find out for you.

MR. EBERSOLE: Why don't we do that as a generic problem. We would like to know whether the operators know whether or not the system -- I don't mean running down to the basement to fix it. Whether in the automatic mode these are critical to the HPSI pumps.

Is that all of that?

MR. GIITTER: That is it.

MR. KERR: Is there a 100 percent certainty that with these valves open this system would not work?

MR. EBERSOLE: I gather there is.

MR. KERR: I haven't heard anybody say that.

MR. EBERSOLE: You remember the case of --

MR. KERR: I am not talking about a case of anything. I am asking a question as to whether there is 100 percent certainty.

MR. EBERSOLE: You are on the wrong side of the coin. The certainty needs to be that you know that they will, not that you know that they don't.

MR. KERR: I want to know, is there a 100 percent certainty that they won't work?

MR. EBERSOLE: I want to know that they will work. I am not willing to deal with that whether they would not.

MR. KERR: I was asking the staff.

MR. MICHELSON: It is a good question because you have to know the relevant pressure from the water storage tank versus the --

MR. KERR: It isn't clear to me at this point.

MR. MICHELSON: On the basis of what is there, you can't tell whether the tank --

MR. ALLISON: The licensee claims that they would.

MR. MICHELSON: Yes. They have done the test.

MR. EBERSOLE: Has he tested it?

MR. ALLISON: Not to our knowledge.

MR. MICHELSON: You got to test it being already

connected to the refueling storage tank as well. I doubt he did it --

MR. EBERSOLE: You are saying that it is maybe sufficient?

MR. KERR: I don't know. The reason I asked the question is because I want to --

MR. EBERSOLE: Has the operator --

MR. GIITTER: The only explanation they gave me was the one I provided to you where they determined how long it would take for an operator to get down and manually close those valves.

MR. EBERSOLE: The inference you can draw from that is they better get with it because it is going to quit working.

MR. REED: I don't think the test of how long it takes the operator to do it has any relevance to the issue. The operator could fall in the stairway and kill himself while en route. Here is a design. If they want to do a test, they look at the relative heads of the two tanks in their position and location and over pressure of the gases. But after all, that is pretty much a standard Westinghouse design for many, many years. I thought everyone had been trained and knew the importance of these valves and their functioning and why they are there.

MR. EBERSOLE: I think it goes without saying,

we will put this on the list to be discussed at the full committee meeting because of its generic inferences.

Is that all there was to that?

MR. GIITTER: That is all I had.

MR. EBERSOLE: Any further comments?

Let's move to the second one.

MS. WEGNER: On January 7, 1986, Carolina Power & Light Brunswick phoned in the following 5072 report. Test results from Wyle Laboratory on 11 SRV, six SRV failed to lift at pressures as high as 200 psi above setpoints. Two SRV were satisfactory and three SRVs lifted outside the tolerance band.

These test results were significantly different from other licensees' results reported to NRC recently. The NRC's concern was heightened because of the number of valves affected and the extent to which the disc appeared to be stuck.

As a bit of background, the two-stage Target Rock safety relief valves were the solution to the problems that had been experienced with the three-stage valves. Two-stage valves exhibited some setpoint drift during testing an became a major concern in July of 1982 when 11 of the Hatch 1 SRVs, of which there are 11, failed to open at pressures well above their setpoints following a scram in isolation. Subsequently, three of the valves opened at

1180 psig and the other valves were manually opened later in the transient.

The owners group that was formed after this event, the General Electric Company and the Target Rock Company, investigated the causes of the setpoint drift and developed recommendations for solutions to the problem. This is a closeup of the two-stage topworks.

(Slide.)

MS. WEGNER: This shows the labyrinth scale of the disc seating area.

The principal causes were determined to be galling in the labyrinth seal area, and corrosion-induced seat-to-disc bonding in this area.

MR. MICHELSON: Which area did you point to?

MS. WEGNER: Steam side of the disc. The seat area here.

MR. EBERSOLE: You said corrosion induced bonding?

MS. WEGNER: Yes.

The recommended solutions were an enhanced maintenance program and a replacement disc of a material whose oxide film would be less likely to bond to the oxide film of the disc. The seat, sorry. Both the seat and the disc were of a stellite 6 or 6B.

MR. EBERSOLE: How long a period had it been

before they were unseated? How long does it take for this corrosion and bonding method to stick the valves?

MS. WEGNER: As little as three weeks or it has been observed.

MR. EBERSOLE: Normally these things don't get exercised except once every two months.

MS. WEGNER: The Hatch valves had gone the entire cycle without being exercised.

MR. EBERSOLE: What sort of bonding does that bring about?

MS. WEGNER: All 11 valves failed to operate until three valves lifted at 1180 psig. Since then we have seen some valves, mostly on the test stand, actuate very much higher than that. Or not actuate at all.

MR. REED: Those valves are valves that are brought in from the field. They haven't been exercised but or are they reassembled valves in the test facility? And don't actuate?

MS. WEGNER: Okay. The topworks are taken off of the -- off of the main and sent to Wyle Labs. I don't know whether any other facility can do the test or not. That is the only one I am familiar with. The topworks are not disassembled until they get to Wyle. The first test -- I am getting ahead of myself here, but the first test that they were doing to determine the setpoint was when the

valves came in, they had a body there at Wyle. They took the topworks, mounted it to that body, heated it up until it was in thermal equilibrium and simulated reactor conditions. They did a -- a long time ago they used to do a full-flow steam test on it.

MR. REED: They did an as-found-from-the-field test.

MS. WEGNER: I am not talking about the Brunswick test. This is the previous test. There has been some changes which create some problems. I am getting ahead of myself there.

MR. REED: What I want to know is, how did they do the Brunswick test where the relieve-it point was much elevated?

MS. WEGNER: The Brunswick test and the current test are done, were done first by backing the stem off of the disc so the disc would be free-floating, pressurizing under the seat here with nitrogen to determine how much the disc may or may not be stuck. A free-floating disc shouldn't take any more than 5 pounds of nitrogen to lift it.

The facility limit at Wyle was around 200 psi. Six of the Brunswick valves did not lift within the facility limit. They shut down the test.

MR. REED: That is the point I want to key in on.

These are what I call internal pilot-operated relief valves. I don't have much use for those kind of valves unless the environment in which they work has been studied. This environment in which these valves work is probably an environment of hydrogen and oxygen and steam in a condensing recirculating mode. And stripping of the oxygen and hydrogen is taking place up in those cap areas, I would expect. Therefore, when you talk about corrosion bonding, I think you have a very nice set-up for it. You have high oxygen and hydrogen and you are probably going to get all kinds of corrosion and friction factor changes like crazy. I am not so sure that this is a good application of a valve.

MR. EBERSOLE: It is not apparent what the main motivating force is for opening the main bonnet.

MS. WEGNER: This disc, the pressure under this disc, when it exceeds the set pressure, is determined by the spring and bleed off the pressure here. And that allows this to compress its spring and open.

MR. EBERSOLE: So there is an opening factor to the main steam pressure on the big piston. The steam pressure applied to the large piston against the spring. You simply don't show that in the illustration there.

MS. WEGNER: It is applied to the pilot.

MR. ALLISON: The question is, what lifts the main disc?

MR. EBERSOLE: Yes.

MS. WEGNER: Frank, can you help me on this? As I understand it, when lifting the pilot takes the pressure off here and allows the main --

MR. EBERSOLE: It is just through shaft leakage?

MR. CHURNEY: Frank Churney from NRR. It is not shown on this particular slide. There is a tinge on there called the plain piston. It is about the center of the picture. When you open the pilot, you create a delta P.

MR. EBERSOLE: Where is the supply side of the delta P?

MR. CHURNEY: Maybe the picture doesn't show it real well. There is a supply side to this.

MR. EBERSOLE: Okay.

MR. CHURNEY: You bleed it off and get enough of a delta P so it pops off.

MS. WEGNER: This maintenance program that I was talking about has been in effect for at least one fuel cycle at most of the plants. Since the issuance of information notice 82-83 in December of 1983, the testing results that we have seen for a typical plant show a small number of valves within the tech spec limits which are set pressure plus or minus 1 percent. The majority are within a set pressure plus or minus 5 percent. And most plants would have a single valve stuck. Some maybe more than one.

MR. REED: So enhanced maintenance is not the answer to the question.

MS. WEGNER: It is not the entire answer.

MR. REED: I will go right back to say the problem is the environment in which the valves are being asked to function and the nature of the valve.

MS. WEGNER: It is beginning to look like that very much. As the data base grew, it was apparent that the setpoint drift hadn't gone away. The director of division of licensing, NRC NRR, sent a letter to the owners group in March of '85 recognizing the benefits of the enhanced maintenance program but concluding that maintenance alone was inadequate to solve the problem.

The next step taken by the owners group was the selection of a replacement material for the pilot disc which would have less tendency to develop this oxide bond with the seat. The material which they eventually selected was a precipitation-hardenable stainless steel disc. The replacement disc had recently become available and had been installed in 50 percent of the Hatch 1 and the Brunswick valves which were tested at Wyle towards the end of 1985. And the beginning of 1986.

MR. MICHELSON: Wasn't this the same problem that they had with the three-stage valve and the sticking problem?

MS. WEGNER: No, sir. The three-stage valves popped open and stayed open.

MR. MICHELSON: They had a problem also of gluing shut.

MS. WEGNER: I really don't recall that. I know that has been a problem with the electromatics but not these.

MR. MICHELSON: To fix them from going from three-stage to two-stage was just to get away from the sticking open situation?

MS. WEGNER: That is right.

MR. MICHELSON: Now they got one that sticks closed.

MR. REED: I think designers traditionally try to look for different materials but if you haven't changed the environment, is anyone looking at plugging the capillary connection to the pilot and putting a loop seal in that, have an external line come out on the pilot and put a loop seal in there to prevent the oxygen/hydrogen concentration phenomena?

MS. WEGNER: No. I have heard mention of putting a pressure switch in there so that the valves would open at a certain pressure, no matter what. But that, too, is talk.

MR. MICHELSON: The problem is that I think when

you open this valve, you will blow your loop seal out because it vents at the bottom of the pilot there.

MR. REED: It certainly will refill. As long as the valve functions properly and seal the hydrogen and oxygen concentration, maybe it will work. I am just making a suggestion. Here is another one of these things that we have talked about. We talked about them among PWR and the problem of internal operating valves.

MR. MICHELSON: Well, if we knew what to swing on, we would.

MR. EBERSOLE: These valves are typical of boilers.

MS. WEGNER: Yes. Quite an enlarged number of the boilers have the Target Rock's. The PWR relief valves, the majority of them are just spring lift valves that I have --

MR. EBERSOLE: They are not -- well, but you have electric --

MS. WEGNER: I am talking about the main steam safety.

MR. EBERSOLE: I am talking about second --

MR. REED: She is saying the majority are spring action.

MR. CHURNEY: There is one PWR that has pilot-operated safety valves on the primary side. That is

Beaver Valley 1.

MR. REED: Did you say secondary?

MR. CHURNEY: On the primary side. They aren't exactly like this design. My understanding is they are also a two-stage valve. They are more like the valves that are used in the Navy.

MR. EBERSOLE: They have to face the problem of corrosion?

MR. REED: Well, you got your answer.

MR. CHURNEY: They haven't had any real problem with theirs.

MR. CHURNEY: A big U-shaped arrangement of the pipe itself.

MR. REED: You can just do it on the pilot.

MR. CHURNEY: It could be a little tricky on these because they are very small valves. They are right on very short rises on the steam lines.

MR. MICHELSON: I think they clear the loop seal also when they actuate.

MR. EBERSOLE: The valves on the PWR that you state are there, those are not the safety valves, are they?

MR. CHURNEY: On Beaver Valley 1, there are three safety valves on the pressurizer. Not exactly this design, but they are pilot-operated.

MR. EBERSOLE: And they are in a borated

environment with loop seals?

MR. CHURNEY: Yes.

MR. REED: I believe that is a Westinghouse design?

MR. CHURNEY: It is the only one in the country with those valves.

MR. REED: Did you ever ask them why they went to loop seals on pilot-operated relief valves, which is also on another new plant that we just reviewed here about six months ago.

MR. CHURNEY: Well, they have gone -- that is true, Westinghouse has gone to pilot-operated PORV on some of the new plant with --

MR. REED: With loop seals.

MR. CHURNEY: I haven't checked on that.

MR. REED: We had another licensee come through here four months ago.

MR. CHURNEY: I would imagine it is for the reason you state, to prevent against the boration problem.

MR. EBERSOLE: On the secondary sides of the PWR, they have safety valves and then they have to have electromatics to intercept. What do they use for the pilot or manual remote operation of the electromatics?

MS. WEGNER: Did you say BWR?

MR. EBERSOLE: PWR. The ones which are manual

remote operated.

MS. WEGNER: PORVs?

MR. CHURNEY: They are large PORVs. A lot of them are large air-operated type.

MR. EBERSOLE: They are not required to be safety valves.

MS. WEGNER: A lot of them use their safeties for that.

MR. CHURNEY: They have large relief valves on them.

MR. EBERSOLE: But they are just straight air-operated. They are not given the prerogative of being safety valves?

MR. CHURNEY: Yes.

MR. EBERSOLE: Yet they are probably better than this.

MR. CHURNEY: The thing is, it would be very difficult to change these out, for example, and put the large valves on, that are on the BWR 5 and 6s which are more like the big spring safety valves. These are about one-third the size of those. It would be a very difficult thing to change these out. These are mostly on the BWR 4s and a couple of the 5s.

MR. EBERSOLE: I guess they don't test these things every X days because they will stick open.

MS. WEGNER: After the Hatch incident, there was a requirement for Hatch to open the valves. I believe that was their finding. They did stick open.

MR. EBERSOLE: That is a stiff penalty for them to fix the valves.

MR. CHURNEY: As she was saying earlier, some of these things start to stick in a couple weeks. So in terms of finding the frequency that would really be from a cost-benefit point of view advantageous, it is not really obvious that it really buys you that much.

MR. EBERSOLE: You buy some trouble when you stick them open.

MR. REED: I could make a suggestion. We did this at Yankee Rowe 25 years ago when we were studying why pilot-operated relief valves didn't work. You could take an external thermocouple, put it on that solenoid outside the body of the operator, pilot operator, and follow the temperature reduction in time of the external casting of the pilot. That would begin to tell you when you have saturated the space with hydrogen and oxygen and you have stopped the condensate flow and the steam flow up in there, and it will tell you how long it takes to get the sticking condition.

In other words, you have got pure hydrogen and oxygen up there. I expect it doesn't take too long. It is

amazing how low those temperatures will get. I saw the temperature get hot on Yankee Rowe to 280 degrees Fahrenheit. That tells you something is happening.

MR. CHURNEY: That is certainly one variable. On the other hand, most of the test data that we see has a whole spectrum of variation. These things don't just get to one temperature and all of a sudden stick. There is something else going on there, too.

MR. REED: If you have a microleak, you will not get complete strip out. You will continue to keep condensation and keep the temperature up and you won't get solid --

MR. CHURNEY: Those that do leak, and there seems to be fewer and fewer of them as we go along in time, they don't stick as bad.

MR. REED: Hey, you got your answer. All the answers are right in what you just talked about here today. All you got to do is stamp it with a conclusion.

MR. CHURNEY: I think the owner group solution, if it turns out that changing the material to one that won't bind, that sounds like a pretty good option. It is a fairly cheap option.

MR. EBERSOLE: We could use a buffer gas --

MS. WEGNER: That is one of the gases, the particular element in the environment that is giving them

trouble.

This is a detail of the nitrogen lift pressures of each of the valves at the Brunswick plant as per steam line. It is interesting to note that the C steam line had the best track record. It is even more interesting in view of the fact that the C steam line was isolated for six weeks prior to the outage and was opened approximately a week before the outage. Also mentioned was a minor water chemistry transient during that period of time.

MR. MICHELSON: I am confused.

MS. WEGNER: It may not be related.

MR. MICHELSON: Isn't the safety valve on the reactor side of the steam line isolation valves? So they are all pressurized all the time. The fact that you aren't drawing steam through the C line might be interesting or significant. That is all this is saying. That line was inactive.

MS. WEGNER: The line was isolated --

MR. REED: Was there leakage through the valve to the main disc on the C line?

MS. WEGNER: No. None of these valves leaked. On the vessel.

MR. REED: No microleakage or anything?

MS. WEGNER: There was no leakage reported. They have acoustic detectors as well as temperature

detectors on the tailpipes to determine leakage. I am not sure of their sensitivity. Temperature rise is what used to be used at Browns Ferry to determine when we were, when the valves were leaking. They were fairly sensitive.

MR. MICHELSON: How does the tailpipe temperature detect --

MR. REED: I think I have a theory. If you do not pass the steam through that C line and if you have the intimately connected safety valve, no steam is going through there, then stochiometric quantities are not going through there. You are not feeding the cells, so to speak. You have stopped the flow. And that may relate to it.

MR. MICHELSON: You can also argue that the hydrogen and oxygen is rising up and collecting in that area. It is a noncondensable gas. Therefore it is all hydrogen and oxygen up there.

MR. EBERSOLE: That is the other side of the coin.

MR. MICHELSON: So it also depends on how much that main poppet is leaking.

MR. EBERSOLE: Why do you get sticking on the pilot seat without the counterpart and much more serious sticking of the main seat?

MS. WEGNER: The main disc? It doesn't see the steam.

MR. EBERSOLE: Sure it does. The back side does. Is it that it is a better seal and doesn't leak?

MR. MICHELSON: It is that this is a Swiss watch. The pilot of the Swiss watch and the other is a large gear, so to speak.

MR. EBERSOLE: That brings up the old Salem problem of what are the force furctions and what is the margin to do. This is -- do you remember Salem, the margin of force to make the circuit breakers open? Maybe the margins of force to execute valve clearance are not enough for the pilot.

MS. WEGNER: This is a very thin film that forms here. However, the area in which it forms is very small, too.

MR. EBERSOLE: What is the forcing function that makes it open? Is it big enough?

MS. WEGNER: For one thing, this side of the disc doesn't see the --

MR. EBERSOLE: I am talking about the pilot function over here. When I have an overpressure transient, what are the forces versus the static friction and gluing forces that I have to overcome? What is this force balance in this design and what are the margins to overcome this sticking?

It is analogous to the old UV trip on the Salem,

the ATWSS case, where they had to analyze what force do I need in this case to make those UV trips work versus how much spring tension I had to do that. It is an analysis here of one of the forcing functions to make the valve clear versus the sticking forces I do have to clear. Evidently there has been no analysis of the design in this context.

MS. WEGNER: Frank, I don't know about that exact question.

MR. CHURNEY: I think the way that the main disc is opened, that what you are talking about there is that there is much more margin -- I don't think that anyone has done anything real analytical with regard to this. I think the operating experience, though, in the way this bonding forms, I think it is fairly intuitively obvious that what you are saying is correct.

MR. EBERSOLE: It is almost, it is a mechanical version of the Salem case.

MR. MICHELSON: Yes.

MS. WEGNER: I know that Salem had a problem. But I am not familiar with the --

MR. EBERSOLE: I am talking about the breakers. But it is the same old thing. I got to move something. With what force do I do it?

MR. CHURNEY: When it doesn't corrode, it works

fine.

MR. EBERSOLE: But you have to make an allowance for corrosion?

MR. CHURNEY: That is what we are trying to get rid of. In order to do that, I think there would be much more substantial design changes.

MR. EBERSOLE: I am sure.

MR. CHURNEY: That is why I said, changing the material, assuming that works, is probably the cheapest way.

MR. EBERSOLE: Maybe you just need bigger pistons.

MR. REED: I tell you what bothers me: We talk about root causes, what really causes valves to malfunction, and we try to set up committees to search for root causes. And we got root causes all around us. But our recognition is pretty damn poor. I have had recognition of this problem for 25 years. And I have been talking about it for 25 years. Nobody knocks me down. Any time one of you guys want to knock me down about the environment of hydrogen and oxygen and boron and what it does to internal operator relief valves, I would like to be knocked down because I will get off this kick.

MR. EBERSOLE: They will wait until they get a sticking of so many of these. It is just like the list we just had on check valves.

MR. REED: Are we lacking perception to get to root causes?

MR. EBERSOLE: We are not sensitive to anything unless it produces a drastic physical consequence.

MR. REED: I think we are sensitive but we lack the perception to get the real root causes. I could comment on Three Mile Island with respect to that.

MR. MICHELSON: I think one of the difficulties might be, I don't believe the agency has any activities on these types of valves like we do on some of the other valves -- no research activities. I think you are depending upon the industry to be doing this and so what we need to inquire into is what is the industry really doing about it? And what is the status of the situation? I don't know if you people have gotten to that stage yet or not.

MR. ALLISON: I think what Frank was just telling you is what they are doing. They are changing material. The previous work they have done on the labyrinth seals seems to have solved that problem and the material change will solve the problem and that will be a good solution.

MR. CHURNEY: The quickest test program they could come up with was the test on the reactors.

MR. ALLISON: It is pretty close to a root cause

in that you have the seat and the disc are the same material and so what we think is happening is that the oxide films are the same, they are tight films and so they bond together.

MR. EBERSOLE: I didn't hear you say they were the same material.

MR. CHURNEY: They are not exactly the same.

MR. EBERSOLE: Isn't that a no-no in any design?

MR. CHURNEY: I don't think so.

MR. ALLISON: A lot of steam system valves have stellite discs.

MR. EBERSOLE: Okay.

MR. ALLISON: So if the new material is thought to have a weak fragile oxide layer, it is different. And that is the theory anyway.

MR. REED: To me the root cause is the environment, the oxygen and hydrogen, and you choose to build the dam down stream of a different material.

MR. ALLISON: Well, if the material works, you don't have a root cause.

MR. REED: You mean, you don't have an event.

MR. ALLISON: Right.

MR. REED: But you just said the root cause which is hydrogen and oxygen. You really haven't done anything about that.

MR. EBERSOLE: We got to move on. We are going to run out of time.

MR. ALLISON: One last thing, these are the set pressures plus the nitrogen pressures. These are the steam pressure plus the nitrogen pressure for the Brunswick valves. And a recent test on the Hatch 1 valves that had the initial incident can show you basically what BWR water chemistry control can do towards solving part of the problem.

MR. MICHELSON: Have any of the valves failed to open at all at something approaching what the code would say they must open up, the safety valves?

MR. ALLISON: I think the events we are talking about fit what you are asking.

MS. WEGNER: The original Hatch incident had eight valves that were supposed to open from 1080 to 1100 and failed to open, period.

MR. MICHELSON: But as safety valves, they didn't have to open in that range, did they?

MR. CHURNEY: According to the code analysis that was done for Hatch, what she said is correct. They opened quite a bit later.

MR. MICHELSON: Did they have to open at those levels?

MR. CHURNEY: It just so happened that the

particular transient involved was a very mild transient, very slow. So we have never had an event where the code pressure limit has been exceeded.

MR. MICHELSON: That is the question I am asking.

MR. EBERSOLE: Do the ASME code committees on valve, are they aware of the evolution of these valves in actual practice?

MR. CHURNEY: WE haven't reconsidered that. We have substantially increased the initial qualifications of the last couple of years for all the safety valves.

MR. MICHELSON: Does the code address this particular design or is it simply the need for code safety and certain requirements as to what they must meet?

MR. CHURNEY: We have not taken any action to prohibit the use of pilot-operated safety valves. One thing that has come up. You have to realize the way the code is written. The code has to worry about light water reactors. One thing that has come up is the concern about how spring actuated safety valves on liquid. They work lousy on liquid. If you are going to have any liquid transients, these are much better. So there is a lot of tradeoffs.

MR. MICHELSON: So the committees are not yet real worried but they are watching the situation? Is that a correct appraisal?

MR. CHURNEY: They are watching the situation cautiously. As a matter of fact, there used to be a penalty on the use of pilot valves. You had to put more of them on the system. Recently the penalty was removed but the initial qualification requirements were substantially increased.

MR. EBERSOLE: Any further comments on -- what is the committee's feeling about taking this matter to the full committee? Would anyone say no?

MR. REED: I agree with you except that I think we need to keep all kinds of pressure on to get the fragmented valve, this misapplied valve situation straightened out. I think it is one of my burning issues and one I wasn't allowed to put on paper because I only could have one.

(Laughter.)

MR. MICHELSON: You didn't choose to put it on paper.

MR. REED: It wasn't the number 1.

MR. EBERSOLE: Let's go to the third item.

MR. KEISSEL: I am Dick Keissel. I am with the the Office of Inspection and Enforcement. I am here to talk about a series of recent check valve failures that occurred at Turkey Point units 3 and 4. These check valves were in the steam supply system to the aux feed pumps and

as you know, Turkey Point only has steam-driven aux feed pumps at this time.

The safety significance of this problem is the fact that it could have prevented the aux feed system from functioning properly at Turkey Point, but of a more sweeping nature is that it points out again what happens to check valves, and in this particular case, stop check valves, when they are subjected to very low flow conditions.

By way of history, in late November of 1985, Turkey Point became aware of the fact that one of the 12 check valves in question was inoperable. They then ran radiographs of the other 11 and found similar failure to two more of the check valves. When they opened up the valves, they found that all of the valves were experiencing signs of mechanical damage from low flow or what they attributed to be low flow conditions in the lines.

They replaced the stems -- the disc and the disc guide in 10 of the 12 valves and went back into operation.

Then in early January, they discovered that, again, one of the valves had failed. They x-rayed and this time they found that a total of four valves out of the 12 had failed. They failed in a similar manner.

In addition to making the systems inoperable, the loose parts which were generated by this could also have done damage to the turbine pumps.

Because of the generic nature of the problem, that being that we have a potential misuse of this type of valve in an information notice, 82 -- I am sorry, 86-09 was issued, this also referred back to a previous information notice, 82-26 which addressed a similar problem with the steam exhaust valves on the HPSI and RPSI systems which were being subjected to low flow conditions when the pumps were run in a test mode.

MR. EBERSOLE: I think there is no question. We want to take this to the full committee and present it as an immediate backup to the relevant problem of check valve failures that we spent the whole morning on. This was again a case of reduced flow and the chattering and damage to valves which caused five of them to fail concurrently. This is just another case of the same sort of event that happened at SONGS.

MR. MICHELSON: You just said the licensee thought reduced flow was the problem. I haven't heard any explanation of how reduced flow can cause a failure of this type of valve.

MR. EBERSOLE: I read low flow caused vibration and chattering.

MR. MICHELSON: It is a little harder to see how low flow damages this valve.

MR. KEISSEL: The manufacturer of the valve

recommends that they not be used where they see flows of less than 10 percent. These valves are used as isolation valves around the main steam isolation valves to the pumps. And they attribute the low flow condition to leak-by of the isolation valve itself.

MR. EBERSOLE: Were you here this morning?

MR. KEISSEL: No, sir.

MR. EBERSOLE: The failure of the five valves was attributed to a 10 or 15 percent flow reduction.

MR. MICHELSON: These are stop checks. Look at the drawings.

MR. EBERSOLE: These were deviant from the full design flow conditions just as this is. At least this is blamed as a causative factor that causes resident action and damage to those valves which in a general context is what you are telling me here.

MR. KEISSEL: Yes, sir.

MR. EBERSOLE: Does the valve vendor state that he will not warrant his valves at flow rates less than X?

MR. KEISSEL: I don't know about the statement that he will not warrant the valves below a certain amount. He recommends that they not, that this style valve not be used with less than 10 percent flow.

MR. EBERSOLE: So there are full requirements on all valves like this? On those valves? You have full

requirement, a recommendation rather than requirement, that they not use these with flows in that range?

MR. KEISSEL: Right. The unusual thing here is, of course, we are -- when the system was designed, it was assumed there would be zero flow. And what we were experiencing here was just leak-by of a closed gate valve.

MR. EBERSOLE: What is the consequence of failure of these valves.

MR. KEISSEL: The system would be declared inoperable -- well, if the valves failed in the open position, there is a possibility that the system would not respond properly to a line break upstream of them. And you couldn't, because you need to be able to pressurize with steam from the other plant.

Let me show you. I have a schematic of the aux feed system.

MR. EBERSOLE: It would leave it as an open system then on aux feed?

MR. KEISSEL: Yes, sir. Now you have three aux feed pumps that share their steam source between the two units. If there were a steam -- if there were a line break upstream, up in this area, failure of these valves then prevent being able to service the pumps with steam from the other unit.

MR. EBERSOLE: I see. So I lose the header

pressure for steam supply to the aux pumps?

MR. KEISSEL: That is correct.

MR. EBERSOLE: On a single point basis. Just the failure of one of those valves?

MR. KEISSEL: No. Both of them.

MR. EBERSOLE: Why is that?

MR. KEISSEL: Well, if one of these holds, then a breakdown here would be isolated.

MR. EBERSOLE: I thought they were cross-connected.

MR. KEISSEL: The cross-connect is through these headers here. So that your steam flow pattern would be from one unit through the header and out.

MR. REED: I am having some difficulty recognizing this valve that is called a stop check load valve as a stop check. What it looks to me like is a not too well designed and is used as a stop valve. How tight is that little thing down there? It looks to me like it is tight.

Now, didn't you just say that the manufacturer says that it shouldn't be used at flows under 10 percent and didn't you say that its real application is to be open or closed? Not to throttle?

MR. KEISSEL: That is correct. It is not a throttling valve.

MR. REED: Well, then it is a stop valve.

MR. KEISSEL: No, sir. It is a stop check. The valve is in the normally open mode and, therefore, functions as a check valve.

MR. REED: It is normally open and -- well --

MR. KEISSEL: And therefore, it will act as a check valve, blocking flow above the disc but permitting flow from below the disc.

MR. MICHELSON: The drawing is showing it in the locked/closed position.

MR. REED: I see there is some way that this assembly can move up and down on the lower part of the stem that isn't obvious in the drawing.

MR. MICHELSON: Well, you know how they work. It is obvious, but you have got to crank it out. It is in a locked position.

MR. REED: Do you see what I am talking about?

MR. KEISSEL: Yes, sir. If this stem is pulled back up, then the disc is allowed to ride up on it. And it is a bit confusing in that the conventional globe valve would have the stem actually filling this piece down here, giving you the color which would permit you to pull it off.

MR. REED: And a little looseness down there for a seat.

MR. MICHELSON: This one slides up and down.

The damage can only occur if the flow is so low that the valve, the weight --

MR. KEISSEL: It just sits here and simmers.

MR. MICHELSON: That will wear the seat out. But that is an altogether different problem than having damage to the thing from being in the open position and banking against something. This thing here, as long as it isn't fluttering on the seat -- well, what is happening is, it is breaking off that guide stem because it is fluttering enough to --

MR. EBERSOLE: I fail to see why one of these valves failing would deny steam supply to the aux feed pumps. Did you not say that was the case?

MR. KEISSEL: No, sir. It provides a path by which steam can be lost. And it is not just one. They would have to take out two.

MR. EBERSOLE: I didn't see where the blocking device was to prevent -- these are not commonly headered together. I see other check valves up there. Now, it takes more than one valve failure to deny steam pressure to the turbines of the aux feed pumps, right?

MR. KEISSEL: Yes, sir.

MR. EBERSOLE: It would take more than one valve failure?

MR. KEISSEL: Yes.

MR. EBERSOLE: So our problem would only be in front of -- this is a safety problem if we lost more than one?

MR. KEISSEL: Yes, sir. But we are talking about a common mode.

MR. EBERSOLE: You lost one and you might have, with a little less luck, lost them both?

MR. KEISSEL: That is correct.

MR. EBERSOLE: If I had lost two, would I have lost my turbine pumps?

MR. KEISSEL: You could have lost your flow to them assuming that you also then did not have your --

MR. EBERSOLE: I believe you said at Turkey Point, I must depend on these three turbine pumps. I don't have any motor pumps?

MR. KEISSEL: That is correct.

MR. EBERSOLE: So we have another mode of loss of inventory to turbine-driven pumps?

MR. KEISSEL: Yes.

MR. MICHELSON: Where are the stop checks on your drawing?

MR. KEISSEL: Stop checks are this column of valves and this column of valves. (Indicating.)

MR. MICHELSON: You are sure those are stop check and not just open gates?

MR. KEISSEL: They are defined as stop checks and they were --

MR. MICHELSON: Those are normally just maintenance valves for the purpose of repairing that motor operator valve. I didn't think they were stop checks at all. I don't know why they are.

MR. KEISSEL: I don't know if this drawing is the current one. These valves were installed in 1983 or 1984. And they may have been installed as a replacement for the valves that are there.

MR. MICHELSON: I would have guessed the stop check was the valve to the right without a number on it.

MR. EBERSOLE: Well --

MR. KEISSEL: No, sir. Because the --

MR. MICHELSON: Go to the topmost line. I thought that was the stop check.

MR. KEISSEL: No, sir.

MR. MICHELSON: Okay. I will stand corrected. I am just surprised.

MR. KEISSEL: These were the valves and they have been checked by the number. The reports were by valve numbers. These are the two that failed on unit 4. These are the two that failed on unit 3 in January.

MR. MICHELSON: They don't need to be check valves. Not by the original design. The checking function

is provided by the check valves shown on the drawing.

MR. EBERSOLE: I am looking at the two check valves that are on the right side of your drawing there.

MR. KEISSEL: Just there and there. However you also have a steam cap back this way. It is a very -- one word might be sophisticated, another word, complicated system.

MR. MICHELSON: I don't think there is a steam path back that way though, from the drawing, unless mine is different than yours. That is the same line, it just branches into two branches. If you go further to the right, then there is another check valve.

MR. KEISSEL: I don't see the other check valve.

MR. MICHELSON: We have probably exceeded the usefulness of this discussion?

MR. EBERSOLE: Any comments, anybody?

Okay. Thank you.

Before you get to this presentation, I suggest that we include in our full committee presentation with the added aspect of defining the degree of jeopardy to aux turbine feed supply which was not at all clear from the presentation from the E&ID. Okay? I don't know what the degree of jeopardy here is. It was quite clear in the SONGS case what was with us. But it is not here.

MR. ALLISON: We can explain that a little more.

MR. MICHELSON: Would you reconfirm with the licensee which are the stop checks on the drawing and why they are stop checks when they are shown as gates on these drawings.

MR. ALLISON: We will do that. I think the discussion of the back leakage has tended to drag out a little bit. I think probably a more realistic threat is loose parts in the turbine.

MR. EBERSOLE: That could be, thank you.

MR. LICITRA: I am Manny Licitra with NRR. I am here to discuss an event at Palo Verde unit 1 on January 9, 1986. Palo Verde unit 1 at the time was in the last stages, final stages of its power ascension program. It was attempting to do a turbine trip test from 100 percent power. For that test, there is a certain feature in the plant called a reactor cutback system. For that test -- they, this enabled the reactor cutback system, because it had been successfully tested from a loss of load test at 100 percent -- I also believe they tried it at 80 percent and it succeeded at that level -- so for this test, they did not have the reactor cutback system in operation. What they expected was to have a reactor trip due to high pressurized pressure. That is, the turbine would trip. There would be an automatic transfer of nonessential loads to the grid. The pumps, reactor coolant pumps would stay

on and heat would continue to generate. You raise the pressure and you get a trip due to that event.

What they got instead is that the house loads did not automatically transfer to the grid because of a frequency mismatch between the grid and the 1E buses.

MR. EBERSOLE: Was this a delayed -- we just got through talking about the necessity to hold until you had a voltage decay or you had a very rapid transfer. Which kind of transfer is this?

MR. LICITRA: It is a synchronization match. There is also a time element in there. I don't fully understand the circuit myself.

MR. EBERSOLE: I think if they do it fast enough, they do it fast enough before it gets out of sync. If they wait too late, they have got to wait for a voltage decay.

MR. LICITRA: I think time comes into it.

MR. EBERSOLE: You don't have any idea of the time spans we are talking about?

MR. LICITRA: No. What I do know is that the system performed as designed. It was not a fault in how it was set up. It was set up according to design, but because of the way it was designed, it did not allow this transfer. When you didn't get the transfer, you lose power to the reactor coolant pumps, the main feedwater pumps, the circulating water pumps, the steam dump for the bypass

control system. They are all on non-1E buses.

MR. EBERSOLE: Aren't the main feed pumps turbine driven?

MR. LICITRA: At least one is. I think they both are.

MR. EBERSOLE: So when you say you lose the main feed pumps, you mean the booster pumps?

MR. LICITRA: I believe so, yes.

Now, when that happened, the pump didn't have power. It is coasting down. And that caused a reactor trip due to a projected, flow projected low DNBR.

The steam bypass control system valves did receive a quick open signal but wouldn't close because they lost power.

One of the four lowest setting main steam safety valves lifted when the steam generator pressure increased rapidly. And the setting for those lowest safety valves is 1250 psig. But only one of the four lifted. It stayed open for about 43 seconds. When that valve receded, five steam bypass control system valves modulated open, stayed open for about 45 seconds, closed and then they had two additional open/close cycles for the next 40 seconds.

MR. EBERSOLE: How did they operate when they refused to operate just prior to that? You said but reclosed almost immediately due to loss of power.

MR. LICITRA: I believe it is the quick open signal that is on the nonsafety bus.

MR. EBERSOLE: Are you telling me the bypass will open in the absence of circulating water? Until you get condenser, low vacuum?

MR. LICITRA: That is what happened.

MR. EBERSOLE: That is strange, isn't it? You denied the rapid opening right away due to loss of power. Is that a -- is that the forced function, the loss of power that denies bypass?

MR. LICITRA: That is what denied it?

MR. EBERSOLE: It wasn't loss of vacuum?

MR. LICITRA: Well --

MR. EBERSOLE: If it isn't the loss of vacuum, I guess I will ask you why, because as long as you have got vacuum, you can bypass.

MR. LICITRA: I guess I do not have an answer to that.

MR. EBERSOLE: Okay. Go ahead. And then later, you opened -- they modulated open again.

By the way, you mentioned that the safety valves, the main steam safety valves lifted. What about the electromatics that are interposed between the safeties and the steam pressure. They are the ones that Palo Verde advertised as being QA and safety grade, et cetera. Why

didn't they intercept the opening of the main safety valves on the secondary side or are they, are these valves serving no purpose?

MR. LICITRA: Only one of the four valves opened.

MR. EBERSOLE: Is it a dual purpose electromatic or PORV safety valve or not? You say safety valves.

MR. LICITRA: Yes, they are the safety valves.

MR. EBERSOLE: Are they the PORVs, or are they electromatic or are they intended to be operational in front of the safeties?

MR. LICITRA: As you know, Palo Verde does not have PORVs.

MR. EBERSOLE: I am talking about on the secondary side.

MR. LICITRA: I am not able to --

MR. EBERSOLE: I have to infer from what you haven't said is that these are multipurpose valves that perform both a safety function and a pressure function which they claim is safety grade. Is that true?

MR. LICITRA: The atmospheric dump valves?

MR. EBERSOLE: That is the one I am talking about.

MR. LICITRA: These are safety valves.

MR. EBERSOLE: The atmospheric dumps, then, did not work because they are not automatically piloted to

preclude opening of the safeties? Do you follow me? We put electromatics or dump valves on the secondary side to intercept opening of the safeties because we had reclosing problems with the safeties. All right?

What did they do? You tell me the main steam safety valves lifted but you didn't tell me what the dump valves did.

MR. LICITRA: Let me see if I can get an answer out of the report.

MR. EBERSOLE: I wouldn't look it up. I would rather have you plow on through it and we will look at it later.

MR. LICITRA: Okay.

MR. REED: You are right. Normally the atmospheric dumps will open for the safeties the same as the safeties, unless the pressure, rate of pressure rises too fast. Then the safety will go.

MR. EBERSOLE: Right.

MR. REED: This is a strange trip here. It is loss of a lot of vital equipment and it may be that the atmospheric dumps also went open but couldn't take care of the release quantity. So a safety valve opened.

MR. EBERSOLE: If so, it is not mentioned. Let's go ahead. We won't have time to pursue the details. Go ahead.

MR. LICITRA: Well, after the steam bypass control system valves did modulate open, the operators at that time took manual control of the individual valves and stopped a cooldown. Simultaneously or about the same time, they got a main steam isolation signal due to low steam generator pressure. After the cooldown stabilized, the operators reestablished cooldown via a steam generator, two atmospheric dump valves and one auxiliary feedwater pump.

MR. EBERSOLE: How did they get low steam generator pressure? It was excess opening of the bypasses? Was it --

MR. LICITRA: They didn't have those valves open. They were losing steam through the valves.

MR. EBERSOLE: When you say that they were under manual control, they modulated open. They stayed open for 45 seconds. And is that, in the course of doing that, did they lose pressure control? You say -- the bullet below that says manual control was taken?

MR. LICITRA: About that time, they got a main steam isolation signal, automatic signal.

MR. EBERSOLE: Due to low pressure?

MR. LICITRA: Yes. They tried to take over control to stop that from happening but it occurred.

MR. EBERSOLE: Did the operators inadvertently let the steam pressure ride through the modulation valves

too long? I am just curious how they got to low pressure.

MR. LICITRA: I do not know the answer.

MR. MICHELSON: The inference in your chart is that the steam bypasses were open when they shouldn't be open. That is how you got the low pressure?

MR. LICITRA: Yes. The steam bypass was open.

MR. MICHELSON: So it must be the steam bypass stuck open?

MR. EBERSOLE: That is right. They modulated open and stayed open.

MR. MICHELSON: But not because of need but because they --

MR. EBERSOLE: They just malfunctioned.

MR. MICHELSON: -- malfunctioned and kept oncoming down instead of closing again.

MR. LICITRA: That is one of the things they are looking into. Why they did open.

MR. EBERSOLE: Had the circulating water pumps started?

MR. LICITRA: It is not discussed in their report.

MR. EBERSOLE: Okay. Go ahead.

MR. LICITRA: About three minutes after the event, at 13:28, they restored power to the 1E buses. And at 14:08, after doing other activities, they got the

reactor coolant system pumps started again. One at a time. And the event terminated at 14:49, an hour and a half later.

MR. EBERSOLE: This is not very well known at all as to what really transpired.

MR. REED: I think it ought to be brought back to the next meeting.

MR. EBERSOLE: I think. I think it is too muddy to discuss it any further because there are no positive statements made about what happened. I didn't learn anything about whether or not there were these highly advertised atmospheric dumps which are supposed to be safety grade. Palo Verde was proud of the fact that they said they had safety grade secondary dumps to control the primary pressure up and down. And yet I see no statement to the effect that they worked at all or they were even there. I think just in summary, we will just go to the next one and anticipate seeing this then.

MR. HERNAN: I think Manny does have some more material as far as what action is going on. I would like to remind you, though, that it is the rules that we bring these things down.

MR. EBERSOLE: This is why I am saying, I don't think we know enough facts here.

MR. HERNAN: We will be more than happy to come back. But I kind of get the feeling that the staff is

being criticized for bringing something down too early.

MR. EBERSOLE: I gather that we don't really know enough yet really.

MR. LICITRA: I just received the LER for this event.

MR. EBERSOLE: I am not criticizing you for not having all the data. I am just saying, I don't think you have. I don't see the picture very clear as it was handed to you.

MR. HERNAN: We know that we don't have all the data. But we didn't want to wait until we do to --

MR. EBERSOLE: There are about four or five blank spots here.

MR. REED: I think a true -- all that is true, Jesse. Also, we have some sensitivity with respect to Palo Verde systems. So you can read our letters and perhaps tailor the presentation.

MR. EBERSOLE: We are most interested in Palo Verde and all of its ups and downs.

MR. LICITRA: The utility has not completed its review of this event. They are still looking at different aspects of it. They don't have a bottom line.

MR. EBERSOLE: We will get a rerun of this, say, next month, maybe, or whenever.

MR. HERNAN: These meetings are every two months.

MR. EBERSOLE: Well, who knows. Whatever. Next chance.

MR. ALLISON: Next meeting.

MR. EBERSOLE: Okay. I am going to put "hold over." Thank you. Let's go to the next one.

MR. HERNAN: Did you want him to discuss the action that is going on?

MR. EBERSOLE: You have a page 2. I didn't realize that.

MR. LICITRA: That just gave the follow-up actions that they are doing.

MR. HERNAN: I think that is important to cover.

MR. EBERSOLE: Carry on. I didn't know there was a second page.

MR. LICITRA: As a follow-up action, they are looking into this synchronization check relay circuit to see what can be done about preventing this problem from occurring. They do not have an answer at this time. But they are looking into it.

Because one of the main steam safety valves lifted, they checked the settings on all the valves and they found it okay. So even though the other three didn't lift and one did, apparently it must have been close to that pressure. So they don't see any problem with the settings on the safety valves.

They are also considering providing uninterruptible power to the steam dump bypass control system so they won't run into this problem. They still don't have a final answer on that.

After the event, and in evaluation, they determined that the reactor coolant system pump goes down faster than design and faster than considered in one of the safety analysis. That being the loss of flow safety analysis.

They also found that because of the fast cooldown, the trip, the anticipatory trip occurred faster than they previously projected, so they are competing effects. Since they didn't know what the overall effect is, they made an assumption that the faster cooldown does take place but trips at the same time. They previously assumed and imposed some penalties on themselves until they complete that evaluation.

And finally, they reran the turbine trip test on the 24th, but this time they ran it with the 1, non-1E loads powered off at the startup of transformer. So they wouldn't have this problem.

MR. EBERSOLE: Okay. And did that one turn out all right?

MR. LICITRA: That one worked as expected, as predicted.

MR. EBERSOLE: I see. All right. Thank you. Any comments?

MR. GIITTER: I am Joe Giitter with the Events Analysis branch of the Office of Inspection and Enforcement. The second event I would like to talk about today is a loss of off-site power that occurred on January 9 at the Palisades plant with one diesel being out of service and only one diesel in the standby readiness mode. Let me go ahead right away to the circumstances.

The initial conditions, the unit was in cold shutdown for refueling for about 40 days. About one-third of the fuel was new and two-thirds of the fuel was reloaded, so there wasn't much decayed heat at the time. The head was positioned on the vessel but not tensioned and the vessel was drained about down to the flange level, which is about 12 feet above the top of active fuel. Diesel generator 1-1 was out of service and would have required several hours to make operable. This diesel right here. Emergency diesel generator 1-1.

The sequence of events, at about 2:41 -- these are p.m. Eastern Standard Time -- about 2:41, smoke was observed from the conduit on the 1A bus in this general area here. It was later found out that this smoke was caused when a cable shorted to ground and heated up some water that had gotten in the conduit, apparently rain water,

and it looked as though it was smoke. The operators immediately de-energized bus 1A which is a nonvital 4160 volt bus, this one right here, with the smoke believed to be coming from here. And then because of fear of potential fire or explosion, they decided to go ahead and de-energize this entire rear bus, this 345 KV bus which feeds these three starter transformers. Before doing that though, they -- at 2:56, the operators started and loaded the diesel generator that was in standby readiness. The diesel generator function is required and supplied power to this bus, vital bus 1D. At that point, at 3:08 they did de-energize the rear 345 KV switchyard bus. That caused a loss or caused buses 1C -- well, that was of course not energized -- 1E and 1B to be de-energized.

They did have an RHR pump on this and all of their decay heat removal was accomplished by tying the diesel on this single bus here.

The licensee declared an unusual event at 3:00 and the NRC operations center was notified at about 3:19.

At 4:50, there was a shift change and the second shift came on and decided that they were going to establish backfeed to all the main buses and at 4:50 they accomplished this by backfeeding from this front bus through the main transformer through the startup transformer down to the vital buses and to the nonvital

buses.

And once that was done, they terminated the unusual event and secured the emergency diesel generator that was tied to the 1D bus.

In summary, it was a loss of off-site power caused by a voluntary disconnect and it appears that the licensee did everything right.

MR. REED: Why do you call it a loss of off-site power? It looks to me like a partial loss at the most.

MR. GIITTER: From -- as far as the plant, as far as these buses are concerned, it is a loss of off-site power, even though they didn't physically lose any of these incoming lines.

MR. REED: Those lines are right in the switchyard at the facility, I assume. They are the main outgoing lines and incoming lines.

I am worried about these statistics that keep piling up on a loss of off-site power and total blackout. I worry that the people that are doing the analytical work are not watching their definitions carefully. To me, I don't see why this is tabbed, called a loss of off-site power. To me it is an internal bus fault in the plant which caused maneuvering and changing of electrical supply.

MR. ALLISON: During the time in question, there was no off-site power feeding anything inside the plant.

Is that right?

MR. GIITTER: That is correct.

MR. ALLISON: No off-site power was reaching into the station.

MR. REED: Simply because the diesel was on.

MR. GIITTER: The unit was shut down. There is no power being supplied back through the --

MR. ALLISON: There was no electricity coming from the switchyard into the plant.

MR. REED: Are you telling me that both these 345 lines are out of service completely?

MR. ALLISON: No, they are in the switchyard. I am saying there is no electricity coming from the --

MR. EBERSOLE: They drove themselves into this thing.

MR. GIITTER: They voluntarily disconnected.

MR. EBERSOLE: That is a management decision to drive into it.

MR. REED: I would like this not to appear as a loss of off-site power.

MR. EBERSOLE: This is not a loss in the context that it was a loss without operator participation. They deliberately drove into it.

MR. GIITTER: There may be a terminology problem.

MR. EBERSOLE: They decided to risk for a short

while that the diesel would hold. And it did. So I agree with Glen that it is really not a loss in the usual context. It is a -- we need to do something about the statistics where you drive into a configuration like this. Somehow we need the statistics to show that.

Any other comments?

I don't think we -- we will not carry this to the full committee.

MR. REED: Agreed.

MR. ALLISON: Mr. Ebersole, we will discuss the definition for the statistics purposes with the staff people.

MR. EBERSOLE: I think that would be very much in order.

Let's take a 10-minute break here. We will be back at 25 after.

(Recess.)

MR. EBERSOLE: We will resume the meeting. I believe the next event is the Robinson. Mr. Requa.

MR. REQUA: Right.

MR. REQUA: My name is Bud Requa. I am the project manager for H.B. Robinson Unit 2 and this afternoon I will talk to you about another loss of off-site power coincident with a diesel generator out of service.

MR. KERR: I thought Mr. Reed persuaded the

staff that this would be a loss of off-site power?

MR. REQUA: In quotes.

In January about the time this happened, on January 28, the plant was in a closedown for refueling for early February. With a diesel generator out of service, a fault occurred on the emergency bus E2 which caused a spike on the instrumentation bus number 4. A false rod dropped signal was received, causing a turbine runback. The reactor tripped on high pressurizer pressure from 80 percent power. At about 9:18, a loss of off-site power occurred coincident with a fast transfer to the startup transformer. We originally worded it that way because at the time we didn't know why we lost off-site power but later we found that the C phase balance relay opened due to a phase mismatch.

MR. KERR: What is the significance of the term "loss coincided with fast transfer to startup transformer." Does that mean that it actually didn't transfer to the startup transformer?

MR. REQUA: Yes. It happened so fast it couldn't tell what, which happened. But it turned out that the -- at the time they didn't know what happened.

MR. KERR: Was the connection made to the startup transformer?

MR. REQUA: The connection was made but refused

because the relay had opened, refusing it.

MR. KERR: Okay.

MR. REQUA: At 9:35 -- I am sorry. The A diesel generator automatically started and loaded on bus E1. The reactor coolant pump tripped and natural circulation was established.

At this same time frame we received a safety injection signal and the MSIVs were found closed.

MR. KERR: What does it mean if there is no off-site power, to have the RRCPs tripped?

MR. REQUA: We didn't have them.

MR. EBERSOLE: When you say "fault on emergency bus E2." The bus fault is a rare thing?

MR. REQUA: I understand this is what they were speculating and they still don't know. There are a lot of unknowns on this one still.

MR. EBERSOLE: Suppose it had been a bus fault on E1, what would the emergency diesel have done then? Can it get to E2 or is it locked out by circuitry? What I am trying to say is, if it had a true bus fault, that really kills that bus. If that happened to be the unfortunate diesel generator which was working, we would be in trouble, wouldn't we?

MR. SWENSON: I am Warren Swenson in NRR. It does not appear to have been a true bus fault per se. I

may be playing with the terminology here, but one of the pieces of equipment for something like this that was connected to the bus fault and they had not -- as the day progressed, they had difficulty determining which specific piece of equipment had produced the fault. The whole bus itself apparently did not go dead, but it was sufficient to lock the bus out temporarily.

MR. EBERSOLE: But it would have precluded loading the diesel onto it.

MR. SWENSON: Apparently, yes.

MR. EBERSOLE: If that had been the case and if that had been the diesel still operating, where would we have been then?

MR. SWENSON: A station blackout.

MR. EBERSOLE: Okay. I just wanted to know how close to trouble we might be. Carry on.

MR. REQUA: At this same time frame, about 9:18, we received a safety injection signal, found the MSIVs closed. This is another big unknown. We don't know what happened first as yet. At 9:35 an unusual event was declared and open telephone lines established between the NRC operating center and the Robinson site.

At 9:46, the B diesel generator was restored. At this time the reactor coolant temperature was 535 degrees F, pressure 2000 psig and the plant was stable.

MR. KERR: What good did it do to have the B diesel generator restored if there was a bus fault on bus E2?

MR. EBERSOLE: Right.

MR. REQUA: I --

MR. SWENSON: This is Warren Swenson. As this event progressed, like I said, it apparently was not a true bus fault but the, they shed the loads on that particular bus. And as the event progressed, they reloaded the equipment back on the bus one piece at a time, taking their time and making sure that there were not additional problems that they didn't drop out the bus again. It was not the bus itself, a separate piece of equipment that caused the failure. They didn't want to repeat the process and lose power.

MR. KERR: Okay. Thank you.

MR. EBERSOLE: I guess sooner or later you will tell me where the fault was.

MR. REQUA: I still can't tell you where it was. They still haven't found out.

MR. EBERSOLE: That is another one of these hanging cases.

MR. MICHELSON: Spurious.

MR. REQUA: At 10:52 the pressurizer heaters were restored. At 12:28, not shown on here, the C steam

generator PORV was manually opened to balance loop temperature for natural circulation. The valve stuck open due to freezing of the instrumentation lines and they got another safety injection signal on high steam line delta P. The valve was closed a few minutes later by opening the air supply valve and the partial blowdown was secured. At 12:55, the E1 bus was connected to the off-site power. At 16:03, the E2 bus was connected to off-site power.

Realigning the electrical system was delayed, as Mr. Swenson said, because they wanted to take extreme care to not load equipment on that may be still full.

The plant was taken to hot shutdown, and a day or two later Carolina Power & Light decided to remain shut down for refueling, which is where they are now. Their follow-up is that they are in a 45-day refueling outage and the licensee is investigating the cause of the loss of off-site power with Region 2 reviewing their findings prior to startup.

Also, the licensee's and INPO team are investigating the high frequency of reactor scrams at Robinson, of which they had four in January. Region 2 was also evaluating this along with Carolina Power & Light, and the results will be evaluated prior to startup.

MR. REED: I would like to just clarify. You say up there, stuck open steam generator PORV. If you were

asked to use the words -- instead of PORV -- what does PORV mean to you?

MR. REQUA: That is a pressure-operated relief valve?

MR. REED: I believe the terminology is -- PORV is power-operated relief valve. I have seen in various descriptions around here and a lot of -- and some news releases, that PORV means "pilot." I want to make sure we have our nomenclature straight. I would have called this an ADV. That is what it is, isn't it, an atmospheric dump valve? Isn't it?

MR. REQUA: Yes. Yes.

MR. REED: We sort of, I think the terminology goes like this, that PORV or power-operated relief valves are on the primary side of a PPWR. ADVs are on the secondary side. Just nitpicking, thank you.

MR. REQUA: I appreciate that.

MR. EBERSOLE: Let me ask why a signal is produced caused by closure of main steam isolation valves to produce a safety injection signal, since the first consequence of that is to lose heat transfer and to cause a pressure rise in the primary loop which has nothing to do with inventory.

MR. REQUA: That is another one of these mysteries that have to be solved. We don't know; they do

not still at this time know why they got that safety injection signal. This was one of the reasons why they didn't want to --

MR. EBERSOLE: So that is spurious?

MR. REQUA: Right.

MR. EBERSOLE: The second signal caused by stuck open, that is due to contraction of the -- or you don't say, it hasn't yet -- there is nothing automatically producing a safety injection signal because that occurred, is there?

MR. REQUA: Are you talking about the second --

MR. EBERSOLE: The stuck open S/G PORV. That doesn't automatically produce a safety injection signal. Only the secondary effects of it, which is shrinkage?

MR. REQUA: Right.

MR. EBERSOLE: Which you say was caused by that. As though it were a directly produced signal by this stuck open PORV. Do you mean it is by the secondary effects of it?

MR. REQUA: Yes. When these are condensed down, then it shrinks.

MR. EBERSOLE: So it is the secondary effects of it. So the first signal was -- I don't know why that was caused. The second issue was legitimate?

MR. REQUA: Right.

MR. KERR: Did you say why that fast transfer

failed to be successful?

MR. REQUA: Yes. Because there was a phase mismatch in that the C phase balance relay opened. It says a phase mismatch. I don't know what percentage it is. If it is a certain percent, then the relay opens up and won't accept the transfer, as I understand it. Is that correct?

MR. SWENSON: That is my understanding.

MR. EBERSOLE: In summary, this is a real, this is a close approach to total blackout. Not like the earlier blackout. This was not invoked by the operators. It happened?

MR. REQUA: That is correct.

MR. EBERSOLE: So it is a legitimate off-site power failure down to a degraded emergency power supply. And it happened -- it was lucky enough it was on the right crisscrossed pattern of buses.

One thing that bugs me, I see about 20 minutes there gone while they figured out whether it was a bus fault or not a bus fault. What were they doing, opening the breakers to clear the bus and verify the bus was clear and then put the loads on one at a time?

MR. REQUA: Yes.

MR. EBERSOLE: Did they ever find where the bus fault was?

MR. EBERSOLE: I take it it was a load fault

that didn't clear?

MR. REQUA: As of this moment, we don't know.

MR. EBERSOLE: We don't know yet. Well, it wasn't enough to clear by independent opening of the circuit breakers? They didn't clear it? You say there was a ground fault. Wait a minute. You say, fault on emergency bus E2.

MR. REQUA: No. A spike -- on E2, yes.

MR. EBERSOLE: How did you determine that? What was the signal that says there is a fault on that bus?

MR. REQUA: As we said before, the bus itself probably was not faulted. It was some equipment on there.

MR. EBERSOLE: What is the signal that told you it was a bus fault? How did you conclude it was a fault or even erroneously?

MR. SWENSON: I don't think we have all that information. It was either a degraded voltage or loss of power to the bus. And that is a question that --

MR. EBERSOLE: You don't know whether it was incoming or outgoing problems?

MR. REQUA: No. We don't know.

MR. EBERSOLE: Okay.

Any questions?

MR. REED: No questions. It doesn't look like this should come before committee?

MR. EBERSOLE: No. This is just a close approach to loss of power.

MR. REED: Did you happen to see that Piper Cub that flew into the high lines last night?

MR. EBERSOLE: Maybe it was one of those. A bus fault.

Any further -- no questions. Thank you.

MR. REQUA: Thank you.

MR. EBERSOLE: Our next one is design deficiency. This is a B&W plant?

MR. VISSING: Right.

MR. EBERSOLE: Feedwater in a B&W plant. That is already interesting, say no more.

MR. ALLISON: Mr. Ebersole, I don't think I wrote down an answer about Palisades, but I take it you don't want Palisades --

MR. EBERSOLE: No, you are correct.

MR. VISSING: I am Guy Vissing with the -- project manager for Arkansas Nuclear One, Unit Number 1. I am with the Office of Nuclear Reactor Regulation.

I am going to discuss with you today a design deficiency in the emergency feedwater system of ANO 1 which was discovered January 14. The significance of this event would be a potential loss of all emergency feedwater and blowdown of both steam generators during a steam line break

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with a single failure of one AC bus.

I would like to go through the narrative here and then I will show you some schematics that more -- in more detail explain the event. This deficiency was discovered by an inspection team from the Office of Inspection and Enforcement in which they postulated a steam line break concurrent with a loss of the red AC power bus. The turbine-driven and motor-driven emergency feedwater pumps would be lost and there would be a possible blowdown of both steam generators.

MR. EBERSOLE: How old is this plant? It is many years old?

MR. VISSING: This plant was licensed in April 1974.

MR. EBERSOLE: So we have got 11 or 12 years?

MR. VISSING: Yes. However, I would like to add that this emergency feedwater system has recently been upgraded to a safety grade system and it was partially a result of that upgrade.

MR. EBERSOLE: The upgrade in fact ran it into this failure mode?

MR. VISSING: Well, yes. I would like to get to that.

The design would not meet the single failure criteria. The plant was at 89 percent power at the time,

and concurrent with this they had a failure of the diesel generator and they were shut down.

The licensee confirmed the existence --

MR. KERR: I am sorry. Concurrent with what did the diesel generator fail?

MR. VISSING: The diesel generator tripped on a high pressure in the crank case and they had a controlled shutdown to repair the diesel generator. It was the diesel, really.

The licensee, to correct this, they did indeed install check valves in the steam lines from both steam generators before startup. Our staff at NRR did review this issue and concurred with the inspection team's findings. I would like to show you the emergency feedwater system of Arkansas. You know that Arkansas Unit 1 is a B&W reactor. The emergency feedwater system is a two-train system. It has the adversity in that one pump is motor-driven and one pump is steam turbine-driven.

I wanted you to note that the motor-driven pump is powered from the red bus, and if we take a take a look at the steam system of the turbine --

MR. REED: Where is this newly installed check valve? Is that the one right next to the steam generator?

MR. VISSING: I haven't shown that yet. Yes. This would be in the steam part of the -- the seem steam

system of the emergency feedwater steam turbine.

MR. HERNAN: It may be helpful, if you want to use both projectors, you can turn the other one on there.

MR. VISSING: Okay. Let's see here. That is the hydraulic part of the system.

MR. EBERSOLE: Before you go further, isn't it true that if you have only a two-pump aux feed system, if you lose steam supply to the turbine-driven pump and then you invoke a theoretical AC power supply to the single remaining pump and it doesn't have a transfer capability, you are dead?

MR. VISSING: That is right.

MR. EBERSOLE: That is just built into the design configuration to begin with?

MR. VISSING: That is right.

MR. EBERSOLE: So was that not accepted as a basis?

MR. ALLISON: The difference here is the steam line break loses the supply to the pump and that is the accident that the system is there to mitigate. It is true you can go down and fail at -- you can fail the steam-driven pump, but that is a failure. However, when you have a steam line break, that is not really a failure.

MR. EBERSOLE: Why not?

MR. ALLISON: You then need to have a redundant

aux feed system to deal with the steam line break.

MR. EBERSOLE: Well, then that is not done.

MR. ALLISON: Yes, they missed it here.

MR. EBERSOLE: Are there not quite a few plants that only have two pumps, aux feed pumps, one of which is turbine-driven?

MR. VISSING: That is right.

MR. EBERSOLE: Doesn't every one of them face the possibility of loss of steam flow to one pump?

MR. ALLISON: If you wait a little, he gets to the check valve question.

MR. EBERSOLE: Maybe it is something --

MR. VISSING: I would like to show you the scenario that would happen with this configuration if they had a steam line break in the 36-inch line here, and you will note that this valve is a normally opened valve and it is on the red bus. And the motor over here is on the red bus. Now, having the break here, you would then have the valve here normally opened and the valve here opened and you would have a blowdown of the steam generator and no steam to the auxiliary -- the turbine-driven pump.

MR. EBERSOLE: Right.

MR. VISSING: Therefore -- and also you wouldn't have power to the red -- to the motor-driven pump. So, therefore, you would lose, first you would lose your main

feedwater because it would trip off because of the steam line break.

MR. EBERSOLE: Sure.

MR. VISSING: And then you would lose your emergency feedwater pumps.

Now, what is normally done here is what they did do to correct this situation: they, Arkansas, recognized what they did and they installed check valves here and here (indicating) and they switched this, these two pumps, these two valves around so that this would be a closed valve, locked, closed valve and this would be the operating valve that would operate off the red bus.

Now, therefore, if you have a postulated break in the 36 inch line, you would then lose the main feedwater pumps, and you would lose -- if you had a single failure in the red bus, then you would lose this valve here, but it would have a check valve and you wouldn't have the blowdown of the steam, of this part over here and you would still have steam to your turbine of the turbine-driven motor.

What happened here, in their upgrade of their system, they did initially have in their design such check valves. However, subsequently, because of problems that they had experienced on Unit 2, with check valves chattering, they thought that they were going to have a much more reliable system if they didn't have those check

valves, and not recognizing that they didn't adequately look at the single failure criteria.

And so they did indeed install check valves, the swing-type check valves. And they are going to upgrade those check valves with -- when they can -- with the designs that are similar to what they used in Unit 2 to correct the problem.

MR. REED: Is there any traceability on -- apparently a design decision was made way back when to remove some valves in a very critical system.

MR. VISSING: Yes.

MR. REED: Is there any traceability as to who was responsible, who verified, approved such a design change?

MR. VISSING: Well, that is an issue that isn't altogether clear to us. They explained their -- the scenario during the inspection process and it wasn't altogether clear as to the thoroughness of their review for that change.

MR. REED: Do you think it was a designer failure or a utility/owner failure? Who was -- was it the utility that caused the deletion of the valves or the designer?

MR. VISSING: The design was originally there. The valves were originally in the design. Subsequently,

they were indeed removed and there was a conscious decision to remove them from the design.

Now, who indeed were the contributing people that made that decision, I don't know.

MR. EBERSOLE: In a turbine-driven aux pump, isn't it just like breathing? When I look at the steam line, I will always see two checks?

MR. VISSING: Yes.

MR. EBERSOLE: But it wasn't done?

MR. VISSING: That is right.

MR. EBERSOLE: So they weren't breathing very well. That is astonishing, isn't it?

MR. VISSING: Yes.

MR. EBERSOLE: How did they escape all this ferocious review in their engineering section? I want to know how it got through?

MR. VISSING: I can't answer that. I don't know.

MR. EBERSOLE: How did it escape NRC?

MR. VISSING: It did not escape NRC. Their original design was submitted to NRC --

MR. EBERSOLE: I mean the modification?

MR. VISSING: The modification, they did it under a 5059.

MR. EBERSOLE: What happens to that?

MR. VISSING: We did not review that.

MR. REED: I don't see how this could go into 5059. It certainly degrades, changes the safety analysis by a lot.

MR. MICHELSON: They just didn't recognize it.

MR. EBERSOLE: But it is up to them to recognize that. So here is a basic flaw in the whole rationale. Unless they can see it, you will never know. And this would apply to anything.

MR. MICHELSON: You see it at the end of the year when you get the yearly report from the utility.

MR. REED: This was done some time ago, apparently. This is an old plant.

MR. VISSING: The design was originally submitted to us in the latter part of 1981.

MR. REED: This modification to take out the checks?

MR. VISSING: No, the initial design of the upgrade of the system was submitted to the NRC in December of 1981.

MR. EBERSOLE: And it escaped that --

MR. VISSING: It had the check valves in that design.

MR. EBERSOLE: It had the check valves in 1981?

MR. VISSING: Yes.

MR. EBERSOLE: And then they took them out.

MR. VISSING: The upgrade was installed in the last refueling outage which was in December of -- it was December of 1984.

MR. REED: I think I am becoming confused. I think you are talking about the checks being put back in and I am focusing on when, how, and what time were they deleted from the original design. That must have been back in 1974 or -5?

MR. VISSING: No. They had submitted an upgrade of the emergency feedwater system in December of 1981. At that time that design had had check valves in the system. The system was not installed until December of 1984.

MR. EBERSOLE: You mean it took three years --

MR. VISSING: Yes.

MR. EBERSOLE: -- for that to materialize?

MR. VISSING: That is right. And between the time that the design was submitted to the NRC and the time it was installed, there was a change made in the design.

MR. EBERSOLE: How did that get supervised?

MR. VISSING: We did not see that change.

MR. EBERSOLE: That doesn't come to you?

MR. VISSING: No.

MR. REED: I guess what you are telling me is that this 12-year old plant had a Three Mile Island backfit modification going and that didn't hit the fan until 1981?

Is that right?

MR. VISSING: Right.

MR. REED: I wonder what they had prior to Three Mile Island.

MR. VISSING: I tried to find that out. I don't know.

MR. REED: They probably had the right system. That's just a joke.

On this one, I see this only as a statement to point out that auxiliary boiler feed systems are very vulnerable to all kinds of problems and design aspects. I don't know that I support auxiliary feed system as the only way to take care of decay heat removal on PWRs.

MR. EBERSOLE: I was wondering if we should bring this to the full committee of how important things can slip through a supposedly rigorous review system and and then be picked up as late as this.

MR. REED: In the gut issue, the root issue is decay heat removal.

MR. EBERSOLE: But here is an explicit --

MR. KERR: I don't think we have to convince anybody that decay heat removal is important. What is it that we are trying to persuade the full committee to do?

MR. REED: To adopt primary blowdown for certain PWRs?

MR. EBERSOLE: But is there an administrative control or move here that is indicated to close these holes in the review process so that we don't let these things lay around like this? What is the NRC action on this in the generic context.

MR. KERR: My impression was that this was not done deliberately, it was just that somebody overlooked something. Are we suggesting this they go back and review every plant?

MR. EBERSOLE: Aren't there some general rules like breathing, that you always look for two check valves?

MR. KERR: Well, there are some general rules like one never makes a mistake.

MR. ALLISON: This type of inspection is a good thing to do where you go and look at the design of the system in some detail and find cases like this and take enforcement action and publish information notices. That is really our -- what we are doing.

MR. EBERSOLE: What is contemplated here is an enforcement action other than --

MR. ALLISON: I don't know specifically. I guess we can find that out.

MR. EBERSOLE: What about the quality of the utility review?

MR. ALLISON: I think that will get a closer

look by the Region now.

MR. EBERSOLE: Is that enough?

This has been sitting there for how long? Three or four years?

MR. ALLISON: Right. About that long.

MR. REED: Jesse, I would like to repeat my position, it seems to me that it is worthwhile for the full committee to know that here is another example of the vulnerability of aux feed as the only path of decay heat removal for some reactors.

MR. EBERSOLE: Let's put down about five minutes for this one, just to mention it to the full committee.

MR. KERR: I will agree to that will if you will assure the full committee that this is being put to them because of Glen Reed.

MR. EBERSOLE: Okay. I will mention this in a very quick way.

MR. HERNAN: Do you want the presentation or is this just something --

MR. EBERSOLE: No. We will make a statement to this effect. You can make it or I will make it. Whatever. Let's go to the next one.

MR. WEISS: The next item is the River Bend startup review. It is not an event per se, but it is an action that was taken because we saw significant number of

5072 reports for a new plant that was just starting up and we decided to go out and have a look and see if we could divine exactly what was going on.

MR. KERR: Will you forgive me if I ask you what a 5072 report is?

MR. WEISS: A 5072 report is a report to the operations center made over the emergency notification system. A red phone call. If you go in the control room of every nuclear power plant, there is a red phone. You pick it up and call within either one or four hours of an event occurring. The criteria for 5072 are very similar to those required for LERs. So generally they are the same thing.

MR. KERR: Thank you.

MR. WEISS: To get to the bottom line first, we, after our visit we came to the conclusion that we will expect the number of events that this plant is producing to decrease; generally we thought their management appeared sound. However, as a check, we asked the licensee to come to the Region 4 for a progress review, and I believe the date is March 4. There were a number of us that went. I was the only representative from IE. Steven Stern, the project manager who is here, and three other individuals from NRR went. The Region 4 section chief, John Jordan and his project manager showed up. John is here.

When we visited the site on January 28 through 30, we went over with the staff their recent events. Recent scrams and other significant events, examined root causes. We witnessed a shift turnover in the control room. We had a tour of the plant, including the diesel generators and the feedwater system. The feedwater system was particularly important since this plant is having a significant number of problems with their feedwater system.

We went to the Fancy Point substation, which was significant because they had a loss of off-site power event early in January that was precipitated at this substation. We witnessed a surveillance procedure on the reactor water coolant system. The largest single contributor was inadvertent isolation of reactor water cleanup. It was also significant from the point of view of the demonstrating their ability to deal with jumper control problems.

We also got a walkdown of the control room panels, and that was revealing in a number of ways, not only to familiarize ourselves with the plant, but to also with some of the problems they had. I noticed that they had a 3 by 5 card pasted on the panel with instructions warning the operators to be careful not to have the suction to shut down cooling and LPSI open at the same time. They, like most plants, have had one of those draindown events.

In the exit interview, the project manager, Steve Stern, urged the licensee to continue open communication with the NRC. We thought this was important because the licensee was indeed making a significant number of reports that were voluntary in nature and weren't really required by the regulations and we wanted that to continue.

There were such things as half scrams and half isolations that told us something about the plant or generic component problems and so forth.

We urged the licensee to resolve the outstanding equipment and human factors problems. He is undergoing a significant number of problems, as would any plant during startup. And we did criticize them and urge them to upgrade their communication with other plants, because we thought there was a significant number of problems that if they were not preventable, at least they could have found out earlier by having more open, more frank discussions with their counterparts at other plants, other similar boilers with a year or two more experience. We urged them to conduct these particularly at the lower levels rather than just at the highest levels of plant management.

I could discuss a large number of plant-specific problems, but I just want to mention a few to give you a flavor for some of the things that we found. They had a temporary alterations program that I guess was best

characterized like a quick 5059 that was maybe appropriate during the latter stages of construction but we thought should come to a halt soon. It was appropriate for an operating plant.

They had, like most new boilers, they have more instrumentation than the old boilers and a small number of annunciator windows. One of those small windows had something like a diesel generator problem. There were, I think, over 15 different conditions that could have caused that window to flash, some of which were more in the nature of nuisance alarms and others were more serious. However, they wouldn't get a reflash when a more serious problem came in, and the operators that had asked that this be fixed and it is being fixed.

The feedwater problems we thought were the most serious of those that we had seen out there. They had vibration, for example, in the short-cycle loop that this had actually torn an anchor out of the wall. They had a packing leak in that room when we visited it.

MR. KERR: What is a short cycle?

MR. WEISS: That is a loop that takes you back to the condenser for the purpose of cleaning up the feedwater system. There is both a short cycle and a long cycle, depending on how much of the feedwater system you want to clean up. They had erosion in the long-cycle loop

and were installing heavier schedule piping, more erosion-resistant piping. They had sticking of the feedwater regular valves, actual mechanical binding. The operators noticed that they did not actually have real valve position indication in the control room. They had a demand signal. That was being changed.

They had steam tunnel ventilation problems. And I understand that they are being rectified with additional coolers. But one of the most interesting aspects of the visit was this was a particularly productive visit in terms of identifying generic problems for us. We are lucky if we get one good problem out of a visit that we can tell other plants about. Here we came up with three that seemed to be interesting.

You may recall from reading regional daily reports earlier this year they had a feedwater size isolation valve with its operator fallen off on the floor. There were about three contributing factors to that event that ultimately I think will result in an information notice. There was inadequate thread engagement. The bolts were too short. There is a difference of opinion as to what was the proper torque value between the various vendors involved. And there was also a question of whether the valves should operate or should have been torqued with a valve on its, seat because that appeared to have

transferred some torque into the valve stem. Then when the valve was cycled, the torque was relaxed on those bolts.

Another generic problem occurred with a particular variety of temperature switch they were using in their reactor water cleanup isolation. The licensee informed us of the need to retrofit in a significant number of capacitors and to add a resister for the purpose of noise suppression. And it is our understanding at this point that General Electric will be issuing a service information letter on that subject.

Now, the fiber --

MR. KERR: Was that a new design?

MR. WEISS: No. That was one of the things that we thought that they could have learned about if they had been speaking to other late model boilers, would have had similar problems.

MR. KERR: GE didn't know about it because GE is just now issuing something from what you said. Or did I misunderstand?

MR. WEISS: That is correct. I understand that GE will be issuing something on the subject. Exactly how much GE knew and when they knew it, I am not in a position to say. But we have seen --

MR. KERR: Had others previously had the problem and done repairs?

MR. WEISS: Yes.

MR. KERR: But they just haven't told anybody about it.

MR. WEISS: Sometimes when you replace a component, you think, well, maybe it is a random component failure. But after one sees a pattern at several plants, one becomes disillusioned with that point of view and begins to think that there is a generic problem.

MR. KERR: I thought it was a matter of redesigning the circuit.

MR. WEISS: There was a noise suppression resistor that River Bend thinks is advisable so that when one moves a switch to read to function on the -- this particular temperature switch, one doesn't get a noise spike that causes an inadvertent isolation.

I think, I don't know whether other plants have adopted that change. But it is certainly worth telling other people about.

There is an interesting problem in the switchyard. The plant added a diverse method of controlling breakers in the switchyard which consisted of a fiber optics multiplex system. They suffered a loss of off-site power event around January 1 of this year. There were no valid targets set that -- in other words, there were no real faults identified, electrical faults

identified to cause this event. Testing subsequent to the event has shown that any one of several different kinds of hand-held walkie-talkies will cause tripping of the tone-relaying equipment that is associated with the fiber optics system.

The utility has adopted shielding; they installed some plywood panels and put copper shielding around the room that contains the equipment. They have installed an events recorder so that, should it happen again, they will have much more detailed information about it. They have put up signs in the building and on the fence around the substation and installed nonduplicating locks at the entrance. They have trained their people, cautioned their people and supplied additional procedures in the substation for resetting breakers, should the need arise.

They are also doing some filtering and separation of power supplies.

In general, --

MR. KERR: I am a little puzzled. Fiber optics, I thought, transmitted rather high frequency electromagnetic radiation.

MR. WEISS: That was the impression I was under. And every time I bring the subject up with an electrical engineer, he says it can't happen. That is why we will

probably write an information notice on the subject. What people forget is that at the end of the fiber optics line, at the end of this glass or fiber tube, you will have a demodulator receiver of some kind which will operate in the radio frequency regime.

MR. KERR: My point was, it seems to me the difficulty is not with the fiber optics but with the termination?

MR. WEISS: With the end, which is exactly right. You may recall that there was two loss-of-off-site-power events, I believe, at Palo Verde that were caused by fiber optics systems.

I am reasonably sure that the licensee there did not evaluate whether a walkie-talkie could have precipitated that problem. But that licensee decided to remove breaker control by fiber optics. Now their breaker control is strictly hard wire.

In any case, there were several things that pleased us when we went out there. One I wanted to mention is their condition report system. Apparently anyone in plant can write a condition report on any condition and this goes to, directly to the upper level management, which then makes a decision on its ultimate disposition. One of the strengths of their system is their ability to focus resources quickly.

One of the things that I was most concerned about when I arrived was their jumper control. They have instated a strong program there of locking cabinets and then when one wants to perform a maintenance procedure or surveillance of some kind, then you go to the jumper control officer who then gives you a serialized tag for each jumper that is installed and there is an open log entry for each jumper which then must be cleared. That is in addition to the double verification that would be performed on order STPs.

Another interesting aspect of their jumper control problem was they had a significant number of events that were caused by alligator clips falling off the terminal strips. They tried several different things. They settled on a solution that appears to be quite desirable. They used banana plug, banana jack-type assemblies. They install a banana jack beneath the terminal lugs that are to be used in a surveillance procedure. Then when the banana plug is used for a test instrument, it will not fall out.

It also has the added feature, I noticed, of when you open the cabinet, you may see several hundred possible connections, but you see only four brightly colored banana jacks, which greatly cuts down on the possibility for putting it in the wrong place.

I have already mentioned the work they did with fiber optics. Another thing I notice that they did in the control room was a large number of their meters were colored in such a way that the normal operating range was something like green and abnormal range was in some other color. And I asked the assistant operations supervisor whose idea that was. He said it was mine. So I think that indicates that the operators are having a substantial feedback into their plant design and management is listening to them.

MR. EBERSOLE: That was the plant that provided the number 3 diesel cooling water from the number 2 diesel supplies? Did they ever fix that?

MR. WEISS: Yes.

MR. EBERSOLE: Okay.

MR. WEISS: I noticed a number of interesting design features out there. I haven't gone to that many plants, but it was interesting.

MR. KERR: That location is fairly remote. You may be getting all those telephone calls just because they get lonesome and want to call somebody up.

MR. REED: Let me try another theory. It seems to me that what I am hearing, because they have lots of events, reportable events, is that there is an aggressive utility, informed, perhaps capable organization that see

weaknesses and vulnerabilities in design and they are not just accepting that and laying down with it. Is that what I am hearing?

MR. WEISS: That is an entirely possible thing. We have the project manager here who has a series of charts that show that on a month-to-month basis, they have a fair number of reports. But if you look at it from a milestone-to-milestone basis, they are doing pretty well.

MR. REED: They have a convergent seam.

MR. WEISS: One of the benefits of going out there is that you get some individuals willing to speak very frankly. I had my eyes opened up to some of the techniques that plants use to eliminate reportable events. For example, if you have got a system that you know is cantankerous and is likely to cause a red phone call, one thing you can do is take it out of service, declare it inoperable and perform your surveillances on it while it is inoperable. Then it doesn't become reportable. I hadn't heard of that twist before but now I am a little bit wiser. The plant, by the way, has not been doing that.

MR. KERR: I bet Mr. Reed has heard of it.

(Laughter).

MR. REED: I hadn't really.

MR. KERR: I didn't mean that you had done it. I just meant, I bet you had heard of it.

MR. REED: I am getting the feeling that you have a couple people probably selected by aptitude testing, well trained, who are causing quite a stir with this new plant and finding a lot of problems.

MR. EBERSOLE: You are an optimist.

MR. KERR: We have got to have another report on aptitude tests.

MR. EBERSOLE: Is that it?

MR. WEISS: That's it. Thank you very much.

MR. EBERSOLE: I believe we have had a substantial increase in interest. We would like to introduce Bob Benaro, director of the PWR licensing division.

MR. BENARO: I would like to get into the Perry earthquake event. This event occurred on January 31st and simultaneously I approached people on the committee to get on the agenda for the committee's reaction or review of what we were doing in safety evaluation of the event. And simultaneously two members of the Congress wrote to you and I think you must have seen those letters by now. I will be talking to Mr. Fraley and others in order to do what I can to expedite the ACRS attention to this matter.

The first formal report of the event has just become available today. We had an all-day meeting at the site yesterday on the event. And the owner of the plant is

here with his consultants and I have a good number of staff members here and I would like to go right in by turning it over to the owner who will give you a substantive briefing on what happened and what their evaluation of the event shows.

So Mr. Murray Edelman, who is the senior vice president, if I recall his title correctly, of Cleveland Electric Illuminating Company, I will turn the floor over to him.

MR. EDELMAN: Good afternoon. I am Murray Edelman, vice president of CEI in charge of the Perry project in our nuclear program. I am glad to be here this afternoon on what actions we have taken with respect to the earthquake that occurred in the vicinity of our plant on January 31.

Yesterday afternoon, we had the opportunity to brief the NRC staff and their consultants at the site as to what actions we have taken since the earthquake occurred on January 31st. This is the overhead that we used yesterday. What I plan to do is to cover the introduction and the overview of what actions we have taken the plant status and our response that was given by our operations general supervisor yesterday. And a little bit about the earthquake analysis and seismicity that was performed for us by Weston Geophysical. Mr. Holt is here. I am not a

geologist but I will try and cover that portion of the program. Then we would like to have Dr. Chen, who was responsible for the design and seismic analysis of plant for Gilbert Associates, our architect engineer, review our design criteria and actual experience that we had during the earthquake on January 31st.

As an overview, the event occurred on the 31st, about 11:45 in the morning. Our plant people, even though we do not have an operating license and are about 10 case days away from, in our opinion from receiving a license, even though we're not in an operating mode, chose to go into our emergency procedures for several reasons. One, it is the best way we have learned, through training over the last several years, of marshaling all the resources of CEI, the county, the NRC and the state of Ohio. It gives us the opportunity to ensure accountability of anybody who was in the plant at the time. Since our security system is in full effect, we used that to get a site accountability of everybody on site and we went into our emergency plan and activated our facilities as well as the facilities in Chicago and Washington.

We then also sent our people out in the plant after that to assess the status of the plant. We ran numerous plant inspections with our operating people to insure that the plant was in a safe condition. We then

downgraded our emergency event from a site emergency down to an alert and after conferences with the state, the NRC and ourselves, we terminated the emergency about 2:30 in the afternoon. At which point in time we instituted our recovery organization, the same as we would have in a normal emergency if one had existed. We are still in that mode where I am the recovery manager to bring the site and response to the technical issues raised from the earthquake.

We did conduct numerous walkdowns by our people. About 65 of our people spent the entire evening and next day while the NRC sent people in from both Chicago and Washington in response to a mutually agreed upon confirmatory action letter from Region 3. We froze all equipment and inspected everything and Washington came in to inspect the equipment. After inspections by our people and the NRC, the confirmatory action letter was modified which allowed us to go back to finish our testing and operation of the plant.

In addition, immediately after the event occurred, we got a hold of all our special consultants who have worked on the Perry project. This included Gilbert Associates, Weston Geophysical, Mr. Engdahl who manufactures the seismic information; the people from Kinemetrics; Dr. Hall, Mr. Stevenson and a number of other outside consultants to review the plant conditions.

Close examinations were also prepared in nearby areas to record any aftershocks that may have occurred after the event took place.

We also started to compare the plant data that we had recorded during the earthquake to see if it had any engineering significance in our overall design. We did record some high frequency accelerations which we didn't consider to be a problem, which is a conclusion of our report which you have in front of you, because they were low energy, very short duration and low velocity. The subject of the high frequency accelerations is a generic subject which we know the staff is looking at at a number of sites around this country. We have reviewed and are still reviewing the plant structures and equipment to make sure that there is enough conservatism in the design of the plant and considering our loading combinations and allowable stresses.

I would like to go through what our operations people looked at in terms of what actions we have taken.

As the event started, we had ongoing testing and calibration activity going on. We were preparing to run our division 2 diesel generator testing. We were ready to move the startup sources from the upper pool toward the reactor and we had a number of systems in standby and operating mode throughout the plant. What occurred during,

after and while the operation of the event took place is all systems remained operable. All safety systems that were designed to be operating through an earthquake operated through the earthquake -- all relays, all the safety systems we had in the plant.

Immediately following that --

MR. MAOL: What was the status of the plant at the time the earthquake occurred?

MR. EDELMAN: We were in our final surveillance testing. We had moved fuel from the lower pool to the upper pool. We were calibrating our source range monitors in the upper pool and running our final surveillance test on some of our major systems such as the final test that you have to run on diesel generators.

We sent our operators out after the event took place and examined the plant to see if there was any structural damage. We did walkdown by maintenance people. A systematic identification of everything in the plant. We asked our people to identify any abnormalities they might see, no matter what they were, whether it was a burned-out lightbulb or a cracked or peeled paint. Each one was evaluated to see if it had any impact on the earthquake itself.

We ran site surveys and settlement surveys which we normally run on a monthly basis. Part of our recovery

plan. We asked those surveys to be taken the next day to see if we had seen any settlement throughout the plant. We looked at our cooling tower and walked down the structures. We have certain areas in the plant where we have identified what we call seismic violations. One pipe may be too close to another. There were 26 of those that we had not yet fixed so we went and looked at those specifically to see if anything had interacted. Nothing was identified.

We studied all the relays that were energized to make sure that they stayed energized through the event, and they were.

And we in addition, and consistent with our confirmatory action letter, we put a new procedure in as we were running our surveillances. If any surveillance test didn't pass, we would go back and analyze that to make sure that was not a cause of the earthquake.

Since that time, we are running these surveillances, we have not identified any that have worked without our management review or NRC review because of the earthquake itself.

Today we are back finishing our surveillances and our preopen test right now. We met with the staff and reviewed our operational readiness. We anticipate to be complete, ready for fuel loading in about 10 days.

Now I would like to cover some general USGS data

on the status of the earthquake itself.

MR. KERR: Excuse me. One of the things that we have encountered in discussion of seismic event and fix, at least those that might be hypothesized, is the possibility that relay chatter will occur and relays will put things in modes of operation that are unexpected or unusual.

Did your investigation include any effort to see whether that sort of thing had occurred?

MR. EDELMAN: To the best of my knowledge, yes, and to the best of my knowledge we found nothing where that occurred. The relay that did trip in the field was one that was on our generator breaker which was supposed to trip and the volt damage to it was off at the time and would not go back. But of all o our safety systems, we had no relay chatter or anythin go into a different mode because of the earthquake.

MR. KERR: Thank you.

MR. MICHELSON: Were all of the safety systems fully energized and armed at the time of the earthquake so that you could see if relay chatter was causing anything unusual to happen?

MR. EDELMAN: I don't think all of our systems were energized but I can give you a list of about 40 that were. Our control rod drives --

MR. BENARO: For convenience, it is on page 3.6

of that big report. Section 3, page 3.6.

MR. EDELMAN: I think we listed in there all the safety systems that were energized.

MR. MICHELSON: Of those that were energized, was there any potential for relay chatter to even occur?

MR. BENARO: Ask him, not me.

MR. MICHELSON: Have you actually analyzed your system to see if there was a potential to observe the effects of relay chatter to know if there was even a question about it?

MR. EDELMAN: Our manager of engineering is here.

MR. STEAD: We have looked at that. In fact, at the time the event occurred, we were in preparation for a response time testing and it was fully lined up and in a standby mode ready to do the test. This is where you actually initiate a signal to the bus and put all the loads 't after the diesel starts. All the systems were lined up, ready to go into operation, waiting for a LOCA signal to initiate. So all those relays were in position whereby any movement to have started a LPSI pump or a high pressure injection pump or caused a valve to open or close. So those things are all in that position. So any of those systems that are related to emergency core cooling, which I think is probably the best test, in response to your question, were in a mode that that could have happened and

wouldn't be immediately recognized by the control room operators by indication in the control room. We saw no indication of that during the event or after the event.

MR. MICHELSON: Not knowing the details of your particular test, it is a little hard to see how the systems were fully armed, though, because in order to be so, there must be certain temperature pressure permissives, et cetera, in the system. You went in and jumpered out a great deal of ECCS initiating pressure or temperature.

MR. EDELMAN: If a plant is not running, you can't have some of those.

MR. MICHELSON: But within the spectrum of what you had, you did not see any relay chatter effects and you were already armed and ready to go on that?

MR. STEAD: That is correct.

MR. MICHELSON: Thank you.

MR. EDELMAN: The data I am showing up here, all this information and slides that we have are all in the report that we have. What we have handed out to the staff and yourself today is the USGS data which locates the type and latitude and longitude of the earthquakes, the depth of the earthquake and based on 64 stations worldwide, the magnitude on the Richter scale was about 4.96. It was located in Geauga County about 11 miles from the plant site.

MR. MICHELSON: I am assuming that you observed

no nonsafety systems' spurious operation?

MR. STEAD: No, we did not. We had a strip of our instrument air compressor on high vibration.

MR. MICHELSON: I am not sure what you mean. Was it induced by the earthquake?

MR. STEAD: Yes. It was a shaft vibration line.

MR. MICHELSON: That is the sort of thing you are looking for, just like relay chatter is induced by the earthquake.

MR. STEAD: This was in the normal mode of its operation. It is a centrifugal compressor.

MR. MICHELSON: What I am asking you, was there anything unusual that happened in the plant that might have been coming from the effects of the earthquake?

MR. EDELMAN: The only two things that tripped that we describe in our report are the instrument air compressor and a building heating boiler which tripped. All other equipment remained in operation.

MR. STEAD: I think on the next page is a list of all the nonsafety systems that were in operation at the time and gave no indication of any spurious operations.

MR. MICHELSON: Your air system was nonessential?

MR. STEAD: That is right.

MR. KERR: I know you don't like this, but it seems to me the data that you are collecting is rather

important in terms of sort of a general indication of what happened.

MR. MICHELSON: What you have to compare it with is what is the design basis seismic situation, which I assume you are going to tell us about in a minute.

MR. SIESS: Have you got any idea of what the range is all about?

MR. EDELMAN: I didn't hear the question.

MR. SIESS: Do you have any idea of what the range of natural frequencies are for relays?

MR. EDELMAN: For relays? I do not know.

MR. KERR: That is an easy question to answer. The answer is no.

MR. EDELMAN: The map of the site area which is in your book shows the plant site on the top, a 5- and a 10-mile radius ring. The epicenter was here on January 31st. The recorded aftershocks were about three miles west of the epicenter. The highest one I believe is about 2.4. We had no additional readings from these aftershocks on any of our plant instrumentation.

This slide, the site at Perry is located -- and this is kind of a skewed picture -- we are showing the focus of the epicenter about 11 miles offset from the site and about 5- to 6000 feet below or 30,000 feet below the surface of the earth in the Precambrian rock. The site is

actually located on top of undifferentiated Paleozoic sedimentary rocks on shale, as I call it.

During the -- we did extensive review of the seismology during the PSAR and FSAR stage of the plant. We did find geological anomalies which were glacially formed folds in the sedimentary rock well above the Precambrian rocks and the earthquake occurred down here. Our people have looked at it and the conclusion that our geologists and seismologists have reached from their analysis is that the tectonic approach that we took in our PSAR and FSAR stage is still valid. That there is no capable fault, no tectonic structure. That our safe shutdown earthquake intensive scale is 7. That our site-specific spectrum was 5.3.

We did see, based on extensive instrumentation that we had in place and operating at the time, an exceedance and short duration of a high frequency. That is about 20 Hertz of a spike on the upper levels of the containment building. Dr. Chen will go through that in detail. But that was, again, high frequency and was above, exceeded the 84 percentile spectrum that was put in the FSAR.

Dr. Chen now, I have gone through this pretty quickly to allow Dr. Chen enough time to review the details of the specifics of the site and the design of the plant.

I would like to turn it over now to Dr. Chen, who is our structure recall designer for our plant.

MR. CHEN: I am the manager of the civil structural department of Gilbert. Before we go into a detailed discussion of the comparison of this 1986 Ohio earthquake event versus our design, I would like to summarize the nature of the earthquake which will give all of us a proper perspective of the size of the earthquake.

The recorded 1986 Ohio earthquake is of the nature of high frequencies, low velocity, low displacement and low energy, short duration earthquake. In comparison with the kind of design earthquake we used for Perry, it is of the broad frequency band: long duration, high velocity, high displacement and high energy earthquake.

To show you how we reached this conclusion, we will show you the comparison of the recorded time history versus the time history we used in design. This time history at the top is our design time history in the north/south direction on the top of the foundation mat.

The one at the bottom is the recorded time history in the same direction, at the same location. As you can see, at the top the design time history has a duration of 22 seconds. The recorded time history with the str ng motion, part of the record is less than one second. And the one at the bottom has much higher frequency content.

The high frequency content which translates into low velocity and low displacement.

This is the north/south component. We will show you the rest of the components.

This one is east/west. As you can see, the expiration value for east/west is also lower than the design time history. Not only the duration is shorter. The next one would be the vertical component. Also of similar nature. Also the scores were low.

MR. REED: In the lower graph, does it actually stop and then is it a straight line on the right or?

MR. CHEN: Totally stopped. Whatever is left is the background noise.

MR. REED: Okay.

MR. CHEN: Next we show you the comparison of the upper elevation which is at the steel containment. This one is also north/south component.

Similar conclusion, that the duration of the recorder is much shorter and that frequency is much higher. There is some exceeders in the expiration which is a 20 Hertz region. We will come to discuss that comparison later.

The next one is in east/west direction. As you can see, the acceleration is also less than what we used in design.

This one is in the vertical direction, a similar conclusion.

Then in comparison, we can see our seismic design basis is of the nature of broad band frequency design with smoothed 84.1 percentile spectrum. And it has much higher energy and long duration.

The spectrum we used I think everybody is familiar with.

Next we will show you the location of those instruments. And the type of the instrument. At the top of the Kinemetrics SMA-3, time history recorder. They are located at the top of the reactor building map and also at the elevation 686 of containment. The three at the bottom are the Engdahl recorders. One out of the three during the earthquake was in maintenance. That is why there was no record of the other one.

There are four more instruments on this table. These four instruments are Engdahl response recorders. They are located at reactor building foundation mat, reactor building 630 platform and the other two are at the basement of aux building.

This one shows you the location in the plant view of those recorders. 1 and 2, for example here, are the triaxial time history recorder at the top of the foundation mat and at the steel containment and the rest

are Engdahl instruments.

This one shows you the elevation view of the location of the instruments. 1 and 2 are Kinemetrics triaxial time history recorders. And the rest are Engdahl instruments.

MR. MOELLER: The 4, 7 and 8 are different, are they?

MR. CHEN: 4, 7, and 8. 3 and 4 -- let me see. 3 and 4 are the peak spectrum recorders. 5, 6, 7, 8 are the triaxial response spectrum recorders. One records the peak. One records the response at 12 discrete frequencies.

All those instruments with regard were used in the comparison with our design data. The first design barrier we compare with is the so-called ZPA variable which means zero period acceleration.

The conclusion of this comparison is that the recorded ZPA value varied from way below OBE value to 74 percent of the SSE values.

Except at containment vessel elevation 686, as we showed earlier in the time history comparison. But at this point, the relative displacement, in other words, the stresses is very low.

The reason for that is because the nature of the earthquake is of high frequency and low displacement earthquake.

Now we will show you the table, how we reached this conclusion.

There are the five columns and the five different locations of the recorders with the ZPA values. ıne first column is at the foundation mat of the aux building. The second column is at the toundation level of the reactor building. The third column is at the top of the recirculation pump. The fourth column is at the elevation 630 of the reactor building. And the fifth column is elevation 686 of the steel containment. And the first row here is in the north/south direction. And the second row is in the east/west direction. And the third row is in the vertical direction. Since during the seismic design we considered three components simultaneously, the proper comparison is the square root of sum of squares which is at the fourth row here.

If we look at the SRSS comparison here, at a foundation mat of aux building, the recorded value is comparable to OBE value. At the foundation mat of the reactor building, the recorded value is between OBE and SSE. At the top of recirculation pump, the recorded value is way below OBE value. And at elevation 630 of reactor building, the record in the vertical direction cannot be determined. So the only thing we can compare is in the north/south and east/west direction. In these two directions, the recorded

values are below OBE value.

The only thing of some exceedance is in the last column in the reactor building elevation 686. The SRSS value of the recorded acceleration is comparable to SSE with slight, less than 4 percent exceedance.

Now we were looking to dislocation about its relative displacement. At elevation 686, with the recorded value exceeding the design value by less than 4 percent, we are looking at a displacement. The first column is the displacement at the foundation mat. The second column is the displacement at the elevation 686. And the third column is the difference which is the so-called relative displacement. Relative displacement is a measure of stresses in the containment.

If we look at a comparison in the last row, which is the SRSS value, the relative displacement is about one-half of OBE value. So that is why we said that the nature of this earthquake is of very small displacement, even though the acceleration is high, the induced stress is low.

In addition to the comparison of ZPA values, we also compared response spectra. This is the conclusion of our response spectra comparison.

First, Perry design spectra are far above the recorded spectra in the frequency range below 14 Hertz.

Second, certain recorded response spectra exceeded design spectra around the frequency region of 20 Hertz. But, this exceedance at 20 Hertz corresponding to very small displacement, for example, at the foundation mat, which is less than 3/100ths of an inch, because of this small displacement at 20 Hertz, it doesn't really have any engineering significance. Also, the recorded velocity spectra, which is a measure of energy input, is also much lower than the design velocity spectra. The north/south components of the foundation mat of the reactor mat -- by the way, this is a paper used in 160. I presume everybody is familiar with this type of paper.

At the frequency region below 14 Hertz, this is the design spectra. This is the recorded spectra. As we can see here, the design spectra is far above the recorded spectra, except with this spike here, which corresponds to 20 Hertz. But if we look at the displacement value at this spike, which is somewhere here, it is 2 times 10 to the minus 2 inches, that is .02 inch, 2/100th of an inch spectra displacement. It doesn't really have any engineering significance because of the low displacement.

This north/south component -- we will look at the east/west components in the same location. The exceedance is even less in the high frequency region here. Same thing in the low frequency region. Design is well

exceeded by the design spectra. Again, the peak displacement here is about 1.5 times 10 to the minus 2. That means .015 inch, which is very small.

Next we will look at vertical component at the top of the foundation mat. Same thing. Below 14 Hertz, the recorded spectra is way under the design spectra. Again, with a 20 Hertz spike here which corresponded to about 1.5 times 10 to the minus 2. In other words, .015 inch. Which is very small.

I will now show the rest of the comparisons which are all in the book.

After this comparison, we want to show you why this earthquake is insignificant. First of all, all the energized equipment went through the earthquake as designed. Second, as to the other industrial design criteria. During the blasting or pile-driving operation, the criterion used for nondamage, nondamage threshold of non-engineered building is 1 inch per second. The maximum recorded velocity at the site on the foundation mat is only .87 inch per second. That means that the maximum recorded velocity on site is less than what people would experience during a pile-driving operation.

Another thing is, in comparison with IEE 344 which is the seismic qualification of class 1E components in a nuclear power plant, section 77.5 of this IEE 344

prohibits the use of shock tests to qualify the equipment for the reason of high frequency and short duration input.

MR. MICHELSON: I guess one can say from that that I guess we can't derive too much comfort from the fact that there were no electrical disturbances in terms of relay chatter or whatever. So I guess it still leaves us without any good data on responses of electronic equipment of that type.

MR. EBERSOLE: My experience in listening to this is that you thought maybe in the beginning you had something here and then when you analyzed it, you found you didn't.

MR. CHEN: From the plant safety point of view, you are right. It is nothing.

MR. EBERSOLE: But you didn't know that.

MR. CHEN: We didn't know until we looked into it.

MR. EBERSOLE: I guess we have to take that sort of news to the full committee. There is a special session set up for it.

MR. CHEN: Yes, sir.

MR. MICHELSON: So it is inconclusive from the viewpoint of providing us some indirect evidence of survivability of these plants from the electrical relay chatter viewpoint at least.

MR. CHEN: You are right, sir.

MR. EBERSOLE: And that is because of the absence of any duration.

MR. CHEN: Short duration and high frequency input.

MR. SIESS: I still think the high frequency might have been a better test of relay chatter, but something else.

MR. KERR: You only need one half cycles.

MR. SIESS: You had several cycles of the high frequency. But somebody has got to tell me more about --

MR. MICHELSON: You could go back and do some --

MR. SIESS: With respect to relays --

MR. MICHELSON: You might be able to do some laboratory work to correlate --

MR. SIESS: If it turns out they are sensitive to high frequency, we could probably do some pile-driving and not wait for an earthquake.

MR. ETHERINGTON: Or just a couple sticks of dynamite at the appropriate distance.

MR. EBERSOLE: But you did find one point where you had high acceleration on top of containment?

MR. CHEN: Yes, but that high acceleration corresponded to 20 Hertz with very low velocity and displacement.

MR. SIESS: What was the frequency of those angle brackets that were held --

MR. CHEN: After the earthquake event, the vendor, which is Kinemetrics, came in and took some tests of the brackets. The conclusion at that time was around 100 Hertz.

Well, there are other slides. I don't think there is any need to show you here because I think you have seen them 100 times.

MR. EBERSOLE: In view of the fact there is going to be a full committee presentation by some sort of edict that I don't really understand, I guess we can cover that tomorrow.

MR. CHEN: Okay.

Now the conclusion here we have is, again, this 1986 Ohio earthquake is of such a short duration, high frequency, low velocity, low displacement and low energy event it really doesn't have any engineering significance. Thank you.

MR. EBERSOLE: Thank you.

MR. ETHERINGTON: Using the same basis of evaluation, how much does the addition of this earthquake to the previous history of earthquakes add to the SSE acceleration?

MR. CHEN: How does this compare with SSE?

MR. ETHERINGTON: Is it a significant addition to previous history of seismicity in the region?

MR. EDELMAN: Would you like to respond to that, Dick? Dick Holt from Western Geophysics.

MR. HOLT: It certainly adds to our data base of eastern United States earthquakes of which we have very few. We have some records from the New Brunswick series of earthquakes and some from New Hampshire. This is the other addition.

MR. EDELMAN: The inclusion of this event would not change our design. That is what I would like to go through now briefly. This proof test, which may not have proved everything that we might have looked for as far as equipment chatter since it was such a short duration, took place at the plant, the structures and the systems were unaffected by the earthquake. It does not change any conclusion as to the geology and seismology of our site. The design earthquake bounds of the 1986 event and that the inclusion of the event does not change the design basis; that the plant seismic capability is there to accommodate the January earthquake, specifically this earthquake of short duration which we have covered; and that we have a number of follow on confirmatory programs that we at CEI are participating with the rest of the industry to gather as much information as we can.

We have worked with EPRI to gather as much data as they can and worked with us and provide our data to the staff and to EPRI and to USGS for additional what we think might be useful to the industry. Even though it proved there was no damage to our plant. We are part of the seismic owners group, CEI, and we are dealing with them in providing the data. They have been on site. The EPRI people looking at our data, analyzing, and we are providing all that information to them. We worked closely with all these people to provide that data. We think it will provide additional basis of information for the industry. I am not very happy that we were the plant to do a proof test 10 days prior to fuel load, but it did prove that the plant was designed and capable of doing that.

MR. EBERSOLE: What happened to the people in the control room? Did the swivel chairs roll at all and the emergency manuals fall out of the bookcases.

MR. EDELMAN: Nothing. No, sir. None of that happened.

MR. EBERSOLE: You couldn't even feel it.

MR. EDELMAN: We looked at our control room, no spurious annunciators. When I was recovery manager on site that evening, I talked to the shift that actually went on, I talked to the shift on after that. They said they noticed no spurious alarms in the control room. They knew

the buildings shook.

MR. EBERSOLE: No lights flickered?

MR. EDELMAN: No. We did have the alarm that went off which allowed us ˍo go into our emergency procedures. But they reacted very well and knew what to do and followed the procedures. There were no spurious actions in that respect.

MR. KERR: If you are asking was it felt, it was felt.

MR. EDELMAN: It was felt.

MR. EBERSOLE: In the control room they even felt it?

MR. EDELMAN: Yes. It was felt.

MR. SIESS: Could I follow up on Mr. Etherington's questions. You have a map in here, figure 4.3, and there is nothing in the area of the site the size of 6 historically.

MR. EDELMAN: I think I have a copy right there.

MR. SIESS: This shows 5 as the highest.

MR. EDELMAN: The site -- there were two earthquakes used to bound our site consideration. One is the Attica earthquake in New York and the one is the Anna earthquake.

MR. SIESS: But in your immediate vicinity, this is the highest?

MR. EDELMAN: Yes.

MR. SIESS: There would be no instrumental records for any of those other earthquakes?

MR. EDELMAN: Not to my knowledge.

MR. SIESS: This just bears out my theory that nuclear plants attract earthquakes.

(Laughter.)

MR. EDELMAN: I guess I will not comment on that.

That concludes our remarks and with some direction from you or from your staff as to how much detail we covered tonight that you would like covered, it was about a half hour presentation to the full committee.

MR. EBERSOLE: That is about the best I can define it.

MR. EDELMAN: I don't know if the staff wishes to make any comments now. Mr. Stefano is our project manager.

MR. STEFANO: Good afternoon. I am John Stefano. I am the Perry project manager. Have been the project manager since February 1982. I had the pleasure of presenting the initial SSER to the SACRS Subcommittee which was headed at the time by Dr. Ray and Dr. Ebersole was on that committee.

All I intend to do right here is to try to give you some sort of a feeling for where the NRC staff is

coming from; what they are going to be doing since we have received this information. The books that you have before you are hot off the press. We received them the same time you did. There are additional copies here which can be disseminated tomorrow if need be or anybody else you need to disseminate them with today. We have additional copies back at my office.

I would like to say that immediately upon being notified of the earthquake, an inspection team was immediately assembled by Region 3. They were dispatched immediately, the next day, to the plant site to begin their investigation of plant damage. NRR joined them the same day. I was a member of the NRR team that went out there. We spent the first and second day going through the plant from a preliminary standpoint. Our initial findings in walking through the plant at that time, which was around the 2nd, 1st and 2nd of February, was to confirm pretty much what the utility found by way of minimal damage. We did see some hairline cracks in concrete, most of which I understand the utility has records of as having occurred prior to the earthquake. We did confirm the leak in the flange of the hot water space heating tank which is nonseismic supported. And I understand that certain of us went out to check the circuits that tripped which were designed to trip.

Where we are right now is in the very preliminary early stages of our review. We really do not have a position on what the utility has analyzed thus far. We will certainly use what the utility has given us. We will also use the resources we have available within research, within NRR, our consultants, to do our own independent assessment of what we found and to hopefully present -- come to a conclusion that is similar to the utility's. We recognize very well that they are in a near-term licensing condition and we do intend to try to support that the best we can. We want to assure you, we are not going to rush this because they are 10 days away from licensing.

Essentially, the actions planned are as noted on this sheet. We are in the initial process of characterizing what that earthquake means to basically reaffirm the seismological and geological assumptions that we used in initial plant design basis, as is described in the FSAR and as we have evaluated in our SER, issued in May have 1982.

In terms of this, we will also include a structural design review where we will be comparing the measured versus predicted responses, the effect and impact of this so-called short duration high frequency exceedance to the design basis of the structures themselves.

In addition to that, we will be looking at piping and equipment which may be sensitive in this high frequency region to see if there is any impacts there.

We have amassed a team of our research people, people within NRR -- to name a few, Dr. Leon Rider; we have Mr. Bob Hermsn, who is heading up our mechanical equipment qualification design group; we have Phyllis Sobel and we have Arnold Lee, who are also, by the way, here to answer any specific questions you may have.

All this we hope will culminate in obtaining some preliminary internal findings by the 21st of this month which will be a week from this Friday. I plan to amass all of the people within NRR, NRC only, to try to come up with a picture of where we are.

I want to emphasize that the follow-up dates on SSER issue and Perry licensing are strictly very rough estimates and targets. Those dates really can't be firmed up until we meet on the 21st to decide where we really are on this thing.

As I understand it from our seismologists and geologists, this is the first plant outside of California where an earthquake occurred and there were instruments inside that were used to record it as opposed to trying to set charges outside someplace to see what the response would be in the building. So we do have -- we really have

a unique set of data here. I am not sure at this point whether we really understand what it means. So I think when you look at SSER data and Perry licensing date, that is really at this point sort of red.

However, we do wish to indicate to the ACRS that we will try to get your oversight of this draft SER as we put it out. Hopefully before the 7th. And we would request your assistance in that regard, if you could provide that to us.

Let us know where we are going, is there anything we are not covering, what is good and what is not good before we finalize that. It is my intent to try to beat that 7/7 date with a draft SER if we can do that.

One last point. Just coming here, in fact after coming here we found out that there are some preliminary unconfirmed report received from the USGS that there are some indications -- at least they think that the earthquake may have been precipitated by an underground injection plant which apparently is located somewhere between the plant site and the site of the earthquake. We don't know any specifics about that. As I say, this is just unconfirmed. We are looking into that. As I understand it, the injection is of agricultural waste which is injected quite a large depth into the ground, I understand something like 6000 feet below ground. So I don't know if that has any

input at all as to what we are finding here or any measure whatsoever. I just thought I would let you know that because we just found that out.

Are there any questions?

MR. SIESS: I have got two questions. I don't know who can answer that. The first one is fairly simple. Was there any instrumentation in the free field?

MR. STEAD: To the best of my knowledge, no.

MR. RIDER: There was a dam some seven kilometers to the south. There is a hospital in Erie, Pennsylvania which may have recordings of that. There are two records in distance which were not in the free field.

MR. SIESS: The seismic analysis and design, what structures, components or systems would be affected by the portion of the reg guide spectrum in the range above 15 Hertz?

MR. CHEN: We just found reactor building as second higher.

MR. SIESS: How much would that affect the design?

MR. CHEN: As we just showed, in comparison with the steel containment, even though acceleration was exceeded, relative displacement is below OBE value.

MR. SIESS: The design of the building would be forces from mode 22, I think that is the one, the 18 Hertz

mode --

MR. CHEN: We took the -- in the original design we took the modes out to 33 Hertz.

MR. SIESS: How much contribution did you get from the modes above --

MR. CHEN: The contribution, the participation factor at the mode of 18.4 Hertz is somewhat, about one-third to one-half of the first one. So as to contribution to which building, we haven't looked into them and determined that. So the only one we have looked into in detail is the containment which we showed the displacement.

MR. SIESS: And the piping wouldn't be falling in that range?

MR. CHEN: Piping, usually the frequency range is lower. Especially for this kind of displacement, usually the piping has a gap of about 1-1/16 of an inch.

MR. SIESS: Thank you.

MR. EBERSOLE: Any further questions?

MR. BENARO: I would just like to add one thing. This is indeed a very valuable data point for eastern earthquakes and we are trying as we go into this thing to break our attention into two categories. One is Perry plant-specific questioning, whether the seismic design basis and structural design are appropriate for Perry. But

secondly, there are a lot of things that we could learn from this about earthquake instrumentation or it doesn't make sense to put so many frequency range meters of earthquake response in a control room and have the shift supervisor interpret what they mean. There are many generic lessons to be learned. We are trying to not lose those and I am sure the committee will be interested in that for its follow-on value.

MR. EBERSOLE: Would that be something that might be mentioned tomorrow?

MR. BENARO: I was just going to make the same point tomorrow.

MR. EBERSOLE: Well, all I can point out is that tomorrow -- well, we are only 30 minutes tomorrow. From what [illegible] heard, the operator deserves at least 20 of that and the staff 5 and perhaps you can cover the other 5. How is that?

MR. BENARO: I would say 25 and 5.

MR. STEFANO: Exactly.

MR. EBERSOLE: That is the kind of ratio we are heading for.

MR. HERMAN: We will take a look at the equipment.

MR. SIESS: That was the containment building.

MR. CHEN: Yes, sir.

MR. SIESS: To what extent do the seismically calculated stresses govern the design of that construction?

MR. CHEN: Very small. The loss of coolant pressure design, thickness of the shell -- it is a small portion of that.

MR. EBERSOLE: Thank you.

There may be people who would want to go out. We are going into other topics. I will just pause for a minute if you want to clear.

(Pause.)

MR. GREENMAN: Good evening. I am Ed Greenman from the division of reactor projects in Region 3. I have with me Jeffrey Wright. The topical area on the agenda is the Fermi 50.54 improvement program, which while again not an event, was the culmination of a number of events and a number of problems at the Fermi station. I think it might be helpful in the way of background to go over a very, very brief history of what has transpired at the Fermi station since March of last year culminating in the issuance of the 50.54 letter in late December this year.

The lower power license for Fermi was issued on March 20, 1985. It was a Commission briefing for the full power license on July 10 of last year. And the license was issued on July the 15th.

On July 1, the evening of July 1, into July 2, a

premature criticality event occurred. The NRC was not aware of that event at the time of the Commission briefing. Nor was it mentioned at the Commission briefing. That event was not significant in and of itself technically.

MR. KERR: I have heard reports of this statement before. It seems to me, in fairness to the licensee, you should point out that although you perhaps were not aware of it being a criticality event, I must say I am still skeptical that the thing ever went critical. You certainly did know that an event had occurred.

MR. GREENMAN: That is a correct statement. Nonetheless the Commission was not briefed on that event by either the NRC or the licensee at the time.

MR. KERR: The resident inspector was. He knew the details of the event, he knew about the rad withdrawal and that there had been instrumentation observed.

MR. GREENMAN: As I know the Subcommittee is aware, that subject is still under discussion and is part of an OIA investigation internal to the staff.

At the time the full power license was granted, the NRC's view of licensee performance was favorable. There had been minimal operational-type events. There had been minimal difficulties in making a transition from an engineering organization into an operational phase. Since the time the license was issued and subsequent to

notification of the premature criticality, a confirmatory action letter was issued by Region 3 that restricted power operation of that utility to 5 percent power. The utility still has not operated above 5 percent power. There was no operation up until a scheduled shutdown which occurred in October of last year which was scheduled to do work on a remote shutdown panel. That outage was originally anticipated to have been concluded late November, early December of 1985 and the utility still remains in a shutdown condition today.

Projected availability from the standpoint of resolution of technical issues is now anticipated to be sometime in late February or early March, dependent upon resolution of some engineering issues which I will discuss a little bit later.

MR. MOELLER: When you say the utility or the plant remains shut down, that is in accord with NRC regulations?

MR. GREENMAN: They are shut down with the constraints of the confirmatory action letter which restricts them to 5 percent power. They are also shut down within the constraints of the 50.54 approach. However, there are a number of technical issues. I guess my view is from what we know today, the diesel generators which we briefed the Subcommittee on a month or so ago still remains

as a critical path issue for the utility, dependent upon the resolution of the engineering issues.

As I mentioned, we briefed you on the diesel generators a month or so ago. Nothing really new has happened with respect to the diesel generators. They have embarked upon a demonstration program for the NRC involving two of the four Fairbanks Morris diesels. They recently encountered a situation where they identified a portion of a scraper ring on the top of one of the crank lines, reevaluated and reinspected all of the pistons and did not find any damage. We have also recently identified a case of a viscosity change in the oil, which was a Shell product, an additive type product, where foaming was encountered in the oil. And they are involved in their test runs.

The utility has submitted an action plan to the NRC. We have just received that within the last few days and we have not made a determination as to the acceptability of that plant.

It involves a series of starts and extended runs to demonstrate the reliability of the diesel generators. It will also involve our assessment of what types of inspections they do after the fact to assure the operational capability of those diesel generators.

Since the plant started up, we have also encountered cracking in the turbine bypass lines. That

piping has been replaced with nominal one-inch wall thickness. NRC inspection of that area is still in progress.

There was a major failure of one of the two feedwater pumps at that utility. The south feedwater pipe, it is a TDI product. It experienced excessive vibration. The root cause, the licensee and we have attributed to failure on the part of the engineering department to translate warmup instructions into the operations procedure. Also an inadequate engineering assessment of the meaning of the vibration alarms and the fact that the computer that monitored what the vibration alarms were telling them was scaled off by about 2.5.

This resulted in catastrophic bearing failures, problems with the pedestal and a number of other issues. That particular feed pump has been restored and is ready to be returned to service and I think it will perform acceptably.

There is an outstanding issue with NRR that the licensee has requested a waiver for inability to meet general design criteria 56 on the traversing in-core probe. NRR is reviewing that request. One possible solution to that would be the installation of two ball check valves outside of containment and a request for relief to meet GEC 56 until the first refueling outage.

One personnel error has occurred that involved a rupture of the condensate storage tank up near the top of that particular tank and the loss of about 100,000 gallons of water.

The weld failure has not been repaired. The lie licensee is planning on using that as is and we are still in the process of evaluating the acceptability of that.

MR. REED: On one bullet up there you have the south reactor feed pump turbine. Is that a main feed pump turbine?

MR. GREENMAN: Yes.

MR. REED: Then you have the condensate storage tank. Are these safety-related pieces of equipment?

MR. GREENMAN: They are balance of plant equipment. Feedwater pumps and turbines are not safety-related. The same thing is true of the condensate storage tank.

MR. REED: And they could run at 60 percent power with that south reactor feed pump out of service?

MR. GREENMAN: They could run at reduced power on one pump. However, I would emphasize that the engineering repairs and the reviews of those repairs have been completed and that is simply awaiting testing now to return to service.

The same problem was not encountered on the

north feed pump. When we looked at all of the engineering calculations, all of the alignments, they were all aligned properly.

MR. REED: I am having a little problem why you looked at all the engineering calculations on the pump. It is not in the NRC jurisdiction, is it?

MR. GREENMAN: We have encountered a number of problems with calculations at the Fermi station. That was also the subject -- with respect to the south feedwater and pump, that was also the subject of a specific allegation that the NRR received from an outside source.

MR. REED: If it was an allegation, would it be in your horizon?

MR. GREENMAN: As far as a regulatory requirement, no, I don't have any regulatory requirement.

MR. REED: I know there is a study going on or will be a study on regulatory involvement in balance of plant. It seems to me that we are already involved. So why the study?

MR. GREENMAN: I think we are already involved. I think it is also of value to note that even though that is nonsafety-related, every time you have a problem with components of that type, you also put yourself at risk on challenges to safety systems in the plant.

MR. REED: If you ran at 60 percent power, one

boiler feed pump.

MR. EBERSOLE: Loss of one source of coolant water, you are not supposed to run that.

MR. REED: I am worried about empire building here.

MR. EBERSOLE: If you only have one feed pump continuously, wouldn't you worry about that?

MR. REED: I can look at these as indicators that they have equipment trouble on the secondary side and they have equipment trouble on the safety-related side. But I worry that what we are doing is creating a stairway to total involvement in, unto the toilet.

MR. EBERSOLE: On the other hand, the feedwater system is an entity.

MR. REED: The reason I worry about it is because I really think that the Nuclear Regulatory is over, in over their heads a great deal right now. That if we were to settle down to the real safety-related stuff and stay with it and do it right, we wouldn't have any time to be fiddling around in these other areas.

MR. EBERSOLE: If you did that, what you would have is great safety stuff that would run every time you needed it, every day.

MR. REED: Right. And that takes care of the core and we wouldn't have any problems with the core.

MR. EBERSOLE: But every day is too often.

MR. KERR: I am going to urge that this discussion continue.

MR. EBERSOLE: Thank you, sir.

MR. GREENMAN: Other problems that have been encountered by the utility involve either an absence of sufficient documentation at the time to support the reviews of environmental qualifications and absence of documentation to support the adequacy of seismic reviews, absence of reviews and potential problems with the calculation for concrete embedments and the hangers that they are hooked to, which have a potential impact on safety systems, problems with one of the RHR pumps, which appears to be in one case a manufacturing defect, which has been resolved with replacement of the motor. There is a potentially generic issue there which the NRC is reviewing on whether or not the laminated material was the correct application.

MR. EBERSOLE: You are looking at this plant pretty much the way Vogtle was being looked at for startup readiness review.

MR. GREENMAN: Yes, sir. And in this case I think it has been a combination of initiation on the part of the NRC, various questions and problems identified and then an approach by the licensee to resolve them.

MR. EBERSOLE: What is wrong with the remote shutdown panel?

MR. GREENMAN: The problem with that, that installation, that is what the October shutdown was scheduled to do. The startup or the completion of the outage was delayed when during the design reviews it was identified that there was some inadequacies in a couple of areas with separation and fire protection issues. And that has been resolved and the last couple of modifications on that particular issue are just undergoing a final details design review right now.

The most recent development is a finding that occurred in late January where changes, modifications, stress reports and calculations to drawings were not updated since September of 1984. The impact of this is not fully known at this point in time. The licensee is doing an extensive evaluation and will be meeting with the NRC this Friday to discuss the results of their evaluation.

It did result in an immediate impact from the standpoint of the Detroit Edison Company in declaring certain systems inoperable. This involved fire protection systems and ECCS systems.

MR. MICHELSON: What is the problem on the RHR pump?

MR. GREENMAN: The RHR pump motor problem, there

was a seizure that broke off a couple of components in the pump. It was shipped off-site and a spare motor from Brown's Ferry was brought in.

MR. MICHELSON: This related to a bearing clearance problem and so forth?

MR. GREENMAN: The initial installation, the pressure plate when it was installed in the pressure fingers were unbalanced on the Fermi pump and they broke and that gave a short circuit in the motor. It is these.

MR. MICHELSON: The thrust bearing failed? Is that what you are saying?

MR. GREENMAN: No. Not on the Fermi facility. It was a problem in windings.

MR. MICHELSON: I am sorry. Okay. It is entirely a motor problem.

MR. GREENMAN: Yes, sir.

MR. MICHELSON: That is not generic to any other plant then?

MR. GREENMAN: Well, the use of an epoxy might be generic to some other facilities. And the Office of Inspection and Enforcement is looking at that right now.

MR. MICHELSON: I see. Thank you.

MR. EBERSOLE: Is that it?

MR. GREENMAN: On the technical issues themselves.

Performance issues at the utility can be characterized into several broad areas. Detroit Edison has experienced a number of personnel errors; they have had problems with implementation and direction of their security program. While that particular topic was very briefly addressed in their response to the 50.54 efforts, a separate improvement program and a meeting will be held with the NRC to discuss that. So their resolution of that area is not known at this point in time.

An inability to disposition known problems, this is all part and parcel of the engineering issues that we have encountered. It is taking them a long time to resolve the feedwater pump problem. It is taking them a long time to resolve the diesel generator problem. It is taking them a long time to work their way out of engineering calculations and design reviews to satisfy themselves if all of their initial reviews were adequate. On the up side of that, to date none of the engineering reviews, reviews that have been done, have resulted in any situation that has required any hardware change at that plant. Phrasing that another way is that there was no hardware impact as the result of not doing the appropriate level of review or not documenting the appropriate level of review to date.

Ineffective communications, I think the example of the feedwater pump, I think problems with communications

that resulted in transfer of information with respect to the premature criticality all fall into that arena.

To give you an idea of the personnel errors, we did a review from March 20 to December 25 of the 78 licensee event reports. 41 of those involved personnel errors. Taking two subsets of that, from July 10 through September 10, nine of the 25 involved personnel errors; from September to November 25, almost half, eight of the 17, involved personnel errors.

Personnel errors in and of themselves on a new plant, people don't always perform perfectly. It has manifested itself at Fermi in a situation where there have been multiple violations of technical specification requirements, problems with meeting limiting conditions for operations in one case it involved having an ECCS system out of service.

MR. MICHELSON: Have you looked at other plants from the viewpoint of seeing whether this number of personnel errors in relation to total LERs is out of normal?

MR. GREENMAN: The number of personnel errors is out of normal. I wouldn't be in a position to say that total number of LERs for that time frame is out of normal.

MR. MICHELSON: You say that personnel error were out of order. I was kind of under the impression that about 40 percent of the LERs involve personnel error and

looking at your chart, about 40 percent of them involve personnel error.

MR. GREENMAN: In some cases we have exceeded 50 percent.

MR. MICHELSON: Give or take 10 on something like that is not unusual.

MR. GREENMAN: I think that figure is a figure that you have to be careful for. I say that for this reason: There is a disparity in the way licensees classify personnel errors. And --

MR. MICHELSON: These are the new LERs we are dealing with here. Whose classification of personnel error are you using? The one on the data base that NRC maintains or are you using the one off the LER?

MR. GREENMAN: The one off the LER.

MR. MICHELSON: If you use it off the sequence coded data base, that is where the 40 percent numbers come from?

MR. GREENMAN: That is correct.

MR. MICHELSON: That is not unusual to see that high number of other plants. That is why I wondered why Fermi is different.

MR. GREENMAN: It is unusual when you look at what the personnel errors caused.

MR. MICHELSON: This is just a nose count here.

MR. GREENMAN: I am not trying to compare utility A to utility B.

MR. MICHELSON: I was just trying to figure out why Fermi was pointed out for this particular problem.

MR. GREENMAN: It is the types of personnel errors.

Given all of this, in December of this year, the NRC collective views of Region 3, NRR and IE, we issued a 50.54 letter to the utility requesting additional information on how they would address certain issues. The basic items that that particular letter addressed, requested the utility to provide information on were the adequacy of their management structures and systems to operate the plant, the actions that they had taken and planned to take to insure their readiness to restart and support power ascension to operate the plant, and what short-term and long-term actions they plan to take to improve not only their regulatory performance but their operational performance.

The licensee responded formally by letter, January 29 of this year, and acknowledged that there was a need for improvement.

The basic bullets that were contained in their response were as follows: The chief executive officer, chairman of the board, in conversations with Region 3 and

in his letter, assumed the direct management and control to support the restart effort. As a matter of fact, he is planning as chairman of the board to spend three to four mornings a week at that plant. He also directed the president of the company to oversee the quality assurance program as another senior manager.

He tasked the organization to form an independent overview committee consisting of outside consultants, including a member of the MAC organization that had been on site before, to independently make recommendations to him and to the board of directors. That oversight committee held their first meeting and briefing on February 7 a week ago last Friday. We are expecting to have a copy of their report shortly. We don't have it at this date so I don't have any information as to what the overview committee told the board of directors and the chairman.

He has also established a reactors operations improvement plan which contains some 60-odd different areas which the utility has addressed to improve all aspects of their operation. A number of those have been completed and are under review by the Region 3 office. That particular program is in place.

MR. EBERSOLE: Is there anything like a safety review staff like we hear about with TVA that is

contemplated?

MR. GREENMAN: Not what I would characterize as a safety -- the independent oversight committee, but not a safety review staff that goes beyond the normal organization.

There have been some managerial changes that have been made to strengthen their performance and there are more contemplated. As an example, the utility is going outside to seek a senior vice presidential level type individual to move into that organization. That would result in a pyriamid of almost two-on-one senior vice presidents on-site reporting to the corporate office. Both in the engineering support as well as operations.

The utility is also developing a nuclear operations improvement plan with full implementation of that anticipated in May of this year.

One thing that is significant, in order to go into a power escalation phase, the combination of recommendations from the on-site management organizational structures, the combinations of recommendations and changes that are put in place and accepted from the overview committee will all be given to the chief executive officer and he will personally authorize any change in power.

MR. REED: I am sitting here having some deep thoughts about what I see up here. The CEO of this company

is a long time involved, old-time nuclear person. We won't mention any names. That company established years ago a safety review board right at the top, very top level of the company and had a nationally known person who was a member of the board with the CEO as president. It also had an in-between plateau of safety experts and reviewers who met on a regular basis. What I see up here is more of the same. I would be looking for a different solution. It seems to me over the years they have had all this stuff with great coupling and nationally known people and so on. Maybe there is something different rather than more plateaus and reshuffling of review boards and things like that.

I have to say that I may have a slight conflict of interest here. I did serve for a short term on one of their boards. Then I unfortunately made the mistake of joining the ACRS. So do you think this is going to determine something or improve something? To me, I am quizzical.

MR. GREENMAN: I think the fact that the Detroit Edison Company perceives that strengthening of management controls are needed is positive. I think the fact that they have established an independent overview group is positive. I am not of the opinion, based on the results that we have seen to date -- and part of this program is very, very new -- that the problems are solved at Fermi.

They have identified a number of goals.

MR. REED: They are the company that in addition -- which I thought was way up front in management -- they are the company that put all their offices on the job site in a special building and only a mile away from the plant. What I would think from my recollection is that this thing was just overloaded with management review and attention and so on. Is there something fundamental here that maybe is related to their large number of LERs or human errors or something?

MR. KERR: Wait a minute. We don't necessarily have a large number of LERs.

MR. REED: The proportions are the same.

MR. GREENMAN: I think I can only respond that given all of that, their performance has been disappointing.

MR. EBERSOLE: Can you come in on the accountability problem. Who is in charge of what? Can you check back to the accountability locus in this organizational structure?

MR. GREENMAN: I think accountability, I think accountability, it is clear that accountability and where accountability rests in response to the 50.54 letter, I think it was clear that accountability was established between engineering organizations and operational organizations at the time the facility was licensed. It is

not clear that communications were adequate. It is not clear that the organizations talked to each other properly.

MR. EBERSOLE: That is the perennial problem: Who is in charge of that?

MR. REED: I just can't figure this out. These people were the first people to put their total engineering and offices on the job site right at the plant. All kind of company were supposed to be there, if I was an inspection person, I would be looking for some key here. If they in fact have gotten more real problems than they should have at this point in time.

We talked about River Bend. It is a boiling water reactor and somebody said, they have got a lot of problems. We tried to explain that away. But here is another boiling water reactor, perhaps a little earlier vintage. I am not saying it is reactor-related. You look at the diesels, hey, we know what that is related to: lousy manufacturing and design. And that is throughout the industry. Maybe if you threw away some of these discardable aspects, you could get down to whether or not they need to do a complete reorganization of what looked to be outstanding at one time.

I could throw in the other thing, do they use aptitude testing for their rank and file people and their management?

MR. EBERSOLE: Is that it?

Any further comments?

Let me make a proposal to shorten this meeting by asking Ron Hernan a question. I believe we are going to be obligated to take the tech spec improvement program to the full committee in any case. And that your observations on it will be just as comprehensive then as they would be now.

MR. HERNAN: The way it is, there is agreement, the leader of the tech spec improvement group will be here Friday with his people to make the presentation.

MR. MICHELSON: How much time is he going to have?

MR. HERNAN: We would like to have a half hour for him.

MR. EBERSOLE: Since we have an hour and a half anyway.

MR. MICHELSON: Are we going to get the fire protection questions out of the way at that time then? They have questions on fire protection in tech spec changes?

MR. HERNAN: I personally don't think we are going to get into that much detail.

MR. MICHELSON: Well, when are we going to hit it?

MR. HERNAN: Our purpose of being here today and

Friday is to follow up on Harold Denton's comments to the full committee last month.

MR. MICHELSON: I thought this was following up on my request.

MR. HERNAN: Mr. Denton said that we are kind of at a milestone point in the tech spec program and we will come down and tell the committee.

MR. MICHELSON: I had some concerns on taking the fire protection aspects out of the tech specs and I guess putting them back in to the FSAR itself somehow, and how this was going to work in view of the kind of details associated with fire protection tech specs.

MR. HERNAN: We may be able to address that question fairly simply on Friday. There is some discussion in the Commission paper which we --

MR. MICHELSON: That wasn't too hopeful. I don't think they have much details understood of what a fire protection tech spec was.

MR. EBERSOLE: Let me poll the full committee for a second. We have an internal conference going on.

MR. MICHELSON: Let's ignore them.

MR. EBERSOLE: I am granting a 30-minute presentation for the tech specs Friday, okay?

MR. HERNAN: Okay.

MR. EBERSOLE: We will put out a block of time

for a half an hour or so for that. There are three other topics then that I have marked which I am going to suggest we carry to tech specs, that is McGuire, Turkey Point, ANO 1. And they will get about 10 to 15 minutes apiece. That will leave ample time to cover the tech specs.

MR. REED: I thought we had McGuire, Brunswick --

MR. EBERSOLE: We have McGuire, Brunswick, and ANO 1.

MR. REED: Yes.

MR. EBERSOLE: And that is it. We are going to automatically get Perry, the earthquake, the prior day. That ought to do it. Any further comments? No further comments?

MR. HERNAN: On Brunswick and ANO 1, you do not need a staff presentation for that?

MR. EBERSOLE: No.

MR. HERNAN: So you need staff on McGuire and Turkey Point.

MR. EBERSOLE: Yes.

MR. ALLISON: No staff on ANO 1?

MR. EBERSOLE: Let's see. That is a design deficiency. I didn't say any staff support on that. That is that single failure design problem. I think we can just get staff support on that.

MR. HERNAN: We -- you decided earlier in the meeting that we didn't need it.

MR. EBERSOLE: That is the single failure --

MR. REED: What we said was, have a full committee quick statement of the complexity and vulnerability of auxiliary feed and whether or not maybe we ought to be thinking beyond auxiliary feed.

MR. KERR: Do you want all this recorded?

MR. EBERSOLE: Off the record.

(Whereupon, at 6:00 p.m., the meeting was concluded.)

CERTIFICATE OF OFFICIAL REPORTER

This is to certify that the attached proceedings before the UNITED STATES NUCLEAR REGULATORY COMMISSION in the matter of:

NAME OF PROCEEDING: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE ON REACTOR OPERATIONS

DOCKET NO.:

PLACE: WASHINGTON, D. C.

DATE: WEDNESDAY, FEBRUARY 12, 1986

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission.

(sigt) Rebecca E. Eyster

(TYPED)

REBECCA E. EYSTER

Official Reporter

ACE-FEDERAL REPORTERS, INC.

Reporter's Affiliation

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT: PERRY EARTHQUAKE

DATE: January 31.1986

PRESENTER: JOHN J. STEFANO

PRESENTER'S TITLE/BRANCH/DIV: PROJECT MANAGER/PD#4/DBL/NRR

PRESENTER'S NRC TEL. NO.: 492-9473

SUBCOMMITTEE: OPERATIONS

- RESULTS OF INDEPENDENT NRC WALKDOWN OF PLANT - CONFIRMS CEI FINDINGS OF NO SIGNIFICANT PLANT DAMAGE

- NRC ASSESSMENT OF EVENT IN EARLY STAGE OF REVIEW - ACTIONS PLANNED ARE:

  - CHARACTERIZING EARTHQUAKE TO REAFFIRM SEISMOLOGIC/GEOLOGIC ASSUMPTIONS USED FOR PLANT DESIGN BASIS AS DESCRIBED IN FSAR/SER

  - STRUCTURAL DESIGN REVIEW:

    - COMPARISON OF MEASURED/PREDICTED RESPONSES

    - EFFECT/IMPACT OF SHORT DURATION/HI FREQUENCY EXCEEDANCE OF DESIGN BASIS SPECTRA ON STRUCTURES

  - PIPING/EQUIPMENT DESIGN REVIEW:

    - COMPARISON OF MEASURED/DESIGN BASIS SEISMIC LOADS

    - EFFECT AND QUANTITATIVE ASSESSMENT OF IMPACT OF SHORT DURATION/HI FREQUENCY EXCEEDANCE ON PLANT PIPING/EQUIPMENT

- SCHEDULED ACTIONS

  - NRC STAFF/CONSULTANTS PRELIM. FINDINGS/DATA NEEDS 2/21/86

    - PERRY SPECIFIC DESIGN BASIS

    - OTHER RECOMMENDED GENERIC ACTIVITIES

  - SSER ISSUED 3/7/86

  - PERRY 1 LICENSING TARGET 3/14/86

PROPOSED REVISION OF 10 CFR PART 20

"STANDARDS FOR PROTECTION AGAINST RADIATION"

ROBERT E. ALEXANDER
CHIEF, HEALTH EFFECTS AND OCCUPATIONAL RADIATION PROTECTION
OFFICE OF NUCLEAR REGULATORY RESEARCH

JANUARY 16, 1986

REASONS FOR REVISING PART 20

- UPDATE PRESENT PART 20 PROMULGATED IN 1957
- IMPLEMENT APPROPRIATE CURRENT RECOMMENDATIONS OF ICRP
- IMPLEMENT EPA PROPOSED FEDERAL RADIATION PROTECTION GUIDANCE FOR OCCUPATIONAL EXPOSURES
- IN GENERAL --

  TO ESTABLISH A SCIENTIFICALLY SOUND AND EXPLICIT HEALTH PROTECTION BASIS FOR PART 20 STANDARDS AND OTHER NRC REGULATORY ACTIONS

NEEDED IMPROVEMENTS IN THE PRESENT PART 20

- o UNIFORM RISK BASED STANDARD FOR DOSE & OTHER LIMITS
- o UPDATE NUCLIDE INTAKE LIMITS (MPC'S)
- o DELETE 5(N-18) RULE PERMITTING 12 REMS/YR
- o INTEGRATED LIMITS FOR INTERNAL AND EXTERNAL DOSES
  - ESPECIALLY IMPORTANT FOR NMSS LICENSEES
- o PROVIDE EXPLICIT DOSE LIMITS FOR MEMBERS OF THE PUBLIC
- o PROVIDE A CUTOFF ON COLLECTIVE DOSE (PERSON-REM) CALCULATIONS RATHER THAN INTEGRATE OVER INFINITE TIME AND SPACE

RISK COMPARISON: CURRENT 10 CFR PART 20 AND ICRP 26

1. CURRENT 10 CFR PART 20 PERMITS EXTERNAL WHOLE BODY DOSE OF 5 REM/YEAR (IN CERTAIN INSTANCES UP TO 12 REM/YEAR) AND IN ADDITION 15 REM/YEAR TO ANY ORGAN FROM INTERNALLY DEPOSITED RADIONUCLIDES (30 REM/YEAR TO BONE AND THYROID, 5 REM/YEAR TO GONADS).

2. ONLY THE DOSE TO THE ORGAN RECEIVING THE MAXIMUM DOSE (THE "CRITICAL ORGAN") IS LIMITING. RISKS TO ORGANS RECEIVING LOWER DOSES FROM THE SAME RADIONUCLIDE INTAKE ARE NOT CONSIDERED.

3. ICRP-26 LIMITS THE DOSE FROM ALL SOURCES: EXTERNAL PLUS INTERNAL (CONSIDERING ALL IRRADIATED ORGANS) TO 5 REM/YEAR "EFFECTIVE". 5 REM/YEAR "EFFECTIVE" MEANS THAT THE TOTAL RISK TO THE INDIVIDUAL EQUALS TO RISK FROM 5 REM/YEAR FROM EXTERNAL RADIATION ALONE.

4. THE PARAMOUNT CONSIDERATION IS THE RISK TO AN INDIVIDUAL AND NOT THE DOSE TO A SINGLE ORGAN. IF ONLY A SINGLE ORGAN IS IRRADIATED, HIGHLY UNLIKELY IN PRACTICE, THE ICRP DOSE LIMITS FOR THAT ORGAN ARE IN CERTAIN CASES HIGHER THAN PERMITTED BY CURRENT 10 CFR PART 20. THE RISK UNDER THOSE LIMITS, HOWEVER, NEVER EXCEEDS THAT OF 5 REM/YEAR OF EXTERNAL WHOLE BODY IRRADIATION.

5. IN ACTUAL CASES WHERE MULTIPLE ORGANS ARE IRRADIATED, FROM INTERNAL AND EXTERNAL RADIATION, THE TOTAL EFFECTIVE DOSE AND THE RISK WILL BE LOWER UNDER ICRP-26 SYSTEM OF DOSE LIMITATION.

RISK COEFFICIENTS COMPARISON

		Risk per rem
ICRP 26		
CANCER	MODEL USED: LINEAR ABSOLUTE PROJECTION	12.5×10^{-5}
GENETIC EFFECTS (FIRST TWO GENERATIONS)		4×10^{-5}
BEIR III		
CANCER	MODELS PRESENTED: LINEAR ABSOLUTE PROJECTION	16.7×10^{-5}
	LINEAR RELATIVE PROJECTION	50.1×10^{-5}
	LINEAR QUADRATIC ABSOLUTE PROJECTION	7.7×10^{-5}
	LINEAR QUADRATIC RELATIVE PROJECTION	22.6×10^{-5}
GENETIC EFFECTS (FIRST TWO GENERATIONS)		3×10^{-5}

NOTE: ALL MODELS ASSUME "NO THRESHOLD"

BRIEF HISTORY OF RULEMAKING

- 1977 PUBLICATION OF ICRP-26
- 1979 NMSS MEMO TO RES SUGGESTED REVISION OF PART 20
- INTEROFFICE WORKING GROUP, SCOPED EFFORT, IDENTIFIED ISSUES
- MARCH 1980 ANPR
- ESTABLISHMENT OF DRAFTING GROUP
- EPA DRAFT GUIDANCE ON OCCUPATIONAL EXPOSURES -- SECY-81-232 & COMMISSION RESPONSE
- MEETINGS WITH INTERESTED PARTIES
- NCRP/ACRS RECOMMEND WAITING FOR NCRP RECOMMENDATIONS, NEG. EDO RESPONSE
- INTERNAL NRC REVIEW, COMMENT, CONCURRENCE

ICRP-26 SYSTEM OF DOSE LIMITATION

- o JUSTIFICATION OF ANY RADIATION EXPOSURE
- o OPTIMIZATION OF RADIATION PROTECTION
- o LIMITATION OF DOSES
  - "EFFECTIVE DOSE EQUIVALENT" CONCEPT
    COMBINES INTERNAL AND EXTERNAL DOSES
    INCLUDES ALL ORGANS, WEIGHTING FACTORS
    BASED ON RISK

MAJOR CHANGES IN PROPOSED REVISION

EXTERNAL OCCUPATIONAL DOSE LIMITS, WHOLE BODY

1.25 REMS/QTR

OR

3 REMS/QTR AND

5 REMS/YR AVERAGE

5(N - 18) FORMULA

CHANGE TO

3 REMS/QTR

AND

5 REMS/YR

AND

"PLANNED SPECIAL EXPOSURES"

5 ADDITIONAL REMS/YR
WITH SPECIAL JUSTIFICATION

25 REMS LIFETIME LIMIT FOR SUCH
EXPOSURES

ANNUAL LIMIT INCLUDES INTERNAL DOSE

0 SLIGHTLY MODIFIED ICRP-26 CONCEPT

OTHER EXTERNAL OCCUPATIONAL DOSE LIMITS

EXTREMITIES:	18-3/4	REMS/QTR	CHANGE TO	50	REMS/YR
SKIN:	7-1/2	REMS/QTR	CHANGE TO	50	REMS/YR
LENS: *	1.25	REMS/QTR, OR	CHANGE TO	15	REMS/YR
	3	REMS/QTR AND			
	5	REMS/YR AVERAGE			

0 CHANGES FULLY CONSISTENT WITH ICRP-26

* TREATED AS PART OF WHOLE BODY

INTERNAL OCCUPATIONAL DOSE LIMITS

QUARTERLY INTAKE LIMITS FOR EACH NUCLIDE	CHANGE TO	ANNUAL INTAKE LIMITS FOR EACH NUCLIDE
(COMPLIANCE PREVENTS MOST HIGHLY EXPOSED ORGAN FROM EXCEEDING ITS OLD ICRP DOSE LIMIT)		(COMPLIANCE PREVENTS RISK TO ALL AFFECTED ORGANS FROM EXCEEDING THAT FROM 5 REMS/YR TO WHOLE BODY)
		LIMITS MUST INCLUDE EXTERNAL DOSE

0 CHANGES FULLY CONSISTENT WITH ICRP-26

INTERNAL OCCUPATIONAL DOSE CONTROL

- CHANGES WOULD PRIMARILY AFFECT NMSS LICENSEES
- RISK FROM AIRBORNE NUCLIDES AT NPP'S IS SMALL (98% ORGAN BURDENS <2% OF PERMISSIBLE)
- INTAKE LIMITS FOR SOME NUCLIDES WOULD BE LOWERED, PRIMARILY ALPHA EMITTERS ENCOUNTERED IN NMSS-LICENSED FACILITIES, E.G. URANIUM - FACTOR OF 6,
THORIUM - FACTOR OF 60,
AMERICIUM - FACTOR OF 50
- INTAKE LIMITS FOR SOME NUCLIDES WOULD BE INCREASED, PRIMARILY BETA EMITTERS
- SPECIAL PROVISIONS IN REVISED PART 20 FOR ALPHA EMITTERS DIFFICULT TO MEASURE AT NEW LIMITS (ANNUAL DOSE CONTROL RATHER THAN 50-YEAR INTEGRATED DOSE)
- ALL CHANGES CONSISTENT WITH ICRP-26

OCCUPATIONAL ALARA CONCEPT

§20.1 (c) SAYS EXPOSURES <u>SHOULD</u> BE MAINTAINED ALARA	<u>CHANGE TO</u>	REQUIREMENT TO DEVELOP, DOCUMENT AND IMPLEMENT A RADIATION PROGRAM THAT INCLUDES ALARA

INVESTIGATION LEVELS

CONCEPT NOT USED IN PRESENT PART 20	<u>CHANGE TO</u>	INVESTIGATION LEVELS BELOW REGULATORY LIMITS REQUIRED -- SET BY LICENSEE

DE MINIMIS EXPOSURE LEVELS

PRESENT PART 20 SETS LEVELS BELOW WHICH CERTAIN REQUIREMENTS DO NOT APPLY: THESE LEVELS ESTABLISH CONTROL POINTS, NOT TRIVIALITY

RETAIN SIMILAR LEVELS BUT ADD:

PROPOSED PART 20 WOULD SET A CUT OFF LEVEL OF 1 MREM FOR COLLECTIVE DOSE CALCULATIONS

NO DE MINIMIS LEVEL IS PROPOSED FOR INDIVIDUAL WORKER EXPOSURES

UK NATIONAL RADIOLOGICAL PROTECTION BOARD IN JANUARY 1985 ADVISED USE OF THE FOLLOWING DE MINIMIS LEVELS:

5 MREM/YR	INDIVIDUAL MEMBERS OF THE PUBLIC (1% OF ANNUAL DOSE LIMIT FOR PUBLIC)
0.5 MREM/YR	WHERE THE INDIVIDUAL MAY BE EXPOSED TO SEVERAL "DE MINIMIS" SOURCES

PROTECTION OF THE PUBLIC

RADIATION LEVELS IN UNRESTRICTED AREAS

PRESENT PART 20 (TO AN INDIVIDUAL):

- 2 MREMS IN ANY HOUR
- 100 MREMS IN ANY 7 DAYS
- 500 MREMS IN A CALENDAR YR (IMPLIED)

RADIOACTIVITY IN EFFLUENTS

PRESENT PART 20 (ANY TYPE FACILITY):

- PRESCRIBED CONCENTRATION VALUES, AVERAGED OVER 1 YR
- ADDITIONAL LIMITATIONS IF FOOD PATHWAY INCREASES DAILY INTAKE
- COMPLIANCE WITH 40 CFR PART 190 REQUIRED OF FUEL CYCLE/LWR LICENSEES

CHANGE TO

RADIATION LEVELS AND RADIOACTIVITY CONCENTRATIONS IN UNRESTRICTED AREAS

- 500 MREMS/YR, EXPLICIT, EFFECTIVE DOSE FROM ALL EXTERNAL/INTERNAL SOURCES
- 100 MREM/YR REFERENCE LEVEL

- PRESCRIBED CONCENTRATION VALUES MODIFIED AND RETAINED

CONTROVERSIAL ISSUES

IN THE PROPOSED

10 CFR PART 20

- Protection of Embryo/Fetus
- Annual Exposure Reports
- De Minimis Levels
- Optional Implementation Period
- Committed vs. Annual Dose

PROTECTION OF AN EMBRYO/FETUS

ISSUE: PROTECTION OF AN EMBRYO/FETUS AGAINST OCCUPATIONAL RADIATION EXPOSURE OF PREGNANT WORKER.

EFFECTS OF MATERNAL FACTORS ON PREGNANCY OUTCOME

Factor	Effect	Number Occurring From Natural Causes	Excess Occurrences Due To Maternal Factor
Radiation Risk			
Childhood Cancer			
Radiation dose of 1000 millirems received before birth	Cancer death	1200 per million	600 per million
Abnormalities			
A. Radiation dose of 1000 millirads received during specific periods after conception.			
4-7 weeks	Small Head Size	40 per thousand	5 per thousand
8-11 weeks	" " "	40 per thousand	9 per thousand
B. Radiation dose of 1000 millirads received during the following period after conception			
8-15 weeks	Mental Retardation	4 per thousand	4 per thousand

EFFECTS OF MATERNAL FACTORS ON PREGNANCY OUTCOME (CONT.)

Factor	Effect	Number Occurring From Natural Causes	Excess Occurrences Due To Maternal Factor
Nonradiation Risks			
Occupation			
Work in high risk occupations	Stillbirth or spontaneous abortion	200 per thousand	90 per thousand
Alcohol Consumption			
2-4 drinks per day	Fetal Alcohol Syndrome	1-2 per thousand (total births)	100 per thousand
4-10 drinks per day	Fetal Alcohol Syndrome	1-2 per thousand (total births)	200 per thousand
More than 10 drinks per day (chronic alcoholic)	Fetal Alcohol Syndrome	1-2 per thousand (total births)	350 per thousand
More than 10 drinks per day (chronic alcoholic)	Perinatal	23 per thousand	170 per thousand

EFFECTS OF MATERNAL FACTORS ON PREGNANCY OUTCOME (CONT.)

Factor	Effect	Number Occurring From Natural Causes	Excess Occurrences Due To Maternal Factor
Smoking			
-	Babies at birth tend to weigh less than babies of non smokers	-	-
Less than 1 pack per day	Perinatal Infant death	23 per thousand	5 per thousand births
One pack or more per day	Perinatal Infant death	23 per thousand	10 per thousand births

PROTECTION OF EMBRYO/FETUS

PRINCIPAL ALTERNATIVES

- INFORMED CONSENT (PRESENT NRC POSITION)
  - ONUS FOR PROTECTION ON WORKER
- LOWER DOSE LIMIT FOR ALL WORKERS
  - HIGHER COLLECTIVE DOSES AND COSTS
- LOWER DOSE LIMIT FOR ALL WOMEN
  - JOB DISCRIMINATION, INAPPROPRIATE LIMIT FOR SOME WOMEN
- LOWER DOSE LIMIT FOR FERTILE WOMEN
  - JOB DISCRIMINATION, INVASION OF PRIVACY
- LOWER DOSE LIMIT FOR WOMEN KNOWN TO BE PREGNANT
  - JOB DISCRIMINATION, INVASION OF PRIVACY, MODERATE OVEREXPOSURE POTENTIAL
  - ONUS FOR PROTECTION ON EMPLOYER
- LOWER DOSE LIMIT FOR WOMEN DECLARED TO BE PREGNANT
  - JOB DISCRIMINATION, HIGHER OVEREXPOSURE POTENTIAL
  - ONUS FOR DECLARATION ON WORKER

EPA IS PROPOSING THE FINAL ALTERNATIVE IN NEW GUIDANCE TO FEDERAL AGENCIES

PROPOSED EMBRYO/FETUS LIMIT

- PROVIDES DOSE LIMIT OF 0.5 REM TO EMBRYO/FETUS FOR ENTIRE GESTATION PERIOD.
- DECLARATION OF PREGNANCY WOULD BE VOLUNTARY.
- ONCE DECLARED, THE RESPONSIBILITY OF MEETING LIMIT WOULD SHIFT TO LICENSEE.
- AN ADDITIONAL 0.05 REM TO THE EMBRYO/FETUS PERMITTED IF EMBRYO/FETUS LIMIT HAS BEEN EXCEEDED BEFORE DECLARATION OF PREGNANCY.
- WOMEN WHO CHOSE NOT TO DECLARE THEIR PREGNANCIES WILL BE SUBJECT TO THE SAME OCCUPATIONAL LIMITS AS ANY OTHER NUCLEAR WORKER.
- CONSISTENT WITH EPA PROPOSED FEDERAL GUIDELINES, NCRP, ICRP, PREVIOUS COMMISSION GUIDANCE AND OTHER FEDERAL AGENCIES.

DE MINIMIS LEVELS

ISSUES: 1. SHOULD CALCULATIONAL "CUTOFF" LEVEL FOR COLLECTIVE DOSE ASSESSMENTS BE ESTABLISHED?

2. IF ESTABLISHED, AT WHAT LEVEL SHOULD THE CUTOFF BE SET?

USE OF DE MINIMIS DOSE CONCEPT

- GENERALLY, DE MINIMIS DOSE MEANS A DOSE WHICH HAS AN ATTENDANT RISK SO LOW THAT IT DOES NOT WARRANT REGULATORY ATTENTION.

- THERE ARE MANY POTENTIAL APPLICATIONS OF DE MINIMIS CONCEPT.

- NEITHER EXISTING NOR PROPOSED FEDERAL GUIDANCE INCLUDES ANY APPLICATION OF DE MINIMIS CONCEPT.

- DE MINIMIS DOSE CONCEPT IN PROPOSED 10 CFR PART 20 REVISION IS VERY LIMITED - ONLY FOR CALCULATIONAL "CUTOFF" FOR COLLECTIVE DOSE ASSESSMENTS.

- COLLECTIVE DOSE ASSESSMENTS ARE A MAJOR CONSIDERATION IN DECISIONS ABOUT:

  - BALANCE BETWEEN INDIVIDUAL DOSE AND POPULATION DOSE IN DETERMINING ALARA.

  - CHOICES AMONG AVAILABLE OPTIONS FOR PROVIDING RADIATION PROTECTION.

DE MINIMIS LEVELS IN UNITED KINGDOM

- 5 MREMS/YR, INDIVIDUAL DOSE FROM ALL NON-MEDICAL, CONTROLLED SOURCES (RISK OF 10^{-6}/YR, MORTALITY)
- 0.5 MREM/YR, INDIVIDUAL DOSE FROM ANY ONE NON-MEDICAL, CONTROLLED SOURCE
- INDIVIDUAL DOSES LESS THAN 0.5 MREM/YR NOT INCLUDED IN COLLECTIVE DOSE ESTIMATES USED FOR APPROVAL PURPOSES IF COLLECTIVE DOSE IS LESS THAN 100 PERSONREMS

THESE LEVELS WILL AFFECT:

- GOVERNMENT APPROVALS
- DESIGN AND PLANNING
- OPERATIONS
- WASTE MANAGEMENT
- DECOMMISSIONING

- THE DEGREE OF REASON EMPLOYED IN RADIATION PROTECTION
- UNNECESSARY EXPENDITURE OF RESOURCES
- DECISIONS OTHERWISE AFFECTED UNDULY BY TRIVIAL CONCERNS
- EXAGGERATED PROTECTIVE MEASURES THAT EXACERBATE OTHER HAZARDS
- PUBLIC UNDERSTANDING THAT RADIATION RISK IS DOSE-DEPENDENT

PROPOSED RULE IN PART 20 WOULD ALLOW A 1 MREM/YR CUT OFF IN COLLECTIVE DOSE CALCULATIONS (QUASI DE MINIMUS LEVEL)

QUASI DE MINIMIS RULE IN PROPOSED 10 CFR PART 20

- 1 MREM/YR CUT OFF IN COLLECTIVE DOSE CALCULATIONS MADE TO OBTAIN NRC APPROVALS (COMPARABLE UK LEVEL IS 0.5 MREM/YR)

- WOULD ELIMINATE CONSIDERATION OF VERY LOW DOSES DELIVERED AT VERY LOW DOSE RATES OVER EXTREMELY LONG PERIODS OF TIME TO EXTREMELY LARGE NUMBERS OF PEOPLE

- COLLECTIVE DOSES CALCULATED FROM INDIVIDUAL DOSES LESS THAN 1 MREM/YR ARE A HIGH PERCENTAGE OF THE TOTAL

- RISKS ASSOCIATED WITH 1 MREM/YR OR LESS SHOULD BE OF NO CONCERN TO THE EXPOSED INDIVIDUAL (ONE-FIFTH OF THE COMPARABLE LEVEL ADOPTED IN UK), ARE TRIVIAL WITH RESPECT TO REGULATORY CONCERN, AND SHOULD NOT AFFECT DECISION-MAKING PROCESSES

- IF A LONG-LIVED RADIONUCLIDE WERE RELEASED SUCH THAT EVERY US CITIZEN CONTINUOUSLY RECEIVED 1 MREM/YR (WORST CASE), THE THEORETICAL NUMBER OF PREMATURE DEATHS PER YEAR WOULD BE ABOUT 21 IN A POPULATION OF 220,000,000 PEOPLE.

- THE US DEATH RATE FROM CANCER IS ABOUT 400,000 PER YEAR AND FROM ALL CAUSES IS ABOUT 2,000,000 PER YEAR.

WHY ESTABLISH CALCULATIONAL CUTOFF DOSE?

- WILL PROVIDE A CONSISTENT BASIS FOR LIMITING THE EXTENT OF ASSESSMENT OF COLLECTIVE DOSES.
- WILL REMOVE SUBJECTIVITY OF UNKNOWNS AND UNCERTAINTIES LEADING TO POTENTIALLY INCORRECT JUDGEMENTS:
  - UNCERTAINTIES ASSOCIATED WITH DISTANCES FAR FROM THE SOURCE.
  - UNCERTAINTIES ASSOCIATED WITH LONG PERIODS OF TIME.
  - VANISHINGLY SMALL RISKS TO THE EXPOSED INDIVIDUAL, I.E., DE MINIMIS.
- WILL SIGNIFICANTLY REDUCE RESOURCE COMMITMENTS.

APPLICATION OF DE MINIMIS CONCEPT TO INDIVIDUALS

- WOULD ESTABLISH ANNUAL DOSE LEVEL SO SMALL THAT INDIVIDUALS USUALLY WOULD NOT CONSIDER THE RISK IN ARRIVING AT DECISIONS REGARDING THEIR ACTIONS.
  - APPLICATIONS WOULD BE MUCH BRAODER THAN CALCULATIONAL "CUTOFF".
  - NO REGULATORY ACTIONS WOULD BE IMPOSED FOR DOSES BELOW THIS LEVEL.
  - COST REDUCTIONS WOULD BE APPRECIABLE (E.G., RADWASTE DISPOSAL, EFFLUENT PROCESSING).
- WAS CONSIDERED DURING DEVELOPMENT OF PROPOSED RULE.
- COULD RESULT IN:
  - UNAPPROVED INCORPORATION OF RADIOACTIVE MATERIALS INTO CONSUMER PRODUCTS.
  - RELEASE OF VERY LOW-LEVEL WASTE STREAMS WITHOUT EVALUATION.
  - RAISING LEVELS, AS IN NPP TECHNICAL SPECIFICATIONS, AT WHICH CERTAIN CONTROLS OR EQUIPMENT MUST BE INSTALLED OR OPERATED.
- SUPPLEMENTAL INFORMATION OF PROPOSED RULE DISCUSSES AND SPECIFICALLY REQUESTS COMMENTS.

STAFF RECOMMENDATIONS

- ESTABLISH A CUTOFF LEVEL OF 1 MREM/YR ON INDIVIDUAL DOSES TO BE USED IN THE CALCULATION OF COLLECTIVE DOSE ESTIMATES FOR REG PURPOSES.

- SPECIFICALLY REQUEST PUBLIC COMMENT ON THIS ISSUE AND ON A DE MINIMIS LEVEL FOR INDIVIDUALS.

REPORTING OF WORKER DOSES

ISSUES: 1. SHOULD THE NRC REQUIRE ALL LICENSEES TO REPORT ANNUAL DOSES OF ALL MONITORED WORKERS TO NRC?

2. SHOULD THE NRC REQUIRE ALL LICENSEES TO REPORT ANNUAL DOSES TO WORKERS (WITHOUT REQUEST)?

EXPOSURE REPORTS

- THE NATIONAL INTEREST
  - The Federal Government is Expected to be Informed, and to Disseminate Information, Regarding Health and Safety in the Workplace
  - DOL Bureau of Labor Statistics (radiation exposure not included)
  - Exposure Data are Vital in Planning for the Administration of Justice (Workmans Compensation, Tort Law)

- NRC INTERESTS
  - NRC is Primary Source of Information for Congress, Administration, Other Federal Agencies, State Governments, Labor Unions, Scientific and Industrial Organizations, Special Interest Groups, News Media, United Nations, Foreign Governments, Among Others
  - Data Base for Decisions Regarding: Budget Needs; Research and Standards Development Priorities; Evaluating Licensee Performance; Areas to Emphasize in Licensing; Inspection and Enforcement Priorities
  - Data Base for Factoring Worker Risks into Decisions on Plant Safety Requirements
  - Data Base for Controlling Potential Transient Worker Problems
  - Triggering Timely Corrective Action
  - Data Base for Evaluation of Radiological Risks in NRC-Licensed Activities and Licensee Performance

EXPOSURE REPORTS...CONTINUED

- INDUSTRY INTEREST
  - Unbiased Government Data are Used by Industry to Verify that Safe Working Conditions are Maintained
  - Data are Used by Industry to Identify its Strong and Weak Performers and to Effect Improvements
- WORKER INTEREST
  - Assurance that Health Protection is Adequate, as Supervised by Government
  - Confidence that Adequate Legal Records are Available if Needed
  - Government Termination of Unsafe Conditions

PRESENT WORKER DOSE REPORTING REQUIREMENTS

0 ANNUAL STATISTICAL SUMMARY REPORTS--FROM 7 CATEGORIES OF LICENSEES

- LICENSEE IDENTIFICATION ONLY.
- NUMBER OF WORKERS RECEIVING DOSES WITHIN SPECIFIED DOSE RANGES.

0 REPORTS ON TOTAL DOSE RECEIVED BY WORKERS WHO TERMINATED EMPLOYMENT--FROM SAME 7 CATEGORIES OF LICENSEES.

- LICENSEE IDENTIFICATION
- WORKER IDENTIFICATION (NAME, SS NO., BIRTH DATE)
- PERIOD OF EMPLOYMENT
- MAGNITUDE AND TYPE OF RADIATION DOSE.

0 REPORTS ON DOSES RECEIVED BY WORKERS WHEN LIMITS ARE EXCEEDED

- LICENSEE IDENTIFICATION
- WORKER IDENTIFICATION
- TIME OF EXPOSURE
- MAGNITUDE AND TYPE OF RADIATION DOSE.

0 REPORT TO WORKER FROM ANY LICENSEE UPON REQUEST FROM WORKER

- MAGNITUDE AND TYPE OF RADIATION DOSE RECEIVED IN YEAR.

0 COPY REPORT TO WORKER WHENEVER INFORMATION REPORTED TO NRC BY NAMED INDIVIDUAL.

STAFF USES OF REPORTED RADIATION EXPOSURE DATA

THE DATA OBTAINED FROM PRESENT REPORTING REQUIREMENTS PERMIT ESTIMATES OF:

- O SIZE OF RADIATION WORKFORCE.
- O MAGNITUDE OF ANNUAL COLLECTIVE DOSE.
- O TRENDS OF EXPOSURES
  - COMPARISON OF DATA BY TYPE OF LICENSEE, AND BY LICENSEE WITHIN EACH TYPE
  - DOSE DISTRIBUTION VARIATION WITH TIME.
- O TRANSIENT WORKER DOSES.

EXAMPLES OF SPECIFIC USES OF REPORTED EXPOSURE DATA

0 MONITORING OF TRANSIENT WORKER OVEREXPOSURES

- INDIVIDUAL WORKING AT MULTIPLE PLANTS DURING A QUARTER IS COMMON.
- OVEREXPOSURES ARE KNOWN TO BE EXTREMELY RARE.

0 EXTENT OF AIRBORNE RADIOACTIVITY PROBLEM FOR NPP'S WORKERS

- NO OVEREXPOSURES REPORTED TO DATE.
- 98% OF THE REPORTED MEASUREMENT RESULTS FOR INTERNALLY DEPOSITED RADIOACTIVITY ARE LESS THAT 2% OF "PERMISSIBLE" QUANTITY.
- THEREFORE FEW NRC RESOURCES NEED BE EXPENDED FOR PROTECTION IN THIS AREA.

<u>ALTERNATIVES</u>

0 CONTINUE PRESENT REQUIREMENTS

- ANNUAL STATISTICAL SUMMARY REPORTS TO NRC FROM 7 LICENSEE CATEGORIES
- TERMINATION REPORTS TO NRC FROM 7 LICENSEE CATEGORIES (COPY TO WORKER).

0 EXPAND PRESENT REQUIREMENTS TO INCLUDE <u>ALL</u> LICENSEES

0 REQUIRE ANNUAL EXPOSURE REPORTS FOR EACH INDIVIDUAL WORKER TO THE <u>WORKER</u> FROM:

- 7 LICENSEE CATEGORIES
- ALL LICENSEES.

0 REQUIRE ANNUAL EXPOSURE REPORTS FOR EACH INDIVIDUAL WORKER TO THE <u>WORKER AND NRC</u> FROM:

- 7 LICENSEE CATEGORIES
- ALL LICENSEES.

0 TIMING OF REQUIREMENT FOR ANNUAL REPORTS OF NAMED INDIVIDUAL WORKER:

- INCLUDE IN FINAL RULE (PART 20)
- REVISE PART 19 SEPARATELY (IF REPORT ONLY TO WORKER IS REQUIRED BUT NOT IN PART 20).

REPORTING OF WORKER DOSES CONSIDERATIONS

O PRESENT REPORTING REQUIREMENTS PROVIDE INCOMPLETE DATA, E.G.,:

- NO ROUTINE OCCUPATIONAL EXPOSURE INFORMATION IS REPORTED BY MANY NRC LICENSEES, INCLUDING MEDICAL INSTITUTIONS. (MEDICAL WORKERS RECEIVE ABOUT HALF OF THE ANNUAL COLLECTIVE OCCUPATIONAL DOSE).

- NO ROUTINE OCCUPATIONAL EXPOSURE INFORMATION IS REPORTED BY THE AGREEMENT STATE LICENSEES. (AGREEMENT STATES REGULATE LICENSEES HAVING ABOUT 200,000 OF THE 500,000 WORKERS MONITORED).

- DATA SAMPLE FOR INTERNAL EXPOSURES TO ALPHA EMITTERS TOO SMALL AND UNTIMELY FOR STAFF NEEDS.

O THE EPA PROPOSED FEDERAL GUIDANCE WOULD REQUIRE EMPLOYERS TO REPORT WORKER ANNUAL DOSES TO THE WORKER WITH NO REQUIREMENT ON REPORTING TO ANY FEDERAL AGENCY.

- EPA PROPOSED FEDERAL GUIDANCE MAY NOT BE FINALIZED FOR SOME TIME.

<u>STAFF RECOMMENDATIONS</u>

0 CONTINUE THE PRESENT REQUIREMENT FOR ANNUAL STATISTICAL SUMMARY REPORTS AND TERMINATION REPORTS FROM 7 CATEGORIES OF LICENSEES WHOSE EMPLOYEES ARE CONSIDERED TO ACCEPT THE HIGHEST RISKS.

0 SPECIFICALLY REQUEST COMMENTS ON THIS ISSUE.

0 COMPLY WITH NEW FEDERAL GUIDANCE (ANNUAL EXPOSURE REPORT TO EACH WORKER) BY AMENDING REGULATIONS AFTER NEW FEDERAL GUIDANCE IS ISSUED.

<u>OPTIONAL IMPLEMENTATION PERIOD</u>

ISSUE: SHOULD THERE BE A TRANSITION PERIOD, FOLLOWING PUBLICATION OF THE NEW PART 20 AS A FINAL RULE, DURING WHICH COMPLIANCE WITH EITHER THE PRESENT OR NEW PART 20 WOULD BE ACCEPTABLE?

COMMITTED VS ANNUAL DOSE

ISSUE: SHOULD CONTROL OF LONG-LIVED RADIOACTIVE MATERIAL INTAKES BE BASED ON THE DOSE ACTUALLY RECEIVED EACH YEAR OR ON THE DOSE INTEGRATED OVER A PERIOD OF 50 YEARS?

COMMITTED VS ANNUAL DOSE FOR CONTROL OF LONG-LIVED AIRBORNE RADIONUCLIDES BY FEDERAL AGENCIES

- Controversy HIGHLIGHTTED by EPA/DOE/NRC reaction to ICRP-26
- ICRP has always used committed dose to internal organs (50-yr integrated dose), including ICRP-26
- AEC-Reg./NRC have always used committed dose
- AEC/ERDA/DOE have used combination of committed and annual dose
- EPA, in new guidance to Federal Agencies, is proposing combination of committed and annual dose
- 10 CFR Part 20 revision would use combination of committed and annual dose
- Federal agencies are in agreement on this issue
- The degree of protection provided for NRC-licensee and DOE-contractor workers would continue to be virtually the same.

COMMITTED VS ANNUAL DOSE CONTROVERSY IN HEALTH PHYSICS COMMUNITY

- THIS ISSUE INVOLVES CAREER INTERFERENCE, INTERNAL DOSE RECORDS AND REPORTS TO EMPLOYEES IN THE MANAGEMENT OF LARGE DEPOSITIONS; NOT COVERED IN FEDERAL GUIDANCE.

- HEALTH PHYSICS COMMUNITY DIVIDED ON THIS ISSUE.

- ISSUE WILL BE VOTED ON BY THE HEALTH PHYSICS SOCIETY -- THE FIRST TIME VOTING OF THIS TYPE HAS BEEN HELD.

- THE RESULTS OF THIS VOTE WILL BE INFLUENTIAL ON THE STAFF AS PUBLIC COMMENTS ON 10 CFR PART 20 ARE ANALYZED.

CAREER INTERFERENCE: RECORDING: REPORTING

- A WORKER ACCIDENTALLY RECEIVING A SUFFICIENTLY LARGE INTAKE WOULD RECEIVE AN EFFECTIVE DOSE EQUIVALENT LARGER THAN THE 5 REMS PER YEAR LIMIT THE REST OF HIS/HER LIFE.
- SINCE THE RISK FROM THIS DEPOSITION WOULD NOT BE ALTERED BY SUBSEQUENT INTAKE OR EXTERNAL EXPOSURE LIMITATIONS, WHILE CAREER INTERFERENCE COULD IMPOSE SEVERE ECONOMIC (AND OTHER) PENALTIES, PRESENT NRC REGULATIONS (AND THE NEW PART 20) WOULD, AT THE BEGINNING OF THE NEXT QUARTER (YEAR) DISREGARD THE PRESENCE OF THIS DEPOSITION.
- THE LICENSEE WOULD BE REQUIRED TO RECORD THE INTAKE AND AN ESTIMATE OF THE COMMITTED DOSE EQUIVALENT (EFFECTIVE COMMITTED DOSE EQUIVALENT AFTER REVISED PART 20 BECOMES EFFECTIVE).
- THE LCIENSEE WOULD BE REQUIRED TO REPORT THIS INFORMATION TO THE NRC AND TO THE WORKER.
- FOR CERTAIN NUCLIDES DIFFICULT TO MEASURE IN QUANTITIES ASSOCIATED WITH THE DOSE COMMITMENT, RECORDING AND REPORTING OF THE ANNUAL EFFECTIVE DOSE WOULD BE ALLOWED BY THE NEW PART 20, AND LARGER INTAKES WOULD BE ALLOWED UNDER PRESCRIBED CONDITIONS.
- INDIVIDUAL DETERMINATIONS WILL BE CONTINUED FOR DOE WORKERS SO EXPOSED, INCLUDING COMPENSATORY DOSE LIMITATIONS IF CONSIDERED NECESSARY BY DOE MEDICAL AUTHORITIES.
- DOE WILL CONTINUE TO RECORD AND REPORT ANNUAL DOSES.

ESTIMATED COST TO LICENSEES* OF IMPLEMENTING 10 CFR PART 20 REVISION

$33,000,000 INITIAL COST

$ 7,800,000 ANNUAL COST

BASED ON NRC/EPA CO-SPONSORED CONTRACT THAT SURVEYED LICENSEES

* INCLUDES AGREEMENT STATE LICENSEES

PERSPECTIVE ON ESTIMATED COSTS

- MUCH OF ECONOMIC IMPACT THAT IS ASSIGNED TO PROPOSED REVISION OF PART 20 HAS BEEN, IS BEING, OR WILL BE COMMITTED WHETHER OR NOT REVISION IS PROMULGATED.

- MANY LICENSEES VOLUNTARILY IMPLEMENTING RECOMMENDATIONS OF ICRP 26 AND 30, BECAUSE RECOGNIZED AS "GOOD PRACTICE" AND LIKELY TO BE HELPFUL IN MITIGATING LIABILITY CLAIMS

- COST OF SPECIAL EQUIPMENT, SUCH AS LUNG COUNTERS, PROCESS CHANGES, ROBOTICS, AND OTHER MAJOR MODERNIZATION ACTIVITIES, HAVE AND WILL BE "CHARGED" TO THE PART 20 REVISION, ALTHOUGH THEY ARE NOT REQUIRED AND WOULD BE INCURRED ANYWAY FOR OTHER REASONS

- ESTIMATED COSTS TAKE NO CREDIT FOR SAVINGS FROM CHANGES IN TECH SPECS, LIC. CONDITIONS, ETC. THAT COULD RESULT FROM PROMULGATION OF PROPOSED REVISION

BENEFITS OF PROPOSED REVISION

- REGULATION WILL REFLECT ICRP COHERENT RISK-BASED SYSTEM AND USE WIDELY-ACCEPTED CONTEMPORARY SCIENTIFIC KNOWLEDGE

- ANNUAL AND LIFETIME DOSES TO WORKERS RECEIVING HIGHEST EXPOSURES & WORKERS IN URAN. MILLS & FUEL FABR. WILL BE REDUCED

- WILL PROVIDE METHOD FOR SUMMING EXTERNAL AND INTERNAL EXPOSURES -- ESPECIALLY IMPORTANT FOR SOME NMSS LICENSED ACTIVITIES

- PUBLIC DOSE LIMITS ARE CLEARLY IDENTIFIED

- WORKERS AND PUBLIC SHOULD BETTER UNDERSTAND HEALTH RISK BASE AND PROTECTION PROVIDED

- CUTOFF ON COLLECTIVE DOSE EVALUATIONS WOULD ELIMINATE CONSIDERATION OF INSIGNIFICANT HEALTH RISKS

- IMPROVES REQUIREMENTS ON RADIATION SAFETY; E.G., TO PREVENT ACCESS TO VERY HIGH RADIATION AREAS, POSTING OF AREAS USED FOR MEDICAL RADIATION TREATMENTS, REQUIRED APPLICATION OF ALARA

- WOULD INTRODUCE SI (METRIC) RADIATION UNITS INTO NRC REGS

STAFF RECOMMENDATIONS

- CONCURRED IN BY ALL OFFICES INVOLVED
- PROPOSE EXTENDED IMPLEMENTATION PERIOD OF 5 YEARS FROM PUBLICATION OF FINAL RULE
- PUBLISH PROPOSED REVISION OF PART 20 FOR PUBLIC COMMENT
- SPECIFICALLY REQUEST COMMENTS ON CONTROVERSIAL ISSUES
- ALLOW EXTENDED PERIOD (120 DAYS) FOR COMMENT

Agenda for ACRS Subcommittee
Meeting on February 12, 1986
1:30 p.m.
Room 1046, H Street

RECENT SIGNIFICANT EVENTS

Date	Plant	Event	Presenter/Office telephone	Page
11/3/85	McGuire 1	Start-Up with Degraded HPSI System	J. Giitter, IE 492-9001	2
1/7/86	Brunswick 2	Target Rock Two-Stage SRV Setpoint Drift	M. Wegner, IE 492-4511	4
1/7/86	Turkey Point	Stop Check Valve Failures	R. Kiessel, IE 492-8119	5
1/9/86	Palo Verde 1	Reactor Trip	E. Licitra, NRR 492-8599	8
1/9/86	Palisades	Loss of Offsite Power	J. Giitter, IE 492-9001	10
1/28/86	Robinson	Loss of Offsite Power	G. Requa, NRR 492-9798	13
1/14/86	ANO 1	Design Deficiency in Emergency Feedwater System	G. Vissing, NRR 492-8796	14
1/29/86	Riverbend	Review Of Start-Up Test Program	E. Weiss, IE 492-9005	18
1/31/86	Perry	Earthquake	J. Stefano, NRR 492-9473	20
	Fermi	50.54(f) Improvement Program	E. Greenman, Reg 3 312-790-5518	NOT IN PACKAGE
	----	Technical Specification Improvement Program	R. Hernan, NRR 492-9519	21

McGUIRE UNIT 1 - START-UP WITH DEGRADED HPSI SYSTEM
NOVEMBER 3, 1985 (J. GITTER, IE)

- PROBLEM: FAILURE TO REPAIR VCT ISOLATION VALVE MOTOR OPERATORS PRIOR TO START-UP WOULD HAVE PREVENTED VCT ISOLATION ON SI SIGNAL.

- SAFETY SIGNIFICANCE: HPSI MAY NOT HAVE FUNCTIONED AS REQUIRED ON A REAL DEMAND.

- DISCUSSION:

- AT 0640 ON NOVEMBER 2, A LOSS OF INST. AIR (SHARED BY BOTH UNITS) CAUSED BOTH UNITS TO TRIP AND SAFETY INJECTION IN UNIT-1.

- VCT ISOL. VALVES CLOSED AS REQUIRED; HOWEVER, THE VALVE MOTOR OPERATORS WERE LATER FOUND BURNED OUT.

- PRIOR TO START-UP, OPERATORS MANUALLY OPENED VALVES, BUT DID NOT REPAIR THE MOTOR OPERATORS.

- UNIT START-UP COMMENCED AT ABOUT 0600 ON 11/3. THE UNIT WAS IN MODE 2 FOR ABOUT 6 HOURS.

- TWO MAJOR CONCERNS 1) IS BORON CONC. LESS THAN THAT ASSUMED IN SAFETY ANALYSIS? AND 2) COULD VCT COVER (H_2 AND N_2) BECOME ENTRAINED IN THE CHARGING PUMP SUCTION PATH RESULTING IN GAS BINDING OF PUMPS?

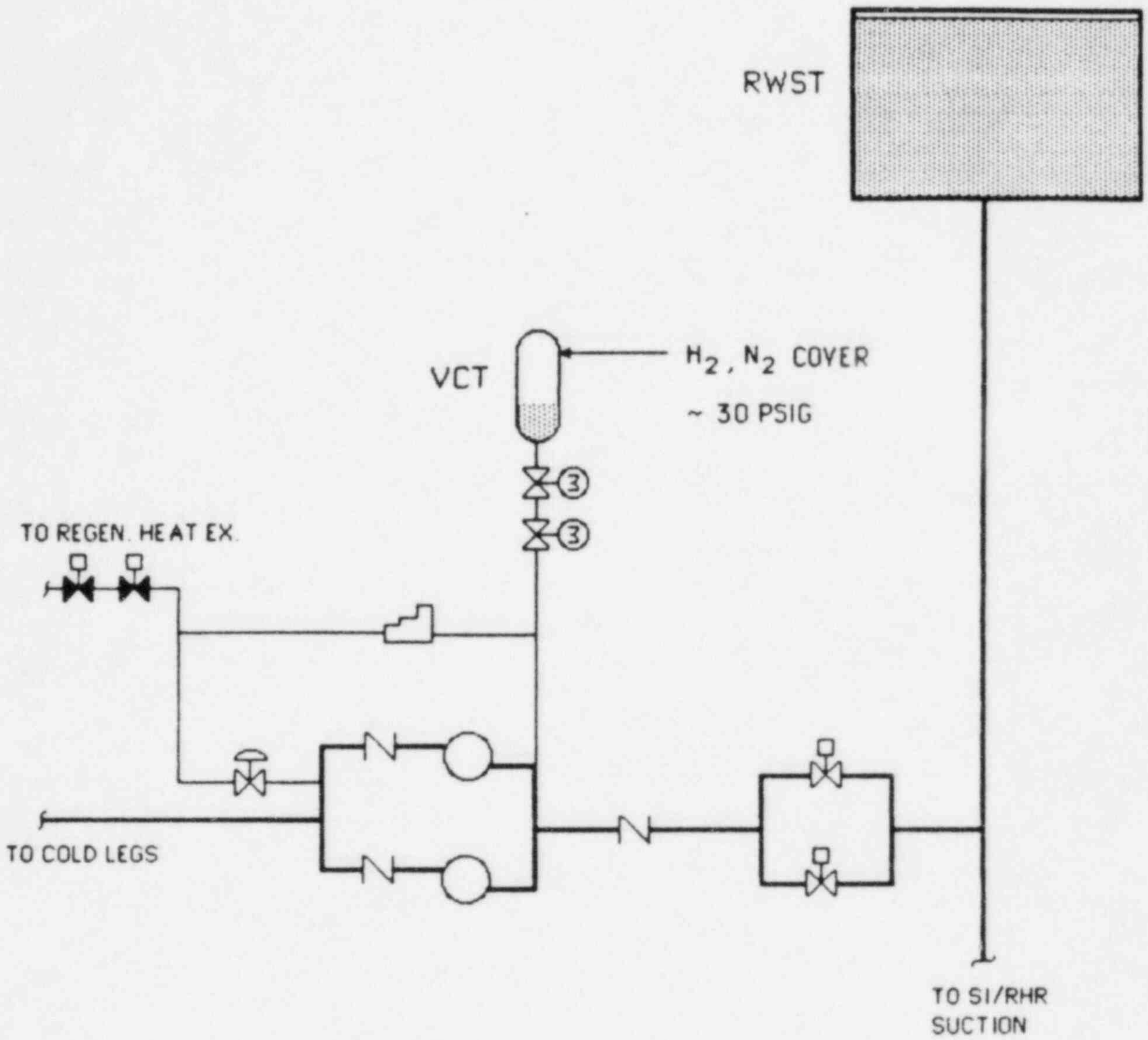
- NRC (OP CENTER) WAS NOTIFIED AT 1218 ON 1/14/86.

- FOLLOW-UP:

- VALVE OPERATOR REPAIRED.

- REVIEWED DESIGN OF OTHER MOTOR OPERATED VALVES IN PLANT.

- PERFORMED TEST TO DETERMINE TIME REQUIRED FOR OPERATOR TO CLOSE ISOLATION VALVES. TIME REQUIRED WAS 19.4 MINUTES. AT 20 MINUTES VCT LEVEL WAS AT 20%.


RWST
VCT
H_2, N_2 COVER
~ 30 PSIG
3
3
TO REGEN. HEAT EX.
TO COLD LEGS
TO SI/RHR
SUCTION

BRUNSWICK 2

TARGET ROCK TWO-STAGE SRV SETPOINT DRIFT, 1/7/86, MARY S. WEGNER

- 1/7/86 CP&L REPORTED 6/11 BSEP 1 VALVES LIFTED AT 200 PSIG OVER SETPOINT AT WYLE LAB

- HISTORY

  THREE STAGE VALVE SPURIOUS OPENINGS AND FAILURE TO RESEAT

  TWO STAGE REPLACEMENT BEGUN 1978

  HATCH 1 HAD 11 OF 11 FAILED TO OPEN AT SETPOINT

  OWNERS GROUP FORMED TO INVESTIGATE PROBLEM, POSE SOLUTION

  CAUSE ATTRIBUTED TO LABYRINTH SEAL GALLING AND SEAT-TO-DISC BONDING

  ENHANCED MAINTENANCE SUGGESTED

  RECURRING PROBLEMS

  NRR LETTER TO OWNERS GROUP

  SEAT REPLACEMENT PROGRAM

- THE BRUNSWICK INCIDENT

  WYLE TESTING

  OBSERVATIONS AT WYLE

- HATCH 1 RESULTS

- MILLSTONE 1 RESULTS

- GENERIC ISSUE B55

TURKEY POINT UNITS 3 AND 4 - STOP CHECK VALVE FAILURES NOV. 1985 THRU JAN. 1986 - (R. KIESSEL, IE)

PROBLEM: AFW STOP CHECK VALVE GUIDE PIN FAILURES

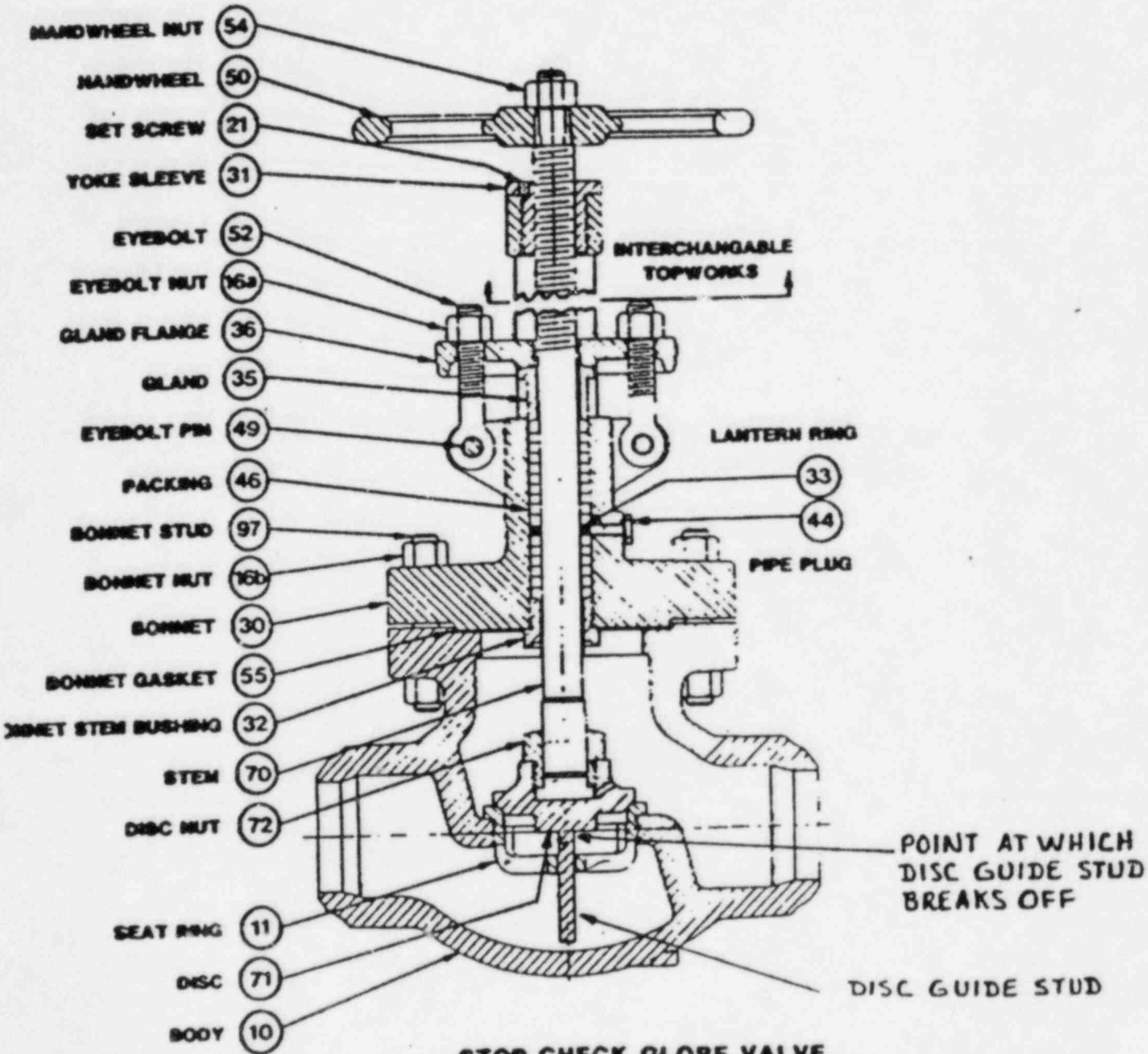
SAFETY SIGNIFICANCE: AFW VALVE/SYSTEM CONCERNS AND GENERIC IMPLICATIONS OF LOW FLOW THROUGH CHECK VALVES

CIRCUMSTANCES:

- BETWEEN NOV. 1985 AND JAN. 1986, NUMEROUS FAILURES OF STOP CHECK VALVES IN STEAM SUPPLY SYSTEM TO AFW PUMPS
- STOP CHECK VALVES LOCATED UPSTREAM AND DOWNSTREAM OF MOTOR OPERATED VALVES THAT OPENS WHEN REQUIRED TO INITIATE FLOW
- STOP CHECK NORMALLY OPEN - PREVENTS BACKFLOW IN EVENT OF STEAM LINE BREAK
- MODE OF FAILURE - DEGRADATION OF DISC AND DISC NUT
- FAILURE DUE TO LOW STEAM FLOW CONDITIONS CAUSED BY SLIGHT LEAKAGE OF NORMALLY CLOSED MOV
- LOW FLOW CAUSED VIBRATION AND CHATTERING BREAKING DISC GUIDE FROM DISC
- LOOSE DISC GUIDE PREVENTED FULL CLOSURE AND FULL OPENING OF VALVE. ALSO FREE TO TRAVEL

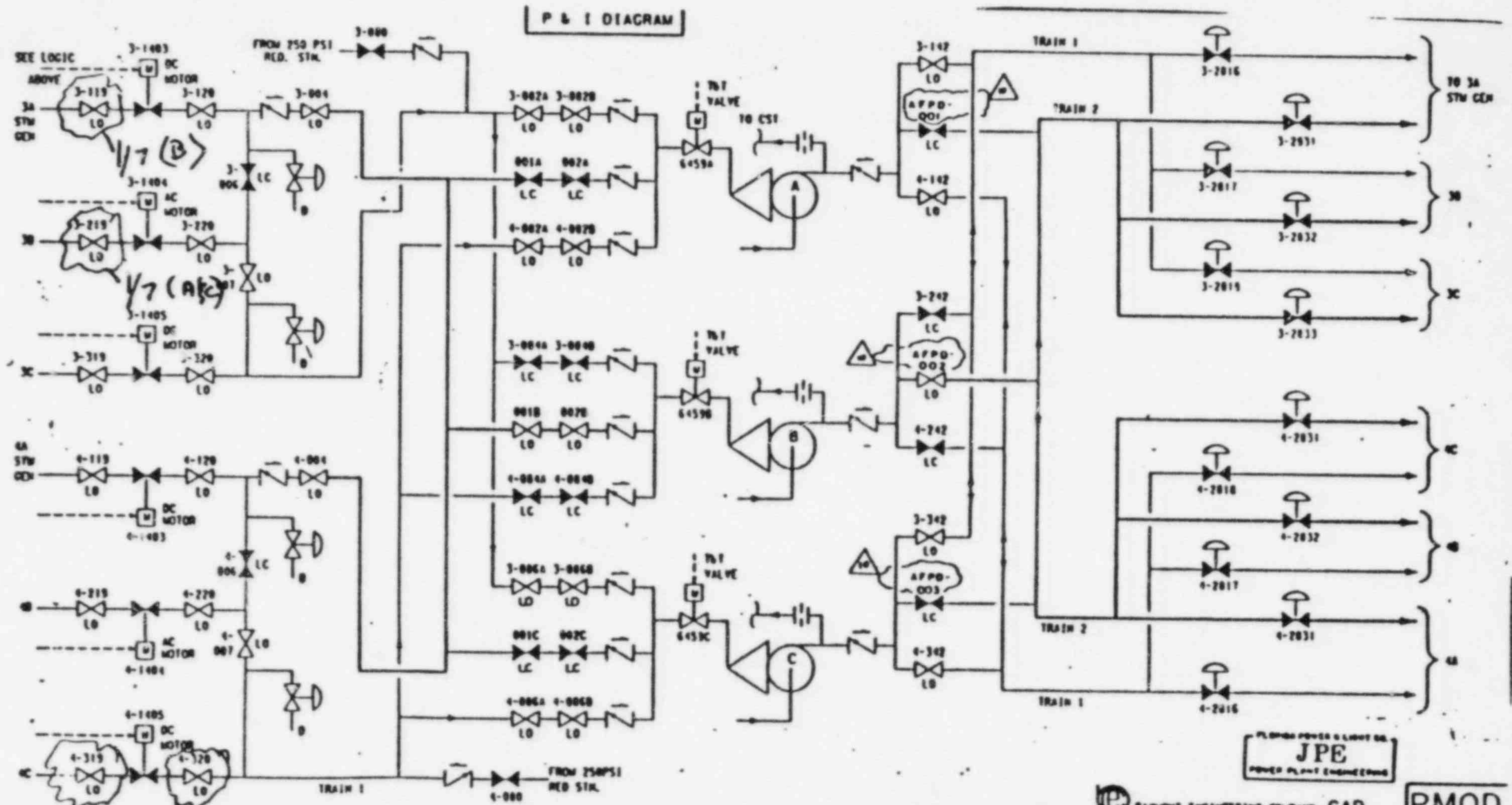
FOLLOWUP:

- AFW SYSTEM WAS INSPECTED
- ALL MISSING GUIDE PINS LOCATED AND REMOVED
- ALL VALVES REPAIRED WITH HIGHER STRENGTH MATERIAL USED IN DISC GUIDE
- FAILURE ANALYSIS AND METALLURGICAL EXAMINATIONS PERFORMED
- LICENSEE COMMITTED TO REGULAR RADIOGRAPHIC EXAMINATION OF VALVES
- LICENSEE CONSIDERS THIS AS INTERIM REPAIR PENDING COMPLETION OF THE STUDY BY ITS AFW ENHANCEMENT TASK FORCE
- INFORMATION NOTICE 86-09, "FAILURE OF CHECK AND STOP CHECK VALVES SUBJECTED TO LOW FLOW CONDITIONS" WAS ISSUED FEBRUARY 3, 1986



STOP CHECK GLOBE VALVE

P & I DIAGRAM



NO.	DATE	REVISION	BY	CK.	COR.	APP.
10	7/?/??	AS-BUILT PER DWP-461				
9	-/?/??	AS-BUILT PER DWP-155				
8	?/?/84	ISSUED AS-BUILT FOR PC/M 80-44				
7	5/4/84	[illegible]				

BECHTEL ENGINEERING COMPANY - CAD

PMOD

FLORIDA POWER & LIGHT CO.

TURKEY POINT PLANT
UNITS 3 & 4
AUXILIARY FEEDWATER
PUMP START
LOGIC, P&I AND CONTROL SYST.

DRAWN BY:
CHECKED BY:
APPROVED: WCM
DATE:

5610-T-L1
SHEET 15

REV.
10

JPE
POWER PLANT ENGINEERING

7

PALO VERDE UNIT 1 - COMPLICATIONS FOLLOWING TURBINE TRIP TEST FROM 100% POWER JANUARY 9, 1986 - (E. LICITRA, NRR)

CIRCUMSTANCES:

- TURBINE TRIP TEST STARTED FROM 100% (2250 PSI AND 565°F COLD LEG) AT APPROXIMATELY 13:25
- NON 1E HOUSE LOADS FAILED TO TRANSFER TO OFFSITE POWER DUE TO FREQUENCY MISMATCH BETWEEN GRID AND NON 1E BUSES
- POWER LOSS AFFECTED RCS PUMPS, MFW PUMPS, CIRCULATING WATER PUMPS AND STEAM DUMP/BYPASS CONTROL SYSTEM
- REACTOR TRIP DUE TO FLOW PROJECTED LOW DNBR
- STEAM BYPASS CONTROL SYSTEM (SBCS) VALVES RECEIVED QUICK OPEN SIGNAL BUT RECLOSED ALMOST IMMEDIATELY (DUE TO LOSS OF POWER)
- ONE OF THE FOUR LOWEST SETTING (1250 PSIG) MAIN STEAM SAFETY VALVES LIFTED WHEN SG PRESSURE INCREASED RAPIDLY - STAYED OPEN FOR 43 SECONDS
- WHEN SAFETY VALVE RESEATED, FIVE SBCS VALVES MODULATED OPEN - STAYED OPEN FOR 45 SECONDS WITH TWO ADDITIONAL OPEN/CLOSE CYCLES DURING NEXT 40 SECONDS
- MANUAL CONTROL TAKEN OF INDIVIDUAL SBCS VALVES TO STOP COOL-DOWN - ALSO MSIS OCCURRED DUE TO LOW SG PRESSURE
- AFTER STABILIZATION, OPERATOR RESTABLISHED COOLDOWN VIA ONE SG, TWO ADVs AND ONE AFW PUMP
- POWER RESTORED TO NON 1E BUSES AT 13:28 (3 MINUTES AFTER START OF EVENT)
- RCS PUMP FLOW RESTORED AT 14:08
- EVENT TERMINATED AT 14:49

FOLLOW UP:

- CONSIDERING DESIGN CHANGE TO SYNCHRONIZATION CHECK RELAY ASSOCIATED WITH AUTO TRANSFER
- MAIN STEAM SAFETY VALVE SETTINGS CHECKED AND FOUND OK
- CONSIDERING UNINTERUPTABLE POWER TO STEAM DUMP/BYPASS CONTROL SYSTEM
- RCS PUMP COASTDOWN FOUND FASTER THAN DESIGN - ONLY AFFECTS LOSS-OF-FLOW SAFETY ANALYSIS
- COLSS AND CPC PENALTIES IMPOSED WHILE CE EVALUATES FAST COASTDOWN EFFECT
- TURBINE TRIP TEST SUCESSFULLY RERUN ON JAN 24, 1986 (NON 1E LOADS POWERED OFF OF STARTUP TRANSFORMER)

PALISADES - LOSS OF OFF-SITE POWER

JANUARY 9, 1986 (J. G. GIITTER)

° PROBLEM

LOSS OF OFF-SITE POWER WITH 1 DIESEL-GENERATOR OUT OF SERVICE. VOLUNTARY DISCONNECT FROM GRID DUE TO STEAM OBSERVED FROM 4160V SYSTEM.

° SAFETY SIGNIFICANCE:

LOSS OF OFF-SITE POWER WITH 1 OF 2 D/Gs OUT OF SERVICE (TORN DOWN); VESSEL DRAINED DOWN TO APPROXIMATELY FLANGE LEVEL; HEAD NOT TENSIONED.

° CIRCUMSTANCES:

INITIAL CONDITIONS:

- UNIT HAD BEEN IN COLD SHUTDOWN FOR REFUELING SINCE NOVEMBER 30, 1985 (40 DAYS).

- HEAD WAS POSITIONED ON VESSEL BUT NOT TENSIONED. VESSEL WAS DRAINED TO APPROXIMATELY FLANGE LEVEL (~12 FT ABOVE FUEL).

- D/G 1-1 WAS OUT OF SERVICE (WOULD HAVE REQUIRED SEVERAL HOURS TO MAKE OPERABLE).

SEQUENCE: (TIMES ARE PM, EST)

- 2:41 - "SMOKE" (STEAM) OBSERVED FROM CONDUIT ON THE 1A BUS. OPERATORS IMMEDIATELY DE-ENERGIZED NON-VITAL 4160V BUS 1A. IT WAS DECIDED TO DE-ENERGIZE THE R 345 KV SWITCH YARD (S/Y) BUS. (DE-ENERGIZING ALL 3 S/U XFMRS.)

- 2:56 - OPERATORS STARTED AND LOADED D/G 1-2 ON THE 1D BUS.

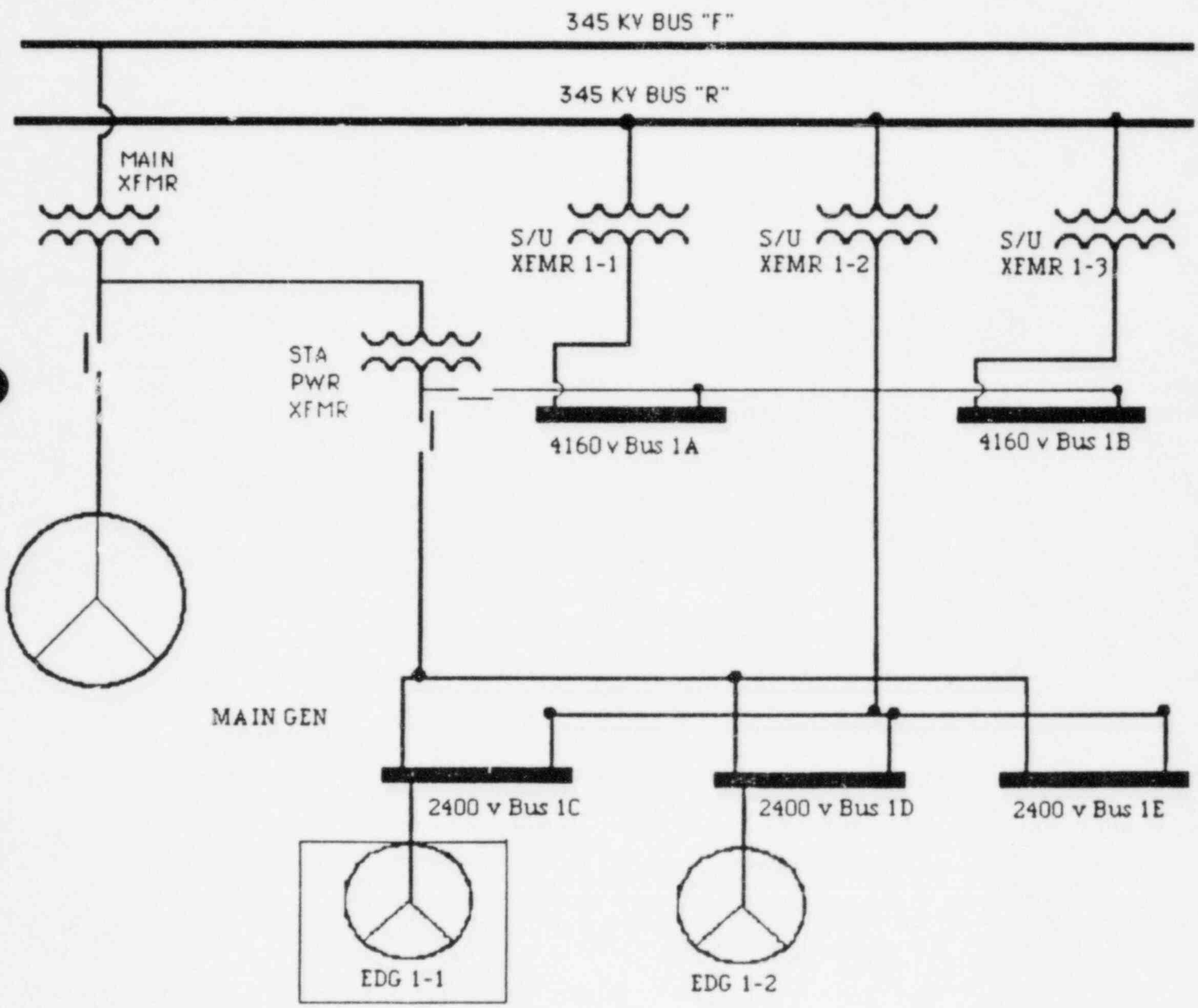
- 3:08 - R 345 KV S/Y BUS DE-ENERGIZED. 1B, (IE NON-VITAL BUSES) LOST 1C VITAL BUS LOST. 1D POWERED FROM 1-2 D/G, UNUSUAL EVENT (UE) DECL'D AT 3:00 AND NRC OP CENTER NOTIFIED AT 3:19.

- 4:50 - BACKFEED ESTABLISHED TO ALL BUSES THRU MAIN TRANSFORMER AND STATION POWER TRANSFORMER. UE TERMINATED AT 5:00.

° FOLLOW UP:
CABLE BETWEEN 4160 V BUS 1A AND STARTUP TRANSFORMER 1-2 WAS REPLACED PRIOR TO ELECTRICAL REAL!GNMENT.

//

SIMPLIFIED ELECTRICAL DISTRIBUTION DIAGRAM FOR PALISADES



H. B ROBINSON UNIT 2 - REACTOR TRIP WITH LOSS OF OFFSITE POWER
JANUARY 28, 1986 (G. REQUA, NRR)

PROBLEM

LOSS OF OFFSITE POWER (LOOP) WITH "B" EDG OUT OF SERVICE

SAFETY SIGNIFICANCE

PRECURSOR TO STATION BLACKOUT

DISCUSSION

09:17	WITH "B" EDG OUT OF SERVICE, FAULT ON EMERGENCY BUS E-2 CAUSED SPIKED ON INSTRUMENT BUS #4. FALSE ROD DROP SIGNAL CAUSED TURBINE RUNBACK. REACTOR TRIPPED ON HIGH PRESSURIZER PRESSURE.
09:18	LOSS OF OFFSITE POWER COINCIDED WITH FAST TRANSFER TO START UP TRANSFORMER. "A" EDG STARTS & LOADS ON BUS E-1. RCP's TRIPPED. NATURAL CIRCULATION ESTABLISHED.
09:35	UNUSUAL EVENT DECLARED - OPEN TELEPHONE LINE ESTABLISHED BETWEEN NRC OPERATIONS CENTER AND ROBINSON SITE
09:46	"B" DIESEL GENERATOR RESTORED
10:52	PRESSURIZER HEATERS RESTORED
12:55	E-1 BUS CONNECTED TO OFFSITE POWER
16:03	E-2 BUS CONNECTED TO OFFSITE POWER. REALIGNMENT OF OFFSITE POWER DELAYED BY 1. TWO SI SIGNALS - 1ST SIGNAL CAUSED BY CLOSURE OF MSIVs; SECOND SIGNAL CAUSED BY STUCK OPEN S/G PORV 2. ADDITIONAL BREAKER PROBLEM BETWEEN #3 BUS & E-2 3. OPERATOR DECISION TO PROCEED WITH CAUTION IN RESTORING ELECTRICAL CONFIGURATION

PLANT WAS TAKEN TO HOT SHUTDOWN. CP&L DECIDED TO SHUTDOWN FOR REFUELING ONE WEEK EARLY

FOLLOWUP

- PLANT IN 45 DAY REFUELING OUTAGE; LICENSEE INVESTIGATING CAUSE OF LOOP; REGION II WILL REVIEW PRIOR TO RESTART
- LICENSEE & INPO TEAM FORMED TO INVESTIGATE HIGH FREQUENCY OF REACTOR SCRAMS AT ROBINSON; REGION II TO EVALUATE RESULTS PRIOR TO RESTART

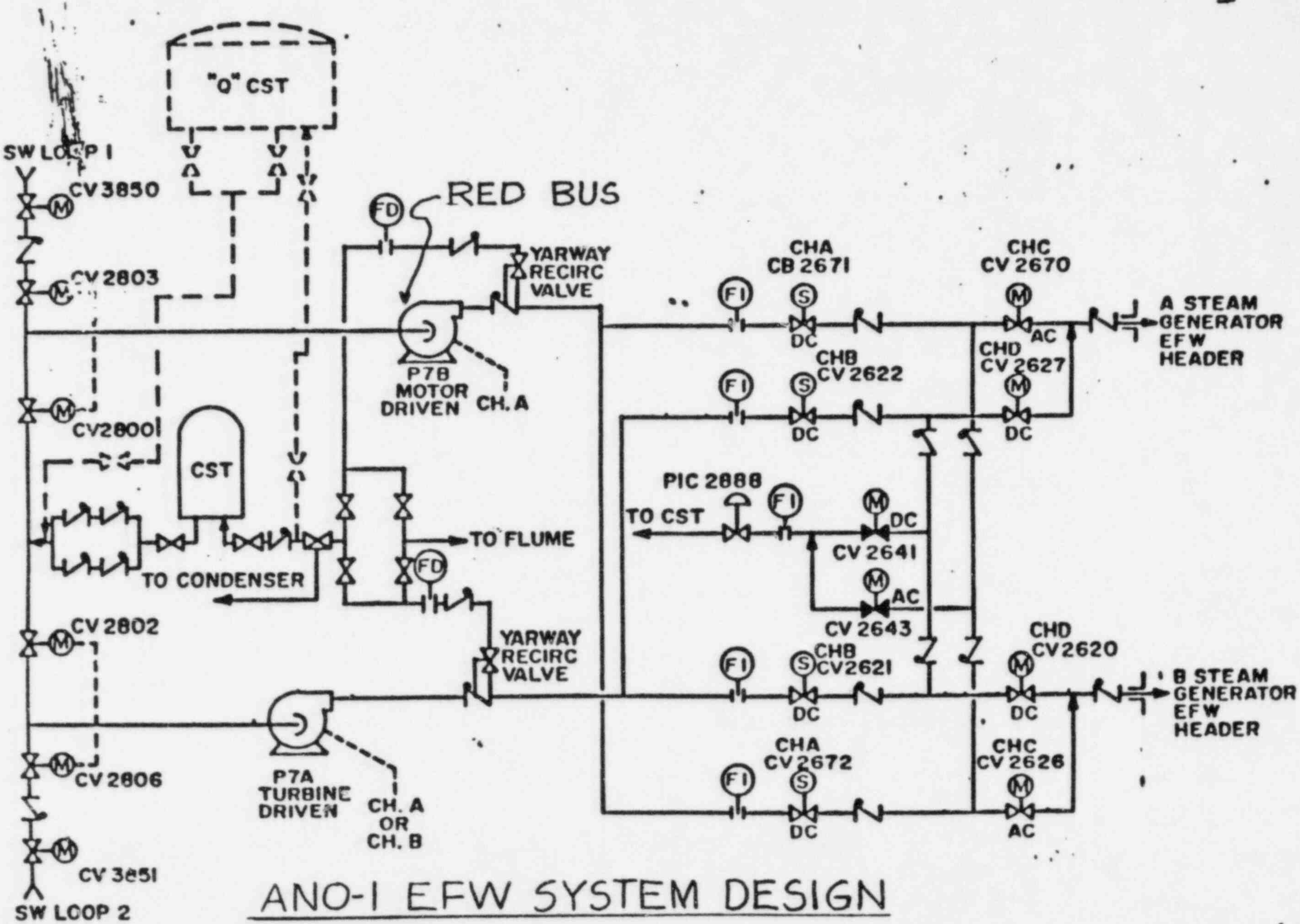
ARKANSAS NUCLEAR ONE UNIT 1 - DESIGN DEFICIENCY IN EMERGENCY FEEDWATER SYSTEM (EFW)
JANUARY 14, 1986 - (G. VISSING, NRR)

PROBLEM:
- DESIGN DEFICIENCY IN THE EMERGENCY FEEDWATER SYSTEM (EFW)

SIGNIFICANCE:
- POTENTIAL FOR LOSS OF ALL EFW AND BLOWDOWN OF BOTH STEAM GENERATORS, DURING STEAM LINE BREAK WITH SINGLE FAILURE OF ONE A/C BUS

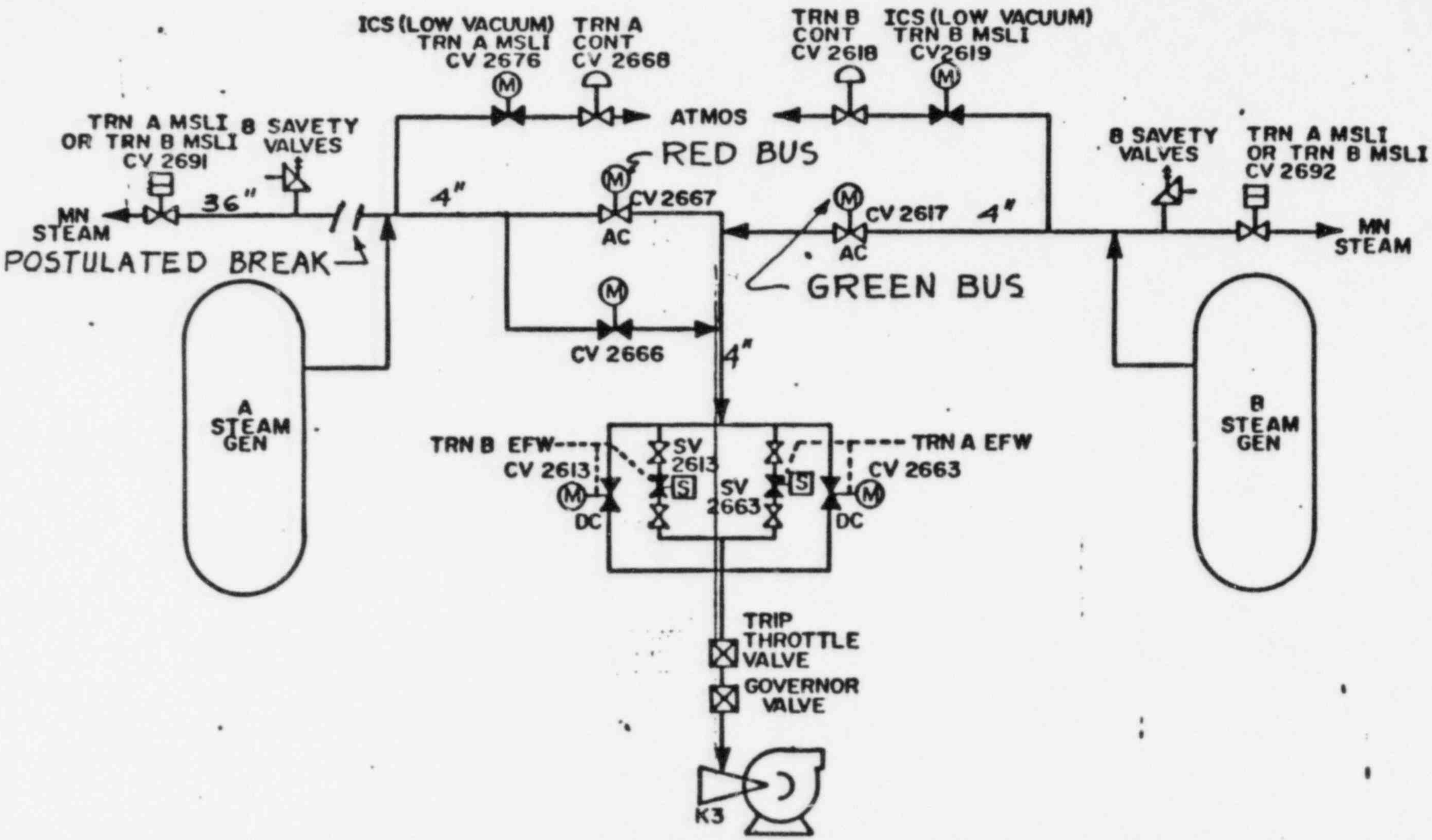
CIRCUMSTANCES:
- DEFICIENCY DISCOVERED BY IE INSPECTION TEAM
- DURING POSTULATED STEAM LINE BREAK CONCURRENT WITH LOSS OF RED A/C POWER BUS, TURBINE DRIVEN AND MOTOR DRIVEN EFW PUMPS WOULD BE LOST
- POSSIBLE BLOWDOWN OF BOTH S/Gs
- DESIGN DOES NOT MEET THE SINGLE FAILURE CRITERION
- PLANT, AT 89% POWER AT THE TIME OF THE DISCOVERY, WENT INTO CONTROLLED SHUTDOWN DUE TO FAILURE OF EDG

FOLLOWUP:
- LICENSEE CONFIRMED EXISTENCE OF DESIGN DEFICIENCY
- LICENSEE TO INSTALL CHECK VALVES IN STEAM LINES FROM EACH OTSG BEFORE STARTUP.
- NRR (PEICSB) REVIEWED ISSUE AND CONCURRED WITH INSPECTION TEAM'S FINDINGS



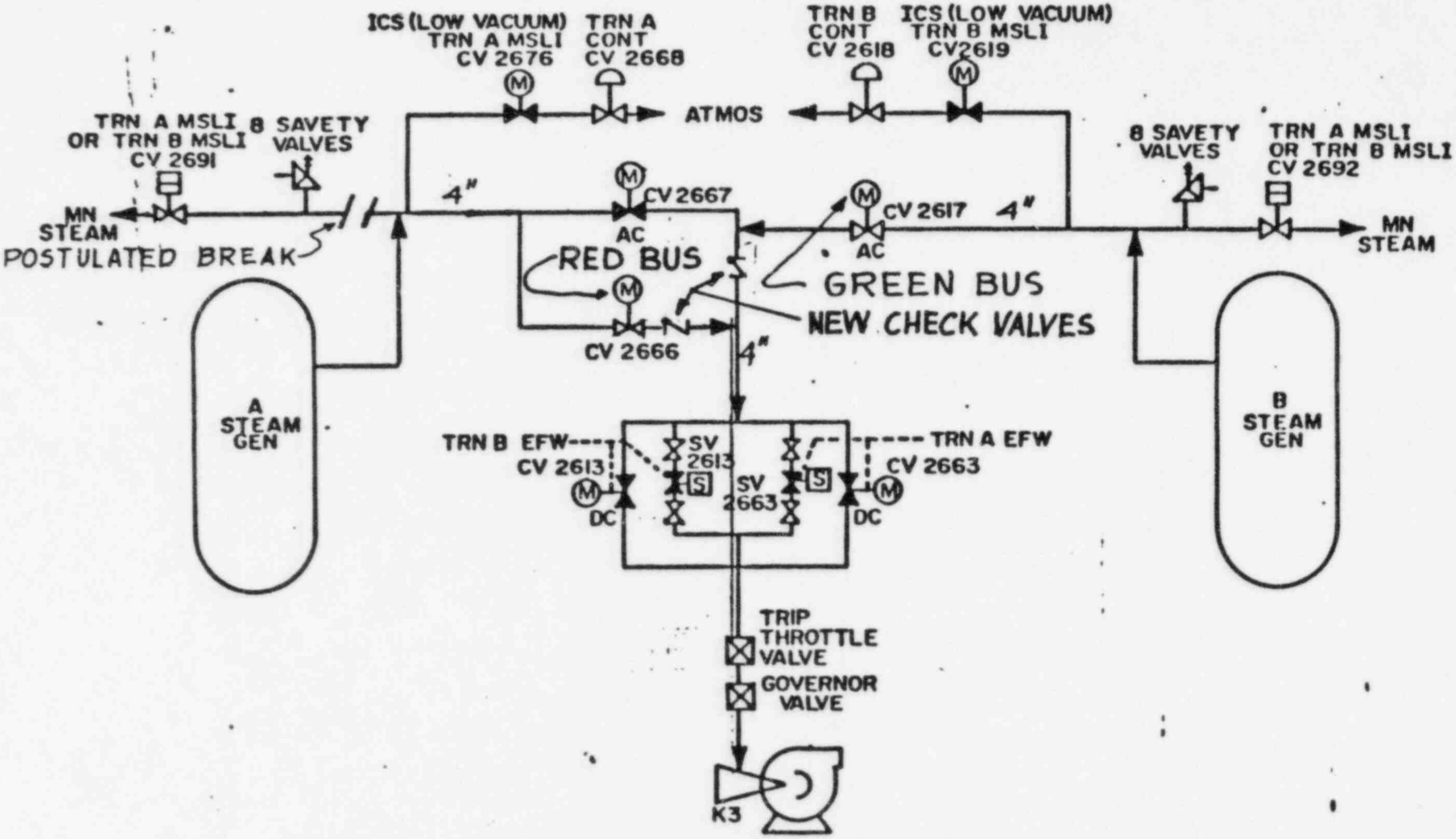
ANO-1 EFW SYSTEM DESIGN

15



ANO-1 EFW DESIGN (STEAM SYSTEM)

16



EFW DESIGN CORRECTION

17

RIVERBEND START-UP REVIEW
(Eric Weiss, IE)

CONCERN: NUMBER OF REPORTABLE EVENTS SINCE OPERATING LICENSE

CONCLUSION: NUMBER OF EVENTS SHOULD DECREASE

RIVERBEND MANAGEMENT APPEARS SOUND

SHOULD MEET IN REGION 4 TO REVIEW PROGRESS IN ONE MONTH

NRC SITE VISIT JANUARY 28 - 30, 1986

REPORTABLE EVENTS REVIEW WITH PLANT STAFF

SHIFT TURNOVER

DIESEL GENERATORS

FEEDWATER SYSTEM

FANCY POINT SUBSTATION

SURVEILLANCE PROCEDURE ON RWCUS

CONTROL ROOM PANEL WALKDOWN

URGED LICENSEE TO:

CONTINUE OPEN COMMUNICATION WITH THE NRC

RESOLVE OUTSTANDING EQUIPMENT AND HUMAN FACTORS PROBLEMS

UPGRADE COMMUNICATION WITH OTHER PLANTS

PLANT SPECIFIC PROBLEMS

TEMPORARY ALTERATION PROGRAM SHOULD BE STOPPED

ANNUNCIATOR REFLASH

FEEDWATER - VIBRATION, EROSION, STICKING FW REG VALVES

STEAM TUNNEL VENTILATION

POTENTIALLY GENERIC PROBLEMS

- VALVE OPERATOR PROBLEM
- TEMP SWITCH PROBLEMS
- FIBER OPTICS SYSTEM INTERFERENCE

EXAMPLES OF PROBLEM RESOLUTION

- CONDITION REPORT SYSTEM
- JUMPER CONTROL - TAGS, LOGS, BANANA PLUGS
- FIBER OPTICS - SHIELDING, GROUND LOOP ELIMINATION, EVENTS RECORDING, SIGNS AND NON-DUPLICATING LOCKS
- COLOR ON METERS

PERRY 1 AND 2 - EARTHQUAKE

JANUARY 31, 1986 (J. STEFANO, NRR)

PROBLEM

EARTHQUAKE MEASURING 5.0 ON RICHTER SCALE WITH EPICENTER APPROXIMATELY 10 MILES FROM SITE.

SIGNIFICANCE

- EVENT MAY BE OUTSIDE DESIGN ENVELOPE

CIRCUMSTANCES

- PLANT CONSTRUCTION COMPLETE-WAS UNDERGOING SYSTEM SURVEILLANCE/ OPERATIONAL READINESS TESTS FOR LICENSING. (NUMEROUS SAFETY SYSTEMS WERE IN OPERATION).
- SEISMIC INSTRUMENTS WERE BEING CALIBRATED AT TIME OF EVENT.
- TWO NON-SAFETY EQUIP. TRIPPED ON PROTECTIVE SIGNAL AS DESIGNED; SOME HAIRLINE CONCRETE CRACKS OBSERVED IN AUX., INTERMED. AND RADWASTE TREATMENT BLDGS. PIPE FLANGE ON WATER TANK IN RADWASTE BLDG. LEAKING.
- PRELIM. DATA SEEMS TO INDICATE SHORT DURATION (<1 SEC) SEISMIC MOTION MAINLY FALLS WITHIN OBE/SSE DESIGN SPECTRUM EXCEPT POSSIBLY AT HIGH FREQUENCY RANGE (>15Hz)

FOLLOWUP

- NRR/REG.III TEAM VISIT TO SITE TO LOOK FOR DAMAGE & OBTAIN SEISMIC DATA ON 2/1/86 - 2/2/86
- NRR/REG.III FOLLOWUP VISIT TO SITE ON 2/5/86 - 2/7/86
- CEI ANALYSIS UNDERWAY TO DETERMINE TO WHAT EXTENT PLANT STRUCTURE/COMPONENT DESIGN WAS EXCEEDED.
- PLANT LICENSING DECISION PENDING NRC STAFF EVALUATION AND DETERMINATIONS RE EVENT.

NRR TECHNICAL SPECIFICATION IMPROVEMENT PROGRAM (TSIP)

(E. Butcher, NRR, DHFT)

NEED FOR TSIP

- TOO MANY SPECS - LESS IMPORTANT ONES THUS DETRACTING FROM IMPORTANT ONES
- SIZE AND COMPLEXITY OF TS
- INDUSTRY AVAILABILITY RECORD
- NOT OPERATOR ORIENTED
- FINDINGS OF NUREG-1024 "TECHNICAL SPECIFICATIONS-ENHANCING THE SAFETY IMPACT"

PROBLEM IDENTIFICATION

- IDENTIFIED PROBLEMS BY INTERVIEWS, DOCUMENT REVIEWS, AND CONTRACTOR ASSISTANCE
- IDENTIFIED THREE PROBLEM AREAS
  - LACK OF WELL-DEFINED CRITERIA FOR TS
  - HUMAN FACTORS AND TECHNICAL WEAKNESSES
  - RELUCTANCE OF THE NRC STAFF TO USE TOOLS OTHER THAN TS

TECHNICAL SPECIFICATION IMPROVEMENT PROJECT RECOMMENDATIONS

(1) A COMMISSION POLICY STATEMENT SHOULD BE ISSUED WHICH DEFINES THE SCOPE AND PURPOSE OF TECHNICAL SPECIFICATIONS AND ENCOURAGES LICENSEES TO IMPLEMENT A PROGRAM TO UPGRADE THEIR TECHNICAL SPECIFICATIONS."

(2) THE NRC SHOULD GIVE INCREASED ATTENTION TO CHANGES MADE BY LICENSEES USING THE 10 CFR 50.59 PROCESS.

(3) THE NRC SHOULD REVIEW AND REVISE THE STANDARD TECHNICAL SPECIFICATIONS TO CORRECT HUMAN FACTORS AND OTHER TECHNICAL WEAKNESSES THROUGH A PROGRAM OF TECHNICAL ASSISTANCE AND DEDICATED IN-HOUSE TECHNICAL RESOURCES.

(4) THE NRC SHOULD ENCOURAGE THE CONTINUED DEVELOPMENT AND APPLICATION OF PROBABILISTIC RISK ASSESSMENT METHODS TO ADDRESS TECHNICAL SPECIFICATIONS REQUIREMENTS."

CRITERIA FOR TS CONTENT

- AN INSTALLED SYSTEM THAT IS USED TO DETECT, BY MONITORS IN THE CONTROL ROOM, A SIGNIFICANT ABNORMAL DEGRADATION OF THE REACTOR COOLANT PRESSURE BOUNDARY.
- A PROCESS VARIABLE THAT IS AN INITIAL CONDITION OF A DBA ANALYSIS.
- A STRUCTURE, SYSTEM, OR COMPONENT THAT IS PART OF THE PRIMARY SUCCESS PATH OF A SAFETY SEQUENCE ANALYSIS AND FUNCTIONS OR ACTUATES TO MITIGATE A DESIGN BASIS ACCIDENT.

ONGOING ACTIVITIES

- TRIAL USE OF TSIP CRITERIA

- MEETINGS WITH INDUSTRY OWNERS GROUPS AND AIF

- SHORT TERM IMPROVEMENTS TO EXISTING STS

  - FIRE PROTECTION TECHNICAL SPECIFICATION

  - ACTION STATEMENTS FOR MISSED SURVEILLANCE TESTS

  - BWR RPS SURVEILLANCE INTERVALS AND AOTs (NEDC-30851P)

  - BWR ECCS INSTRUMENTATION SURVEILLANCE INTERVALS AND AOTs (NEDC-30936P)

- EVALUATION OF COMMENTS ON TSIP REPORT

FUTURE ACTIVITIES

- DETAILED IMPLEMENTATION PROGRAM PLAN - 03/86

- PROPOSED COMMISSION POLICY STATEMENT - 06/86

- ULTIMATE LONG TERM OBJECTIVE

  "A COMPLETE REWRITE/STREAMLINING" OF THE EXISTING STS BASED ON THE RECOMMENDATIONS OF THE TSIP REPORT

BRIEFING FOR ACRS

Fermi 2

Wednesday February 12, 1986

Region III

FERMI HISTORY

Significant Events	Date
Low Power License	March 20, 1985
Premature Criticality Event	July 1, 1985
Commission Briefing	July 10, 1985
Full Power License	July 15, 1985
Unit Shutdown for Planned Outage	October 11, 1985
Projected Availability for Startup	March, 1986**

** This may be delayed, dependent upon resolution of Stress Report and Hanger Design Calculations. (PNO-III-86-011)

TECHNICAL ISSUES

- o Diesel Generators
- o Turbine Bypass Lines
- o South Reactor Feed Pump Turbine
- o TIP Purge Line
- o Condensate Storage Tank
- o Remote Shutdown (3-L) Panel
- o Environmental Qualifications/Review
- o Seismic Review
- o Concrete Embedments
- o RHR Pump
- o Outdated Stress Reports and Hanger Calculations

PERFORMANCE ISSUES

- o Excessive Personnel Errors
- o Security Program Implementation
- o Failure to Disposition known Problems
- o Ineffective Communications

Fermi 2

PERSONNEL ERRORS

A Review of 78 LERs

Covered During	No. of LERs	No. of LERs Caused by Personnel Errors
03/20/85-11/25/85	78	41
07/10/85-09/10/85	25	9
09/10/85-11/25/85	17	8

10 CFR 50.54(f) LETTER

- Adequacy of Management and Management Structures and Systems
- Actions to Ensure Readiness to Restart and Power Ascension
- Actions to Improve Regulatory and Operational Performance

LICENSEE RESPONSE

- o CEO Assumes Direct Management Control
- o Independent Overview Committee
- o Reactor Operations Improvement Plan
- o Nuclear Operations Improvement Plan
- o CEO to Authorize Power Escalation
- o Single A/E On Site
- o Independent Overview by Corporate Officers
- o Plan to Hire A Senior VP

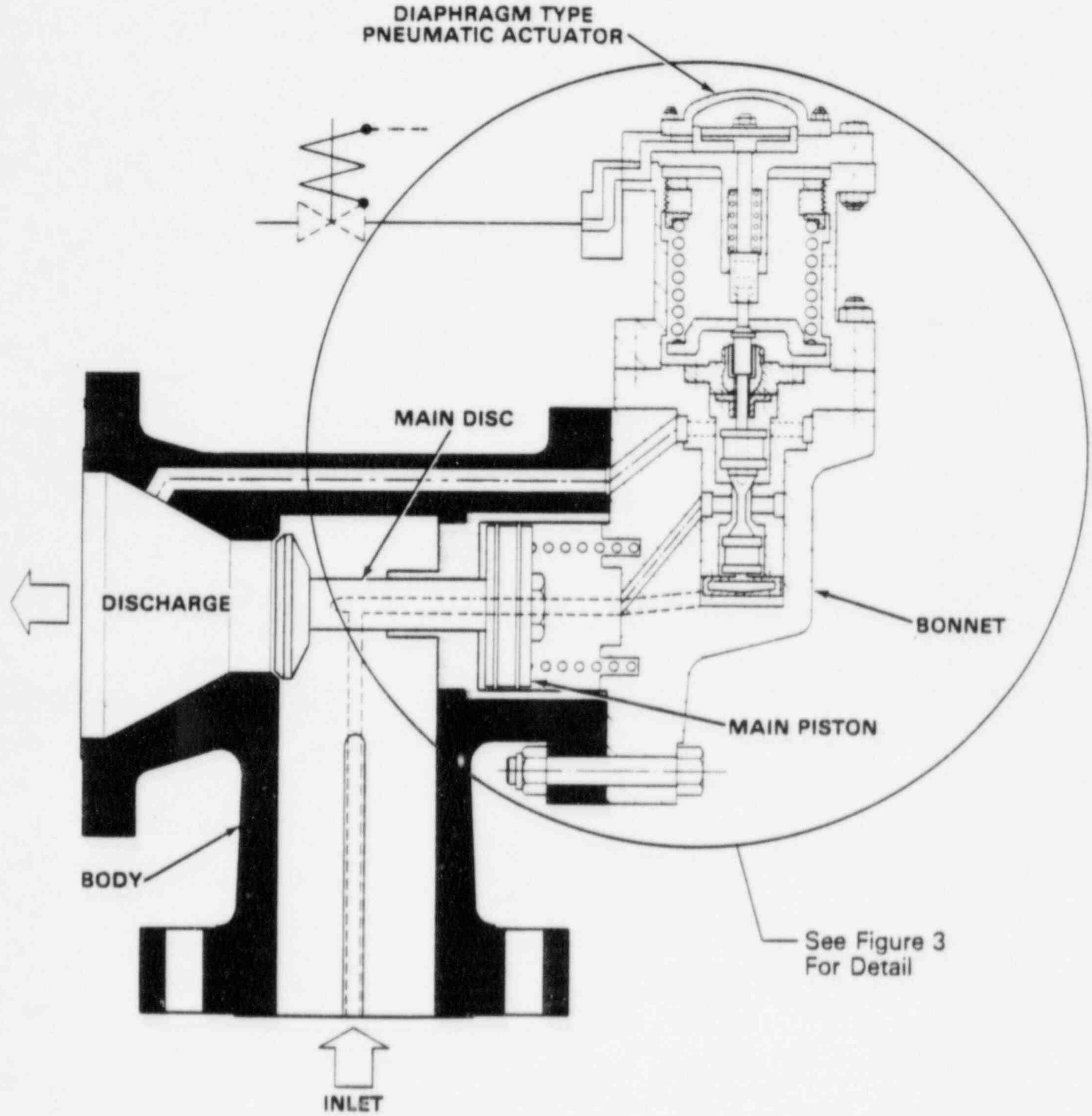


Figure 2 Target Rock Two-Stage Pilot Actuated, Safety/Relief Valve

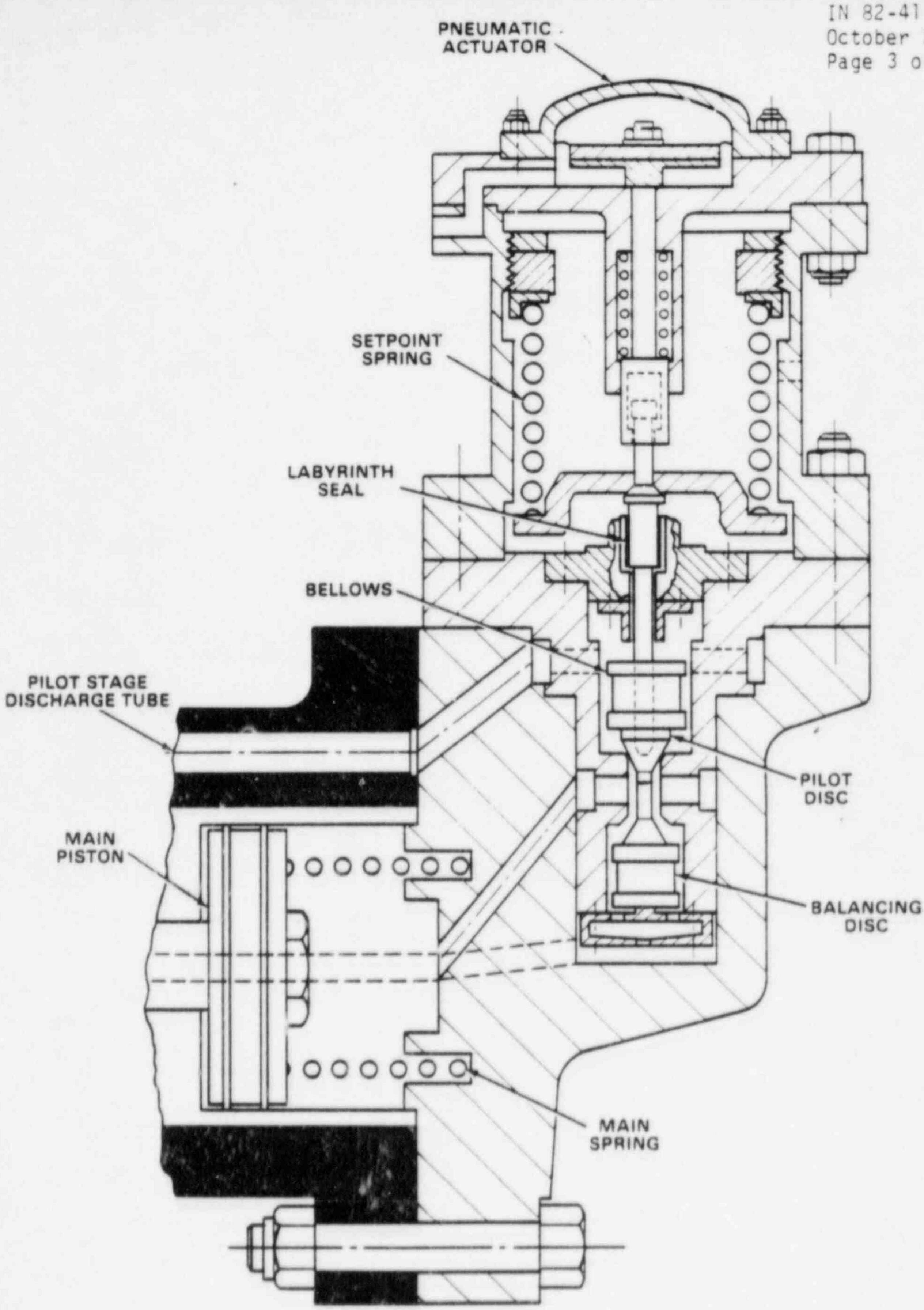
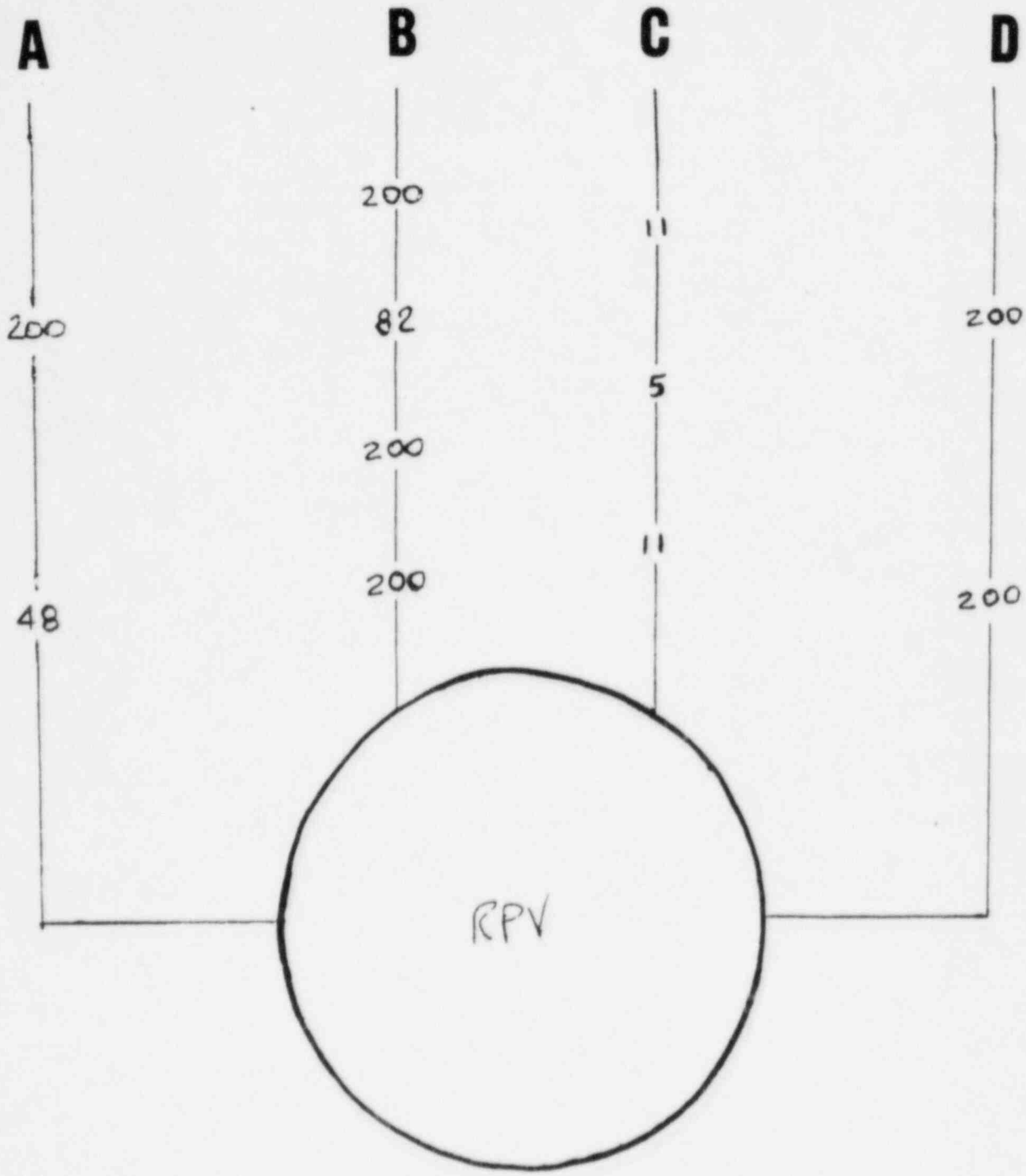
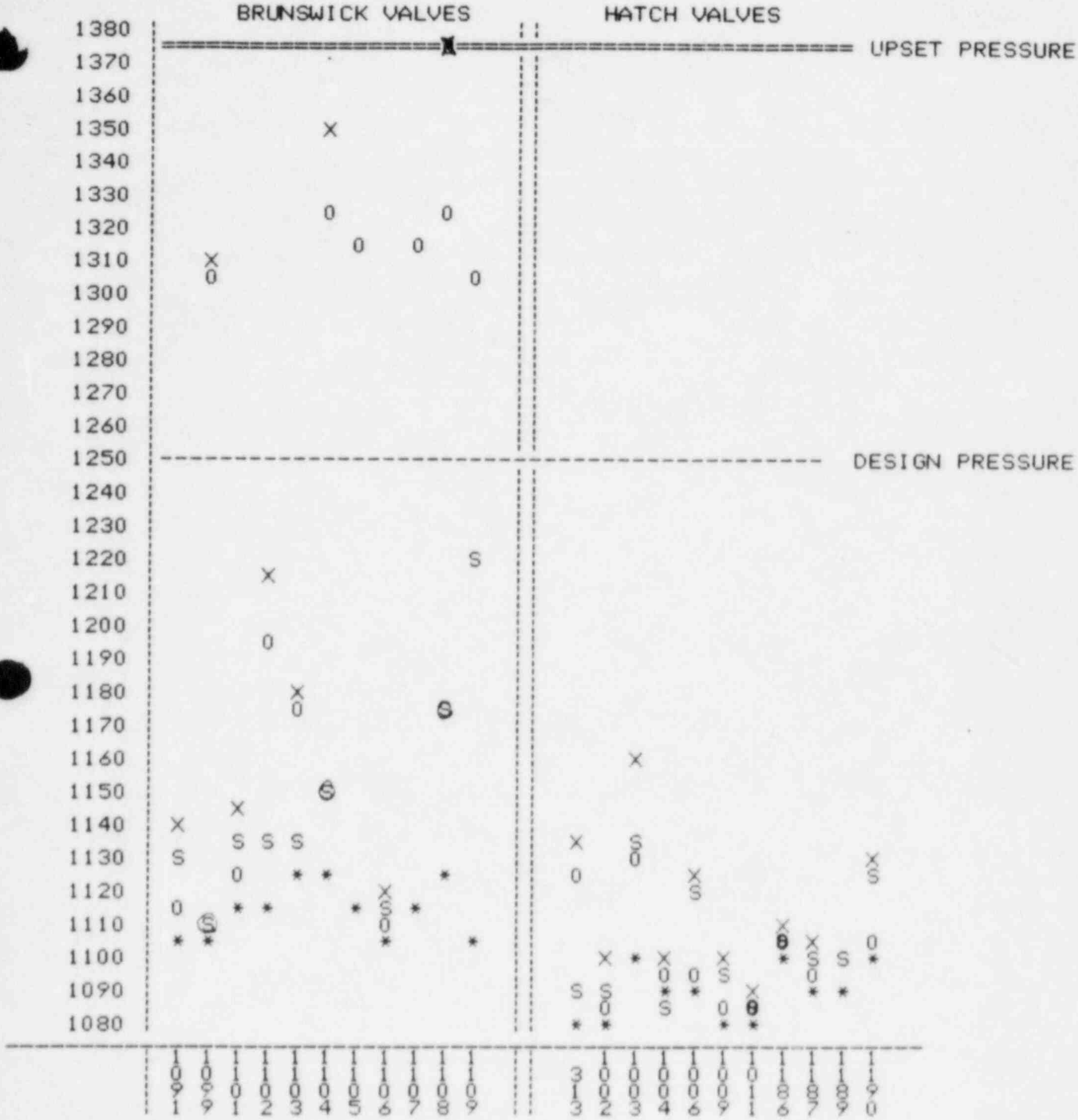


Figure 3 Topworks of Target Rock Two-Stage Safety/Relief Valve



LOCATION OF BSEP 2 SRV BY N_2 PRESSURE

SRV TEST RESULTS

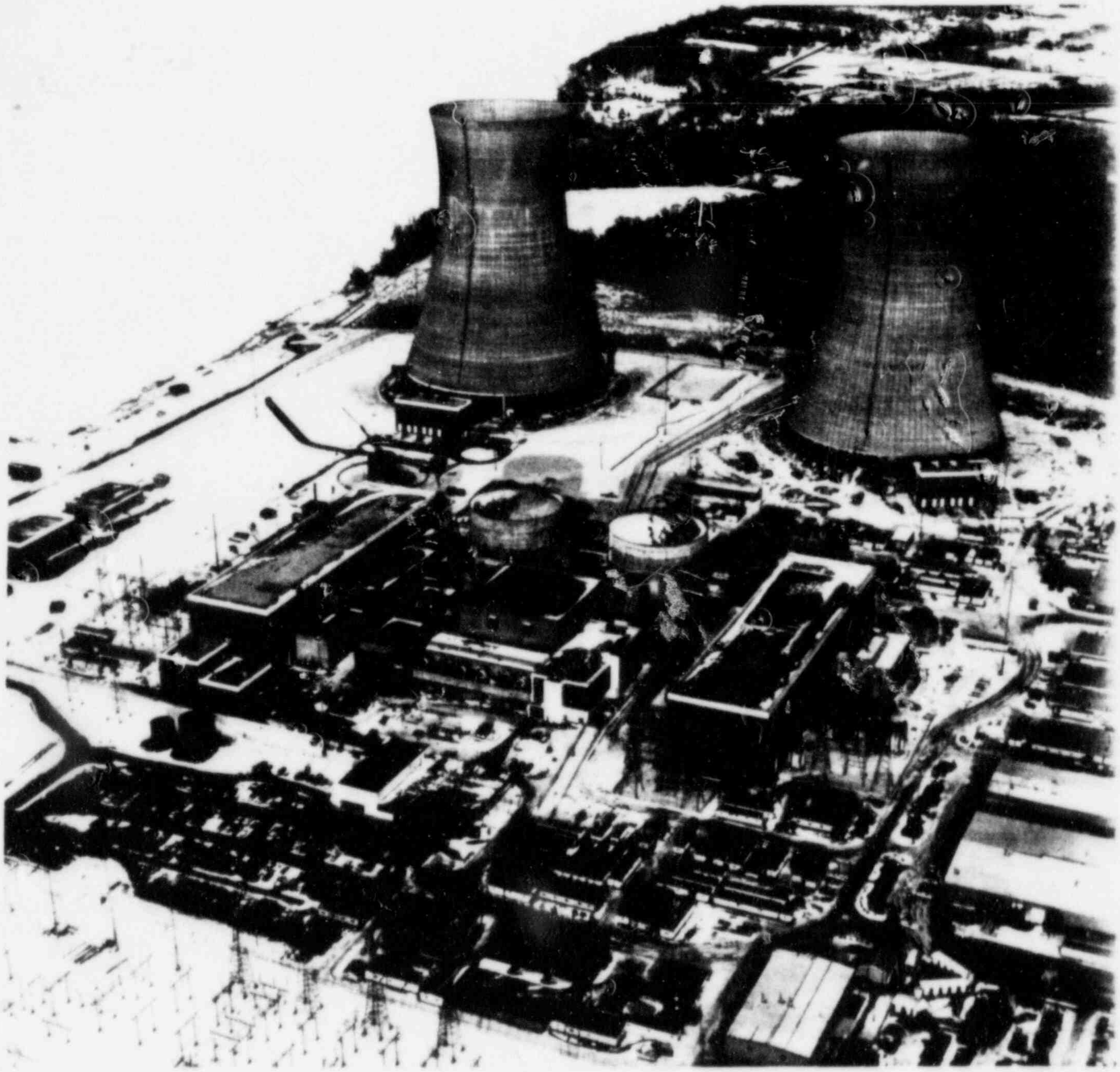


KEY: * - SETPOINT

0 - SP + N2 PRESS

S - STEAM PRESS

X - N2 + STEAM PRESS



PERRY POWER PLANT

JANUARY 31, 1986 EARTHQUAKE
SEISMIC EVENT EVALUATION

SEISMIC EVENT EVALUATION

PEPORT

PERRY NUCLEAR POWER PLANT

DOCKET NOS. 50-440; 50-441

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

FEBRUARY 1986

TABLE OF CONTENTS

1.0 INTRODUCTION

The purpose and scope of this report is to provide the results of The Cleveland Electric Illuminating Company seismic event evaluation for the Perry Nuclear Power Plant. The discussions contained herein provide the basis for CEI's conclusions that the January 31, 1986 earthquake in the vicinity of the Perry site:

1) did not adversely effect the plant structures, systems or components,
2) was within the design capability of the Perry Nuclear Power Plant, and
3) does not change the licensing basis or conclusions regarding the site geology, seismology or design basis earthquake.

This evaluation report addresses the key issues related to the January 31 earthquake including the immediate response to the event, and the plant status and impact assessments following the earthquake. Detailed evaluations of the geological and seismological implications of this event and an analysis of the plant seismic design basis capabilities are presented. In addition, a description is provided of the confirmatory programs to monitor post seismic event activity, to continue the evaluations to identify any earthquake related effects, and to participate in generic industry studies.

2.0 SEISMIC EVENT OVERVIEW

Event

At approximately 11:48 a.m. on January 31, 1986, an earthquake occurred, which was located about 10 miles south of the Perry site and had a Richter magnitude of approximately 5.0. CEI implemented the Perry emergency plan in response to the seismic event as described in the attached chronology. A site area emergency was declared as a precautionary measure for site personnel accountability and for informational notification to local officials. Timely notifications were made and plant staff responded professionally and successfully implemented the plant procedures for this type of an event.

Plant Response and Assessments

Immediately following the earthquake, plant operations personnel were dispatched into the plant to survey for any major damage. The initial reports indicated no damage. Subsequently, a team of approximately 65 engineers and technicians was organized to perform a detailed walkdown of all plant areas. These inspections found no damage to any systems, structures or components. The hairline cracks in concrete walls that were observed have been reviewed and found to be typical of reinforced concrete structures whi h have not experienced seismic events. Numerous safety-related systems in operation or standby readiness continued to operate without incident.

Earthquake Analysis

Based on United States Geological Survey (USGS) recorded data, the earthquake of January 31, 1986 was centered about 10 miles south of the Perry Site and had a Richter magnitude of 4.96. This is a lesser magnitude than the earthquakes for which the Perry Plant has been analyzed and had substantially lower total energy content than the

Perry design response spectra. The January 31 earthquake is consistent with the previously established geology and historical seismicity of the region, as described in the Final Safety Analysis Report. The earthquake does not change the conclusions of the FSAR on the geology and seismicity of the site area.

Seismic Design Evaluation

Acceleration data taken from the in-plant seismic recorders showed recorded floor response spectra in certain locations outside the design spectra at high frequencies. The design spectra are based on a statistical envelope of historical earthquakes (84th percentile) and, therefore, some instances of recorded responses exceeding predicted floor responses are expected. The possibility of high frequencies outside the spectra has been evaluated at other nuclear plant sites and concluded to have insignificant effect on plant structure and components.

CEI analysis shows the high frequency accelerations involved are of a very short duration and the velocities are well below those which could cause damage even to non-engineered structures. The total energy associated with these high frequency accelerations is small, and therefore has no adverse impact on plant structures and equipment. Thus, the high frequency accelerations have no engineering significance and the effects of the earthquake experienced at Perry are well within the seismic capability of the plant.

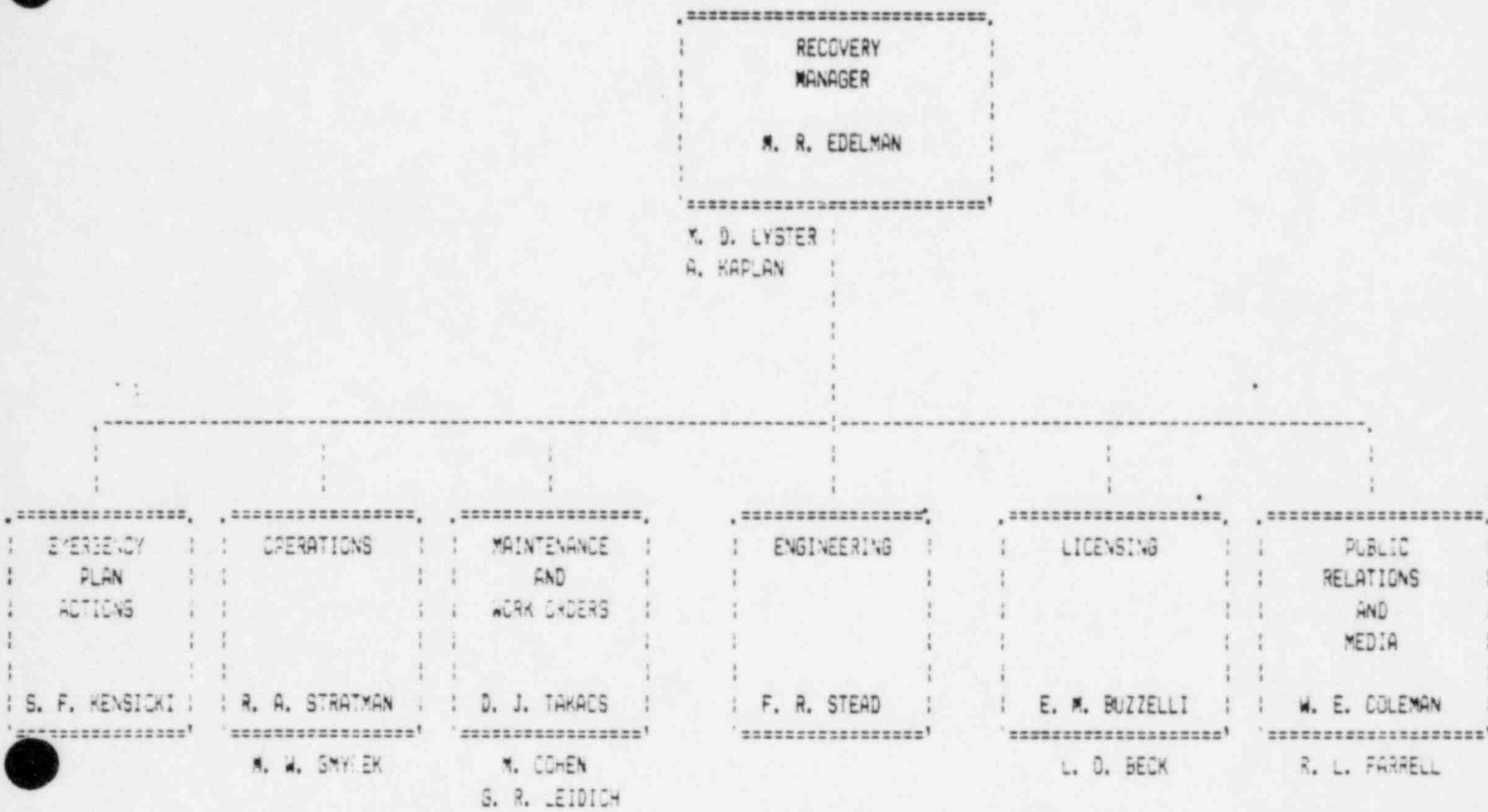
January 31, 1986 Earthquake

Chronological Summary of Events

Time of Occurrence	Event
11.46:42.3 (USGS data)	Seismic event occurs
1148	Control room reports nosie & vibration to to Systems Operation Center
1150	Main generator breaker reported open, isolating main and auxiliary transformer, automatically shifting to startup transformer. Auxiliary boiler trips noted Seismic alarms received in P680
1155	Trip of instrument air compressor noted
1200	Visual inspection of lower areas of Turbine Building, Auxiliary Building, Intermediate Building and transformer yard satisfactory
1201	Shift Supervisor sounds Plant Emergency Alarm
1204	Visual inspection of Turbine Building, Turbine Power Complex, Intermediate Building, Auxiliary Building and Control Complex satisfactory.
1206	Shift Supervisor delcares precautionary Site Area Emergency, makes Evacuation Announcement.
1211	Auxiliary boiler restarted
1216	Notifications to CEI emergency personnel pursuant to Emergency Plan began
1218	Initiated retrieval of seismic plates and magnetic tapes from seismic instrumentation
1219	Visual inspection of service water and emergency service water pump house satisfactory
1225-1240	Initial notifications of Site Area Emergency provided to Lake, Geauga and Ashtabula counties, the State of Ohio, Coast Guard, NRC
1230	Visual inspection of cooling towers and basins satisfactory
1232	Operational Support Center (OSC) activated

Time of Occurrence	Event
1235	Technical Support Center (TSC) activated
1251	Initial inspections of all Unit 1 and Common areas completed with satisfacory results, only minor problems noted.
1254	Visual inspection of suppression pool satisfactory
1257 - 1301	TSC completes precautionary Site Area Emergency Follow-up Notifications to Counties, State, Coast Guard
1300	Walkdown of all Unit 1 areas has finds no major equipment damage and noted minor flange leaks.
1302	Site Area Emergency downgraded to Alert
1303 - 1315	Initial Notification of downgrading to Alert made to Counties, State, Coast Guard and NRC.
1305	Three teams dispatched for additional system walkdowns; six maintenance teams dispatched to investigate equipment
1340 - 1401	Follow-up notification of Alert status provided to Counties, State, Coast Guard and NRC
1341	INPO contacted
1420	NRC - Bethesda and Region III concur on termination of emergency
1425	Termination of Emergency Event
1431 - 1442	Termination of Emergency Event reported to Counties, State, Coast Guard, & NRC.
1440	INPO notified of termination of Emergency Event
1531	Deactivated TSC.
1552	Seismic alarm P969 reset
1630	Recovery organization met to review seismic event, emergency response and confirmatory actions.

RECOVERY ORGANIZATION



3.0 PLANT STATUS AND IMPACT ASSESSMENTS

3.1 PLANT STATUS

Prior to the earthquake that occurred on January 31, 1986, numerous testing, calibration, and work completion activites were being conducted in preparation for fuel load. One major activity was preparation for the Division II Diesel Generator response time testing. As part of this work, all of the safety related components powered from the Division II Diesel were energized and in standby readiness. All of this equipment behaved normally through the event; that is, there were no spurious starts or alarms. Preparations were also underway to move the startup sources. This work had not yet begun when the seismic event occurred. The sources were never actually moved, and remained stored in the upper pools.

In support of the ongoing test and surveillance activities, a significant number of systems were in operation. In addition, numerous other systems were energized and in the standby mode. Lists of the specific safety and non-safety systems energized or operating prior to and during the earthquake are included as Tables 3.1 and 3.2. All of the operating safety-related systems continued to operate through the event. None of the safety-related systems in the standby mode experienced any spurious initiations.

As noted in Table 3.2, a large number of non-safety systems were operating or in the standby mode, and maintained their status throughout the event. Two non-safety items tripped on protective signals as intended by the design. These were the Unit 1 instrument air compressor, which tripped on high vibration, and the auxiliary steam boiler, which tripped due to actuation of one of its protective

circuits. The instrument air compressor is a centrifugal machine that operates at greater than 40,000 rpm and as part of its protective devices has a very sensitive vibration switch. The auxiliary steam boiler has several protective circuits of which one tripped during the earthquake. The boiler was successfully restarted after the event.

The only other non-safety items of equipment that tripped during the earthquake were the Unit 1 main and auxiliary transformers, which tripped due to the closing of the generator protection relays. These relays although open at the time of the seismic event, did not have voltage applied as a result of an ongoing outage. Laboratory testing of these relays since the event has confirmed that the presence of voltage on the relays significantly increases the force required to close these relays. Had the voltage been supplied to these relays, they would not have closed during the event. This is substantiated by the fact that other similar open relays with voltage applied did not close during the event.

Investigation is ongoing to determine the cause of an indicated 1 1/2 inch increase in suppression pool level. No basis for a physical change in the water level has been identified. The water level transmitters were found to be out of calibration, though not enough to account for the entire indicated level increase. The same transmitters in other applications did not show any anomalous behavior.

In addition to the emergency plan actions previously discussed, immediately following the event the plant operators performed initial surveys of the plant. Areas visually inspected included the Transformer Yard, lower elevations of the Turbine, Auxiliary, Intermediate and Radwaste Buildings, as well as the Control Complex, Turbine Power Complex, Heater Bay and Water Treatment Building. The reports back to the Control Room indicated that the areas were found in satisfactory condition with no major damage. In addition, the General Supervisor of Operations and the Senior Operations

Coordinator made a specific survey of below grade areas. They found no unusual or abnormal conditions. Further steps taken to assess and evaluate the status of the plant included additional walkdowns by teams of plant maintenance personnel dispatched from the Operations Support Center.

3.3 PLANT IMPACT ASSESSMENT

As part of CEI's response to the earthquake, a team of approximately 65 engineers and technicians was organized on the evening of January 31 to perform systematic and thorough walkdowns of all plant areas. These walkdowns were performed using drawings of each area and checklists of components to inspect for any abnormal conditions. These included such items as piping, hangers, snubbers, valves, pumps, instrumentation and other components. The results of these walkdowns were recorded and compiled into a list of approximately 480 observations, many of which were later determined to be preexisting conditions. None of the observations involved structural damage to the plant or equipment. The 480 observations are typified by minor hairline cracks in concrete, burned out light bulbs and leaking valve or piping flanges, all of which are normal and expected conditions that would be identified in any comprehensive walkdown.

In the inspections that were conducted following the earthquake, plant personnel were instructed to document all unusual or abnormal conditions. Those conducting the inspections did not attempt to determine whether the conditions were the result of the earthquake; instead, discrepant conditions regardless of potential cause were documented to insure that the status of the plant following the earthquake would be fully documented for subsequent evaluation by engineering. Each of the observed discrepant conditions was subsequently evaluated by engineeering to determine whether the condition was caused by the earthquake and whether rework or repair was required. The engineering evaluation of the items concluded that 77% were preexisting conditions, and only two minor items, were directly attributable to the earthquake. The remainder,

approximately 100 items, have been classified as indeterminate, i.e., it could not be definitively established that the condition existed prior to the earthquake. About 25% of the approximately 480 items will need rework or repair. (See Appendix E). These will be processed in accordance with a special procedure instituted in response to the earthquake.

A number of other inspections were also performed to determine the effect, if any, on specific plant structures and conditions. A site survey was performed to assess any impact of the earthquake on the site environs, and in particular on the shoreline bluff. No evidence of any earthquake impacts could be found.

A survey of settlement monitoring points was ordered to determine if the earthquake had any effect on building settlement. Monitoring points at various locations around the perimeter of the plant buildings are surveyed on a monthly basis to monitor building settlement. The results of the surveys were that the recorded movements were consistent with those measured in the past, including the amount of change from prior surveys and the absolute elevations. For example, a comparison of the Reactor Building reading with that of February 1985, found that the two readings were identical. Thus it is concluded that the earthquake had no impact on building settlement. (See Appendix E).

A walkdown of Unit 1 Cooling Tower was performed to determine whether any damage had resulted from the earthquake. The areas inspected included the basin walls, tower columns and footers, internal support columns, baffle system, discharge pipe, and veil. While all inspections were done from ground level, any significant cracks in

the veil would have been readily apparent since they would have been saturated by the previous day's rain. No structural damage was found in any area of the cooling tower. Water was observed seeping through the north and south vertical joints where the basin plume wall and pump house flume wall meet. Seepage at this joint has been noted in the past and stopped by the application of mastic material. (See Appendix E).

As part of the design program for the plant, seismic clearance criteria were established to assure that a seismic event would not cause any impact on a safety system either by causing swaying or by impact from a non-safety item. Instances of these criteria not being met are termed Seismic Clearance Violations (SCV's). SCV's are forwarded to engineering for evalution to determine whether repair is required. At the time of the earthquake, there were 29 SCV's that had been dispositioned for repair, where the repair had not yet been completed. Following the earthquake, inspectors were directed to reinspect these SCV's to determine whether the seismic event affected the SCV condition. These inspections found neither damage nor dimensional change. (See Appendix E).

As previously noted, the plant systems, both safety related and non-safety related, operated properly during and following the seismic event. Recognizing the sensitivity of electrical components to high frequency response, a detailed engineering study was undertaken to identify the number and types of electrical equipment that was energized during the earthquake. The components included motors, transformers, relays, switchgear breakers, switches, batteries, contacts, valve operators, chargers/inverters, meters, recorders, and transmitters. A wide variety of suppliers was represented. More than 70 separate systems were involved. The study showed that over 47,000 electrical components were energized and experienced no adverse effects in terms of spurious system actuation (See Appendix E).

TABLE 3.1

SAFETY RELATED SYSTEMS ENERGIZED DURING THE SEISMIC EVENT OF JANUARY 31, 1986

SYSTEM	DESCRIPTION
C11	Control Rod Drive
C41	Standby Liquid Control
C71	Reactor Protection System
D17	Plant Radiation Monitors
E12	Residual Heat Removal
E21	Low Pressure Core Spray
E22	High Pressure Core Sray
G41	Fuel Pool Cooling and Cleanup
M15	Annulus Exhaust Gas Treatment
M23	MCC, Switchgear, & Misc. Area HVAC
M24	Battery Room Exhaust
M25	Control Room HVAC
M26	Control Room Emergency Recirculation
M32	ESW Pumphouse Ventilation
M40	Fuel Handling Building Ventilation
M43	Diesel Building Ventilation
P11	Condensate Transfer and Storage
P22	Mixed Bed Demineralizer
P41	Service Water
P42	Emergency Closed Cooling
P43	Nuclear Closed Cooling
P45	Emergency Service Water
P47	Control Complex Chill Water
P49	ESW Screen Wash
P52	Instrument Air
P54	Fire Protection
C95	Emergency Response Information System
P51	Service Air
R14	110 VAC Vital Inverters
R22	Metalclad Switchgear
R23	480 V Load Centers
R24	Motor Control Centers
R25	Distribution Panels - 120, 208 & 480 volts
R42	D. C. System
R43	Standby Diesel Generator (SDG)
R45	SDG Fuel Oil
R46	SDG Jacket Water Coolant
R47	SDG Lube Oil
R61	Main Control Room Annunciator

TABLE 3.2

NON-SAFETY RELATED SYSTEMS
ENERGIZED DURING THE SEISMIC EVENT
OF JANUARY 31, 1986

SYSTEM	DESCRIPTION
F42	Fuel Transfer Equipment
G33	Reactor Water Cleanup
M11	Containment Vessel Cooling
M13	Drywell Cooling
M21	Controlled Access HVAC
M27	Computer Room HVAC
M35	Turbine Building Cooling & Ventilation
M36	Off-Gas Building Exhaust
M41	Heater Bay Ventilation
M45	Circulating Water Pump House Ventilation
N21	Condensate
N23	Condensate Filtration
N24	Condensate Demineralizers
N32	Turbine Control (EHC)
N71	Circulating Water
P20	Water Treatment
P21	Two Bed Demineralizer
P44	Turbine Building Closed Cooling
P55	Building Heating
P61	Auxiliary Steam
P62	Auxiliary Boiler Fuel Oil
P72	Plant Underdrain
C91	Process Computer
C94	Health Physic Computer
P56	Security
R11	Station Transformers
R15	Technical Support Center UPS
R36	Heat Tracing & Anti Freeze Protection
R44	SDG Starting Air
R51	Intra Plant Communications
R52	Maintenance & Calibration
R53	Exclusion Area Paging System
R57	Radio & In-Plant Antenna System
R71	Lighting
S11	Power Transformers
S41	Step Up Station

4.0 EARTHQUAKE ANALYSIS AND SITE SEISMICITY

An earthquake of magnitude 4.96 M_{blg} occurred on January 31, 1986 at 11 hours, 46 minutes, and 42.3 seconds approximately 11 miles (17.7 kilometers) south of the plant. The depth of the earthquake is presently calculated to be 6 miles (10 kilometers) deep and is located at 41.640° W and 81.098° N by the National Earthquake Information Center of the United States Geological Survey (USGS). This location is near the intersection of Highways 86 and 166 in Thompson Township, Geauga County. The location of this earthquake is shown on Figure 4.1 of this report. Earthquakes which have occurred within 200 miles in historical times, and an update for those occurring within 50 miles of the plant site are shown in Figures 4.2. and 4.3.

4.1 BACKGROUND GEOLOGICAL & SEISMOLOGICAL STUDIES RELATED TO THE PERRY NUCLEAR POWER PLANT

As required by the regulations governing the siting of nuclear power plants, a thorough study of the geological and seismological characteristics of the Perry Nuclear Power Plant site and its regional surroundings was made as part of both the Preliminary Safety Analysis Report (PSAR) and Final Safety Analysis Report (FSAR). The purpose of these investigations was to assure that the site was geologically suitable for the construction of a nuclear power plant and to provide a basis for the determination of a Safe Shutdown Earthquake (SSE) and the site ground motion resulting from the occurrence of such an earthquake. The information contained herein is summarized from the detailed discussions contained in Chapter 2 of the PSAR and FSAR, as reviewed and accepted by the NRC in the Safety Evaluation Reports and Supplements.

These studies were extensive, consisting of a compilation and analyses of published and unpublished literature; field geological checking and mapping including wide scale and local geophysical studies to characterize geological conditions at depth; borings; laboratory analyses; and detailed engineering analysis of the site foundation materials.

Based on these studies and following Appendix A of 10 CFR Part 100, a correlation of earthquakes to a particular fault or series of faults which would be designated as "capable" could not be made. In addition, no "large scale dislocation or distortion" of the earth's crust designated as a tectonic structure could be identified to which earthquakes could be correlated. Consequently, earthquakes were identified with a "tectonic province", representative of a region within which there is a relative consistency of geologic structural features.

To select the SSE, a Modified Mercalli Intensity of VII was chosen as the maximum intensity earthquake at the Perry site. This intensity corresponds to an acceleration value of 0.15g, based upon a number of developed relationships which relate peak accelaration to earthquake intensity values; the principal relationship was developed by Trifunac and Brady. (Trifunac, M.D. and Brady, A.G., 1975, on the Correlation of Seismic Intensity Scales with the Peaks of Recorded Strong Ground Motion: Bulletin of the Seismological Society of America, v. 65, No. 1, pp. 139-162). The response spectra representing the SSE were then developed by adopting a NRC Regulatory Guide 1.60 response spectral shape. The design response spectra are shown on Figures 4.4 and 4.5.

During the review of the FSAR, the NRC staff requested that site-specific spectra be constructed for the Perry site. In response to this request, site-specific response spectra were constructed using a set of ground motion accelerograms from actual earthquakes of magnitude range 5.3 $\pm$ .5 recorded on rock (to simulate the foundation conditions at Perry) at epicentral distances of 0 to 25 kilometers; this represents the earthquake "at the site" as required by Appendix A and is shown on Figure 4.6.

Eleven (11) earthquakes representing 22 components of motion were chosen. A subset of records accepted by the staff as representative of an Anna, Ohio type earthquake had an average magnitude of 5.53 $\pm$ .3 at an average distance of 8.5 miles (13.66 $\pm$ 4.5 kilometers). A smoothed 84 percentile of this data set fell below the design

response spectra represented by a Regulatory Guide 1.60 spectra set at an acceleration of 0.15 g. These spectra are representative of free field data recorded at locations away from the influence of buildings and structures, and are shown on Figure 4.7.

4.2 REGIONAL GEOLOGY AND TECTONICS

The Perry site is located in the central part of Eastern Stable Platform Tectonic Province, characterized by an upper Precambrian crystalline basement and overlain unconformably by a sequence of Paleozoic sedimentary rocks. Basement rocks of this tectonic province comprise a complex sequence of high grade metamorphics and include: schists, gneisses, marbles, and granulites consolidated during the Grenville Orogeny (950 mya) onto the North American craton.

The basement rocks are overlain by a 5000' thick sequence of sedimentary rocks, Cambrian to Carboniferous in age, which dips less than 5^{o} to the south. (Fig. 4.8). Sedimentary rocks within this sequence of Paleozoic sediments includes shales, salt, sandstone, dolomites, and limestones. In the epicentral region the sedimentary sequence is approximately 2 kilometers thick with the main shock focus well within the crystalline basement.

A thin veneer, generally less than 100' of variable thick Pleistocene deposits, lies unconformably on the sedimentary sequence. These deposits include a lower till, dense and compact (approximately 30' thick) overlain by less compact till, lacustrine deposits and beach deposits.

Post consolidation tectonic deformation in the province includes the following structual elements. Paleozoic structures include broad upwarps: Cincinnati arch, Findlay arch, Kankakee arch, Ozark uplift, Nashville dome, and intervening Michigan and Illinois basins. Uplift and subsidence produced localized faulting and folding. The north northeast-trending Waverly arch of west central Ohio is the nearest upwarp structure.

Faults in the site region include:

- Chatham sag faults
- Peck fault, Howell-Northville anticline faults
- Bowling Green fault
- Anna Ohio faults
- Cincinnati arch faults
- Eastern Ohio faults
- Western New York faults
- Appalachian Plateau and Northern Valley and Ridge faults

Within the region only the Clarendon-Linden fault system in western New York is considered active.

4.3 SITE GEOLOGY

In conjunction with the PSAR and FSAR preparation and reviews, intensive geological and geotechnical investigations were conducted at the Perry site including:

- test borings (maximum depth 730')
- 42" drilled exploratory shafts
- in-site testing, plate load tests
- permeability determinations
- piezometer installations
- seismic analyses
- seismic refraction and seismic shear wave determinations
- geologic mapping of excavations, tunnels and trenches

Two bedrock structural styles were observed by Gilbert Commonwealth, NRC staff, USGS, and the Corps of Engineers. Gentle northeast-trending folds with two to three foot wavelength and 6" amplitude were attributed to depositional processes. Two larger folds and several related faults were also examined. The folds terminated below foundation grade. Faults with characteristic north over south directed motion become bedding plane detachments at depth. One to

three inch thick gouge occurs in the fault zones. Absence of foreign materials, no recrystallization of country rock or crystallization within fault zone or adjacent fracture zones is interpreted to result from localized low temperature, relatively low stress deformation.

In summary, an approximately 45 foot thick layer, between excavation grade of the deepest onshore foundation excavations and the base of a boulder layer defining the bottom of structureless basal till, experienced deformation (folds, faults) including bedding detachment rotation and buckling, and slight upward thrusting. These features occur in glaciated terraine and are attributed to glacial loading, unloading and/or ice push mechanisms. Similar faulting was studied in the Warner Creek area with the same conclusions.

4.4 DEFORMATION - INTAKE AND DISCHARGE TUNNELS

Three minor low-angle north-northeast striking thrust faults occur in the intake and discharge tunnels to the north beneath Lake Erie. Displacements range between 0.5 and 2.5 feet, upward to the northeast.

Studies undertaken to define tunnel fault geometry included:

- o detailed mapping of tunnel walls
- o reconnaissance of lake bottom
- o lake shore reconnaissance
- o exploratory borings
- o borehole logging, offshore and onshore magnetic surveys
- o review of existing geophysical data
- o isotopic analyses of Lake Erie and fault seepage water

Studies to date fault included:

- o x-ray diffraction
- o clay mineralogic analysis
- o microcrack
- o consolidation of gouge

Miscellaneous studies included:

- borehole stress
- structure contour maps
- interviews with knowledgeable Ohio geologists

Investigations of the vertical and lateral extent of faulting indicated that the faulting did not extend upward to the lake floor. Borings at the projected western shoreline intersection showed no faulting. Conclusions reached from detailed mapping of the tunnel faults, geophysical surveys, borings, and analysis of fault gouge and seepage included:

- faults are genetically related; same fault or an echelon
- faults confined in Chagrin shale; limited lateral and vertical extent
- date of last motion is Pleistocene or older
- motion sense indicates faults originated in northwest directed stress field, approximately 90° from present stress field
- possible mechanisms of nontectonic glacial origin include ice sheet traction, differential downwarp, differential rebound, surficial stress relief ("pop up")
- geologic processes responsible for initiation and latest motion are nontectonic and no longer operative; therefore faults are not capable according to Appendix A to 10 CFR 100

4.5 CURRENT SEISMOLOGICAL AND GEOLOGICAL STUDIES

Immediately after the occurrence of the earthquake, CEI undertook a number of geological and seismological investigations to provide a thorough understanding of the earthquake and assess any impact on previous studies performed for the siting and licensing of the Perry Nuclear Plant.

In addition to the investigations undertaken by CEI, USGS, as well as various universities and private groups, have deployed instruments to study earthquake aftershocks.

Portable Seismographic Network

At the request of CEI, Weston Geophysical Corporation installed six portable analog seismographs (Sprengnether Instrument Co. MEG-800) in the epicenter area of the January 31, 1986 earthquake during the period from approximately 10 hours to 30 hours after the event.

These seismograph stations are located at the Perry Nuclear Plant and in the communities of Chardon, Chesterland, Middlefield, Hartsgrove, and Thompson. A seventh station was installed on February 4, 1986 in the town of Concord. This spatial distribution of the stations is designed to form a symmetrical array around the preliminary epicental area of the main shock, which was located in the basis of more distant stations. All instruments are operated continuously and all seismograms are recovered and analyzed daily. The purpose of this network is to obtain accurate locations of any recorded aftershocks, to refine the original location of the main shock, and to determine whether or not their occurrence reveals anything about the causative geologic structure.

Five other portable instruments integrated into this network are operated by Woodward-Clyde Consultants and deployed in a similar configuration to provide additional locationing capabilities.

Five small microearthquakes have been detected. The parameters of these earthquakes are located on the Table 4-1. Preliminary analyses indicate that the focal depths for these microearthquakes range from 2.3 to 8.9 kilometers. The largest of these microearthquakes, a magnitude 2.4 event on February 6, 1986, was the only event to be felt. These microearthquake locations are slightly to the west of the preliminary location of main shock provided by the National Earthquake Information Center.

Felt Intensity Investigation

A questionnaire survey is being conducted to evaluate the distribution of effects, including a general description of how people experienced the event and accounts of any damage that have been incurred. The questionnaires are being distributed using several parallel approaches to obtain broad coverage of the affected areas. Analysis and compilation of questionnaire results will be used to produce an "isoseismal map" or plot of intensity levels measured on the Modified Mercalli Scale. The purpose of such a map is to enable a comparison of effects of the present event with a well-known epicenter to the effects of some historical events located in the site area that have no well-determined instrumental epicenter.

Weston Geophysical personnel have been conducting personal interviews on perception and other effects of the earthquake in the epicenter region. Questionnaires have been distributed at establishments such as fire departments, grocery stores, schools, etc. with instructions to distribute these to persons near the earthquake epicenter. These reports will be used to recover information on the range of effects.

A preliminary evaluation of returned questionnaires indicates that most of the reports in the epicentral area are evaluated as representative of an Intensity VI on the Modified Mercalli Scale. Maximum observed or reported effects include a few instances of damaged chimneys above the roof line, cracks in concrete and cinder block walls, cracked or fallen plaster, and few broken windows. Some disturbances including silting of well-water have also been reported.

Geologic Studies

Weston Geophysical geologists have conducted preliminary reconnaissance of bedrock exposures in the epicentral area to determine whether or not any surface expression resulted from the earthquake. No significant expression of surface disturbance has been observed. Although several occurrences of minor rock slides and soil slumps have been documented and photographed, these are not considered unusual, since they occur in unstable, undercut steam banks where they could have been caused by ordinary weathering processes or induced vibratory ground motion from the earthquake.

Previously mapped fault locations on Paine Creek have been examined. No evidence of recent fault movement was observed. Also, no slumping or sliding of the steep slope was apparent. No evidence suggestive of a "capable fault" has been observed.

On-going work includes examination of other geological features, as well as an investigation of sites of unusual felt reports such as foundation damage and water-well disturbance. A field observation and evaluation of soil and rock conditions at such sites is being made to determine whether or not there is a correlation between the higher intensity values and geological conditions.

4.6 CONCLUSIONS REGARDING OF THE JANUARY 31, 1986 OHIO EARTHQUAKE

The earthquake, both as regards to magnitude and intensity, is below the maximum earthquake selected to represent the Safe Shutdown Earthquake. The intensity of the Safe Shutdown Earthquake was selected as intensity VII. It is estimated that the present earthquake is best represented by an intensity VI. The magnitude 4.96 M_{blg} of the January 1986 earthquake is below the magnitude of 5.3 $\pm$ 5, used in establishing the site specific response spectra.

Based on the initial data evaluation, it appears that the free field design response spectra constructed to represent the SSE may have been exceeded. An accelerogram at the foundation level of Unit 1 showed a peak acceleration of 0.18 g at approximately 20 Hz on the north-south component. The duration of the motion on foundation above the smoothed ground response spectra (SSE) is less than 0.1 second. Since both the Regulatory Guide 1.60 ground motion and the site-specific spectra represent a smoothed spectra at the 84th percentile for a number of strong motion accelerograms, exceedances above the smoothed spectra are not unexpected.

At the high frequency end of the spectra, where the 20 Hz exceedance exists, it is important to look at the other parameters of ground motion. The particle velocity associated with the 0.18 g. is 0.55 inch per second and the displacement is 0.004 inch. This velocity value would be far less than the 1 inch per second generally accepted by the US Bureau of Mines as the threshold of damage at the 20 Hz frequency: cracking of plaster walls, etc. to ordinary structures. (Siskind, D.E. et al., 1980, Structure Response and Damage Produced by Ground Vibrations from Surface Mine Blasting, Bureau of Mines RI 8507). Structural damage therefore is not a problem.

The area and region in which the January 31, 1986 earthquake occurred is one of low seismicity. Prior to 1986, the largest earthquake to occur within 50 miles of the site occurred in 1943. The 1986 Ohio earthquake is slightly larger in magnitude (4.9 vs. 4.7) and intensity (VI vs. V) than the March 9, 1943 earthquake which occurred approximately 12 miles west-southwest of the 1986 earthquake. Although somewhat larger than historical earthquakes within 50 miles of the plant site, it is smaller than those within 200 miles of the site, as well as those on which the plant design is based. This earthquake is consistent with the seismicity of the area and the area and region are still of low seismicity.

Geological investigations to date have not uncovered any evidence suggestive of a "capable fault" as defined in 10 CFR Part 100, nor has the investigation revealed a cause for any geological concern. The 1986 earthquake does not change the conclusions in the FSAR on the geology and seismology of the Perry site.

Table 4.1

RECENT EARTHQUAKES
IN THE SITE VICINITY

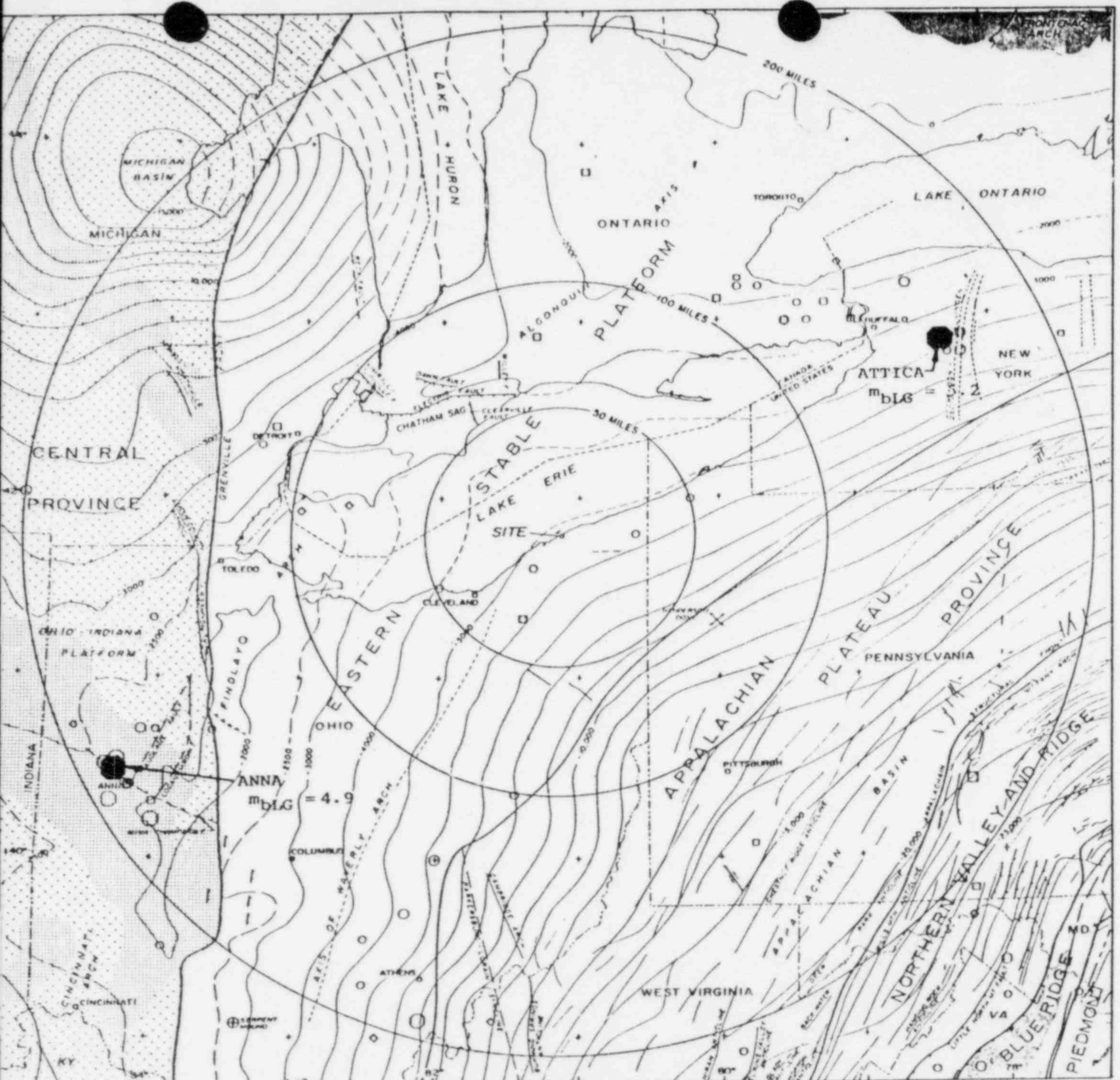
DATE	ORIGIN TIME (1)	LATITUDE	LONGITUDE	PRELIMINARY DEPTH (KM)	MAGNITUDE
22-JANUARY-1983	07:46:57.9	$41^{o}51.24'$	$81^{o}11.46'$	5	$2.7M_{bLg}$ (2)
31-JANUARY-1986	16:46:42.3	$41^{o}38.84'$	$81^{o}05.30'$	10	$4.9\ M_{b}$ (3)
01-FEBRUARY-1986	18:54:49.7	$41^{o}38.39'$	$81^{o}09.99'$	3.1	--
02-FEBRUARY-1986	03:22:49.1	$41^{o}38.37'$	$81^{o}09.81'$	2.3	--
03-FEBRUARY-1986	19:47:19.6	$41^{o}39.19'$	$81^{o}10.27'$	9	--
05-FEBRUARY-1986	06:34:02.4	$41^{o}39.93'$	$81^{o}09.11'$	6	--
06-FEBRUARY-1986	18:36:22.6	$41^{o}38.66'$	$81^{o}09.80'$	5	2.4

(1) UNIVERSAL Time Unless Noted As Local Time

(2) SOURCE: University of Michigan

(3) SOURCE: National Earthquake Information Center (NEIC)


SITE
5 miles
10 miles
FEB. 5
FEB. 3
JAN 31
M_b 4.9
FEB. 6
M_b 2.4
FEB. 1
FEB. 2
0
3 MI
Figure 4.1



REGIONAL TECTONIC ELEMENTS

Grenvillian Basement - Ages around 950 million years (Shaded: Exposed, Non-patterned: Buried)

Keweenawan Basement - Ages around 1100 million years

Elsonian Basement - Ages around 1450 million years

Structure contours in feet drawn on the top of Precambrian basement surface

Thrust fault - teeth on upper plate

Normal fault - hatchures on downthrown side

High angle fault

Inferred fault

Anticlinal axis

Synclinal axis

Intensely disturbed "Cryptoexplosive" structure

(Primary basement - structure sources: Bayley & Muehlberger, 1968, Hinze et al. 1975; Owens, 1967)

0 50 100 150 Miles

0 50 100 150 200 Kilometers

REGIONAL TECTONIC PROVINCES

PROVINCE BOUNDARY

EARTHQUAKES

MAGNITUDE (1930 through 1967)	MAGNITUDE (after 1-1-1968)		INTENSITY	
□	○	3.1-3.5	○	V
□	◇	3.6-4.2	○	VI
□	◇	4.3-4.7	○	VII
			○	VIII

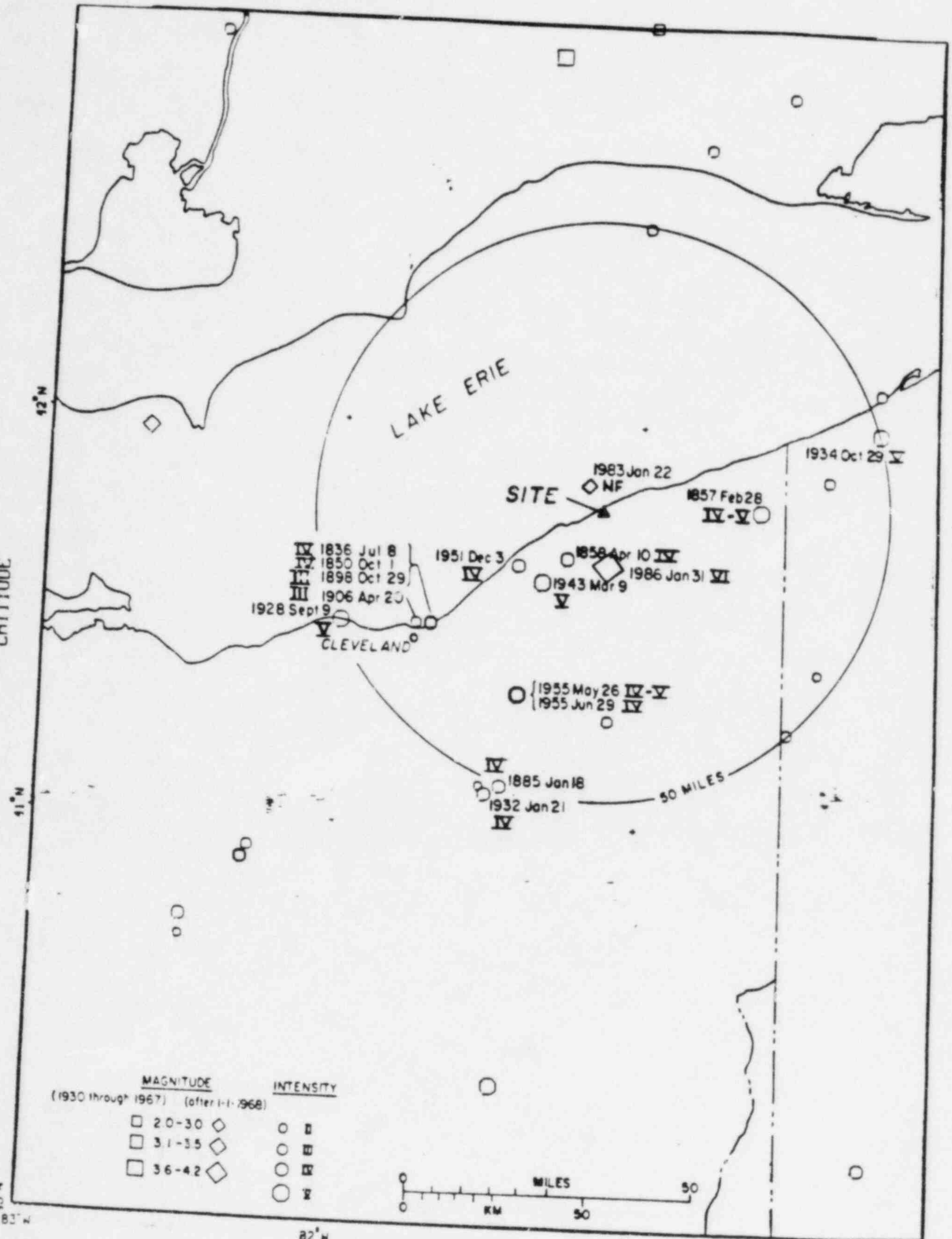
Figure 4.2

Am. 10 (11-29-82)

PERRY NUCLEAR POWER
THE CLEVELAND ELEC
ILLUMINATING COMP

Regional Tectonics -
Earthquake Tectonic Provi

Figure 2.5-59



LAKE ERIE
SITE
1983 Jan 22
NF
1857 Feb 28
IV-V
1934 Oct 29 V
IV 1836 Jul 8
IV 1850 Oct 1
III 1898 Oct 29
III 1906 Apr 20
1928 Sept 9
V
CLEVELAND
1951 Dec 3
IV
1858 Apr 10 IV
1986 Jan 31 VI
1943 Mar 9
V
1955 May 26 IV-V
1955 Jun 29 IV
IV
1885 Jan 18
1932 Jan 21
IV
50 MILES
42°N
41°N
40°N
83°W
82°W
81°W
80°W
LATITUDE
LONGITUDE
MAGNITUDE
(1930 through 1967)
(after 1-1-1968)
2.0-3.0
3.1-3.5
3.6-4.2
INTENSITY
MILES
0
50
KM

Figure 4.3

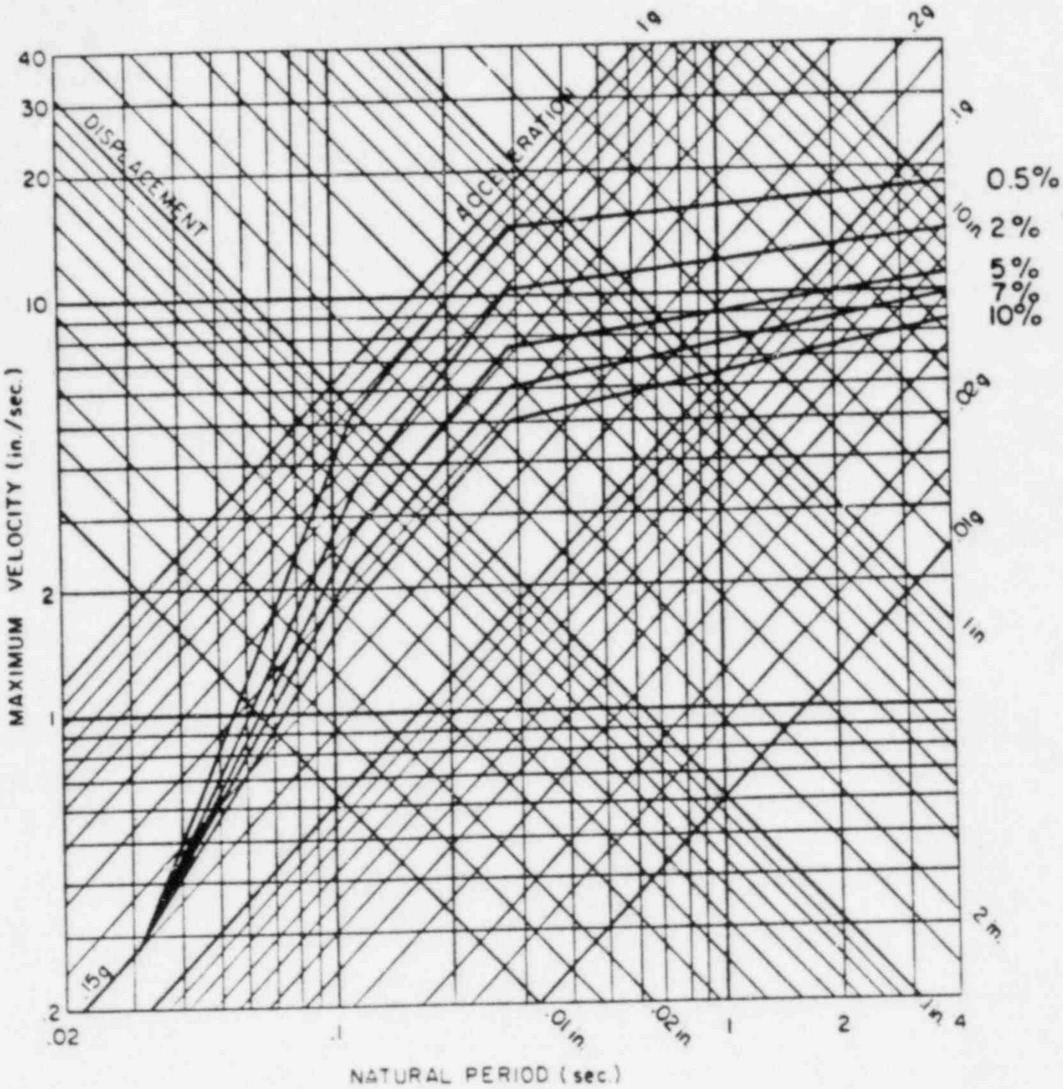


Figure 4.4

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Safe Shutdown Earthquake
Design Response Spectra -
Vertical Motion

Figure 3.7-2

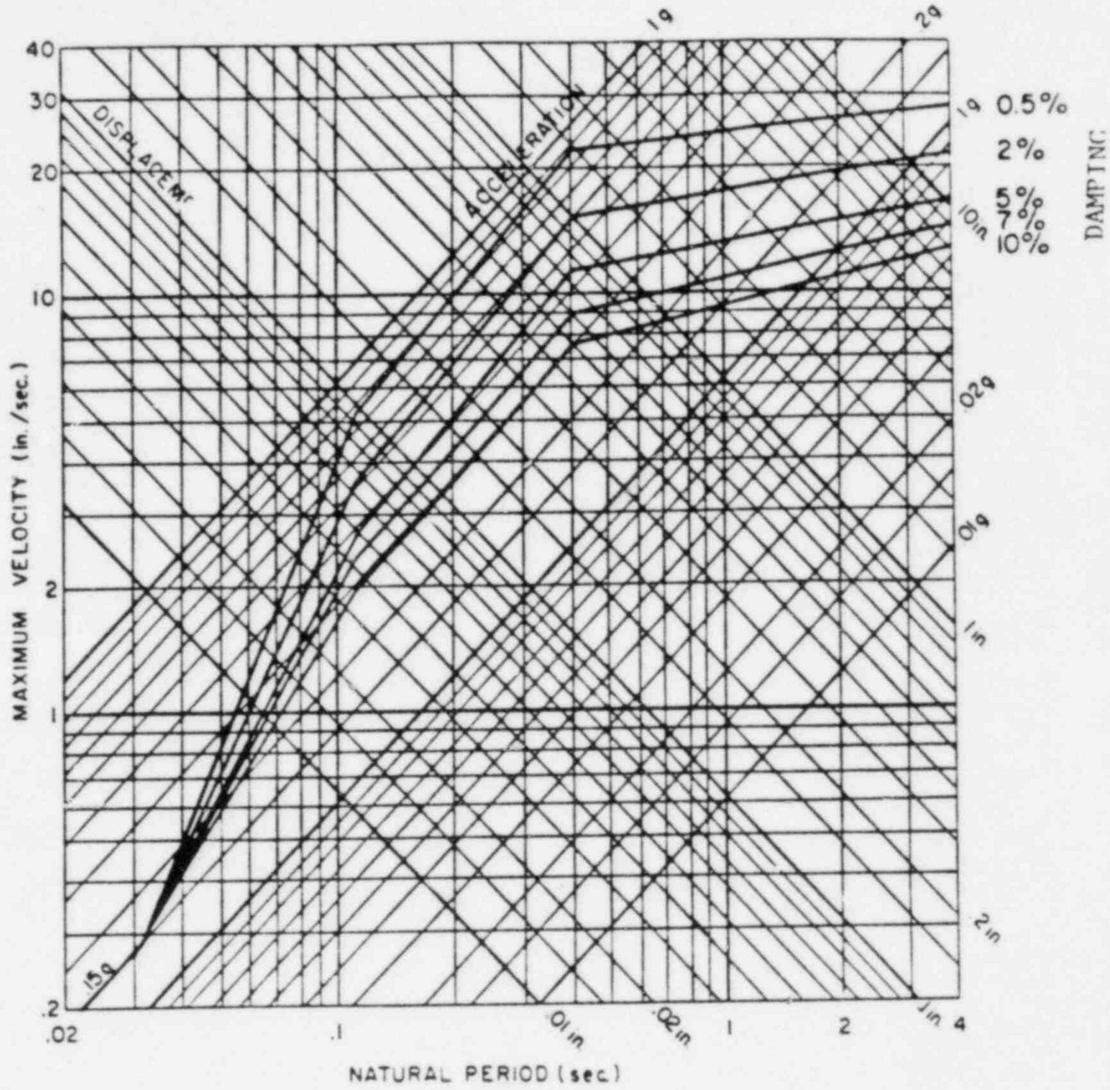
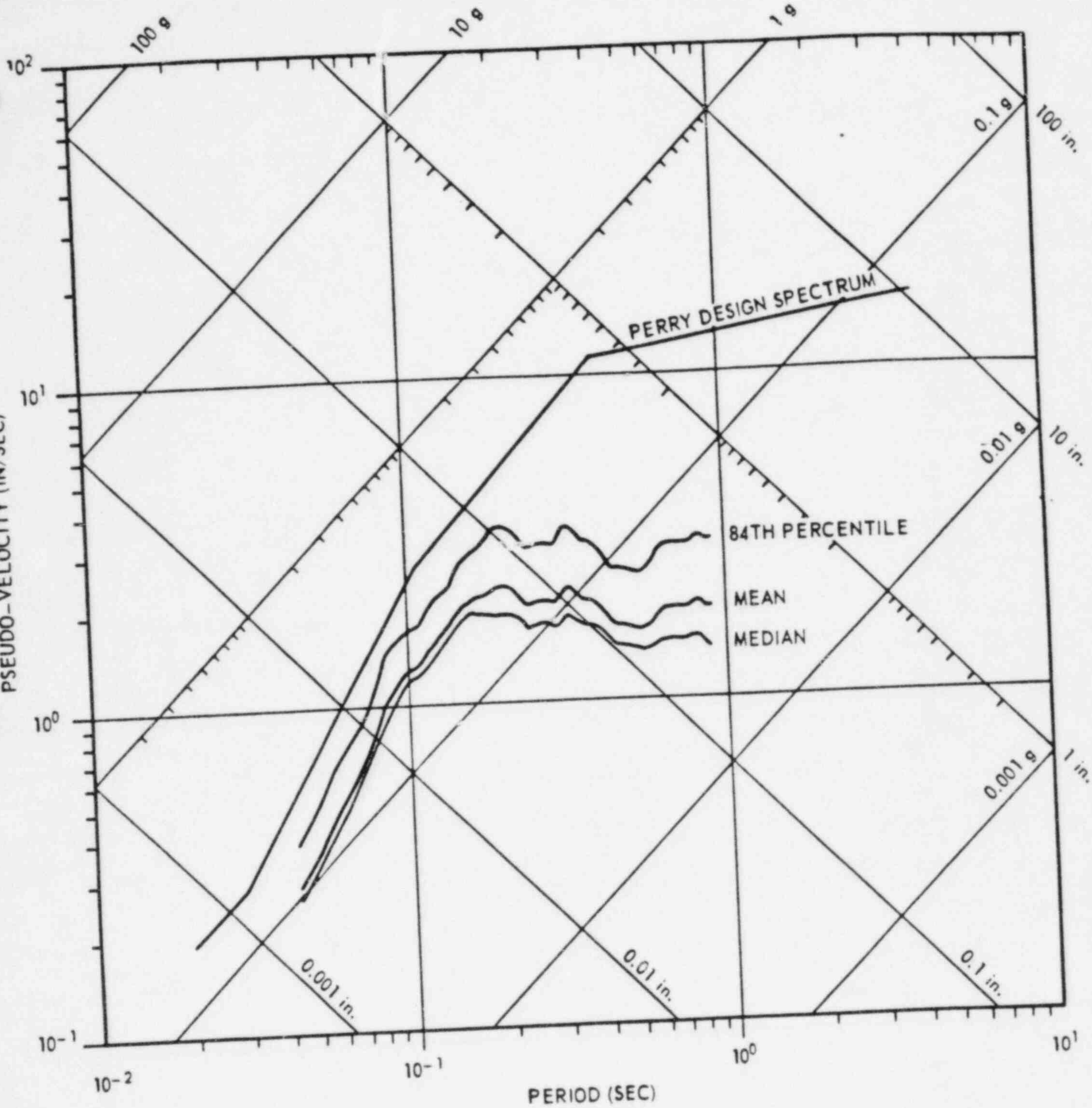


Figure 4.5

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Safe Shutdown Earthquake
Design Response Spectra -
Horizontal Motion

Figure 3.7-1



MEDIAN, MEAN AND 84TH PERCENTILE RESPONSE SPECTRA FOR PERRY (ROCK) SITE. (BASIC SUBSET, 5% DAMPING)

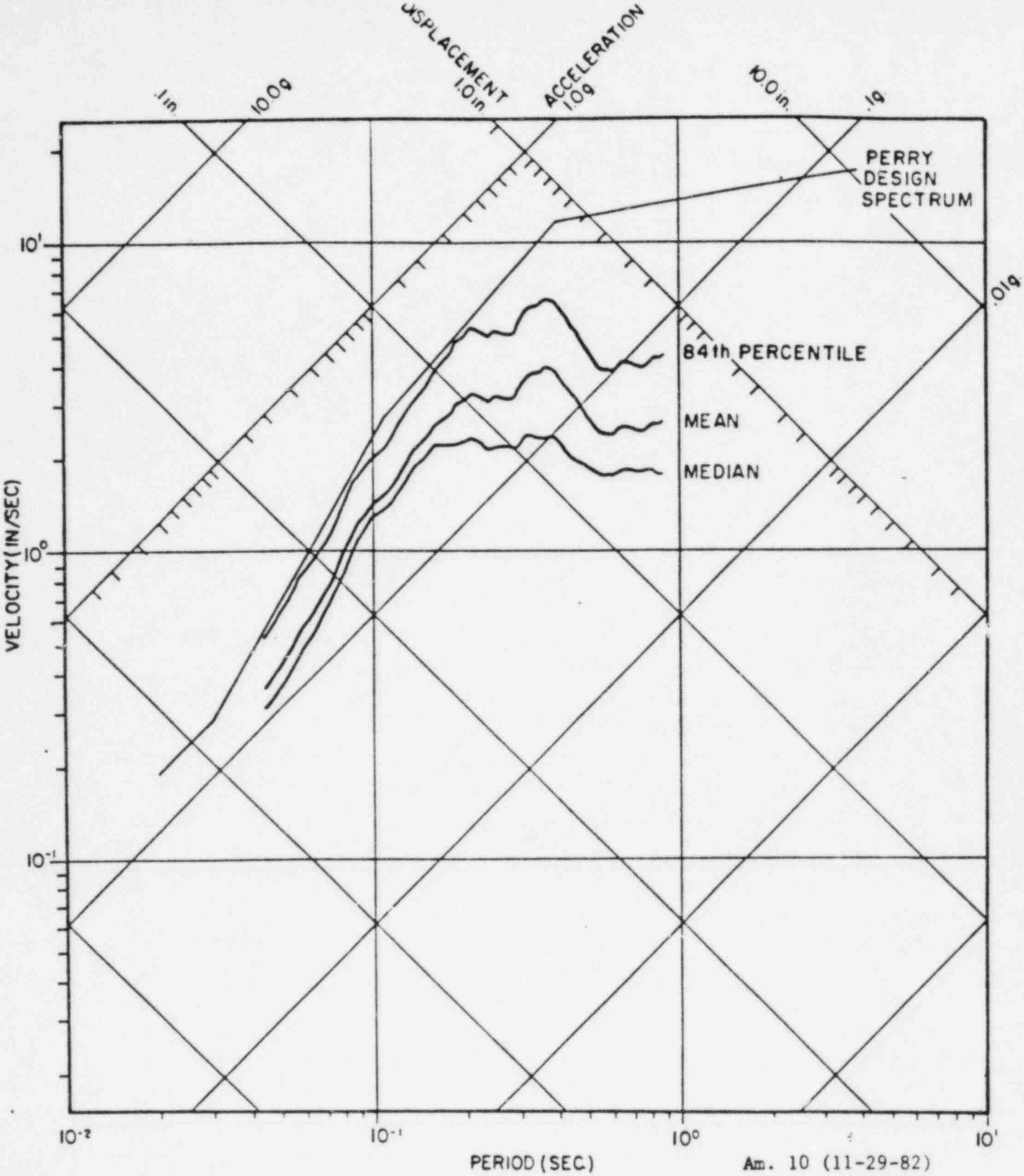
$m_{bLG} = 5.3 \pm .5$

Am. 10 (11-29-82)

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Figure 230.6-2

Figure 4.6



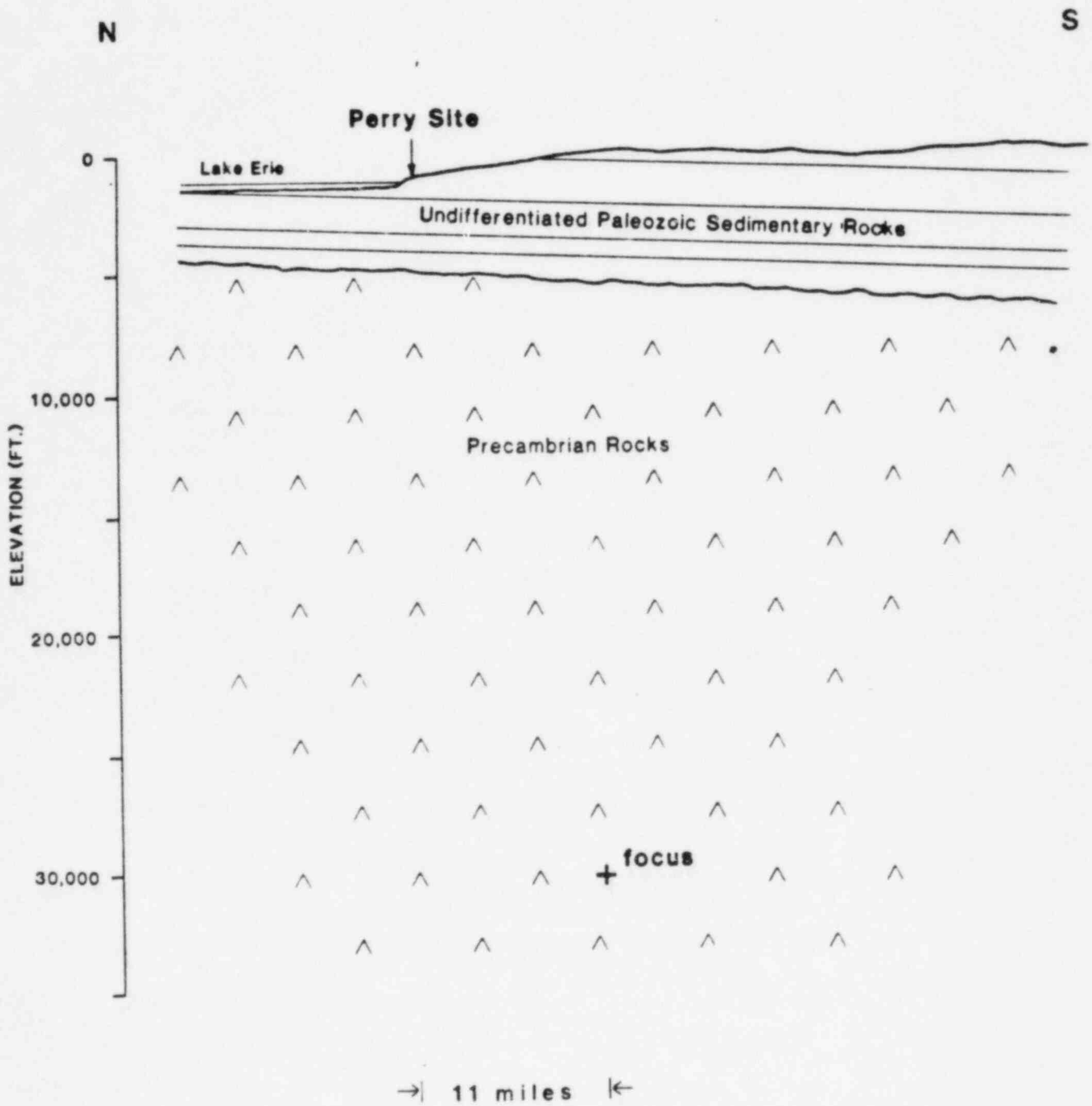
Am. 10 (11-29-82)

MEDIAN, MEAN AND 84TH PERCENTILE
RESPONSE SPECTRA FOR PERRY (ROCK) SITE
(Basic Subset, Parkfield Included, 5% Damping)
$m_{bLG} = 5.53 \pm .3$

PERRY NUCLEAR POWER PLANT THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
Figure 230.6-5

Q&R 2.5-40

Figure 4.7



N
S
Perry Site
0
Lake Erie
Undifferentiated Paleozoic Sedimentary Rocks
10,000
ELEVATION (FT.)
Precambrian Rocks
20,000
30,000
focus
11 miles

Figure 4.8

5.0 SEISMIC INSTRUMENTATION DATA EVALUATION

Three different types of seismic monitoring instrumentation were used to record the 1986 Ohio Earthquake. Table 5.1 and Figure A through H and J delineate the specific instrument number, type and location. One type of instrument used is the Kinemetrics Model SMA-3 strong motion triaxial time-history accelerograph. This system detects and records three mutually perpendicular components of acceleration over the entire duration of the earthquake onto cassette magnetic tape. Power to the unit is supplied by internal rechargeable batteries which are kept in a charged state by 120 VAC line power. Two instruments of this type were used and were located on the Reactor Building Foundation Mat at an elevation of approximately 575 feet. Their latest calibration was December 1, 1985. See Appendix A for further instrumentation details and data tabulation.

The second type of instrumentation used was the Engdahl PSR 1200-H/V response spectrum recorder. This totally mechanical system also records three mutually perpendicular components of acceleration. The instrument used twelve reeds fabricated of varying lengths and weights of spring steel, one for each frequency (ranging from approximately 2 Hz to 25 Hz). A diamond-tipped stylus is attached to the free end of each reed to inscribe a permanent record of its deflection on one of twelve record plates. The record plates are made of aluminum and plated with successive layers of nickel, tin and lead-tin. This system is totally self-contained and requires no outside power source.

Four instruments of this type were used - two on the Auxiliary Building Foundation Mat and an elevation of approximately 568 feet, one at the Reactor Building Foundation Mat at an elevation approximately 575 feet, and one at the Reactor Building Inside Drywell Platform at an elevation of approximately 630 feet. Except for the one instrument located on the Reactor Building platform which was calibrated on January 30, 1986, all instruments of this type were calibrated during January 1985. See Appendix B for further instrumentation details and data tabulation.

The third type of instrument was the Engdahl PAR 400 peak accelerograph. This totally mechanical system records three mutually perpendicular components of peak local ecceleration (i.e., the zero period acceleration). A diamond tipped scriber at the end of an amplifier arm records a permanent mark on a record plate made of aluminum and successive layers of nickel, gold and burnt gold. Again, this system is totally self-contained and requires no outside power source. Two instruments of this type were used and were located on the Auxiliary Building Foundation Mat at an elevation of approximately 568 feet and on the Reactor Recirculation Pump at the elevation of approximately 605 feet. The latest calibration date for the Auxiliary Building instrument was January 30, 1986, while the calibration date for the Recirculation Pump instrument was December 4, 1985. A third instrument of this type was out of service at the time of the earthquake because it was being recalibrated. See Appendix B for further instrumentation details and data tabulation.

All recorded data from the in-plant seismic instruments have been used in the evaluation.

PERRY NUCLEAR POWER PLANT UNIT NO. 1
SEISMIC MONITORING INSTRUMENTATION

TABLE 5.1

Instrument Number	Type	Manufacturer / Model Number	Location	References
D51-N101	(1)	Kinemetrics / SMA-3	Reactor Building Foundation Mat Elevation 575'−10" Azimuth 175°	Figures A and B
D51-N111	(1)	Kinemetrics / SMA-3	Reactor Building Containment Vessel Elevation 686'−0" Azimuth 174°	Figures A and C
D51-R120	(2)	Engdahl / PAR-400	Reactor Recirculation Pump (Inside Drywell, Reactor Building) Elevation 605'−0" (Approximately) Azimuth 145°	Figures A and D
D51-R130	(2)	Engdahl / PAR-400	OUT OF SERVICE	
D51-R140	(2)	Engdahl / PAR-400	Auxiliary Building Foundation Mat (HPCS Pump Room) Elevation 568'−4"	Figures A and E

1. Triaxial Time-History Accelerograph
2. Triaxial Peak Accelerograph
3. Triaxial Response Spectrum Recorder

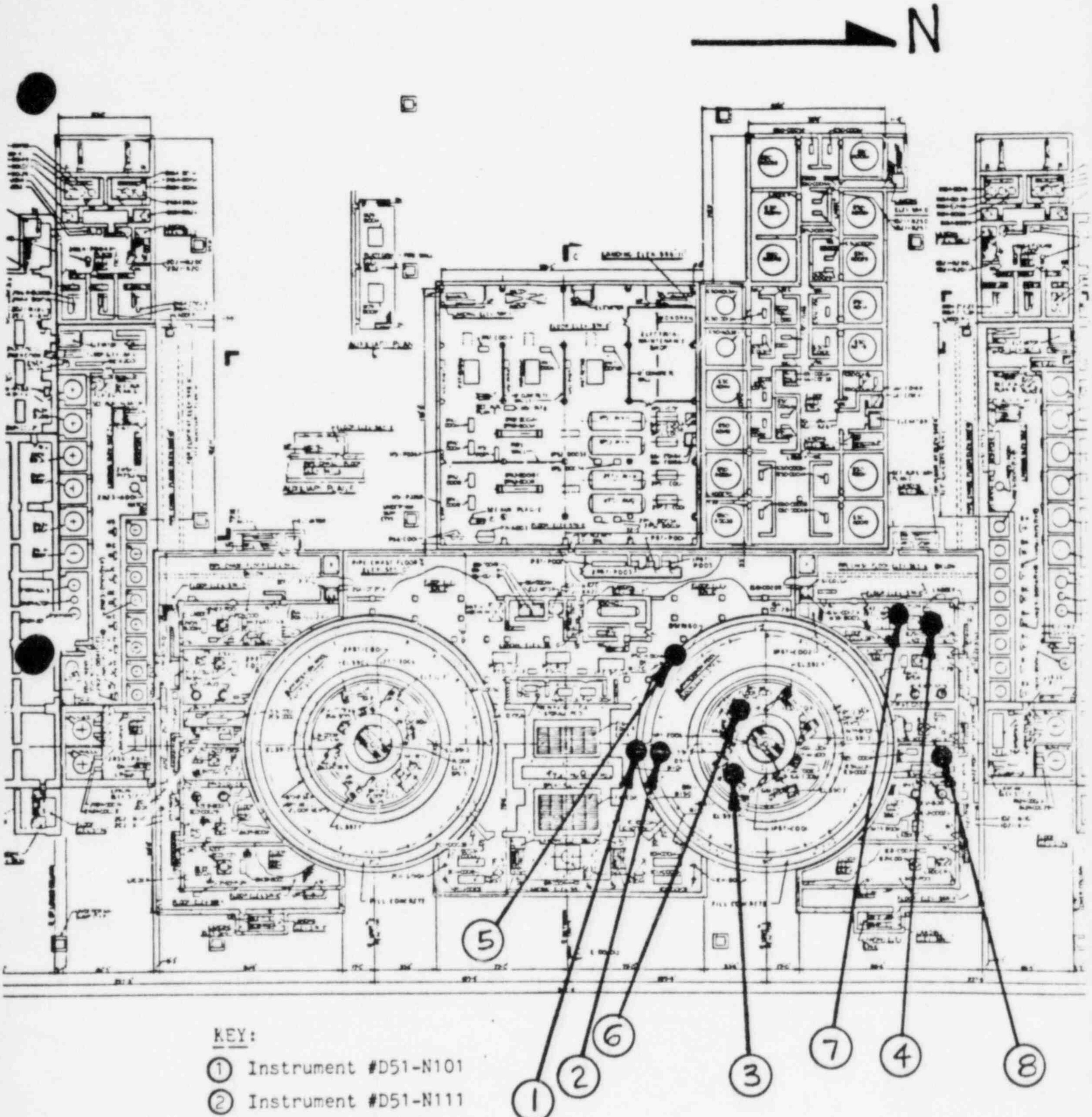
PERRY NUCLEAR POWER PLANT UNIT NO. 1
SEISMIC MONITORING INSTRUMENTATION

TABLE 5.1

Instrument Number	Type	Manufacturer / Model Number	Location	References
D51-R160	(3)	Engdahl / PSR-1200-H / V-12A	Reactor Building Foundation Mat Elevation 574'–10" Azimuth 225°	Figures A and F
D51-R170	(3)	Engdahl / PSR-1200-H / V	Reactor Building 630' Platform (Inside Drywell) Elevation 630'–1" Azimuth 238°	Figures A and G
D51-R180	(3)	Engdahl / PSR-1200-H / V	Auxiliary Building Foundation Mat (HPCS Pump Room) Elevation 568'–4"	Figures A and H
D51-R190	(3)	Engdahl / PSR-1200-H / V	Auxiliary Building Foundation Mat (RCIC Pump Room) Elevation 568'–4"	Figures A and J

1. Triaxial Time-History Accelerograph
2. Triaxial Peak Accelerograph
3. Triaxial Response Spectrum Recorder

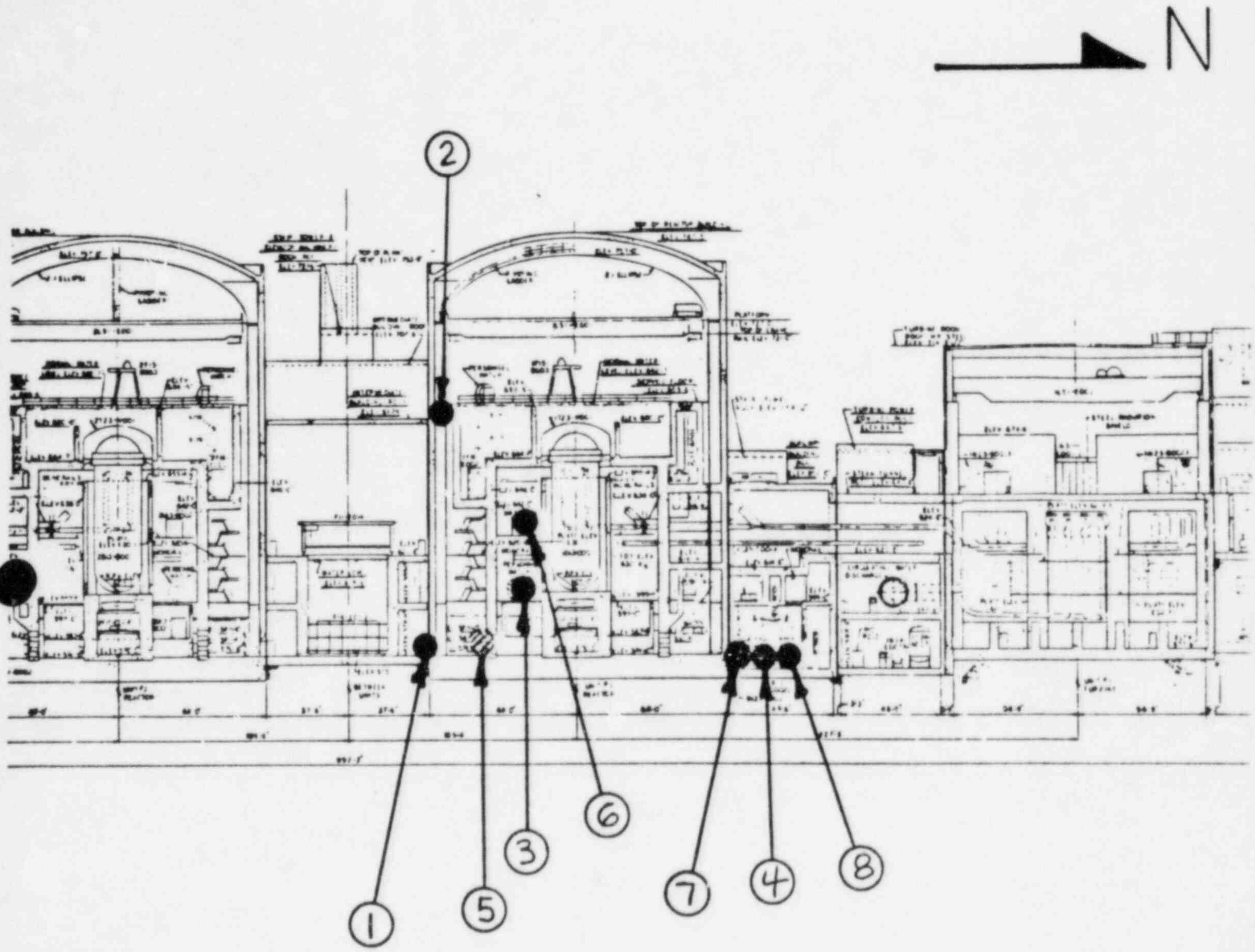
FIGURE A
Sheet 1 of 2



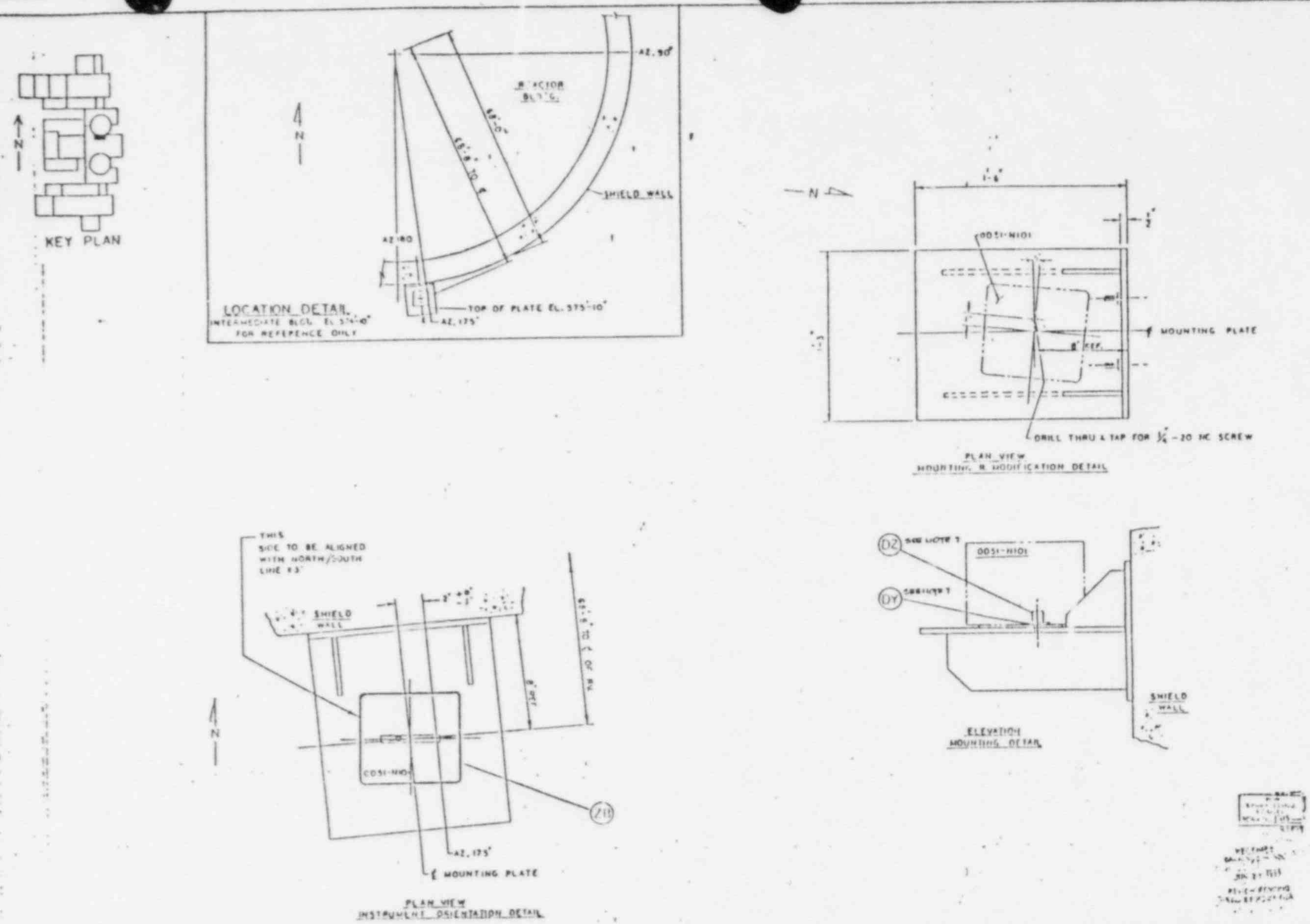
KEY:

1. Instrument #D51-N101
2. Instrument #D51-N111
3. Instrument #D51-R120
4. Instrument #D51-R140
5. Instrument #D51-R160
6. Instrument #D51-R170
7. Instrument #D51-R180
8. Instrument #D51-R190

FIGURE A
Sheet 2 of 2



1. #D51-N101 R/B Foundation Mat, El. 575', Az. 175°
2. #D51-N111 R/B Containment Vessel, El. 686', Az. 174°
3. #D51-R120 Reactor Recirc Pump, El. 605', Az. 145°
4. #D51-R140 A/B Foundation Mat, El. 568'
5. #D51-R160 R/B Foundation Mat, El. 574' Az. 225°
6. #D51-R170 R/B Platform, El. 630' Az. 238°
7. #D51-R180 A/B Foundation Mat, El. 568'
8. #D51-R190 A/B Foundation Mat, El. 568'



NOTES:

1) THIS DRAWING DEPICTS INSTRUMENT MOUNTING ONLY.
2) STRUCTURAL STEEL SPECIFICATION ASTM A-36.
3) UNLESS OTHERWISE SPECIFIED, ALL TOLERANCES ... ± 1/8".
4) NON-SAFETY SEISMIC, MAINTAIN 18" SEPARATION BETW... ANY NON-SAFETY AND NON-SEISMIC IMPULSE LINES ... DEVICES INSTALLED BY JCI.
5) MOUNTING BRACKET SUPPLIED AND INSTALLED BY ...
6) INSTRUMENT TO BE ALIGNED WITH NORTH/SOUTH LINE AS SHOWN ON THIS DRAWING.
7) MATERIAL SUPPLIED BY SP-90 AND INSTALLED BY ...
8) THE FOLLOWING MEASUREMENT TOLERANCE SHALL APPLY TO AS-BUILT DIMENSIONS.
9) ALL DIMENSIONS SHOWN ARE ACTUAL AS-BUILT DIMENSIONS.

THE FOLLOWING DOCUMENTS ARE APPLICABLE TO THIS DWG		
E.C.N.	F.V.A.	K.R.
N/A	N/A	N/A

REV.	DESCRIPTION
0	ORIGINAL ISSUE
1	AS BUILT

REFERENCE DOC	REV.
D714-F63	D

BILL OF MATERIALS

QTY	ITEM	DESCRIPTION
1	ZB	0D51-N101 TIME HISTORY ACCELEROGRAPH
1	DZ	LOCK WASHER
1	DY	[illegible]

JOHNSON CONTROLS

PERRY NUCLEAR POWER PLANT UNITS 1 & 2

SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR 0D51-N101

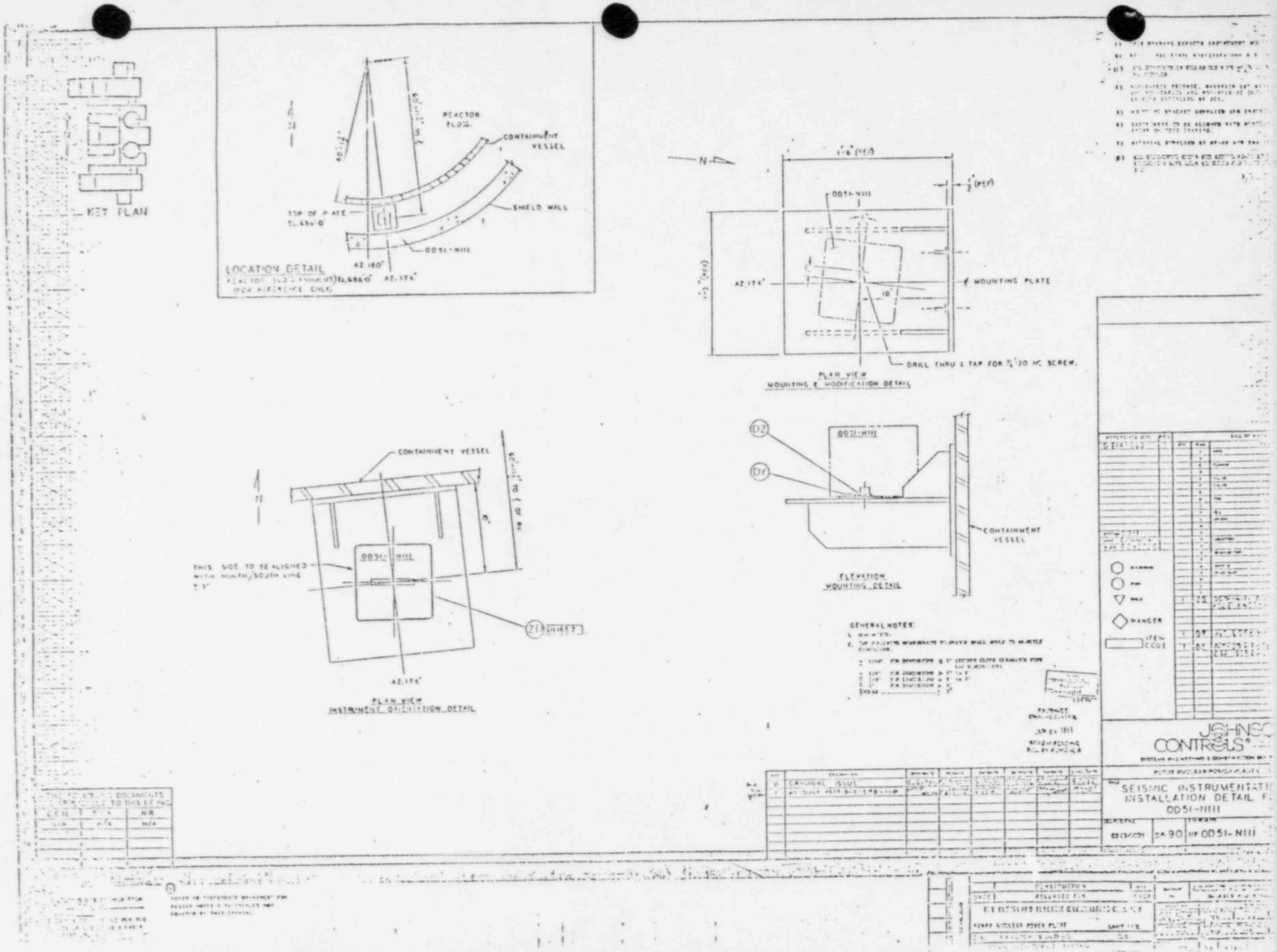
SP-90 | 0D51-N101

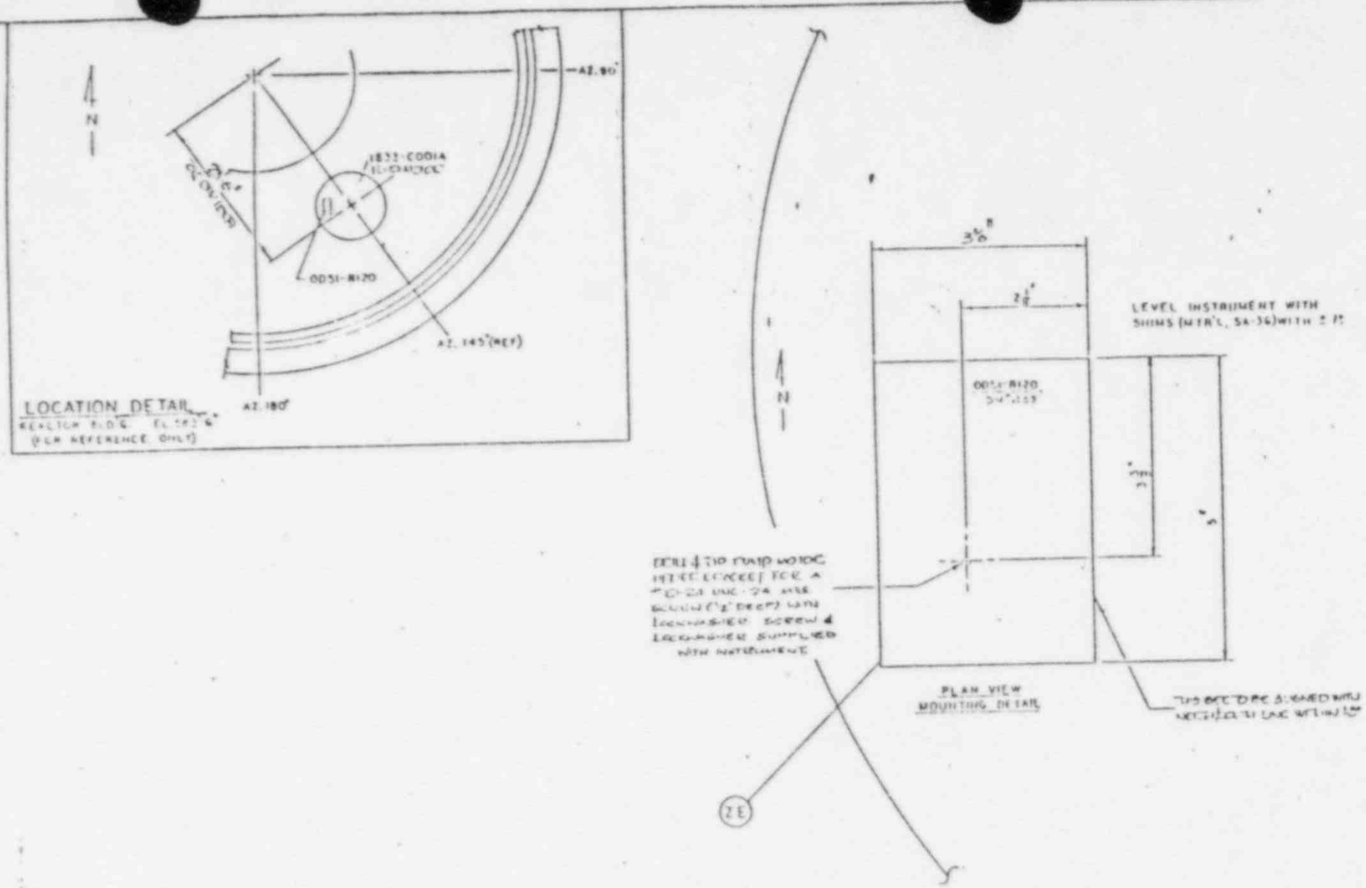
SCALE: NONE

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

PERRY NUCLEAR POWER PLANT UNIT 1 & 2

GILBERT ASSOCIATES, INC.


KEY PLAN
REACTOR FLOOR
CONTAINMENT VESSEL
SHIELD WALL
TOP OF PLATE
OD51-NIII
AZ.180°
AZ.174°
LOCATION DETAIL
(FOR REFERENCE ONLY)
N
1'-6" (REF)
MOUNTING PLATE
DRILL THRU & TAP FOR ¼"-20 NC SCREW.
PLAN VIEW
MOUNTING & MODIFICATION DETAIL
CONTAINMENT VESSEL
60'-11½" TO ℄ OF RV
THIS SIDE TO BE ALIGNED WITH NORTH/SOUTH LINE
AZ.174°
PLAN VIEW
INSTRUMENT ORIENTATION DETAIL
DZ
DY
ELEVATION
MOUNTING DETAIL
GENERAL NOTES:
HANGER
ITEM CODE
ORIGINAL ISSUE
JOHNSON CONTROLS
SEISMIC INSTRUMENTATION INSTALLATION DETAIL
OD51-NIII
OD51-NIII



NOTES:

1) THIS DRAWING DEPICTS INSTRUMENT MOUNTING ONLY.

2) UNLESS OTHERWISE SPECIFIED, ALL TOLERANCES SHALL BE ± 1/8".

3) NON SAFETY SEISMIC ... INSTALLED BY JCI.

4) PLAN AND LIFT FIXTURE SUPPLIED BY OTHERS.

5) INSTRUMENT TO BE ALIGNED WITH NORTH/SOUTH LINE AS SHOWN ON THIS DRAWING.

6) [illegible]

7) [illegible]

REFERENCE DOC	REV
D-814-153	2
OD51 COMMON SP-R120	

ITEM	DESCRIPTION
2E	RECTIFIED PEAK ACCELERATION RECORDER

HANGER

JOHNSON CONTROLS®
SYSTEMS ENGINEERING & CONSTRUCTION DIV. POWER UNIT
PERRY NUCLEAR POWER PLANT UNITS 1 & 2

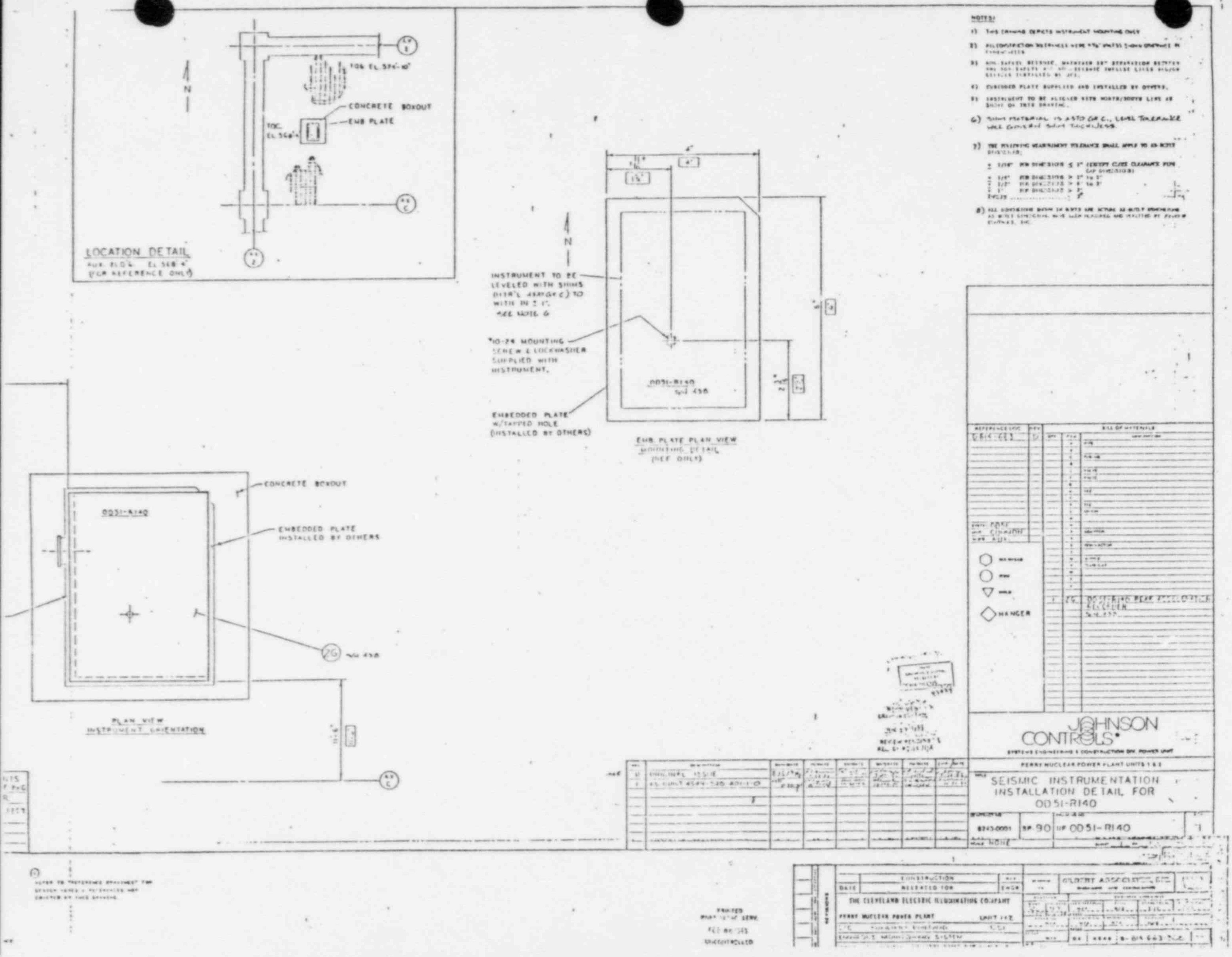
SEISMIC INSTRUMENTATION
INSTALLATION DETAIL FOR
OD51-R120

REV	DESCRIPTION
0	ORIGINAL ISSUE
1	[illegible]

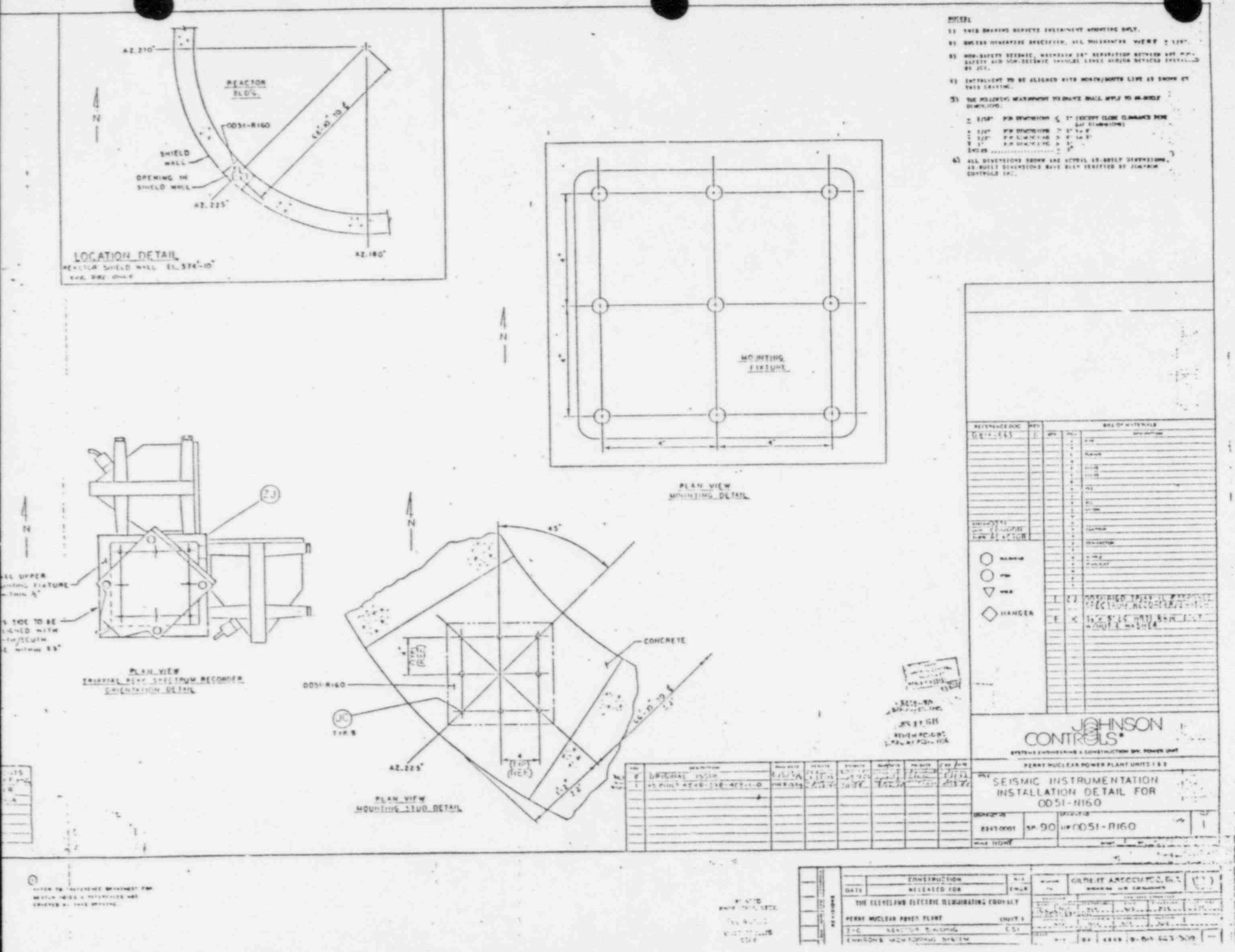
8243-0001 | SP. 90 | OF OD51-R120

JUN 27 1985
REVIEW PENDING

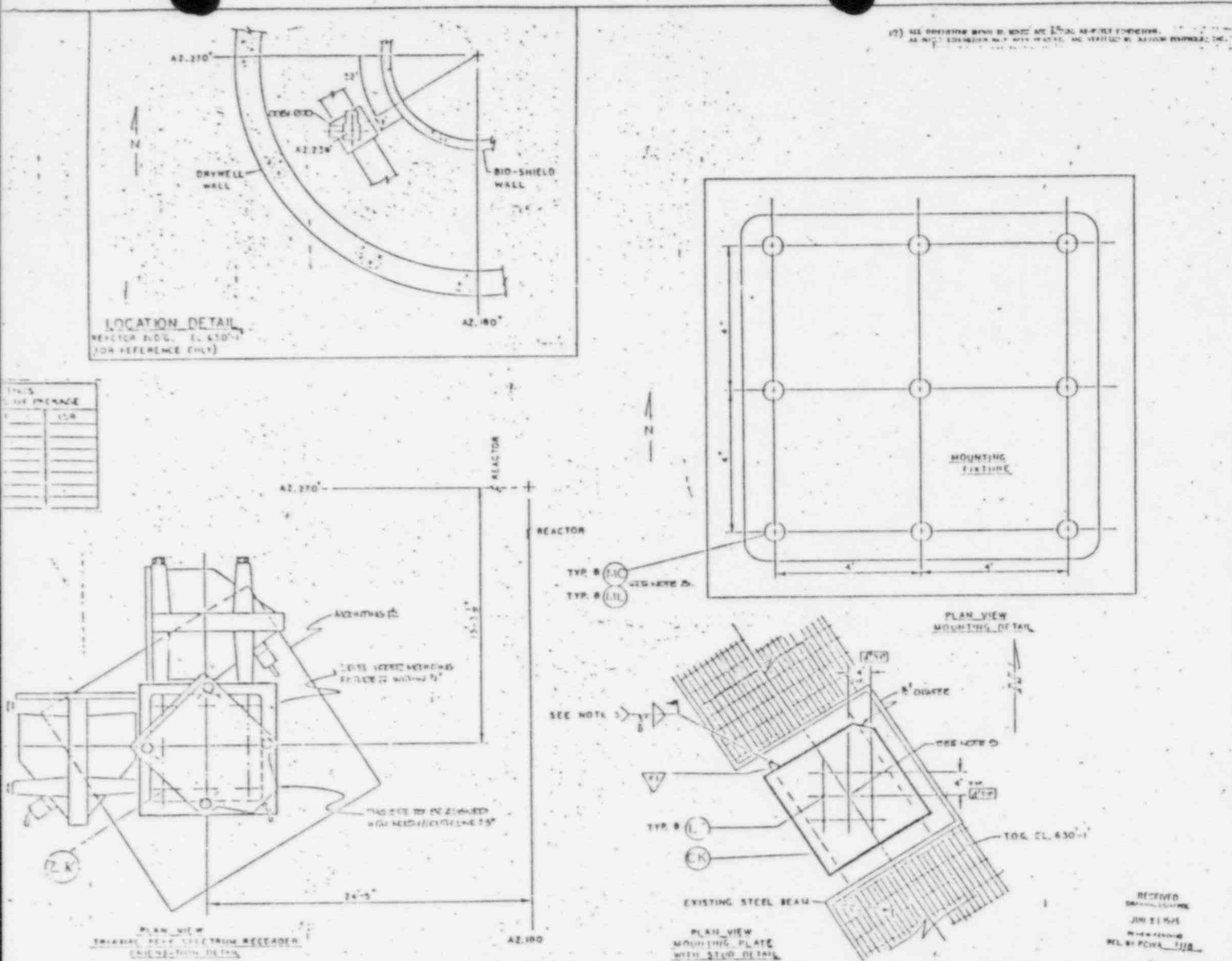
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
PERRY NUCLEAR POWER PLANT UNIT 1 & 2
REACTOR BUILDING
SEISMIC MONITORING SYSTEM
GILBERT ASSOCIATES, INC.
CONSTRUCTION
RELEASED FOR
D-511-563-306



NOTES:
1) THIS DRAWING DEPICTS INSTRUMENT MOUNTING ONLY
4) EMBEDDED PLATE SUPPLIED AND INSTALLED BY OTHERS.
LOCATION DETAIL
AUX. BLDG. EL 568'-4"
(FOR REFERENCE ONLY)
CONCRETE BOXOUT
EMB PLATE
INSTRUMENT TO BE LEVELED WITH SHIMS
SEE NOTE 6
10-24 MOUNTING SCREW & LOCKWASHER SUPPLIED WITH INSTRUMENT.
EMBEDDED PLATE W/TAPPED HOLE (INSTALLED BY OTHERS)
0D51-R140
EMB. PLATE PLAN VIEW
MOUNTING DETAIL
(REF ONLY)
CONCRETE BOXOUT
EMBEDDED PLATE INSTALLED BY OTHERS
PLAN VIEW
INSTRUMENT ORIENTATION
HANGER
ORIGINAL ISSUE
JOHNSON CONTROLS
PERRY NUCLEAR POWER PLANT UNITS 1 & 2
SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR 0D51-R140
SP 90
0D51-R140
CONSTRUCTION
RELEASED FOR
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
PERRY NUCLEAR POWER PLANT
GILBERT ASSOCIATES, INC.
PRINTED
UNCONTROLLED



LOCATION DETAIL
REACTOR SHIELD WALL EL. 574'-10"
REACTOR BLDG.
AZ. 270°
AZ. 180°
AZ. 225°
0D51-R160
SHIELD WALL
OPENING IN SHIELD WALL
MOUNTING FIXTURE
PLAN VIEW
MOUNTING DETAIL
4"
PLAN VIEW
TRIAXIAL PEAK SPECTRUM RECORDER
ORIENTATION DETAIL
PLAN VIEW
MOUNTING STUD DETAIL
CONCRETE
45°
0D51-R160
JC
TYP. 8
2J
HANGER
JOHNSON CONTROLS
PERRY NUCLEAR POWER PLANT UNITS 1 & 2
SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR 0D51-N160
SP-90
0D51-R160
ORIGINAL ISSUE
CONSTRUCTION
RELEASED FOR
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
PERRY NUCLEAR POWER PLANT
GILBERT ASSOCIATES, INC.



NOTES:

1) STUDS TO BE WELDED TO STEEL PLATE IN ACCORDANCE WITH ...

2) THIS DRAWING DEPICTS INSTRUMENT MOUNTING ONLY.

3) STRUCTURAL STEEL SPECIFICATION: ASTM A-36.

4) UNLESS OTHERWISE SPECIFIED, ALL TOLERANCES WERE ± 1/8".

5) ALL WELDING OF THIS DRAWING, EXCEPT STUDS, SHALL BE PERFORMED PER THE FOLLOWING PROCEDURE:

WELD NO.

6) NON-SAFETY SEISMIC, MAINTAIN 18" SEPARATION BETWEEN ANY NON-SAFETY AND NON-SEISMIC IMPULSE LINES AND/OR DEVICES INSTALLED BY JCI.

7) INSTRUMENT TO BE ALIGNED WITH NORTH/SOUTH LINE AS SHOWN ON THIS DRAWING.

10) MOUNTING PLATE SHALL HAVE STAMPED ON FACE ... "PLANT NORTH"

RECEIVED
JUN 21 1985

JOHNSON CONTROLS

PERRY NUCLEAR POWER PLANT UNITS 1 & 2

SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR OD51-R170

SP-90 1-OD51-R170

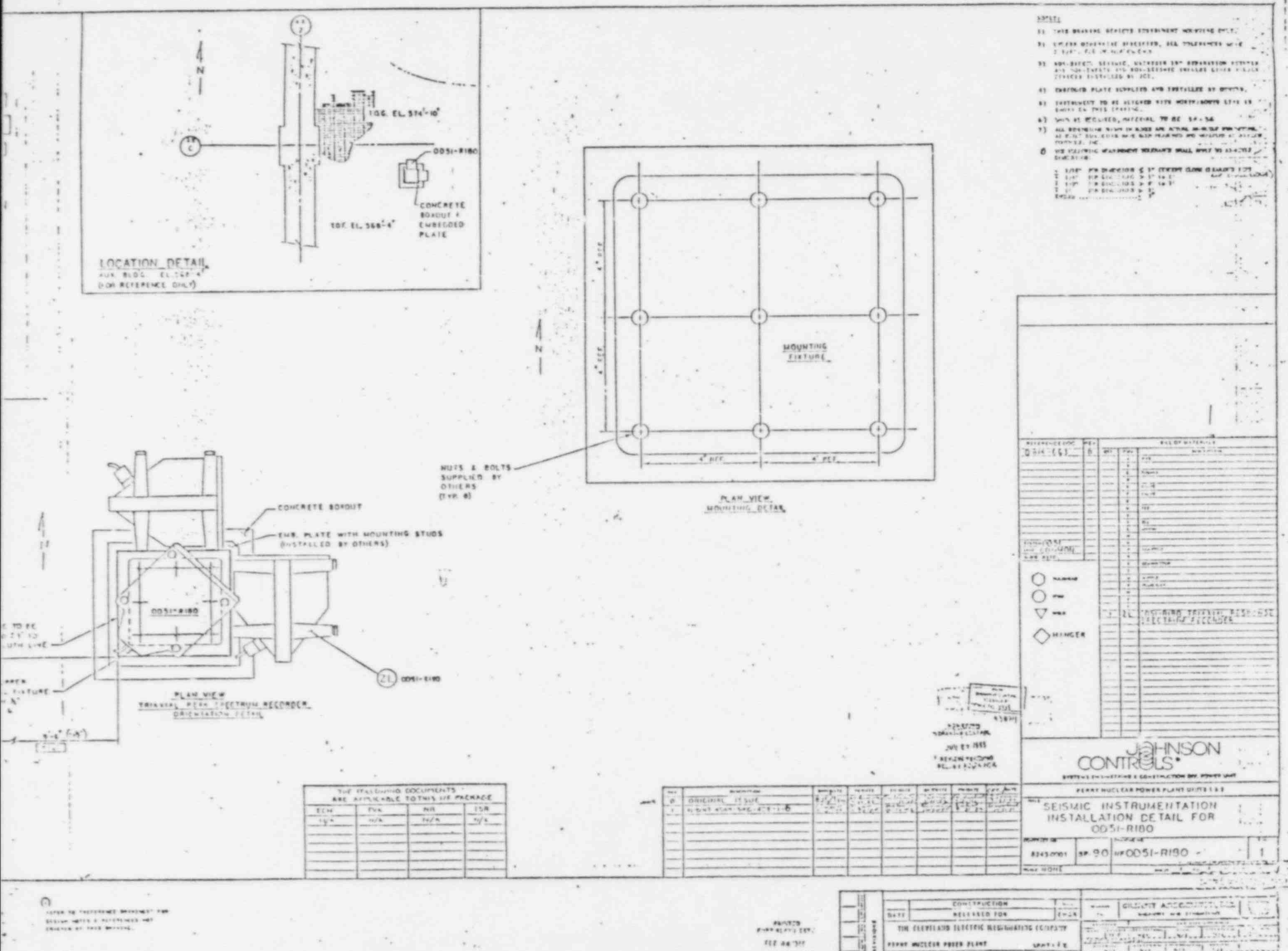
0	ORIGINAL ISSUE

GILBERT ASSOCIATES, INC.

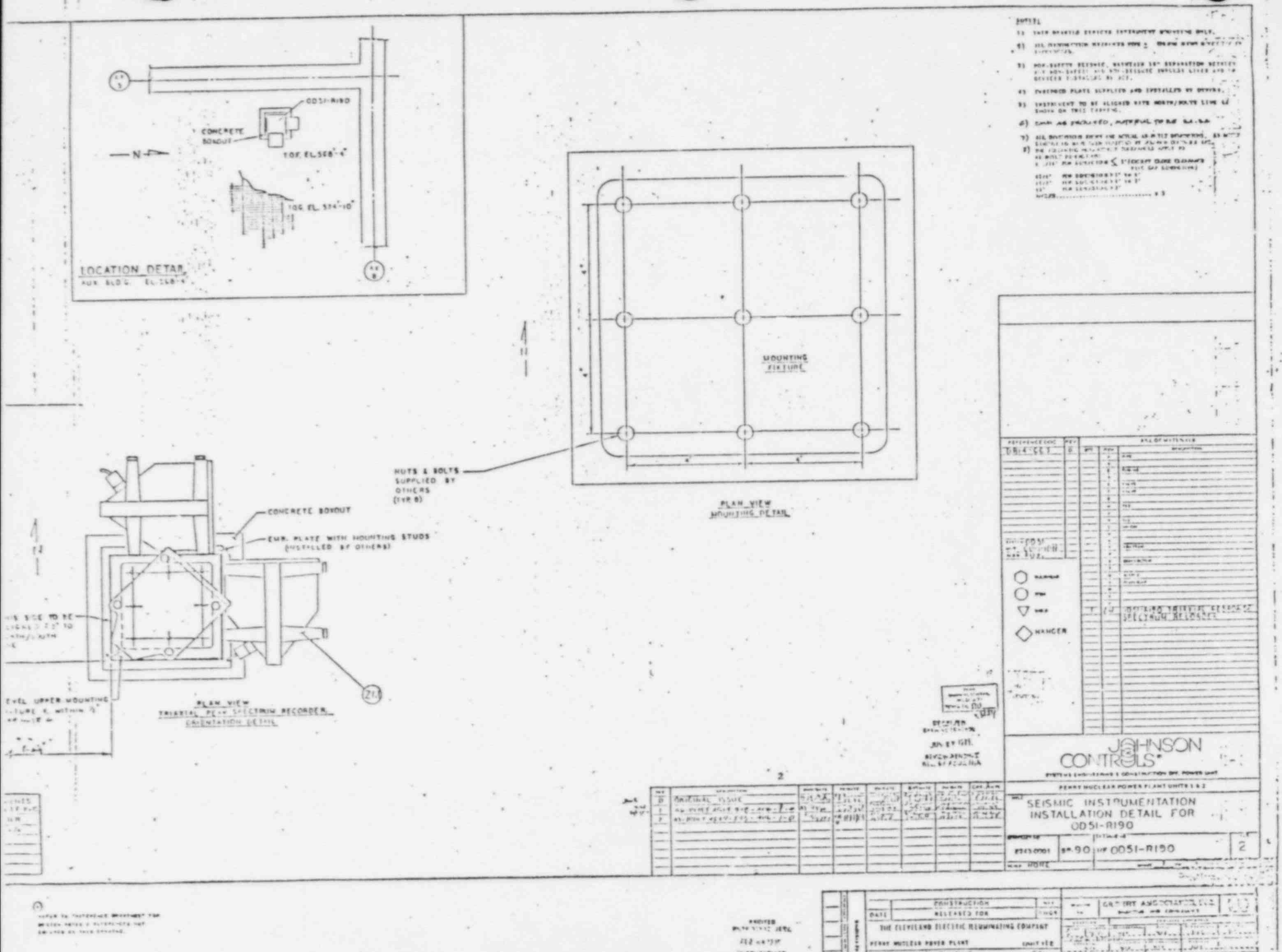
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

PERRY NUCLEAR POWER PLANT

REACTOR BUILDING



LOCATION DETAIL
AUX. BLDG. EL. 568'-4"
(FOR REFERENCE ONLY)
TOG. EL. 574'-10"
TOF. EL. 568'-4"
0051-R180
CONCRETE BOXOUT & EMBEDDED PLATE
N
MOUNTING FIXTURE
4" CC
PLAN VIEW
MOUNTING DETAIL
NUTS & BOLTS SUPPLIED BY OTHERS (TYP. 8)
CONCRETE BOXOUT
EMB. PLATE WITH MOUNTING STUDS (INSTALLED BY OTHERS)
0051-R180
ZL 0051-R180
PLAN VIEW
TRIAXIAL PEAK SPECTRUM RECORDER ORIENTATION DETAIL
THE FOLLOWING DOCUMENTS ARE APPLICABLE TO THIS I/P PACKAGE
ECN FCN NR DSR
N/A N/A N/A N/A
ORIGINAL ISSUE
HINGER
JOHNSON CONTROLS
PERRY NUCLEAR POWER PLANT UNITS 1 & 2
SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR 0051-R180
SP-90
0051-R180
CONSTRUCTION
RELEASED FOR
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
PERRY NUCLEAR POWER PLANT
GILBERT ASSOCIATES


LOCATION DETAIL
AUX. BLDG.
0D51-R190
CONCRETE BOXOUT
TOF EL. 568'-6"
TOC EL. 574'-10"
N
MOUNTING FIXTURE
PLAN VIEW
MOUNTING DETAIL
NUTS & BOLTS SUPPLIED BY OTHERS (TYP 8)
CONCRETE BOXOUT
EMB. PLATE WITH MOUNTING STUDS (INSTALLED BY OTHERS)
21.6
PLAN VIEW
TRIAXIAL PEAK SPECTRUM RECORDER ORIENTATION DETAIL
NOTES:
HANGER
ORIGINAL ISSUE
JOHNSON CONTROLS
PERRY NUCLEAR POWER PLANT UNITS 1 & 2
SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR 0D51-R190
0D51-R190
2
NONE
CONSTRUCTION
RELEASED FOR
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
PERRY NUCLEAR POWER PLANT
GILBERT ASSOCIATES, INC.

6.0 PLANT SEISMIC DESIGN EVALUATION

The seismic design basis for the Perry Nuclear Power Plant is established by requirements in 10 CFR Part 100, Appendix A and NRC Regulatory Guide 1.60. These regulations require nuclear plant structures and safety class systems and components to be designed to withstand loads induced by a "Safe Shutdown Earthquake" (SSE) for the particular site. The SSE is the strongest earthquake in terms of magnitude of vibratory ground motion that is ever expected to occur at a particular site. The SSE is the design basis earthquake considered for plant licensing. A second seismic event also considered in designing nuclear plants is the "Operating Basis Earthquake" (OBE). The OBE is the strongest earthquake considered likely to occur at a particular site and is at least one-half of the SSE. Operations may resume following an earthquake which exceeds the OBE after demonstrating that no functional damage has occurred to safety-related plant features. (10 CFR Part 100, Appendix A, III(c), V(a)).

The SSE can be described by means of a "response spectrum," which depicts the maximum acceleration, velocity or displacement response to an input excitation (here the SSE) at a specified damping value for single degree-of-freedom oscillators of varying natural frequencies. The high frequency end of a response spectrum indicates the "zero period acceleration" (ZPA) associated with the event. The ZPA is equal to the maximum ground acceleration of the SSE itself.

In the design of any plant, it is difficult to predict the exact shape of postulated earthquake acceleration time-histories and associated ground response spectra. Appendix A of 10 CFR Part 100 therefore requires an expected SSE to be developed by statistically combining the response spectra from multiple historical earthquakes. Following this guideline, the NRC has provided in Reg. Guide 1.60 standardized response spectra that can be used in lieu of spectra developed for each site (see Fig. 6.1). These standardized spectra were derived by normalizing and combining spectra calculated from

numerous sets of historically recorded acceleration time-histories. From these sets of spectra, smoothed response curves (acceleration, velocity and displacement) were generated at a level equal to one standard deviation greater than the mean of the responses. This method provides an 84% level of statistical confidence that responses at any particular frequency will not be exceeded by any future event.

Thus, in lieu of having to develop site-specific SSE ground response spectra, the standardized response spectra of Reg. Guide 1.60 can be used. The standardized spectra need only be scaled up or down to reflect the effective maximum ground accelerations (i.e., ZPA's) expected for the SSE at that site. The SSE design response spectra are used to dynamically analyze a lumped-mass model of the power plant structures.

6.1 DESIGN OF THE PERRY PLANT

The Perry design response spectra were derived by using the standard response spectra of Reg. Guide 1.60 scaled to a ZPA of 0.15 g determined for the Perry site. These spectra served as the design response spectra at the foundation elevations for use in designing the plant buildings.

From these spectra, a simulated SSE time-history of ground accelerations was developed for each directional component (N-S, E-W, and vertical). The conservatism of these simulated time-histories was checked and confirmed by assuring that the response spectra generated from the simulated time-histories envelop the Reg. Guide 1.60 design response spectra (see Fig. 6.2).

Seismic Category I structures were analyzed by applying the simulated time-histories to a lumped-mass model of the entire structure, as shown in Figure 6.3. From this analysis, time-history accelerations at each floor elevation were also derived. These time-histories were then used to derive response spectra for each floor of each main building. The floor response spectra were used in designing the safety class equipment, components, and systems.

In addition to the conservatism included in the derivation of response spectra, there were numerous other conservatisms included in the overall structural design of the Perry structures, systems and components. Examples of some of the more significant conservatisms are as follows:

1. Broadening the Envelope of Floor Response Spectra

   Frequency bands of floor responses spectra were artifically broadened (typically by 15%) to account for possible frequency variations. Responses used for design were thus overestimated for systems having more than one dominant frequency falling into the broadened frequency bands of the floor response spectra.

2. Equipment Qualification by Test

   Equipment qualified by shake table testing used time-histories simulated from the floor response spectra. The simulated time-histories were generated in such a way that their calculated response spectra envelop the broadened floor response spectra, which in turn already envelop the original design response spectra. The conservatism of the time-histories is increased by this "envelope on top of an envelope" process. Moreover, this process results in simulated time-histories with maximum accelerations much higher than the ZPA's of the floor response spectra.

3. Strain Hardening Not Accounted For and Static Allowables Used for Dynamic Load

   In equipment design, material is assumed to behave linearly up to the yield point, then to deform continuously to collapse when the

external load is maintained. All material used in equipment design exhibits characteristics of strain hardening. This means that resistance to deformation increases after the deformation exceeds the yield point. Furthermore, even if no strain hardening is assumed, the material can resist dynamic loads having peak values higher than the yield strength through the absorption of energy in the plastic region.

4. Loading Combinations

The plant was designed to withstand loading combinations with a very low probability of simultaneous occurrence. For example, some load combinations included seismic loads, hydrodynamic loads, and loads due to a hypothetical loss of coolant accident. This results in design capability well above the loads associated with seismic alone.

5. Allowable Stresses

Computed seismic stresses used in design were considered to be primary, non-self-limiting stresses instead of secondary stresses with a self-limiting nature. The actual behavior of seismic stresses is somewhere between a primary and secondary nature. Consideration of seismic stresses as primary stresses results in overestimated values used for design.

6. Damping Values

Conservative damping values were employed at Perry pursuant to NRC Regulatory Guide 1.61. The recent ASME code case N-411 allows increased damping values to be used in the design of nuclear power plant piping systems.

One example of just how significant these types of design conservatisms are is the response of the El Centro Steam Plant (in California) to the 1979 Imperial Valley earthquake. The El Centro Steam Plant was designed to withstand a 0.2 g static lateral load. The recorded peak horizontal load at the site was 0.5 g. The station tripped when station power was lost. One unit was restored to service in 15 minutes and another one in 2 hours. According to calcuations performed by Lawrence Livermore Laboratories, the actual loads experienced by the plant were 2 to 9 times higher than the design values. The plant, however, suffered essentially no damage. The El Centro case shows that an engineered structure can indeed resist seismic loads many times higher than their design values.

6.2 EVALUATION OF THE JANUARY 31 EARTHQUAKE

The USGS determined the magnitude of the January 31, 1986 earthquake to be $M_b = 4.9$ with an epicenter at about 11 miles (17.6 Km.) south of the Perry Power Plant site. This is of much less magnitude than the earthquake for which the plant was designed (the SSE) and contained substantially lower total energy than the Perry SSE. Evidence of the low energy content of the January 31 earthquake is shown by a comparison of the acceleration time-histories it induced at various elevations with the corresponding design acceleration time-histories. (See Figs. 6.4 through 6.9). The time-histories used for design are 22 seconds long and of sustained high magnitude (strong motion). By contrast, the January 31 time-histories are about 5 seconds long and contain strong motion in only less than a one-second interval (total) of the event.

A comparison of Figures 6.1 and 6.10 gives a further indication of the low energy content of the January 31 earthquake. These figures show that the Reg. Guide 1.60 spectra used for design have much broader frequency contents than those of the recorded earthquake, which contain strong motion only at high frequencies. The design earthquake therefore contains much greater total energy.

The maximum relative displacements from the recorded time-histories of the recorded earthquake are shown in Table 6.1. A comparison of the total square-root-of-the-sum-of-the-squares (SRSS) recorded relative displacements with the SSE and OBE values shows that the recorded displacements were all far below those values. For example, the overall relative displacement shown in the Table is 0.36 cm for the SSE and 0.10 cm for the actual event. Since stresses in the structures are proportional to relative displacements, and the recorded relative displacements were far less than the SSE design values, the stresses induced by the 1986 earthquake were all well within design capabilities.

Table 6.2 compares the structural response ZPA's of the recorded data with those of the SSE and OBE. The SRSS comparison indicates that the recorded values of the 1986 earthquake vary from significantly below OBE values to 74% of SSE values, except at elevation 686 feet of the Reactor Building Containment Vessel. At that location, the N-S and Vertical acceleration components exceed SSE values, while the E-W acceleration component is less than the SSE value. However, the recorded relative displacements are far less than their design values, as shown in Table 6-1. In addition, recorded response spectra accelerations show that the design response spectra accelerations in certain instances were exceeded at the high frequency end of the spectra. At lower frequencies (at or below approximately 14 Hz) the recorded accelerations are all well under the design values (see response spectra comparisons in Appendix D).

The measurement of accelerations outside the predicted responses at the high frequency ends of certain response spectra has no engineering significance. This is explained by the interrelationships among the frequencies, accelerations, velocities, and displacements associated with a seismic event. In general, high frequency acceleration responses have correspondingly low velocity and displacement responses. The 1986 earthquake accelerations occurred at very high frequencies. Therefore, despite some recorded maximum

acceleration responses which exceeded SSE values at higher frequencies, corresponding velocities and displacements (and resulting stresses) were nevertheless acceptably low.

As discussed, the significant indicators of structural stresses are the relative displacements, and Table 6.1 indicates that relative displacements (and thus stresses) caused by the 1986 earthquake were very small. This is consistent with the high frequency nature of the disturbance. The high frequencies combined with the short duration resulted in an earthquake that contained very low total energy compared to the SSE.

The maximum recorded velocity at the top of the Reactor Building foundation mat during the 1986 earthquake was 0.87 inches/sec (2.21 cm/sec). This can be compared with the Bureau of Mines (BOM) velocity threshold for no damage to non-engineered buildings, which is 1 inch/sec (2.54 cm/sec). This shows that the BOM considers it acceptable for blasting work to induce velocity waves in nearby residential housing foundations that are greater than the maximum velocities induced by the 1986 earthquake at the Perry Plant. This example helps provide perspective on just how low the velocities and energy content of the 1986 event were.

As discussed earlier in this report, extensive plant inspections have indicated that no structural damage resulted form the 1986 earthquake. This is as expected based upon the low energy, short duration, and low velocity and displacemnt of the event. Although some hairline cracks in the structural concrete were documented during plant walkdowns, this does not constitute damage. Reinforced concrete structures are expected to show hairline cracks. Regardless of their cause, such cracks have no effect on the strength and integrity of the structures. Moreover, such cracking is judged not to be attributable to the 1986 earthquake because of the low magnitude of the event.

Section 7.5 of IEEE 344 "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," was employed at Perry. This standard recognizes that short duration/high frequency/low energy input motions will not cause significant structural stresses. Instead, it requires qualification by long duration/broad band frequency/high energy testing to provide conservatism.

As discussed earlier in this report, all energized plant equipment functioned during this event as designed. To confirm the lack of impact of the high frequency accelerations on plant equipment, CEI is comparing the qualification data for equipment listed in Table 6.3 against recorded response spectra. Although still ongoing, the evaluation to date shows that the original conservatism in the equipment qualification was more than adequate to accomodate the recorded event.

6.3 EVALUATION OF SPECIFIC DATA

In light of the above discussion, recorded responses at particular locations can be evaluated. At all four instrument locations recording response spectra, SSE design spectra are all well above the recorded spectra in the frequency range of 1 Hz to 14 Hz (see Figs. D1 through D12). These figures compare recorded data with the appropriate design spectra at adjacent elevations. These figures also compare the data from different types of seismic instrumentation at the same elevation.

At higher frequencies, the design spectra are exceeded by recorded values in certain cases. However, the corresponding displacements based on recorded data are all extremely small (on the order of several one-hundredths of an inch) at 20 Hz. These extemely low displacements conform to the above analysis demonstrating that the stresses at higher frequencies are insignificant despite acceleration exceedences.

In evaluating all the spectra data recorded at the various locations, it was noted that the acceleration responses at the Reactor Building Platform outside the Biological Shield Wall varied from the general pattern of responses recorded at the other three locations. The recorded N-S and E-W acceleration components for this location are all well-enveloped by the entire range of the SSE spectra while the recorded vertical acceleration component exceeds the SSE spectra at the high frequency end (see Figure D-9). This response may be due to the fact that this particular Engdahl PSR-1200 instrument is located near multiple supports and piping system snubbers and components. Actuation of snubbers or local loads induced by nearby components may thus have influenced the recorded vertical response. Such impacts would be of a local, secondary nature. Regardless, the low energy, short duration, high frequency nature of the event indicates that these accelerations had no structural significance. Indeed, the recorded displacement spectrum value is only 0.023 inches (0.06 cm) at 25 Hz at this location.

In general, the high frequency acceleartion content of ground motion will be filtered out by buildings and thus will not appear at higher elevations. This is due in part to the low participation factor generally associated with modes at the higher frequencies. This phenomenon is exhibited by the responses recorded at the Reactor Building mat and elevation 686 feet of the Reactor Building Containment Vessel. A very high frequency p-wave was recorded at the Reactor Building foundation mat. The time-histories shown in Figures 6.4 through 6.9 indicate that this p-wave (appearing during the first second or so of the time-histories) was filtered out by the building and did not appear at elevation 686 feet.

There was a response in the range of 20 Hz that was transmitted to the higher elevations. The explanation for this involves the structural characteristics of the buildings on the Reactor Building foundation mat. The Reactor Building consists of multiple structures sitting on

a common foundation mat--a concrete shield building, steel containment vessel, concrete drywell wall, and biological shield wall. The structural response of each building influences the responses of the others. The frequencies, mode shapes and participation factors of the two most dominant vibration modes are at roughly 4 Hz and 18.4 Hz, as shown in Figures 6.11 through 6.13. These two dominant frequencies correspond to the peaks at 4 Hz and 20 Hz on the recorded spectra for the Reactor Building at the mat and elevation 686 feet. The input motion at 20 Hz (corresponding to the s-wave) was amplified by this latter mode with some rigid body motion. The 20 Hz input was thus not filtered out but did appear at the higher elevation. As discussed, the acceleration peaks at 20 Hz at this location correspond to very small relative displacements and thus are not significant in an engineering sense.

6.4 CONCLUSION

The 1986 Ohio earthquake was a low energy, high frequency, short duration, low velocity, and small displacement event. As a result of these characteristics and the above discussions, the 1986 earthquake had no adverse effects on the Perry structures, systems, or components, and no changes to the Perry seismic design basis are required.

TABLE 6.1

Comparison of Design Displacements[1] VS Recorded Displacements[1]

(Expressed in centimeters / one inch = 2.54 cm)

		COLUMN 1	COLUMN 2	COLUMN 2 minus COLUMN 1
		Reactor Building Foundation Mat Elevation 574'–10" SMA-3 (Kinemetrics) D51-N101	Reactor Building Containment Vessel Elevation 686' SMA-3 (kinemetrics) D51-N111	Relative Displacements for the Containment Vessel
NS	Recorded	0.09	0.17	0.08
	SSE	0.044	0.28	0.24
	OBE	0.023	0.17	0.15
EW	Recorded	0.16	0.21	0.05
	SSE	0.044	0.28	0.24
	OBE	0.023	0.17	0.15
VERT.	Recorded	0.05	0.07	0.02
	SSE	0.02	0.37	0.017
	OBE	0.013	0.022	0.009
SRSS[2]	**Recorded**	—	—	**0.1**
	SSE	—	—	**0.34**
	OBE	—	—	**0.21**

1. Displacements based on same time-step to determine relative displacements
2. Square-root-of-the-sum of the squares

TABLE 6.2

Comparison of Design ZPA's[1] VS Recorded ZPA's

(Expressed in g values)

		Auxiliary Building Foundation Mat Elevation 568' PAR 400 (Engdahl) D51-R140	Reactor Building Foundation Mat Elevation 574'–10" SMA–3 (Kinemetrics) D51-N101	Reactor Building Recirculation Pump Elevation 605' PAR 400 (Engdahl) D51-R120	Reactor Building Platform Elevation 630' Inside Drywell PSR 1200 (Engdahl) D51-R170	Reactor Building Containment Vessel Elevation 686' SMA–3 (Kinemetrics) D51-N111
NS	Recorded	.17	.18	.32	.09	.55
	SSE	.17	.18	1.06	.48	.40
	OBE	.10	.10	.86	.40	.24
EW	Recorded	.06	.10	.11	.16	.18
	SSE	.20	.18	1.06	.48	.40
	OBE	.10	.10	.86	.40	.24
VERT.	Recorded	.03	.11	.05	Note 2	.30
	SSE	.20	.18	.47	.28	.24
	OBE	.10	.10	.38	.16	.15
SRSS[3]	**Recorded**	**.18**	**.23**	**.34**	**Note 2**	**.65**
	SSE[4]	**.33**	**.31**	**1.57**	**.73**	**.62**
	OBE	**.17**	**.17**	**1.27**	**.59**	**.37**

1. Zero period acceleration of structural response
2. ZPA indeterminable from available data
3. Square-root-of-the-sum of the squares
4. Licensing basis is SSE

TABLE 6.3

EQUIPMENT LIST AT AUXILIARY BUILDING ELEVATION 568'

1H22P0001	LPCS	Instrument Rack	
1H22P0017	RCIC	Instrument Rack	
1H22P0018	RHR	Instrument Rack	A
1H22P0021	RHR	Instrument Rack	B
1H22P0055	RHR	Instrument Rack	C
1C61N0001		Differential Press Transmitter	
1E12N0007A,B		Differential Press Transmitter	
1E12N0015A,B,C		Differential Press Transmitter	
1E12N0026A,B		Pressure Transmitter	
1E12N0028		Pressure Transmitter	
1E12N0050A,B		Pressure Transmitter	
1E12N0051A,B		Pressure Transmitter	
1E12N0052A,B,C		Differential Press Transmitter	
1E12N0055A,B,C		Pressure Transmitter	
1E12N0056A,B,C		Pressure Transmitter	
1E12N0058 C		Pressure Transmitter	
1E21N0003		Pressure Transmitter	
1E21N0050		Pressure Transmitter	
1E21N0051		Flow Transmitter	
1E21N0052		Pressure Transmitter	
1E21N0053		Pressure Transmitter	
1E21N0054		Pressure Transmitter	
1E31N0075A		Pressure Transmitter	
1E31N0077A		Pressure Transmitter	
1E31N0083A,B		Pressure Transmitter	
1E51N0003		Differential Press Transmitter	
1E51N0050		Pressure Transmitter	
1E51N0051		Differential Press Transmitter	
1E51N0053		Pressure Transmitter	
1E51N0055A,B,E,F		Pressure Transmitter	
1E51N0056A, E		Pressure Transmitter	
1E12C002A	RHR	Pump & Motor	
1E12C002B	RHR	Pump & Motor	
1E12C002C	RHR	Pump & Motor	
1E21C001	LPCS	Pump & Motor	
1E22C001	HPCS	Pump & Motor	

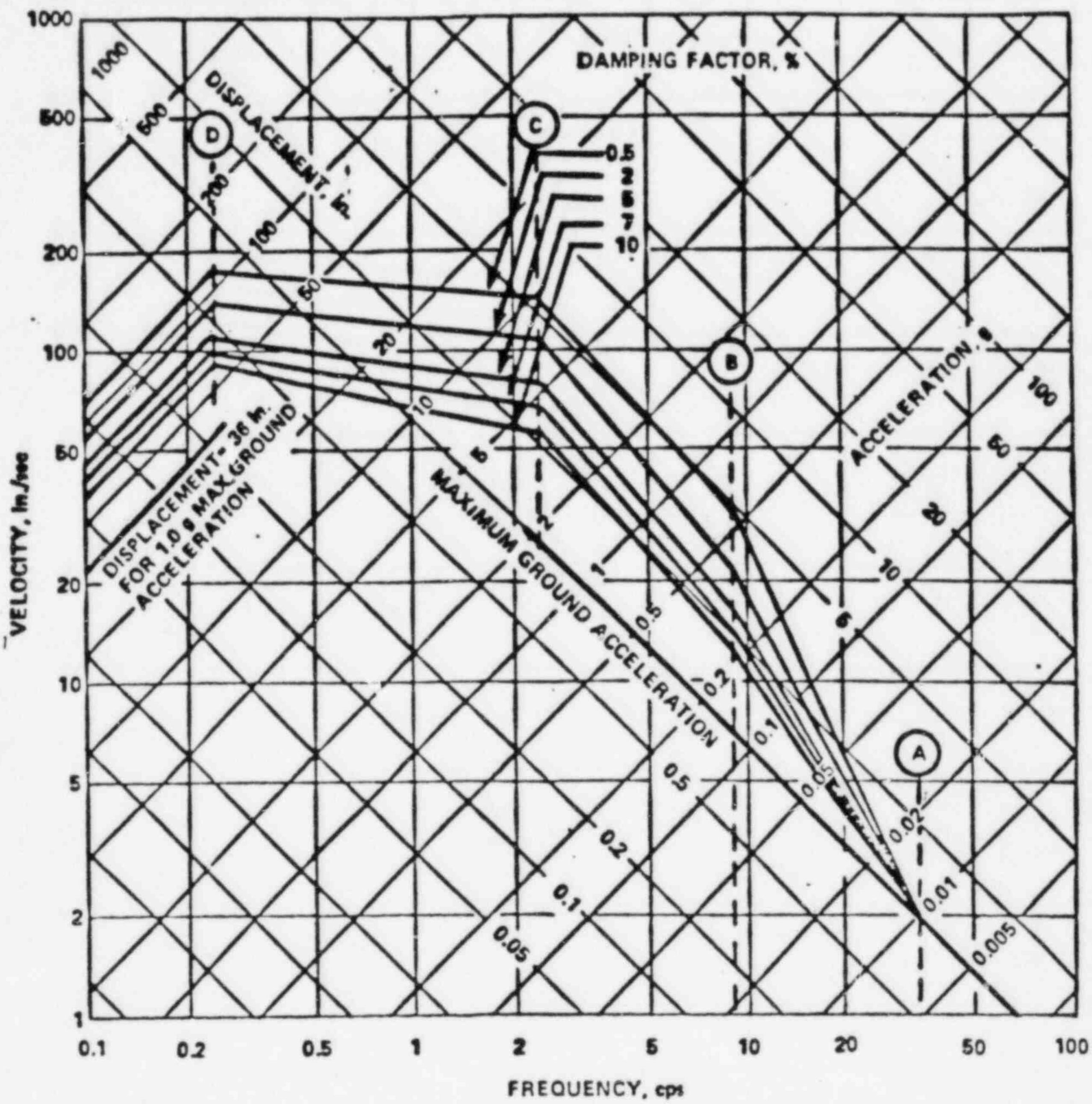
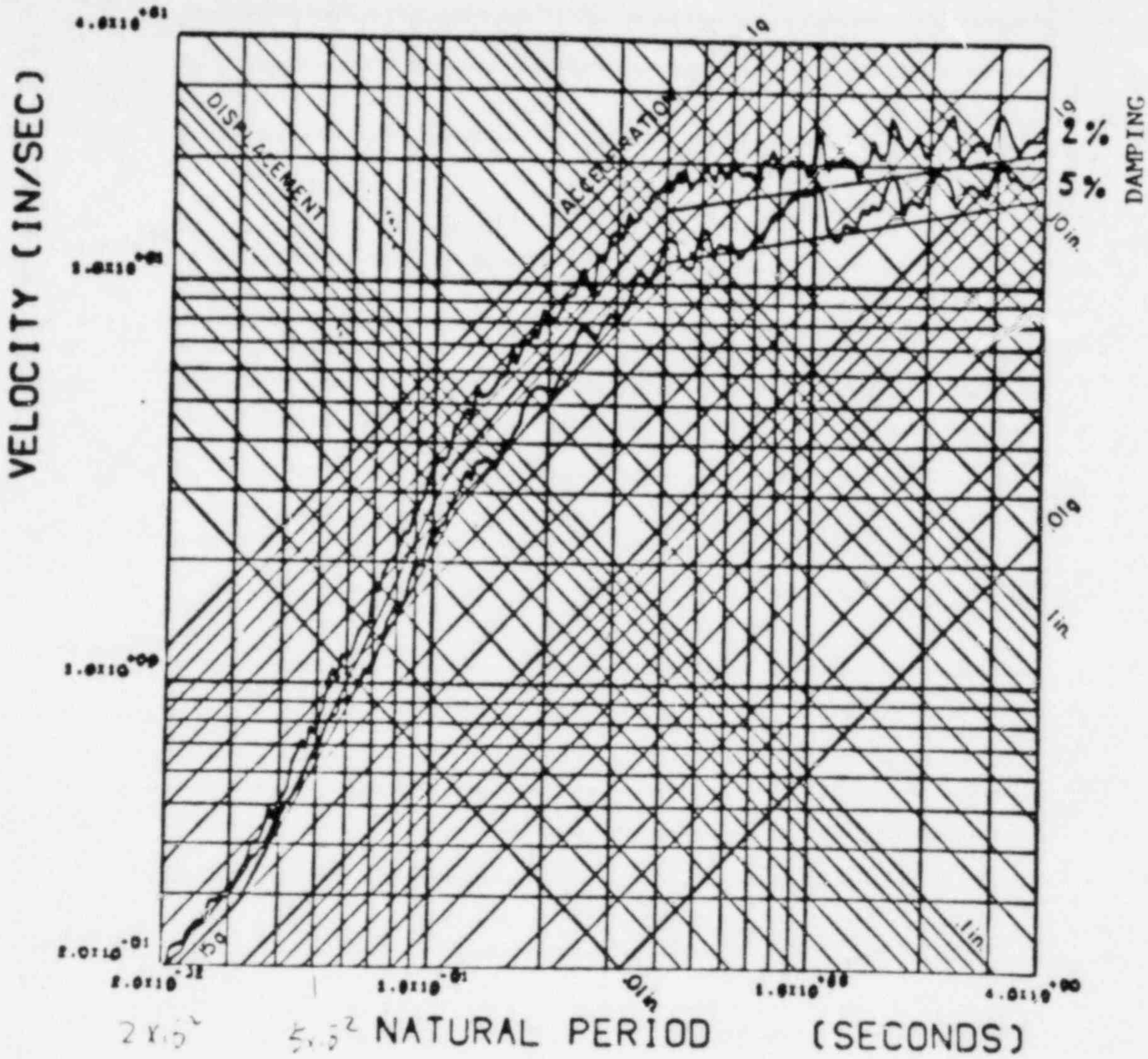


FIGURE 1. HORIZONTAL DESIGN RESPONSE SPECTRA – SCALED TO 1g HORIZONTAL GROUND ACCELERATION

Figure 6.1



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Response Spectra -
Horizontal Motion H1
(2% and 5% Damping)

Figure 3.7-5

Figure 6.2

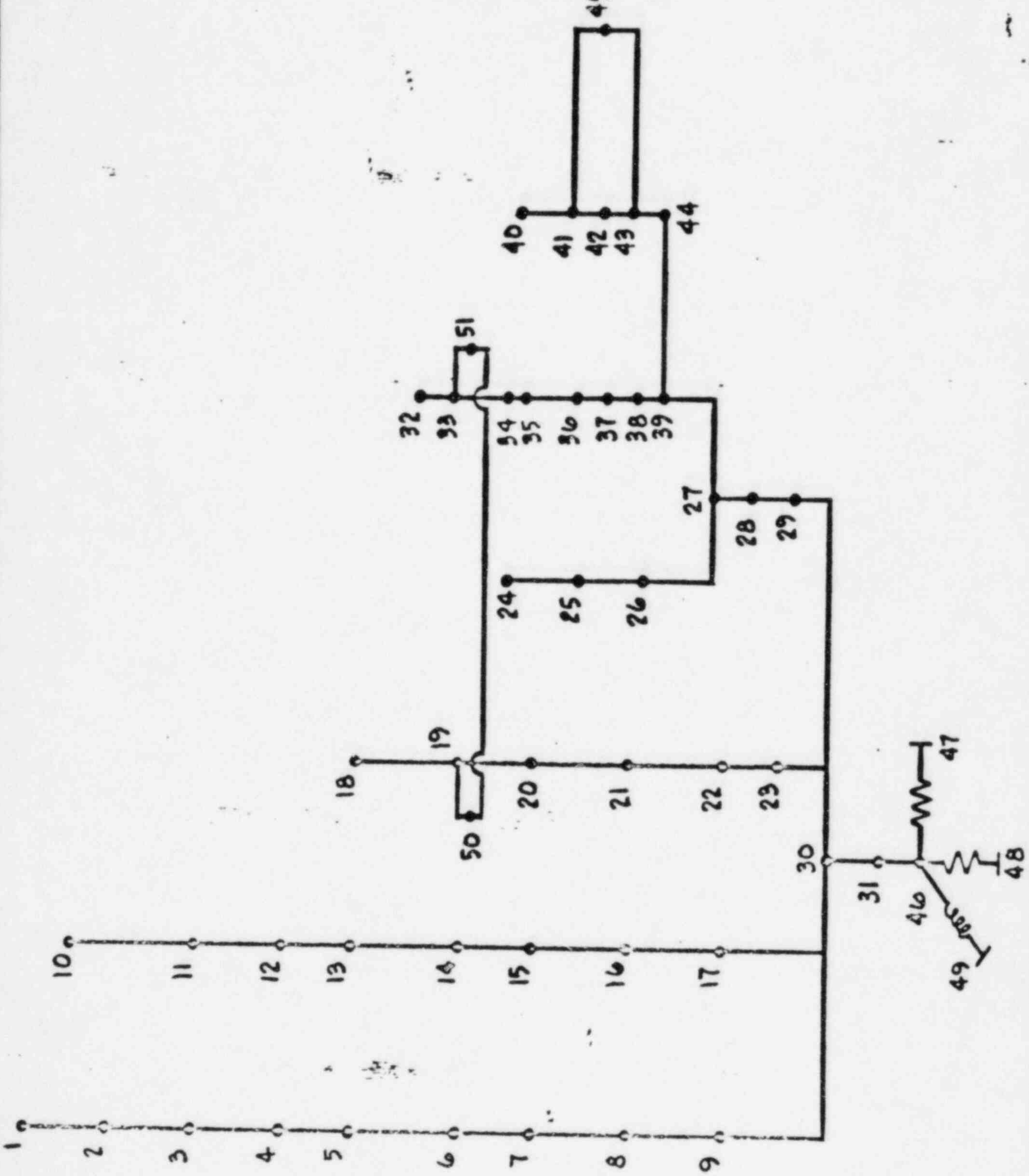

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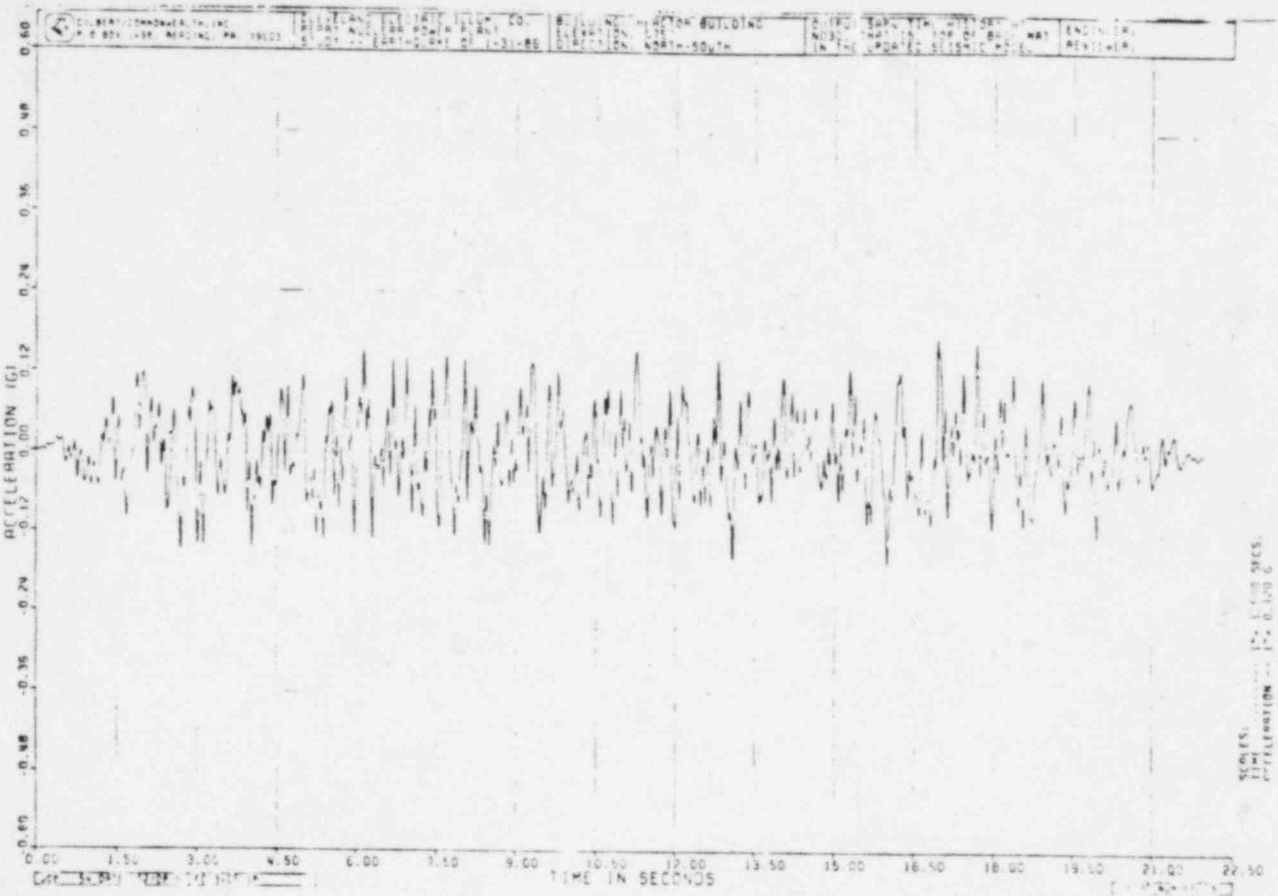
Figure 6.3


ACCELERATION (G)
TIME IN SECONDS
BUILDING: REACTOR BUILDING
DIRECTION: NORTH-SOUTH

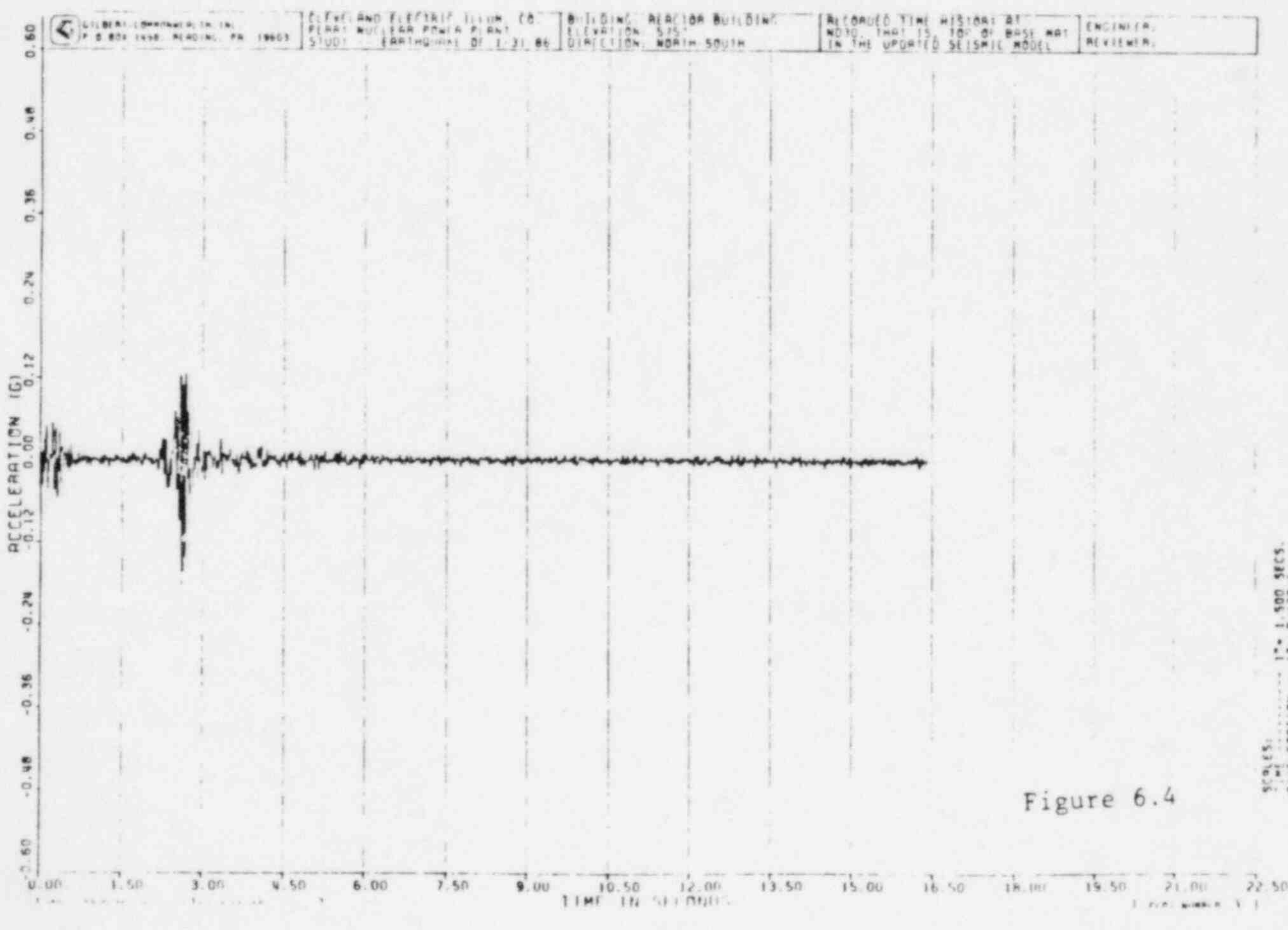
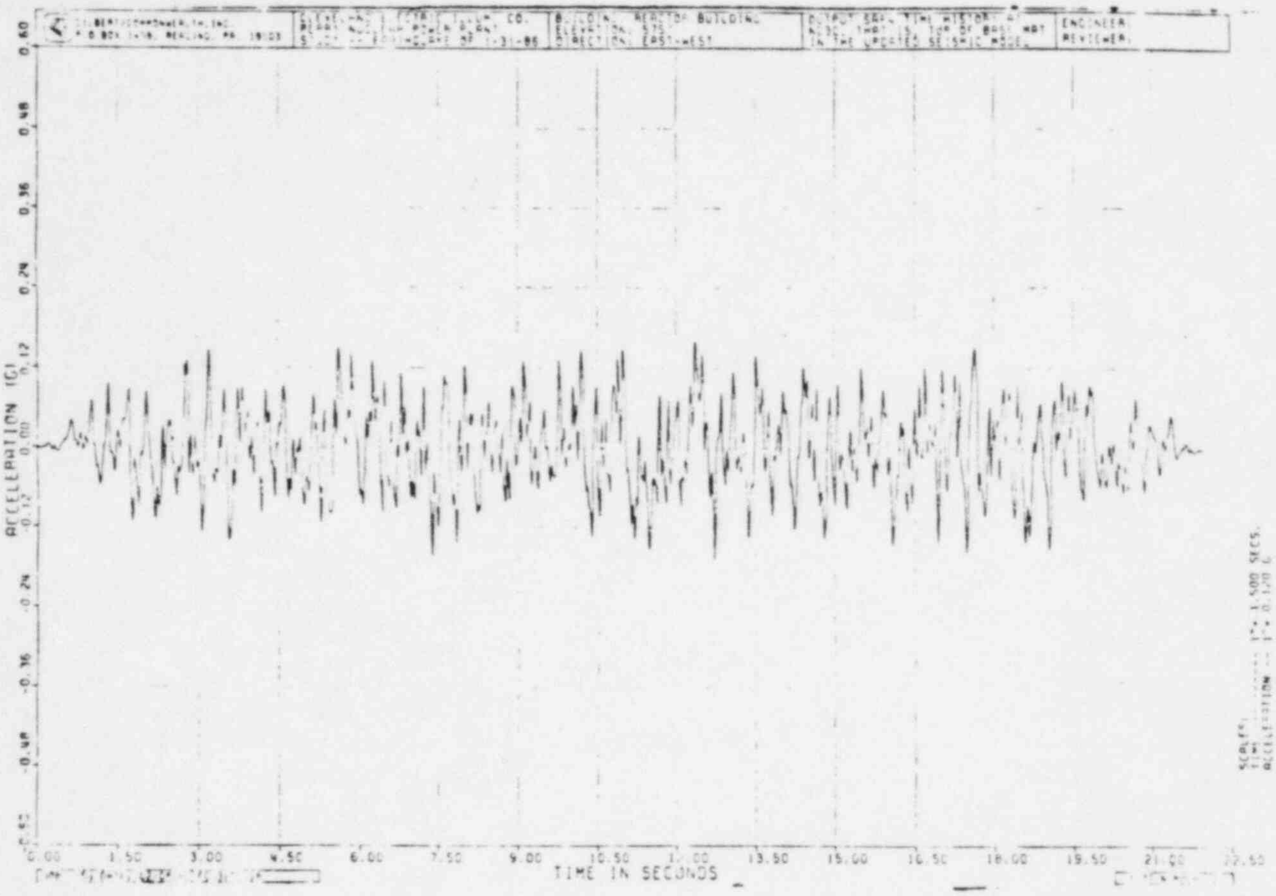

CLEVELAND ELECTRIC ILLUM. CO.
PERRY NUCLEAR POWER PLANT
BUILDING: REACTOR BUILDING
DIRECTION: NORTH-SOUTH
RECORDED TIME HISTORY AT
IN THE UPDATED SEISMIC MODEL
ACCELERATION (G)
TIME IN SECONDS

Figure 6.4



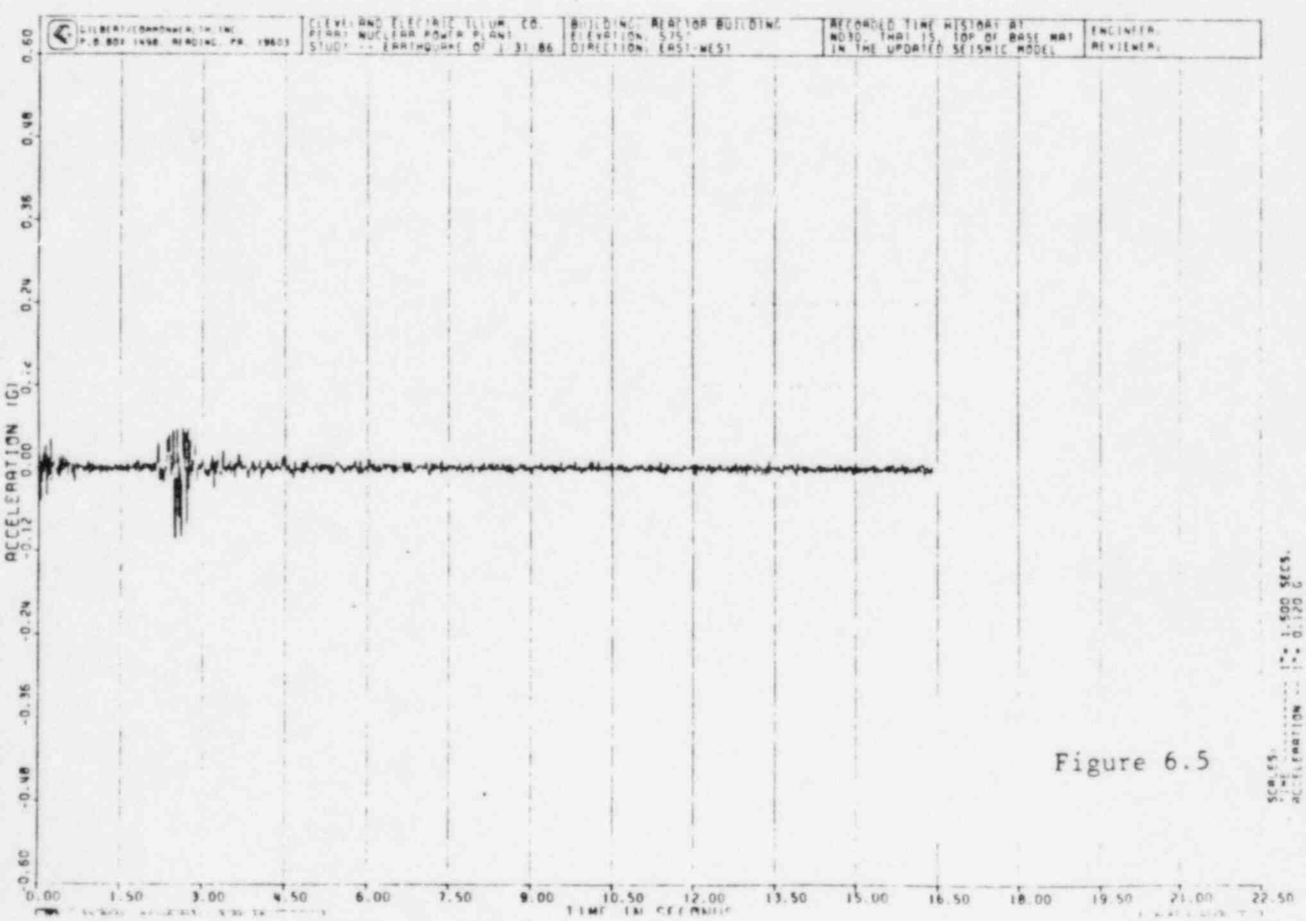
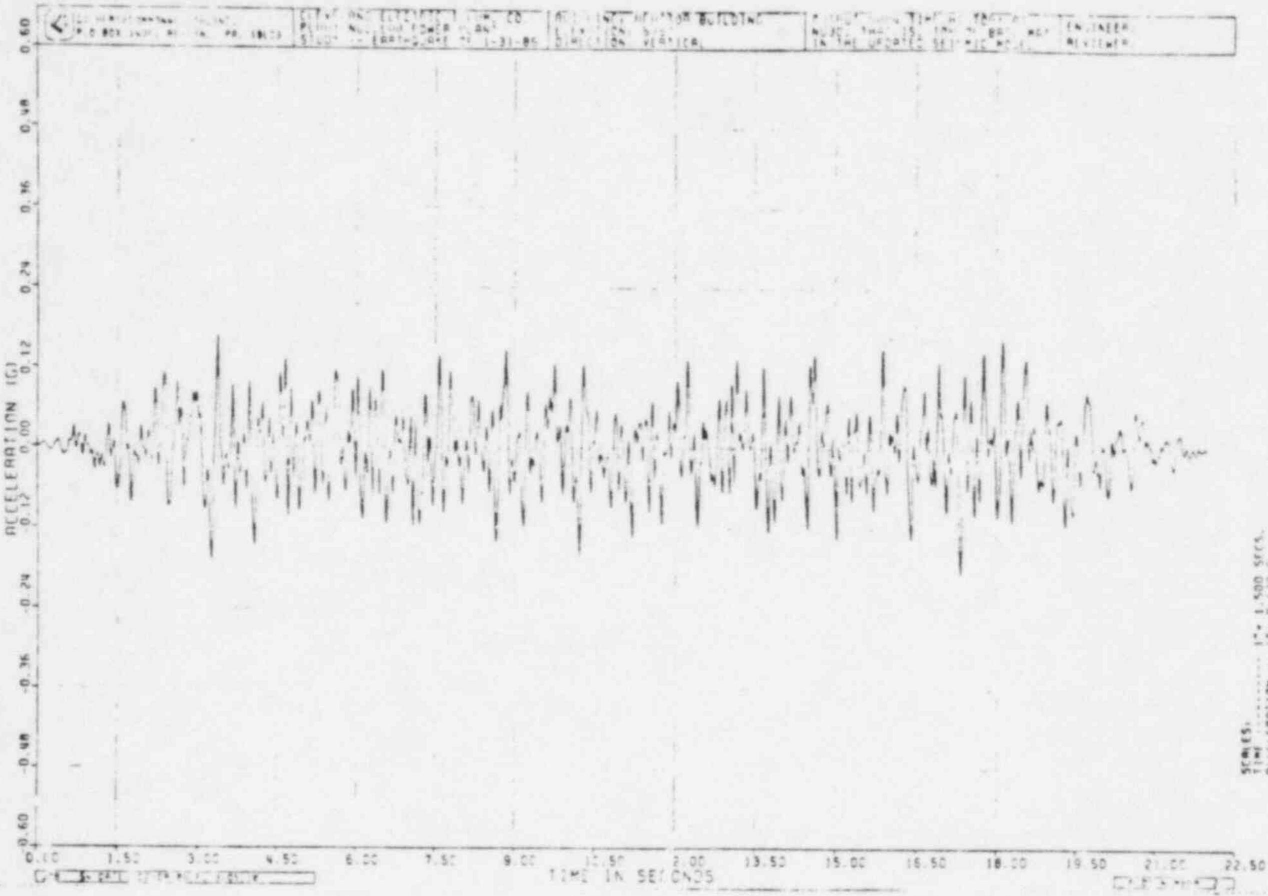


Figure 6.5



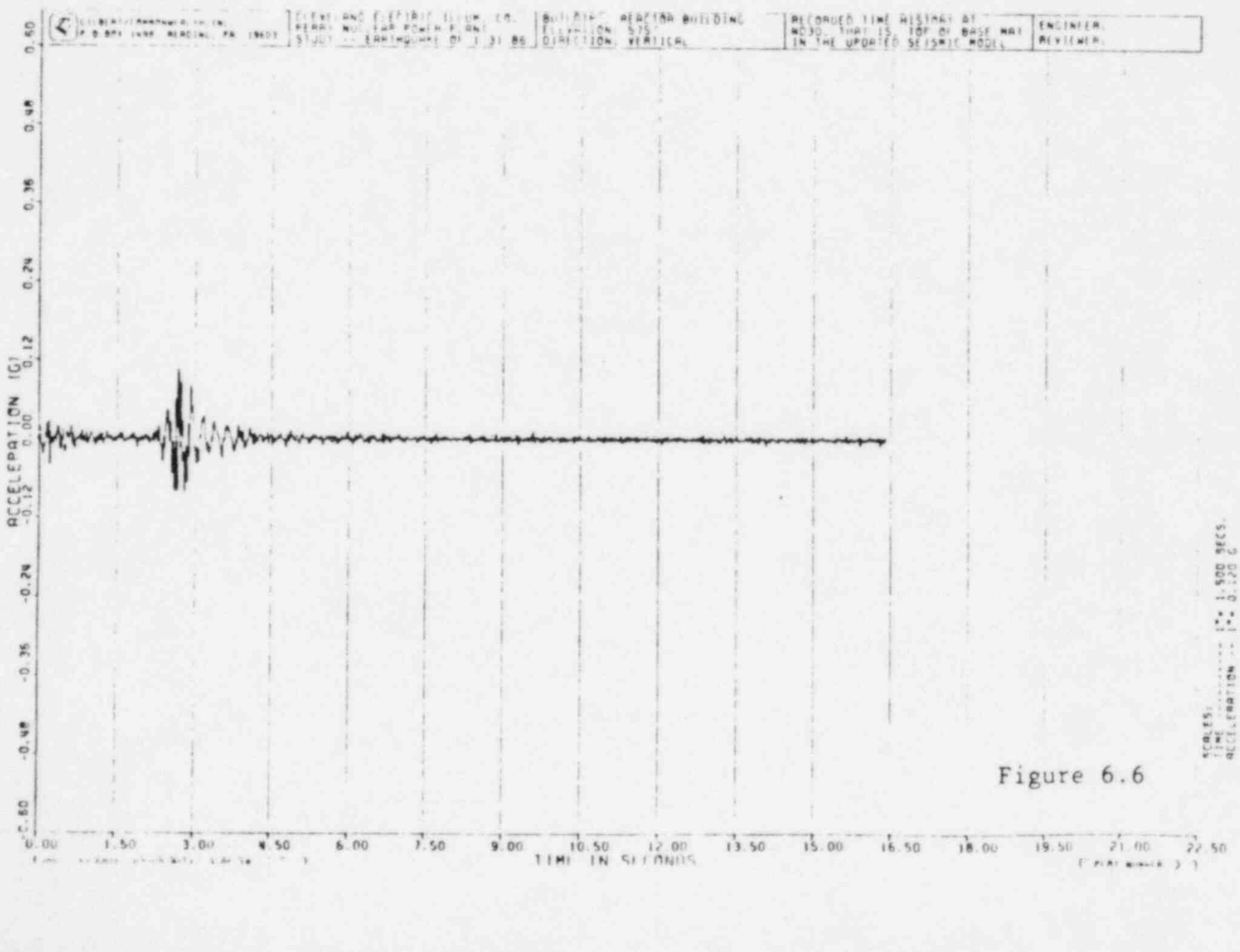
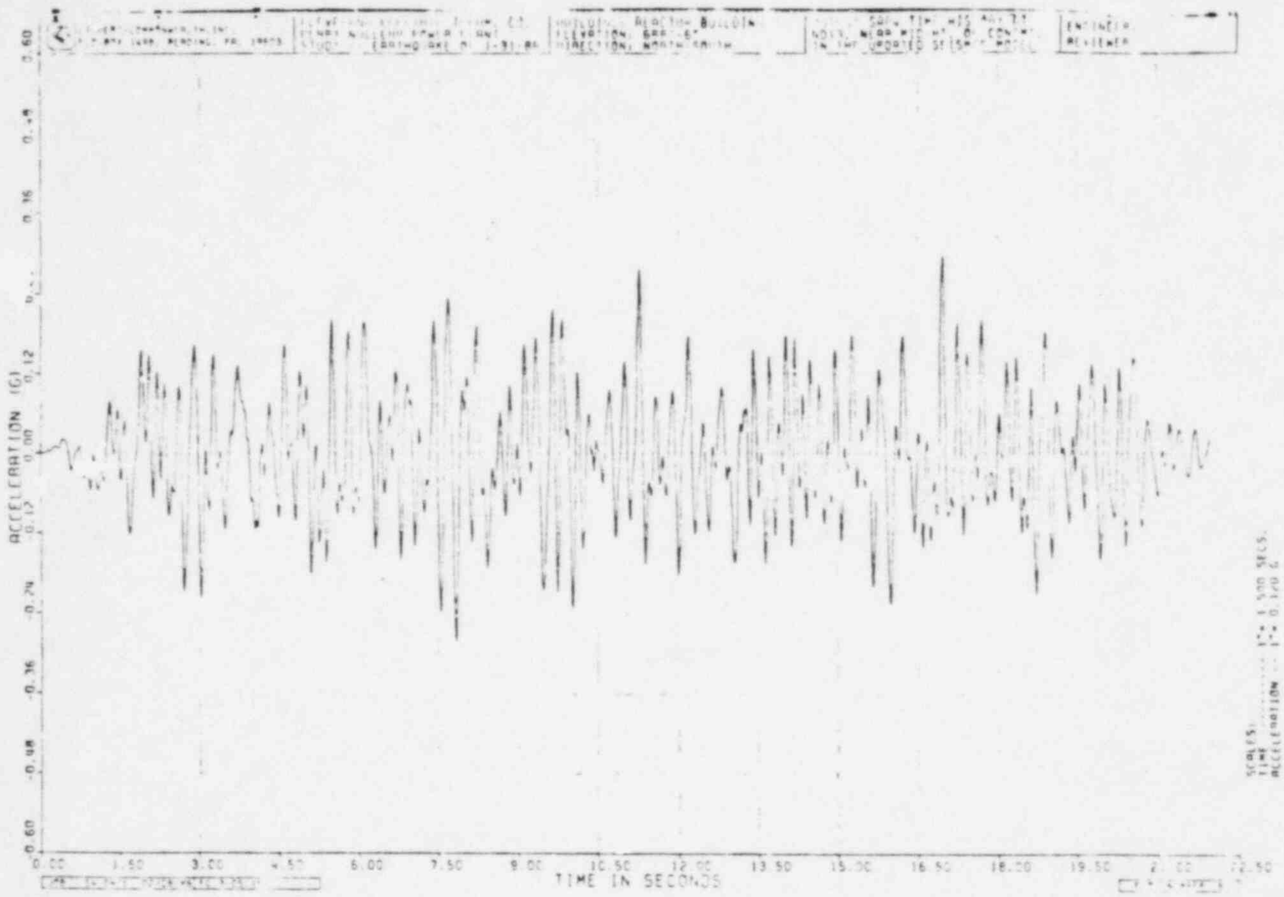


Figure 6.6



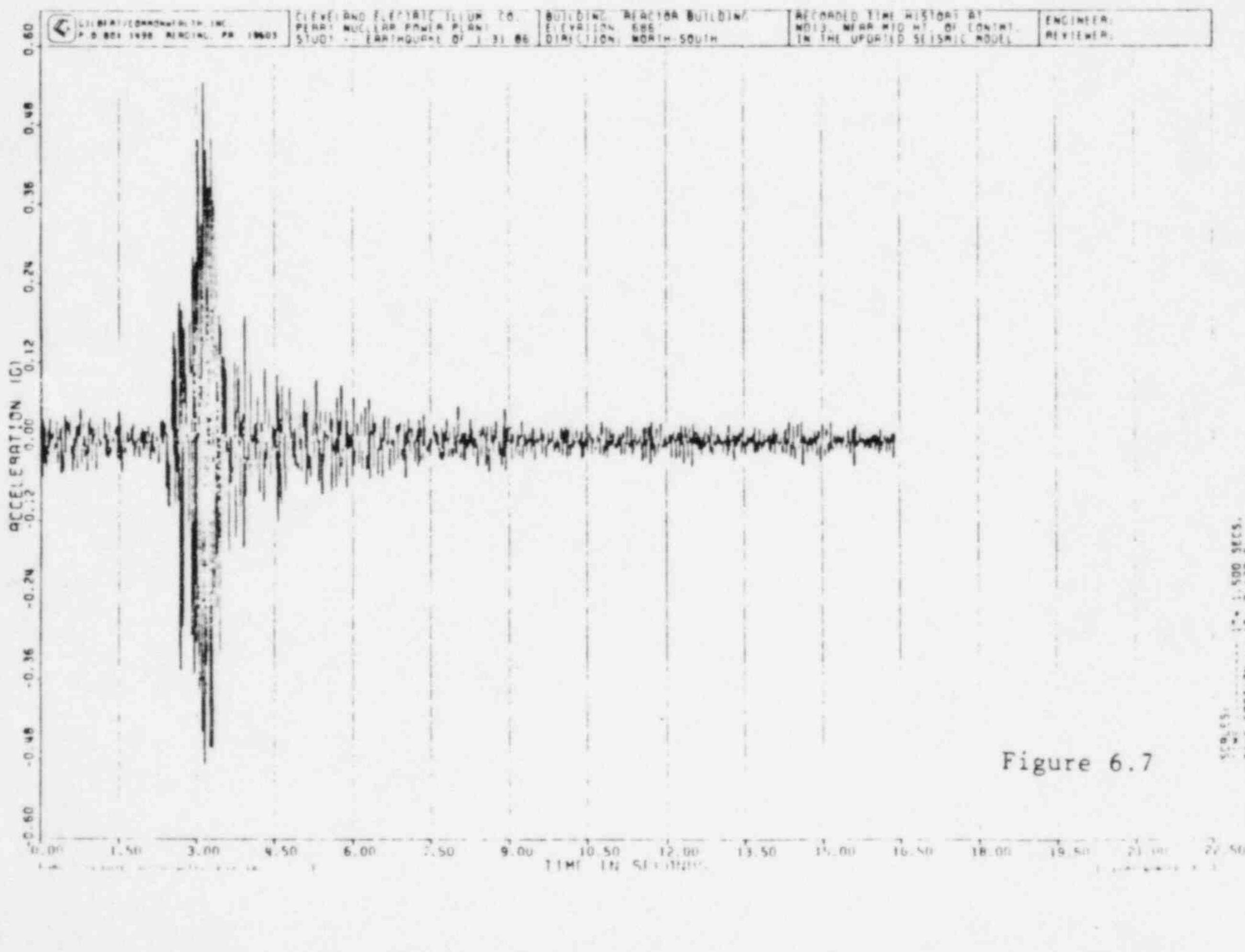
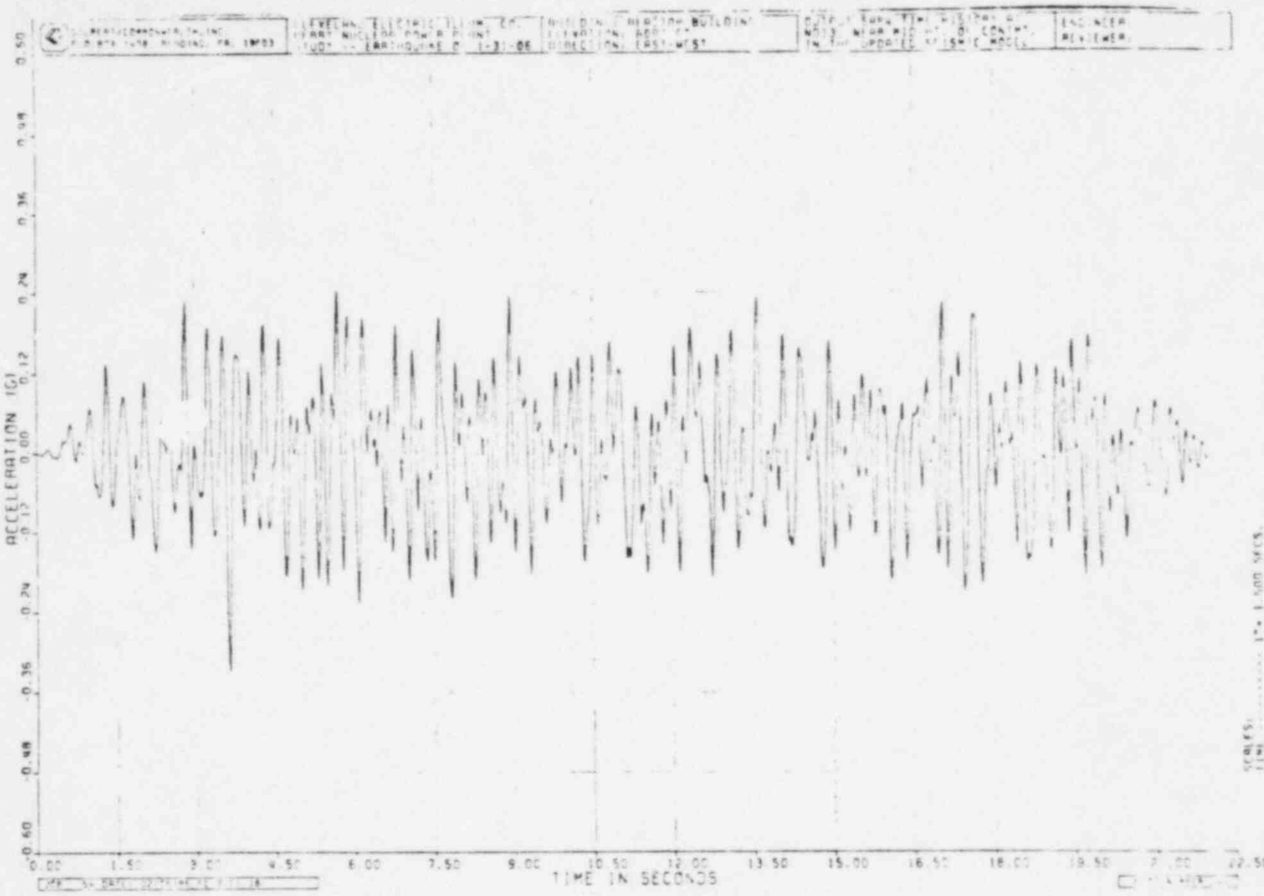


Figure 6.7


CLEVELAND ELECTRIC ILLUM. CO.
PERRY NUCLEAR POWER PLANT
STUDY -- EARTHQUAKE OF 1-31-86
ACCELERATION (G)
TIME IN SECONDS

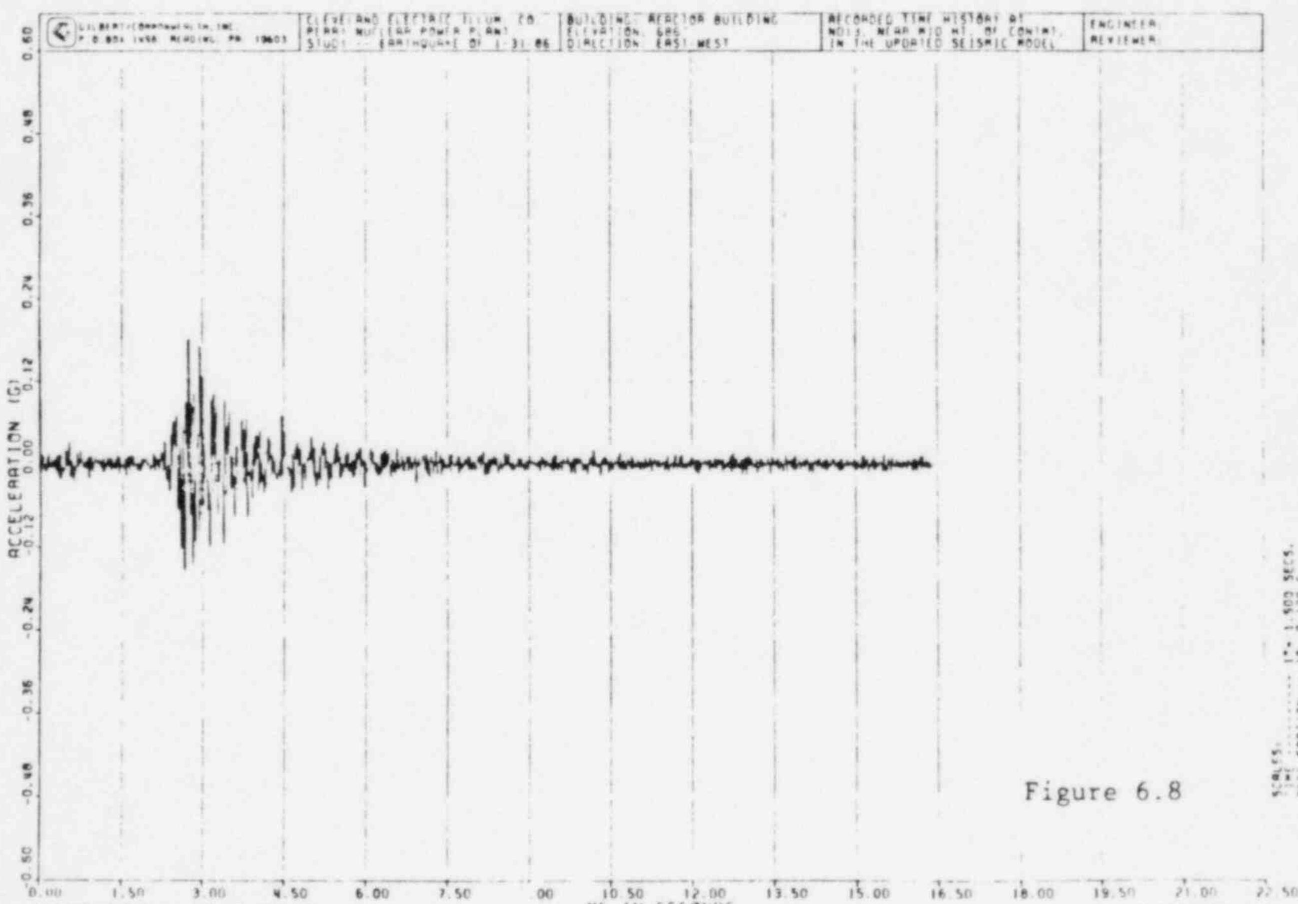
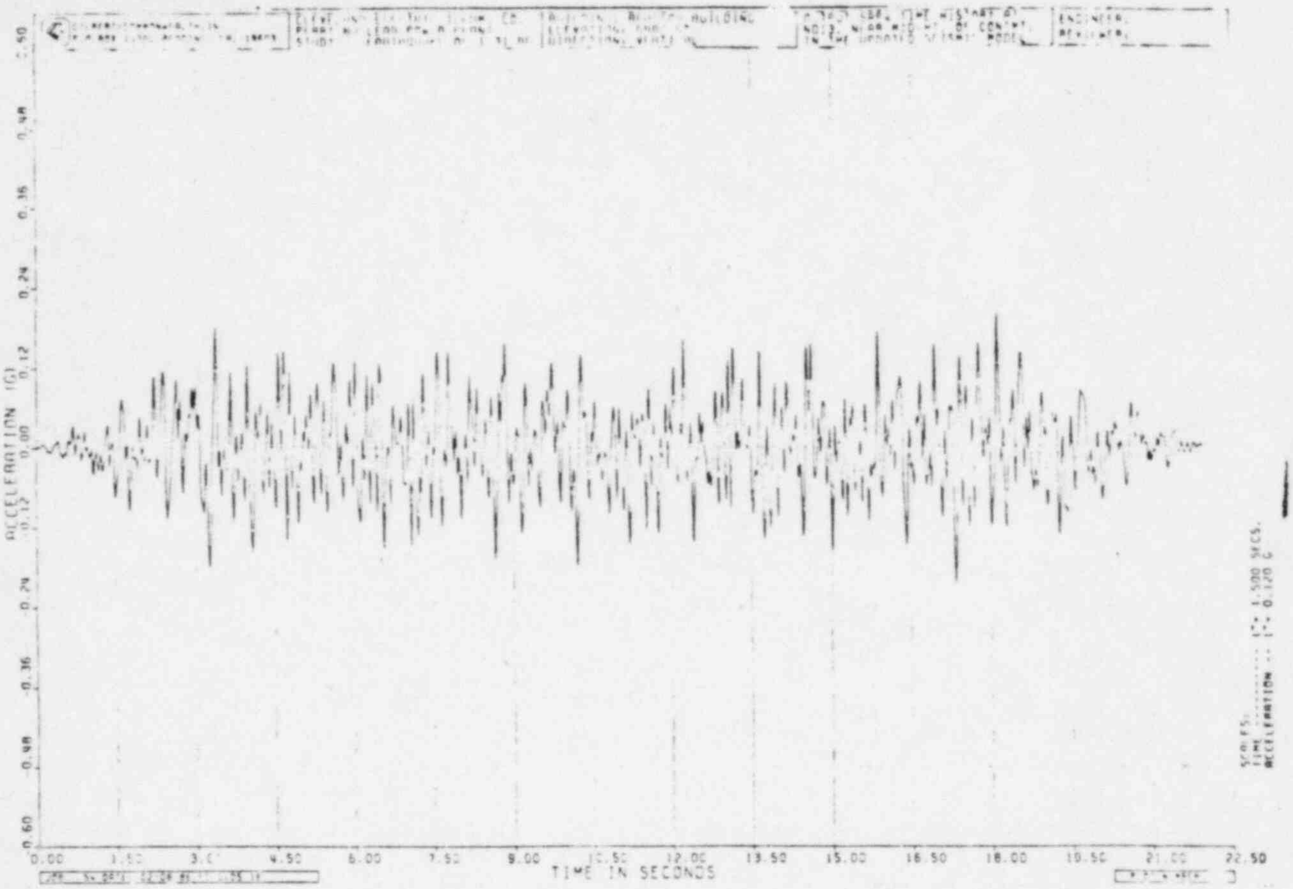

CLEVELAND ELECTRIC ILLUM. CO.
PERRY NUCLEAR POWER PLANT
STUDY -- EARTHQUAKE OF 1-31-86
BUILDING: REACTOR BUILDING
ELEVATION: 686
DIRECTION: EAST-WEST
RECORDED TIME HISTORY AT
NODE 13, NEAR MID HT. OF CONTMT.
IN THE UPDATED SEISMIC MODEL
ENGINEER:
REVIEWER:
ACCELERATION (G)
SCALES:
TIME .... 1"= 1.500 SECS.
ACCELERATION -- 1"= 0.120 G

Figure 6.8


ACCELERATION (G)
TIME IN SECONDS
0.00 1.50 3.00 4.50 6.00 7.50 9.00 10.50 12.00 13.50 15.00 16.50 18.00 19.50 21.00 22.50
SCALES:
TIME .... 1"= 1.500 SECS.
ACCELERATION .. 1"= 0.120 G

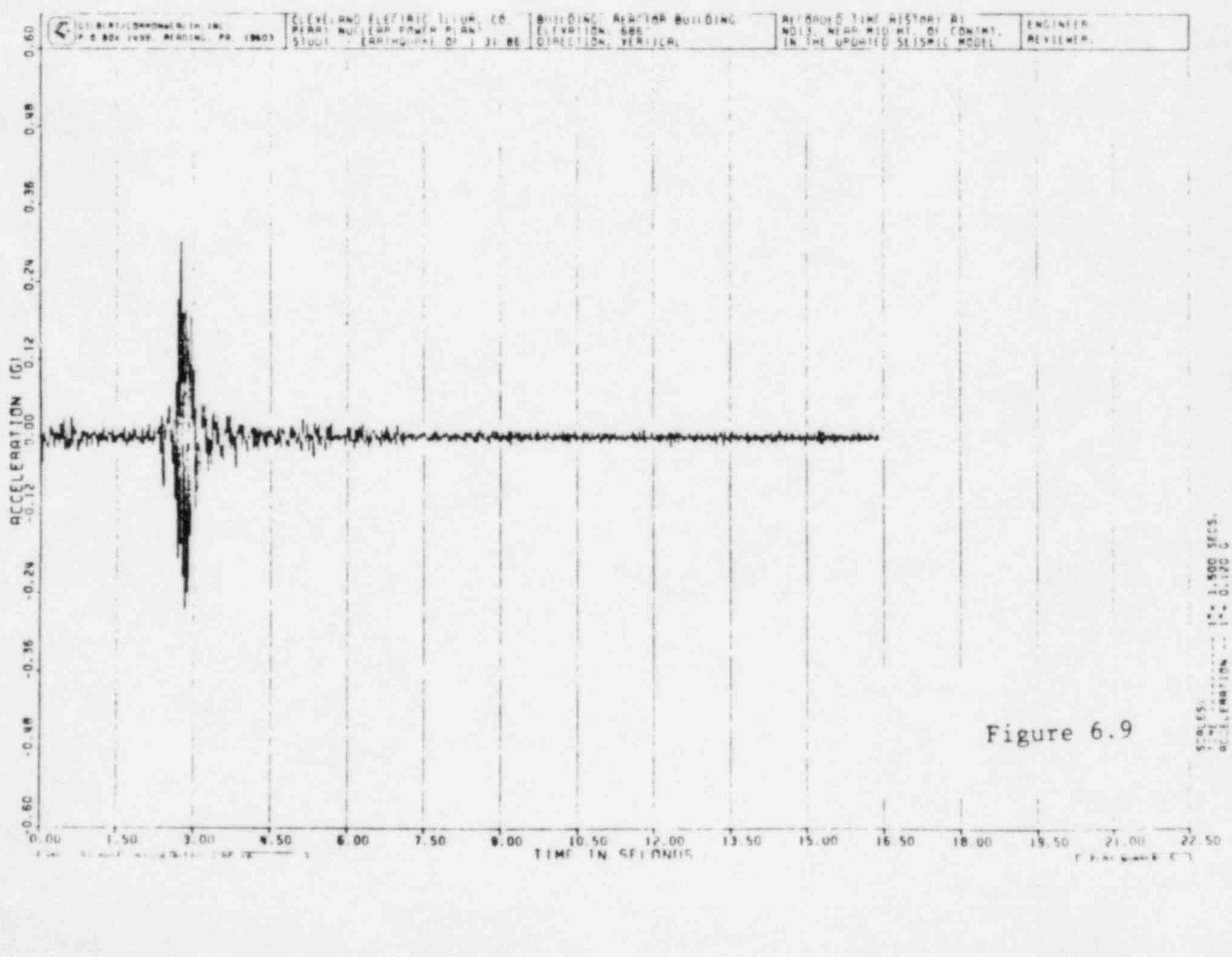
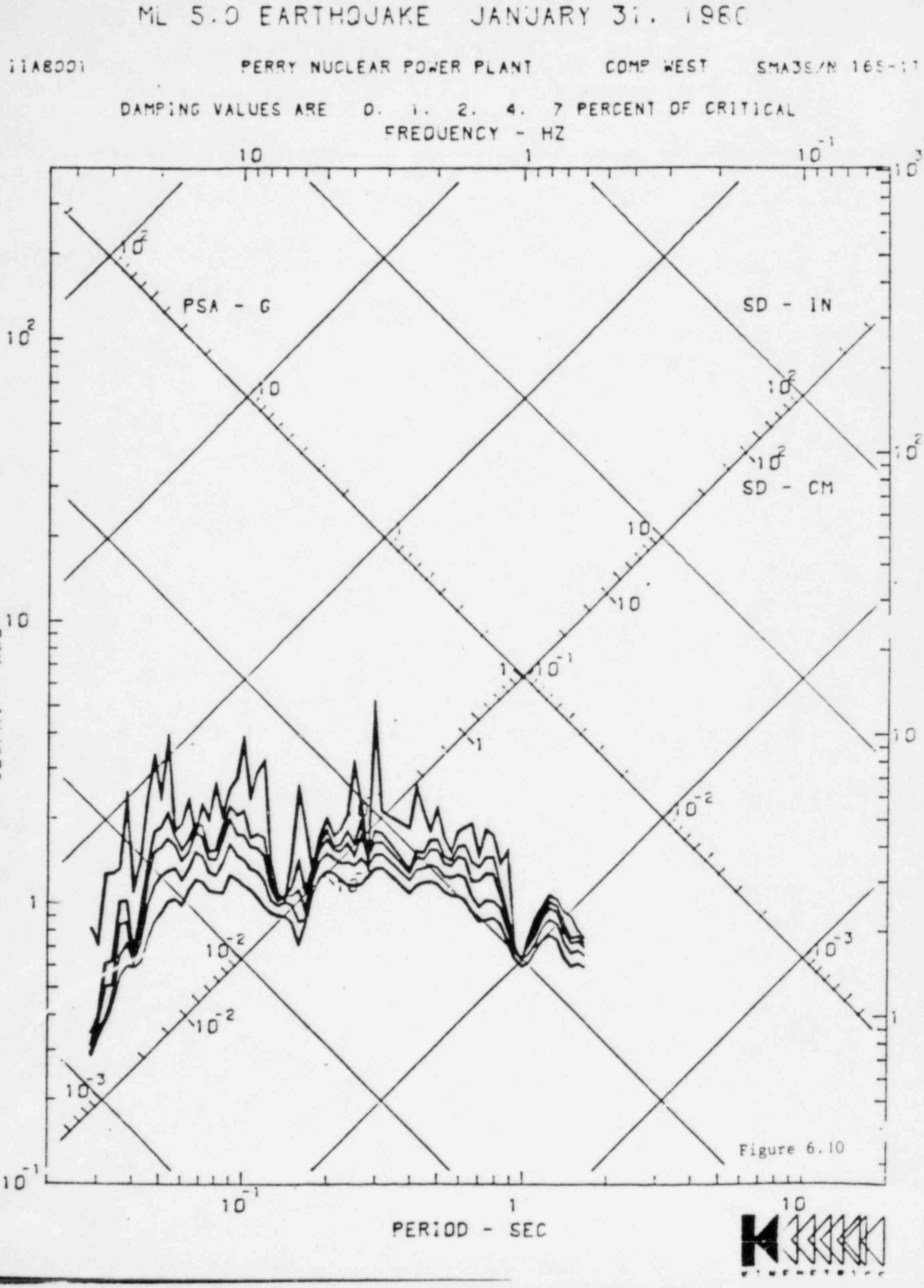

CLEVELAND ELECTRIC ILLUM. CO.
PERRY NUCLEAR POWER PLANT
BUILDING: REACTOR BUILDING
ELEVATION: 686'
DIRECTION: VERTICAL
RECORDED TIME HISTORY AT
NODE 13, NEAR MID HT. OF CONTMT.
IN THE UPDATED SEISMIC MODEL
ENGINEER:
REVIEWER:
ACCELERATION (G)
TIME IN SECONDS
0.00 1.50 3.00 4.50 6.00 7.50 9.00 10.50 12.00 13.50 15.00 16.50 18.00 19.50 21.00 22.50
SCALES:
TIME .... 1"= 1.500 SECS.
ACCELERATION .. 1"= 0.120 G

Figure 6.9



ML 5.0 EARTHQUAKE JANUARY 31, 1980
11AB001 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-11
DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL
FREQUENCY - HZ
PSA - G
SD - IN
SD - CM
PSV - IN/SEC
PERIOD - SEC

Figure 6.10

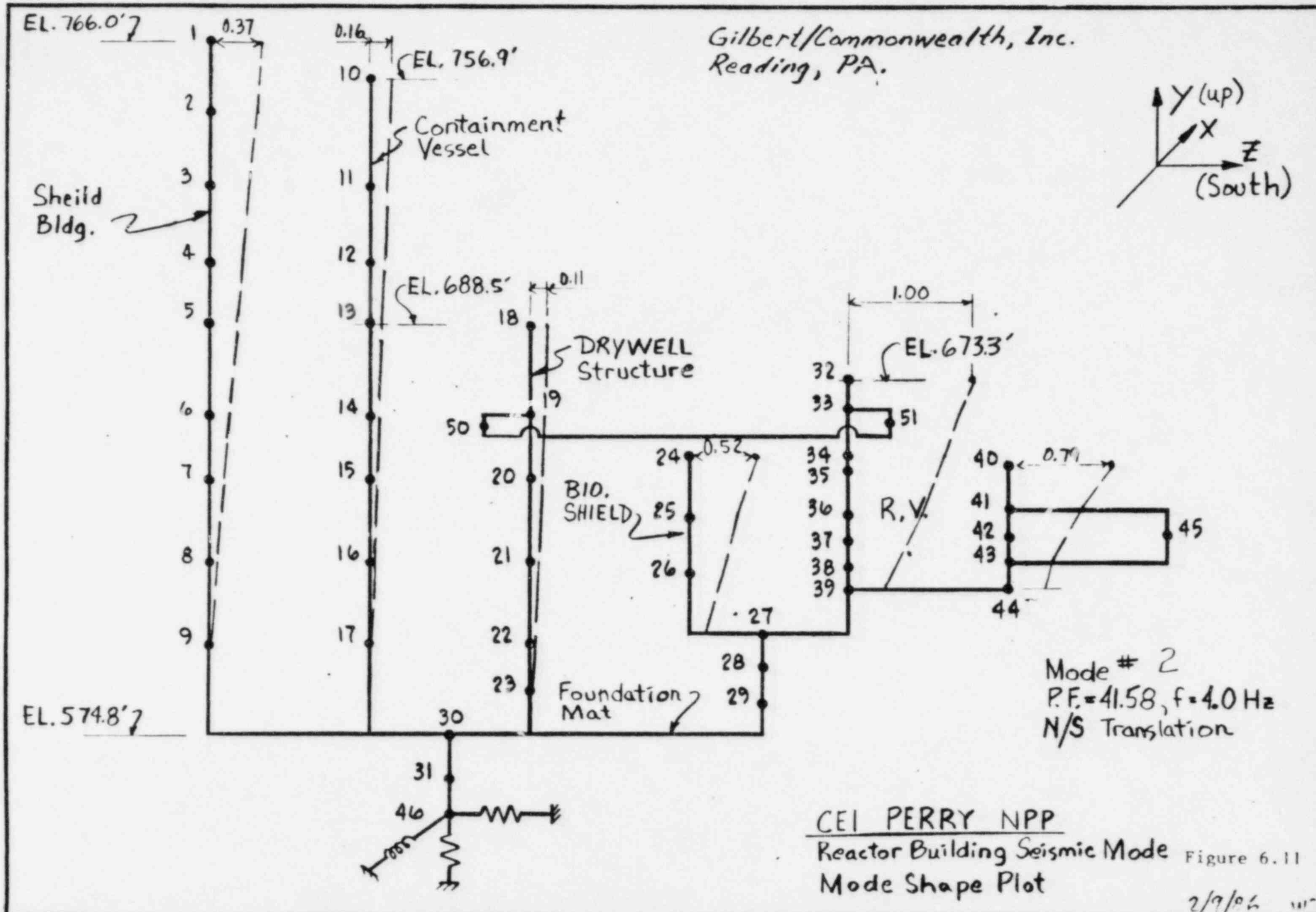
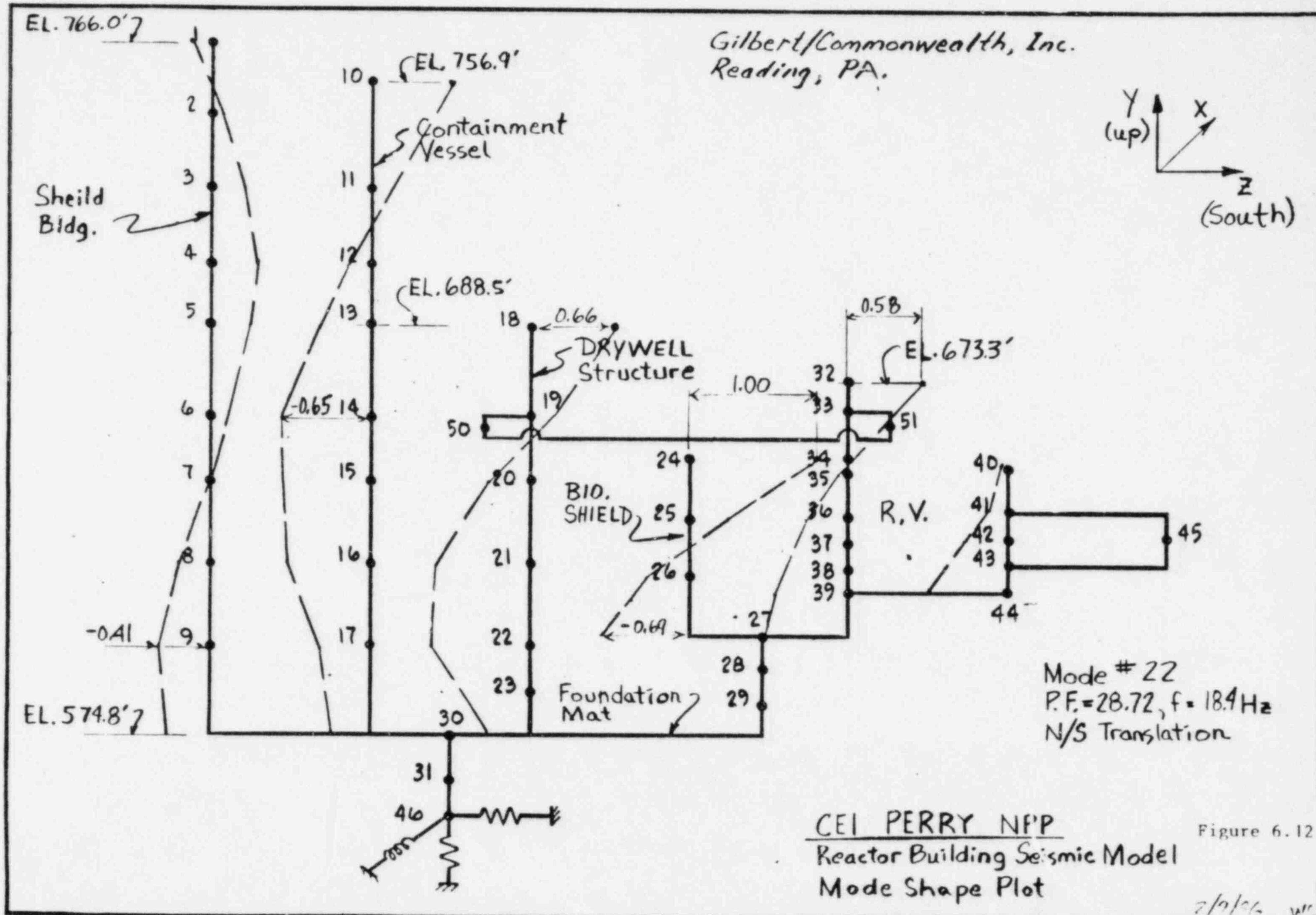
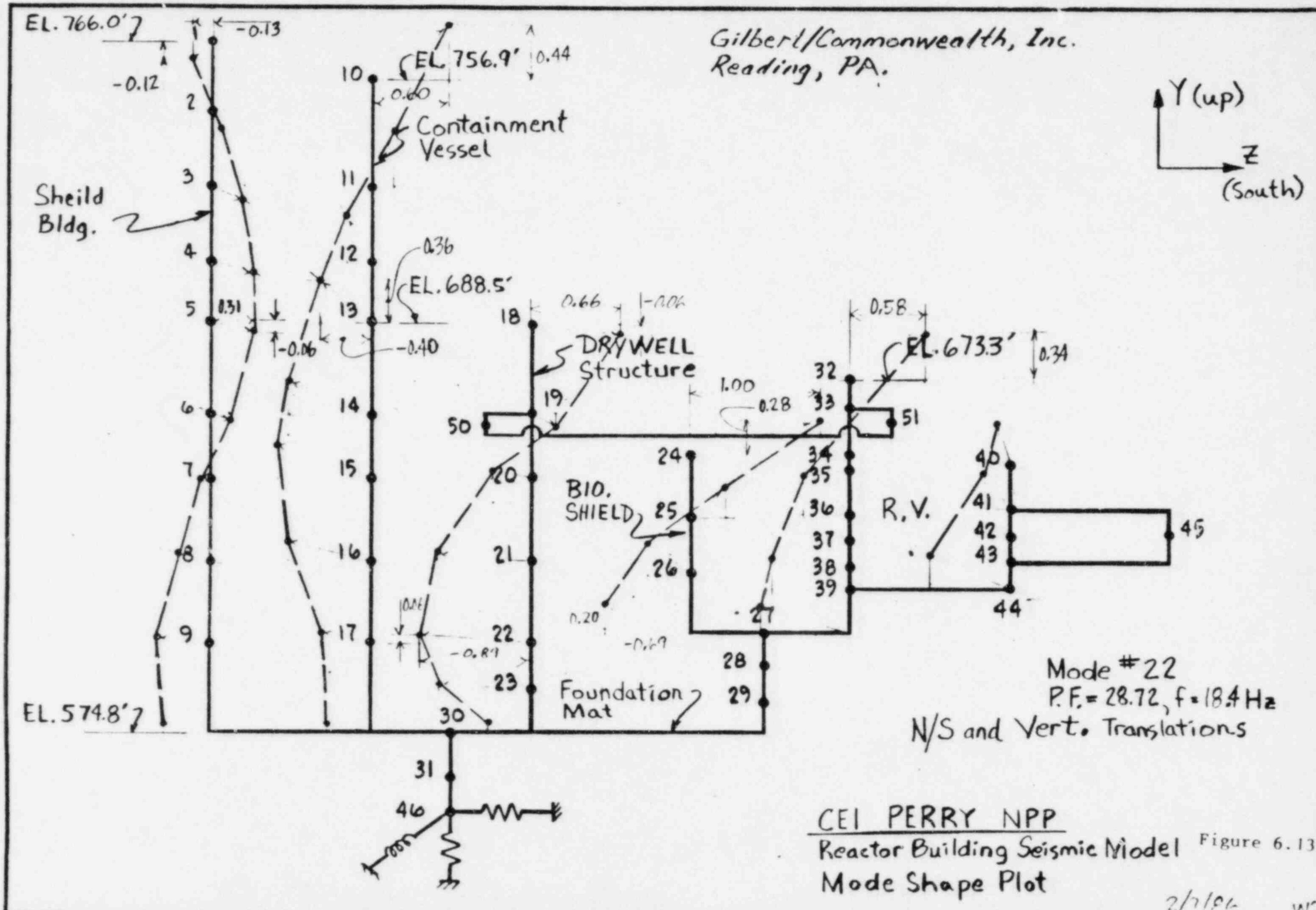

EL. 766.0'
0.37
0.16
EL. 756.9'
Gilbert/Commonwealth, Inc.
Reading, PA.
Y (up)
X
Z
(South)
Containment Vessel
Sheild Bldg.
EL. 688.5'
0.11
1.00
DRYWELL Structure
EL. 673.3'
0.52
0.71
BIO. SHIELD
R.V.
Mode # 2
P.F. = 41.58, f = 4.0 Hz
N/S Translation
Foundation Mat
EL. 574.8'
CEI PERRY NPP
Reactor Building Seismic Mode
Mode Shape Plot
2/7/86

Figure 6.11



Gilbert/Commonwealth, Inc.
Reading, PA.
EL. 766.0'
EL. 756.9'
Containment Vessel
Sheild Bldg.
EL. 688.5'
0.66
DRYWELL Structure
0.58
EL. 673.3'
1.00
-0.65
BIO. SHIELD
R.V.
-0.69
-0.41
EL. 574.8'
Foundation Mat
Y (up)
X
Z (South)
Mode # 22
P.F. = 28.72, f = 18.4 Hz
N/S Translation
CEI PERRY NPP
Reactor Building Seismic Model
Mode Shape Plot

Figure 6.12



Gilbert/Commonwealth, Inc.
Reading, PA.

Mode #22
P.F. = 28.72, f = 18.4 Hz
N/S and Vert. Translations

CEI PERRY NPP
Reactor Building Seismic Model
Mode Shape Plot

Figure 6.13

7.0 CONFIRMATORY PROGRAMS

Within hours of the earthquake, CEI's geophysical consultant had set up seismographs in the area of the epicenter to monitor any aftershocks. These remain in place at this time and the monitoring will continue until it is determined that no further aftershocks are anticipated. In addition, CEI is cooperating with the U.S. Geological Survey and others who are studying the earthquake.

CEI has instituted a specific procedure (OM19A: GTI-003) to ensure proper documentation, review, and reporting of all potentially earthquake related conditions in the plant. Under the procedure, all of the items identified within 24 hours following the seismic event have been documented as Earthquake Inspection Team Items ("EITI's"). Engineering has evaluated each EITI to determine whether the item was a direct result of the earthquake. The results of the evaluation are shown in Appendix E. The two EITI's determined to have been caused by the earthquake, and those with an "indeterminate" cause (i.e., where it cannot be definitively established that the condition existed prior to the earthquake), were identified and documented as discussed above. None of these items is associated with any plant structural damage. It is anticipated that minor rework or repair will be done on some of the items in accordance with CEI's normal program to correct nonconforming conditions. CEI's procedure provides that all potentially earthquake related EITI's will be maintained in the "as found" condition until reviewed by CEI and released by the NRC.

New Work Requests (WR's) (for conditions other than those already covered by EITI's), are also being reviewed in accordance with CEI's new procedure for earthquake related items.

Engineering evaluation results for these items are being documented and tracked. As with the EITI's, any potentially earthquake related conditions associated with new WR's are being maintained in the as-found condition until reviewed by CEI and released by the NRC. CEI has not identified any plant structural damage associated with potentially earthquake related items identified on new WR's.

On a longer term basis, CEI is participating in several industry efforts to study the effects of seismic events on nuclear plants. The organizations performing these studies include the Seismic Owners Group (SOG), the Seismic Qualification Utilities Group (SQUG), and Electric Power Research Institute (EPRI).

These industry groups are examining various gneneric seismic issues which have been under consideration by the NRC. For example, SOG has been focusing on eastern seismicity hazard analysis, with EPRI managing the program effort. SOG will review the Perry earthquake as part of this work. SQUG has focused its effort on the seismic qualification of electrical equipment. SQUG intends to review the Perry data presented in this report, and will integrate this information into their studies. EPRI has been supporting SQUG by sponsoring projects to resolve issues associated with equipment qualification, focusing on test data, adequacy of equipment anchorages, and post earthquake investigation programs.

These industry groups all visited the site shortly after the seismic event. A SOG/EPRI team installed in-plant and field instruments within a day of the seismic event to collect aftershock data. An SQUG team conducted a plant walkdown. The team informed CEI that the seismic event at Perry was much smaller than others they have evaluated (Coalingo, Chile, Mexico City, Morgan Hill), and that the SQUG data base generated from these previous earthquakes would predict no damage from the Janaury 31, 1986 earthquake. This prediction was confirmed by the group's plant walkdown. The EPRI equipment qualification program manager concluded that Perry's response to the seismic event was properly handled. The Perry experience will be used in EPRI's development of generic post-earthquake investigation methods.

8.0 SUMMARY AND CONCLUSIONS

The seismic event which occurred on January 31, 1986 has been thoroughly studied and its effects on the Perry Nuclear Power Plant analyzed in detail. The earthquake itself was of smaller magnitude and intensity than the postulated earthquake which was used as the basis for the plant seismic design. The occurrence of the 1986 earthquake does not change any of the conclusions previously reached as to the geology and seismology of the site. Consideration of this event does not result in any change in the Safe Shutdown Earthquake licensing basis for the Perry plant.

The earthquake confirmed the adequacy of the plant's seismic design. The plant structures and equipment were essentially unaffected by the earthquake. The large number of safety and non-safety related systems which were operating or energized at the time of the earthquake responded in accordance with their design. Extensive plant walkdowns and inspections revealed no structural or equipment damage.

The seismic characteristics of the earthquake have been reviewed and compared the plant's seismic design. The high frequencies which typified the 1986 earthquake are of no significance with regard to the adequacy of the plant's design. In contrast to the seismic design basis, the earthquake was of short duration, with low energy, low velocities and small displacements. Although certain of the recorded response spectra exceeded the design response spectra in the high frequency range, such exceedances are consistent with the analytical methods of Regulatory Guide 1.60 and are of no engineering significance. In the frequency range of significance for plant structural design (below 14 Hz), recorded spectra are far below the design response spectra for Perry.

The January 31, 1986 earthquake, in effect, constituted a proof test of Perry's seismic design. By any standard the Perry Nuclear Power Plant passed that test. The earthquake presents no new information which would change the previously accepted licensing basis for the plant.

APPENDIX A

STRONG-MOTION DATA FROM THE PERRY NUCLEAR POWER PLANT
SEISMIC INSTRUMENTATION
KINEMETRICS

ML 5.0 EARTHQUAKE

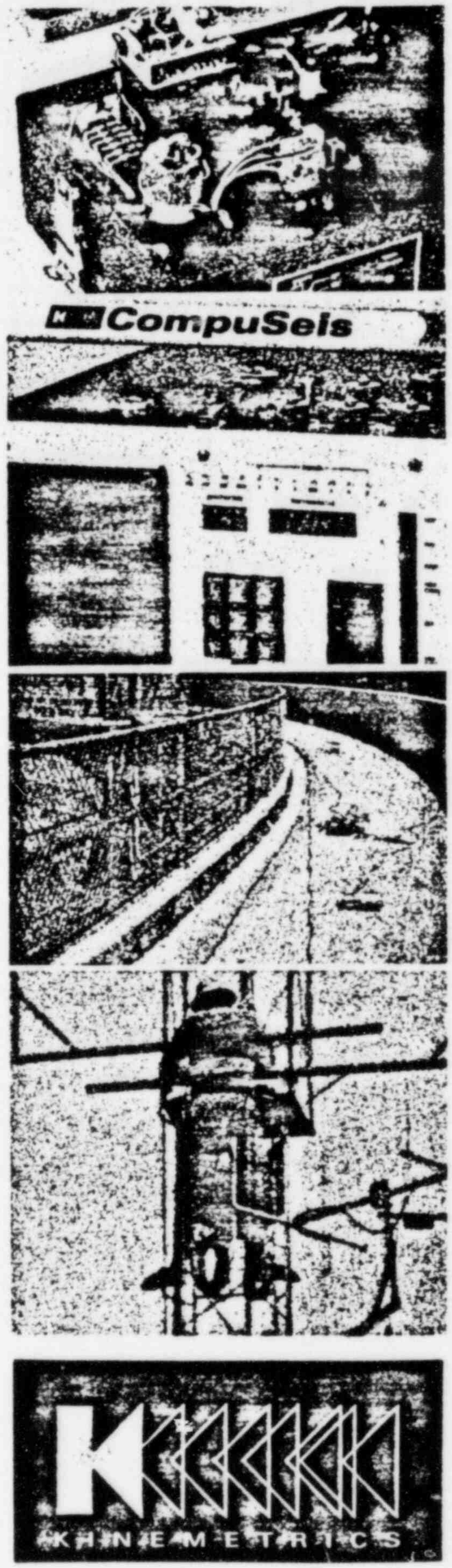
JANUARY 31, 1986

STRONG-MOTION DATA

from the

PERRY NUCLEAR POWER PLANT

SEISMIC INSTRUMENTATION



February 3, 1986

STRONG-MOTION DATA REPORT

for the

M_L 5.0 EARTHQUAKE

of

1147 EST, JANUARY 31,1986

PERRY, OHIO

RECORDED ON THE

PERRY NUCLEAR POWER PLANT

STRONG MOTION ACCELEROGRAPHS

for

Cleveland Electric Illuminating Company

Requisition No. NED-E-860006

by

Kinemetrics/Systems
222 Vista Ave.
Pasadena, CA 91107

Sales Order C-K6028

February 4, 1986

TABLE OF CONTENTS

1.0 INTRODUCTION

On January 31, 1986, a (M_L 5.0) local earthquake was recorded by the strong-motion instrumentation at Perry Nuclear Power Plant, Perry, Ohio. The FM analog magnetic tape cassette records from two Kinemetrics Model SMA-3 accelerographs were retrieved from the instruments and provided to Kinemetrics for analysis.

This report describes the processing of these strong-motion records and presents the results. Included are the uncorrected accelerograms, corrected acceleration, velocity and displacement time series, and response spectra.

2.0 INSTRUMENTATION

2.1 Model SMA-3 Accelerograph

The SMA-3 is a multi-channel, centralized recording, FM analog magnetic tape accelerograph system designed to detect and record strong local earthquakes and record the three orthogonal acceleration signals on cassette tape. The SMA-3 remains in a standby mode until its vertical trigger detects an earthquake. The trigger then actuates recording in less than .10 seconds.

The force balance accelerometers in the SMA-3 have a nominal natural frequency of 50 Hz and damping of 65% critical, providing flat (-3dB) response from DC to 50 Hz. The nominal sensitivity of each of the three channels is 2.5 volts/g with a full scale response of 1.0g. The dynamic range of the accelerograph is nominally 40 dB, giving it a resolution of approximately .01g.

The trigger in the SMA-3 has a flat (-3dB) response from 1 to 10 Hz and a nominal trigger level of 0.01g.

Power is supplied to the SMA-3 by internal rechargeable batteries. These batteries are kept in a charged state by 120 VAC line power.

2.2 Calibration Data

The three Model SMA-3 accelerographs which recorded the event were factory calibrated in January, 1985, and the sensors were recalibrated for sensitivity by the Perry NPP personnel in December of 1985. These most current calibration data are given in Table 1 below.

Ser. No.	Channel	Sens., v/g	Nat. Freq., Hz	Damping % critical
165-1	long	2.48	52.3	65
	tran	2.49	53.7	65
	vert	2.47	50.6	64
165-2	long	2.48	52.6	67
	tran	2.48	52.2	72
	vert	2.65	50.5	66

TABLE 1: Calibration Data

3.0 DATA PROCESSING

Data from the Model SMA-3 accelerographs were played back using a Kinemetrics Model SMP-1 Playback System through a Data Compenstor, digitized using a Kinemetrics Model DDS-1105 Digital Data System and processed as described in Kinemetrics' Aplication Note No. 7 "Conditioning and Correction of Strong Motion Data on Analog Magnetic Tapes", appended to this report.

3.1 Digitization

The magnetic tapes were digitized using the DDS-1105. The 1024 Hertz FM time reference recorded on channel 4 of the cassette is output from the SMP-1 and divided down by four (256 Hz ± deviation) and used as the timing signal for the digital conversion time interval. The multiplexed uncorrected time series are written on 9-track computer-compatible tape.

3.2 VOL1 Processing

The digitized data were demultiplexed and scaled to acceleration units using the Table 1 calibration data. The mean was then subtracted from each acceleration time history. The new time histories were then written in a Kinemetrics' VOL1-format disk file.

The three uncorrected acceleration time histories from each SMA-3 record were then plotted; these plots are included in the data section of this report.

3.3 VOL2 Processing

The recorded accelerograms were then instrument and baseline corrected using Kinemetrics' VOL2 program. This program is based upon the VOL2 program developed at Caltech (Trifunac and Lee, 1973). No major modifications to the original VOL2 algorithms have been made.

The data were bandpass filtered using Ormsby filters. The low-pass filter had a cut-off frequency of 35 Hz and a termination frequency of 40 Hz. The high-pass filter had a cutoff frequency of 0.625 Hz and a termination frequency of 0.4 Hz.

Output of this program consists of a plot of corrected acceleration, velocity and displacement for each component of recorded data. These plots are presented in the data section of this report.

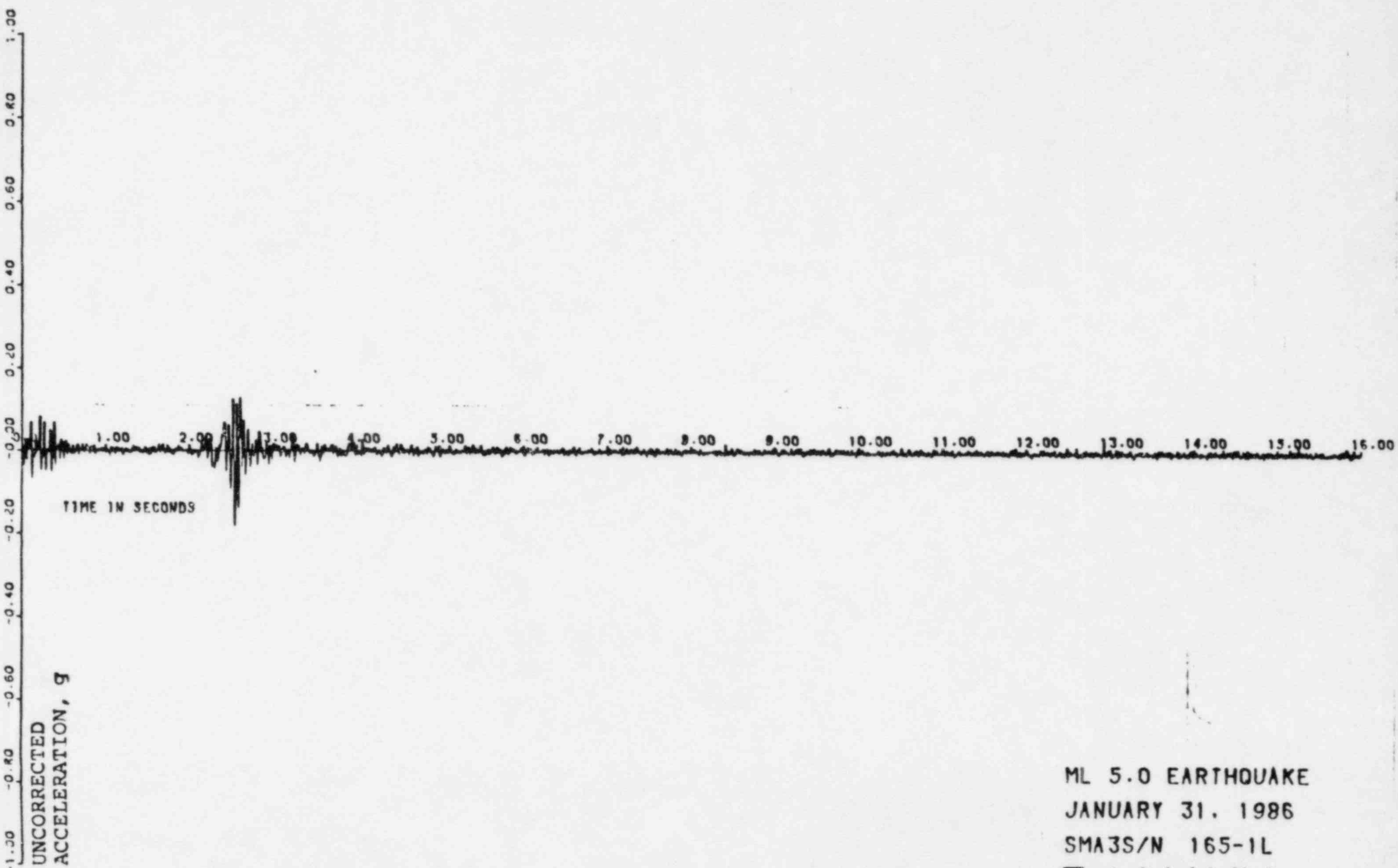
3.4 VOL3 Processing

Linear response spectra were calculated from the corrected acceleration time histories using the algorithms developed by Trifunac and Lee. Response spectra were calculated for damping ratios of 0, 1, 2, 4, and 7 percent. The period range of these spectra was 1.68 to 0.0283 seconds (0.59 to 35.4 Hz) with oscillator response calculated at 1/24 th octave intervals.

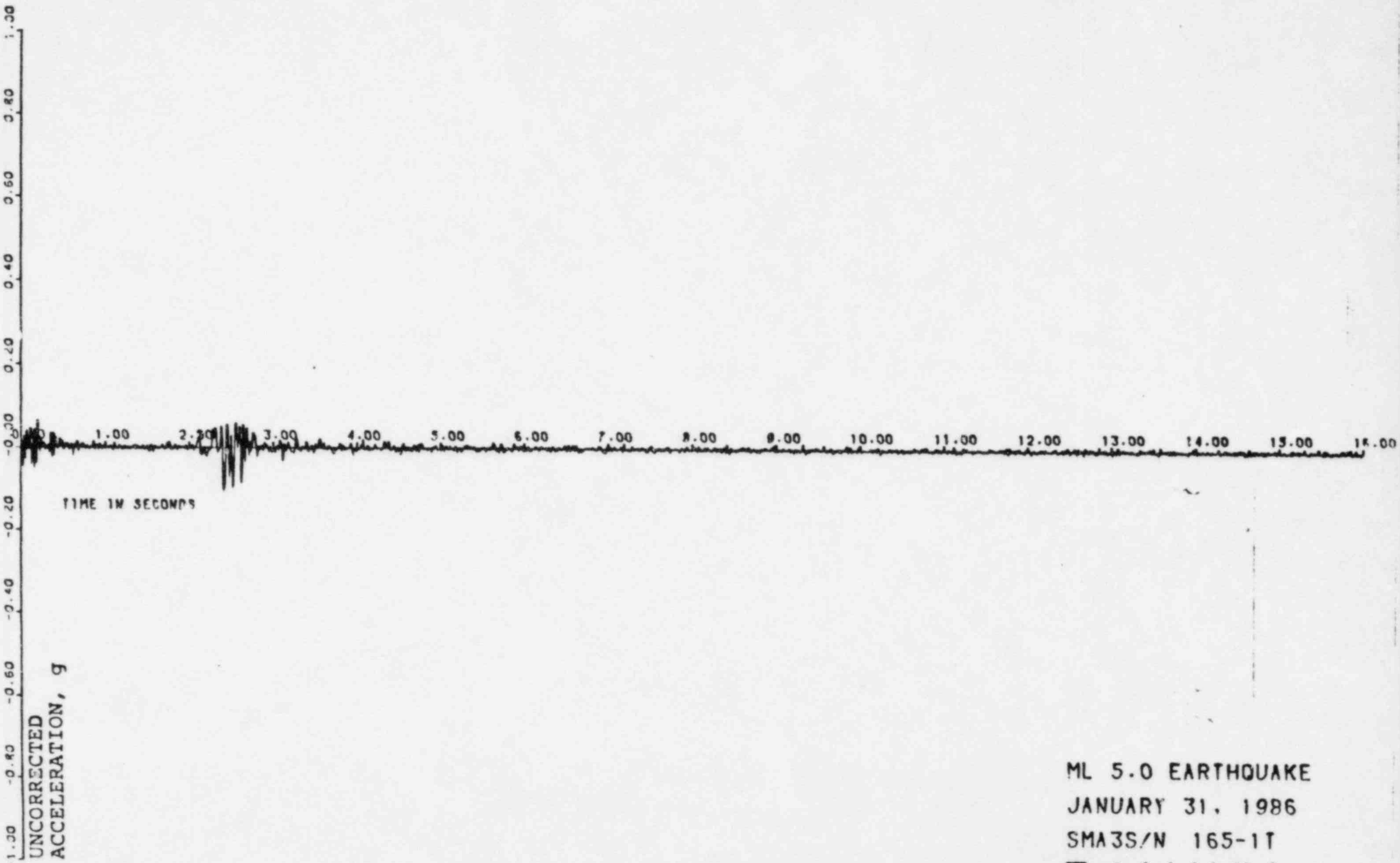
Two types of plots were produced and are included in the data section of this report. The first type is the traditional tripartite log-log plot of pseudo-velocity vs. period. The second is a linear plot of velocity response and Fourier spectrum vs. frequency.

Reactor Building Foundation, Elevation 575 Ft.

SMA-3 Serial Number 165-1
Tag Number D51-N101
Longitudinal Channel - South Orientation
Transverse Channel - West Orientation
Vertical Channel - Up Orientation



ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA3S/N 165-1L



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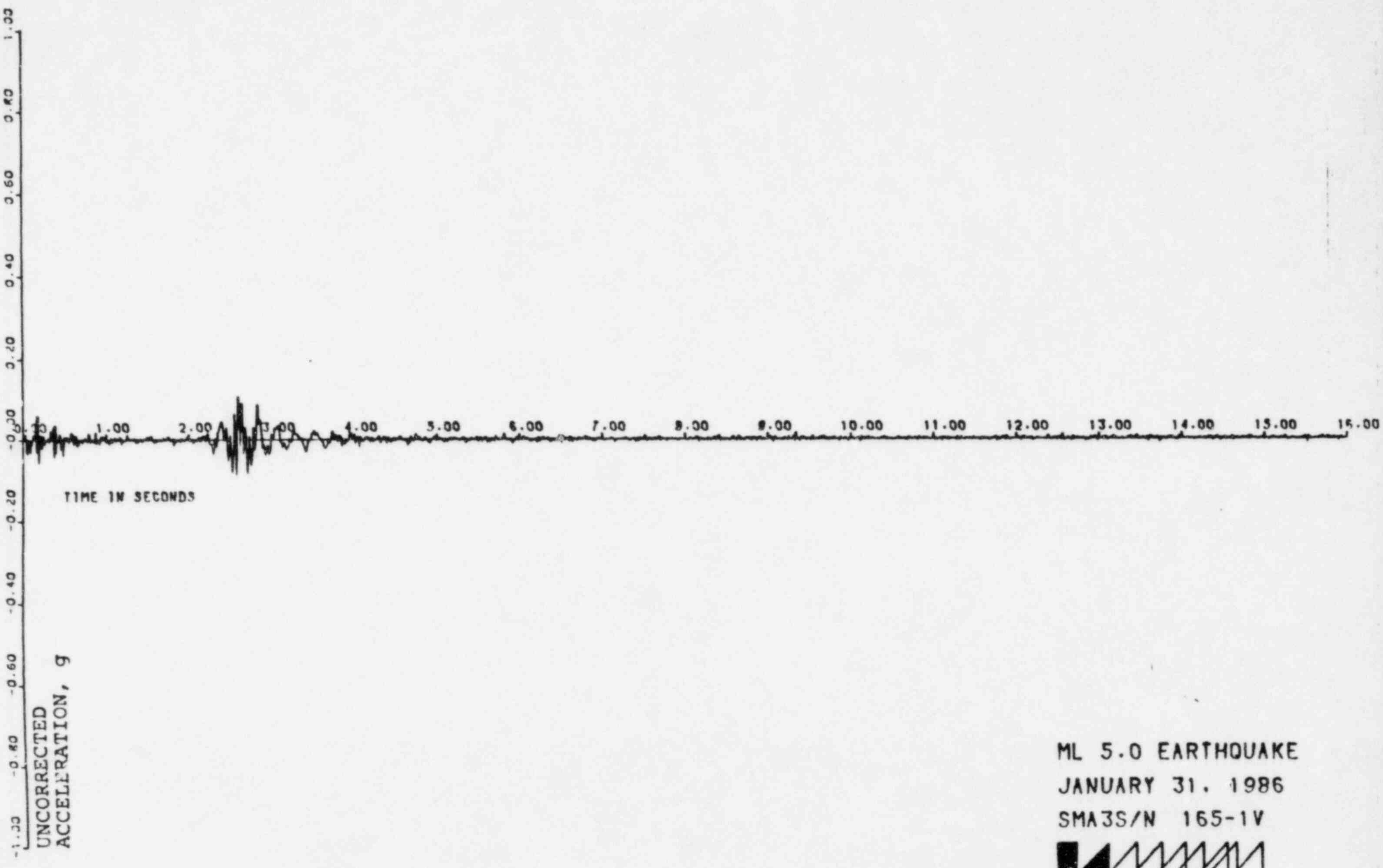
JANUARY 31, 1986
SMA3S/N 165-1T

KINEMETRICS



ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA3S/N 165-1V

KINEMETRICS

ML 5.0 EARTHQUAKE JANUARY 31. 1986

11A8001 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-1L

ACCELEROGRAM IS BAND-PASS FILTERED BETWEEN 0.400- 0.625 AND 35.00- 40.00 HERTZ

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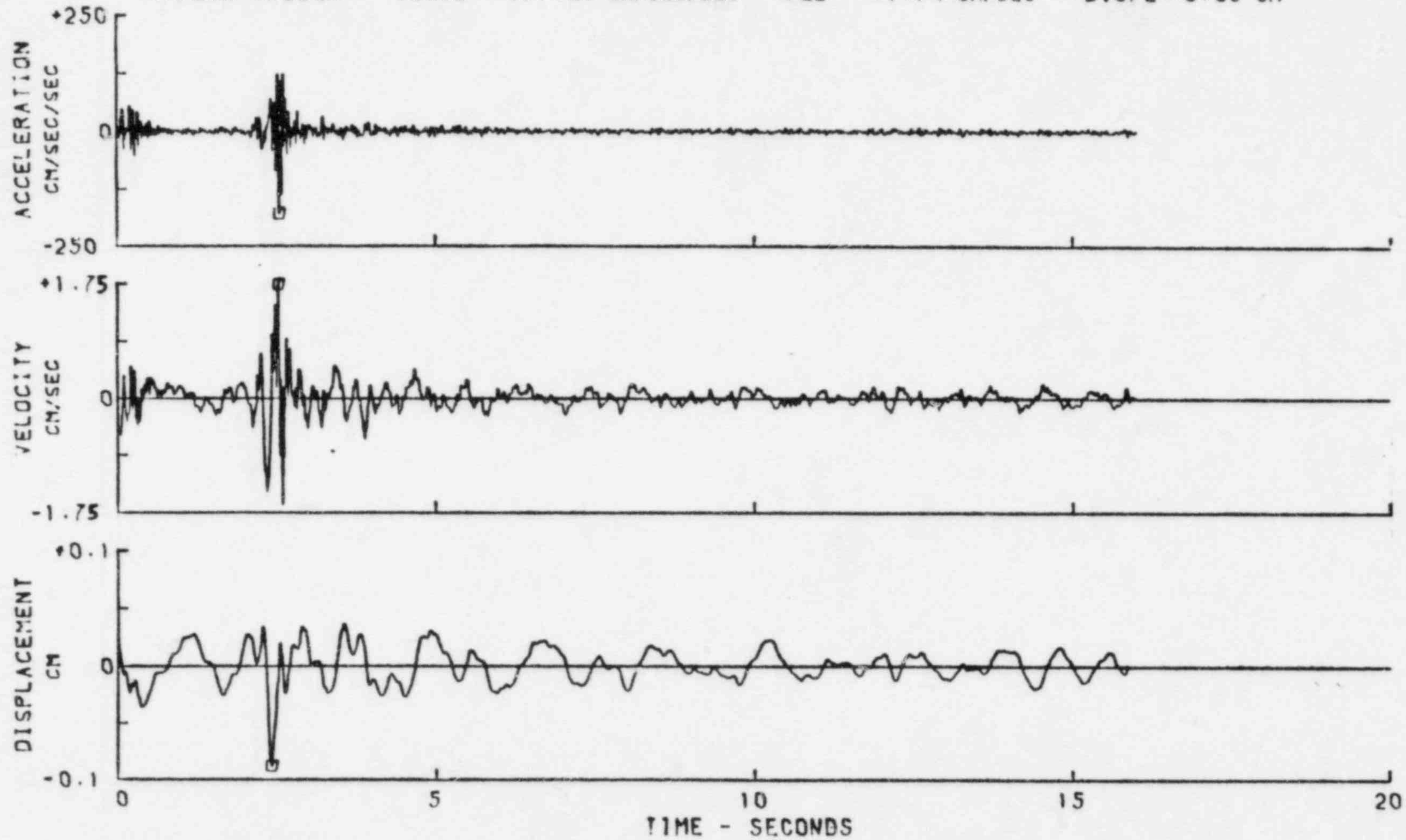

KINEMETRICS

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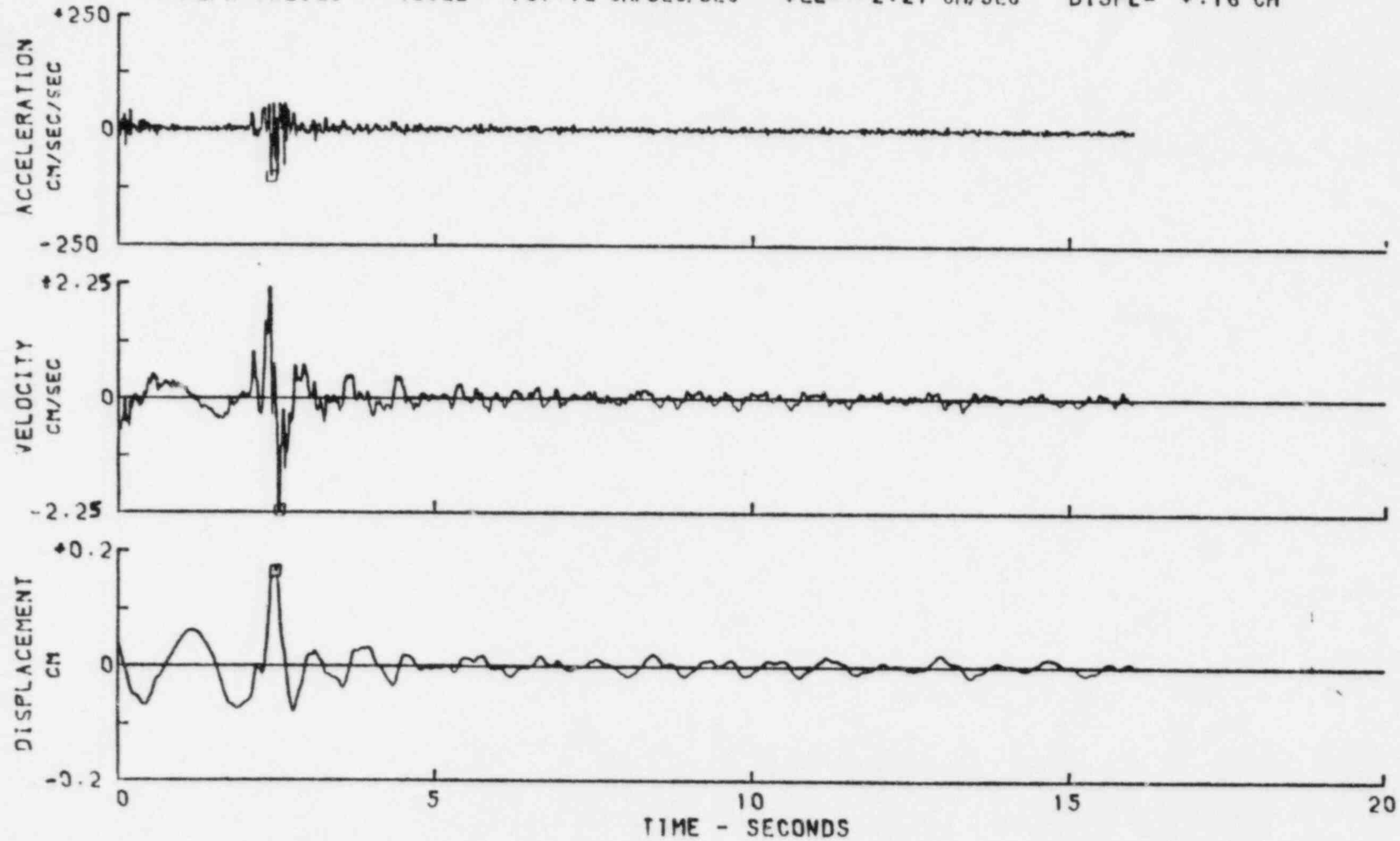

KINEMETRICS

ML 5.0 EARTHQUAKE JANUARY 31, 1986

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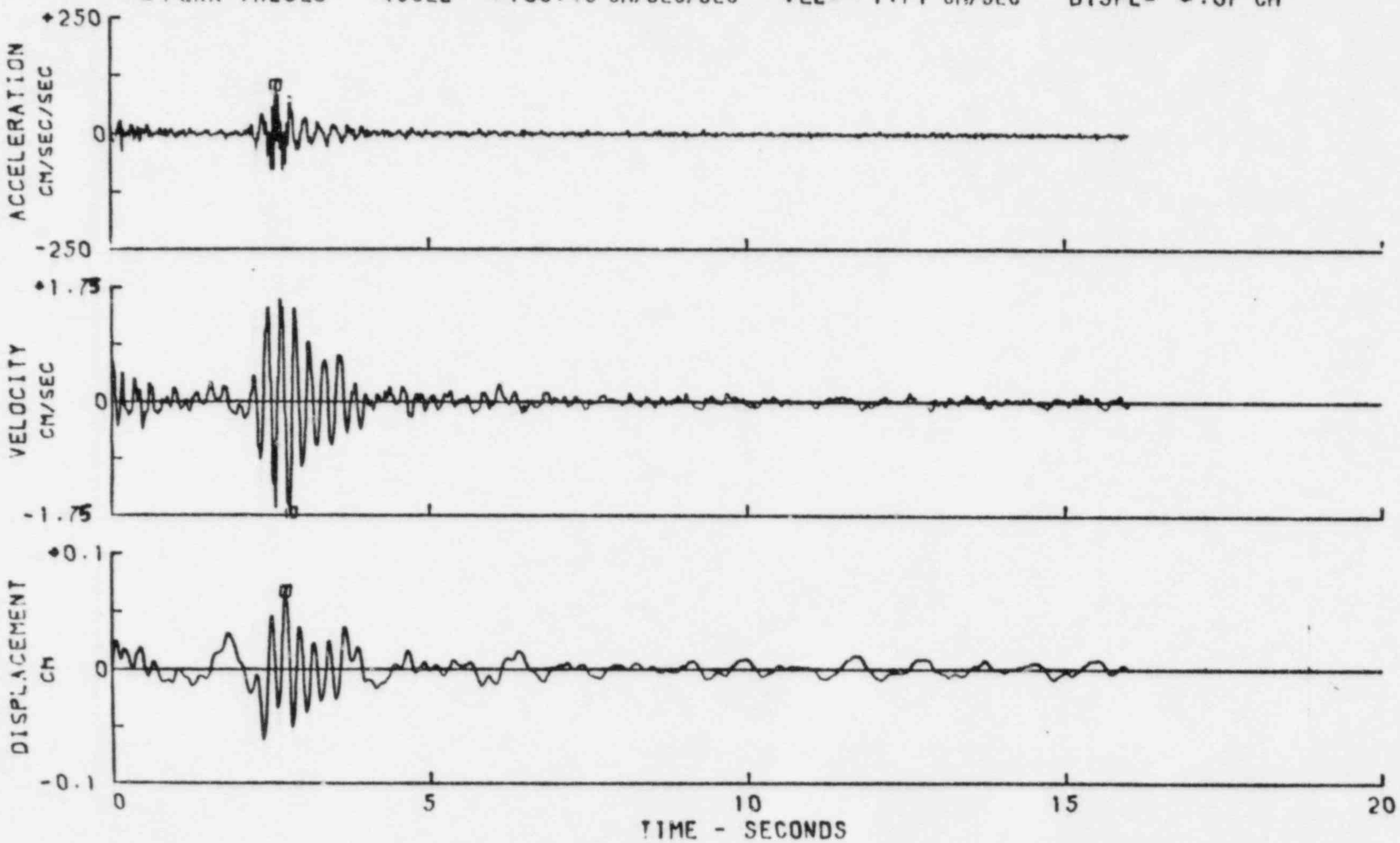

RELATIVE VELOCITY RESPONSE SPECTRUM

ML 5.0 EARTHQUAKE JANUARY 31, 1986

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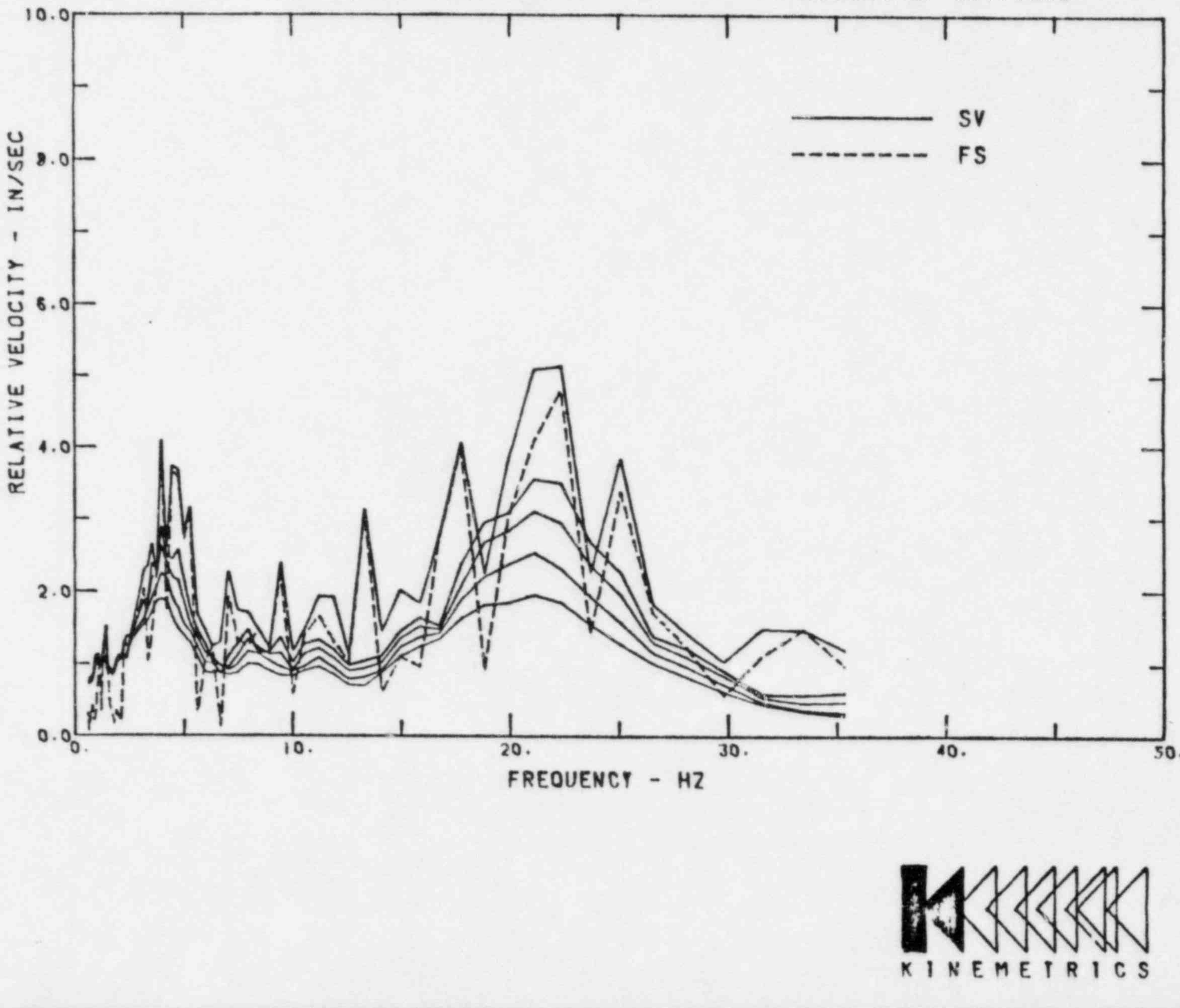

KINEMETRICS

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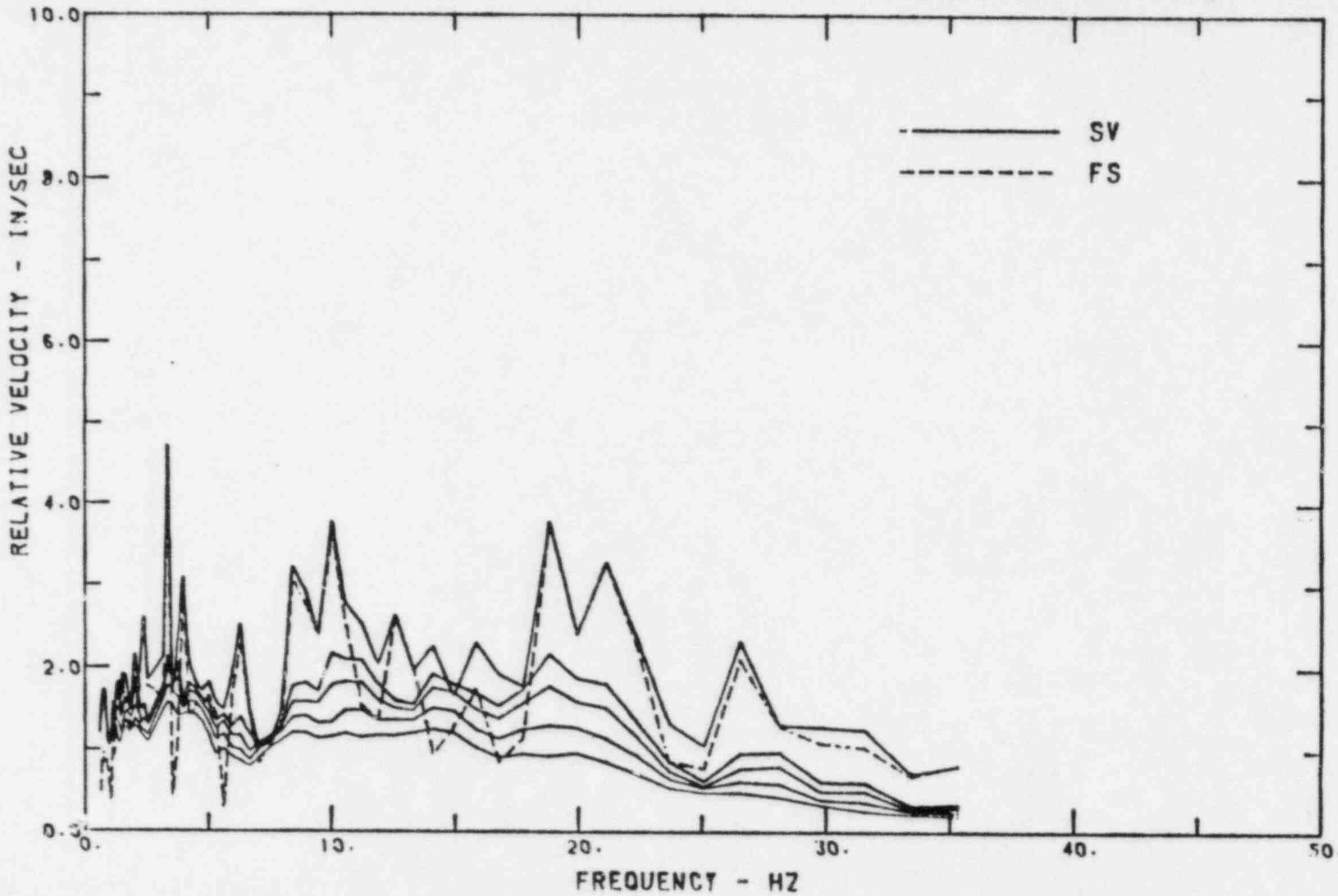

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RELATIVE VELOCITY RESPONSE SPECTRUM

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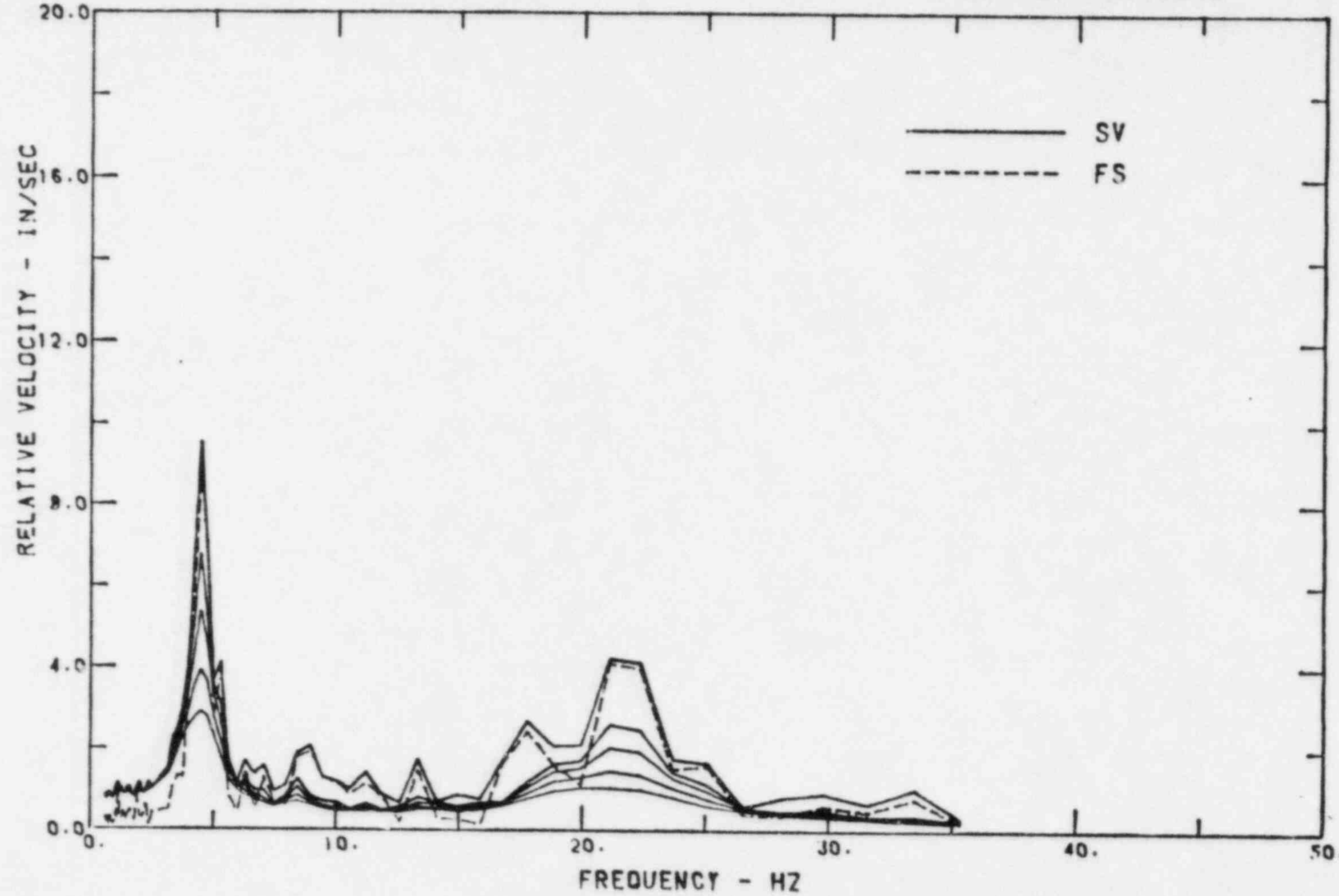

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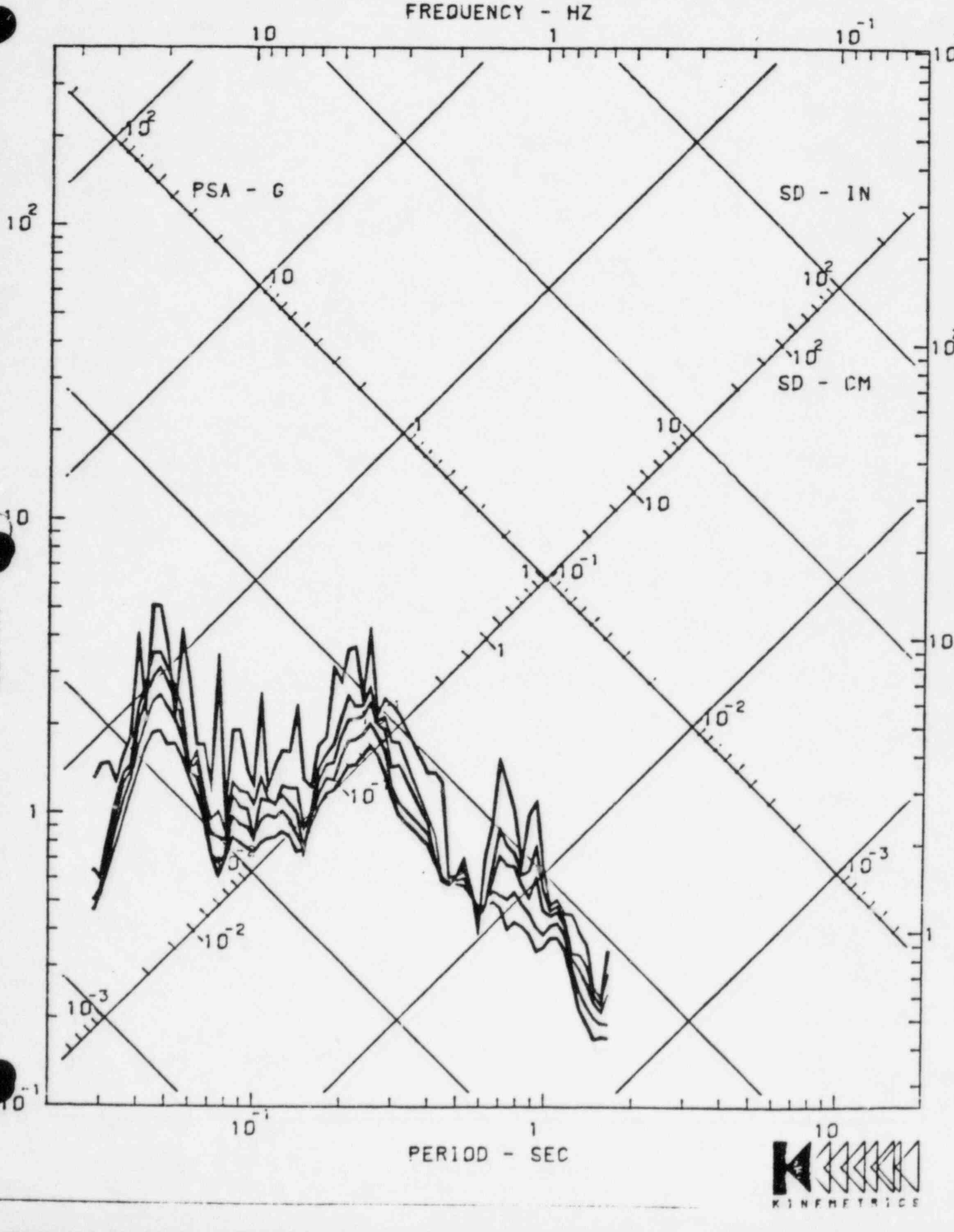

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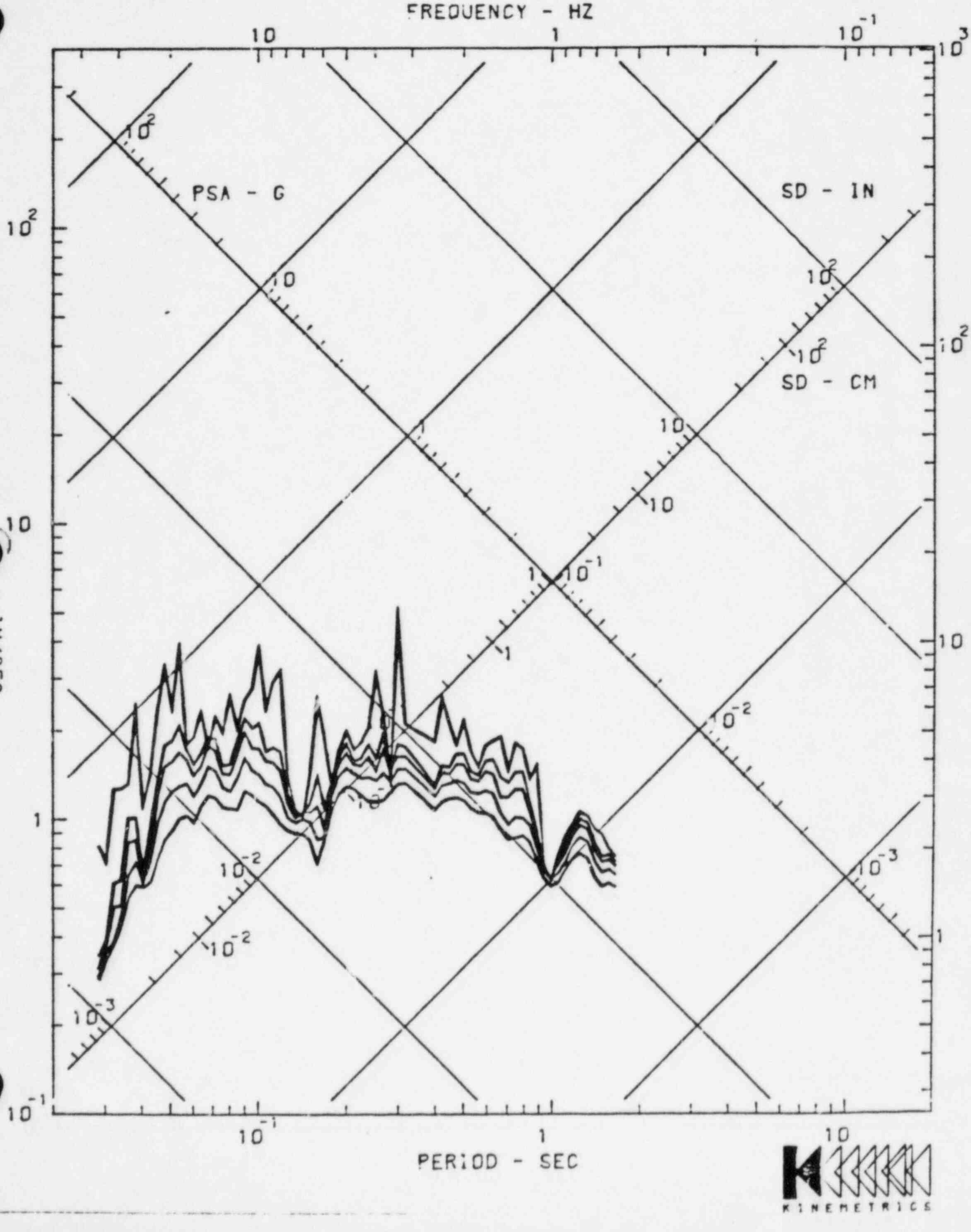

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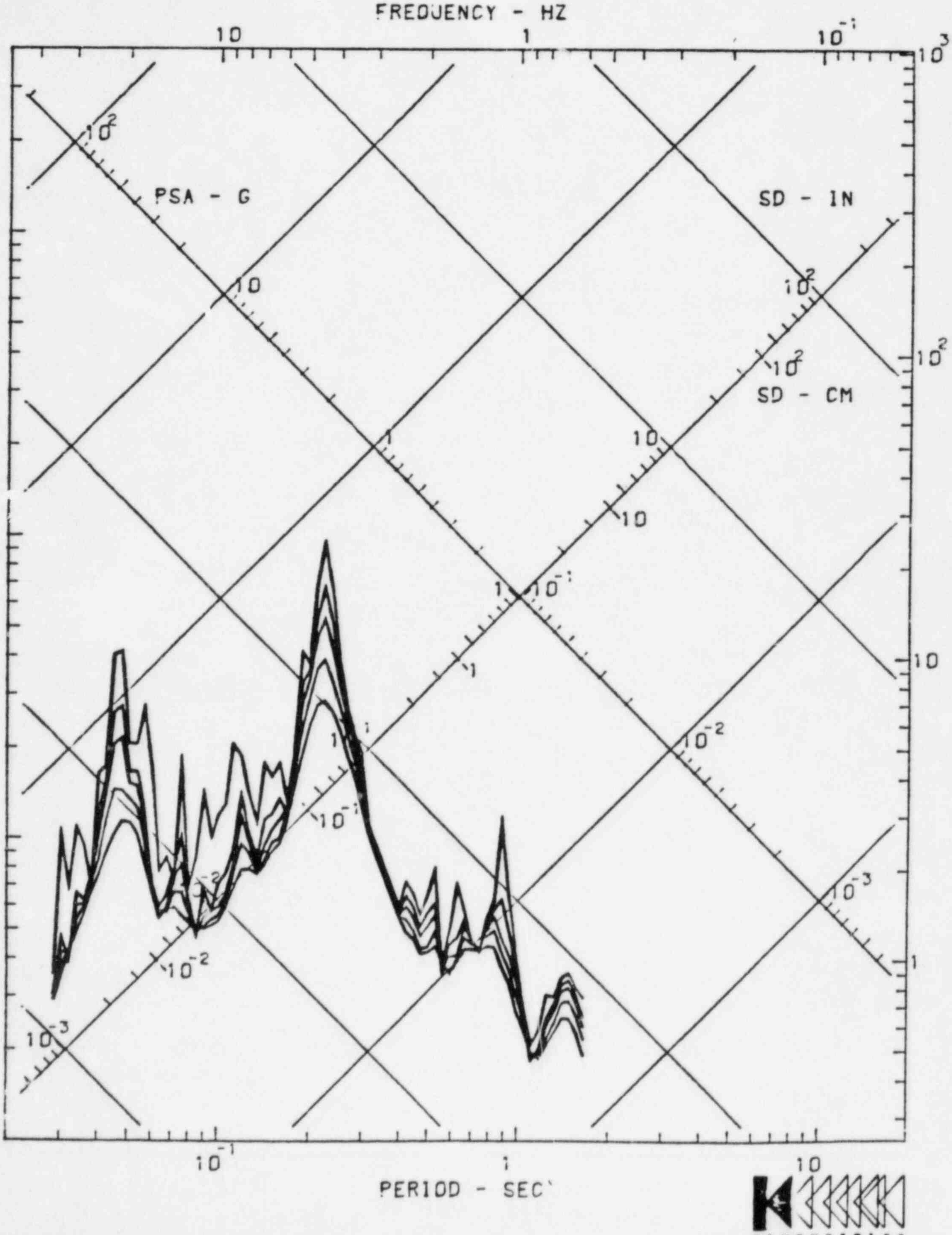

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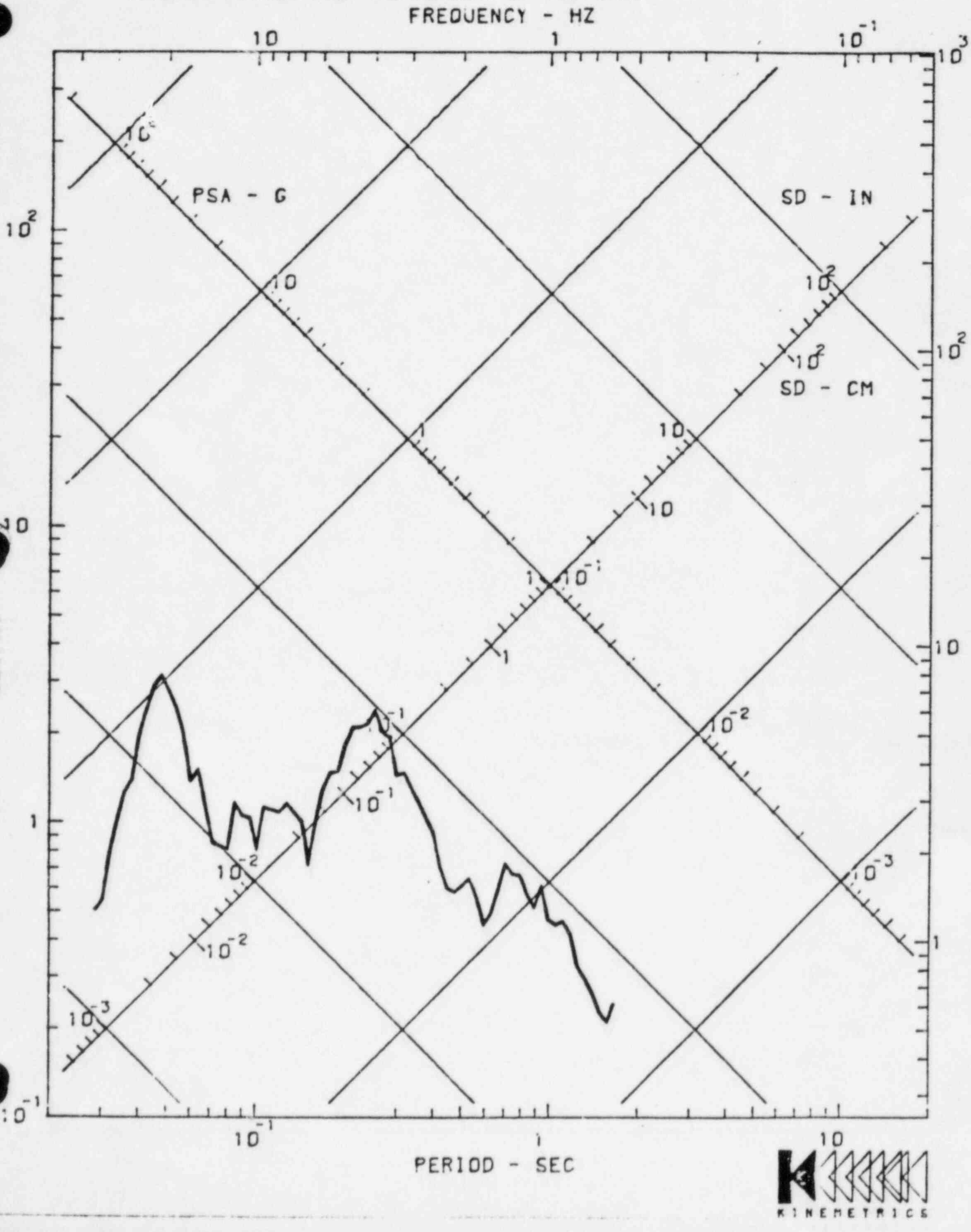

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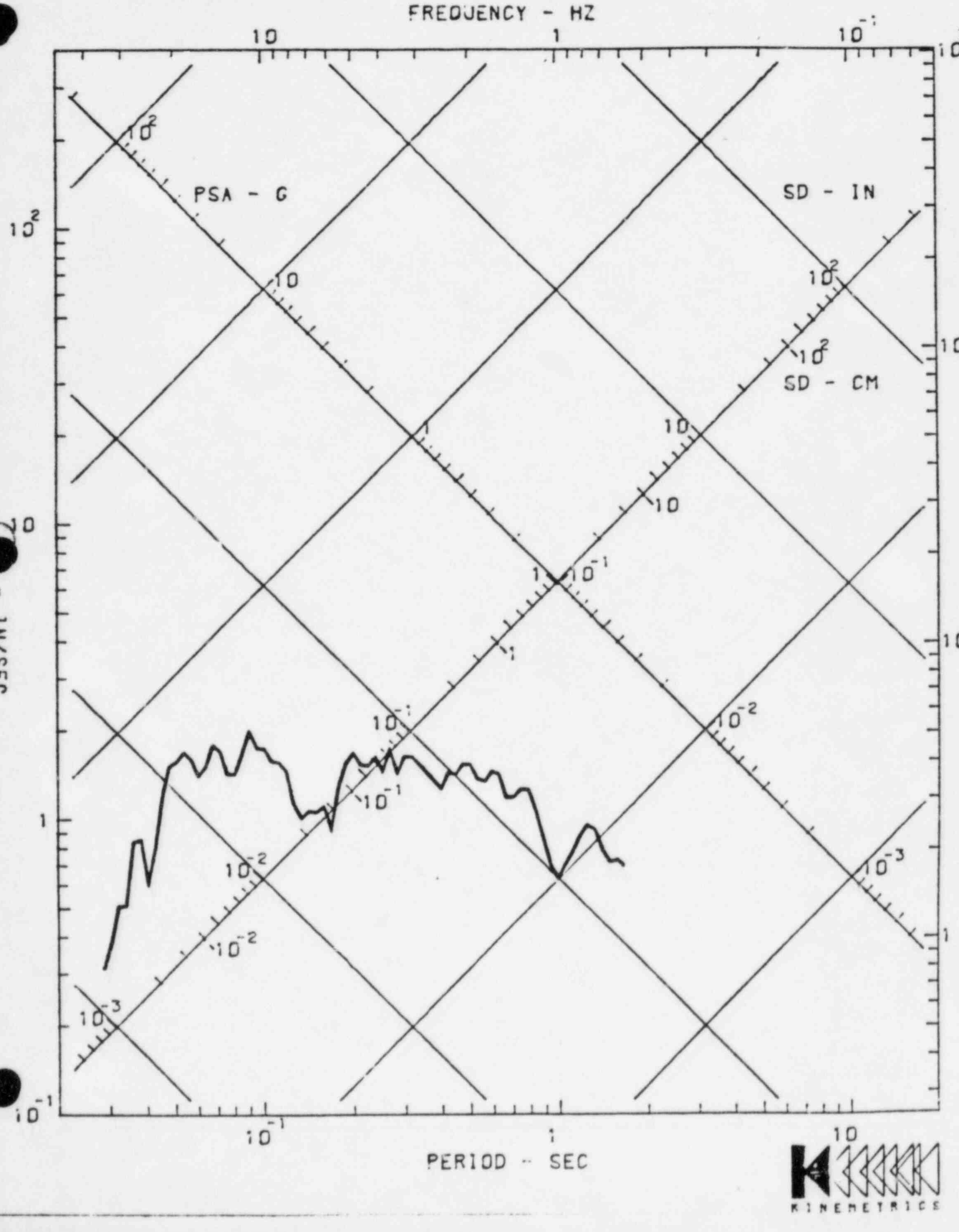

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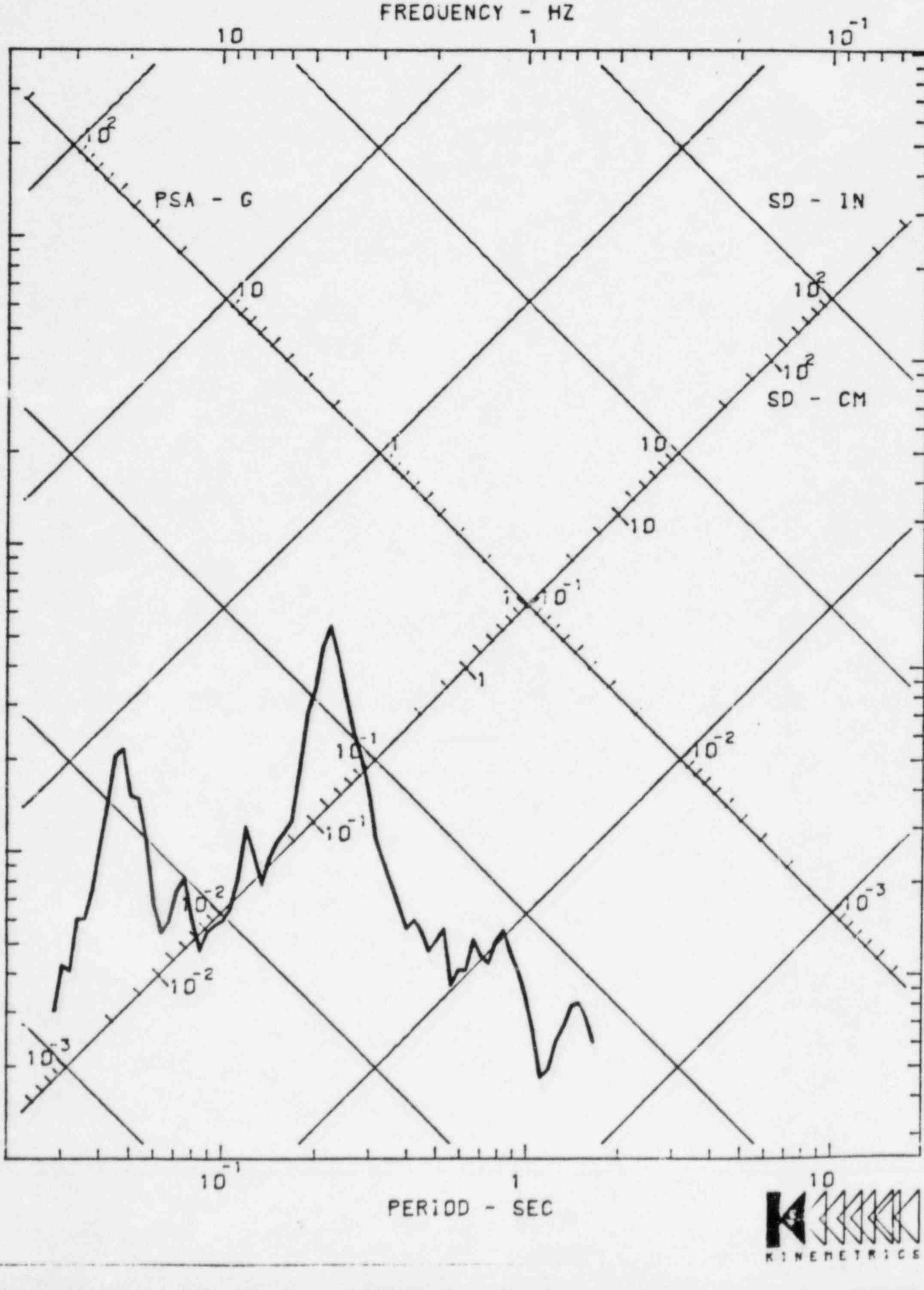

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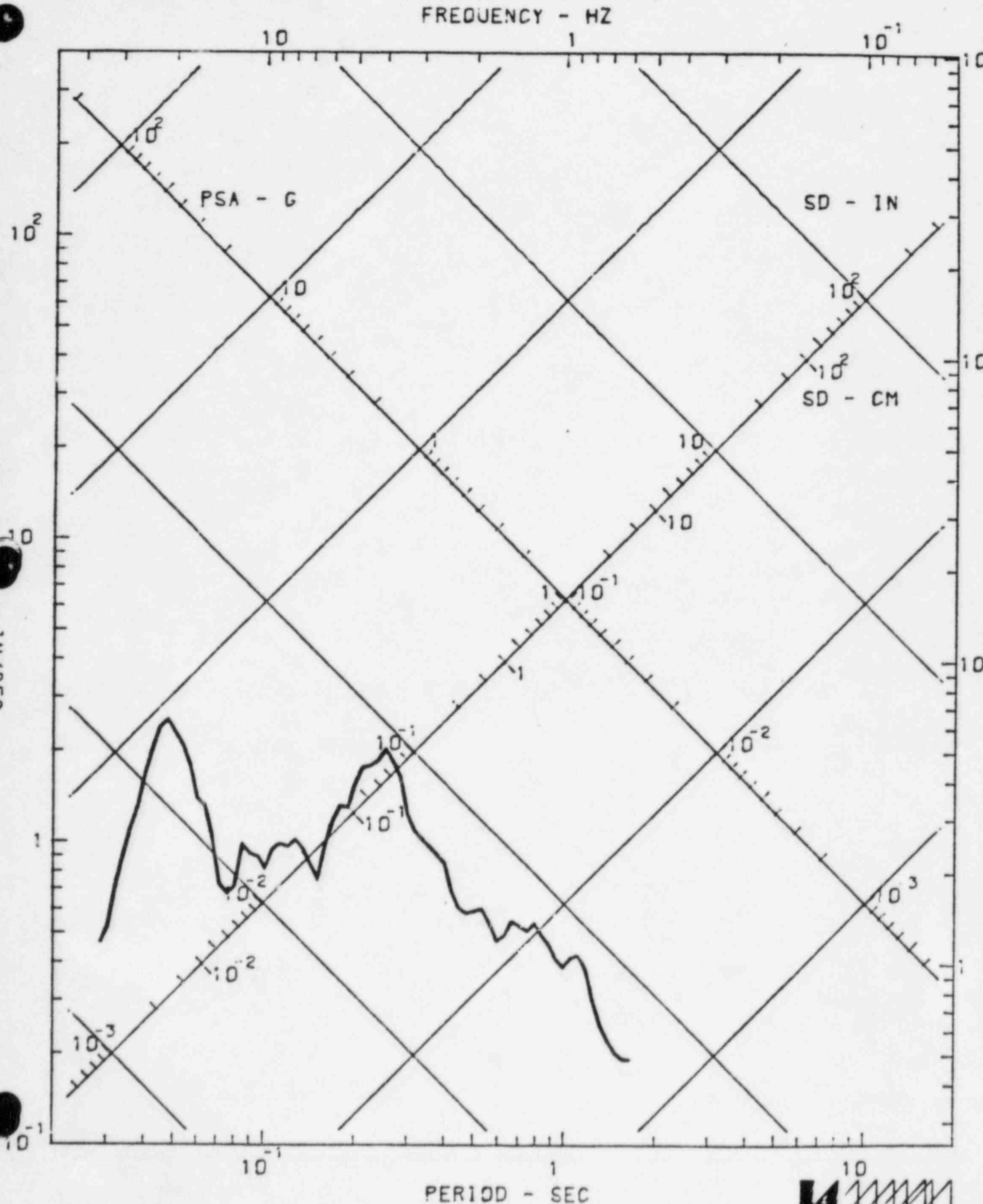

KINEMETRICS

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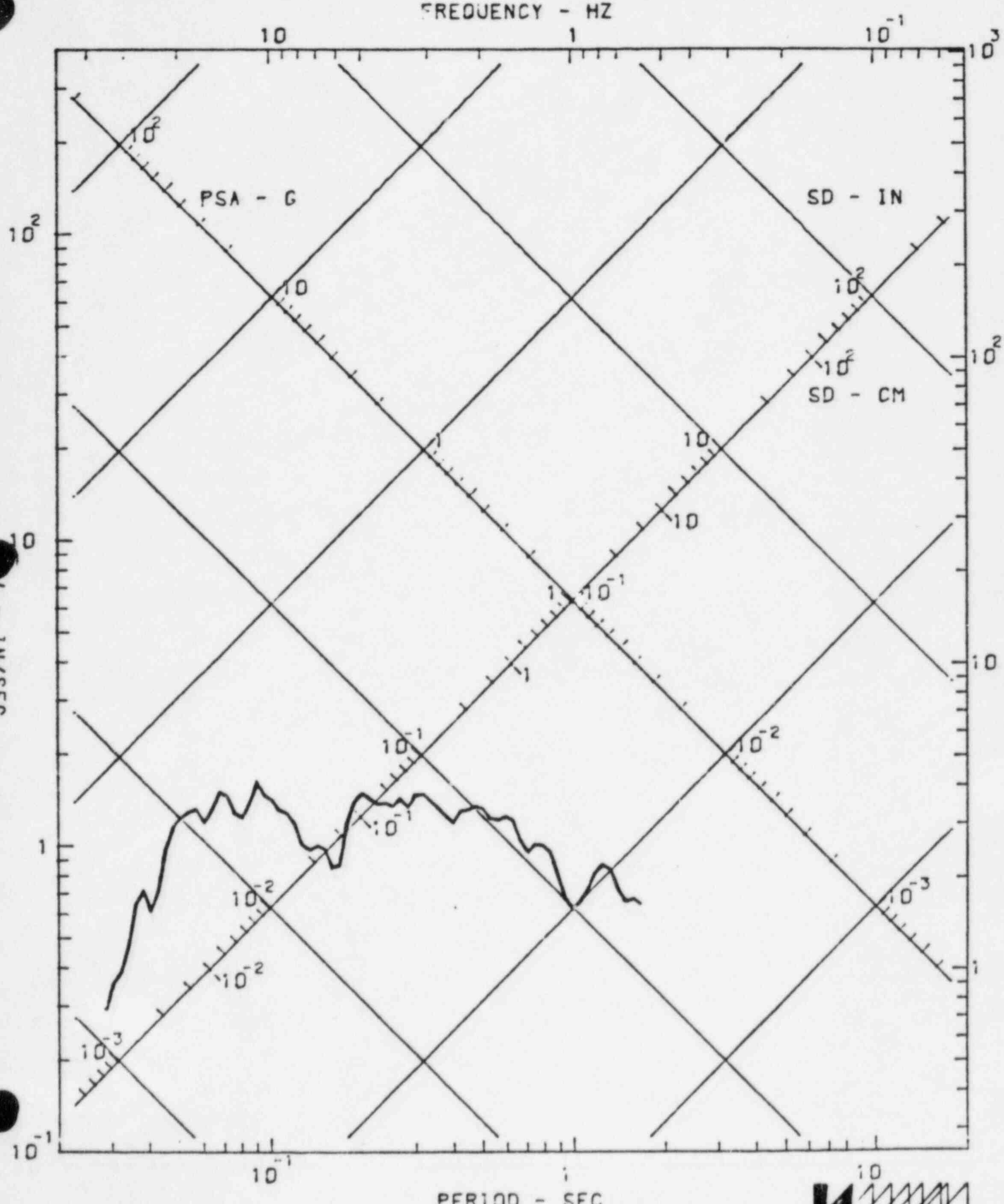

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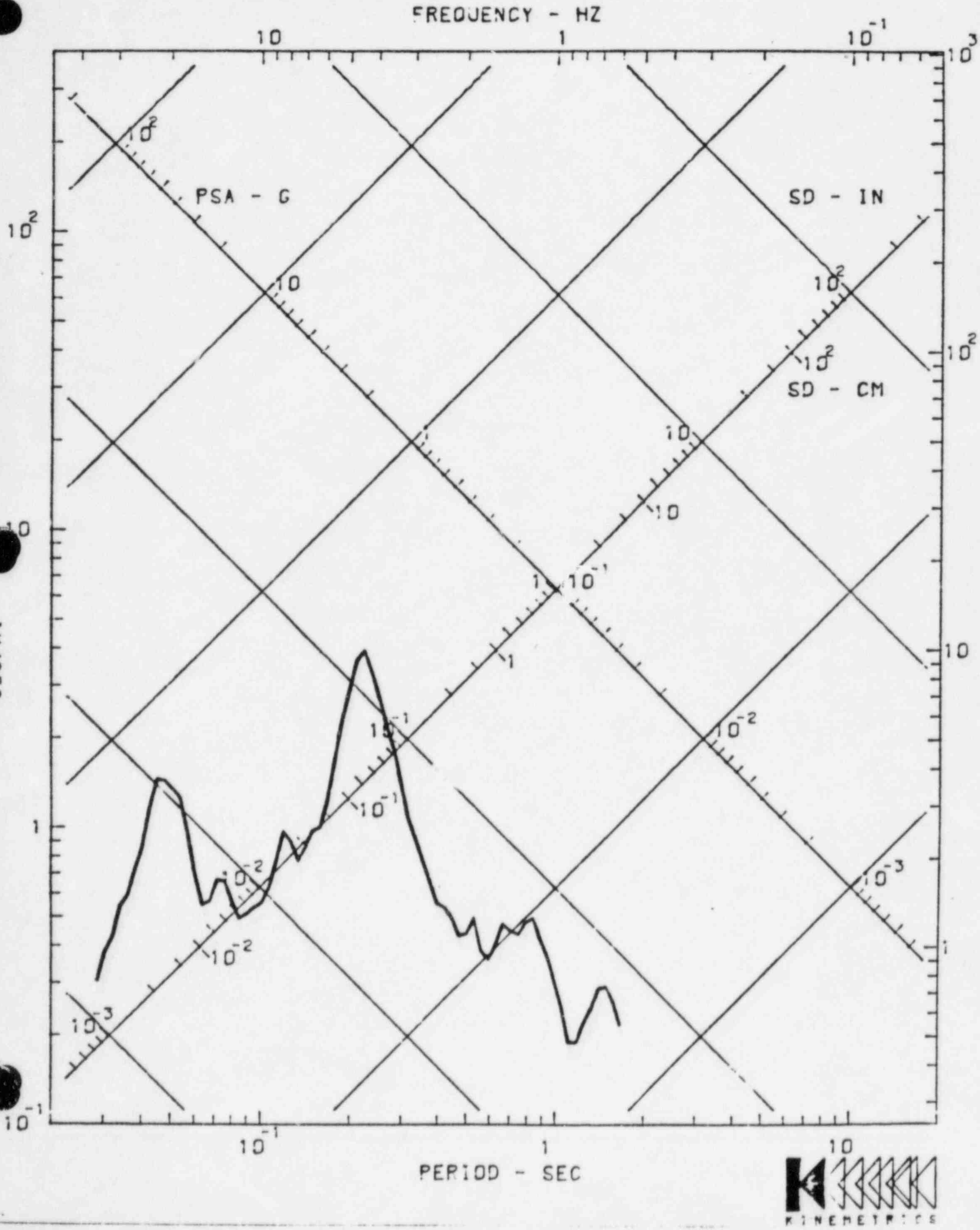
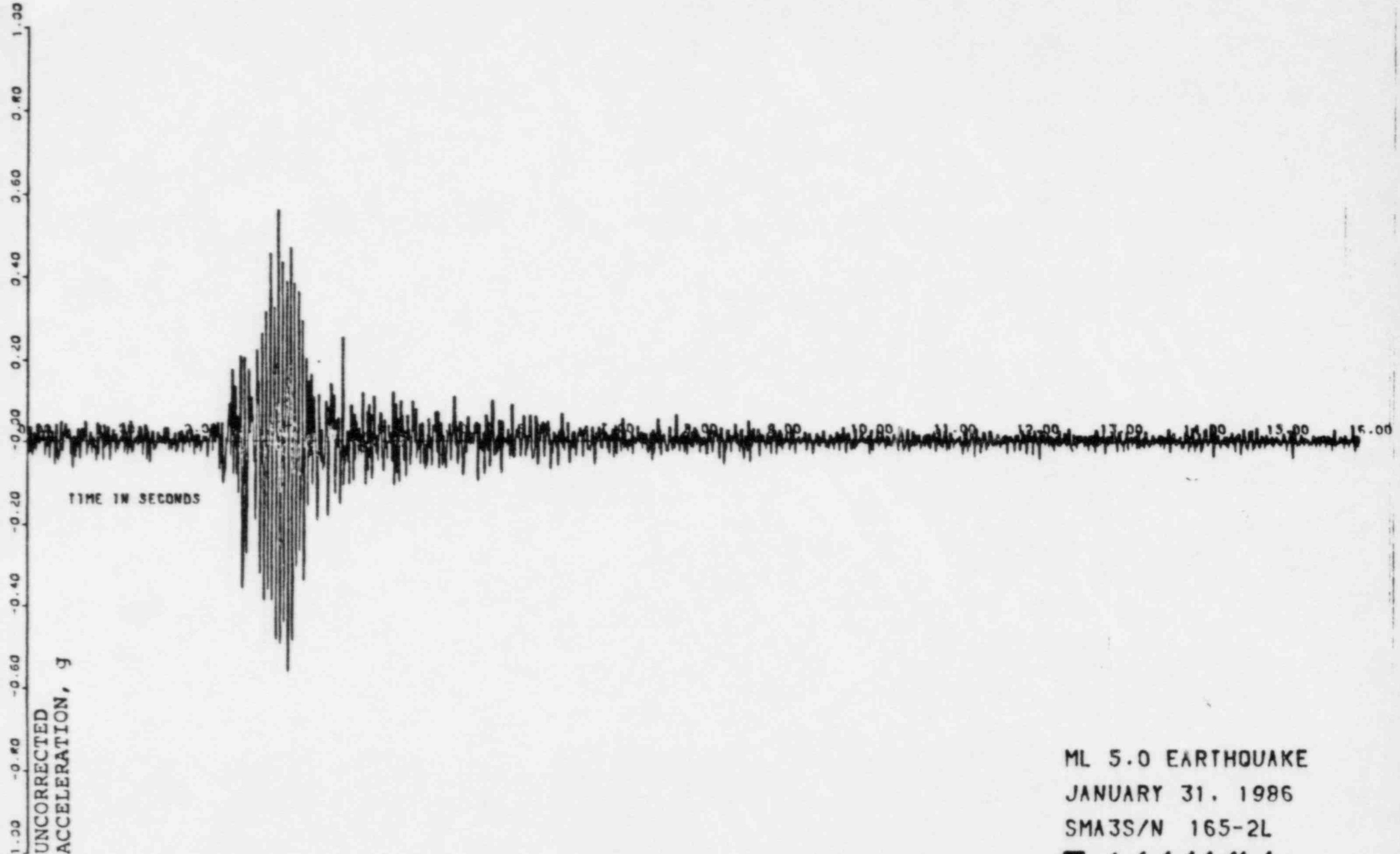
DAMPING VALUES ARE 4 PERCENT OF CRITICAL



Containment Vessel Annulus, Elevation 682 Ft.

SMA-3 Serial Number 165-2
Tag Number D51-N111
Longitudinal Channel - South Orientation
Transverse Channel - West Orientation
Vertical Channel - Up Orientation



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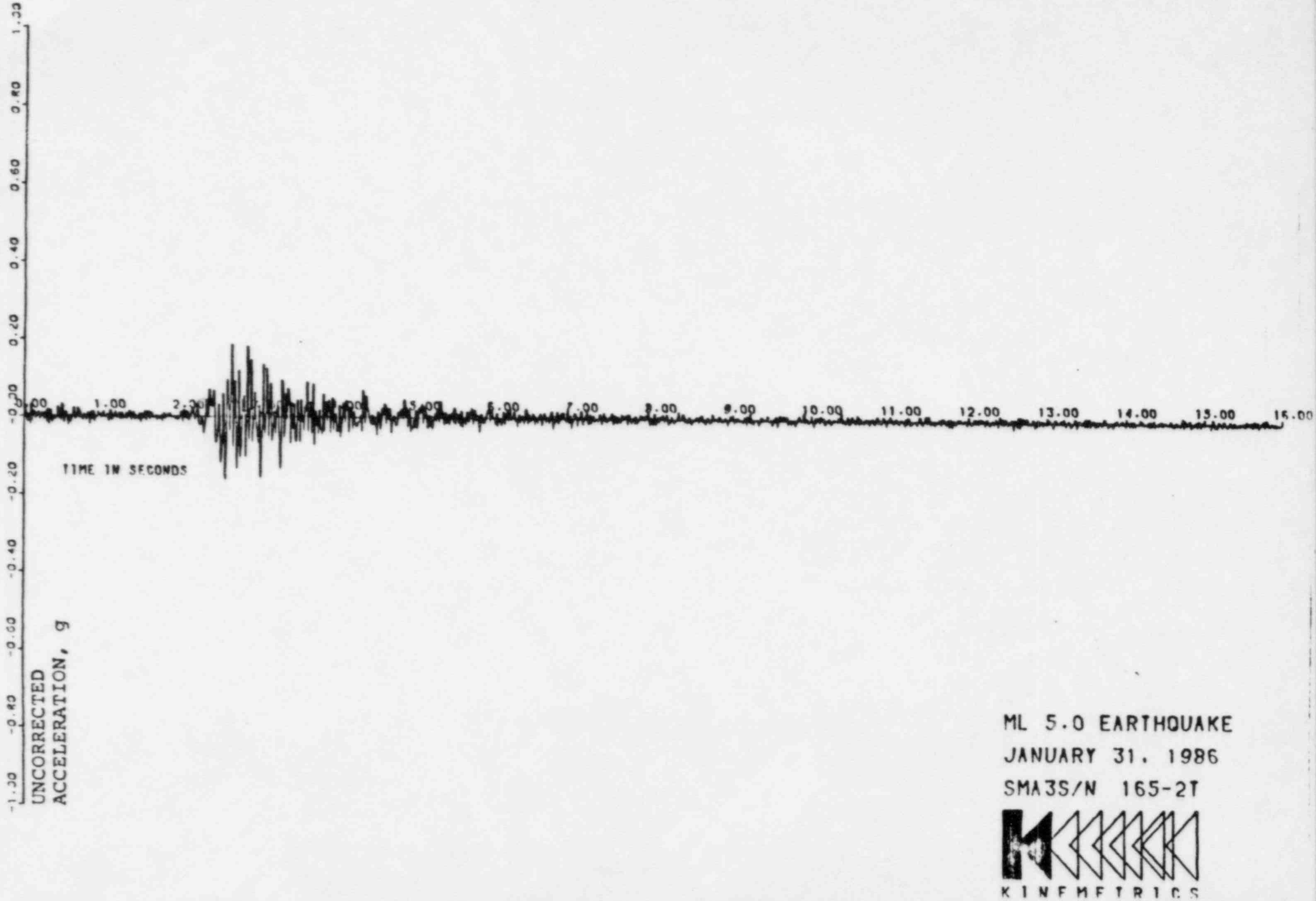
JANUARY 31, 1986
SMA3S/N 165-2L

KINEMETRICS



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ACCELERATION, g
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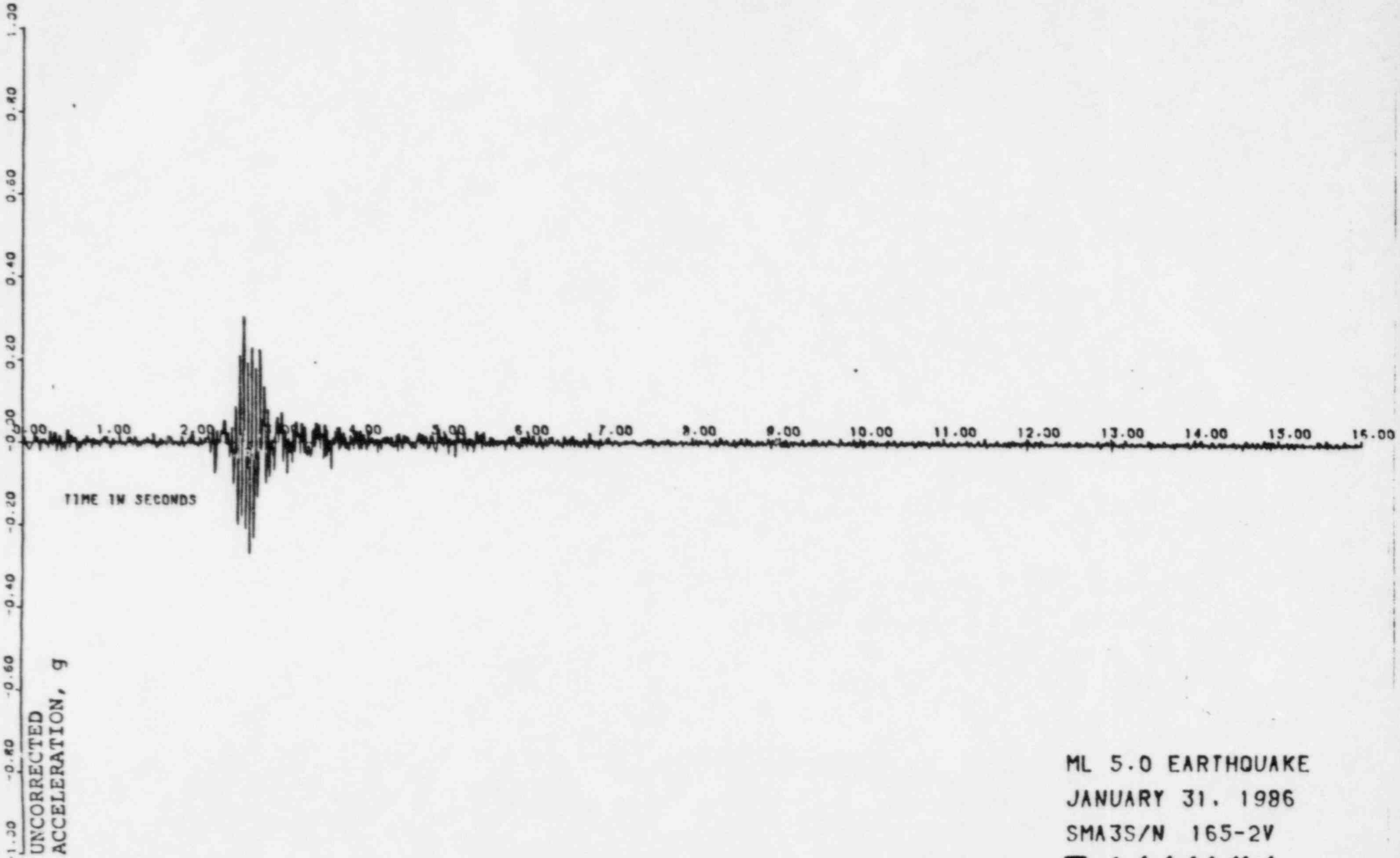

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JANUARY 31, 1986
SMA3S/N 165-2V

KINEMETRICS

KINEMETRICS

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8002 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-2L

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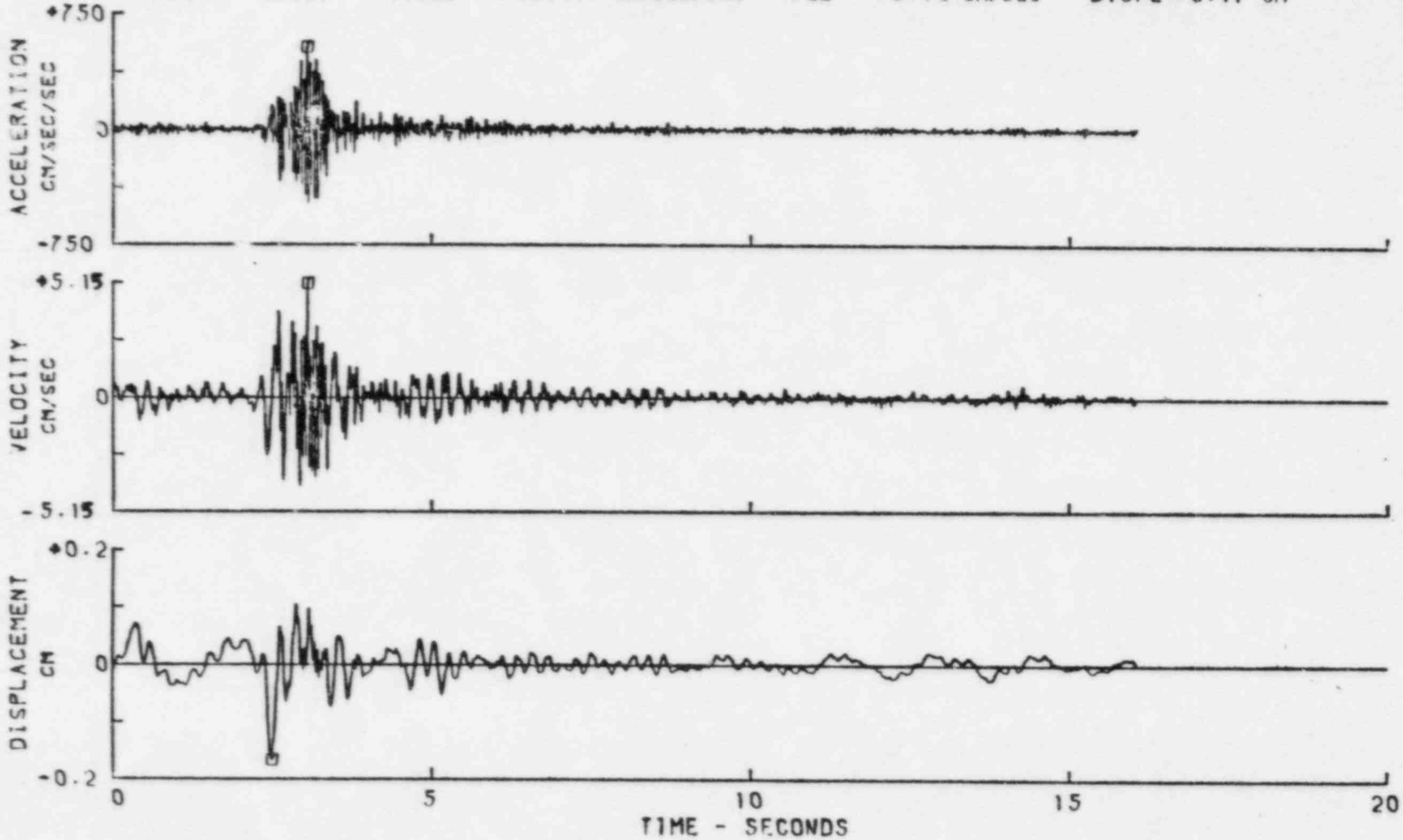

KINEMETRICS

ML 5.0 EARTHQUAKE JANUARY 31, 1986

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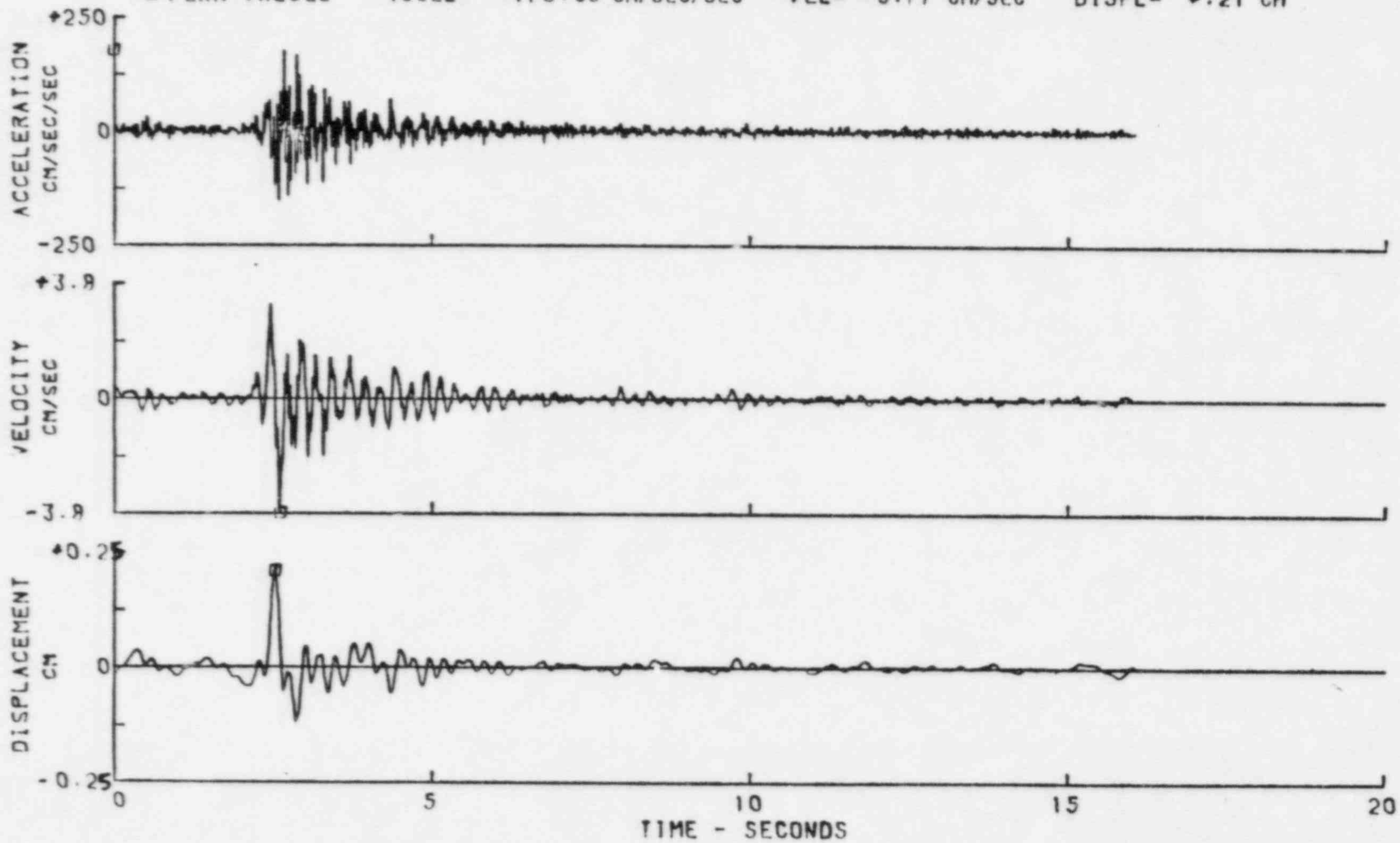

KINEMETRICS

ML 5.0 EARTHQUAKE JANUARY 31, 1986

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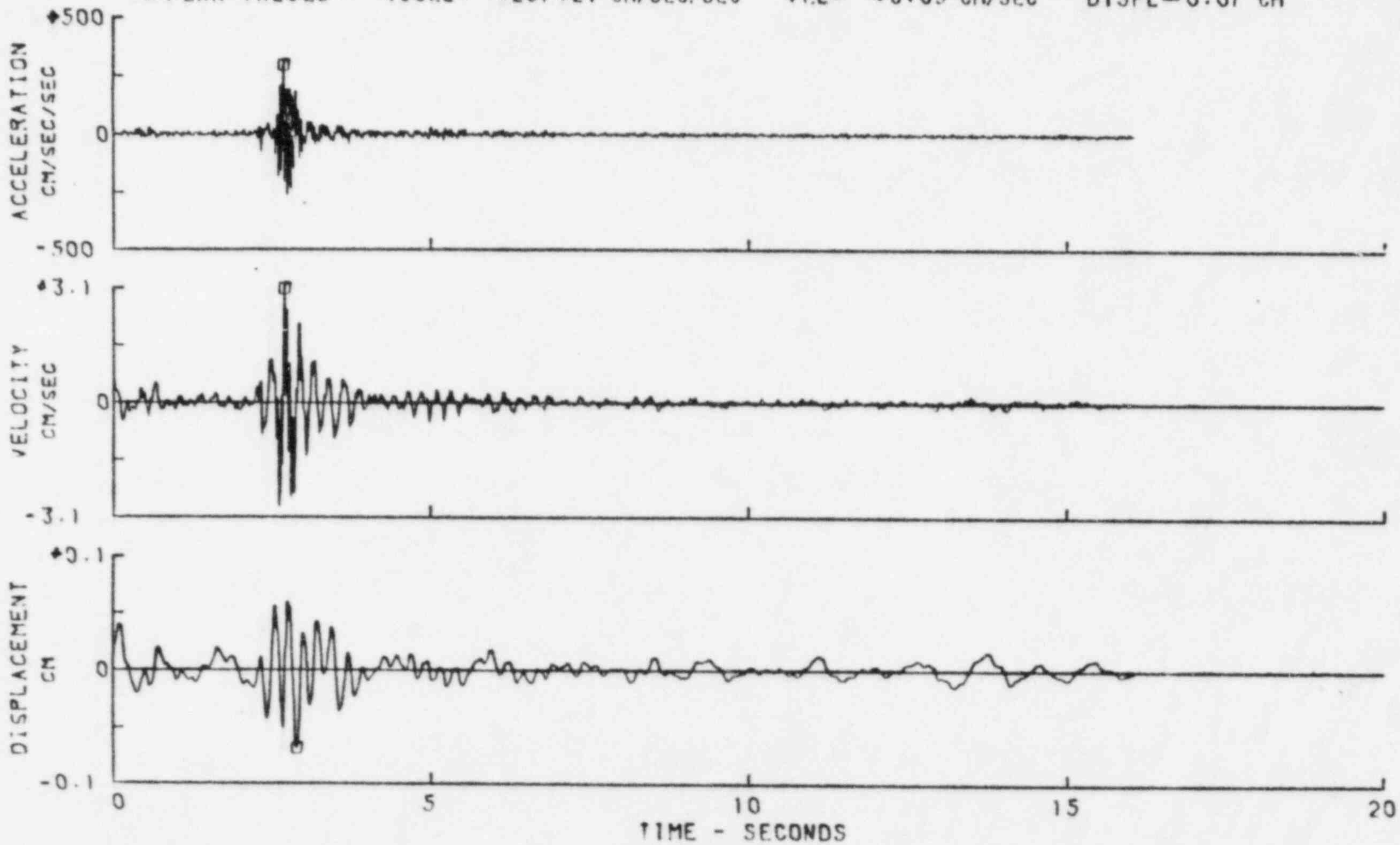

RELATIVE VELOCITY RESPONSE SPECTRUM

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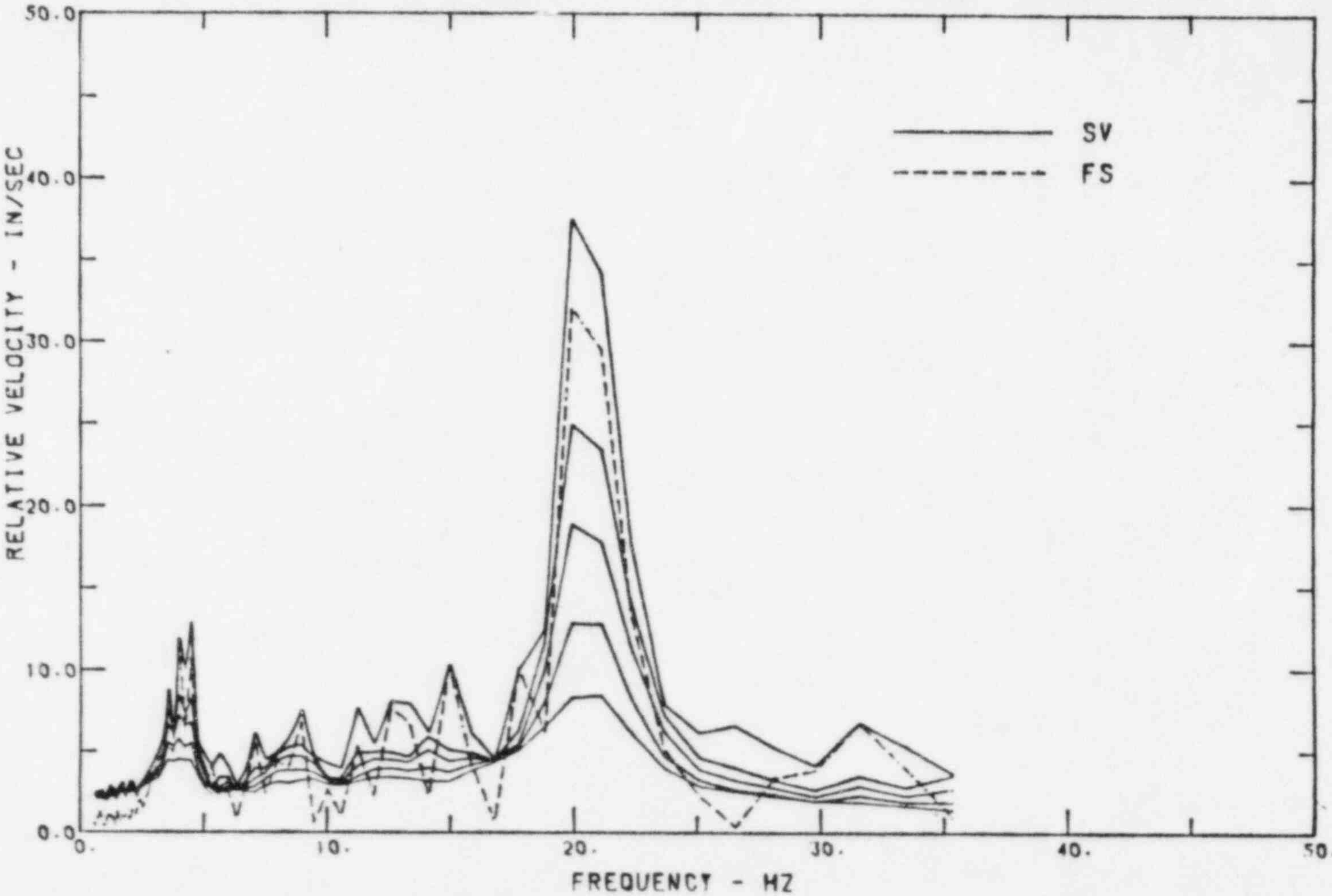

KINEMETRICS

RELATIVE VELOCITY RESPONSE SPECTRUM

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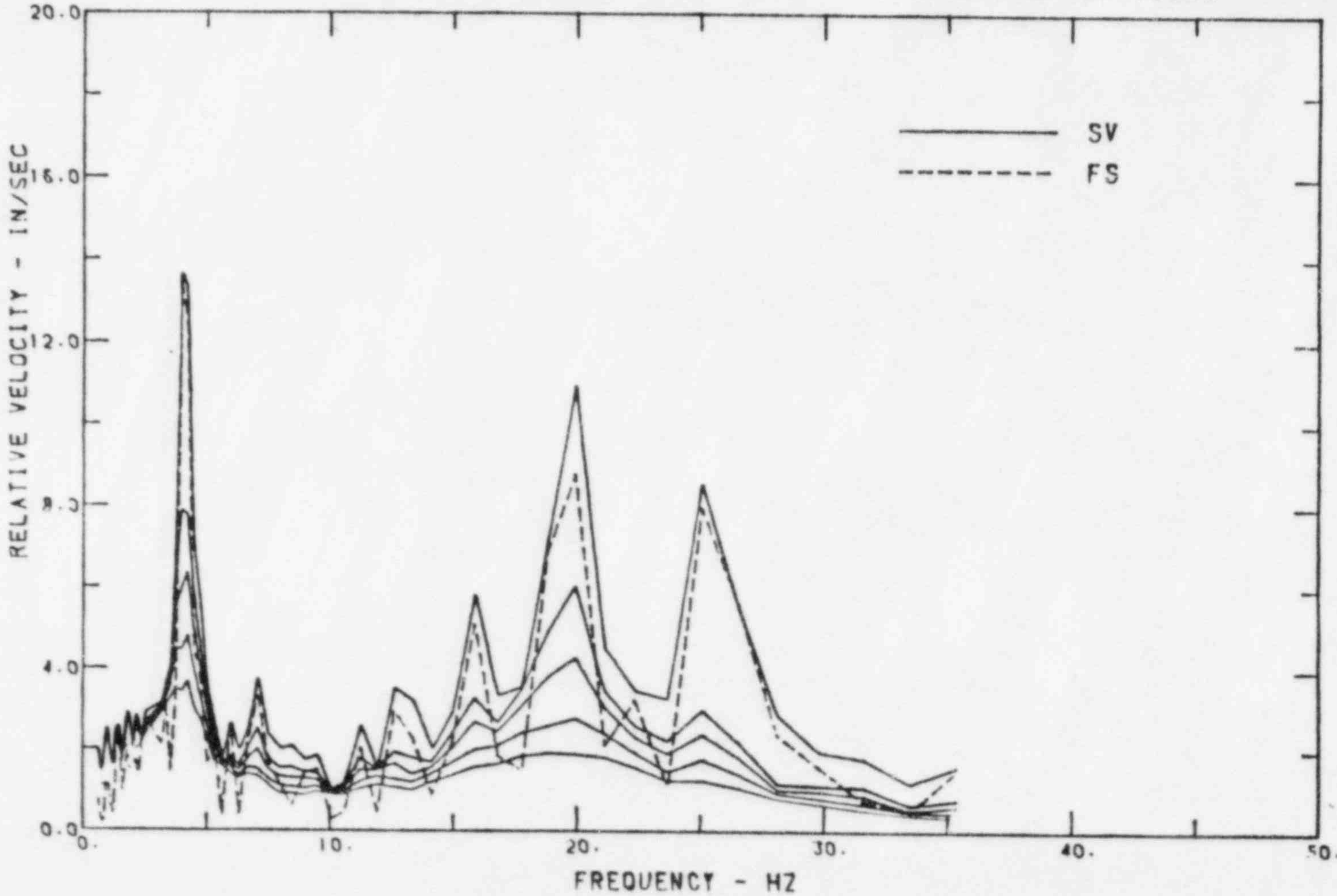

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RELATIVE VELOCITY RESPONSE SPECTRUM

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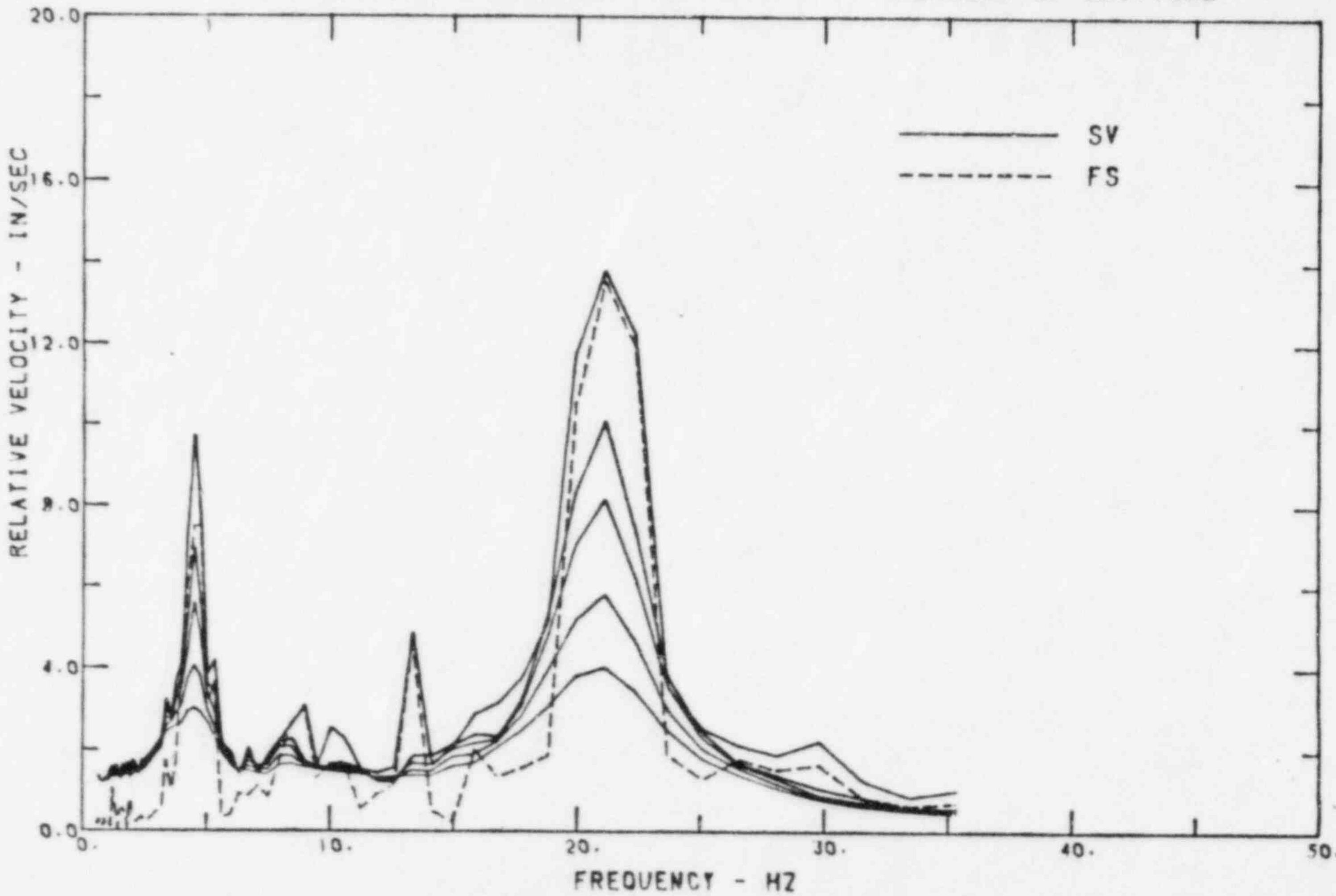

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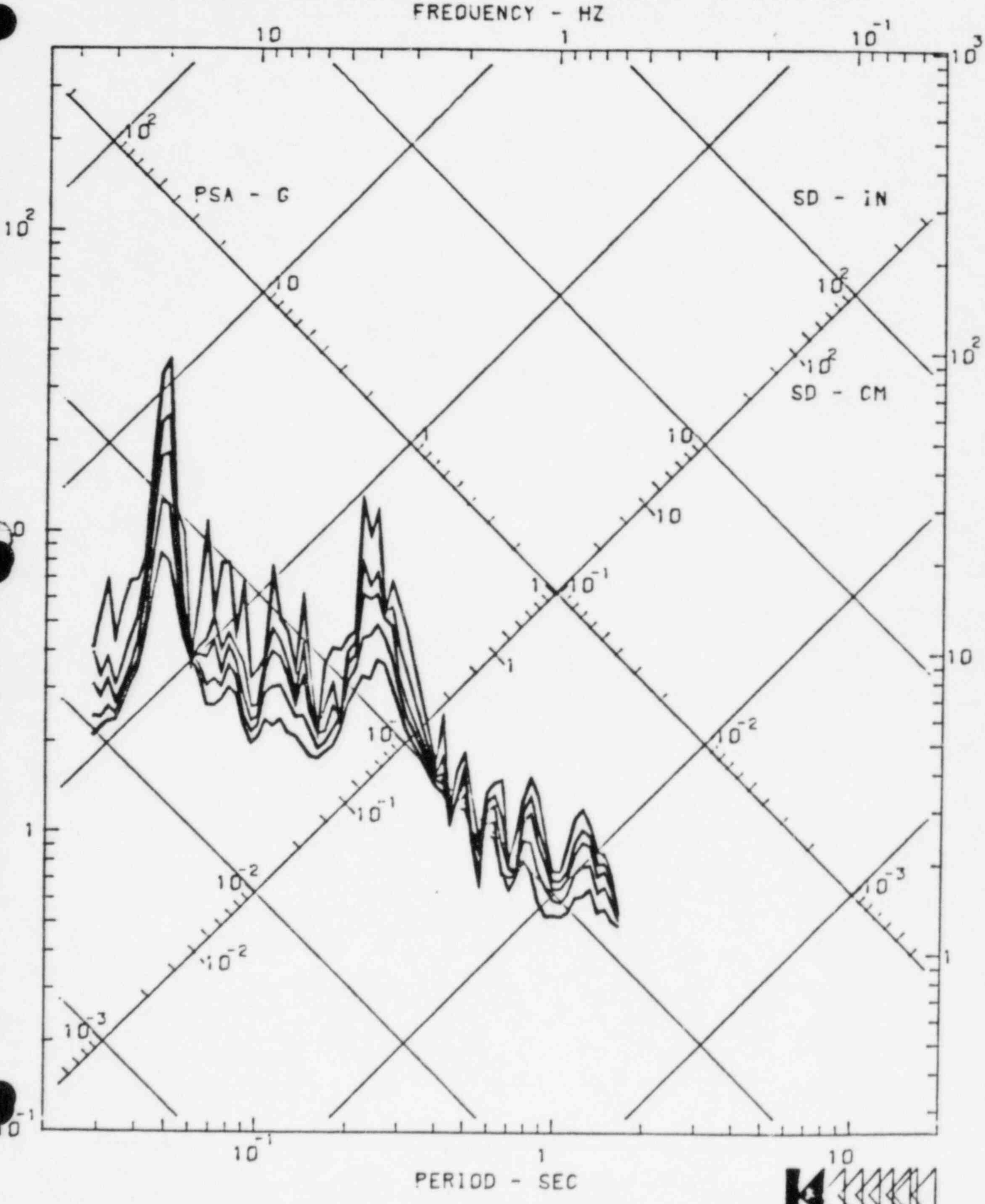

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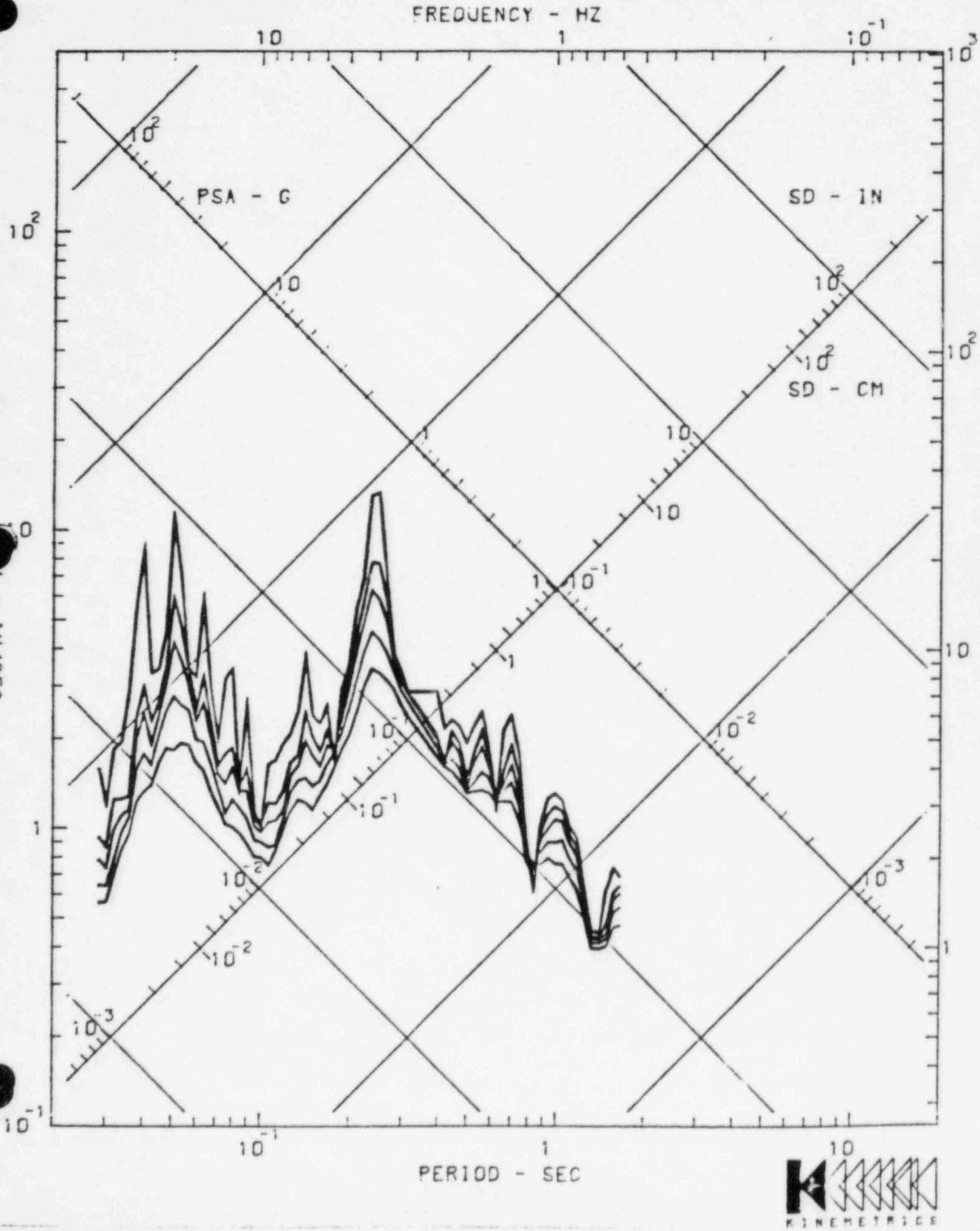
IIAB002 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-2T

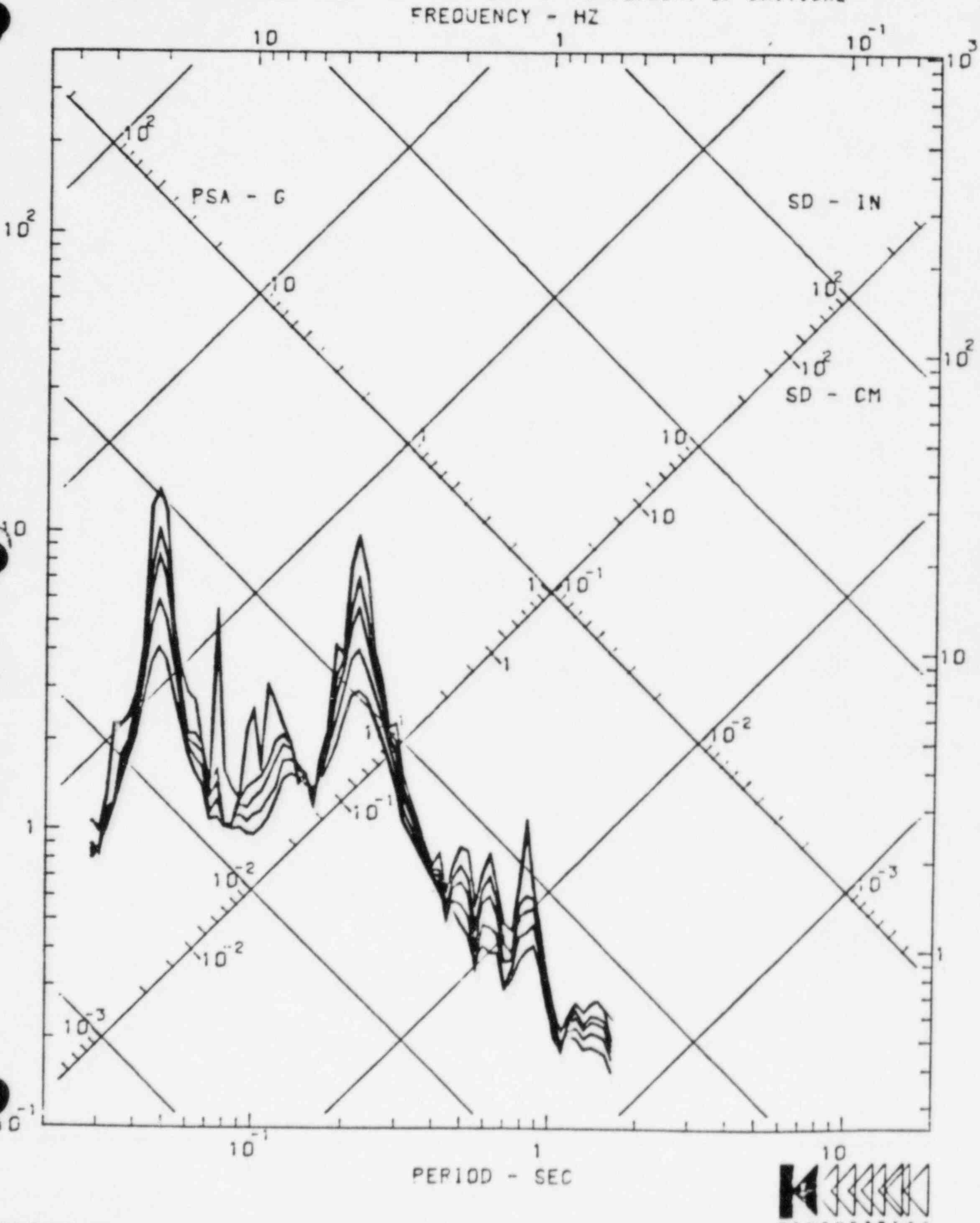
DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8002 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 165-2V

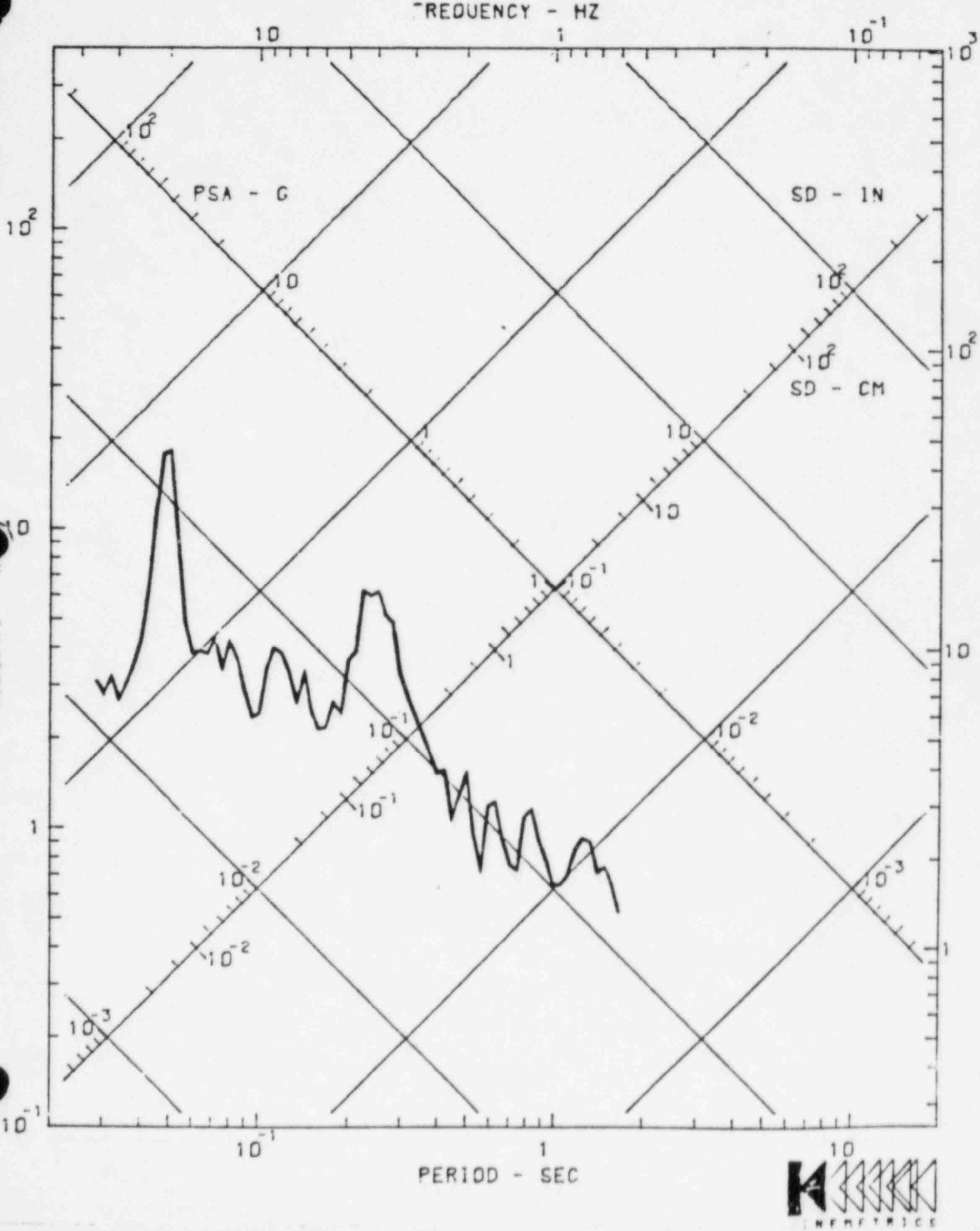
DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8002 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-2L

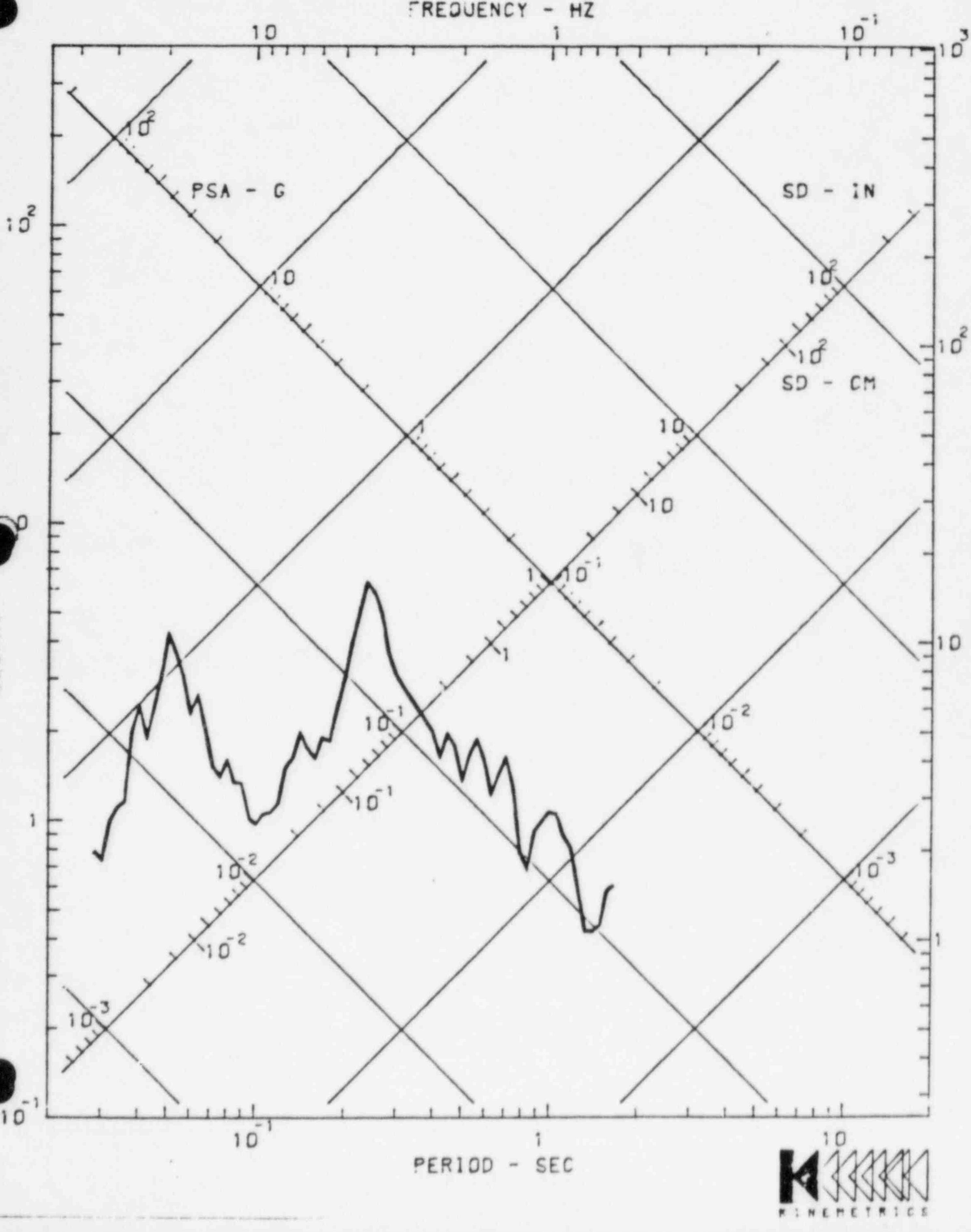
DAMPING VALUES ARE 2 PERCENT OF CRITICAL



ML 5.0 EARTHQUAKE JANUARY 31. 1986

11A8002 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-21

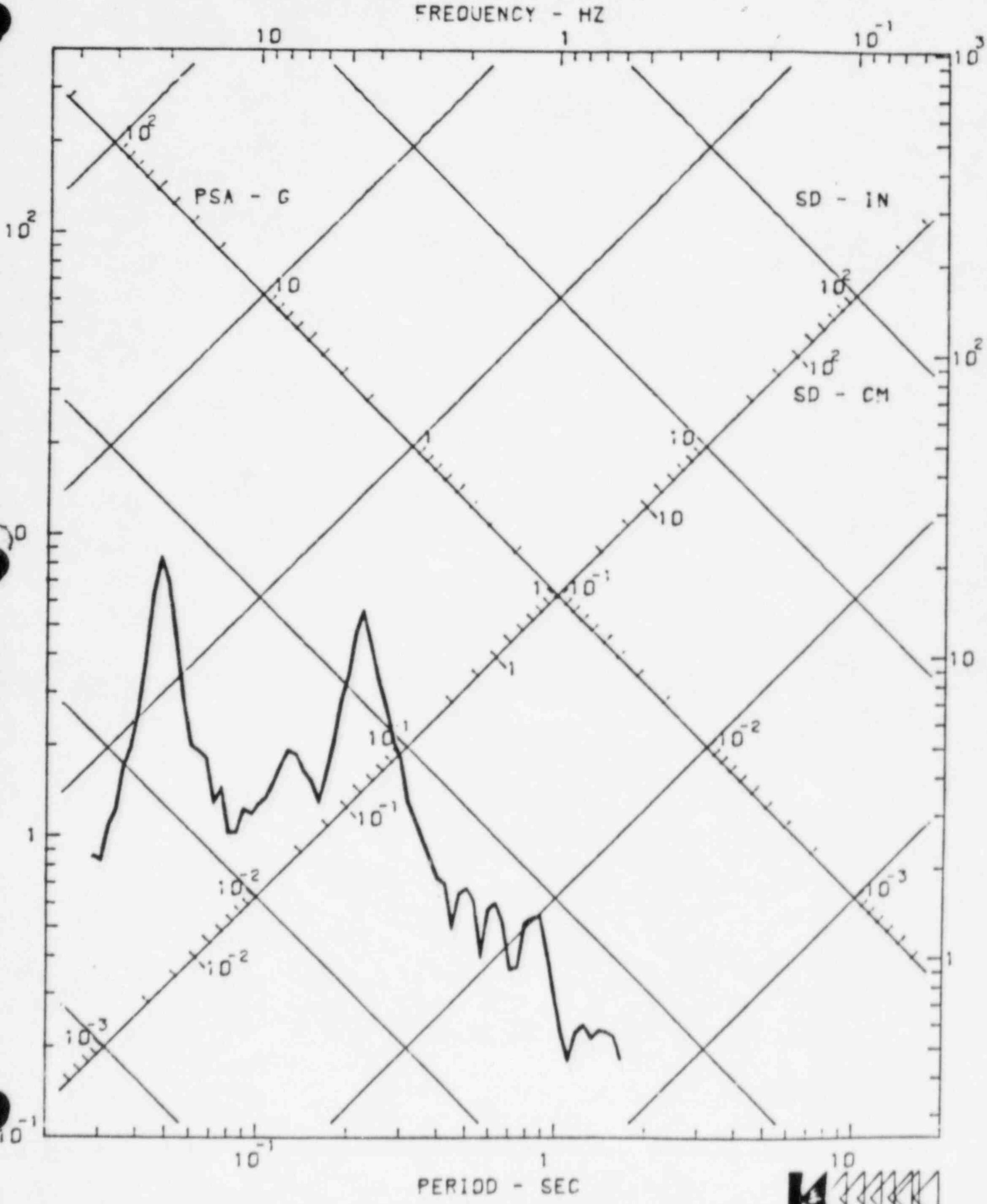
DAMPING VALUES ARE 2 PERCENT OF CRITICAL



ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8002 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 165-2V

DAMPING VALUES ARE 2 PERCENT OF CRITICAL

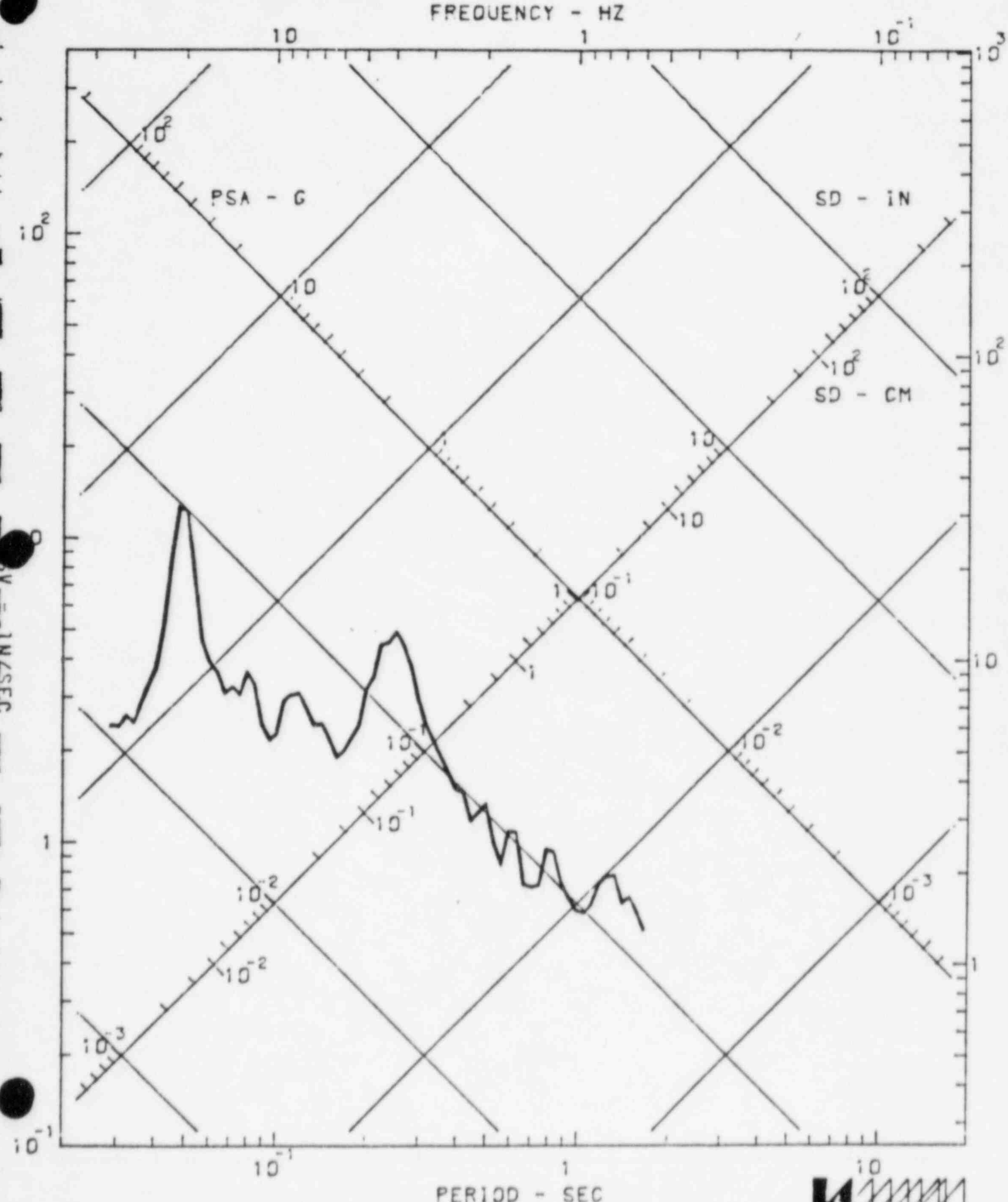


KINEMETRICS

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11AB002 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-2L

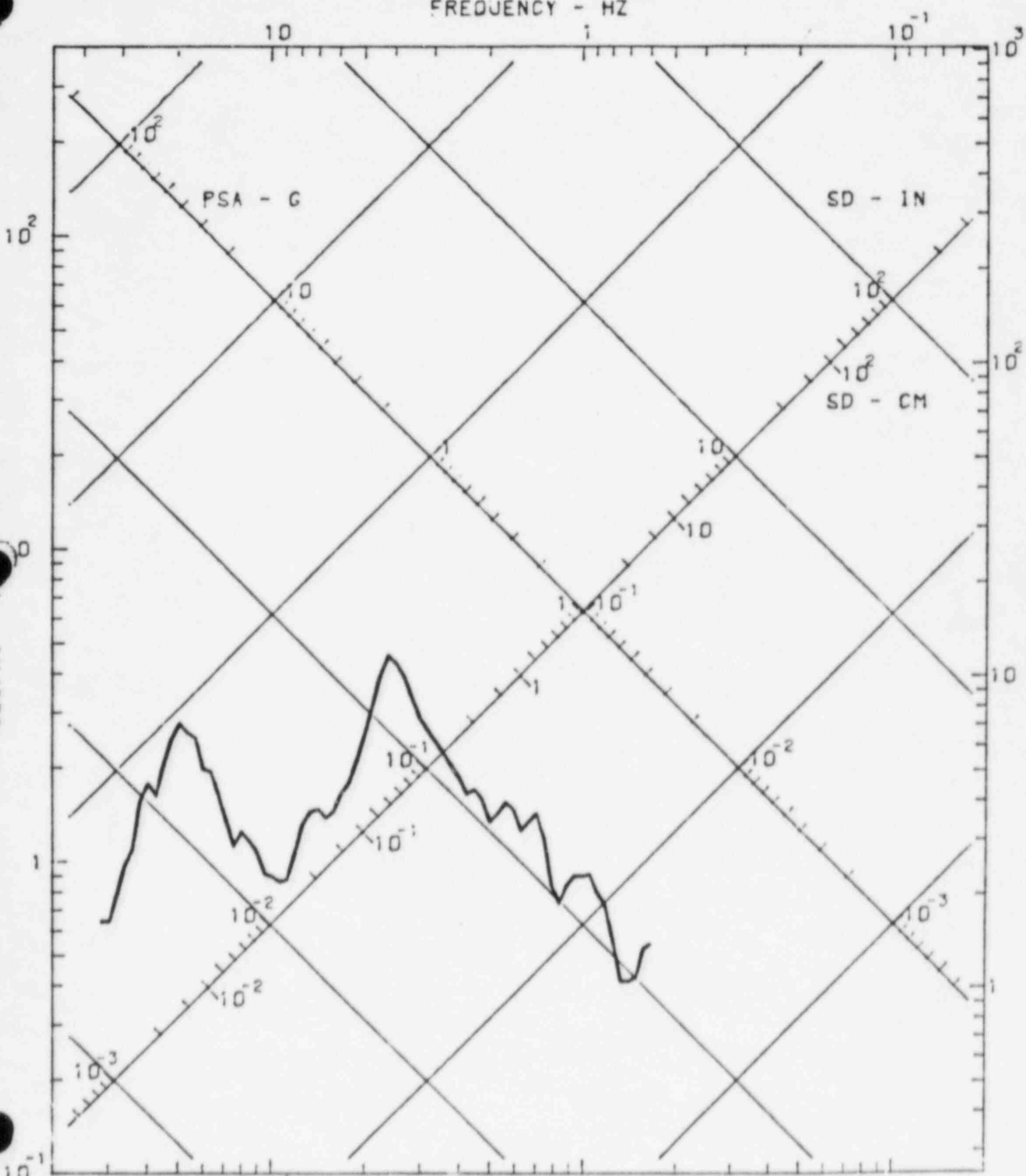
DAMPING VALUES ARE 4 PERCENT OF CRITICAL



ML 5.0 EARTHQUAKE JANUARY 31, 1986

IIA8002 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-27

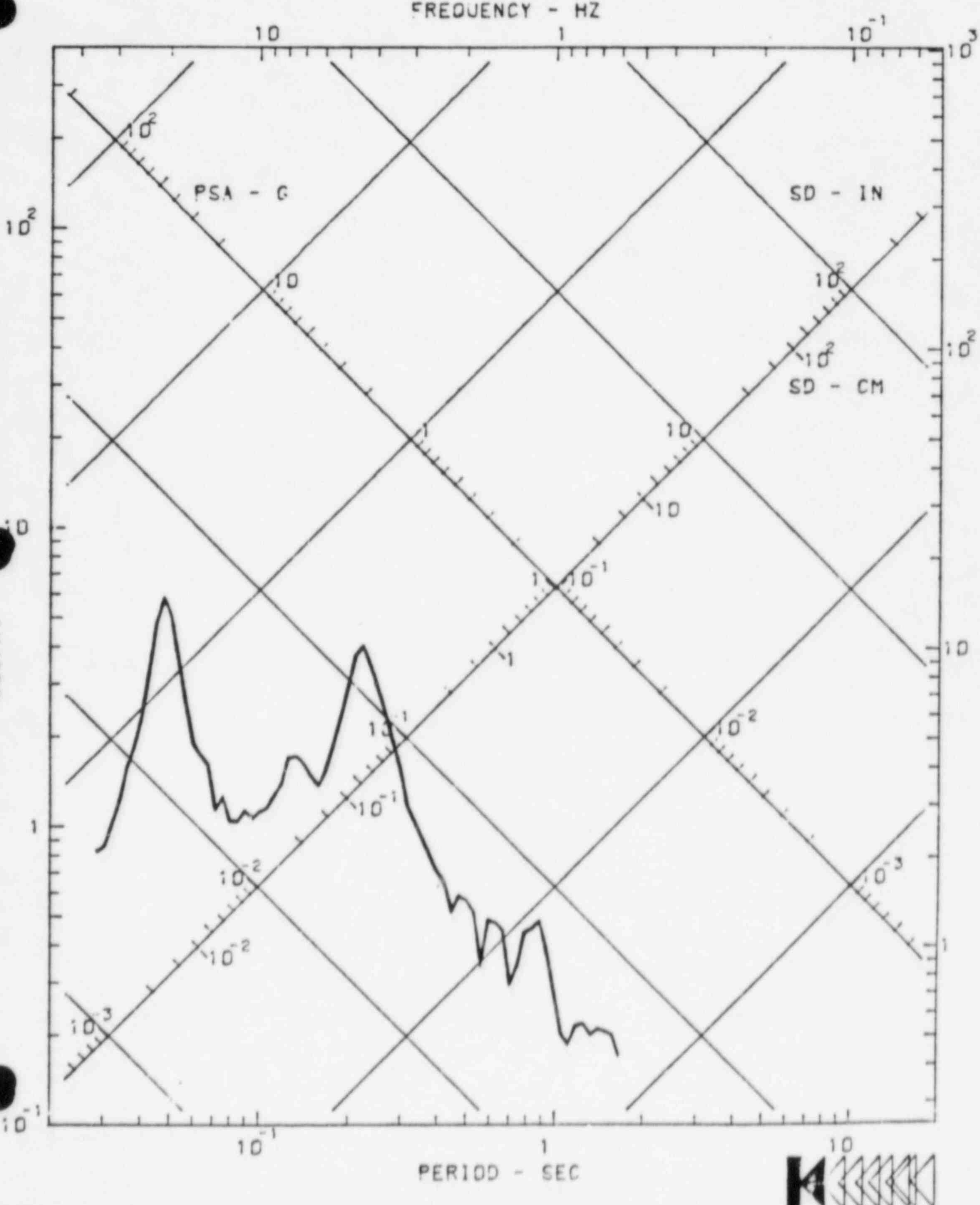
DAMPING VALUES ARE 4 PERCENT OF CRITICAL



ML 5.0 EARTHQUAKE JANUARY 31, 1986

11AB002 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 165-2V

DAMPING VALUES ARE 4 PERCENT OF CRITICAL



APPLICATION NOTE

Conditioning and Correction of Strong Motion Data on Analog Magnetic Tapes

No. 7

Kinemetrics has developed programs for routine computer processing of data recorded on the analog magnetic tape accelerographs, Models SMA-2 and SMA-3. The software from published research for film recording accelerographs (Trifunac & Lee, 1973) has been adapted to the analog magnetic tape recording instruments.

Magnetic tape is used where rapid playback and analysis of data are required. These accelerographs are normally located at large engineered facilities, such as nuclear power plants. Figure 1, "Kinemetrics Earthquake Data Reduction System Flow Diagram," illustrates the specialized services needed to prepare data immediately after an earthquake.

The purpose of this Note is to describe the standard data conditioning and correction used to prepare accelerograms for subsequent response spectrum or time-series analysis. On Figure 1 are references to the following paragraphs: 1.0--Data Playback, 2.0--Analog-to-Digital Conversion, 3.0--Data Conditioning, and 4.0--Data Correction.

There are two "tape speed" errors in all FM analog recording/playback systems. One "error" is a change in apparent amplitude due to unwanted tape speed changes. Correction of this error is called "amplitude compensation". This is shown in Figure 2 and described in Sections 1.0 and 3.0. The second "error" is a change in apparent length of the earthquake due to different tape speeds during recording and playback. Correction of this error is called "time base compensation". This is shown in Figure 3 and described in Section 2.0.

1.0 Data Playback

1.1 The playback system is a Model SMP-1 (Figure 4). If the SMP-1 is used to play out the SMA-2 or SMA-3 tapes, the signals

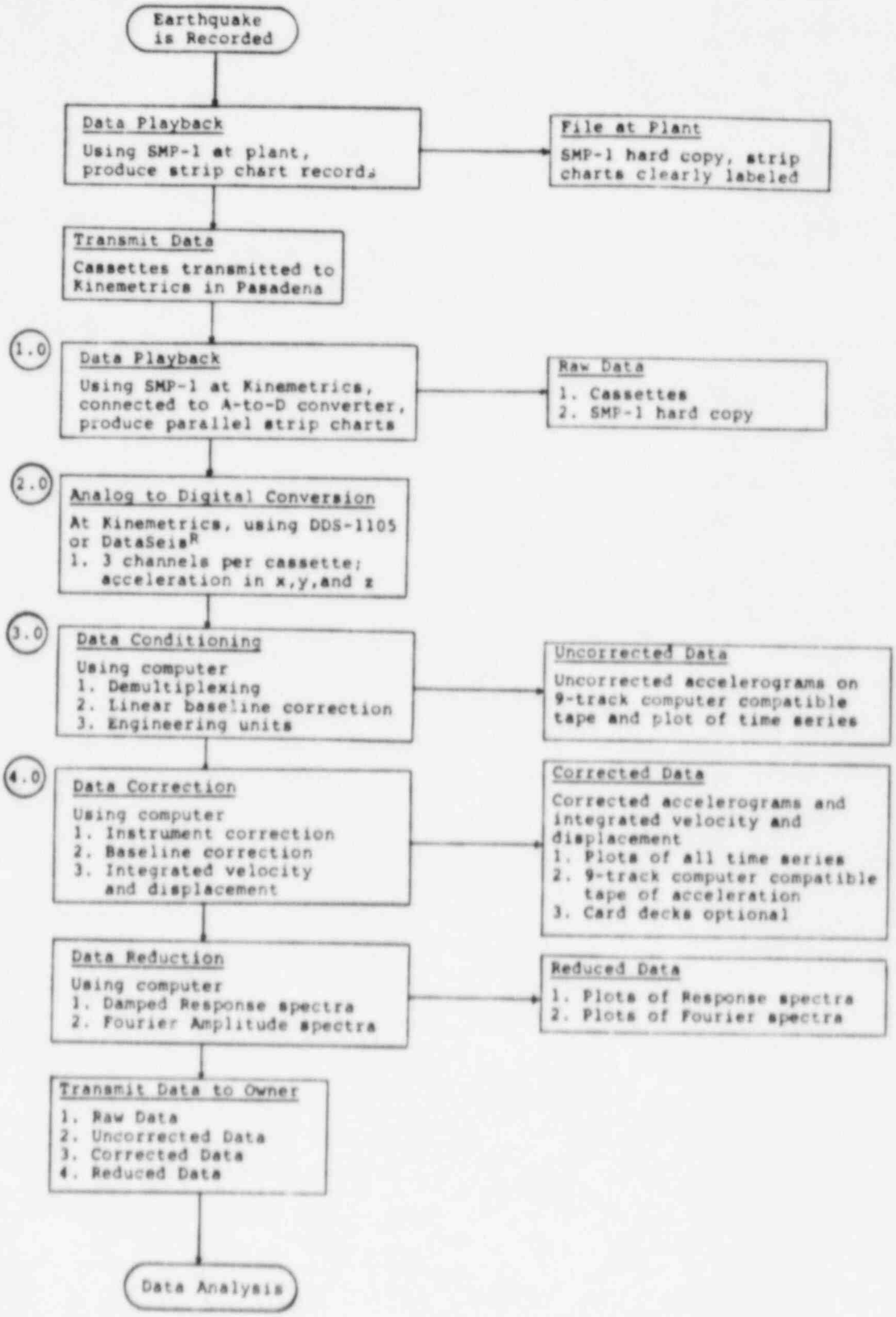


FIGURE 1 Flow Diagram for Kinemetrics E.D.R.S. (Earthquake Data Reduction Sequence)

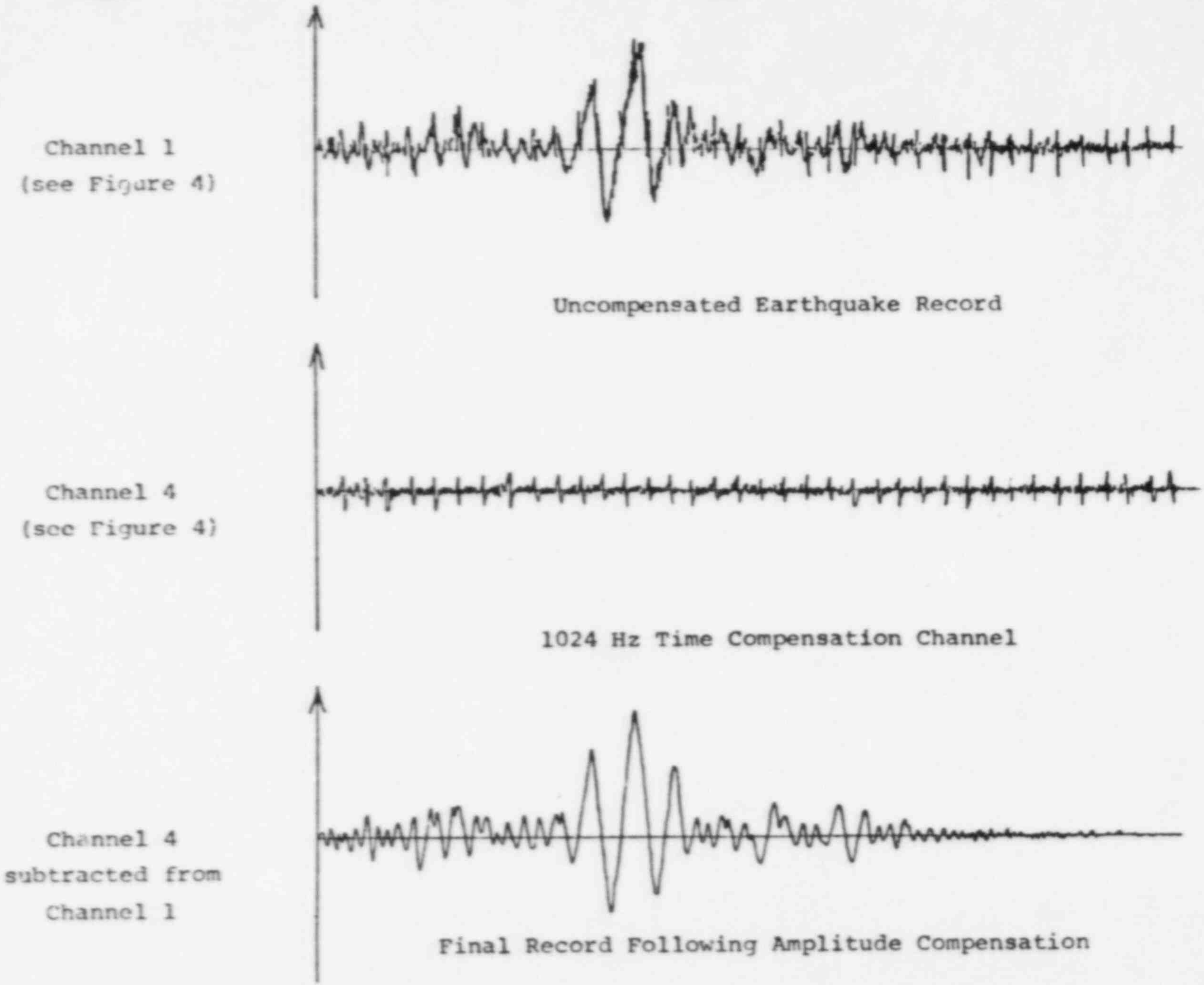


FIGURE 2 Amplitude Compensation

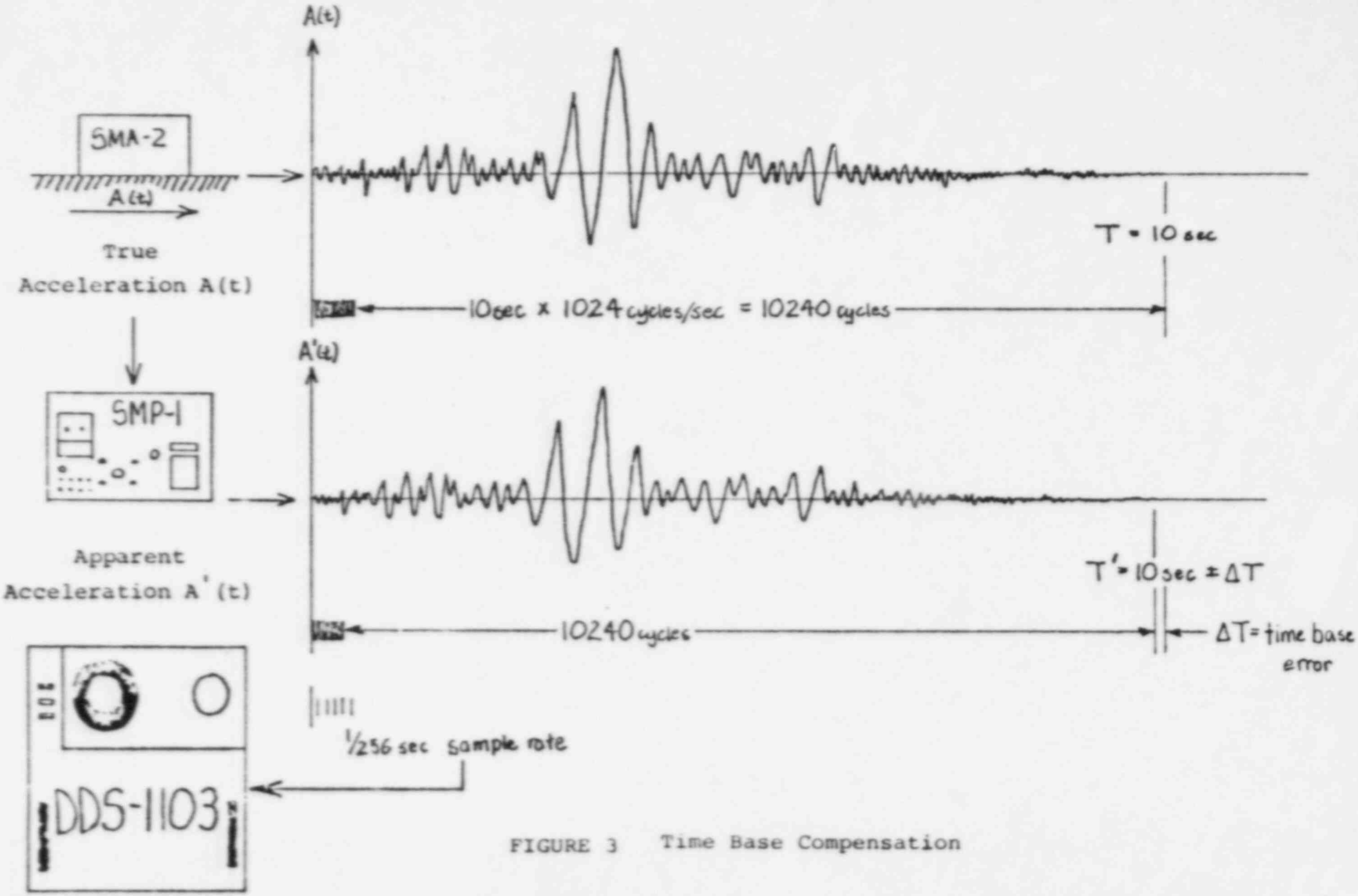
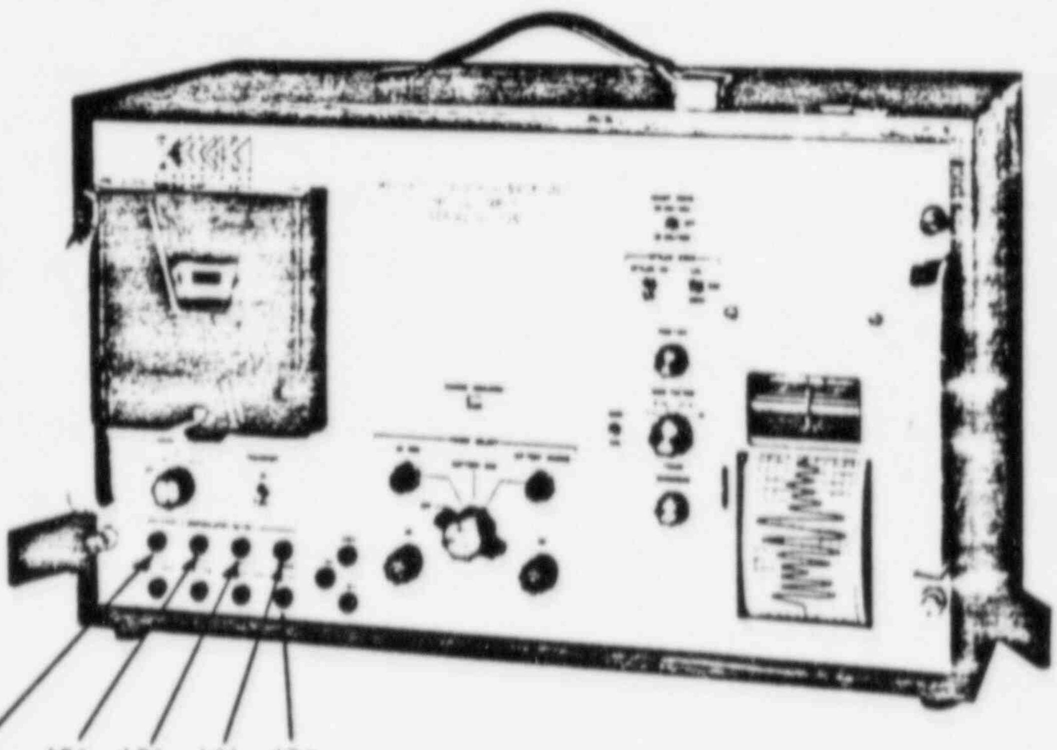


FIGURE 3 Time Base Compensation

SMP-1

Magnetic Tape Playback System

FIGURE 4



(1) (2) (3) (4) (5)

The SMP-1 is a versatile magnetic tape playback system designed for use with the Kinemetrics SMA-2 and SMA-3 Magnetic Tape Acceleration Systems. The combination of the SMA-2 or SMA-3 Acceleration Systems with the SMP-1 Magnetic Tape Playback System meets the applicable requirements of US NRC Regulatory Guide 1.12, and provides immediate visual playback capability of recorded acceleration data.

The SMP-1 is portable and may be operated either from 110 Vac or internal rechargeable batteries. Optionally the unit may be mounted in a standard 19-inch cabinet. An internal battery charger is included with the unit.

which appear on the chart recorder are amplitude compensated.

1.2 The electrical outputs taken from the DEMODULATED OUTPUT jacks (Channels 1, 2, or 3 of Figure 4) are not amplitude compensated. However, Kinemetrics has an electronic Data Compensator which plugs into an SMP-1.

If this Data Compensator is used, the electrical signals are amplitude compensated by electronic subtraction of Channel 4 from Channels 1, 2, and 3. The Data Compensator should be used if the signals are to be recorded on a three-channel strip-chart recorder for display. The signals are not time base compensated.

1.3 If the signals are to be processed on a computer, there are two options:

1.3.1 Use the Data Compensator for amplitude compensation.

1.3.2 Without a Data Compensator, have software perform amplitude compensation.

2.0 Analog-to-Digital Conversion

The following steps are taken at Kinemetrics using the SMP-1 connected to the Analog-to-Digital Converter, Model DDS-1105 or DataSeis^R.

2.1 Three (3) analog outputs of the SMP-1 with Data Compensator are digitized simultaneously: longitudinal, transverse, and vertical (Channels 1, 2, 3 of Figure 4). A 12-bit analog-to-digital converter is used with normal full scale of ±5 volts.

2.2 The FM Time reference output (Channel 5 of Figure 4) is 1,024 Hz plus or minus tape speed error. This signal is divided down by four (256 Hz ± deviation) and used as the timing signal for the digital conversion time interval. Thus, the accelerogram time base is corrected for tape speed error and the voltage values are equally spaced at 1/256 second. This is "time base compensation" and can be done on analog-to-digital converters other than DDS-1105 or DataSeis^R.

2.3 The final uncorrected accelerograms are written on 9-track computer-compatible tape. The three channels are

multiplexed (i.e., 1, 2, 3, 1, 2, 3, 1, 2,...), and are in a 16-bit, offset binary format.

3.0 Data Conditioning

Figure 5 illustrates the flow of the "Data Conditioning" software. Tape speed variations during recording and during playback of FM analog tape change the apparent time base and affect the analog amplitude. The time base has been compensated in the previous section by using the FM time reference output (Channel 5 of Figure 4) as the timing signal for the analog-to-digital converter. The amplitude has been compensated using the Data Compensator module.

The output accelerograms are uncorrected in the sense that no modifications have been introduced which involve any hypothesis of the ground motion character or of the instrument involved.

4.0 Data Correction

Figure 6 illustrates the flow of the "Data Correction" software. The purpose is to present corrected acceleration data and integrated ground velocity and displacement curves in as accurate a form and over as wide a frequency range as is compatible with the original data. The modified data is believed to be the most accurate form of input data feasible to produce from the original record for structural response calculations and for response spectrum determinations.

Instrument correction is introduced to compensate for the accelerometers' frequency response. The Caltech publication EERL 71-05 discusses the approach used. The baseline correction uses an Ormsby high-pass filter. The technique is explained in Caltech publication EERL 70-07.

Figure 7 contains a sample output plot of corrected data for one component of the Santa Barbara earthquake of 13 August 1978, recorded on a SMA-2 accelerograph.

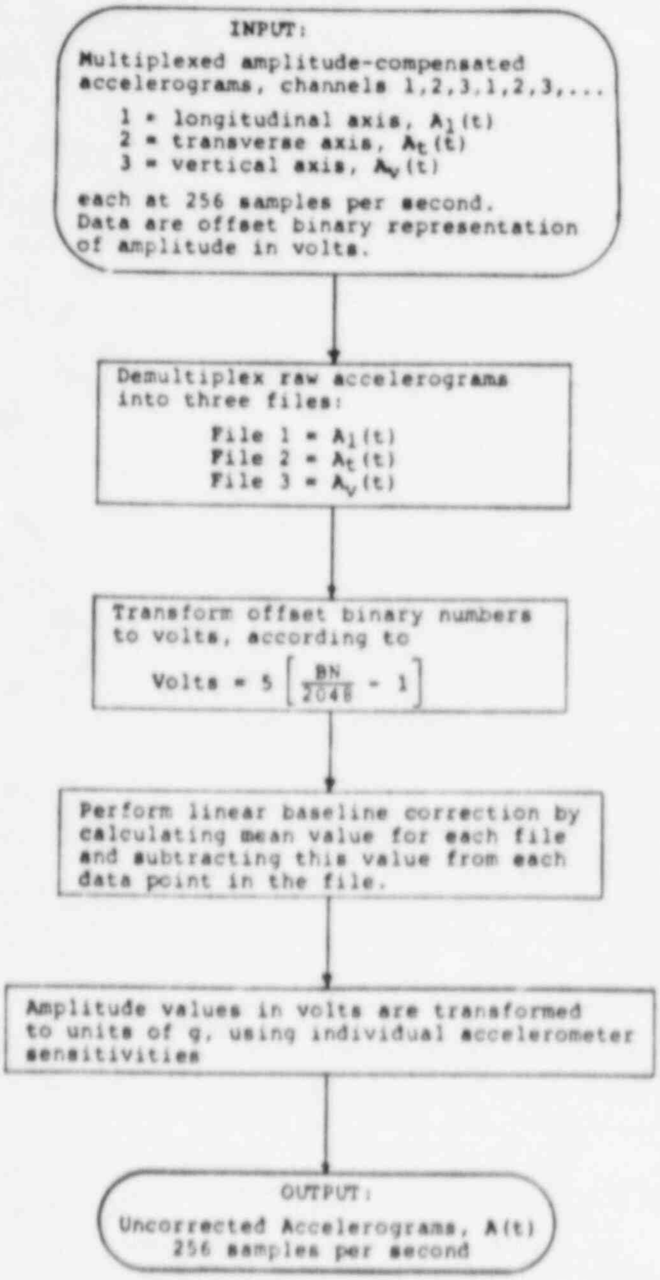


FIGURE 5

Data Conditioning, E.D.R.S.

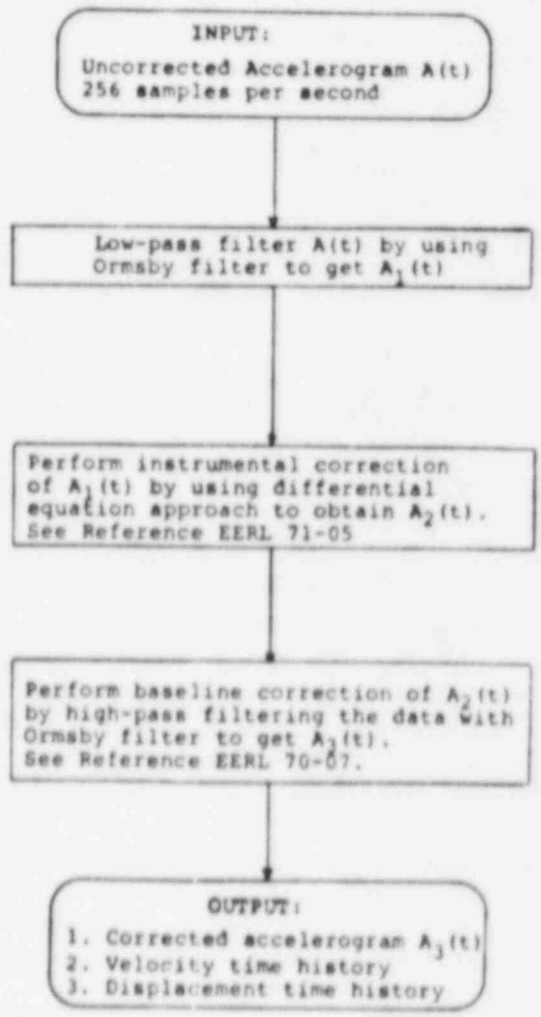
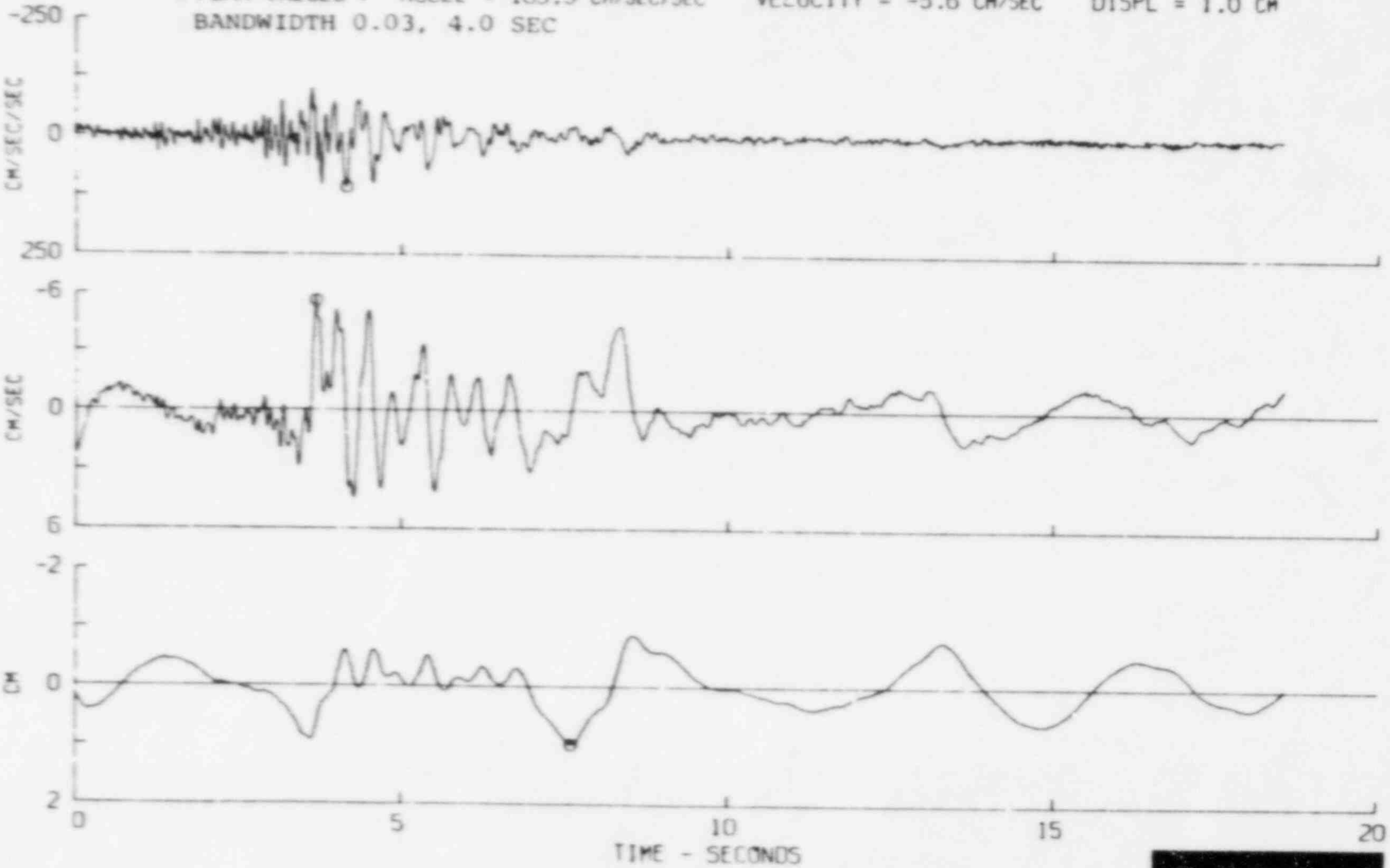


FIGURE 6

Data Correction, E.D.R.S.

FIGURE 7

SANTA BARBARA EARTHQUAKE AUGUST 13, 1978 - 1555 PDT
GOLETA SUBSTATION SCE, 34°28.0'N, 119°53.1'W COMP UP
⊙ PEAK VALUES : ACCEL = 105.9 CM/SEC/SEC VELOCITY = -5.6 CM/SEC DISPL = 1.0 CM
BANDWIDTH 0.03, 4.0 SEC



REFERENCES

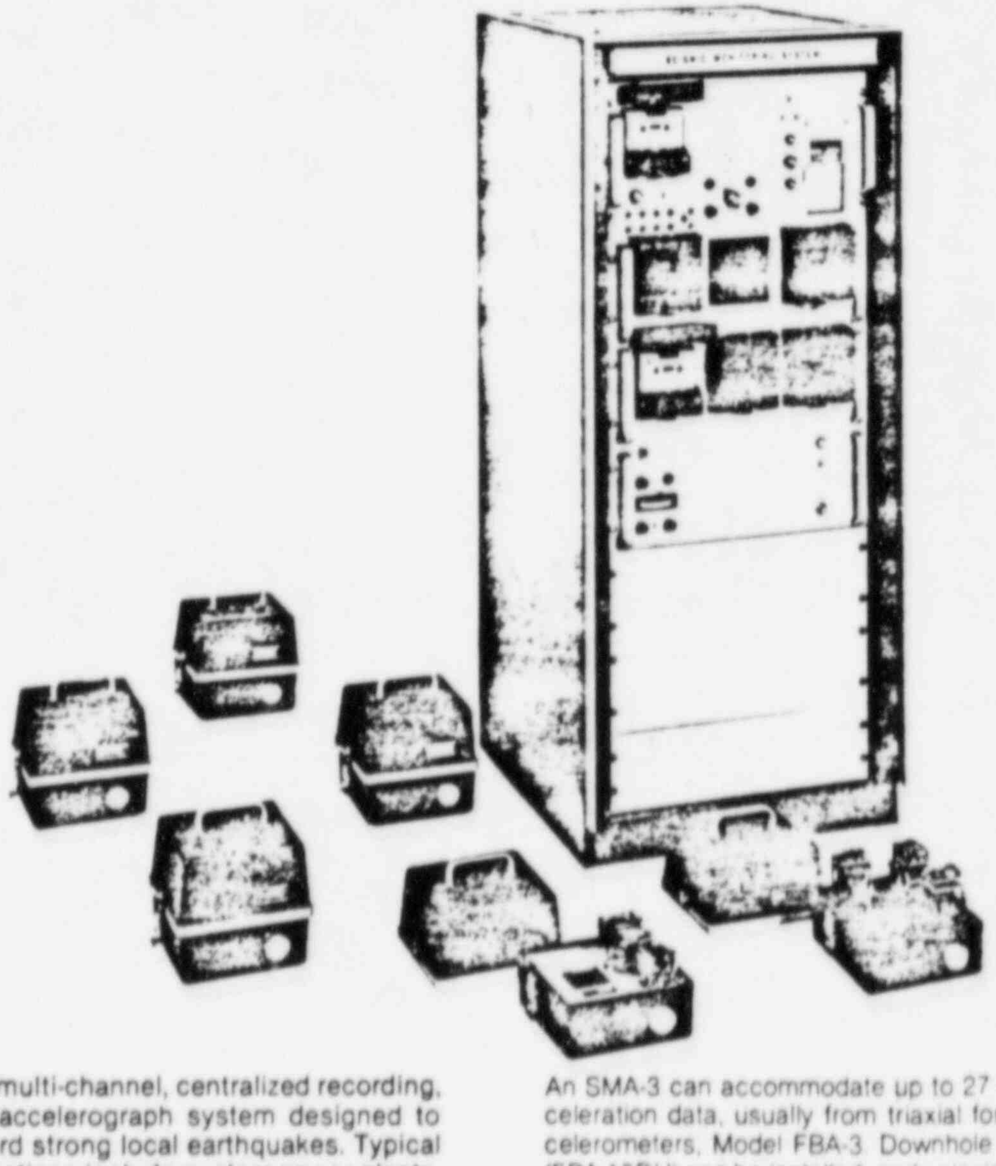
Trifunac, M. D. (1970). Low Frequency Digitization Errors and a New Method for Zero Baseline Correction of Strong-Motion Accelerograms, Earthquake Engineering Research Laboratory, EERL 70-07, pgs. 32-52, California Institute of Technology, Pasadena

Trifunac, M. D., F. E. Udwadia and A. G. Brady (1971). High Frequency Errors and Instrument Corrections of Strong-Motion Accelerograms, Earthquake Engineering Research Laboratory, EERL 71-05, pgs. 33-47, California Institute of Technology, Pasadena

Trifunac, M. D. and V. Lee (1973). Routine Computer Processing of Strong-Motion Accelerograms, Earthquake Engineering Research Laboratory, EERL 73-03, California Institute of Technology, Pasadena

SMA-3

Strong Motion Acceleration System



The SMA-3 is a multi-channel, centralized recording, magnetic tape accelerograph system designed to detect and record strong local earthquakes. Typical structural applications include nuclear power plants, tall buildings, dams, offshore platforms and bridges. The SMA-3, used with the companion SMP-1 Playback System, meets the requirements of U.S. NRC Regulatory Guide 1.12 and is being used at over 90 nuclear power plants around the world.

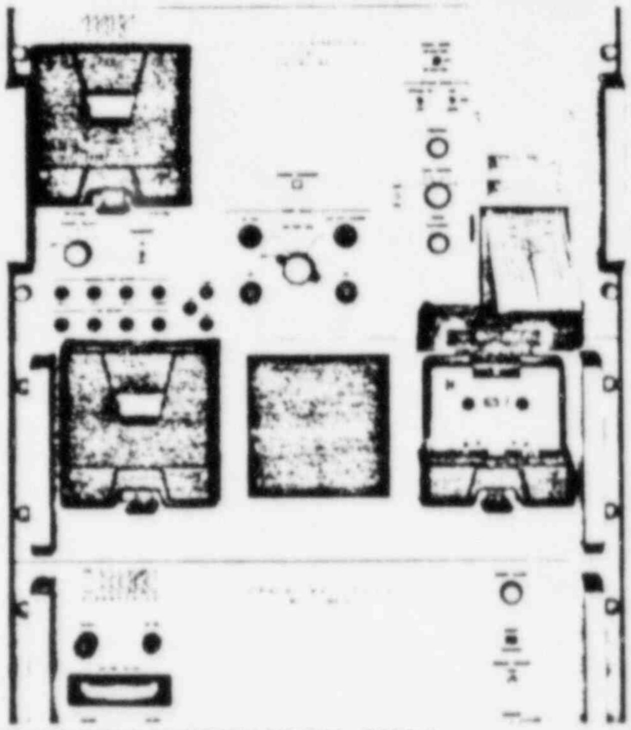
An SMA-3 can accommodate up to 27 channels of acceleration data, usually from triaxial force balance accelerometers, Model FBA-3. Downhole triaxial sensors (FBA-13DH) can be installed, and uniaxial and biaxial accelerometers may also be used. The sensors may be located up to 1500 feet from the central recorder. The TS-3 triaxial seismic trigger is standard with any SMA-3 system. The SMA-3 comes supplied with two cassettes per recording section, and all mounting hardware and mating connectors for the specified number of triggers and accelerometers.

GENERAL DESCRIPTION

The SMA-3 is a versatile multi-channel acceleration recording system. It is self-actuating when a local earthquake exceeds a predetermined level of ground acceleration. When acceleration falls below the preset value, the SMA-3 automatically returns to the standby condition.

The standard FBA-3 triaxial accelerometer package is approximately a 20 centimeter cube. It contains three force-balance acceleration sensors. The accelerometer package accepts calibration commands for damping and natural frequency.

Each accelerometer signal is buffered, frequency modulated, and recorded on an assigned track of a four-track magnetic tape cassette. Three tracks are used for acceleration data and the fourth for a timing signal, which is common for all recording tape transports in the system.



TECHNICAL SPECIFICATIONS

SEISMIC TRIGGERS (Model TS-3)

Type: Triaxial acceleration trigger
Housing: Cast aluminum, waterproof
Set Point: 0.01g standard; field adjustable, 0.005g to 0.05g
Option: Adjustment range of 0.025g to 0.25g
Current Drain: 0.45 mA in standby; 60 mA operating

TRANSDUCERS (Model FBA-3)

Type: Force balance accelerometers
Housing: Cast aluminum, waterproof
Bandwidth: 0 to 50 Hz
Range: ± 1g full scale
Output: ± 2.5 V full scale
Damping: 70% of critical
Natural Frequency: 50 Hz
Calibration: Damping and natural frequency recorded by command
Temperature Range: —20° to 70° C (0° to 160° F)
Temperature Effects: ± 1.5% of full scale over operating range

RECORDING SYSTEM

Type: Frequency modulation
Tape: Four track magnetic tape cassette
Tape Speed: 1-7/8" per second
Recording Time: 30 minutes
Bandwidth: 0 to 50 Hz
Dynamic Range: 40 dB from 15° to 35° C (with SMP-1)
Modulation Frequency: 1000 Hz ± 50% modulation
Timing Frequency: 1024 Hz ± 0.2%
System Accuracy (with SMP-1): ± 5% at full scale, changing linearly to 1.5% of full scale at 0.01g
Start-up Time: Less than 0.1 seconds
Event Alarm: Normally open contacts, rated 1 amp @ 12 Vdc
Event Indicator: Electromagnetic visual display

POWER SUPPLY

Two 12 V internal, rechargeable batteries. An internal battery charger, operating from 110 Vac, is supplied.

OPERATING ENVIRONMENT

Temperature: 0° to 55° C (30° to 130° F)
Humidity: Remote packages: 100% R.H.
Cabinet mounted panels: 80% R.H. non-condensing

ORDERING INFORMATION, SMA-3

Kinemetrics Part Number: 101100
Strong Motion Acceleration System, including:

One triaxial seismic trigger, Model TS-3
Specify triggering threshold (0.01g standard)
Specify number of additional triggers if desired

Up to nine triaxial acceleration sensors, Model FBA-3, 1.0g full scale
Cost Option—Model FBA-11 uniaxial sensor
Cost Option—Model FBA-13DH downhole triaxial sensor
Option—Range 0.25g, 0.5g, 2.0g full scale
Specify number and type of sensors, up to 27 channels

Up to nine triaxial tape recording modules, with cassettes
Cost Option—Flame resistant wiring
Specify number of channels, up to twenty-seven

Control/Power Panel
Cost Option—Conversion to 220 Vac

Accessories:

Interconnecting Cables for seismic trigger(s)
Cost Option—Flame retardant cable
Specify lengths required, up to 1500' to each trigger

Interconnecting Cables for remote accelerometers
Cost Option—Flame retardant cable
Specify lengths required, up to 1500' to each sensor

19-inch Rack Mounting Cabinet
Cost Option—Seismically braced cabinet

Tape Playback, Model SMP-1 (see SMP-1 data sheet)

Spares and Supplies
Magnetic Tape Cassettes, Part #700030
Desiccant Envelopes, Part #700049
12 V Batteries (pair), Part #103413

APPENDIX B

REPORT ON THE PEAK SHOCK RECORDERS AND PEAK ACCELERATION RECORDERS INSTALLED AT THE PERRY NUCLEAR POWER PLANT DURING THE SEISMIC EVENT ON JANUARY 31, 1986 ENGDAHL ENTERPRISES

ENGDAHL ENTERPRISES 2850 Monterey Avenue, Costa Mesa, California 92626, (714) 540-0398

REPORT ON THE PEAK SHOCK RECORDERS AND PEAK ACCELERATION RECORDERS INSTALLED AT THE PERRY NUCLEAR POWER PLANT DURING THE SEISMIC EVENT ON JANUARY 31, 1986

Copy Number 04

Engdahl Enterprises
Costa Mesa, CA 92626

February 7, 1986

CONTENTS

1. INTRODUCTION

On January 31, 1986, the effects of a seismic event were recorded by the Engdahl PSR1200, Peak Shock Recorders and PAR400, Peak Acceleration Recorders at the Perry Nuclear Power Plant located at Perry, Ohio. The record plates were removed from the recorders within hours and new plates were installed by Perry Plant and Engdahl personnel. A preliminary data reduction was completed the same day. A second independent data reduction was made on February 2, 1986. Photographs of all of the scribed records were made on February 2-3, 1986.

This report reviews the status of the instruments at the time of the event, contains the recorded data, and evaluates the data. The report also reviews the present status of the recorders and work to be done in the near future.

2. INSTRUMENT DESCRIPTIONS

2.1 PEAK SHOCK RECORDER (Response Spectrum Recorder and Response Spectrum Switch)

The Model PSR1200-H/V, Peak Shock Recorder, is designed to meet the characteristics of the Response Spectrum Recorder and the Response Spectrum Switch as described in the American Nuclear Society Standard ANSI/ANS-2.2-1978, "Earthquake Instrumentation Criteria for Nuclear Power Plants", and NRC Regulatory Guide 1.12 (Rev. 1), "Instrumentation for Earthquakes". It is a completely passive device covering the range of 2-25 HZ in 1/3 octave increments. Damping of each accelerometer is nominally 2%. It is completely self contained. Three recorders are arranged triaxially.

Twelve reeds of different lengths and weights, one for each frequency, are fabricated from spring steel. A diamond-tipped stylus is attached to the free end of each reed to inscribe a permanent record of its deflection on one of twelve record plates. The record plates are aluminum, plated with successive layers of nickel, tin, and lead-tin.

The Model PSR1200-H/V-12A comprises the standard PSR1200-H/V plus the capability of providing instantaneous warning signals when preset accelerations at selected frequencies have been exceeded. This is achieved by adding dual contacts which are closed by the reed when it is deflected through a predetermined distance.

2.2 PEAK ACCELERATION RECORDER (Peak Accelerograph)

The Model PAR400, Peak Acceleration Recorder, is designed to meet the characteristics of the Peak Accelograph described in ANSI/ANS-2.2-1978 and NRC Regulatory Guide 1.12. It senses and records peak accelerations triaxially. It is a self-contained passive device requiring no external power or control connections and has a minimum band width of 0 to 26 Hertz with a sensitivity as low as .01 g. The recorder is nominally 60% damped. A diamond tipped scriber at the end of an amplifier arm traces a very fine visible permanent record on an aluminum record plate with successive layers of nickel, gold, and burnt gold.

3. DESIGNATIONS, LOCATIONS, AND CALIBRATION STATUS OF INSTRUMENTS

3.1 D51-R160 - REACTOR BUILDING FOUNDATION

Triaxial Response Spectrum Recorder (PSR1200-H/V-12A)
Location - 574' Reactor Building foundation mat, azimuth 210° (see drawing D-811-801 and D-814-663-909)
Active scratch recorder, which alarms on control room panel 1H13-P969, annunciator panel D51-R215
Most recent calibration on 1-14-85. *

3.2 D51-R170 - REACTOR BUILDING I.D.W. 630' PLATFORM

Triaxial Response Spectrum Recorder (PSR1200-H/V)
Location - inside Drywell platform - 630', azimuth, 240° (see drawing D-811-605 and D-814-665-910)
Most recent calibration completed on 1-30-86. *

3.3 D51-R180 - HPCS PUMP BASE MAT

Triaxial Response Spectrum Recorder (PSR1200-H/V)
Location - HPCS Pump Room - Auxiliary Building foundation mat 574' (see drawing D-811-701 and D-814-663-911)
Equipment being calibrated on 1-31-86 during earthquake. (North-South and East-West recorders operable).
Previous calibration on 1-14-85. *

* Calibration interval is established at 18 months by ANSI/ANS - 2.2-1978, "Earthquake Instrumentation Criteria for Nuclear Power Plants."

3.4 D51-R190 - RCIC PUMP BASE MAT

Triaxial Response Spectrum Recorder (PSR1200-H/V)
Location - RCIC Pump Room - Auxiliary Building foundation mat 574' (see drawing D-811-702 and D-814-663-912)
Equipment being calibrated on 1-31-86 during earthquake (all recorders operable).
Previous calibration on 1-14-85.*

3.5 D51-R120 - REACTOR RECIRCULATION PUMP

Peak Acceleration Recorder (PAR400)
Location - inside Drywell - 574' elevation. (see drawing D-811-602 and D-814-663-906). Located on recirculation pump B33-C001A.
Most recent calibration 12-4-85.*

3.6 D51-R140 - HPCS PUMP BASE MAT

Peak Acceleration Recorder (PAR400)
Location - Auxiliary Building - 574'
HPCS Pump Room - Auxiliary Building foundation mat 574' (see drawing D-814-633-908 and D-811-701)
Most recent calibration on 1-30-86. *

4. DATA REDUCTION

The following tabulations on Pages 8 through 13, show the initial data reduction made on January 31, 1986 by Perry Plant personnel and a field representative of Engdahl Enterprises. An independent data reduction made by Engdahl Enterprises on February 2, 1986 is listed alongside the initial reduction.

A total of 129 data point readings were tabulated. A comparison of the two independent data reductions indicates a very close correspondence. Most indicate no significant differences. For those cases where differences exist, the greatest differences (with one exception) are on the order of 0.03g. The largest acceleration difference between the two data reductions was 9% (MPL Number D51-R170, reed number 12, vertical). Even in this case, the difference is within tolerances allowed by industry standards.

5. DATA EVALUATION

The record plates from three of the four triaxial PSR1200 recorders had many scratches and some had multiple zero lines which made them difficult to read. This condition was due to construction work in progress since the recorders had been calibrated and installed in January 1985. Although initial review of these plates indicated that data reduction might be questionable, further review (including comparison with data from the Kinemetrics Time-History recorders**) has established the validity of the data reduction.

**Kinemetrics/Systems, "Strong-Motion Data Report for the M_L 5.0 Earthquake of 1147 EST, January 31, 1986" (February 4, 1986)

5.1 D51-R120, Reactor Recirculation Pump and D51-R140, HPCS Pump Base Mat

The records from these PAR400 recorders were good. D51-R120 had the best records. D51-R140 had poorer zero lines but the results were nonetheless in close agreement with Reactor Building foundation mat data from Kinemetrics Time-History recorder data.

5.2 D51-R160, Reactor Building Foundation

A reading was made for each reed in the horizontal directions. The North/South accelerations were in very close agreement with the response spectrum generated from the Time-History recorder (D51-N101). The East/West did not agree as well but was similar. Only six of twelve vertical data points were readable. All of these values were quite low indicating a low vertical component of acceleration.

5.3 D51-R170, Reactor Building I.D.W. 630' Platform

The most readable of the PSR1200 records were on the Reactor Building I.D.W. 630' Platform. The North/South was especially good with very good zero lines. The East/West and the vertical recorders each had two of twelve records that were difficult to read.

5.4 D51-R180, HPCS Pump Base Mat and D51-R190, RCIC Pump Base Mat

These two installations are both on the Auxiliary Building foundation mat but separated by approximately 80 feet. The resulting North/South response spectra are almost identical. The East/West response spectra were similar. The vertical D51-R180 recorder was not in service due to recalibration activities, so no comparison can be made. The vertical D51-190 recorder is questionable since the zero lines were offset by large amounts in most cases.

5.5 Dual records were noted on some of the record plates. The clearest of these are on D51-R160, East/West. A separate tabulation is made of the six best records (see page 14). A dual record is normally made when the record plate moves a very slight amount (.001 to .002 inches) after one record is made and then a second record is made. It is possible that all six plates moved at low levels and that the second record is just a continuation of the same event. It is also possible that the low level event was recorded and then the plates moved before the second event.

6. CURRENT STATUS

6.1 At present, the instruments are in operation with new record plates except the vertical recorder, D51-R180, which has been removed for recalibration.

6.2 Plans have been made to start the recalibration of all of the instruments on February 10, 1986. This recalibration is in preparation of fuel loading, and not as a result of the seismic event.

7. CONCLUSIONS

Although the records were not always easy to read because of activity at the plant during the construction phase, the records were clear enough in most cases to give very good overall results. Recalibration of the instruments was not required by the seismic event. Recalibration will be performed starting February 10, 1985 in preparation for fuel loading.

Paul D. Engdahl

Paul D. Engdahl

cjw

MPL NUMBER: D51-R120
LOCATION: REACTOR RECIRCULATION PUMP

SENSOR LOCATION	ACCELERATION (g)	
	1-31-86	2-2-86
NORTH/SOUTH (L)	.32	.318
EAST/WEST (T)	.10	.106 *
VERTICAL	.07	.048 *

* Zero lines not clear, best estimate

MPL NUMBER: D51-R140
LOCATION: HPCS PUMP BASE MAT - 574'

SENSOR LOCATION	ACCELERATION (g)	
	1-31-86	2-2-86
NORTH/SOUTH (L)	.15	.167
EAST/WEST (T)	.06	.058
VERTICAL	.04	.029

MPL NUMBER: D51-R160
LOCATION: REACTOR BUILDING FOUNDATION - 574'

REED NUMBER	NOMINAL FREQUENCY (HERTZ)	ACCELERATION (g)					
		North/South		East/West		Vertical	
		1-31-86	2-2-86	1-31-86	2-2-86	1-31-86	2-2-86
1	2.00	.027	.027	.029	.030	.007	**
2	2.52	.038	.038	.046	.046	.013	.011
3	3.17	.062	.060	.039	.040	**	**
4	4.00	.032	.035	.022	.026	**	**
5	5.04	.067	.069	.056	.054	**	.018
6	6.35	.065	.075	.054	.054	**	.016
7	8.00	.143	.133	.056	.051	.010	**
8	10.1	.136	.091	.176	.160	.061*	.053*
9	12.7	.196	.227	.236	.230	.032	.038
10	16.0	.286	.305	.284	.284	.101	.111
11	20.2	1.04	1.02	.605	.586	.224	**
12	25.4	.7657	.766	.540	.513	.329	**

* "C" surface
** Unreadable

MPL NUMBER: D51-R170
LOCATION: REACTOR BUILDING I.D.W. 630' PLATFORM - DW 630', 240°

REED NUMBER	NOMINAL FREQUENCY (HERTZ)	ACCELERATION (g)					
		North/South		East/West		Vertical	
		1-31-86	2-2-86	1-31-86	2-2-86	1-31-86	2-2-86
1	2.00	.047	.048	.049	.051	.007	.007
2	2.52	.082	.082	.086	.084	*	.013
3	3.17	.184	.184	.144	.140	.015	.014
4	4.00	.226	.223	.128	.127	.023	.023
5	5.04	.132	.134	.158	.158	.035	.033
6	6.35	.131	.134	.058	.055	.033	.030
7	8.00	.104	.104	.109	.090	*	.019 (2)
8	10.1	.093	.093	*	.052 (1)	.093	.085 (2)
9	12.7	.188	.182	.166	.080 (2)	.198	.199
10	16.0	.194	.204/ .167	.348	.312	.490	.500
11	20.2	.152	.152	.191	.175	.973	.973
12	25.4	.114	.091	.155	.158	1.7	1.54

* Unreadable
(1) Unusual appearance
(2) Very difficult to read - best estimate

MPL NUMBER: D51-R180
LOCATION: HPCS PUMP BASE MAT - 574'

REED NUMBER	NOMINAL FREQUENCY (HERTZ)	ACCELERATION (g)					
		North/South		East/West		Vertical*	
		1-31-86	2-2-86	1-31-86	2-2-86	1-31-86	2-2-86
1	2.00	.0198	.020	.022	.021	-	-
2	2.52	.0358	.036	.033	.031	-	-
3	3.17	.0677	.068	.045	.048	-	-
4	4.00	.0474	.047	.022	.020	-	-
5	5.04	.0637	.064	.033	.029	-	-
6	6.35	.0735	.068	.054	.050	-	-
7	8.00	.0473	.052	.046	.046	-	-
8	10.1	.0744	.074	.566	**	-	-
9	12.7	.125	.149	.182	.176	-	-
10	16.0	.4582	.449	.253	.214	-	-
11	20.2	.9130	.896/ .432	.413	.429	-	-
12	25.4	.6100	.610/ .293	.191	**	-	-

* Not in service
** Unreadable

MPL NUMBER: D51-R190
LOCATION: RCIC PUMP BASE MAT - 574'

REED NUMBER	NOMINAL FREQUENCY (HERTZ)	ACCELERATION (g)					
		North/South		East/West		Vertical	
		1-31-86	2-2-86	1-31-86	2-2-86	1-31-86	2-2-86
1	2.00	.021	.018	.026	.022	**	**
2	2.52	.039 (1)	.030	.031	.021	**	.013
3	3.17	*	*	.024	.017	**	**
4	4.00	.0367	.031	.028	.023	.029	**
5	5.04	.0305	.045	.037	.038	**	**
6	6.35	.0896	.065	.057	.048	**	**
7	8.00	.0750	.040	.068	.034	.019	.014
8	10.1	*	*	.097	.044	**	**
9	12.7	.130	.124	.142	.136	.053	.024
10	16.0	.409	.400	.162	.162	.082	.055
11	20.2	.810	.794	.237	**	**	.099
12	25.4	.556	.557	**	.156	.256	.256

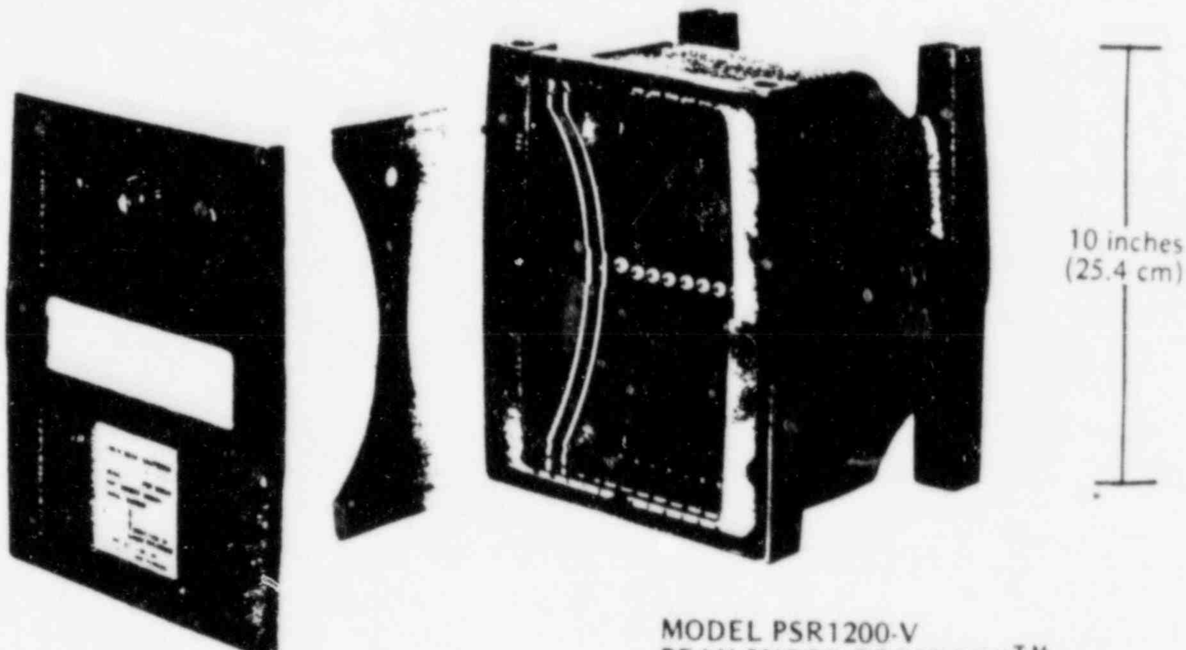
(1) Mathematical error corrected. Originally reported acceleration 0.198.
* Unable to read due to corrosion
** Unreadable

MPL NUMBER: D51-R160
LOCATION: REACTOR BUILDING FOUNDATION - 574'
DUAL RECORDS

REED NUMBER	NOMINAL FREQUENCY (HERTZ)	ACCELERATION (g)					
		North/South		East/West		Vertical	
		1-31-86	2-2-86	1-31-86	2-2-86	1-31-86	2-2-86
1	2.00	-	-	-	.006	-	-
2	2.52	-	-	-	.009	-	-
3	3.17	-	-	-	.010	-	-
4	4.00	-	-	-	.026	-	-
5	5.04	-	-	-	.054	-	-
6	6.35	-	-	-	.035	-	
7	8.00	-	-	-	-	-	-
8	10.1	-	-	-	-	-	-
9	12.7	-	-	-	-	-	
10	16.0	-	-	-	-	-	
11	20.2	-	-	-	-		-
12	25.4	-	-	-	-	-	-

APPENDICES

MODEL PSR1200-H/V-4A and PSR1200-H/V-12A



MODEL PSR1200-V
PEAK SHOCK RECORDER™

SENSES AND PERMANENTLY RECORDS THE SPECTRAL ACCELERATION AT SPECIFIED FREQUENCIES

PROVIDES SIGNALS FOR IMMEDIATE REMOTE INDICATION THAT SPECIFIED PRESET SPECTRAL ACCELERATIONS HAVE BEEN EXCEEDED

- EARTHQUAKES
- STORMS
- EXPLOSIONS

RELIABLE and ECONOMICAL

ENGDAHL ENTERPRISES

2850 Monterey Avenue · Costa Mesa, California 92626 · (714) 540-0398

Introduction

Traditionally, measurement of acceleration has implied measurement with the aid of a device whose resonant frequency was far removed from the frequency range of interest. A typical accelerometer for aerospace applications might have a mass of 10 grams and a resonant frequency of 10 kHz or higher. Such devices were designed primarily for attachment to a structural member to measure its response to shock or vibration. Their low mass was necessary to avoid modifying the characteristics of the device under test, while the resonant frequency had to be at least five times that of the highest frequency of interest. At the other end of the spectrum, earthquakes and other low frequency phenomena are conventionally detected and recorded using instruments whose resonant frequencies are much lower than the frequency range of interest.

A structure such as a large office building, a missile silo or an electrical generating station has many members and subassemblies with a wide range of resonant frequencies, and many of these are lightly damped, i.e., a shock will cause them to "ring" for a relatively long time. To measure the effects of an earthquake or other shock on such a structure in the traditional way, would require a very large number of transducers and a complex data acquisition system followed by computer analysis to digest the raw data and decide whether or not structural damage had been sustained.

To simplify the design of shock resistant structures, dynamicists frequently define shocks and earthquakes in terms of response shock spectra. Basically, a response shock spectrum is a plot of acceleration vs. frequency in which each point represents the peak acceleration experienced by an accelerometer tuned to that specific frequency. The range of frequencies covered by the peak shock accelerometers corresponds to those found in most structures, systems, and components. Since all structural elements possess some low inherent damping, the Peak Shock Recorder™ has been designed with 2% of critical damping. The output obtained is thus directly applicable to structural design and analysis.

A response spectrum may be derived from the conventional acceleration vs. time record of a suitable recording accelerometer, but this involves either digitizing the records followed by computer manipulation of the data or the use of a large amount of auxiliary equipment. The first method is time consuming, while the second is expensive. The Model PSR1200-H/V is an inexpensive instrument requiring no source of power, and virtually no maintenance. It provides a permanent record of data from which the response spectrum may be plotted by a very simple reduction process.

Description

The Model PSR1200-H/V, Peak Shock Recorder™, is a completely passive device covering the range of 2-25 Hz in 1/3 octave increments. Damping of each accelerometer is nominally 2%. It is completely self contained.

Twelve reeds of different lengths and weights, one for each frequency, are fabricated from spring steel. A diamond-tipped stylus is attached to the free end of each reed to inscribe a permanent record of its deflection on one of twelve record plates. A calibration sheet for each recorder lists the resonant frequency and g-sensitivity of each reed.

—V designates a recorder designed for vertical shock recording (compensated for earth's gravitational force). —H designates a recorder designed for horizontal shock recording.

The Model PSR1200-H/V-4A/12A comprises the standard PSR1200-H/V plus the capability of providing instantaneous warning signals when preset accelerations at selected frequencies have been exceeded. This is achieved by adding dual contacts which are closed by the reed when it is deflected through a predetermined distance. Model —4A monitors four selected reeds, while —12A monitors all of the reeds.

Uses

The PSR1200, Peak Shock Recorder™, is useful whenever acceleration measurements are desired at low frequencies. These accelerations may be due to earthquakes, storms, or explosions. The plot of the recorder's twelve individual measurements is the response spectrum of the acceleration to which the recorder was subjected.

The response spectrum switch (–A) version of the PSR1200 is useful whenever remote indications are desired that acceleration limits have been exceeded. The remote indication that four or twelve dual acceleration limits have or have not been exceeded provides immediate information on which to act.

The Peak Shock Recorder™ can be used in connection with:

1. Nuclear power plants
2. Steel mills
3. Refineries
4. Bridges and dams
5. High-rise structures
6. Oil explorations
7. Mines
8. Ships
9. Earth studies
10. Towers

Features

Dzus, quarter-turn fasteners, are used to secure the cover, making it easily removable. The cover is clamped tight enough against the gasket bonded to the watertight housing to provide protection of the unit to 50 PSI (3.6 kg/Cm^2) of water pressure.

The record plates are serialized so only one set of twel. : have the same number. In addition, the plates have two types of slots to allow keying. The narrow key slot allows the plate to slide into only one slot in the housing to its full depth. That is, the plates all have to be in their correct locations in the housing to attach the cover.

The record plates can be inserted four different ways into the housing, allowing four records to be made before using a second set of plates. To prevent mixing the records, all plates must be inserted for the record to appear at A, B, C, or D or, again, the cover cannot be attached. A viewing window is provided, and the appropriate letter A through D will show so the cover need not be removed to know how the plates are inserted. During shipping, a red dot is seen. This means that the plates have been removed and the reed support structure is in place.

Additional keying is provided between the covers and housings in the form of dowel pins. These pins prevent the cover from being put on upside down. They also prevent a cover from a horizontal recorder (–H) being put on a vertical recorder (–V) or a –V on a –H.

Since a lower atmospheric pressure could be created inside the recorder than outside during shipment by air, a jackscrew is provided in the cover to lift a corner of the cover and break the partial vacuum. It will also be of assistance when the unit has been closed for a long period of time as the neoprene gasket may adhere slightly to the cover.

The recorder is reliable because of its simplicity. It does not contain any of the more complex and less reliable components, i.e., batteries, connectors, motors, and bearings. Its rugged structure is fabricated from aluminum alloy. Only a few parts are used. The recorder is self-contained, and requires no start-up time.

The recorder is economical in that no external connections or power are required. The record plates are reusable by replating after four records have been obtained. Maintenance is very low since the unit can be unattended for long periods of time. Data reduction is very simple, requiring only one measurement and one multiplication for each record plate to plot its point on the response spectrum.

The response spectrum switch (—A) version of the recorder has all of the features of the PSR1200.

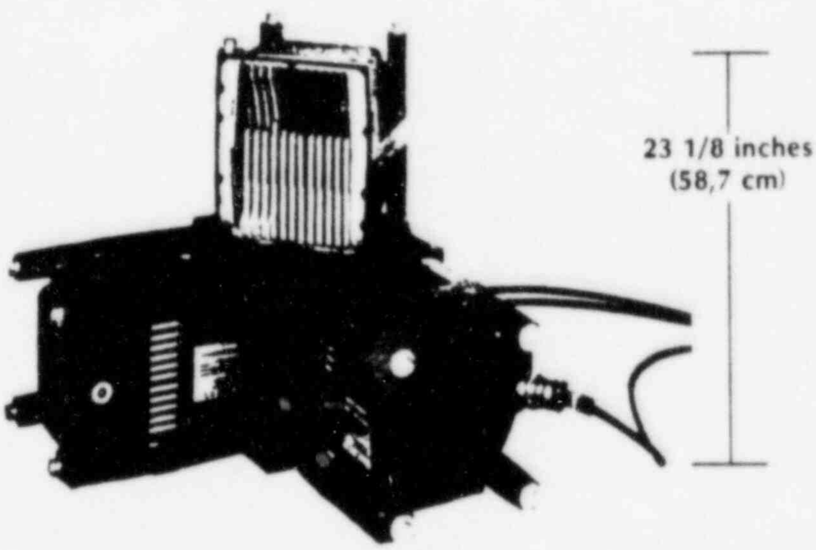
To retain the basic reliability of the PSR1200, no batteries, motors, or bearings have been added. Electrical power is provided from the Peak Shock Annunciator™.

Every effort has been made to achieve the utmost reliability in the switching circuitry so as to match the reliability of the basic Peak Shock Recorder™. Closure of a switch contact sets an electronic latching switch which energizes the appropriate circuit in the annunciator and holds it energized until reset by the key-switch.

High impedance circuitry permits normal operation even if switch contact resistance exceeds several hundred thousand ohms. Ceramic encapsulated integrated circuits offer maximum resistance to the effects of temperature and humidity.

Finally, the heavy cast aluminum housing of the recorder offers protection against radiated interference or spurious mechanical operation caused by striking the recorder.

The recorder can be used singly, biaxially, or triaxially.



TRIAXIAL INSTALLATION OF THREE MODEL PSR 1200-H/V-12A PEAK SHOCK RECORDERS™ ON TRIAXIAL MOUNT

Switch Settings (—A version only)

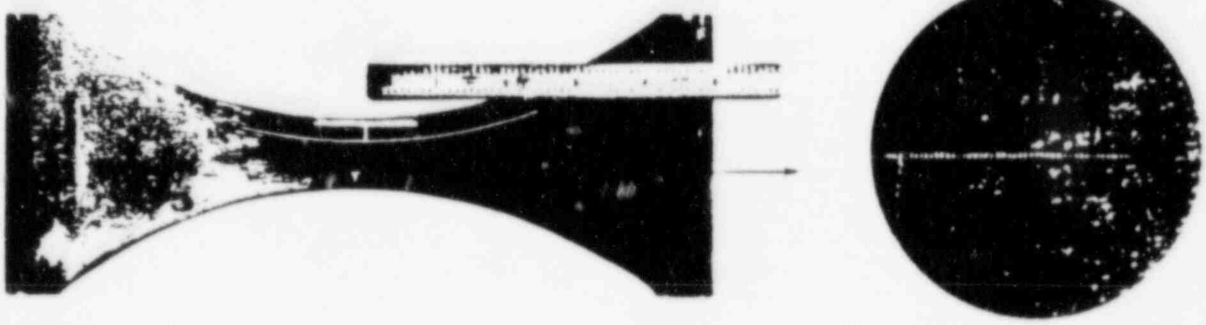
The switch settings are permanently set to positions required by the customer's application. The —4A allows four dual settings, that is, the customer selects four frequencies to be monitored between 2 and 25 Hertz. Two acceleration levels can be selected for switch contacts for each reed frequency, e.g., .47 g and .70 g at 3.2 Hertz. The —12A has twelve dual settings between 2 and 25 Hertz. See the tabulation of "Frequency and Switch Setting Limits" for selection available.

Data Reduction

Data reduction is done by measuring the maximum distance of the scratched record from the zero line. Normally just the maximum is recorded regardless of the direction. List this distance under "Displacement" on the calibration sheet.

Multiply the "Displacement" times the "Acceleration sensitivity" and record in the "Equivalent static acceleration" column. Plot the response spectrum graph.

Large displacement measurements can be made with a six-inch (152 mm) scale with graduations in hundreths (.01) of an inch (.25 mm). Small displacements can be made using a microscope with a reticle having graduations in thousandths (.001) of an inch (.025 mm).

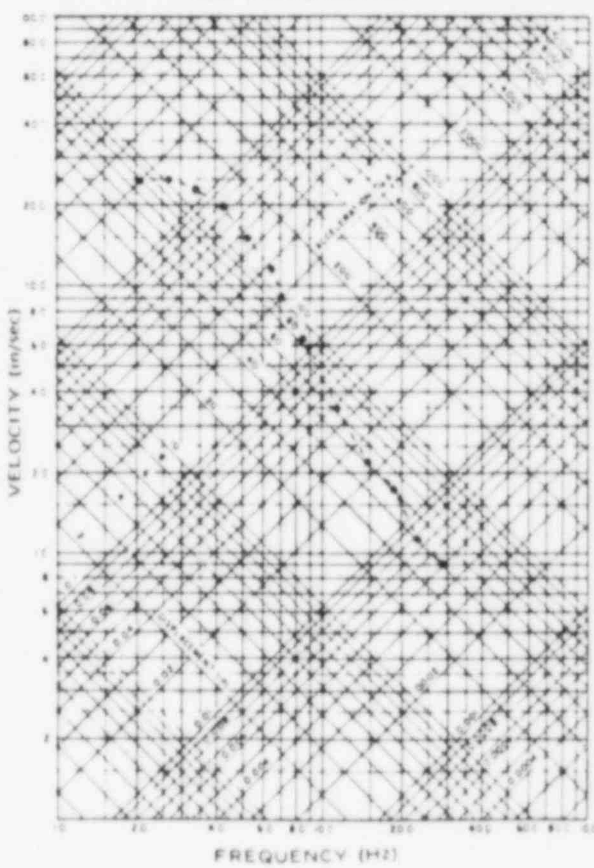


CALIBRATION SHEET AND TEST DATA

Reed Number	Frequency (Hertz)	Acceleration Sensitivity (g/inch)	Displacement (inches)	Equivalent Static Acceleration (g)
1	2.02	.359	2.51	.90
2	2.54	.55	2.00	1.1
3	3.20	.85	1.41	1.2
4	4.02	1.32	.98	1.3
5	4.92	2.34	.55	1.3
6	6.02	3.62	.33	1.2
7	8.08	5.5	.15	.83
8	10.2	7.6	.079	.6
9	12.7	6.6	.078	.5
10	16.2	10.5	.046	.5
11	20.6	17.5	.022	.4
12	26.1	26.8	.015	.4

CALIBRATION — DATA REDUCTION

RESPONSE SPECTRUM



PEAK SHOCK RECORDER™

MODELS PSR1200-H/V-4A and PSR1200-H/V-12A

QUALIFIED TO: GUIDE FOR SEISMIC QUALIFICATION OF CLASS I ELECTRICAL EQUIPMENT FOR NUCLEAR POWER GENERATING STATIONS — IEEE GUIDE 344

Designed to meet the characteristics of the Response Spectrum Recorder and the Response Spectrum Switch described in the American Nuclear Society's Standard, ANSI/ANS-2.2-1978, Earthquake Instrumentation Criteria for Nuclear Power Plants and the U.S. Nuclear Regulatory Commission's Regulatory Guide 1.12, Nuclear Power Plant Instrumentation for Earthquakes, Revision 1. NOTE: Frequency range from 2.00 to 25.4, instead of 1.00 to 30.0.

PHYSICAL

Length	12-27/32 inches (32,6 cm)
Width	11-1/2 inches (29,2 cm)
Thickness	10 inches (25,4 cm)
Weight	34 pounds (15,4 kg) 36 pounds (−A) (16,3 kg)

ENVIRONMENTAL

Temperature	−40°C to +85+C
Altitude	To 50,000 feet (15,240 meters)
Humidity	To 100% RH
RFI	No adverse radiated or conducted RFI
Water-Tight	To 50 PSI (3,6 kg/cm²) To 10 PSI (−A) (,7 kg/cm²)
Nuclear Radiation	No effect on performance of permanent recorder. Switch electronics are not radiation hardened, unless requested at extra cost.

SENSORS

Number of Sensing Elements	12
Damping	2% (Q of 25)
Arrangement of Sensing Elements	Coplanar
Number of Switch Contacts	
−4A	4 Dual Contacts
−12A	12 Dual Contacts

ACCURACY

Frequency	±1%
Acceleration	±3% at 1g
Dynamic Range	See Table
Switch Settings	±3% at 1g

MOUNTING

4 through holes for 1/2 inch bolts

FREQUENCY, RANGE, and SWITCH SETTING LIMITS OF SENSING ELEMENTS

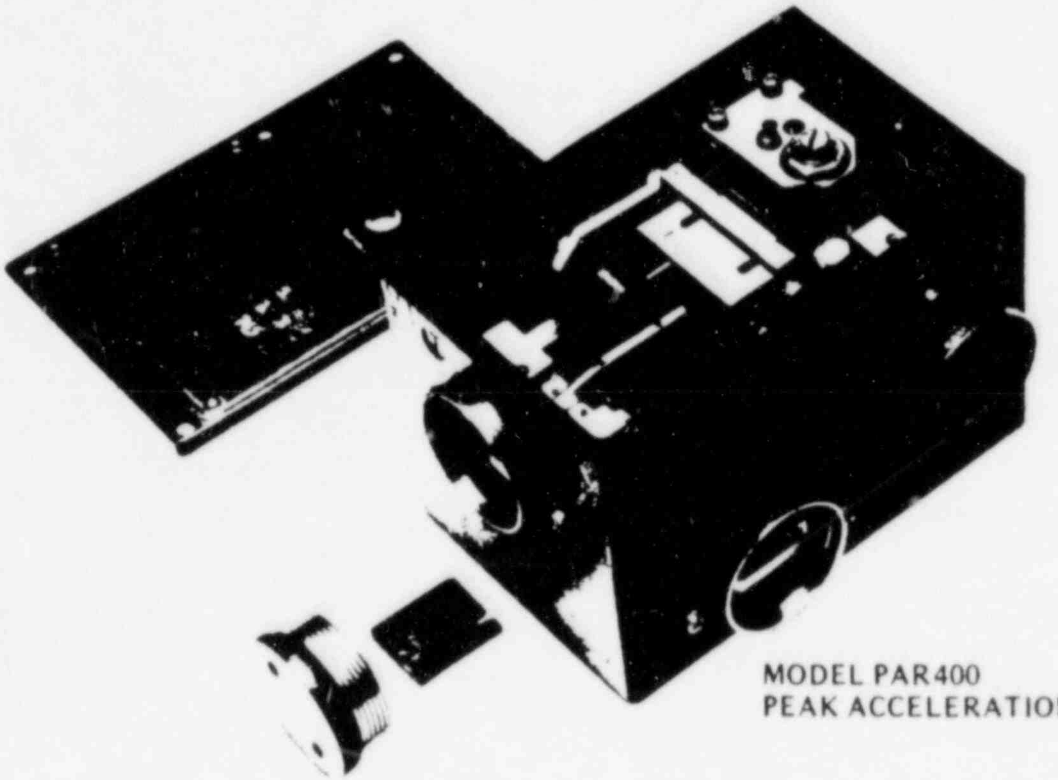
Reed Number	Nominal* Resonant Frequency (Hertz)	Full Scale Acceleration (g)	Dynamic Range (db)	Dynamic Range Ratio	Switch Setting Limits** (g) Minimum*** (Accuracy ±100%)	Switch Setting Limits** (g) Maximum (Physical Stops)
1	2.00	1.6	54.5	530:1	.003	1.6
2	2.52	2.5	55.8	620:1	.004	2.5
3	3.17	4	58.1	800:1	.005	4
4	4.00	6	58.7	860:1	.007	6
5	5.04	10	60.9	1110:1	.009	10
6	6.35	16	61.8	1230:1	.013	16
7	8.00	24	63.0	1410:1	.017	24
8	10.1	34	64.6	1700:1	.020	34
9	12.7	8	54.5	530:1	.015	8
10	16.0	12	55.6	600:1	.020	12
11	20.2	4	46.0	200:1	.020	4
12	25.4	6	49.5	300:1	.020	6

* — 4A Allows choice of 4 frequencies to be monitored from 2 to 25 Hertz
— 12A Allows all 12 frequencies to be monitored from 2 to 25 Hertz
** — Two switch settings for each frequency to be monitored

*** — Do not use PSR1200-H/V-A for settings under 0.10g. For lower settings use RSR1600-H/V-A.

REPRESENTED BY:

MODEL PAR400



3-3/16 inches
(8.1 cm)

MODEL PAR400
PEAK ACCELERATION RECORDER™

SENSES AND PERMANENTLY RECORDS PEAK ACCELERATIONS

- EARTHQUAKES
- STORMS
- EXPLOSIONS

RELIABLE and ECONOMICAL

ENGDAHL ENTERPRISES 2850 Monterey Avenue · Costa Mesa, California 92626 · (714) 540-0398

Introduction

Seismic events are random events, and may occur in remote and inaccessible locations or in built-up areas. Scientists and engineers frequently need to know the acceleration levels associated with these events, and for this reason, have developed instruments requiring no source of power, which can provide permanent records of peak acceleration.

Instruments of this type have been used for many years, but with the advent of the nuclear power plant, higher sensitivity and increased bandwidth are required to measure the accelerations induced in piping and other equipment. Since the older types of peak accelerometers had been pushed to their design limits, an entirely new instrument was required.

This requirement has been met with the Model PAR400, Peak Acceleration Recorder™. It is an inexpensive triaxial unit which requires no power supply, and is virtually maintenance free. Peak accelerations as low as .01 g can be recorded, and the minimum bandwidth extends from 0 to 26 Hertz. Permanent records are scribed by diamond stylii on replaceable metal plates. The peak acceleration is computed by multiplying the maximum excursion of the trace by the acceleration sensitivity of the recorder.

Description

The Model PAR400, Peak Acceleration Recorder™, senses and records peak accelerations triaxially. It is a self-contained passive device requiring no external power or control connections and has a minimum band width of 0 to 26 Hertz with a sensitivity as low as .01 g.

Each sensor of the PAR400 incorporates a new method of mechanical amplification which makes it more than five times as sensitive as previous devices. With the aid of optical magnification, its permanent record can be read to .001 of an inch (.025 mm) or less. With a full scale deflection of .200 inches (5 mm), the −1 version (2 g full scale) has a dynamic range of 200:1 (46 db).

Air damping is used since it is very efficient for its size and weight. Minor adjustments of damping can be made in the field, if required.

Sensors are available in three natural frequencies: 32, 51 and 64 Hertz. The assemblies are mechanically identical and completely interchangeable, so any combination may be included in a triaxial recorder.

The record is scratched permanently on a metal plate which is both serialized and keyed to the recorder to assure that the records are not confused among the three axes. Since the record is scratched, it can be measured without further processing. The record plates are inserted through side holes in the casting without taking off the cover. This minimizes the possibility of damaging the recorder or inadvertently recording on the record plate during insertion or removal by touching the mechanism.

Applications

The PAR400 is useful whenever low frequency peak acceleration measurements are needed. These accelerations may be due to earthquake, storms, or explosions. The three records give the acceleration levels along three mutually perpendicular axes.

The Peak Acceleration Recorder™ can be used in connection with:

1. Nuclear power plants
2. Steel mills
3. Refineries
4. Bridges and dams
5. High-rise structures
6. Mines
7. Ships
8. Off-shore oil rigs
9. Transportation shock

Features

The PAR400 is a very sensitive, wide band, low frequency acceleration recording instrument. The high sensitivity is obtained by using a heavy mass to detect the acceleration, and then mechanically amplifying its motion. A diamond tip scriber at the end of the amplifier arm traces a very fine visible permanent record of the arm's excursions. The scribe line widths are on the order of .0004 inches (.01 mm).

Three plates, stamped L, T, and V, respectively, are used to record the excursions in the three axes. Slotted keyways on the plates match up with pins in the housing so that only the correctly stamped plate can be inserted full depth into the corresponding sensor. Each set of three plates also carries a unique serial number. This permanent identification system eliminates the possibility of confusing the records.

The rugged cast aluminum housing has three pads to contact the mating surface when mounted. A single screw is used for attachment. Shims can be slid under the appropriate pad to level the unit. The screw is then tightened. A clearance hole is provided in the cover for the screw head so the cover need not be removed during mounting of the recorder.

To install the record plates, three plugs are removed from the side walls of the casting and the plates are slipped into the appropriate holders. The plugs are of such a size as to preclude damage to the mechanism during insertion or removal of the record plates. Since the cover does not have to be removed to replace record plates, the mechanism is not exposed to inadvertent damage.

When a record plate is inserted, a spring-loaded pin forces the plate to one side of the track to eliminate any side play which would introduce an error in the recorded acceleration. The insertion produces a zero line on the plate. On removal, a zero line is also scratched. These zero lines should coincide if there is no mechanical shifting between insertion and removal. If there is a shift, the user is made aware that a problem exists.

To obtain wide band response, the instrument is damped to 60% of critical. A preadjusted air damper is used for damping to keep the size and weight of the total package as small as possible.

The recorder is reliable because of its simplicity. It does not contain any of the more complex and less reliable components, i.e., batteries, connectors, motors, and bearings. The recorder is self-contained, and requires no start-up time.

The recorder is economical in that no external connections or power are required. The record plates are reusable by replating. Maintenance is very low since the unit can be unattended for long periods of time.

Materials have been selected for long life even when exposed to nuclear radiation. The cast housing, along with the cover and three plugs, is chemically filmed (alodine) and painted with epoxy paint. The gaskets are made of EPDM to increase resistance to radiation. All hardware is stainless steel. An indicating silica gel desiccant is also provided to decrease the humidity inside the recorder.

Data reduction is very simple requiring only one measurement and one multiplication for each of the three record plates to obtain its maximum acceleration.

Data Reduction

Data reduction is accomplished by measuring the maximum displacement of the scratched record from the zero line. Normally just the maximum is recorded regardless of the direction. List this distance under "Displacement" on the calibration sheet.

Multiply the "Displacement" times the "Acceleration Sensitivity" and record in the "Acceleration" column.

Small displacement measurements can be made using a microscope with a reticle having graduations in thousandths (.001) of an inch. A magnifier with a reticle graduated in tenths (.1) of a mm can be used for medium displacements. Consult Engdahl Enterprises for microscopes.

MAGNIFIED RECORD USING A RETICLE

SAMPLE OF A CALIBRATION AND TEST DATA SHEET

Sensor	Natural Frequency (Hertz)	Acceleration Sensitivity (g/inch) [g/mm]	Displacement (inches) [mm]	Acceleration (g)
L	32.3	14.0 (.551)	.023 (.58)	.32
T	30.9	13.5 (.532)	.010 (.26)	.14
V	33.3	14.2 (.559)	.005 (.13)	.07

CALIBRATION — DATA REDUCTION

PEAK ACCELERATION RECORDER™

MODEL PAR400

QUALIFIED TO IEEE RECOMMENDED PRACTICES FOR SEISMIC QUALIFICATION OF CLASS 1E EQUIPMENT FOR NUCLEAR POWER GENERATING STATIONS, STD. 344-1975

Designed to meet the characteristics of the Peak Accelerograph described in the American National Standard ANSI/ANS-2.2-1978, Earthquake Instrumentation Criteria for Nuclear Power Plants and the U.S. Nuclear Regulatory Commission's Regulatory Guide 1.12, Nuclear Power Plant Instrumentation for Earthquakes, Revision 1.

SENSORS

Number of Sensing Elements	3	
Arrangement of Elements	Triaxial	
Full Scale Acceleration	−1	2 g
	−2	5 g
	−3	10 g
Dynamic Range	−1	200:1 (46 db)
	−2	385:1 (52 db)
	−3	500:1 (54 db)
Natural Frequency (± 5%)	−1	32 Hz
	−2	51 Hz
	−3	64 Hz

Damping 55 to 70% of Critical[1]

Bandwidth	−1	0 to 26 Hz
	−2	0 to 41 Hz
	−3	0 to 52 Hz

Overall Accuracy: Within ± 5% at full scale, changing linearly to ± 1.5% of full scale at 0.01 g

Detail Acceleration Accuracy	−1	.01 to .50 g ± .01 g
		.50 to 1¼ g ± 2%
		1¼ to 2 g ± 3%
	−2	.013 to .65 g ± .013 g
		.65 to 2 g ± 2%
		2 to 5 g ± 3%
	−3	.02 to 1 g ± .02 g
		1 to 3 g ± 2%
		3 to 10 g ± 3%

Spurious Resonances: None within frequency range of interest

Cross Axis Sensitivity: Less than .03 g/g

PHYSICAL DIMENSIONS

Length	5-1/4 inches (13.34 cm)
Width	3-5/8 inches (9.21 cm)
Height	3-11/32 inches (8.49 cm)
Weight	2-3/4 pounds (1.3 kg)

[1]Damping adjusted at nominal atmospheric pressure expected at time of operation.

MOUNTING

One (1) #10-24 Screw.

Level Recorder to ± 1° (1/16 inch in 3½ inches) (1.6 mm in 90 mm) by adding shims under the appropriate mounting pad. "V" will measure the vertical accelerations. Align long side of recorder within 3° (¼ inch in 5-1/8 inches) (6.4 mm in 130 mm) of designated North/South line. "L" (longitudinal) will measure N/S accelerations. "T" (transverse) will measure E/W.

ENVIRONMENTAL

Temperature	−40°C to + 85°C
Humidity	To 100% RH
RFI	Does not radiate or conduct RFI. Not affected by external RFI.
Water	Water-Tight to 70 PSI ($5\ kg/cm^2$)

Nuclear Radiation

The following materials are used in the construction of the PAR400.

1. Metals: Aluminum, Brass, Stainless Steel, Beryllium Copper, Gold, Nickel
2. Non-Metallic Materials

Description	Material	Stress Level	Approx. Stability[2] (RAD)
Paint	Epoxy	Low	1×10^8
Adhesive	Epoxy	Low	1×10^8
Adhesive	Anaerobic	Low	2×10^8
Adhesive	Cyanocrylate	Low	2×10^8
Gaskets	EPDM	Low	1×10^8
Piston	Graphite	Low	2×10^8
Cylinder	Pyrex	Low	2×10^8
Scriber	Diamond	Low	2×10^8

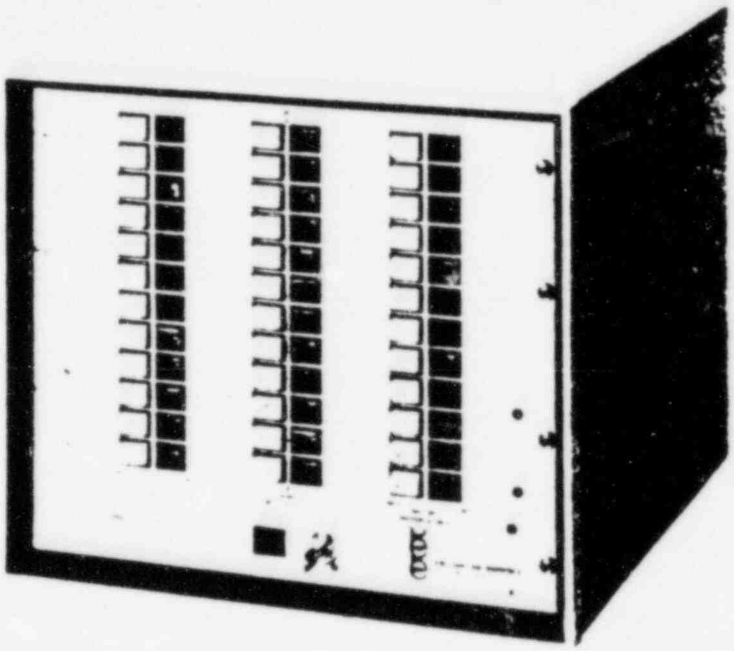
POWER REQUIREMENTS – None

[2]Source: Dow Corning Corporation, Loctite Corporation, Corning Glass Work, E.I. Du Pont De Nemours & Company, Parker Seal Company, Raychem Corporation, General Electric

REPRESENTED BY:

1-85-1M 400700B

MODEL PSA875 and PSA1575



17¼ inches (43.8 cm)

MODEL PSA1575
PEAK SHOCK ANNUNCIATOR™

INDICATES THAT SPECIFIED PRESET SPECTRAL ACCELERATIONS HAVE BEEN EXCEEDED

PROVIDES CONTACT CLOSURES FOR REMOTE INDICATORS OR ALARMS

- EARTHQUAKES
- STORMS
- EXPLOSIONS

RELIABLE and ECONOMICAL

ENGDAHL ENTERPRISES 2850 Monterey Avenue · Costa Mesa, California 92626 · (714) 540-0398

Description

The Models PSA875 and PSA1575, Peak Shock Annunciators™, give visual warning that predetermined acceleration limits, making up a response spectrum, have been exceeded at certain frequencies. They are designed to operate in conjunction with tuned Peak Shock Recorders™, PSR1200-H/V-4A/12A. Both models have three banks of indicator lamps, one bank for each of three mutually perpendicular axes. Amber lights indicate accelerations approaching design limits (normally 70%) while red lights indicate that design limits have been exceeded. Model PSA875 monitors four frequencies per axis while Model 1575 monitors twelve. Both models may be equipped with relays to operate remote indicators or alarms. (See "Options and Accessories".)

Applications

The annunciators may be used whenever it is desired to indicate instantaneously the reaction of a structure to a complex shock such as an earthquake or an explosion. The information provided permits an immediate decision as to whether or not the operation can continue or must be shut down.

The Peak Shock Annunciator™ can be used in connection with:

1. Nuclear power plants
2. Steel mills
3. Refineries
4. Bridges and dams
5. High-rise structures
6. Mines
7. Ships
8. Off-shore oil rigs
9. Transportation shock

Features

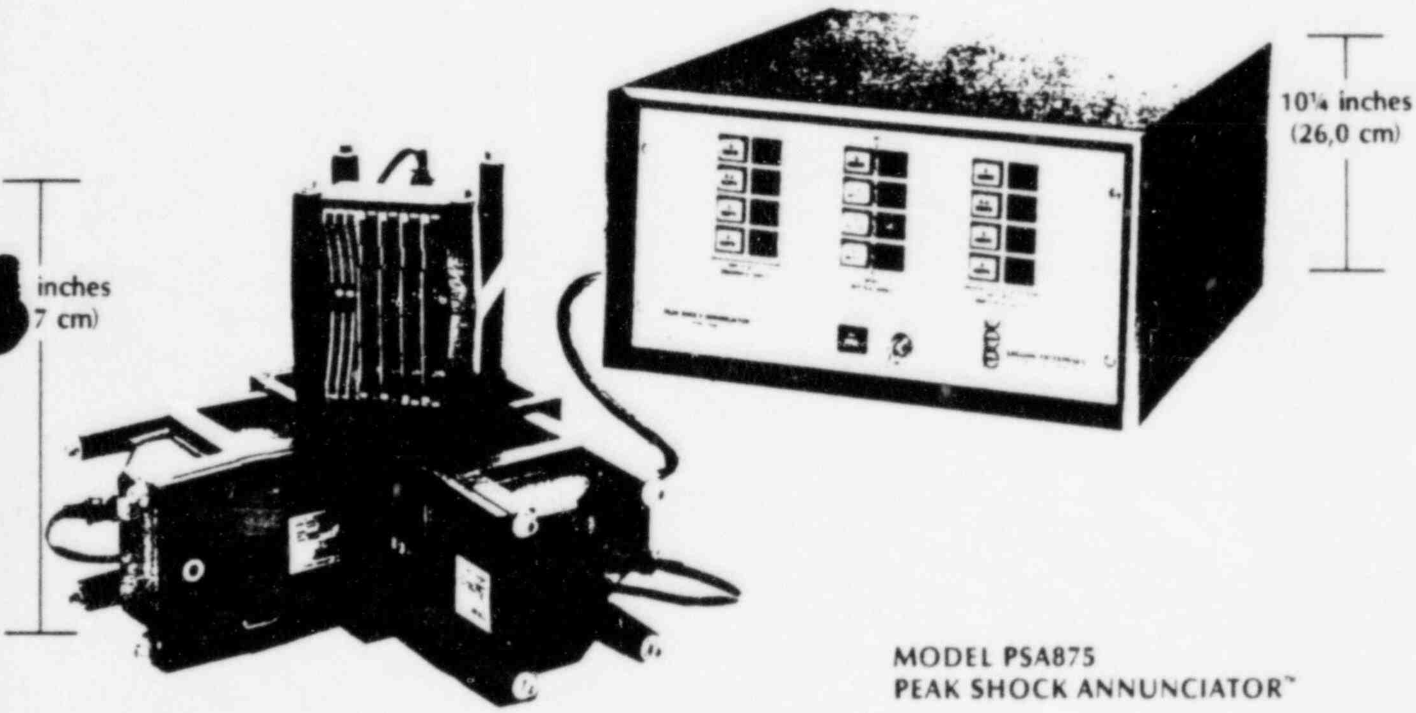
The "AC Power" indicator lamp is fed from the DC power on the printed circuit boards and shows that the incoming power line and the regulated DC supply are both operating normally.

A key-operated test/reset switch is provided. It controls two functions. In the "test" position, all of the indicator lamps should be illuminated and all relays (if provided) should be energized. This permits an immediate check that the annunciator is functioning correctly. When the key is returned to the "reset" position and removed from the switch, all indicators will be de-energized, the latches will be reset, and the annunciator is ready to receive signals from the Response Spectrum Recorder™. Once a signal has been received, the

appropriate lamp and relay, if any, will remain energized until the annunciator is reset with the key.

Where relays are provided for remote indicators or alarms, separate electronic driving circuitry is provided. Dual redundancy is thereby achieved for additional reliability.

Uninterruptible power supplies incorporating batteries for emergency operation can also be provided. If power failure is anticipated, battery operation is strongly recommended since power failure will reset any annunciated signal at the time of failure. Two additional indicators are mounted on the panel. One monitors the AC power at the transformer of the battery charger. The second monitors the charging circuit.



MODEL PSA875
PEAK SHOCK ANNUNCIATOR™

TRIAXIAL INSTALLATION OF THREE
MODEL PSR-1200-H/V-4A,
PEAK SHOCK RECORDERS™
ON TRIAXIAL MOUNT

PEAK SHOCK ANNUNCIATOR™ Models PSA875 and PSA1575

QUALIFED TO: GUIDE FOR SEISMIC QUALIFICATION OF CLASS I ELECTRICAL EQUIPMENT FOR NUCLEAR POWER GENERATING STATIONS – IEEE GUIDE 344

Designed to meet the characteristics of the Control Room Indicator for Response Spectrum Switch described in the American Nuclear Society's Standard ANSI/ANS-2.2-1978, Earthquake Instrumentation Criteria for Nuclear Power Plants and the U.S. Nuclear Regulatory Commission's Regulatory Guide 1.12, Nuclear Power Plant Instrumentation for Earthquakes, Revision 1.

PHYSICAL

	PSA875	PSA1575
Length	19 inches (48,3 cm)	19 inches (48,3 cm)
Width	20⅝ inches (52,4 cm)	20⅝ inches (52,4 cm)
Thickness	10¼ inches (26,0 cm)	17¼ inches (43,8 cm)
Weight	33 pounds (15 kg)	45 pounds (20,5 kg)

INDICATORS

Number of Axes Monitored	3	3
Number of Frequencies Monitored	12	36
Number of Indicators	24	72

ENVIRONMENTAL

Temperature	0 to +70°C
Humidity	To 100% RH
RFI	No adverse radiated or conducted RFI

POWER REQUIREMENTS

Voltage	115 VAC
Current	2½ amperes maximum

MOUNTING

Bench or
Standard 19" (48,3 cm) Relay Rack
8¾" (22,2 cm) high or
15¾" (40,0 cm) high
27 lbs. (12,3 kg)

Options and Accessories (available at extra cost)

1. **Relay closures for remote indication and alarm.** One relay with Form C contacts can be provided for each output indicator. A connector on the back of the chassis facilitates system implementation. The connector is wired for normally open or normally closed operation.

To date, most customers have selected a two-relay system. One relay indicates that the lower level (amber) has been exceeded. The second relay indicates the upper level (red) has been exceeded at least once.

Relays are rated at: 1/10 Hp, 3 amps @ 120 VAC or 3 amps @ 28 VDC resistive.

2. **Uninterruptible power supplies** incorporating batteries for emergency operation can be furnished within the confines of the annunciator. If power failure is anticipated, battery operation is strongly recommended since power failure will reset any annunciated signals at the time of the failure.

REPRESENTED BY:

1-85-1M 875700B

APPENDIX C

A PRELIMINARY EVALUATION OF THE SIGNIFICANCE OF THE SEISMIC EVENT ON JANUARY 31, 1986 AT THE PERRY NUCLEAR POWER PLANT

STEVENSON

STEVENSON & ASSOCIATES

a structural-mechanical consulting engineering firm

9217 Midwest Avenue • Cleveland, Ohio 44125 • (216) 587-3805 • Telex: 980101

Document Number 861401-1
Revision 0 -- 2/10/86

A PRELIMINARY EVALUATION OF THE SIGNIFICANCE OF THE SEISMIC EVENT ON JANUARY 31, 1986 ON THE PERRY NUCLEAR POWER PLANT

February 10, 1986

Perry Nuclear Power Plant
Cleveland Electric Illuminating Co.
10 Center Road
Perry, Ohio 44080

PREPARED BY:

Stevenson and Associates
9217 Midwest Avenue
Cleveland, Ohio 44125
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S&A

1.0 INTRODUCTION

On January 31, 1986, at 11:47 a.m. EST, a brief (approximately 0.75 second strong motion duration) and shallow (10 km focal depth) earthquake with a 4.9 m_b magnitude occurred. Its epicenter was south of Lake Erie, at a distance of approximately eleven (11) miles from the Perry Nuclear Power Plant site at Perry, Ohio.

Stevenson and Associates was retained to analyze the data provided by seismic recorders installed at various locations in the Perry plant, and determine: (1) how the earthquake parameters, as recorded by the instrumentation at the site, compare to those for the Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE) postulated in the design of the Perry plant's buildings, systems and components; (2) the structural significance of the readings by the seismic recorders at the Perry site during the January 31, 1986 earthquake; and (3) the anticipated impact of the earthquake on the plant's buildings, systems and components.

This report contains Stevenson and Associates' preliminary evaluation of the above-described matters. It is based on a physical walkdown of the site, analysis of data recorded by the seismic instrumentation, and discussions with plant technical and operating personnel. Since some of the evaluations of the earthquake are still underway, this report may be supplemented and/or revised at a later date if new information developed during these ongoing activities so warrants.

A resume of the qualifications and experience of Stevenson and Associates is included as Attachment 1 to this report.

2.0 SEISMIC INSTRUMENTATION AT THE PERRY PLANT

The earthquake motion at the Perry site was recorded by three different types of instrumentation. One type of recorder is the Kinemetrics Model SMA-3 strong motion time history recording accelerograph; this system detects and records the three orthogonal components of acceleration signals over the duration of an earthquake. Another type of instrumentation is the Engdahl PSR 1200-H/V response spectrum recorder, which provides the response at selected frequencies in three orthogonal directions. The third type of instrumentation is the Engdahl PAR 400 peak accelerograph, which records the three orthogonal components of peak local accelerations produced by the earthquake. The locations and readings taken by these systems will be discussed separately below.

2.1 Locations and Readings by the Kinemetrics SMA-3 Accelerographs

Two Kinemetrics SMA-3 strong motion time history recording accelerographs installed at the Perry plant provided time history data on the earthquake. One system is located on the Unit 1 reactor containment concrete wall at the basemat at Elevation 575', as shown in Figure 1. The second system is attached to the steel containment vessel wall at Elevation 686', 111 feet

above the first system and offset by less than one degree in Azimuth. The longitudinal axes of both instruments are in the N-S direction.

The time history motions recorded by these two systems are shown in Figures 2 through 8. A detailed interpretation of the readings from these recorders is contained in Reference 1.[1/]

The lower instrument (Elevation 575') gave a peak acceleration of 0.18g in the N-S direction, 0.10g in the E-W direction, and 0.11g in the vertical direction. The upper instrument (Elevation 686') gave a peak acceleration of 0.55g in the N-S direction, 0.18g in the E-W direction, and 0.30g in the vertical direction. It should be noted that both instruments are installed on cantilever brackets off the wall. While the brackets are quite heavy and relatively rigid, they are attached by four 3/8" diameter bolts, approximately 5 inches on center vertically and 8 inches horizontally. This arrangement may result in amplified bracket motion.

2.2 Locations and Readings of the Engdahl Response Spectra Recorders

There are four Engdahl PSR 1200-H/V triaxial response spectra recorders at the Perry plant. This type of recorder includes twelve reeds of different lengths and weights, one for each

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1/ References are listed at the end of this report.

frequency, fabricated from spring steel. A diamond-tipped stylus is attached to the free end of each reed to inscribe a permanent record of its deflection on one of twelve record plates. The record plates are aluminum, plated with successive layers of nickel, tin, and lead-tin.

The four PSR 1200-H/V recorders at the Perry plant are located as follows (all locations are for Unit 1):

1. <u>Reactor Building Foundation</u>: Elevation 574', Reactor Building foundation mat, Azimuth 210°. This recorder was most recently calibrated on January 14, 1985.

2. <u>Reactor Building Drywell Platform</u>: Inside the drywell platform at Elevation 630', Azimuth 240°, mounted as shown in Figure 9. This recorder was most recently calibrated on January 30, 1986.

3. <u>HPCS Pump Base Mat</u>: In the HPCS Pump Room, in the Auxiliary Building foundation mat, Elevation 574'. The equipment was being calibrated at the time of the earthquake. Previous calibration occurred on January 14, 1985.

4. <u>RCIC Pump Base Mat</u>: In the RCIC Pump Room in the Auxiliary Building foundation mat at Elevation 574'. The equipment was being calibrated at the time of the earthquake. Previous calibration was on January 14, 1985.

The readings taken by these four instruments are discussed in detail in Reference 2. Briefly stated, three of the four instruments provided response spectra which were consistent with each other and which were reasonable in light of the time history readings of the Kinemetrics instruments. The fourth spectra recorder, mounted inside the drywell on the Elevation 630' platform (see Figure 9), indicated vertical acceleration response components of .973g and 1.54g at frequencies of 20.2 and 25.4 Hz, respectively. These readings were 8 to 10 times higher than the corresponding horizontal accelerations at the same frequencies measured by the instrument. See Table 1.[2/]

2.3 Location and Readings of the Engdahl Peak Acceleration Recorders

The Engdahl Model PAR 400 peak acceleration recorder senses and records peak accelerations triaxially. A diamond tipped scriber at the end of an amplifier arm traces a very fine visible permanent record on an aluminum record plate with successive layers of nickel, gold, and burnt gold.

2/ Figure 9 shows the mounting of the Engdahl PSR 1200-H/V instrument on the Elevation 630' platform. The instrument is located approximately 6 feet from the face of the reactor vessel shield wall on an outer beam which provides supports for the platform, recirculation and safety injection piping, and a monorail. Given the highly complex nature of the steel platform and support structure on which the instrument is mounted, it is quite possible the instrument may have measured the acceleration caused by a secondary impact resulting from the earthquake.

The two peak acceleration recorders are located as follows:

1. Reactor Recirculation Pump: Inside the drywell at Elevation 574', on recirculation pump B33-C001A. This instrument was most recently calibrated on December 4, 1985.

2. HPCS Pump Base Mat: In the HPCS Pump Room, in the Auxiliary Building foundation mat at Elevation 574', mounted as shown in Figure 10. This instrument was most recently calibrated on January 30, 1986.

The readings by the Engdahl PAR 400 recorders are discussed in detail in Reference 2.

3.0 COMPARISON AND EVALUATION OF RECORDED ACCELERATIONS AGAINST THOSE ASSUMED FOR THE PERRY SSE AND OBE

Table 2 shows a comparison of the zero period acelerations ("ZPAs"), as recorded by the various instruments, with the corresponding SSE and OBE design accelerations. According to the recorded accelerations, the design basis values of ZPA for the OBE, and in a few instances the SSE, were exceeded during the January 31, 1986 earthquake. As will be discussed below, given the short duration and low energy of the earthquake, the exceedences were not significant from an engineering point of view. This is supported by the apparent lack of damage to plant structures and mechanical and electrical components detected as a result of the earthquake. Moreover, inspection of

engineered facilities located near the epicenter and not designed to withstand any earthquake force did not reveal any damage from the earthquake (Reference 3). In order to correlate the short duration, high frequency acceleration that was recorded with the lack of impact on structures and equipment, it is necessary to understand how measured ground acceleration can and should be correlated with design basis accelerations.

In postulating the limiting earthquake conditions for designing nuclear power plant facilities, a key parameter has been the zero period acceleration or Instrumental Peak Acceleration (A_{IP}), which represents the peak acceleration recorded during the entire earthquake motion. As concluded in many studies (References 4 through 11), A_{IP} is a poor indicator of the damage potential of earthquake ground motions. It has been observed that structures performed much better than would have been predicted based on the measured A_{IP} to which the structures were subjected; this phenomenon has been particularly noticeable in connection with short duration, high energy ground motions due to low to moderate magnitude earthquakes, such as the January 31, 1986 earthquake near Perry.[3/] The differences

3/ Examples of this behavior may be found in the records of the 1966 Parkfield earthquake, the 1971 Pacoima Dam earthquake, the 1972 Ancona earthquake, and the 1972 Melendy Ranch Barn earthquake. These earthquakes showed recorded instrumental peak ground accelerations of between 0.5g and 1.2g, yet only minor damage occurred in the vicinity of the recording sites.

between measured ground motion, assumed design levels, and observed physical behavior is so significant that it cannot be attributed to the safety factors which are utilized in the design and in elastic seismic analyses.

Kennedy (Reference 12), based on the work of others (References 13 through 16) has suggested that it is not appropriate to use just measured A_{IP} to define the characteristics of the SSE and OBE. It is necessary to take also into account, in addition to A_{IP}, the dominant frequency of the strong motion excitation and the duration of the strong motion.[4/] He has proposed the following relationship to develop an equivalent design acceleration for the anchoring elastic spectra:

$$A_D = (K_p)\ (\text{rms}),$$

where A_D is the equivalent design acceleration and the other parameters are defined as follows:

$$K_p = \sqrt{2 \ln (2T_D/To)} \geq 2.0$$

$$T_D = \text{Duration of strong motion (sec.)}$$

4/ Thus, for a high dominant frequency and/or short duration earthquake, the equivalent peak acceleration would be significantly less than that predicted on the basis of A_{IP} measurements alone.

T_o = Predominant period of motion (sec.)

rms = $\sqrt{P}$

P = $E(T)/T_D$ = earthquake power (average rate of energy input)

$$E_T = \int_{t_o}^{t_o + T_D} a^2(t)dt = \text{total energy}$$

fed into the structure between times t_o and $t_o + T_D$, and

$a(t)$ = instrument acceleration at time t.

Efforts are underway to compute A_D for the January 31, 1986 earthquake. In the meantime and by way of comparison, four earthquakes similar in magnitude and duration to the Perry earthquake have been selected from Tables 1 and 2 of Reference 12. The characteristics of those earthquakes, and those of the one at Perry, are summarized in Table 3. For the four earthquakes listed, an average ZPA of 0.434g is required to cause the same level of response for elastic structures as that postulated by the NRC Reg. Guide 1.60 (Reference 17) spectra for a .20g ground acceleration. This result suggests that a correction factor of 0.20/0.434 = 0.46 should be applied to the accelerations measured during low to moderate magnitude earthquakes (such as the one near Perry) to obtain elastic responses

that can be compared to those from the limiting Reg. Guide 1.60 earthquake.

If, in fact, a 0.46 correction factor is applied to the accelerations recorded at Perry and shown in Table 2, accelerations well below the SSE and OBE levels are obtained for all locations except for the readings at the Reactor Building Containment Vessel (Elevation 686'), where the corrected N-S and vertical ZPA are approximately equal to the OBE design value. This is shown in Table 4, where the recorded values of Table 2 have been adjusted by a .46 factor.

4. STRUCTURAL SIGNIFICANCE OF THE PERRY EARTHQUAKE AND ANTICIPATED IMPACT OF EVENT ON THE ADEQUACY OF THE PLANT STRUCTURES, SYSTEMS AND COMPONENTS

Table 4 indicates that if the recorded accelerations from the Perry earthquake are corrected to take into account the short duration and low energy of the event, the average elastic response ZPAs are in all but one instance equal to or less than one-third of the OBE design values, and are approximately equal to the OBE values in the remaining case. In light of these results and the design limits placed on the strength of materials for safety applications (<u>i.e.</u>, not to exceed a 0.6 to 0.8 factor of yield during an OBE), all safety-related plant structures, systems and equipment should have remained essentially elastic during an earthquake such as the one experienced on

January 31, 1986, and thus should have emerged undamaged from it. This expectation has been corroborated by physical observation of plant conditions following the earthquake.

Some auxiliary or secondary structural systems, such as suspended ceilings and plaster ceilings and walls, might be expected to sustain some displacement or cracking. One might also expect actuation of instrumentation measuring or sensing changes in liquid levels or the presence of vibration. In addition, one might expect some activation of inertia-sensing relays or switches (fluid or spring loaded), if such controls or instrumentation have not been qualified for seismic operability. If any of these circumstances are determined to have taken place at Perry, their occurrence would only be indicative of the anticipated response of non-seismically qualified structures to moderate earthquake conditions.

TABLE 1 (From Reference 2)

READINGS FROM RESPONSE SPECTRA RECORDER
MPL NUMBER: D51-R170
LOCATION: REACTOR RECIRCULATION
PIPING SUPPORT - DW 630', 240°

REED NUMBER	NOMINAL FREQUENCY (HERTZ)	ACCELERATION(g) North/South 1-31-86		East/West 1-31-86		Vertical 1-31-86	
		North/South 1-31-86	North/South 2-2-86	East/West 1-31-86	East/West 2-2-86	Vertical 1-31-86	Vertical 2-2-86
1	2.00	.047	.048	.049	.051	.007	.007
2	2.52	.082	.082	.086	.084	(*)	.013
3	3.17	.184	.184	.144	.140	.015	.014
4	4.00	.226	.223	.128	.127	.023	.023
5	5.04	.132	.134	.158	.158	.035	.033
6	6.35	.131	.134	.058	.055	.033	.030
7	8.00	.104	.104	.109	.090	(*)	.019
8	10.1	.093	.093	(*)	.052	.093	.085
9	12.7	.188	.182	.166	.080	.198	.199
10	16.0	.194	.204/.167	.348	.312	.490	.500
11	20.2	.152	.152	.191	.175	.973	.973
12	25.4	.114	.091	.155	.158	1.7	1.54

(*) Unreadable

S&A

TABLE 2

COMPARISON OF DESIGN ZPAs (1)
VS RECORDED ZPAs
(Expressed in g values)

		Auxiliary Building Foundation Mat Elevation 568' PAR 400 (Engdahl) D51-R140	Reactor Building Foundation Mat Elevation 574'-10" SMA -3 (Kinemetrics) D51-N101	Reactor Building Recirculation Pump Elevation 605' PAR 400 (Engdahl) D51-R120	Reactor Building Containment Vessel Elevation 686' SMA-3 (Kinemetrics) D51-N111	Reactor Building Platform Elevation 630'-1" Inside Drywell PSR 1200 (Engdahl) D51-R170
	Recorded	.17	.18	.32	.55	.09
NS	SSE	.17	.18	1.06	.40	.48
	OBE	.10	.10	.86	.24	.40
	Recorded	.06	.10	.11	.18	.16
EW	SSE	.20	.18	1.06	.40	.48
	OBE	.10	.10	.86	.24	.40
	Recorded	.03	.11	.05	.30	Note 2
VERT	SSE	.20	.18	.47	.24	.28
	OBE	.10	.10	.38	.15	.16
	Recorded	.18	.23	.34	.65	Note 2
SRSS(3)	SSE	.33	.31	1.57	.62	.73
	OBE	.17	.17	1.27	.37	.59

(1) Zero period acceleration
(2) ZPA indeterminable from available data
(3) Square-root-of-the-sum of the squares

TABLE 3

CHARACTERISTICS AND GROUND ACCELERATION LEVELS REQUIRED TO ACHIEVE EQUAL STRUCTURAL ELASTIC RESPONSE BETWEEN R.G. 1.60 AND SELECTED EARTHQUAKES

Earthquake	Magnitude M_V	Recording Station Epicentral Distance(km)	Peak Inst. Ground Acceleration, g	Strong Motion Duration, sec.	Equiv. ZPGA to the 0.20g R.G. 1.60 Spectra
Parkfield - 1966	5.6	1	0.49	1.4	.3275
Hollister - 1974	5.2	13	0.138	1.1	.4825
Santa Barbara - 1978	5.1	4	0.347	3.0	.2825
Bear Valley-1972	4.7	6	0.520	0.8	.6450
					.434 (Average)
Ohio - 1986	4.9	17	(*)	0.75	--

(*) 0.18g in N-S direction, 0.10g in E-W direction, measured at the foundations.

TABLE 4

COMPARISON OF DESIGN ZPAs (1)
VS CORRECTED RECORDED ZPAs
(Expressed in g values)

		Auxiliary Building Foundation Mat Elevation 568' PAR 400 (Engdahl) D51-R140	Reactor Building Foundation Mat Elevation 574'-10" SMA -3 (Kinemetrics) D51-N101	Reactor Building Recirculation Pump Elevation 605' PAR 400 (Engdahl) D51-R120	Reactor Building Containment Vessel Elevation tion 686' SMA-3 (Kinemetrics) D51-N111	Reactor Building Platform Elevation 630'-1" Inside Drywell PSR 1200 (Engdahl) D51-R170
NS	Recorded	.08	.08	.15	.25	.04
	SSE	.17	.18	1.06	.40	.48
	OBE	.10	.10	.86	.24	.40
EW	Recorded	.03	.05	.06	.08	.07
	SSE	.20	.18	1.06	.40	.48
	OBE	.10	.10	.86	.24	.40
VERT	Recorded	.02	.06	.02	.14	Note 2
	SSE	.20	.18	.47	.24	.28
	OBE	.10	.10	.38	.15	.16
SRSS(3)	Recorded	.08	.11	.16	.30	Note 2
	SSE	.33	.31	1.57	.62	.73
	OBE	.17	.17	1.27	.37	.59

(1) Zero period acceleration
(2) ZPA indeterminable from available data
(3) Square-root-of-the-sum of the squares

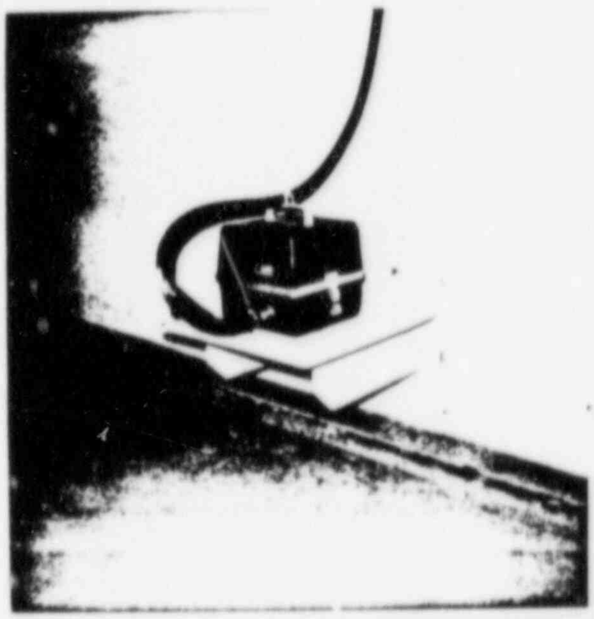

FIGURE 1 - KINEMETRICS [illegible]
ACCELERATION TIME HISTORY [illegible]
SERIAL NO. 165-1, TAG NO. [illegible]
LOCATED AT BASE OF C[illegible]
ELEVATION 575.

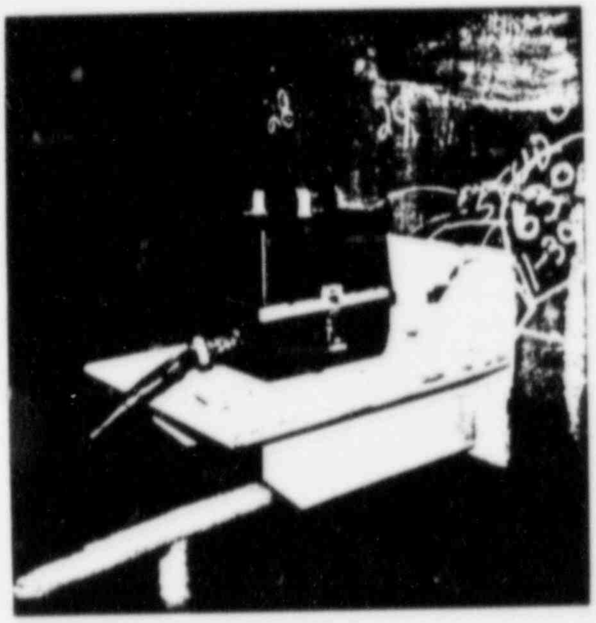

FIGURE 2 - KINEMETRICS SMA-3
ACCELERATION TIME HISTORY [illegible]
SERIAL NO. 165-2, TAG NO. [illegible]
LOCATED ON THE STEEL CONTAINMENT
SHELL AT EL. 692.

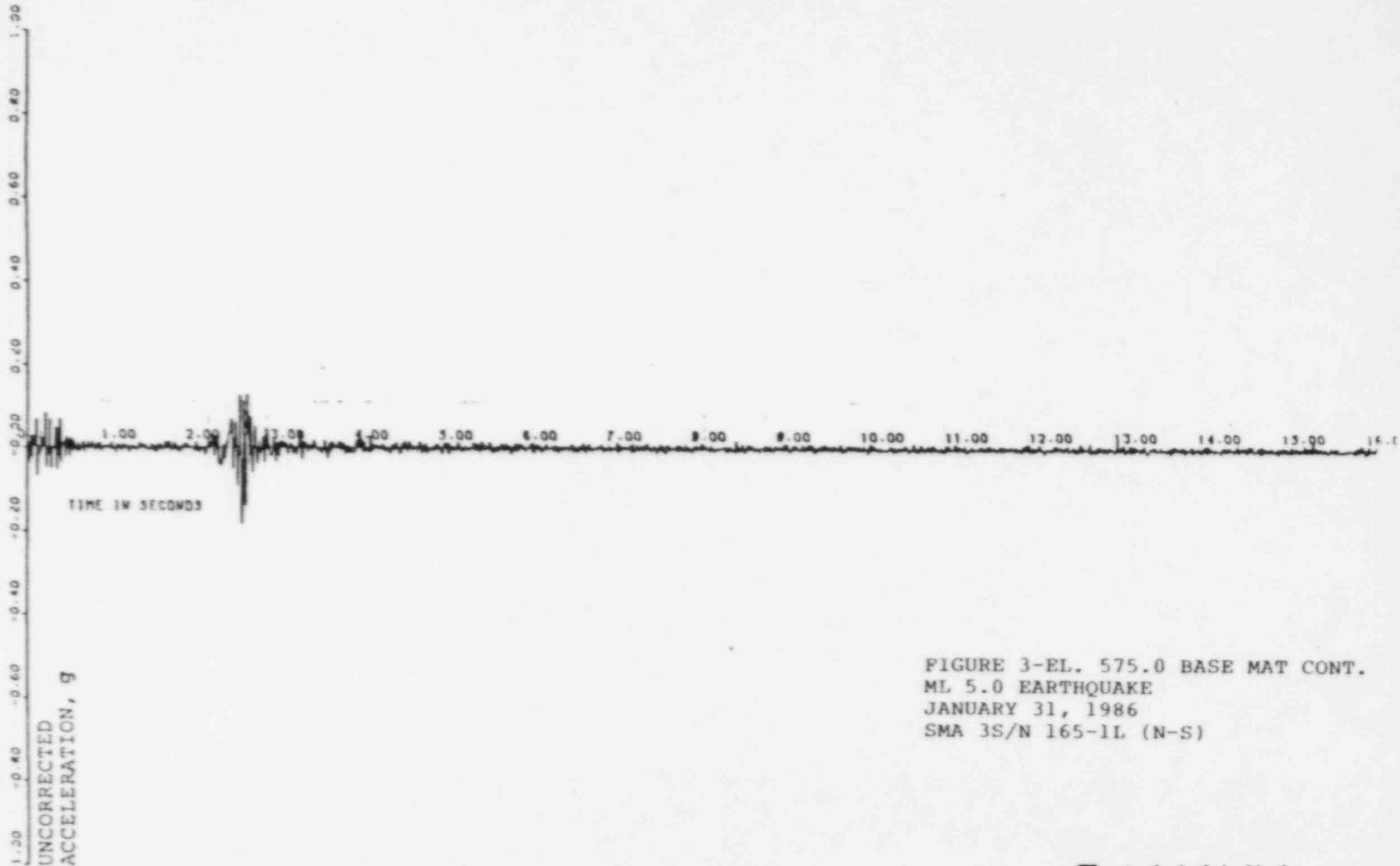

FIGURE 3-EL. 575.0 BASE MAT CONT.
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-1L (N-S)

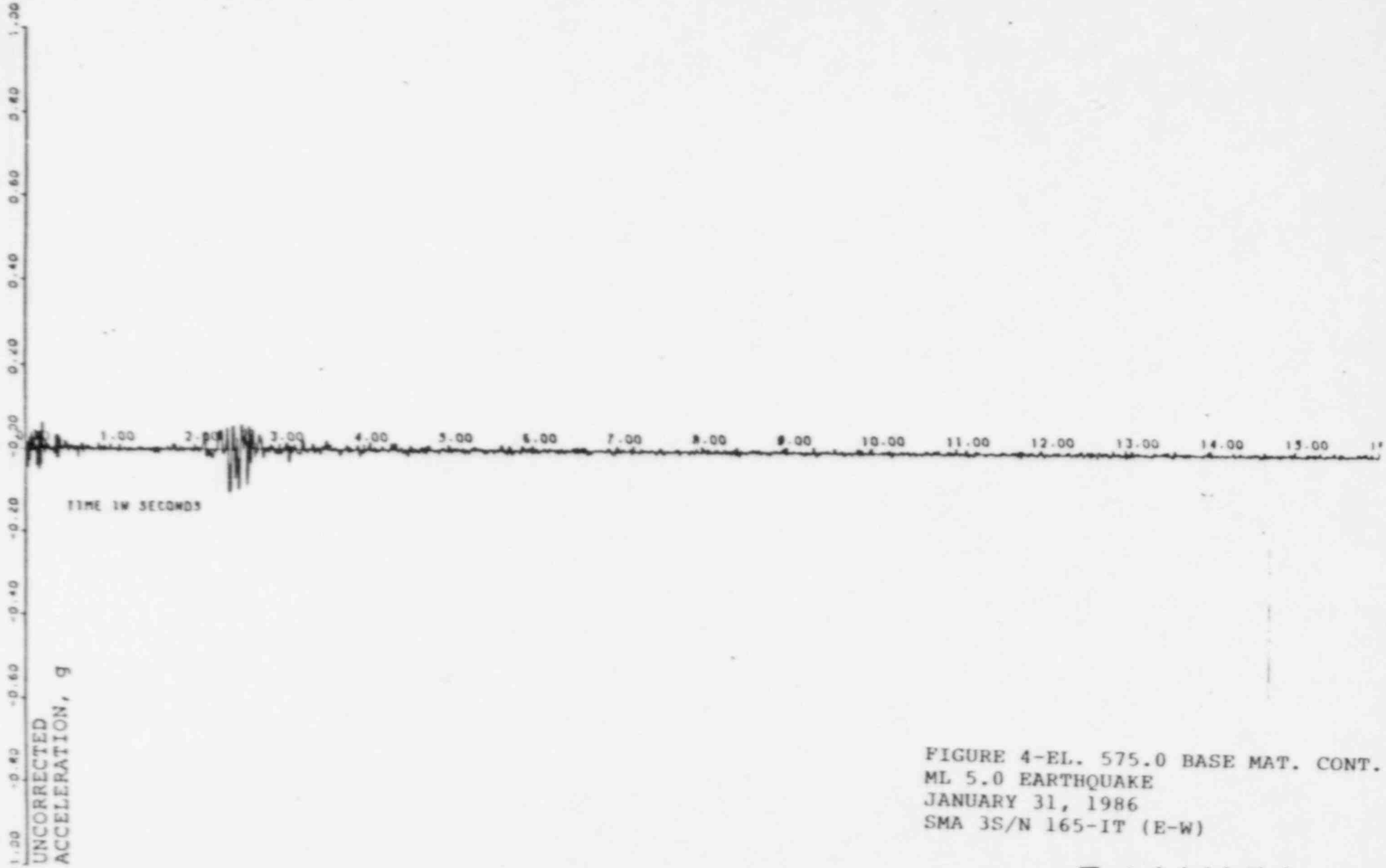

FIGURE 4-EL. 575.0 BASE MAT. CONT.
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-IT (E-W)

KINEMETRICS

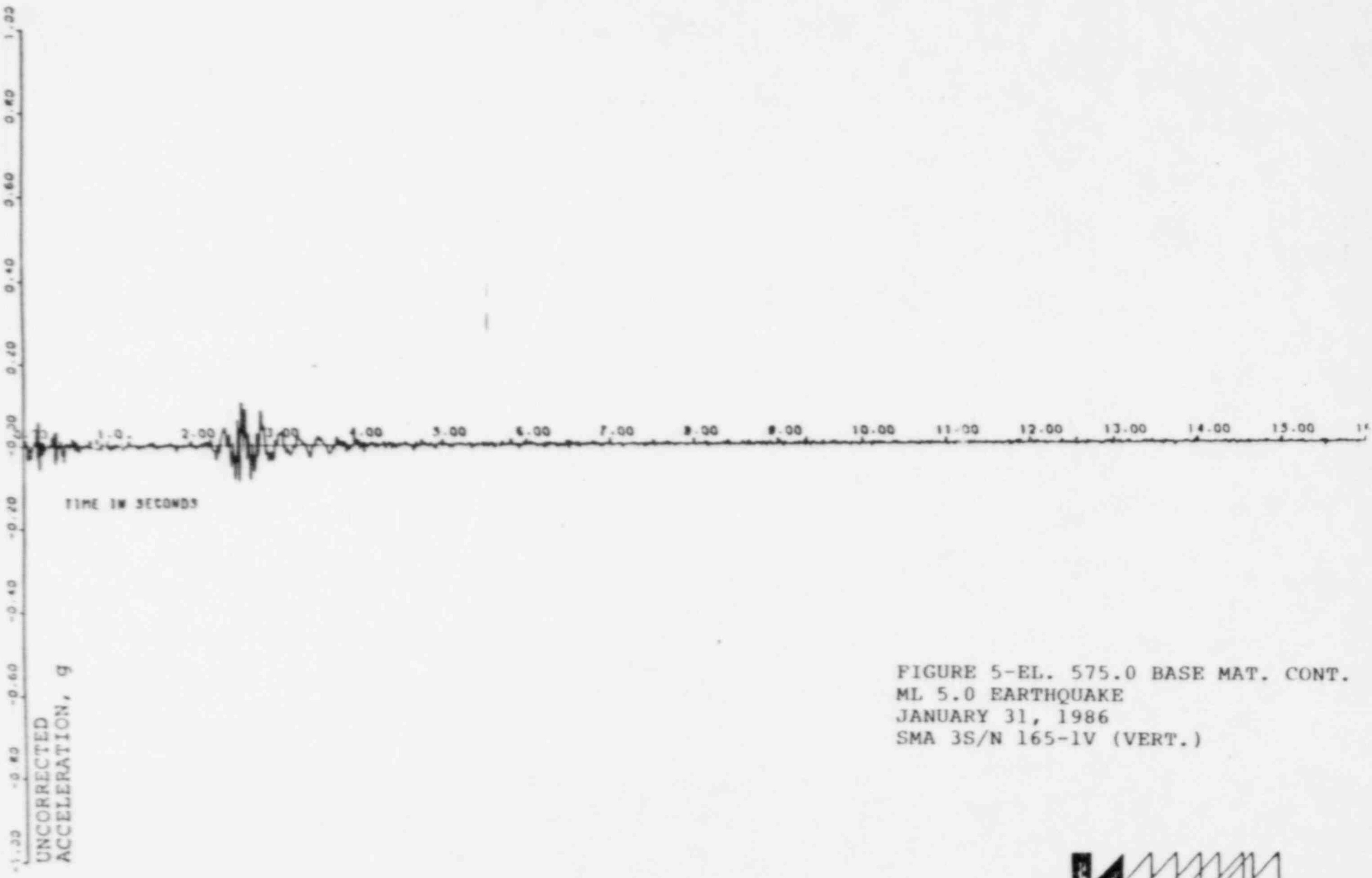

FIGURE 5-EL. 575.0 BASE MAT. CONT.
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-1V (VERT.)

KINEMETRICS

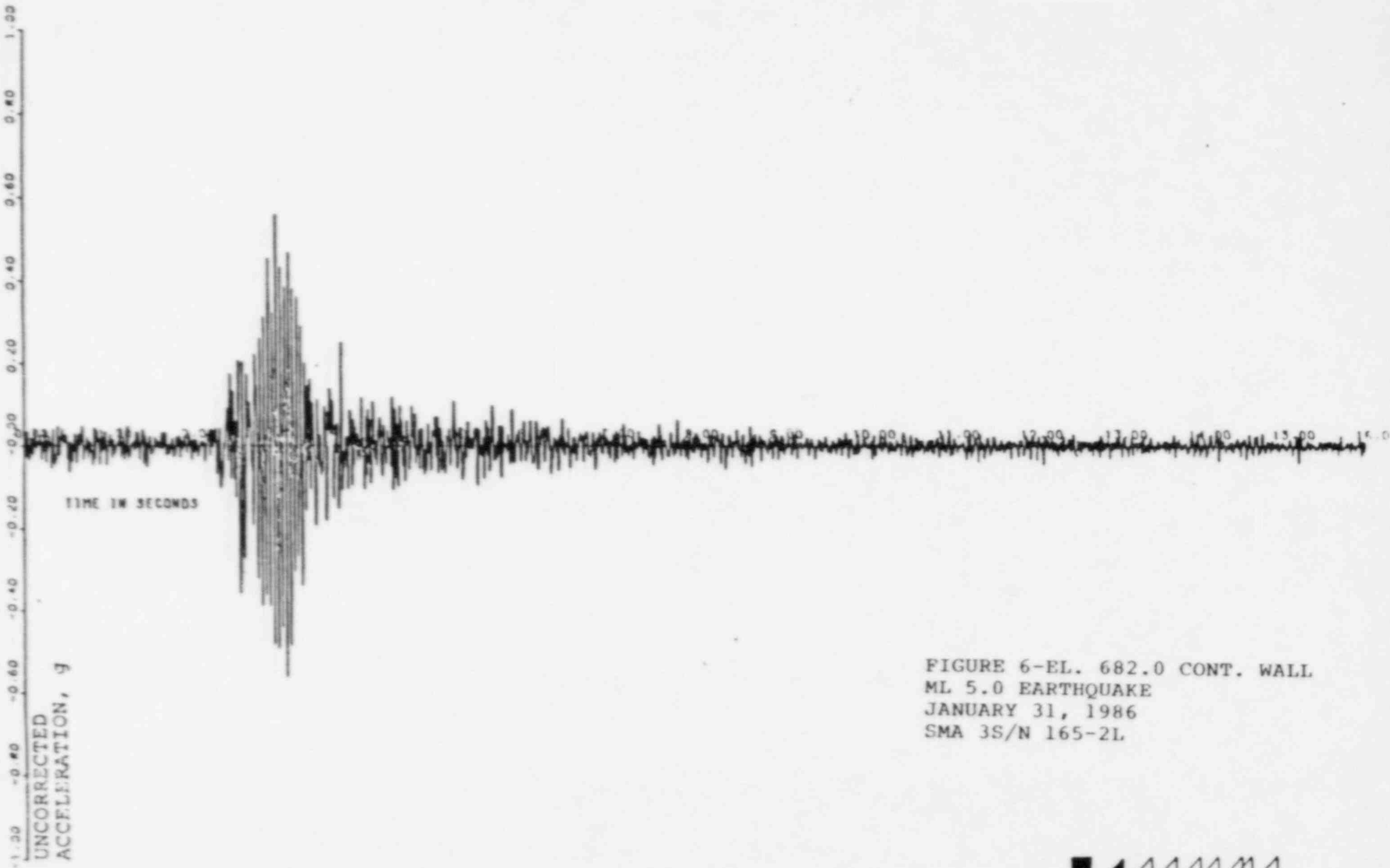

FIGURE 6-EL. 682.0 CONT. WALL
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-2L

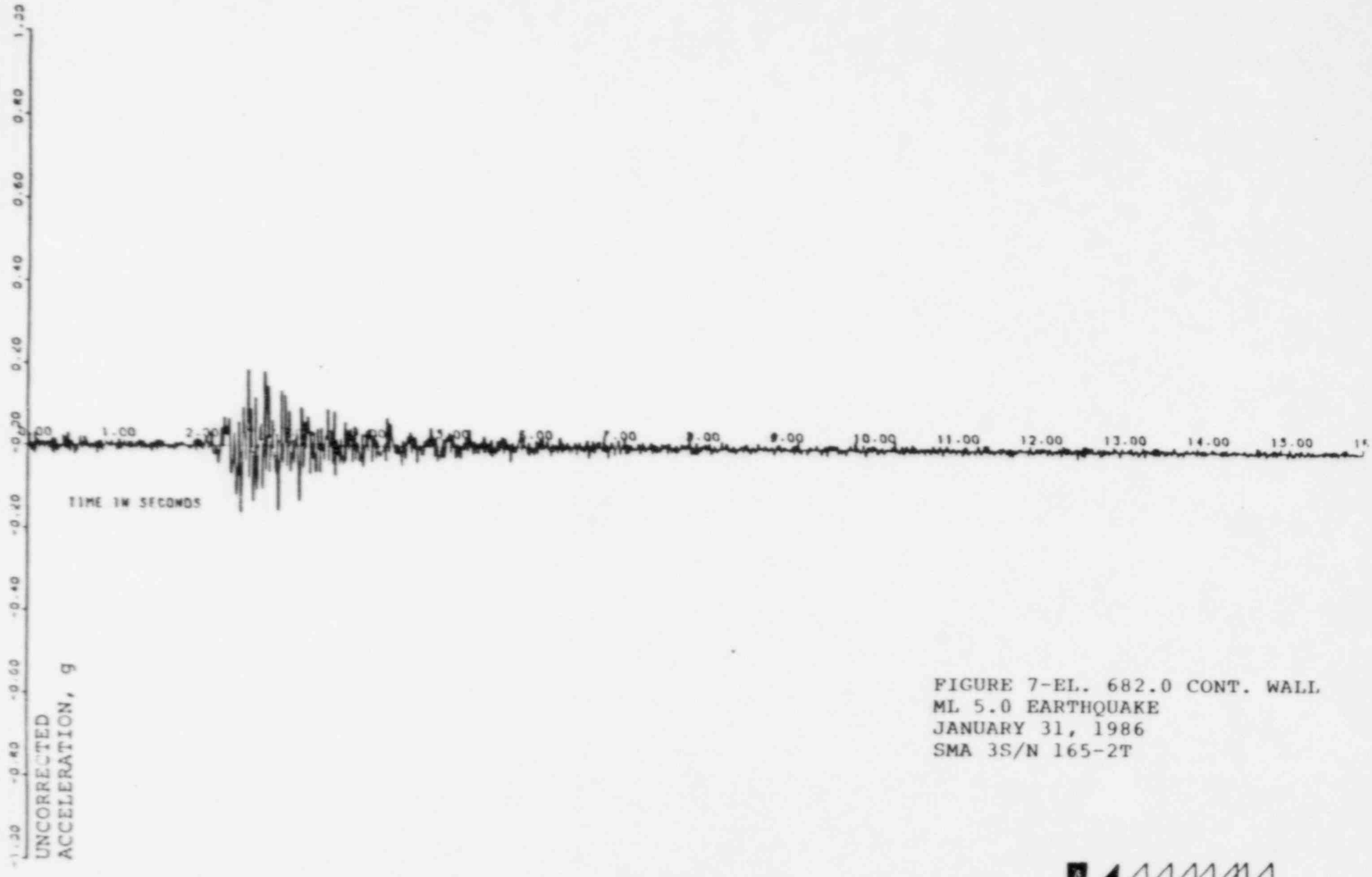

FIGURE 7-EL. 682.0 CONT. WALL
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-2T

KINEMETRICS

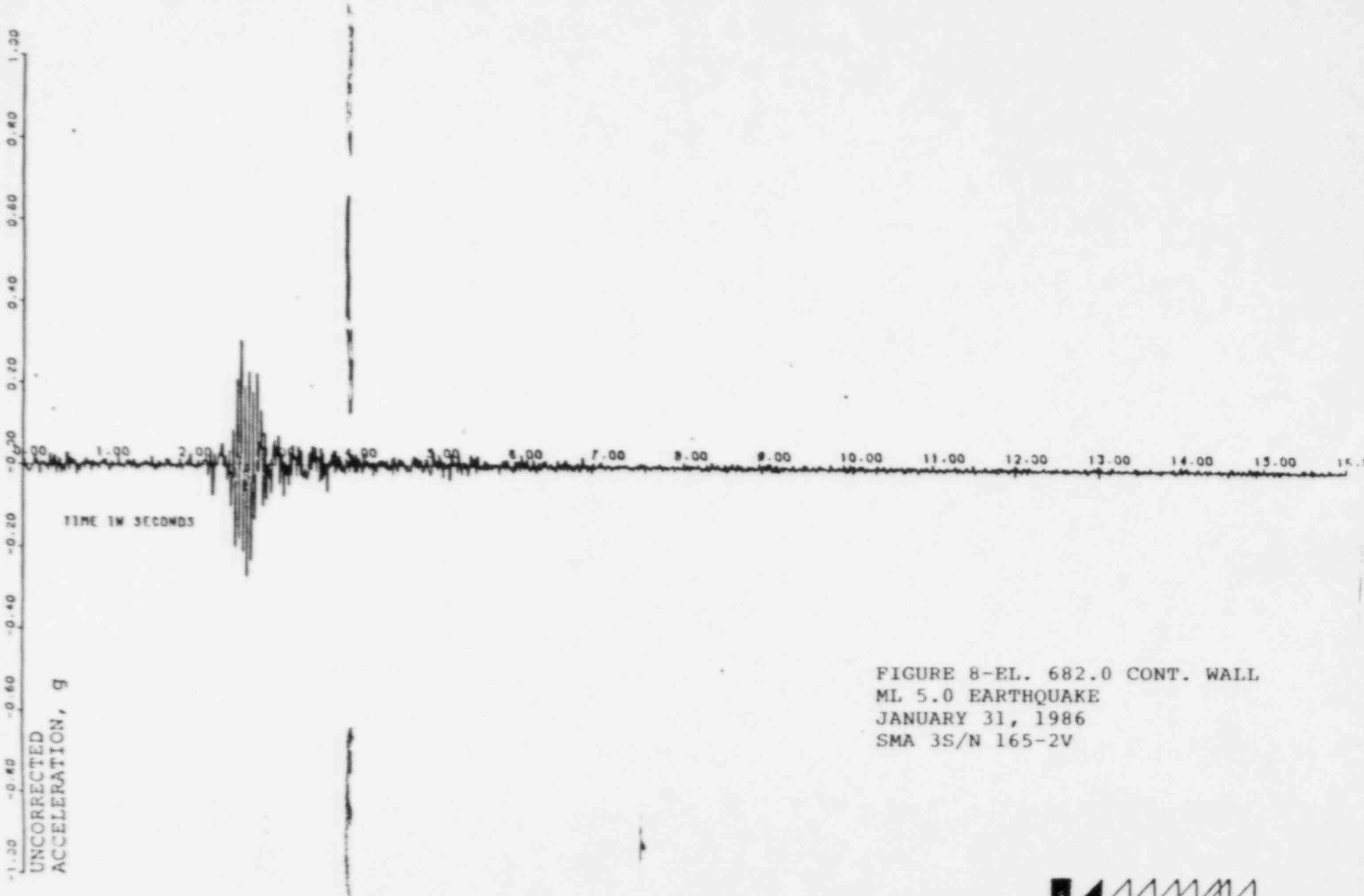

FIGURE 8-EL. 682.0 CONT. WALL
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-2V

KINEMETRICS

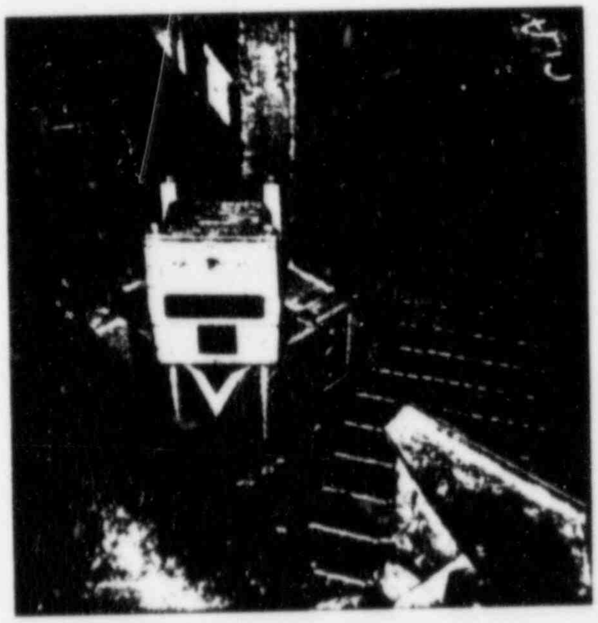

FIGURE 9 - MOUNTING OF ENGDAHL PSR 1200-H/V RECORDER ON THE EL. 43[illegible] DRYWELL PLATFORM.

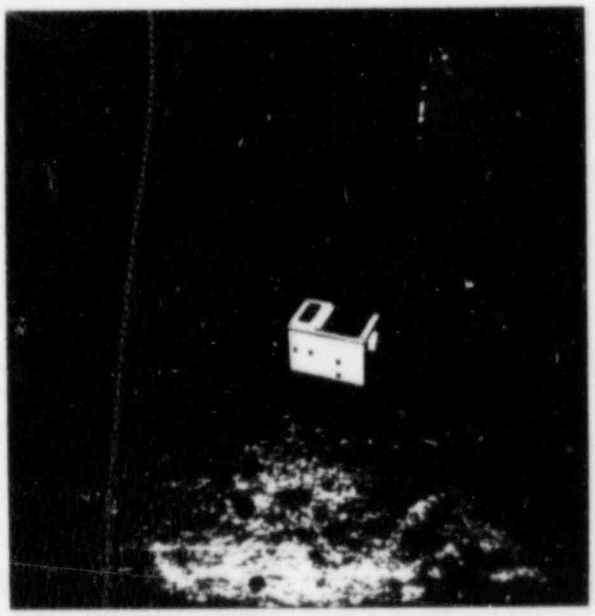

FIGURE 10 - MOUNTING OF
ENGDAHL TYPE PAR 4[illegible]
RECORDER ON [illegible]
BUILDING [illegible]
EL. 563.

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JOHN D. STEVENSON

EXPERIENCE:

PRESIDENT - MANAGING PARTNER	Since November 1981, Dr. Stevenson has managed and has served as President and Senior Consultant to Stevenson & Associates. The firm specializes in high technology consulting and forensic engineering associated with failure analysis of structural and mechanical systems; extreme loads; and nonlinear, dynamic, and probabilistic high temperature analyses.
VICE-PRESIDENT - GENERAL MANAGER 1976 - 1981	As Vice-President, Dr. Stevenson managed and served as Senior Engineering Consultant to the Cleveland Offices of Woodward-Clyde Consultants and Structural Mechanics Associates specializing in areas of high technology applicable to the structural-mechanical design and analysis of systems and components. Prior to this time, the consulting group he headed provided similar services as a Division of Davy-McKee Co. Dr. Stevenson also served as Corporate Manager of Engineering Quality Assurance for Davy-McKee Co.
ASSOCIATE PROFESSOR AND PRINCIPAL MANAGER OF EASTERN OPERATIONS 1974 - 1976	Case Western Reserve University, CWRU, and EDAC, Inc., Cleveland, Ohio. As an Associate Professor at CWRU, Dr. Stevenson served as Director of a program in Design for the Extreme Load Environment and held a joint appointment in the Departments of Civil Engineering and Mechanical Design. He also conducted a number of seminars on Seismic Quality Assurance Scheduling and Manpower Requirements and Mechanical and Electrical Equipment Pipe and Duct Design of Industrial Facilities. Dr. Stevenson was a Principal and managed one of three consulting offices for Engineering Decision Analysis Corp., Palo Alto, California. He was active in marketing and providing consulting services in the area of extreme load, seismic, tornado, high energy systems rupture, and component failure analysis.
CONSULTANT 1973 - 1974	Westinghouse Nuclear Energy Systems, Pittsburgh, Pennsylvania. As a Consulting Engineer for Westinghouse Nuclear Energy Systems, Dr. Stevenson acted as an advisor to the Technical Director on the Executive Vice-President for Nuclear Power Staff. He performed evaluations of balance of plant requirements associated with nuclear power plant design and constructed and represented Westinghouse on a number of Industry Committees associated with nuclear power.

CONSULTANT
1972 - 1973

Westinghouse Water Reactor Divisions, Pittsburgh, Pennsylvania.

As an Advisory Engineer for the Westinghouse Standard Plant Project, Dr. Stevenson acted as a consultant to the Manager of the Westinghouse Standard Plant Project. In this capacity he had responsibility for determining interface requirements with site-related design parameters and set envelope requirements for the standard plant design. He was responsible for nuclear island PSAR text developments and AEC licensing requirements associated with the standard plant layout development.

ADJUNCT PROFESSOR AND PRESIDENT
1970 - 1972

University of Pittsburgh and NSSA Inc., Pittsburgh, Pennsylvania.

As a member of the Civil Engineering Facility of the University of Pittsburgh, Dr. Stevenson was particularly active in the areas of structural dynamic response to earthquake, tornado, missile and fluid jet effects as well as reliability and risk analysis and optimum design of structural systems. Dr. Stevenson was responsible for the development of a graduate study program for the study of structural design and analysis for the extreme load environment.

Dr. Stevenson founded and served as President and Managing Director of Nuclear Structural Systems Associates, Inc. During this period, the firm served as consultants to the nuclear power industry, particularly in the areas of structural and mechanical design and licensing of nuclear plant facilities. Dr. Stevenson was active in developing Standard Plant design concepts and also conducted engineering design seminars for the nuclear industry throughout the U.S., Europe and Japan for over 500 representatives of over 150 companies.

MANAGER STRUCTURAL SYSTEM ENGINEERING
1968 - 1970

Westinghouse PWR Systems Division, Pittsburgh, Pennsylvania.

Dr. Stevenson had overall responsibility within Westinghouse for the development and approval of structural design criteria and layout used in the design of the six nuclear power stations for which Westinghouse had prime design and construction responsibility for product line management of design and development of support structures for major nuclear components.

LEAD ENGINEER 1966 - 1968	Westinghouse PWR Systems Division, Pittsburgh, Pennsylvania. As Lead Engineer, Dr. Stevenson was responsible for liaison with the various architect-engineer-constructor firms which performed the detailed structural design and construction of turnkey plants, and as such he was responsible for design review and approval. Dr. Stevenson was active in representing Westinghouse structural design policy before the Atomic Energy Commission and Advisory Committee on Reactor Safeguards.
GRADUATE STUDENT 1963 - 1966	Case Institute of Technology, Cleveland, Ohio. Work toward a Ph.D. in Structures with emphasis on computer applications and risk analysis applied to structural design.
RESEARCH ENGINEER 1962 - 1963	I.I.T. Research Institute, Chicago, Illinois. Responsibilities included integrated radiation, structural and operational analysis and minimum cost design of nuclear blast resistant underground structures.
ASSISTANT PROFESSOR 1957 - 1962	Virginia Military Institute, Lexington, Virginia. Courses in structural design of concrete and steel structures were taught to Civil Engineering undergraduates. John Hopkins University, Baltimore, Maryland (Part-Time) Research Assistant. Responsibilities included report editing and research in the location, type quantity and packaging of low level solid atomic wastes.
FIELD ENGINEER 1956 - 1957	McDowell Construction Co., Cleveland, Ohio Field Engineer responsible for Technical Supervision and engineering field modifications to construction of a Sintering Plant for U.S. Steel Corp. Youngstown Works.

Dr. Stevenson has been particularly active in the review and evaluation of design adequacy of structures and equipment in nuclear power plants and other industrial facilities. Particular projects where he personally performed such evaluations include the following:

Nuclear Power Plants:

Indian Point Units 2 & 3
H.B. Robinson
R.E. Ginna
Point Beach
Dresden 2
Monticello
D. C. Cook
Palisades
Oyster Creek
Millstone
South Texas Project
Fessenheim - France
Cordoba - Argentina
Mihama - Japan
Conn. Yankee
Maine Yankee
Midland

Other Industrial Facilities:

Tokamac Fusion Test Facility
Purex Facility Hanford
Rocky Flats Processing Facility
Centrifuge Plant
Granger Soda Ash Plant
LMFBR
Hercules Polypropalene Plant
Shuichang Steel Complex
Touss Oil Fired Power Station
Hanford Coal Fired Power Station
Addy Ferro Silicate Plant
Killen Coal Fired Power Station
LNG Storage Facilities - U.S.

EDUCATION:

B.S. - Civil Engineering -
Virginia Military Institute, 1954

AEC Institute on Nuclear Engineering -
Purdue University, Summer 1960

M.S. - Civil Engineering -
Case Institute of Technology, 1962

Ph.D. - Civil Engineering -
Case Institute of Technology, 1968

PROFESSIONAL:

1. Member: American Society of Civil Engineers
   Chairman: Executive Committee Technical Council Codes And Standards
   Chairman: Nuclear Standards Committee
   Member: Structural Division Committee on Nuclear Safety
   Member: Structural Division Committee on Nuclear Structures and Materials

2. Member: American Concrete Institute
   Member: Joint ACI-ASME Subgroup on Design of Concrete Components in Nuclear Service, ASME BPVC-Section III-Div. 2, Corresponding Consultant ACI 349 Safety Class Concrete Structures

3. Member: American Society of Mechanical Engineers
   Member: Subgroup on Design of ASME BPVC-Section III-Div. 1 Nuclear Components
   Member: Subcommittee on Qualification of Mechanical Components in Nuclear Service

4. Member: Nuclear Standards Management Board of ANSI representing ASCE

5. Member: U.S. Respresentative International Standards Committee SC 85/3/7 on Seismic Criteria for Nuclear Plants

6. Member: U.S. Representative International Atomic Energy Agency Working Group on the Development of Seismic Design Standards

7. Vice Chairman: ANS-2, American Nuclear Society Committee on Site Evaluation
   Member: NUPPSCO, American Nuclear Society Committee on Nuclear Power Plant Codes and Standards

8. Member: AISC, American Institute of Steel Construction Committee on Specifications for Structural Steel in Safety Class Nuclear Structures

9. Member: Earthquake Engineering Research Institute

10. Register Professional Engineer: Virginia, Pennsylvania, and Ohio

11. Winner: Moiseiff Award - ASCE, 1971

PUBLICATIONS:

1. Stevenson, J.D., and Haga, P.G., "Pressurized Water Reactor Containment Structures Design Experience," Journal of the Power Division, ASCE, Vol. 96, No. PO 1, Proc. Paper 7037, January 1970.

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5. Stevenson, J.D., and Abrams, J.I., (Editors) "Proceedings of Symposium on Structural Design of Nuclear Power Plant Facilities," University of Pittsburgh, 1972.

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8. Stevenson, J.D. (Editor), "Proceedings of Symposium on Structural Design of Nuclear Power Plant Facilities," ASCE Speciality Conference, Chicago, December 1973.

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10. Stevenson, J.D. (Editor), "Proceedings of Second ASCE Speciality Conference on Structural Design of Nuclear Power Plant Facilities," New Orleans, December 1975.

11. Stevenson, J.D., "Rational Determination of Operational Basis Earthquake and its Impact on Overall Safety and Cost of Nuclear Facilities," Nuclear Engineering and Design, Vol. 135, North Holland Publishing Company, 1975.

12. Stevenson, J.D. "Survey of Extreme Load Design Regulatory Agency Licensing Requirements for Nuclear Power Plants," Nuclear Engineering and Design, Vol. 36, North Holland Publishing Company, 1976.

13. Stevenson, J.D., "External Hazards in Reliability and Risk Assessment of Nuclear Power Plants," Proceeding 2nd International Conference on Structural Safety and Reliability, Munich, Sept. 1977.

14. Stevenson, J.D., "Preliminary Design of a Containment to Withstand Core Melt for a 1300 MWe LWR System," Nuclear Engineering and Design, Vol. 48, North Holland Publishing Company, June 1978.

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16. Stevenson, J.D., "Research Needs Associated with Seismic Load on Nuclear Power Plants," Nuclear Engineering and Design, Vol. 50, North Holland Publishing Co., October 1978.

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18. Stevenson, J.D., "Standards - Status and Development in the Nuclear Industry," Proceedings of ASCE Speciality Conference on Design of Nuclear Plant Facilities, Boston, April 1979.

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23. Stevenson J.D., "Structural Damping Values as a Function of Dynamic Response Stress and Deformation Levels," Nuclear Engineering and Design, Vol. 60 No. 2, September 1980.

24. Stevenson, J.D. "Nuclear Standards Applicable to the Civil-Structural Design of Nuclear Power Plants," Proceedings of Speciality Conference on Experience with the Implementation of Construction Practices, Codes Standards, and Regulations in Construction of Power Generating Facilities, Pennsylvania State University, September 1981.

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30. Stevenson, J.D., "Seismic Design of Nuclear Plant Facilities at Low Seismicity Sites," Lawrence Livermore Laboratories Sponsored Conference on Seismic Design of Industrial Facilities, San Francisco, May 1983.

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32. Stevenson, J.D., "Designing for Extreme Loads - The Impact on Cost and Schedule", Nuclear Engineering International, July 1984

33. Stevenson, J.D., "A Review of Procedures Available to Seismically Requalify Operating Nuclear Plant Structures, Equipment and Distribution Systems", Paper K 936 To be Presented at 8th SMIRT, August 1985

34 Stevenson, J.D. "A Summary of Snubber Failure Experience in Nuclear Power Plant Facilities", Paper F1 935, To be Presented at 8th SMIRT, August, 1985

35. Stevenson, J.D., "Rational Seismic Design of Nuclear Power Plant Piping at Low and Moderate Seismicity Sites", Paper K937, To Be Presented at 8th SMIRT, August 1985

APPENDIX D

PERRY SPECIFIC
RESPONSE SPECTRA PLOTS

FLOOR RESPONSE SPECTRA
DESIGN VERSUS RECORDED

TABLE OF CONTENTS

Instrument Number	Location	Direction	OBE/SSE	Damping Percentage	Figure
D51-R170	Inside Drywell Reactor Building Platform—630'	N-S	SSE	2	D-7
D51-R170	Inside Drywell Reactor Building Platform—630'	E-W	SSE	2	D-8
D51-R170	Inside Drywell Reactor Building Platform—630'	VERT	SSE	2	D-9
D51-N111	Reactor Building Containment Vessel—686'	N-S	SSE	2	D-10
D51-N111	Reactor Building Containment Vessel—686'	E-W	SSE	2	D-11
D51-N111	Reactor Building Containment Vessel—686'	VERT	SSE	2	D-12

PNPP UNIT NO.1
AUXILIARY BUILDING
RESPONSE SPECTRA (SSE)
N/S DIRECTION
ELEVATION 568'-4"

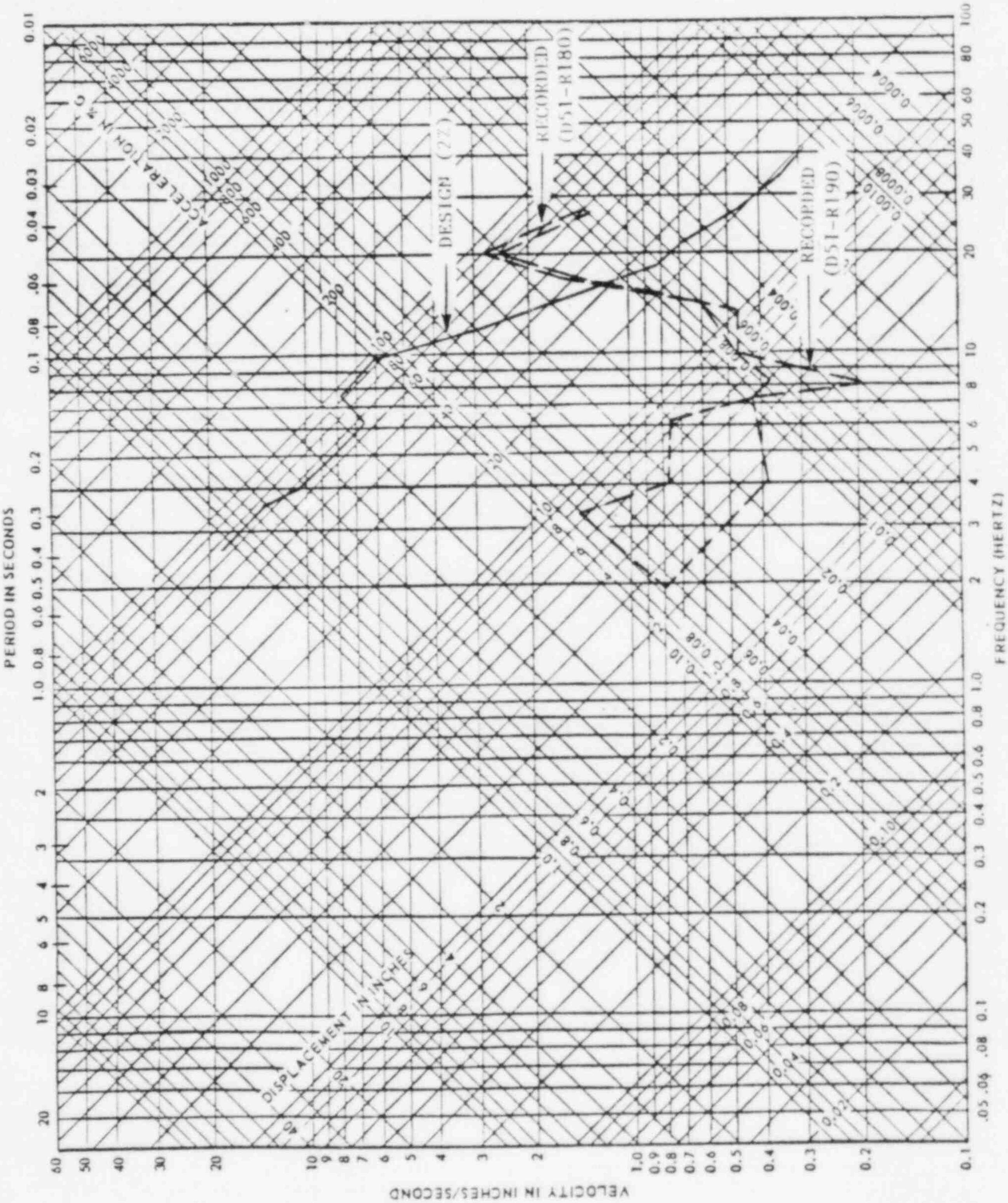


FIGURE D-1

PNPP UNIT NO.1
AUXILIARY BUILDING
RESPONSE SPECTRA (SSE)
E/W DIRECTION
ELEVATION 568'-4"

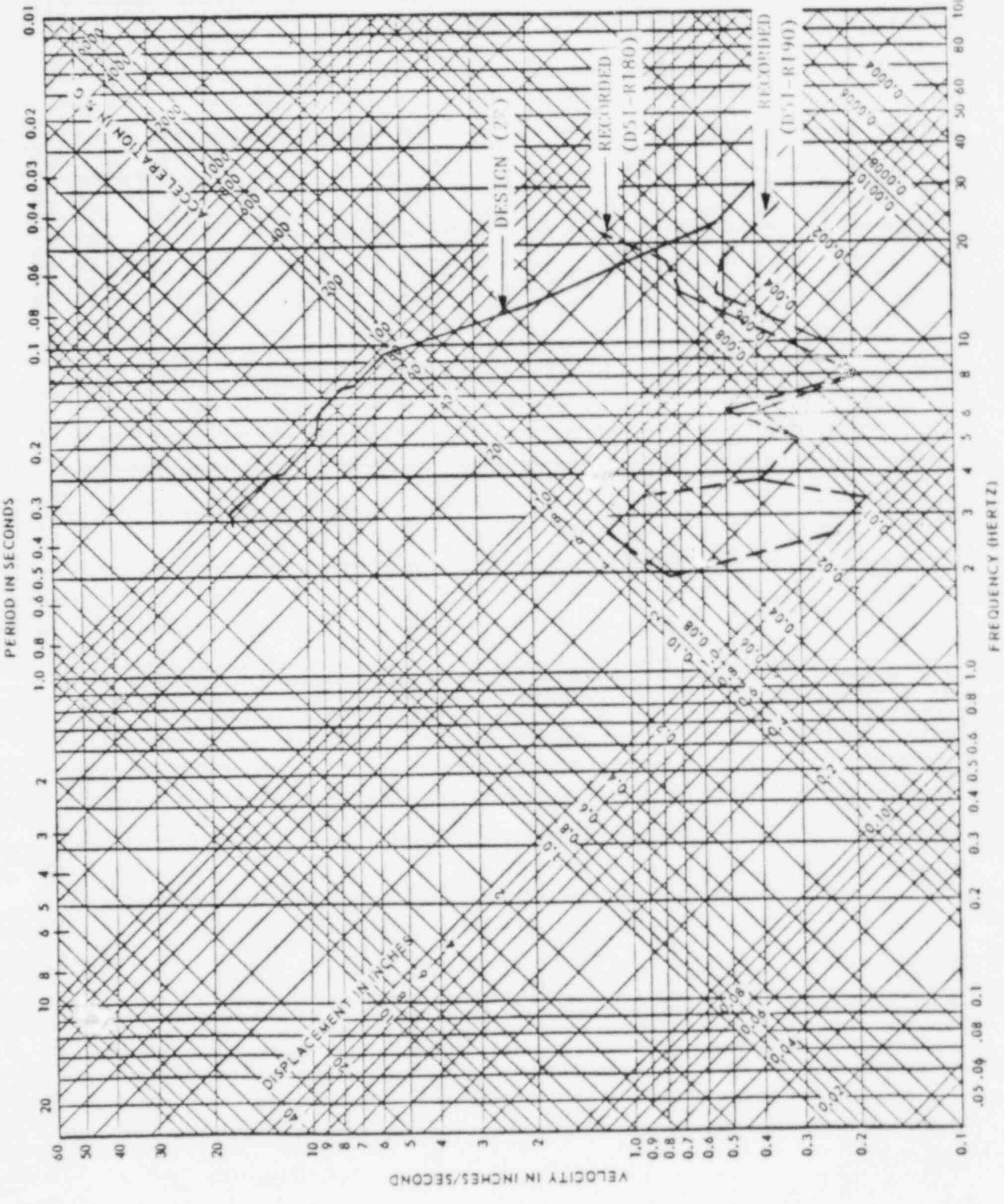
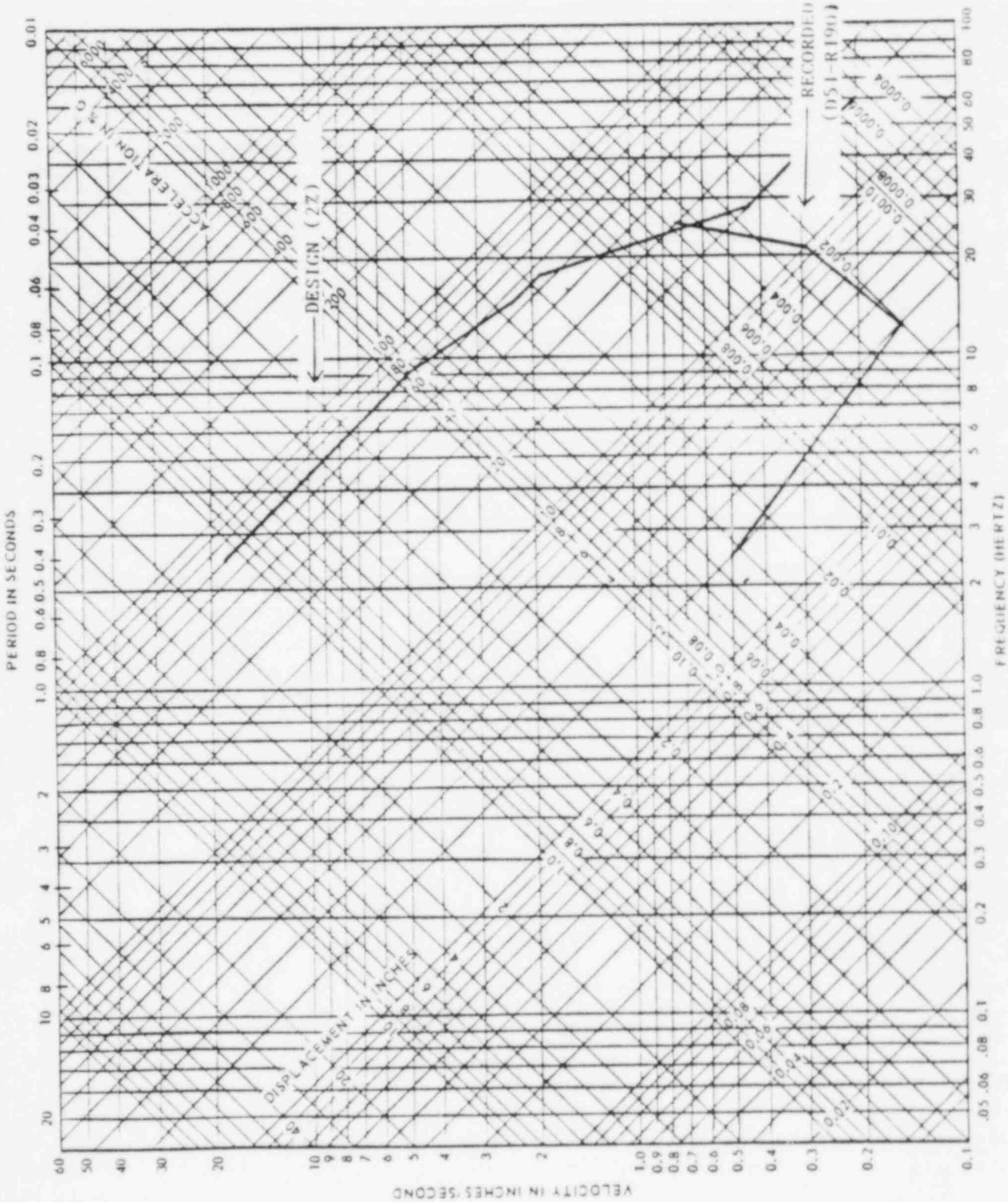


FIGURE D-2

PNPP UNIT NO. 1
AUXILIARY BUILDING
RESPONSE SPECTRA (SSE)
VERTICAL
ELEVATION 568'-4"



NOTE: DSI-R180 (Vertical) Out of

FIGURE D-3

EARTHQUAKE JANUARY 31, 1986

IIA6001 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-1L

DAMPING VALUES ARE 2 PERCENT OF CRITICAL

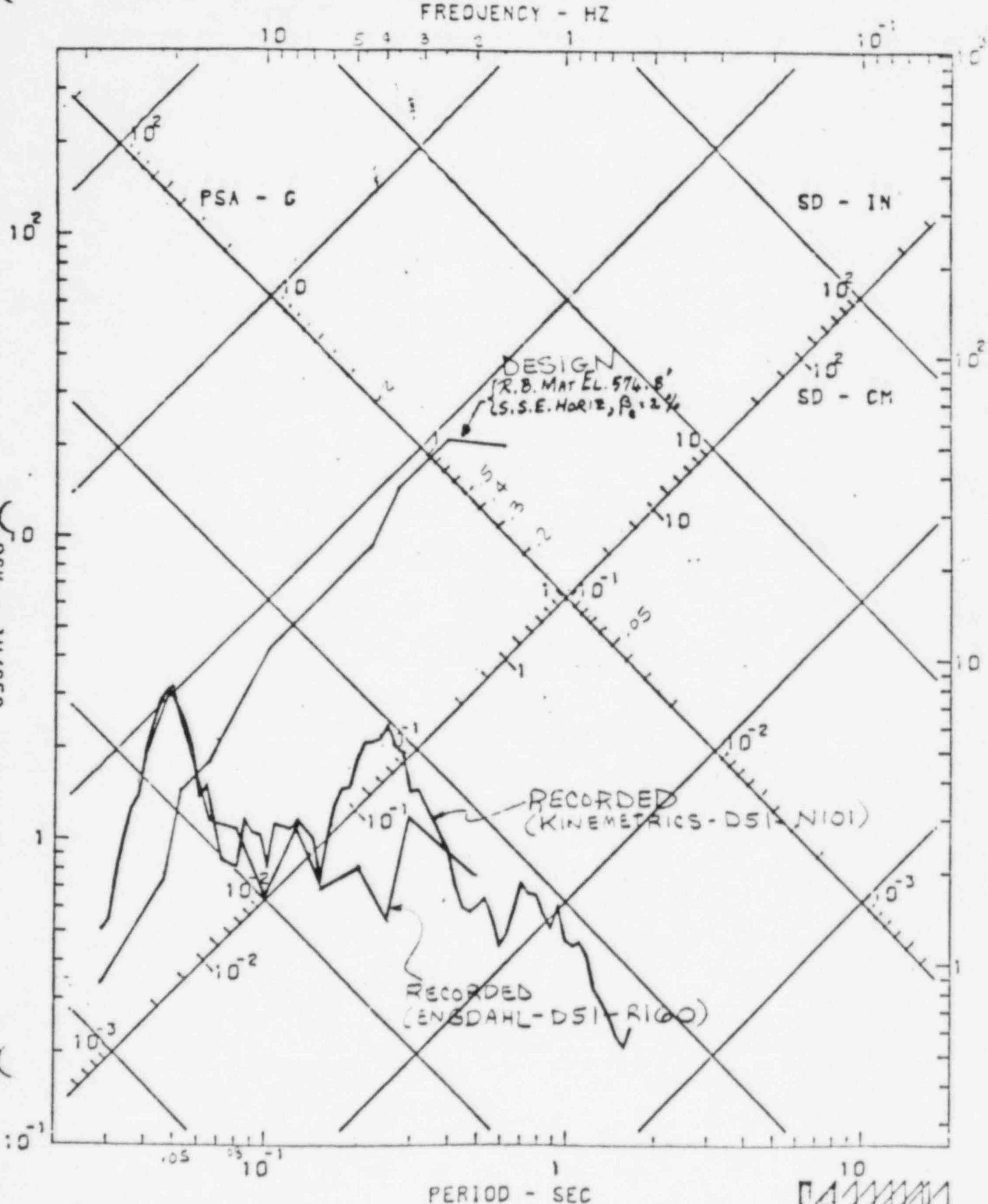


FIGURE D-4

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8001 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-17

DAMPING VALUES ARE 2 PERCENT OF CRITICAL

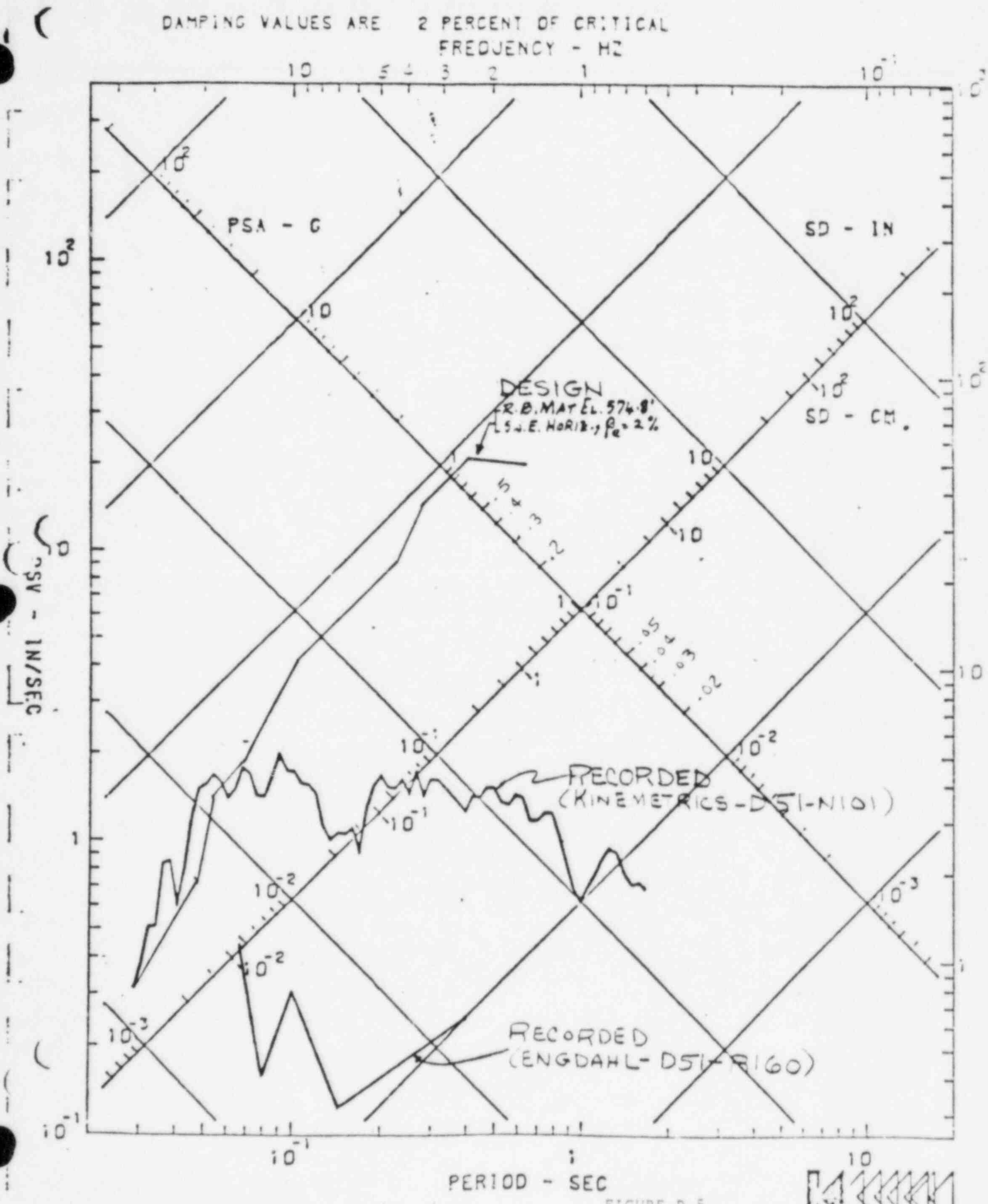


FIGURE D-5

DAMPING VALUES ARE 2 PERCENT OF CRITICAL

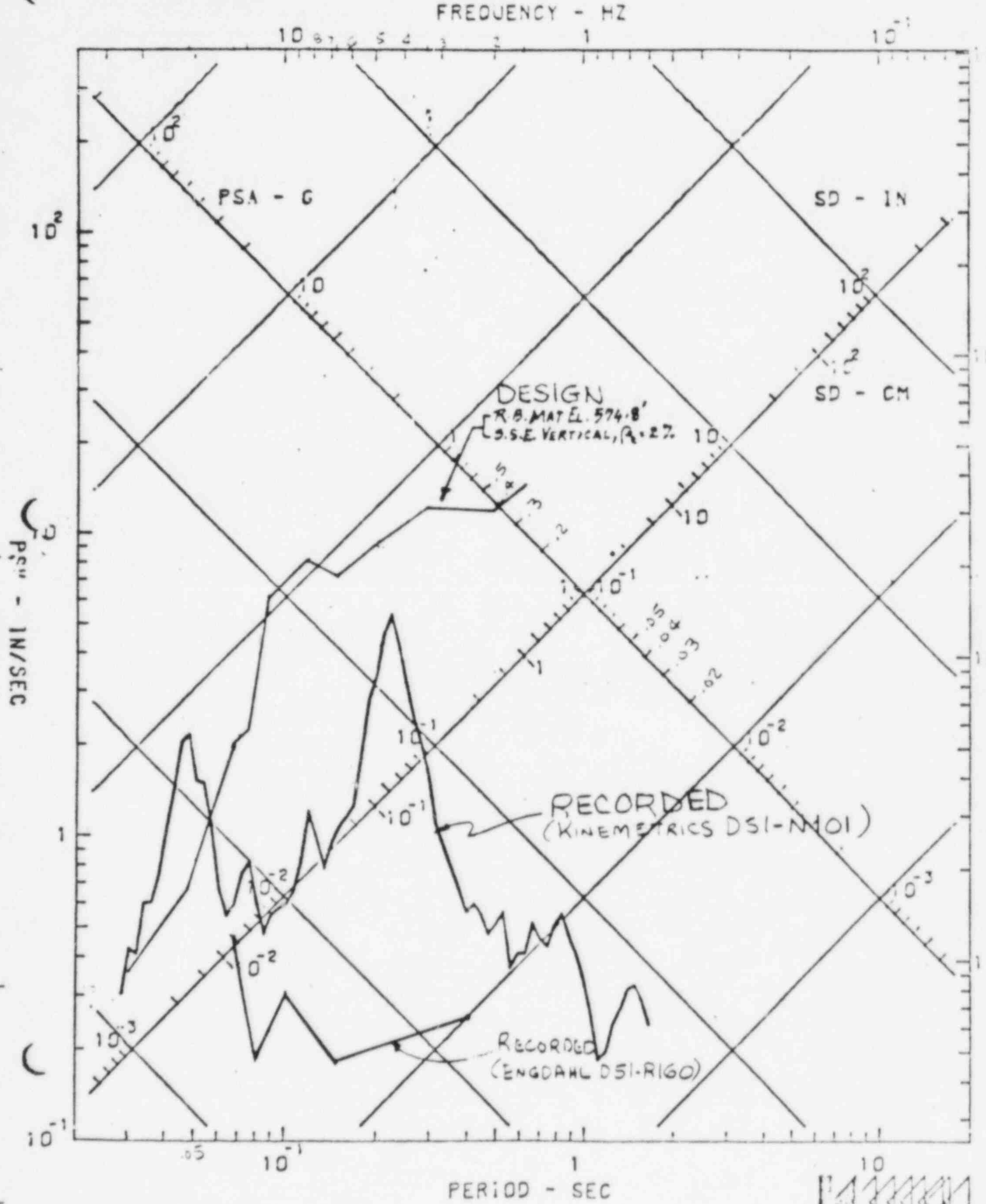


FIGURE D-6

PNPP UNIT NO. 1
REACTOR BUILDING
RESPONSE SPECTRA (SSE)
NORTH-SOUTH
EL [illegible] PLATFORM

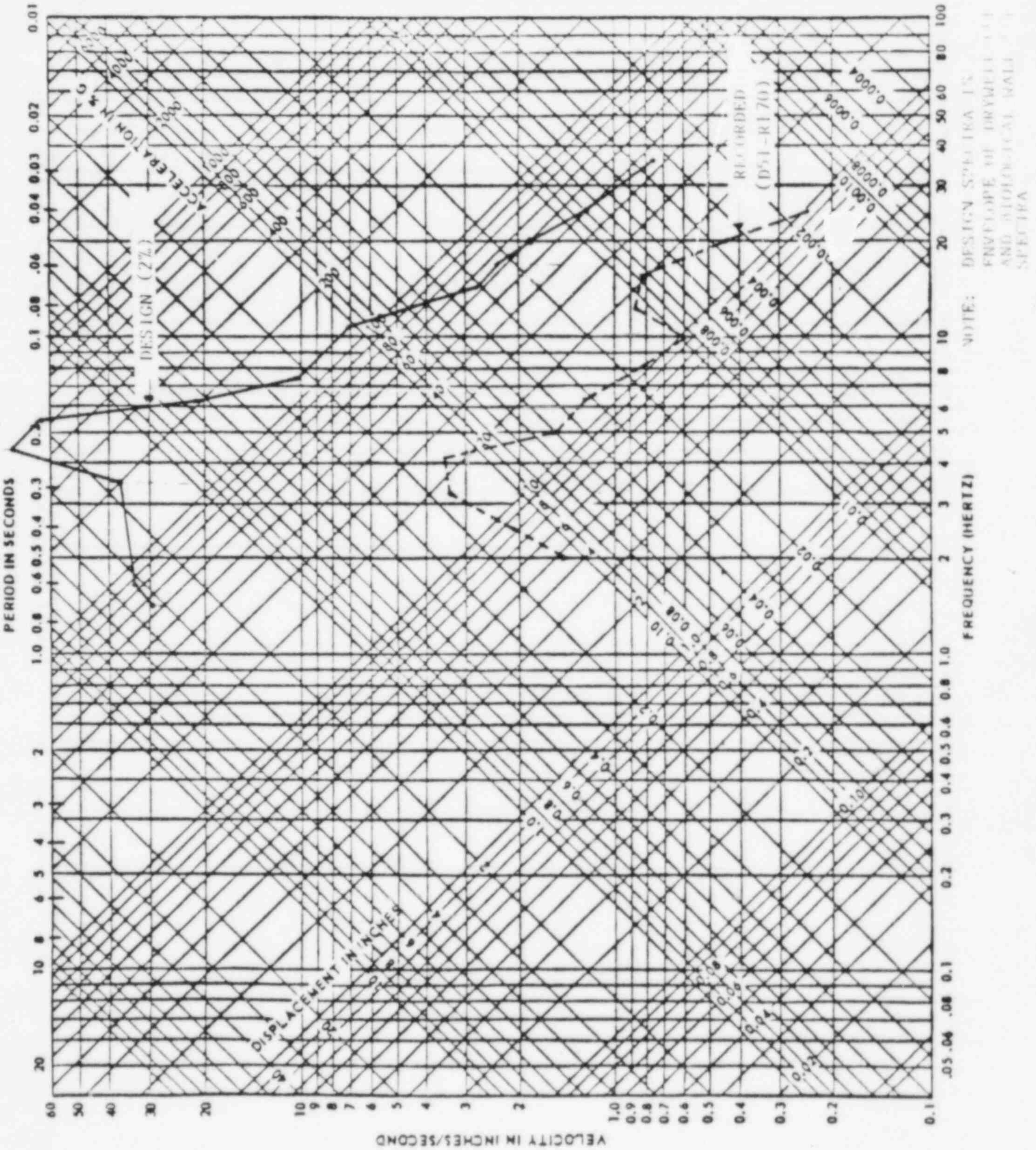


FIGURE D-7

PNPP UNIT NO. 1
REACTOR BUILDING
RESPONSE SPECTRA (SSE)
EAST-WEST
EL 631' PLATFORM

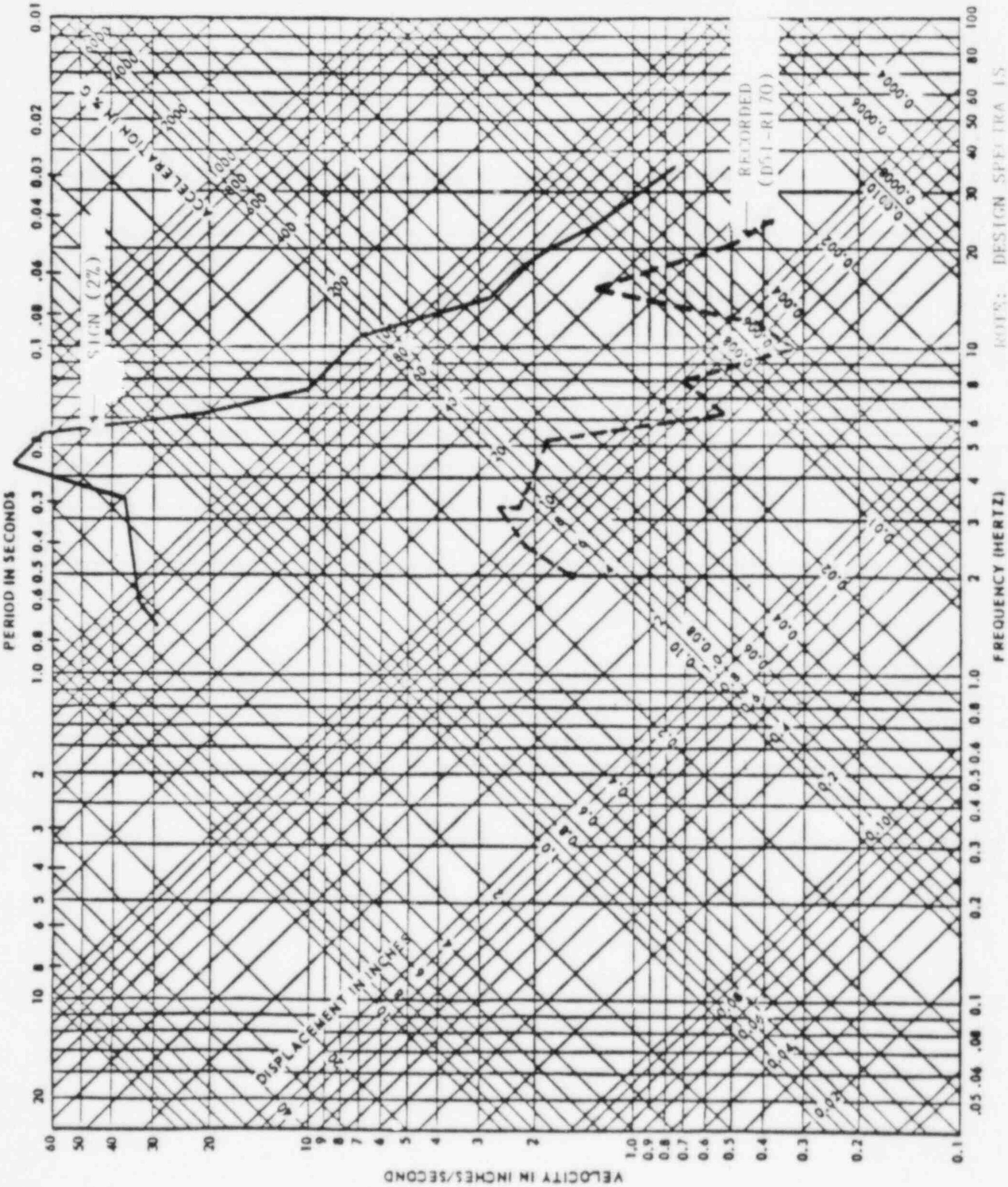


FIGURE D-8

PNPP UNIT NO. 1
REACTOR BUILDING
RESPONSE SPECTRA (SSE)
VERTICAL
EL 631' PLATFORM

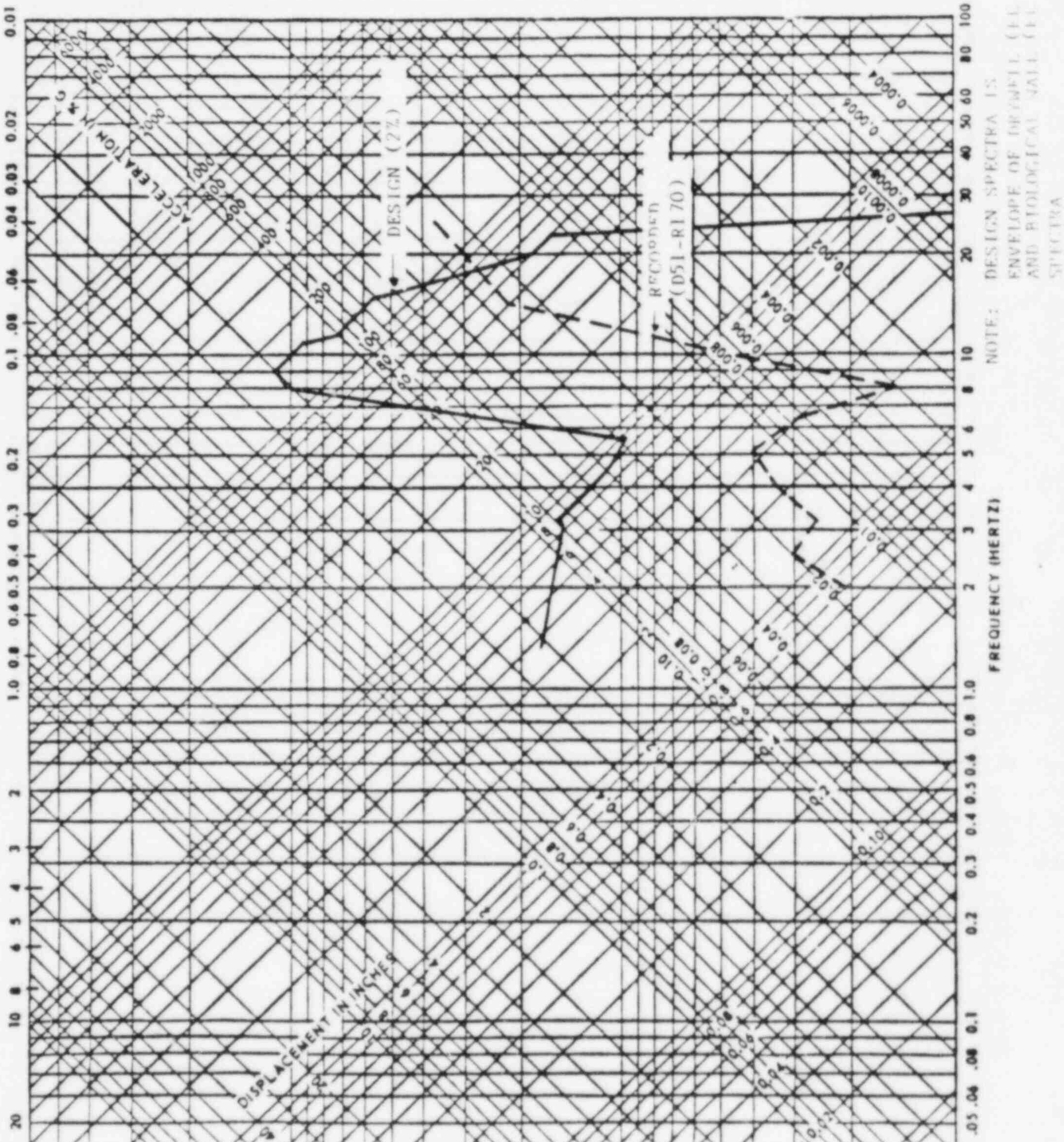


FIGURE D-9

ML 5.0 EARTHQUAKE JANUARY 31, 1986

IIA8002 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-2

DAMPING VALUES ARE 2 PERCENT OF CRITICAL

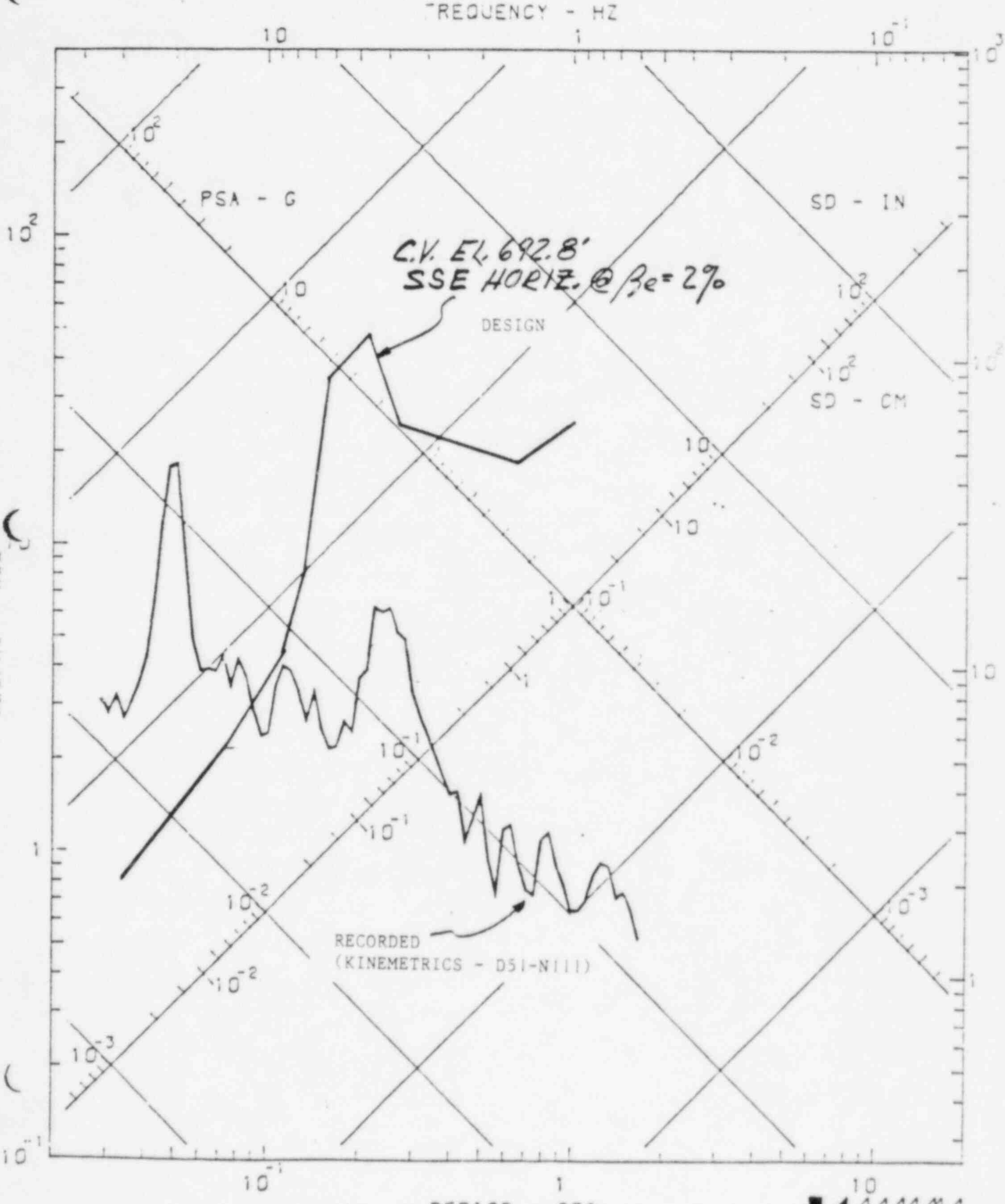


FIGURE D-10

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11AE002 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-21

DAMPING VALUES ARE 2 PERCENT OF CRITICAL

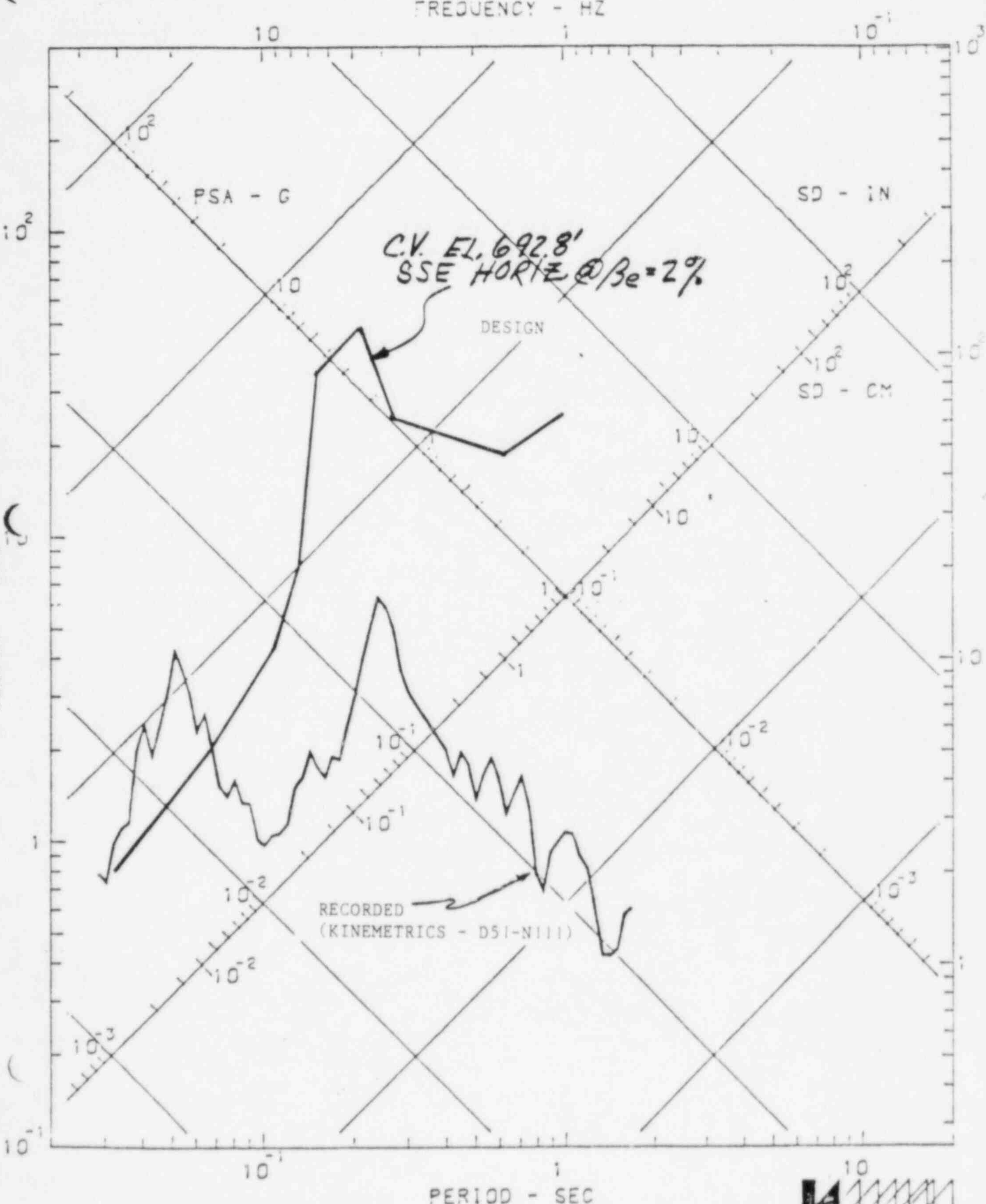


FIGURE D-11

ML 5.0 EARTHQUAKE JANUARY 31, 1986

IIA6002 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 165-2V

DAMPING VALUES ARE 2 PERCENT OF CRITICAL

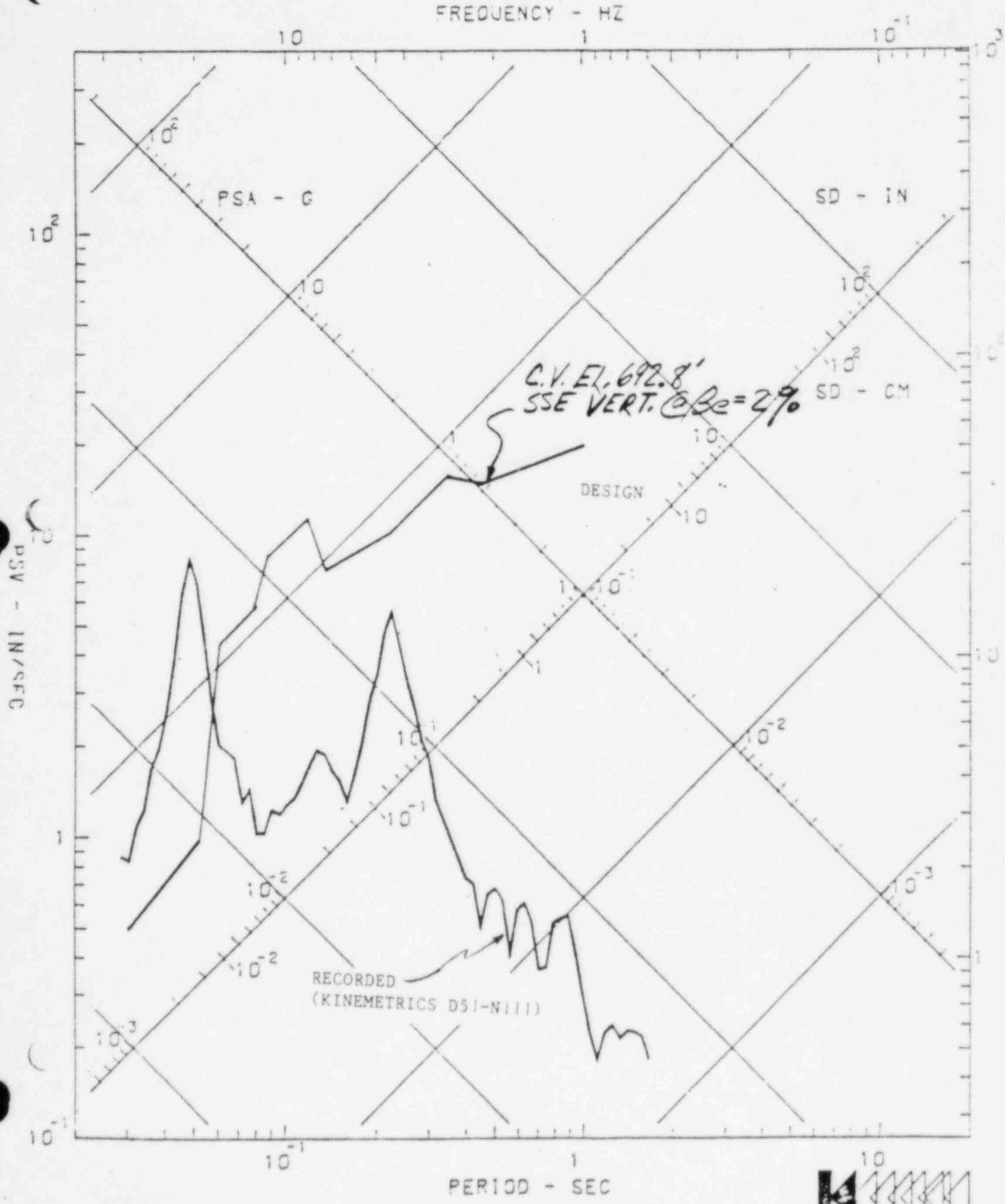


FIGURE D-12

APPENDIX E

RESULTS OF SPECIFIC INSPECTIONS

EVALUATION OF WALKDOWN ITEMS
PLANT SETTLEMENT READINGS
SEISMIC CLEARANCE WALKDOWN
COOLING TOWER WALKDOWN
REVIEW OF ENERGIZED CIRCUITS

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

G-2
REV. 1-82

MEMORANDUM

I no longer wish to receive this material.

TO F. R. Stead ROOM E270 FROM K. R. Pech DATE 11-Feb-86
PHONE 5246 ROOM W220
SUBJECT Evaluation of Walkdown Items

As a result of the plant walkdowns conducted the evening of 1/31/86, Perry Plant Technical Department prepared a list of the observations of the inspection teams. These observations were given the title Earthquake Inspection Team Items or EITI's. The list of EITI's was then forwarded to Engineering for determination of whether the item was a result of the earthquake and whether or not the item needed to be repaired. The assessment of the need for repair and the documentation of that decision whether on a Non-Conformance Report or a Work Request was done in accordance with POP-1501.

The evaluation of all 473 items was completed this afternoon, and the final summary of determinations is presented in the attached table. Each item was placed in one of three categories with respect to its relationship to the earthquake.

1. Caused by the earthquake

2. Indeterminate

3. Not caused by the earthquake

As shown in the summary, 375 items were determined not to be caused by the earthquake, 96 to be indeterminate, and 2 to be caused by the earthquake. With respect to the latter two items, one was the trip of the main transformer, noted in the walkdown of electrical bus L10. The second was a non-safety heater exchanger drain valve that was found dripping water during the walkdown and was reported to be closed and not dripping prior to the earthquake.

In addition each item was categorized as to its final disposition using the procedures contained in POP-1501. Through this process, 330 items were determined to require no repair, 119 to be repaired via a Work Request and 24 items were determined to require dispositioning via a Non-Conformance Report. Of the 24 NR's, 20 are anticipated to be use-as-is and the remainder constitute cosmetic repairs to concrete and drywall walls.

By copy of this memo, the Engineering evaluation of the EITI's is being issued to the Perry Plant Technical Department for preparation of approiate documentation and inclusion in their Condition Report.

KRP:jg

EITI LIST EVALUATION SUMMARY

13:15 2/11/86

DISCIPLINE	TOTAL	C	I	N	NR	WR	NA
ELECTRICAL	35	1	22	12	0	25	10
I & C	18	0	10	8	0	15	3
MECHANICAL	83	1	29	53	2	53	28
PIPING	23	0	18	5	2	10	11
STRUCTURAL	314	0	17	297	20	16	278
TOTAL	473	2	96	375	24	119	330

C = Caused by Earthquake
I = Indeterminate
N = Not Caused by Earthquake
N/A = No Action Required

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

G-2
REV. 1-82

MEMORANDUM

☐ I no longer wish to receive this material.

TO K. Pech ROOM W210 FROM M.R. Kritzer DATE 2-6-86
PHONE 6460 ROOM TQ6
SUBJECT PLANT SETTLEMENT READINGS

M.R. Kritzer

Per our discussion, plant settlement readings were taken on February 5, 1986 (Attached). No significant difference in the building elevation before and after the seismic event was observed. The maximum change occurred in the Reactor Building #1. This change was only a minus (-)0.006 of a foot or 1/16 of an inch. The maximum growth was +0.003 of a foot or 1/32 of an inch, which occurred in the Radwaste Building.

A review of settlement readings taken last February 15, 1985, revealed that the Reactor Building #1 was at the same elevation as it is today.

The minute changes in plant elevation can be expected due to structural growth as a result of weather.

cc: E. Riley
J. Eppich
C. Angstadt
T. Keaveney
S. Dodeja
302.MRK

.RRETT & ASSOC. REG. ENGINEERS & SURVEYORS

PERRY NUCLEAR POWER PLANT
PERRY, OHIO

SKETCH OF: PLANT SETTLEMENTS

DATE: (M) | SCALE: | FIELD BOOK:

DISC # \ DATE	1/16/86	2/5/86				
1-F	625.406	625.402				
2-D	624.339	624.333				
3-D	624.363	624.359				
4-F	626.391	626.386				
5-D	624.540	624.538				
6-D	622.105	622.108				
7-C	621.300	621.298				
SP-8	624.458	624.459				
SP-9	624.924	624.921				
SP-10	624.450	624.453				
SP-11	624.491	624.487				
Initial						

DISC # \ DATE						
1-F						
2-D						
3-D						
4-F						
5-D						
6-D						
7-C						
SP-8						
SP-9						
SP-10						
SP-11						

GARRETT & ASSOC. REG. ENGINEERS & SURVEYORS

PERRY NUCLEAR POWER PLANT
PERRY, OHIO

SKETCH OF: PLANT SETTLEMENTS

DATE: (M)

SCALE:

FIELD BOOK:

PT \ DATE	AUG 20, 1985	SEPT. 25. 1985	OCT. 18, 1985	NOV. 14, 1985	DEC. 17, 1985	JAN. 16 86	
1.F	625.402	625.410	625.411	625.413	625.404	625.406	
2.D	624.341	624.342	624.340	624.345	624.338	624.339	
3.D	624.358	624.361	624.365	624.366	624.362	624.363	
4.F	626.388	626.390	.390	626.387	626.388	626.391	
5.D	624.538	624.544	24.544	624.548	624.540	624.540	
6.D	622.106	622.106	622.104	622.103	622.106	622.105	
7.C	621.303	621.301	621.304	621.300	621.292	621.300	

PT \ DATE							
1.F							
2.D							
3.D							
4.F							
5.D							
6.D							
7.C							

PT \ DATE							
1.F							
2.D							
3.D							
4.F							
5.D							
6.D							
7.C							

GARRETT & ASSOC. REG. ENGINEERS & SURVEYORS

PERRY NUCLEAR POWER PLANT PERRY, OHIO

SKETCH OF: PLANT SETTLEMENTS

DATE: (M)

SCALE:

FIELD BOOK:

PT.	SEPT 27, 83	OCT 27, 1983	NOV. 22, 83	DEC 21 83	JAN. 5, 1984	FEB 10, 84	MAR. 24, 84
1-F	625.396	625.395	625.402	625.407	625.405	625.406	625.401
2-D	624.332	624.329	624.335	624.338	624.334	BLOCKED	624.331
3-D	624.372	624.365	624.365	624.363	624.361	624.362	624.361
4-F	626.389	626.384	626.388	626.388	626.385	626.391	626.381
5-D	624.543	624.547	624.547	624.549	624.550	624.546	624.542
6-D	622.101	622.098	622.096	622.101	622.093	622.101	622.099
7-C	621.295	621.295	621.294	621.303	621.295	621.290	621.281

PT.	APRIL 28, 84	MAY 29, 84	JUNE 20, 1984	JULY 10, 84	AUG. 21, 84	SEPT 14, 84	OCT 10, 84	NOV. 23, 84
1-F	625.397	625.401	625.400	625.402	625.405	625.403	625.406	625.402
2-D	624.335	624.334	624.337	624.336	624.338	624.341	624.339	624.333
3-D	624.365	624.365	624.369	624.368	624.369	624.370	624.366	624.363
4-F	626.385	626.383	626.394	626.390	626.389	626.393	626.391	626.391
5-D	624.537	624.544	624.545	624.546	624.552	624.553	624.550	624.535
6-D	622.102	622.102	622.105	622.108	622.106	622.106	622.103	622.094
7-C	621.287	621.291	621.300	621.298	621.298	621.301	621.300	621.297

PT.	[illegible]	1/10/85	2/15/85	3/22/85	4/12/85	5/15/85	6/25/85	7/9/85
1-F	[illegible].405	625.403	625.406	625.401	625.403	625.396	625.398	625.398
2-D	624.335	624.334	624.333	624.341	624.343	624.336	624.333	624.330
3-D	624.36[illegible]	624.358	624.361	624.359	624.360	624.360	624.357	624.364
4-F	626.389	626.384	626.389	626.380	626.382	626.384	626.381	626.384
5-D	[illegible]46	624.542	624.544	624.547	624.543	624.542	624.546	624.543
6-D	622.096	622.094	622.105	622.103	622.106	622.100	622.101	622.108
7-C	[illegible]	621.296	621.292	621.297	621.297	621.297	621.306	621.299

GARRETT & ASSOC. REG. ENGINEERS SURVEYORS — PERRY NUCLEAR POWER PLANT PERRY, OHIO.

SKETCH OF

FIELD BOOK | SCALE | DATE

PERRY POWER PLANT
FOUNDATION SETTLEMENT CHART

MARK NO	DEC. 18, 78	JAN 22	FEB 22	MARCH 20	MAY 6, 1979	MAY 21 79	JULY 10 79	SEPT 10, 79	OCT 5, 79	OC[illegible]	
1-E	625.286	625.274	625.270	625.270	625.270	—	—				
1-F	NEW DISC →		→	625.393	625.397	625.395	625.406	625.399	625.420	625.418	
2-D	624.351	624.336	624.331	624.332	624.336	624.337	624.343	624.342	624.352	624.346	
3-D	624.423	624.403	624.388	624.394	624.401	624.347	624.398	624.395	624.403	624.396	
4-C	596.003	COVERED		596.007	COVERED	COVERED	COVERED				
4-D	REVISED READING →	604.626	604.626	604.626	604.620	COVERED	"				
4-B	COVERED —			596.947	COVERED	COVERED	"				
5-C	603.615	603.607	COVERED	603.605	TEMPORARILY COVERED	COVERED	"				
5-D				NEW DISC →	624.561	624.551	624.548	624.548	624.551	624.554	
6-C	619.025	COVERE	D —								
6-D	NEW DISC. →		→	622.086	622.088	622.096	622.085	622.091	NEW MARK 622.112	622.109	
7-A	COVERED	—	→		COVERED	COVERED —					
7-B	NEW MARK →		→	616.733	616.735	616.732	616.737	616.729	616.725	(1) 621.300	
4-E	NEW →				620.464	620.465	620.462				
4-F							626.412	626.409	NEW 626.407	626.405	

MARK NO	OCT 31, 1979	NOV 19, 1979	JAN 21, 1980	MAR 27 1980	APR [illegible] 1980	JUNE 23, 80	AUG 12, 1980	SEPT 18, 1980	OCT 22, 1980	DEC 15, 1980	JAN 21, 1981
1-F	625.418	625.413	625.402	625.414	625.415	625.419	625.407	625.392	625.390	625.400	625.385
2-D	624.346	624.349	624.337	624.355	624.350	624.354	624.344	624.336	624.333	624.336	624.326
3-D	624.396	624.391	624.382	624.386	624.388	624.400	624.385	624.372	Blocked	624.370	624.362
4-F	626.405	626.409	626.370	626.403	626.407	626.409	626.397	626.387	626.390	626.414	626.379
5-D	624.554	624.551	624.540	624.547	624.549	624.549	624.546	624.554	624.552	624.548	624.541
6-D	622.104	622.104	622.099	622.098	622.100	622.101	622.099	622.108	622.095	622.104	622.091
7-C	621.300	621.300	621.303	621.288 (621.276) RECHECKED	621.308	621.311	621.310 WALL	621.314 WALL	621.369 WALL	621.303	621.301 0.1 [illegible]

GARRETT & ASSOC. REG. ENGINEERS SURVEYORS

PERRY NUCLEAR POWER PLANT PERRY, OHIO.

SKETCH OF [illegible]

FIELD BOOK

SCALE

DATE

POINT / DATE	FEB 9, 1981	MARCH 22, 1981	APRIL 15 1981	MAY 22 1981	July 2, 1981	July 26 1981	AUG 11, 1981	SEPT 2 1981	OCT 21, 1981	NOV 27, [illegible]	
1-F	625.395	625.395	625.394	625.395	625.392	625.395	625.387	625.396	625.396	625.391	
2-D	624.332	624.325	624.325	624.327	624.332	624.333	624.335	624.337	624.330	624.335	
3-D	BLOCKED	624.366	624.365	624.366	624.370	624.371	624.369	624.375	624.367	624.369	
4-F	626.390	626.390	626.384	626.388	626.390	626.387	626.391	626.389	626.387	626.383	
5-D	624.547	624.521	624.543	624.541	624.538	624.538	624.540	624.540	624.547	624.546	
6-D	622.096	622.096	622.100	622.099	622.096	622.099	622.095	622.103	622.094	622.093	
7-C	621.306	621.312	621.310	621.311	621.314	621.314	621.315	621.318	621.289*	621.290	
	WALL	WALL	WALL	[illegible]	WALL	[illegible]	WALL	WALL	MON		

POINT / DATE	JAN 5 1982	JAN 15 1982	FEB 19, 82	MAR. 31, 1982	APRIL 26, 82	MAY 17 1982	JUNE 22 1982	July 28, 1982	8-25-82	9-82	10-21-82
1-F	625.393	625.390	625.390	625.394	625.392	625.394	625.391	625.390	625.394	625.398	625.398
2-D	624.329	624.325	624.322	Blocked	624.328	624.333	624.324	624.331	624.330	624.335	624.335
3-D	624.365	624.362	624.352	624.361	624.366	624.365	624.358	Blocked	BLOCKED	BLOCKED	BLOCKED
4-F	626.385	626.381	626.368	626.384	626.384	626.383	626.374	626.381	626.383	626.385	626.385
5-D	624.545	624.543	624.540	624.539	624.541	624.538	624.541	624.541	624.544	624.538	624.538
6-D	622.092	622.095	622.096	622.096	622.095	622.096	622.098	622.099	622.098	622.097	622.097
7-C	621.281	621.276	621.276	621.269	621.284	621.289	621.282	621.289	621.290	621.290	621.294

POINT / DATE	NOV 30, 82	DEC. 8, 1982	JAN 25, 1983	FEB 16, 83	MAR 28, 1983	APR 20, 83	MAY 10, 83	MAY 24, 83	JUNE 23, 83	July 26, 1983	AUG. 24 83
1-F	625.395	625.392	625.392	625.399	625.400	625.401	625.~~407~~	625.393	625.399	625.403	625.399
2-D	624.334	624.328	624.331	624.328	624.333	624.330	624.~~337~~	624.321	624.333	624.339	624.332
3-D	BLOCKED	Blocked	BLOCKED	624.367	624.367	624.361	624.~~368~~	624.356	624.363	624.374	624.367
4-F	626.387	626.374	626.379	626.382	626.382	626.380	626.~~399~~	626.380	626.383	626.394	626.388
5-D	624.539	624.5[illegible]	624.537	624.539	624.543	624.536	624.535		624.543	624.541	624.542
6-D	622.097	622.101	622.097	622.097	622.101	622.096	622.098		622.100	622.104	622.101
7-C	621.288	621.296	621.288	621.294	621.291	621.286	621.288		621.295	621.269	621.295

PERRY NUCLEAR POWER PLANT

FOUNDATION SETTLEMENT CHART CONTINUED BY GARRETT [illegible]

SHEET NO. 5-C

MARK NO.	FEB.16,78	MARCH 13,78	APRIL 7, 1978	MAY 9, 1978	MAY 25,78	JUNE 15 78	JULY 20 78	JULY 31, 78	AUG. 17, 78	AUG 28.78	SEPT 27, 78	[illegible]
1-B	COVERED	COVERED	WILL BE CLEAR NEXT MONTH	NOT CLEAR	595.625	COVERED	NOT CLEAR	COVERED	COVERED			
2-D PER	624.362	NOT ABLE TO READ	624.360	624.355	624.360	624.360	624.348	624.358	624.357		624.360	624.35[illegible]
3-B	599.046	599.041	599.040	599.044	599.052	RAISED 5' TO 604.046	604.047	604.033	604.041			
4-B	596.940	596.938	596.939	596.940	COVERED	COVERED	COVERED	COVERED	COVERED			
5	COVERED	COVERED	COVERED	COVERED	COVERED	COVERED	COVERED	"	"			
6-B	602.188	602.170	602.191	602.195	602.188	602.118	602.200	602.197	602.210			
7	NOT ABLE TO READ	NOT ABLE TO READ	NOT ABLE TO READ	NOT ABLE TO READ	COVERED UNDER [illegible]	COVERED	COVERED	COVERED	COVERED			
5-A		NEW POINT SET ON WALL →	APRIL 1978 591.524	COVERED	COVERED	COVERED	COVERED	"	"			
2-C		600.144			600.146	COVERED	COVERED	"	"			
1-C			NEW	593.517	593.514	593.517	COVERED	"	"			
[illegible]-B				598.805	598.794	COVERED	COVERED	"	"			
1-D					NEW 614.287	COVERED	COVERED	"	"		614.291	614.295
[illegible]-C					623.514	623.507	6[illegible].506	"	"			
4-C					596.018	COVERED	COVERED	595.997	595.999			596.007
7-A	MARK	ON WALL	15' W. OF SE	COR	NEW 563.143	562.122	563.113	COVERED				563.100
3-D					PERMANENT	DISC. SET.	REACTOR	No2 (270°)	→	INITIAL 624.425	624.422	624.411
4-D											NEW →	604.625
1-E											NEW →	625.292
6-C											NEW →	619.028
5-C											NEW →	603.600 (SEE NOTE A)

[illegible]ACKED OR CANNOT READ

[illegible] REVISED [illegible] IN NOV.

PERRY NUCLEAR POWER PLANT

FOUNDATION SETTLEMENT CHART CONTINUED BY GARRETT & ASSOCIATES INCORPORATED

SHEET NO 2

NO	JULY 8, 77	July 15, 77	JULY 21 77	July 29	8/5/77	AUG, 11 77	AUG 19, 77	Aug 29, [illegible]	SEPT, 14, 77	SEPT 26, 77	OCT 12 77	[illegible]
1A	582.090	582.093	582 092	582 095	582.095	582.087	582.092	582.091	COVERED	—	—	[illegible]
1B	595.619	595617	595.622	595.625	595.627	595.615	595.624	595.620	595 620	595.619	595.622	595.[illegible]
2	572.484	572.983	COVERED			572.779	572.984	COVERED	—	—	—	—
2A	[illegible]	578.323	578.335	COVERED		—	COVERED	COVERED	—	—	—	—
2B	589.980	[illegible]	589 981	589.981	589.984	589 978		589.982	589.981	COVERED	—	—
3	572.985	572.982	572.984	572.987	[illegible]	572.986	572.770	572.989	577.987	COVERED	—	—
	585.471	585.493	585.474	585.493	585.493	585.493	585.493	585.493	585.495	585.494	585.476	585.498
	572.906	572.904	572.906	572.905	572.904	572.904	572.906	572.899	572.903	572.898	572.891	572.891
A	587.475	587.476	587.478	587.481	587.496	587.427	587.478	587.475	587.474	587.477	587.474	587.469
3A	—					590.267	590.267	590.265	590.266	590.262	590.270	590.265
2-C	new mark —								600.146	600.144	600.144	—
2-D =	BRASS DISC —							SEPT 12.77 624.359	624.361	624.360	624.365	624.358

	NOV. 17, 77	DEC 2, 77	Dec 19, 77	DEC 29, 77	JAN 23, 78	Feb 16, 78
• 1-B	COVERED BY [illegible]	Covered	Covered [illegible]	covered	COVERED	—
• 2-D	624.358	624.358	624361	624.362	624.358	624.362
3-A	COVERED	—	—	—	—	—
• 4-A	COVERED BY [illegible]	—		—	—	—
• 5	COVERED	Covered	[illegible]	Covered	COVERED	—
• 6-A	587.470		587.4?0	587.460	VOID ~~587.416~~	587.446
• 4-B	596.943	596942	596.938	596.940	596.940	596.940
• 2-C	600.142	—	600.143	600.155	600 155	—
• 1-B	602.183	602.180	602.200	602.194	VOID ~~602.160~~	602.188
3-B	599.045	599.048	599.042	599.044	599.049	599.046

CONTINUED ON SHEET "5-C"

PERRY NUCLEAR POWER PLANT

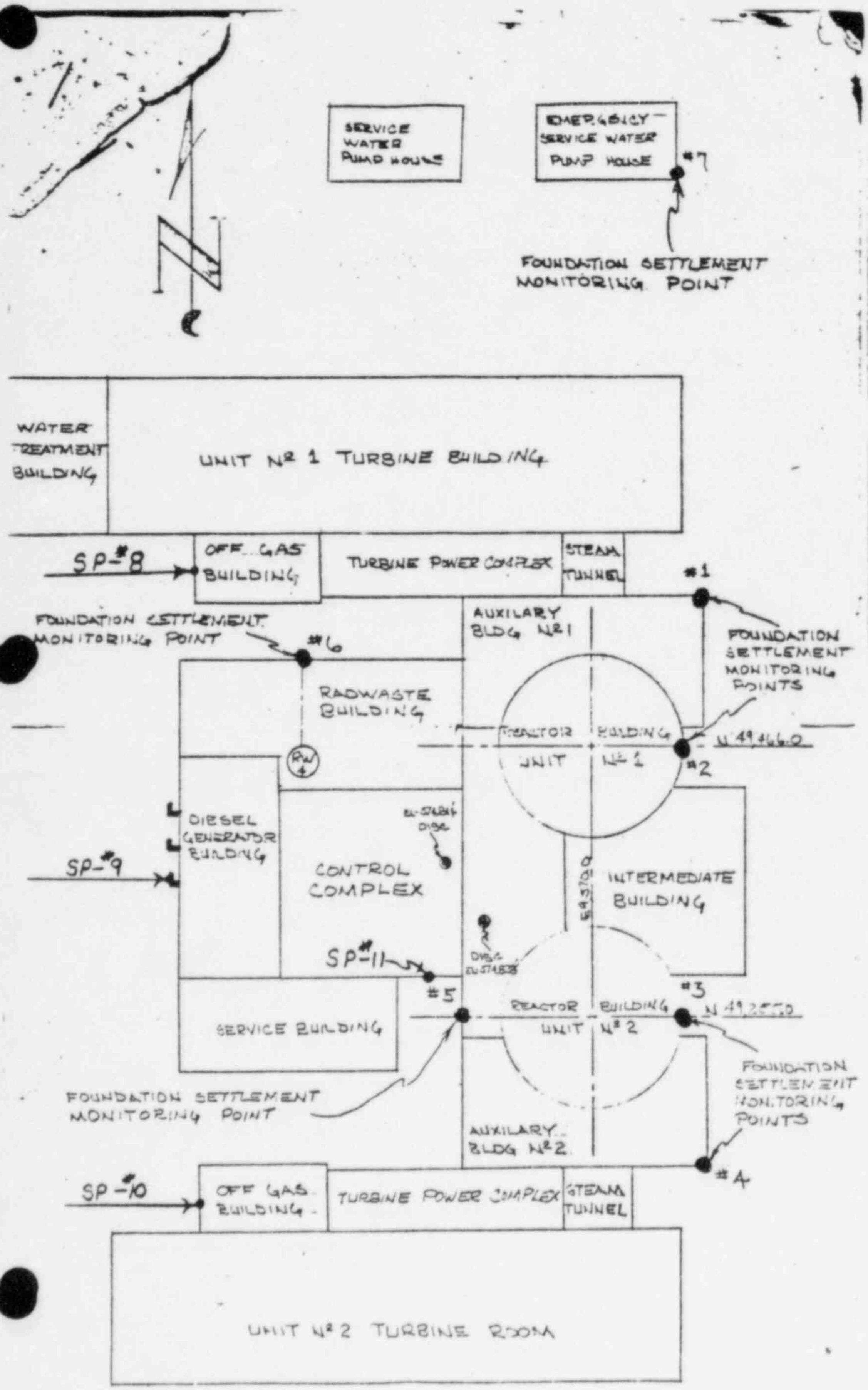
FOUNDATION SETTLEMENT CHART BY GARRETT + ASSOC. INCORPORATED

STARTED JAN 18, 77

NOTES. 1) USE IN CONJUNCTION WITH GILBERT PRINT RDB 121416.
2) ELEVATIONS ARE IN FEET AND DECIMALS.
3) REF. - GARRETT BOOK Nº 28
SHEET Nº 1

MARK Nº	JAN 18, 77	JAN 24, 77	FEB. 1, 77	FEB 2 77	FEB. 15, 77	FEB 21, 77	FEB 28, 77	MARCH 8, 77	MARCH 17, 77	MARCH 25 77	MARCH 30 77	APRIL 6, 77
1	568.334	568.346	568.339	568.333	568.334	568.334	568.335	568.333	568.333	568.318	568.333	568.333
2	572.992	572.993	572.990	572.991	572.990	572.988	572.987	572.988	572.988	572.987	572.986	572.982
3	572.998	572.996	572.996	572.996	572.995	572.994	572.992	572.993	572.989	572.986	572.984	572.987
4								SET MARCH 14, 77	568.295	568.291	568.287	568.287
5												
6					SET FEB 18, 77	573.934	573.931	573.936	573.938	573.931	573.930	573.937

MARK Nº	APRIL 14, 77	APRIL 20, 77	APRIL 27, 77	[illegible] 6, 77	MAY 12, 77	MAY 19, 77	MAY 27, 77	JUNE [illegible] 77	JUNE 10, 77	JUNE 15, 77	JUNE 22 77	JULY 1 77
1	568.333	568.330	568.331	568.330	568.334	1-B = 582.0[illegible] 568.335	1-A: 582.097	582.0[illegible]	582.095	1A 582.0[illegible] 1B 575.618	[illegible] 585.619	1-A 582.0[illegible] 1-B 575.624
2	572.986	572.986	572.986	572.986	2-A: 578.316 572.986	2-A: 578.[illegible] 572.984	2-A: 578.335	578.337	578.380	2 [illegible] 2-578.384 2C 587.783	2A 578.[illegible] 2B 585.983	2A 578.336 2B 587.782
3	572.987	572.984	572.987	572.988	572.990	572.986	572.985	572.989	572.988	572.990	572.987	572.984
4	568.287	568.284	568.285	568.289	568.288	568.287	568.291	568.290	568.286	4A 586.493	585.493	4A 586.493
5		SET MAY 27. 77	572.905	572.905	572.906	572.905	572.904	572.899	572.896	572.905	572.905	572.904
6	573.938	573.936	573.939	587.492 573.935	573.935	573.932	587.499 573.932	6-A: 587.4[illegible] 6-COVERED	587.478	587.495	587.493	587.469


SERVICE WATER PUMP HOUSE
EMERGENCY SERVICE WATER PUMP HOUSE
#7
FOUNDATION SETTLEMENT MONITORING POINT
WATER TREATMENT BUILDING
UNIT Nº 1 TURBINE BUILDING
SP-#8
OFF GAS BUILDING
TURBINE POWER COMPLEX
STEAM TUNNEL
#1
AUXILARY BLDG Nº1
FOUNDATION SETTLEMENT MONITORING POINT
#6
FOUNDATION SETTLEMENT MONITORING POINTS
RADWASTE BUILDING
REACTOR BUILDING
UNIT Nº 1
#2
RW 4
DIESEL GENERATOR BUILDING
SP-#9
CONTROL COMPLEX
INTERMEDIATE BUILDING
SP-#11
#5
#3
REACTOR BUILDING
UNIT Nº 2
SERVICE BUILDING
FOUNDATION SETTLEMENT MONITORING POINT
FOUNDATION SETTLEMENT MONITORING POINTS
AUXILARY BLDG Nº2
#4
SP-#10
OFF GAS BUILDING
TURBINE POWER COMPLEX
STEAM TUNNEL
UNIT Nº 2 TURBINE ROOM

G-2
REV. 1-82

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

MEMORANDUM

☐ I no longer wish to receive this material.

TO K. Pech ROOM W210 FROM M. Kritzer/J. Messenger DATE 2-5-86

PHONE 6460 ROOM TQ6

SUBJECT SCV WALKDOWN AFTER SEISMIC EVENT
Ref. documents NIR's C-23642,
C-23661 thru and including C-23666
also C-23677

Due to the seismic event which occurred on 1/31/86, the Seismic Clearance Group has conducted a walkdown of all open "Repair" dispositioned Seismic Clearance Violations.

The scope of the walkdown was to evaluate if any dimensional changes from the original SCV, and to note any structural or system damage in these particular areas that could have been attributed to the earthquake.

The results of the evaluation, as shown on the above referenced inspection reports, indicates that there were no dimensional changes from the document SCV's. There was also no plant damage associated with the unrepaired SCV's.

It should be noted that 7 of the ~~28~~ 29 MRK 2/6/86 SCV's have at minimum, partial work complete.

Any questions, please feel free to contact us.

attachments: 8

cc: E. Riley
S. Dodeja
302.MRK
302. JWM
J. Eppich
C. Angstadt

Construction Quality Section
Nonsafety Inspection Report (NIR)

PNPP No. 5978

N/A	N/A	C-23642
CONTRACTOR	SPEC. NO.	INSPECTION REPORT NUMBER
SCV INSPECTION WALKDOWN AFTER SEISMIC EVENT		~~See Below~~ N/A
SYSTEM/COMPONENT/ACTIVITY		M.P.L. NUMBER
AX-1 599' EL	21	2-3-86
LOCATION	INSP. TYPE	DATE

DUE TO THE SEISMIC EVENT WHICH OCCURED ON 1-31-86 THE FOLLOWING OPEN SCVs HAVE BEEN FIELD VERIFIED TO DETERMINE IF ANY DIMENSIONAL CHANGES HAVE OCCURED.

SCV 6215
SCV 6292
SCV 6550

CORRECTIVE ACTION DOCUMENTATION - (D.R.'s): N/A

MEASURING/INSPECTION TOOLS I.D.: MISC INSPECTION TOOLS

REMARKS NO DIMENSIONAL CHANGES DIRECTLY ATTRIBUTED TO THE SEISMIC EVENT WERE FOUND AND NO DAMAGE ENCOUNTERED.

cc: S. DODEJA - E180

Mike Hagel 2-3-86
INSPECTOR DATE

John W Messenger 2-4-86
REVIEWED - LEAD INSPECTOR DATE

441/T/1/ba

Nonsafety Inspection Report (NIR)

PNPP No. 5978

N/A	N/A	C-23661
CONTRACTOR	SPEC. NO.	INSPECTION REPORT NUMBER
Walkdown of open S.C.U.s after the seismic event of 1-31-86		N/A
SYSTEM/COMPONENT/ACTIVITY		H.P.L. NUMBER
Control Complex, 679'-6" Floor elev.	21	2-3-86
LOCATION	INSP. TYPE	DATE

Due to the seismic event which occured on 1-31-86, the following open S.C.U.s have been field verified to determine if any dimensional changes had occured: SCU # 5978

S.C.U. # 6060 rev. 1

S.C.U. # 6061 rev. 1

NO dimensional changes or damage, directly attributed to the seismic event, were found.

CORRECTIVE ACTION DOCUMENTATION - (D.R.'s): N/A.

MEASURING/INSPECTION TOOLS I.D.: Miscellaneous inspection Tools.

REMARKS N/A

cc: S. Dodeja E-180

F.C. Manno 2-4-86
INSPECTOR DATE

John W. Messenger 2-4-86
REVIEWED - LEAD INSPECTOR DATE

441/T/1/ba

N/A	N/A	C-23662
CONTRACTOR	SPEC. NO.	INSPECTION REPORT NUMBER
Walkdown of open S.C.U.s after the seismic event of 1-31-86		N/A
SYSTEM/COMPONENT/ACTIVITY		M.P.L. NUMBER
Intermediate Bldg., All elevations	21	2-3-86
LOCATION	INSP. TYPE	DATE

Due to the seismic event which occured on 1-31-86, the following open S.C.U.s have been field verified to determine if any dimensional changes had occured:

S.C.U. # 6500 — 620'-6" Floor elev.
S.C.U. # 6536 — 620'-6" Floor elev.
S.C.U. # 6639 — 599' Floor elev.
S.C.U. # 6688 — 599' Floor elev.
S.C.U. # 5968 - 574'-~~6"~~ 10" Floor elev. (2-4-86)

No dimensional changes or damage, directly attributed to the seismic event, were found.

CORRECTIVE ACTION DOCUMENTATION - (D.R.'s): N/A.

MEASURING/INSPECTION TOOLS I.D.: Miscellaneous inspection tools.

REMARKS S.C.U. # 5968 has 3 of the 4 repair supports added.

cc: S. Dodeja E-180

H. Manno 2-4-86
INSPECTOR DATE

John W Messenger 2-4-86
REVIEWED - LEAD INSPECTOR DATE

Nonconformity Inspection Report (NIR)

PNPP No. 5978

N/A	N/A	C-23663
CONTRACTOR	SPEC. NO.	INSPECTION REPORT NUMBER
Walkdown of open S.C.D.s after the seismic event of 1-31-86		N/A
SYSTEM/COMPONENT/ACTIVITY		M.P.L. NUMBER
Auxiliary Bldg # 1, All elevations	21	2-3-86
LOCATION	INSP. TYPE	DATE

Due to the seismic event which occured on 1-31-86, the following open S.C.U.s have been field verified to determine if any dimensional changes had occured: S.C.U.# 6563 - 620'-6" Floor elev.

S.C.U.# 6136
S.C.U.# 6262 } 599' Floor elev.

NO dimensional changes or damage, directly attributed to the seismic event, were found.

CORRECTIVE ACTION DOCUMENTATION - (D.R.'s): N/A.

MEASURING/INSPECTION TOOLS I.D.: Miscellaneous inspection Tools.

REMARKS S.C.U.# 6262 has repair work in-progress.

cc: S. Dodoja E-180

H. Meano	2-4-86
INSPECTOR	DATE
J. Messenger	2-4-86
REVIEWED - LEAD INSPECTOR	DATE

441/T/1/ba

PNPP No. 5978

N/A	N/A	C-23664
CONTRACTOR	SPEC. NO.	INSPECTION REPORT NUMBER
WALKDOWN OF OPEN SCV'S AFTER SEISMIC EVENT OF 1/31/86		N/A
SYSTEM/COMPONENT/ACTIVITY		M.P.L. NUMBER
IB @ FLOOR EL 599' & 682'	21	2-3-86
LOCATION	INSP. TYPE	DATE

DUE TO THE SEISMIC EVENT WHICH OCCURED ON 1-31-86 THE FOLLOWING OPEN SCV'S HAVE BEEN FIELD VERIFIED TO DETERMINE IF ANY DIMENSIONAL CHANGES HAD OCCURED: SCV# 596 *, 597 *, 5518 *, 6633 — EL 599'; 4727 *, 4729 * — EL 682'

NO DIMENSIONAL CHANGES OR DAMAGE, DIRECTLY ATTRIBUTED TO THE SEISMIC EVENT, WERE FOUND.

CORRECTIVE ACTION DOCUMENTATION - (D.R.'s): N/A

MEASURING/INSPECTION TOOLS I.D.: MISC. INSPECTION TOOLS

REMARKS * ALL REPAIR WORK WAS COMPLETE AT TIME OF WALKDOWN.

cc: S. DODEJA : E-180

G.P. Shaner, Jr. — INSPECTOR — DATE 2/4/86

John W Messenger — REVIEWED - LEAD INSPECTOR — DATE 2-4-86

441/T/1/ba

PNPP No. 5978

N/A	N/A	C-23665
CONTRACTOR	SPEC. NO.	INSPECTION REPORT NUMBER
WALKDOWN OF OPEN SCV'S AFTER SEISMIC EVENT OF 1/31/86		N/A
SYSTEM/COMPONENT/ACTIVITY		M.P.L. NUMBER
AX-1 @ FLOOR EL. 599'	21	2-3-86
LOCATION	INSP. TYPE	DATE

DUE TO THE SEISMIC EVENT WHICH OCCURED ON 1-31-86 THE FOLLOWING OPEN SCV'S HAVE BEEN FIELD VERIFIED TO DETERMINE IF ANY DIMENSIONAL CHANGES HAD OCCURED: SCV # 6833 (NR # CQCN-0126)

NO DIMENSIONAL CHANGES OR DAMAGE, DIRECTLY ATTRIBUTED TO THE SEISMIC EVENT, WERE FOUND.

CORRECTIVE ACTION DOCUMENTATION - (D.R.'s): N/A

MEASURING/INSPECTION TOOLS I.D.: MISC. INSPECTION TOOLS

REMARKS N/A

cc: S DODEJA : E-180

G. P. Shaner, Jr. 2/4/86
INSPECTOR DATE

John W Messenger 2-4-86
REVIEWED - LEAD INSPECTOR DATE

441/T/1/ba

Construction Quality Section
Nonsafety Inspection Report (NIR)

Sheet 1 of ___

PNPP No.: 5978

N/A	N/A	C-23666
CONTRACTOR	SPEC. NO.	INSPECTION REPORT NUMBER
WALKDOWN OF OPEN SCV'S AFTER SEISMIC EVENT OF 1/31/86		N/A
SYSTEM/COMPONENT/ACTIVITY		M.P.L. NUMBER
CC @ FLOOR EL. 679'	21	2-4-86
LOCATION	INSP. TYPE	DATE

DUE TO THE SEISMIC EVENT WHICH OCCURED ON 1-31-86 THE FOLLOWING OPEN SCV'S HAVE BEEN FIELD VERIFIED TO DETERMINE IF ANY DIMENSIONAL CHANGES HAD OCCURED: SCV # 6014

NO DIMENSIONAL CHANGES OR DAMAGE, DIRECTLY ATTRIBUTED TO THE SEISMIC EVENT, WERE FOUND.

CORRECTIVE ACTION DOCUMENTATION - (D.R.'s): N/A

MEASURING/INSPECTION TOOLS I.D.: MISC. INSPECTION TOOLS

REMARKS N/A

cc: S. DODEJA : E-180

G.P. Shaner, Jr. 2/4/86
INSPECTOR DATE

John W. Messenger 2-4-86
REVIEWED - LEAD INSPECTOR DATE

441/T/1/ba

Construction Quality Section
Nonsafety Inspection Report (NIR)

PNPP No. 5978

N/A	N/A	C-23677
CONTRACTOR	SPEC. NO.	INSPECTION REPORT NUMBER
WALKDOWN OF OPEN SCV'S AFTER SEISMIC EVENT OF 1/31/86		N/A
SYSTEM/COMPONENT/ACTIVITY		M.P.L. NUMBER
CC & IB @ VARIOUS ELEVS.	21	2-5-86
LOCATION	INSP. TYPE	DATE

THE ATTACHED MEMO DATED 2-5-86 CONCERNING THE WALKDOWN PERFORMED BY ENGINEERING TO EVALUATE CERTAIN OPEN REPAIR DISPOSITIONED SCV'S (EIR'S) FOR POTENTIAL DIMENSIONAL CHANGES CAUSED BY THE EARTHQUAKE OCCURING ON 1-31-86.

RESULTS INDICATED - NO NOTICABLE CHANGES.

CORRECTIVE ACTION DOCUMENTATION - (D.R.'s): N/A

MEASURING/INSPECTION TOOLS I.D.: N/A

REMARKS N/A

cc: S DODEJA - E180

G.P. Shaner, Jr. 2/5/86
INSPECTOR DATE

John W Messenger 2/5/86
REVIEWED - LEAD INSPECTOR DATE

441/T/1/ba

NIR C-23677 SH. 2 of 2

memorandum Gilbert/Commonwealth

February 5, 1986

to J. W. Messenger/M. R. Kritzer

from H. Dharia/S. C. Dodeja

subject Walkdown of Open EIR's for Potential Earthquake Effect

The following EIR's were walked down to see if the previously reported condition had changed due to the earthquake event.

EIR #	SCV #
CC-620-3	6712
CC-679-7	6720
IB-654-4	6797
CC-574-1	6708
CC-679-17	6730
CC-679-13	6726
Cc-679-1	6714

The conclusion of the walkdown was that there had been no noticeable change due to the earthquake.

H. Dharia

H. Dharia

S. C. Dodeja

S. C. Dodeja

CC: C. R. Angstadt
K. R. Pech

G-2
REV. 1-82

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

MEMORANDUM

"E"SO/-2712

☐ I no longer wish to receive this material.

E. M. Mead	ROOM E280	FROM	I. B. Babiak	DATE	February 3, 1986
T. M. Jameson jmg	E260	PHONE	6699 ROOM E260		
T. P. Keaveney	E230	SUBJECT	N71, Circulating Water System Walkdown to Assess the Intensity of the January 31, 1986 Earthquake		

On February 1, 1986, a system walkdown was performed on the N71 Circulating Water System and the following was observed:

No yard flooding was observed above the buried 12' diameter FRP on both supply to the condensers, and return line to cooling tower. No water was present in the Oil Storage Tank Dyke or beneath the temporary (trailers) lunch room building or in the Sodium Hypochlorite Storage Tank Dyke.

A walkdown was also performed inside the Turbine Building basement to assess if any damage was present to the system, and none was observed.

A walkdown of the cooling tower basin wall was performed and two vertical (minor) leaks were observed. One located on the south cooling tower forebay flume wall and other near the cooling tower raiser manifold (entry into the basin).

The severity of the vertical seam leak in the forebay flume wall exhibits approximately 1 to 2 gpm flow rate out into the yard and with approximately (less than) 1 gpm flow rate through the second vertical seam leak.

The Civil/Structural element is to advise on the severity of the leak with a repair solution.

clm

cc: R. A. Newkirk - E280
E. B. Ortalan - E260
N71 System File - E280
PO/DC - R290

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

☐ I no longer wish to receive this material.

S-8
REV. 1-82

MEMORANDUM

"C"SO-2773

TO K. R. Pech ROOM W220 FROM E. C. Christiansen ECC DATE February 7, 1986
PHONE 5467 ROOM W245
SUBJECT Review of Energized Circuits During 1/31/85 Seismic Event

Per your request, NCES - Electrical has reviewed the circuits that were energized during the January 31, 1986 seismic event. The intent of the review was to determine the number of active electrical components in the energized circuits. Active components were categorized in seven subgroups for this study. The groups were motors, power sources, switches, instruments, relays, transformers, and miscellaneous. The later category included lamps, fuses, resistors, diodes, etc. Attachment I lists major suppliers of equipment in each subgroup. Passive devices such as cables, lugs, terminal boards, conduits, and trays wre not considered in this study.

A listing of systems that were operating during the seismic event was obtained from Perry Plant Operating Department. This list is included as Attachment II of this memorandum. Upon an engineering review of this list it was determined that it was incomplete. Power sources, communication, security, and computer systems were added by Engineering. Attachment II also reflects these additions.

The total number of active components in the energized circuits was 47,460 . Attachment III contains Electrical Device Lists used by each engineer in their review of each energized system. A breakdown of the total active components by subgroup follows:

Motors	775
Power Sources	6493
Switches	6962
Instruments	4721
Relays	6968
Transformers	1885
Miscellaneous	19656

Total Devices 47,460

ECC/mcw

Attachment I

Vendors

Motors

General Electric

Siemens-Allis

Westinghouse

Reliance

U.S. Electrical

Transformers

Westinghouse

General Electric

Brown Boveri

Relays

General Electric

Westinghouse

Brown Boveri

Cutler-Hammer

Agastat

Potter Braumfield

Switchgear Breakers

Brown Boveri

General Electric

Cutler-Hammer

Switches

Allen Bradley

General Electric

States

Electroswitch

Batteries

C & D

Exide

Contractors

Cutler-Hammer

Allen Bradley

General Electric

Attachment I

MOV Operators

Limitorque

Rotorque

EIM

ITT

Chargers/Inverters

C & D

Power Conversion

Topaz

CYBREX

Fuse Disconnects

Cutler-Hammer

General Electric

Cutler-Hammer

Limit Switches

Limitorque

Instrument Switches

Magnetrol

Rosemount

ITT Barton

Mercoid

Meriam

MSW Instruments

Meters

General Electric

Brown Boveri

Westinghouse

Weksler

Instruments (T/C, RTD's)

Weed

Recorders

Leeds & Northrup

Transmitters

Rosemount

Gould

Foxboro

Magnetrol

Weed

Molded Case Breakers

General Electric

Westinghouse

Cutler-Hammer

Gould (Brown Boveri)

Attachment II

Systems Energized During Seismic
Event of January 31, 1986

System Supplied of PPOD

System	Description
C11	Control Rod Drive
C41	Standby Liquid Control
C71	Reactor Protection System
D17	Plant Radiation Monitors
E12	Residual Heat Removal
E21	Low Pressure Core Spray
E22	High Pressure Core Spay
F42	Fuel Transfer Equipment
G33	Reactor Water Cleanup
G41	Fuel Pool Cooling and Cleanup
M11	Containment Vessel Cooling
M13	Drywell Cooling
M15	Annulus Exhaust Gas Treatment
M21	Controlled Access HVAC
M23	MCC, Switchgear, & Misc. Area HVAC
M24	Battery Room Exhaust
M25	Control Room HVAC
M26	Control Room Emergency Recirculation
M27	Computer Room HVAC
M32	ESW Pumphouse Ventilation
M35	Turbine Building Cooling & Ventilation
M36	Off-Gas Building Exhaust
M40	Fuel Handling Building Ventilation
M41	Heater Bay Ventilation
M43	Diesel Building Ventilation
M45	Circulating Water Pump House Ventilation
N21	Condensate
N23	Condensate Filtration
N24	Condensate Demineralizers
N32	Turbine Control (EHC)
N71	Circulating Water
P11	Condensate Transfer and Storage
P20	Water Treatment
P21	Two Bed Demineralizer
P22	Mixed Bed Demineralizer
P41	Service Water
P42	Emergency Closed Cooling
P43	Nuclear Closed Cooling
P44	Turbine Building Closed Cooling
P45	Emergency Service Water
P47	Control Complex Chill Water
P49	ESW Screen Wash
P52	Instrument Air
P54	Fire Protection
P55	Building Heating
P61	Auxiliary Steam
P62	Auxiliary Boiler Fuel Oil
P72	Plant Underdrain

Attachment II

Systems Added by Engineering

System	Description
C91	Process Computer
C95	Emergency Response Information System
P51	Service Air
P56	Security
R11	Station Transformers
R14	110 VAC Vital Inverters
R15	Technical Support Center UPS
R22	Metalclad Switchgear
R23	480 V Load Centers
R25	Distribution Panels - 120, 208 & 480 volts
R36	Heat Tracing & Anti Freeze Protection
R41	Instrumentation
R42	D.C. System
R43	Standby Diesel Generator (SDG)
R44	SDG Starting Air
R45	SDG Fuel Oil
R46	SDG Jacket Water Coolant
R47	SDG Lube Oil
R51	Intra Plant Communications
R52	Maintenance & Calibration
R53	Exclusion Area Paging System
R57	Radio & In-Plant Antenna System
R61	Main Control Room Annunciator
R71	Lighting
S11	Power Transformers