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310TH GENERAL MEETING

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

310TH GENERAL MEETING

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

Thursday, February 13, 1986

The 310th General Meeting convened at 8:45 a.m., Mr. David A. Ward, chairman, presiding.

ACRS MEMBERS PRESENT:

MR. DAVID A. WARD
MR. JESSE C. EBERSOLE
DR. MAX W. CARBON
MR. HAROLD ETHERINGTON
DR. WILLIAM KERR
DR. HAROLD W. LEWIS
DR. CARSON MARK
MR. CARLYLE MICHELSON
DR. DADE W. MOELLER
DR. DAVID OKRENT
MR. GLENN A. REED
DR. FORREST J. REMICK
DR. PAUL G. SHEWMON
DR. CHESTER P. SIESS
MR. CHARLES J. WYLIE

PUBLIC NOTICE BY THE

UNITED STATES NUCLEAR REGULATORY COMMISSIONERS'

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

THURSDAY, FEBRUARY 13, 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.

1 P R O C E E D I N G S

2 MR. WARD: We are ready to go to the first
3 agenda, radioactive waste management and reactor
4 radiological effects. Mr. Moeller?
5 MR. MOELLER: The Subcommittee on Reactor
6 Management and Radiological Effects had a combined meeting
7 during the period January 15 through 17, 1986. We had an
8 excellent representation from the full committee, including
9 Carbon, Ebersole, Kerr, Mark, Remick, and Shewmon. Not
10 everyone was there every day, but at least those people
11 were there one day or the other. And then we had a team of
12 consultants who backed us up for the meeting.
13 We covered some six different topics, beginning
14 with the revisions to 10 CFR 60, which is the high-level
15 waste disposal criteria. We also looked at research
16 activities of the NRC on high-level wastes and low-level
17 wastes; we looked at the proposed revisions to 10 CFR part
18 20, standards for protection against radiation, and then we
19 had three rather interesting additional items.
20 One was a report from the Committee on
21 Interagency Radiation Research and Policy Coordination,
22 which was, as I say, just a very interesting presentation.
23 Similarly, we had a presentation by a group from the Atomic
24 Industrial Forum on the work of their national
25 environmental studies project. And then lastly we had an

1 advance briefly -- do you recall -- on the initial phases
2 of the work within EPA, to develop standards for low-level
3 waste disposal.
4 Let me offer a few comments on each of these
5 items.
6 With regard to 10 CFR, part 60, one of the
7 primary efforts of the NRC staff in revising it is to make
8 it compatible with the new EPA high-level waste standards.
9 Although the two agencies work closely together, as EPA was
10 developing their new standards there were certain words or
11 phrases and terminology and so forth that were used in the
12 new EPA standards which the NRC wanted to incorporate into
13 their part 60. Yes, Dr. Shewmon?
14 MR. SHEWMON: There was talk about $33 million
15 to make some changes. When you get to that change, will
16 you let me know? Or is that this one?
17 MR. MOELLER: No. That's the 10 CFR 20, where
18 the estimates are. It might cost that much.
19 Just a few instances or examples. The EPA used
20 the words in their high-level waste standards about a
21 "reasonable expectation of meeting the standards." The NRC
22 staff uses the words "reasonable assurance of meeting the
23 standards."
24 The new rule -- well, we noted that and the
25 subcommittee tended to like EPA's words, "the reasonable

1 expectation."

2 We noted that the new rule is not consistent

3 with either ICRP publication 26 or with the proposed

4 revisions to 10 CFR 20.

5 Dr. Mark pointed out some, not necessarily

6 discrepancies, but easily misinterpreted points in table 11

7 to 10 CFR 60, specifically where they set the releases for

8 plutonium, the allowable releases at some 10 times the

9 level for thorium. And he thought that should be explained,

10 which, indeed, they agreed to do. And then he also raised

11 some questions on the release limits for carbon 14. So we

12 offered them guidance on those points. And during our

13 meeting, we agreed that rather than providing them written

14 comments on 10 CFR 60, that a transcript of the meeting

15 would serve as our report or the expression of our thoughts

16 to them.

17 We also, as part of 10 CFR 60, reviewed two

18 draft generic technical positions. One was related to the

19 disturbed zone, the definition and so forth of the

20 disturbed zone and the second one was related to

21 groundwater travel time. And we, again with our

22 consultants, concluded that the two generic technical

23 positions were very well written but nonetheless, as would

24 be anticipated, we had some remarks to make which we did

25 pass along, again through means of the transcript of our

1 meeting.

2 On the research on high-level wastes and

3 low-level wastes, we discussed the modeling development

4 underway within the NRC and we, of course, raised questions

5 about the quality and quantity of the data that would be

6 available, or the data that are available for testing these

7 models. We also noted that where they have, perhaps, good

8 models -- or they are moving along pretty well for models --

9 in the development of models for basalt, they really have

10 some way to go yet for media such as Tuff, T-u-f-f.

11 We also discussed how accurate the models had to

12 be, and the validation of the models.

13 They also discuss models for estimating the

14 corrosion of waste containers. And these models at the

15 moment are primarily for saturated media, and of course we

16 raised the questions about what are the models for

17 unsaturated media.

18 In terms of looking at the high-level waste

19 research, several subcommittee members raised questions

20 about the procedures used by the NRC staff in setting

21 priorities for their research, and we questioned that they

22 had a sound approach for doing this, or at least it wasn't

23 that clear to us.

24 Relative to low-level waste and low-level waste

25 research, the NRC staff has pressure -- I guess that's the

1 proper word -- pressure being played upon them now because
2 of the Low-Level Radioactive Waste Policy Amendments Act of
3 1985. These amendments require that within 12 months the
4 NRC must identify disposal methods other than shallow land
5 burial; and within, you know, the acceptable or best ways
6 other than shallow land burial -- and within 24 months, or
7 two years, they must provide technical information to the
8 states on how to proceed with alternate disposal practices.
9 So, both of these, as I say, are a real
10 challenge to the NRC staff and they are moving ahead,
11 exploring as alternatives to shallow land burial such
12 approaches as mined cavities, above-grade engineered vaults;
13 below-grade engineered vaults; augered holes, concrete
14 bunkers, and so forth.
15 Again, we reviewed and commented on that and
16 they agreed to use the transcript of the subcommittee
17 meeting.
18 MR. KERR: Will these be dedicated bunkers?
19 MR. MOELLER: It depends. If they are built in
20 Germany, they probably would be.
21 The proposed revisions to 10 CFR 20, that was
22 our next topic, and today we are going to have
23 presentations by the NRC staff on that. So I will not
24 dwell upon it.
25 The main point to make here is that the

1 subcommittee will be asking the full committee to approve
2 and forward to the NRC staff, some comments prepared by the
3 subcommittee on 10 CFR part 20 in its revised form. And we
4 are doing this specifically in response to a request from
5 the NRC staff, because the proposed rule has been issued
6 for comment and they would like our comments to be a part
7 of the written sets that they receive. And so we are going
8 to ask you to consider that on Saturday and see about
9 submitting them to the staff.
10 One other item. I'm sure the NRC staff in
11 reviewing the proposed revisions will mention the de
12 minimis proposal that is incorporated within these proposed
13 standards, proposed revised standards. And I wanted to
14 mention that specifically because Dr. Lewis had proposed a
15 couple of months ago that the committee send a letter to
16 the Commission on the subject of de minimis.
17 I mention this because it is my belief that the
18 revision in the proposed -- the proposed revision in 10 CFR
19 20 covers the very points that Dr. Lewis is making.
20 The briefing by the Atomic Industrial Forum
21 group on their national environmental studies project was
22 led by David Harward, who is the director of that program.
23 And let me just cite as examples several of the topics that
24 they've either recently completed or that they have under
25 way and which the subcommittee heard about.

1 One was guidelines for radiological record-keeping.
2 This pertains to the manner in which records on radiation
3 exposure for workers are prepared and maintained. And they
4 are giving particular attention to assuring that they have
5 an adequate record in case of litigation.
6 They are working secondly on criteria, and they
7 have issued a report on criteria to airborne release dose
8 model selection. Excuse me, they are still working on that.
9 What they are going to do there is to develop
10 problems, much as is done with other codes, and circulate
11 these to the utilities and ask them to run through them and
12 submit their estimates of doses and so forth that result.
13 And so they hope ultimately to develop an acceptable set of
14 generic models with the weaknesses and strengths of each
15 clearly delineated.
16 Thirdly, we heard a very interesting
17 presentation by William Brown of Westinghouse on the
18 environmental consequences of higher fuel burn-up. For me,
19 personally, I found this a fascinating report because he
20 looks and projects ahead that you use the fuel for 60,000
21 days per ton, versus, say, 33,000. Then he looks at the
22 impacts of this upon uranium mining requirements, milling
23 requirements, and the nature of the wastes that would be
24 disposed. And we have a copy of that report which we will
25 gladly share with those who want it. Obviously it results

1 in a decreasing requirement in terms of uranium mining and
2 milling.
3 He also even factored into his equation his
4 projections on the percent uranium content of the ores that
5 will be mined in future years. So it was just, in my
6 opinion, an excellent report. And then he also factors in
7 the reduction in occupational dose because you are
8 refueling on a less frequent basis. And all of this, then,
9 he has put into a revised table, S-3 and S-4.
10 Other reports -- let me be brief; I want to give
11 plenty of time to our staff members from the NRC who are
12 here -- but they are working on methodologies for
13 classification of low-level wastes; on decommissioning of
14 nuclear power plants, specific emphasis on dose assessments;
15 on the occupational exposure implications of NRC-initiated
16 plant backfits; the evaluation of the potential for
17 deregulation of the disposal of certain classes of
18 low-level wastes that might be de minimis. They are
19 working on methods for improving the accuracy of the
20 prediction of doses to workers and regulatory
21 considerations involved in extending the operating life of
22 nuclear power plants.
23 We have heard so much recently about going from
24 40 years to 70 or 100. Well, they are right on top of it.
25 Next we heard from this Committee on Interagency

1 Radiation Research and Policy Coordination, Dr. Alvin Young.
2 The chairman of the committee was here as well as the
3 director of the research activities within that agency,
4 which is Dr. Randy Catwell of the National Bureau of
5 Standards.
6 They are coordinating research on radiation
7 protection that's under way among the various federal
8 agencies. I guess I should say they are attempting to
9 coordinate it. I think they are moving forward quite well.
10 They are meeting frequently. They have a budget of about
11 $1 million a year, which comes from the contributions from
12 each of the federal agencies that are participating in that
13 activity, except for one. OMB participates but does not
14 contribute.
15 Some of the topics that they are working on:
16 The health effects of high-level radiation. They are
17 working on de minimis, quality factors for neutron
18 exposures, the use of SI units and the overall assessment
19 of U.S. population dose from radiation. That's an effort
20 they are doing cooperatively with the NCRP. And in this
21 effort, they are looking at the total U.S. population, the
22 total rad exposure they receive, and they are delineating
23 exactly where it comes from, the contribution from each of
24 the sources, natural background as well as artificial. And,
25 in terms of perspective and education of the public, I

1 think this would be very helpful.

2 MR. MARK: This will include indoor radon?

3 MR. MOELLER: Yes, sir. It will include indoor

4 radon.

5 Their report, which they gave us copies of,

6 listed some eight or so what they consider the more

7 important radiation protection problems in the U.S. Let me

8 just name a couple of them: Rad waste was one; de minimis;

9 radiation education and awareness; radiation measurement

10 recording and control; rad protection standards; radiation

11 risk assessment; and then radon exposures of the population.

12 That was listed as number 2. Then they have a footnote,

13 though, which I don't like to quote which says these items

14 are not listed in order of priority.

15 Lastly, the EPA standards for low-level waste,

16 we heard a report from both Dr. Shelley Meyers, who is

17 director of the EPA Office of Radiation Programs, and then

18 Floyd Galpin, one of his scientific supporting staff

19 members, on their current thinking on this subject. I was

20 pleased to hear this because we are trying to keep abreast

21 of what they are doing so that later we don't find

22 ourselves suddenly not knowing what has been done. We'll

23 know in this case if we can continue these meetings and

24 continue the interaction.

25 Let me ask at this point if any of the other

1 subcommittee members who were there at the meetings have
2 any comments to add or questions before we move into the
3 formal presentations?
4 MR. SHEWMON: You had a good memo on this
5 meeting, Max. When do we discuss that?
6 MR. CARBON: It's in the thing here. I don't
7 care. It's here if it needs discussing.
8 MR. MOELLER: I can summarize that. Max had
9 three points in his memo. One was to repeat the question
10 he raised at the subcommittee meeting concerning the
11 procedures used by the NRC staff in setting priorities for
12 research on high-level wastes. I thought that was an
13 accurate and, you know, important point. And it has been
14 incorporated into the reactor safety research draft report
15 which you will see later today. It's in Chuck Siess' -- in
16 the waste management chapter of that report.
17 His second point was concerning -- I hope I'm
18 taking them in order -- it was low-level waste and Max
19 raised the point as to why is the NRC going out and asking
20 for this $200,000 or whatever dollars they are spending for
21 a contractor to look at alternatives to shallow land burial?
22 I think Max was asking not only why are they doing it, but
23 is the -- what will be the quality of the product that
24 results? Is it money well spent?
25 There, I think -- we are not going to be

1 discussing that this morning -- at least the staff is not
2 going to be covering it.
3 MR. SHEWMON: The one that I think I was
4 interested in is the justification for voting -- how much
5 effort the committee should devote to low-level waste.
6 MR. MOELLER: That's right. And he asked. That
7 was another primary question. Should the committee be
8 involved? And, again, whatever the committee decides is
9 fine with me. I think the NRC staff, though, has no choice.
10 There is -- Congress has passed a law requiring them to
11 meet this one-year and this two-year deadline and they have
12 to do it.
13 If you want them to do it independently of the
14 committee, they might do it. My own thinking is that we
15 should, perhaps, specifically address that question to the
16 Commissioners and say: Do you want our involvement or not?
17 And that would settle it.
18 MR. SIESS: And will they pay for it?
19 MR. MOELLER: And are they willing to pay for it.
20 His third comment pertained to 10 CFR 20, in
21 questioning whether the revisions of the -- necessary
22 revisions of 10 CFR 20 might be done by patching up, or,
23 you know, minimum revisions of the existing document versus
24 an overhaul, a total overhaul of the document? We will --
25 that will be covered this morning. So, if it's all right,

1 why don't we call upon the NRC staff.

2 We have with us, then, for the first

3 presentation, which should go no more than a total of 45

4 minutes, which will be roughly until 10 of 10:00, our first

5 presentation is on 10 CFR 20. We will have, then, two

6 additional briefer presentations, one on the international

7 cooperative programs that are under way on high-level waste

8 research, and the third one on international comparisons of

9 waste management codes, specifically the hydrologic code.

10 The first presentation will be by Robert

11 Alexander and Alan Brodsky.

12 MR. ETHERINGTON: On the Westinghouse long

13 burn-up study, was there any significant benefit with

14 regard -- you are going to have some burn-up over gamma

15 captured by things like strontium 90 and so on?

16 MR. MOELLER: What I would do, we'll give you a

17 copy of the report. It's not that thick. I cannot answer

18 your question. Can anyone else who was there answer? Paul?

19 MR. SHEWMON: You get some benefit from burning

20 up the plutonium. It's the only way we are likely to do it.

21 I don't know if that's an end gamma or not.

22 MR. MOELLER: You meant you might burn out some

23 of the fission products -- strontium 90 to 91? Is that

24 what --

25 MR. ETHERINGTON: Yes.

1 MR. MOELLER: I don't recall that he covered
2 that.
3 MR. ETHERINGTON: Well, it's not important.
4 MR. MICHELSON: As long as you are on that
5 subject, did they consider the increased fragility of the
6 fuels when you go to the very long burn-up and the effect
7 on contaminating pools, et cetera?
8 MR. MOELLER: Yes. He had consideration of that
9 factor.
10 MR. MICHELSON: You have to guess, of course,
11 what the problem is going to be to in the very long burn to
12 the clad -- it could be severe.
13 MR. SHEWMON: They have run a lot of it and have
14 not had a lot of problems.
15 MR. MICHELSON: 60-, 70,000 megawatt days --
16 it's a very long burn.
17 MR. SHEWMON: Westinghouse has run a lot of
18 stuff like that, more than the other vendors had.
19 MR. MICHELSON: I was thinking of the boiling
20 water reactor.
21 MR. MOELLER: Okay. Bob Alexander has the floor.
22 Welcome. Go ahead.
23 MR. ALEXANDER: Thank you, Dr. Moeller. It's a
24 pleasure to address the ACRS this morning about our
25 proposed revision of 10 CFR 20, which is the Commission's

1 major standard on radiation protection, both for workers
2 and for members of the public.
3 I would like to make as objective a presentation
4 as possible, although I have to admit that we are seeking
5 ACRS support for our regulation. We are also seeking
6 recommendations from the ACRS which would help us improve
7 the document. And we hope to receive those from you.
8 I would like, then, to just mention the safety
9 improvements and other improvements that we see in the
10 proposed revision, and then for more detailed presentation
11 I'll call on the project manager for the major revision,
12 Dr. Alan Brodsky the first safety improvement I would like
13 to mention is the dose limit for external radiation to the
14 whole body is being reduced from a maximum of 12 rems per
15 year to a maximum of 5 rems per year. This is in
16 compliance with ICRP recommendations made several years ago
17 and being adopted around the world.
18 The 12 rems per year is certainly considered to
19 be, in these days, too much radiation for a person to
20 receive and 5 rems per year, in fact, is considered to be
21 an absolute maximum. The dose average -- average dose to
22 workers that the ICRP recommends is more on the order of a
23 half rem per year.
24 In our nuclear power industry, the average dose
25 is about .65 rems per year. We hope that this reduction in

1 the limit per year will result in an associated reduction
2 in the average dose to workers in the nuclear power
3 industry.
4 Flexibility, when it's really needed for the
5 nuclear power industry, is retained by providing for
6 planned special exposures, whereby, under very special
7 conditions and with adequate reports to the staff, the 5
8 rems limit can be exceeded.
9 The second safety improvement I would like to
10 mention is for the first time the Commission has proposed a
11 limit to protect the embryo fetus. Our policy --
12 MR. REED: You, of course, realize the more you
13 lower the limit, the more you probably increase the total
14 radiation to all individuals involved, in that you've got
15 to turn over people, you have to shuffle -- intermittent
16 work and so forth. Accumulated radiation goes up.
17 The fact that the industry was able to, let's
18 say, have an average of .56 doesn't mean that it's in the
19 best interests of safety or quality work -- for total
20 exposure, to keep down the maximum allowable per person.
21 MR. ALEXANDER: Well, of course the collective
22 dose consideration we have been aware of for years and we
23 have looked very hard at that. We believe, based primarily
24 on data submitted to us by the industry, that no rise in
25 the collective dose would occur until the dose limit got

1 somewhat below 2.5 rems per year. That's because of the
2 rather conservative approach that the industry takes to
3 exposures anyhow. Most of them already have a limit of 5
4 rems per year that they impose.
5 We do not expect any increase in the collective
6 dose as a result of this reduction.
7 MR. REED: What, now, is going to be the limit?
8 The same as it has been, per quarter?
9 MR. ALEXANDER: The limit will remain at 3 rems
10 per quarter, but there will be a 5 rems per year limit
11 rather than 12. It will be 3 rems in any one quarter and 5
12 for the year.
13 MR. REED: So, really, except for the endpoint,
14 there's no change in what is being practiced?
15 MR. ALEXANDER: We expect very little change as
16 a result of this regulation in the nuclear power industry.
17 Because, as I said, they are -- most of them operate under
18 a 5 rems per year limit or less already.
19 MR. REMICK: Am I correct in the last year there
20 have been no individuals over 5 rems?
21 MR. ALEXANDER: I think in the past year there
22 were no exposures more than 3 rems per quarter at any one
23 facility.
24 MR. REMICK: How about the 5 rems? Were there
25 any exceeding 5 rems?

1 MR. ALEXANDER: Yes. Yes.

2 MR. CARBON: How many? Half a dozen?

3 MR. ALEXANDER: No, no. Much more than that. I
4 looked at that number recently. I should have brought it
5 with me. But let me go ahead and make a guess, even though
6 it's a little dangerous to do so. I think it does not
7 exceed 200. There's quite a number that exceed the 5 rems.

8 MR. KERR: Under the new proposal it would be
9 possible to go over 5, but this would involve a reporting
10 after the fact?

11 MR. ALEXANDER: Yes. It requires a decision at
12 a fairly high level of management at the plant. It
13 requires a documentation of the justification and a
14 notification to the regional office, so that the inspector
15 at his next inspection can ask to see the justification.

16 We don't think that any use will be made of this.
17 If you have ever been to one of those plants and looked at
18 the preparations that they do for an outage, with every
19 detail planned so carefully, it's inconceivable that a
20 planned special exposure would be undertaken.

21 MR. KERR: I can remember at least two cases in
22 which people with responsibility for plant operation had
23 gone into areas of fairly high radiation level because they
24 thought an emergency existed, recognizing at least the
25 possibility of overexposure. They made the decision,

1 certainly without having any committee meetings and
2 high-level participation. I think in at least in one case,
3 and maybe both cases, the company was subsequently fined by
4 the NRC.
5 It would seem to me there still might be
6 situations in which responsible persons would have to make
7 such a decision.
8 MR. ALEXANDER: Yes, the planned special
9 exposure does not apply to emergency conditions. It
10 applies to conditions where adequate planning and higher
11 level management approval can be obtained.
12 MR. KERR: What does apply to emergency
13 situations?
14 MR. ALEXANDER: We don't really apply
15 limitations to emergency conditions.
16 MR. SHEWMON: They may fine them after the fact,
17 but they don't have any plans; is that what you are saying?
18 It's totally undefined in your regulations?
19 MR. ALEXANDER: Yes. That's true, it is -- of
20 course it is true that under emergency conditions a
21 regulatory dose limit might be exceeded for good reason.
22 If that takes place then an overexposure in violation of
23 the regulations has occurred. That's certainly true.
24 The fact that it was an accident or emergency
25 doesn't relieve one from complying with the regulations.

1 However, all of those limits are based on normal operating
2 conditions and we do not place any limitations on emergency
3 action.
4 MR. MARK: Has it ever happened that someone got
5 10 rem, or whatever, in an emergency situation, where the
6 staff has agreed that that was a reasonable thing to do and
7 no fine was appropriate?
8 MR. ALEXANDER: Dr. Mark, I don't know the
9 answer to that question. I've never, of course, worked in
10 the inspection and enforcement program. But I think it's
11 reasonable to assume that the conditions of the
12 overexposure would be taken into consideration in
13 determining the staff's action.
14 MR. MARK: I agree it would be reasonable. I
15 wasn't sure that it happened.
16 MR. ALEXANDER: Well, I'm sorry. I can't answer
17 that.
18 MR. CARBON: You speak of, I guess, the apparent
19 desire to get the average exposure from 650 millirems per
20 year --
21 MR. MARK: 9 rems.
22 MR. ALEXANDER: .65 rems per year is
23 approximately the average dose that workers in the nuclear
24 power industry receive now.
25 MR. CARBON: Considering the conservatism that

1 seems to be built into setting the limits, this doesn't
2 really sound like a very high amount to begin with. Can
3 you put this in some sort of terms that I can grasp. What
4 is the significance in reducing from 650 to whatever you
5 hope to achieve? How much do you hope to push this down?
6 How important is it?
7 MR. ALEXANDER: Well, what is being striven for,
8 not only in this country but elsewhere, is to achieve the
9 recommendation of the International Commission on
10 Radiological Protection, that the radiation risk to workers
11 be no more than 1 times 10 to the minus 4 per year, or 1 in
12 10,000 per year -- that is death, fatalities. Well,
13 fatalities and serious genetic effects for the first two
14 generations; those two are lumped together. Fatal cancer,
15 then -- and serious genetic effects that the worker would
16 experience himself, or his grandchildren -- children. 1
17 times 10 to the minus 4, or 1 out of 10,000 per year is the
18 criterion that's used. And that is associated
19 approximately with the average dose, average dose of .5
20 rems per year.
21 So that at an average dose of .65 rems per year,
22 the ratio of .65 over 5 is above the ICRP's criteria of 1
23 in 10,000 per year. That's a somewhat arbitrary figure of
24 theirs, based on risks observed in what are considered to
25 be the safer industries. We would like our -- this

1 industry that we regulate to be considered a very safe
2 industry. We feel that at .65 it is a safe industry, but
3 the 1 in 10,000 is the goal that we would like to achieve.
4 MR. ETHERINGTON: Do you have a number for death
5 from all other causes?
6 MR. ALEXANDER: Sir?
7 MR. ETHERINGTON: Do you have a number for the
8 number of fatalities from all other causes?
9 MR. ALEXANDER: Yes, I do. Such information is
10 difficult to get for the nuclear power industry alone
11 because the Department of Labor statistics cover utilities
12 and other -- even other service industries in general
13 without singling out the nuclear power industry. But from
14 information that I have been able to gather, the overall
15 risk to workers at the nuclear power plants already would
16 be something on the order of 1 times 10 to the minus 4
17 without the radiation. Perhaps a little more.
18 MR. WARD: Harold, could you talk in the
19 direction of the microphone, please?
20 MR. ETHERINGTON: Yes. I've finished, anyway.
21 MR. CARBON: The nuclear power plants really are
22 very, very safe places at which to work. How does this
23 compare with industry in general? How does it compare with
24 the teamsters? Obviously driving a truck on the highways
25 isn't as safe --

1 MR. WARD: Max, I'm sorry, but I think the
2 reporter is having trouble hearing you, too.
3 MR. ALEXANDER: We have published information of
4 that nature in our Regulatory Guide 8.29. I don't have it
5 with me, but of course I do remember. I can give you a
6 general answer that even that -- even with the radiation
7 risk in the nuclear power industry, that overall annual
8 risk is considerably lower than many of the occupations --
9 more dangerous occupations.
10 But our criterion, as I mentioned, is to make
11 the radiation risk comparable with the safer industries.
12 So the fact that these workers are already far below mining
13 and quarrying and agriculture and some of the more
14 dangerous industries -- we would like to lower the average
15 radiation dose.
16 MR. KERR: Can you achieve something significant,
17 in your view, by going from 0.65 to 0.5?
18 MR. ALEXANDER: Would you repeat that?
19 MR. KERR: I say, in your view, would you
20 receive a significant increase in safety by going from an
21 average of 0.65 to an average of 0.5?
22 MR. ALEXANDER: The way we look at it, it would
23 be the ratio of 0.65 over 0.5.
24 MR. REED: I don't know that you got the full
25 significance of what I said.

1 MR. ALEXANDER: Could I try again?

2 MR. REED: Let me try to say it again. If you
3 cause more worker turnover in the, shall we say, repair of
4 a steam generator -- that is, the throughput becomes
5 greater, in my opinion, the quality of the work goes down
6 and the total exposure goes up.

7 Now, when you push down the quality of the work
8 because you just can't move lots of bodies, and pretty soon
9 you are moving warm bodies rather than quality bodies
10 through a highly skilled work activity -- when you push the
11 volume of bodies through, you are bound to have quality
12 deterioration.

13 Now, when you have quality deterioration, you
14 can run up the -- not your kind of radiation safety risks,
15 but you can run up the safety risk at the facility. It
16 might even involve public safety, by leakage or tube
17 failure or things like that.

18 Quite frankly, I would not like to see the
19 workplace radiation exposures go down because I think they
20 will be negative in their impact on the total overall
21 safety for the public and they will increase the total
22 exposure for the work force.

23 MR. ALEXANDER: Well, we of course have looked
24 very hard at that question for a number of years, including
25 plant safety considerations. And we have coordinated very

1 carefully with the nuclear power industry itself.

2 We'll find out during the public comment period

3 how they feel about that in a more comprehensive manner,

4 but the information we have today is that the problem --

5 what we are proposing in 10 CFR 20 will not create or

6 exacerbate either the plant safety problems or the

7 collective dose in the nuclear power industry.

8 MR. REED: I think I would take a different

9 position. I would like to point out that I just read two

10 reports on two different facilities, one facility where

11 they had 280 people in the maintenance organization and the

12 other facility where they had 60 maintenance people in that

13 maintenance organization.

14 The 280-man group -- maybe they are paying more

15 attention to getting the radiation exposure per individual

16 down -- the 280-man group is reported by the NRC staff as

17 being very poor. The thing is not going well, the turnover

18 is high and the maintenance thing is bad. Yet the 60-man

19 group is doing fine.

20 You have to look at those kind of reports if you

21 are interested in the overall nuclear plant safety issue.

22 MR. ALEXANDER: I feel like we have. The

23 highest exposure people are people who work for service

24 companies like Westinghouse and General Electric,

25 Combustion Engineering, Babcock & Wilcox, highly skilled

1 people who go from plant to plant.

2 A few years ago, the average dose of one of

3 those crews -- I think the one with the larger number you

4 mentioned -- was 6 rems per year. We expressed

5 dissatisfaction at that, and that dose is now -- average

6 dose is about cut in half now for that group, down to about

7 3 rems per year.

8 There are two ways that that can be done,

9 essentially. One is to reduce the dose rates or the

10 working times, in which case true safety improvement occurs.

11 Another way is to add additional workers, in which case the

12 risk is just spread out among additional workers.

13 This reduction that I mentioned, we checked into

14 that and found out that about 235 helpers had been added to

15 the crew, and the collective dose had gone up almost by a

16 factor of two. Of course, that isn't the way to do it.

17 We feel we are very much aware of this problem.

18 I feel like we are on top of it. We feel that since the

19 plants are already operating -- because of their own

20 radiation liability problems -- at limits at 5 rems per

21 year or less, that this reduction to a maximum of 5 rems

22 will not exacerbate the problems that you are worried about.

23 MR. REED: I'm not concerned about 5. It's

24 there. It is in the workplace. All you did was do away

25 with the ends, and I don't think that's what your problem

1 is.

2 But the thing that bothers me is there's always
3 this tightening of the screw without looking at how strong
4 the bolt is. You know? If you keep tightening the nut,
5 the bolt is going to break. So you have to look at the
6 other side of the issue, what you do to accumulate an
7 exposure in total, and what you do to the quality of the
8 work.

9 MR. ALEXANDER: Well, I agree with you. But I
10 think, sir, that the problem that you are raising is
11 associated with regulatory implementation of the
12 occupational ALARA concept as opposed to the limit. And I
13 would certainly agree with everything you said with respect
14 to regulatory implementation of the ALARA concept. I think
15 the recommendation we are making with respect to upper
16 limits is sound and will not be harmful.

17 MR. REED: In other words, it's based on
18 accumulated good statistics that this is in the best
19 interests of people, that the radiation exposure be, per
20 person, be inched down. You have the background; the
21 International Committee has facts?

22 MR. ALEXANDER: Yes, sir. This recommendation
23 has been made by the International Commission on
24 Radiological Protection and it is being adopted, actually,
25 throughout the world.

1 I believe -- I don't know about behind the Iron
2 Curtain, but in the western world I believe the first
3 country to adopt these recommendations was Great Britain.
4 Their recommendations went into effect the first of this
5 year.
6 In the occupational ALARA area, I think there is
7 more concern, and I expect a lot of public comments on that.
8 We are recommending that the occupational ALARA concept be
9 made mandatory, whereas to date it has been exhortatory --
10 we have exhorted the individual to apply the occupational
11 ALARA concept which says you should reduce doses as far below
12 the limits as is reasonable, considering cost, all types of
13 cost.
14 We are changing that to a mandatory requirement
15 that these plants have occupational -- written occupational
16 ALARA programs that we can inspect against. And there are
17 those who say that that will have a harmful effect because
18 there's no lower limit. That makes it possible for the
19 government to keep pressing lower and lower and lower, no
20 matter how safe a plant already is.
21 I think we do have a problem there.
22 MR. KERR: It seems to me it also effectively
23 eliminates ALARA because you now have written technical
24 specifications in effect for a specific dose rate, perhaps
25 different for each plant, but it's no longer a question of

1 its being reasonably achieveable. It's there. You have to
2 achieve it.
3 MR. ALEXANDER: As far as the 5 rems per year,
4 it is a regulatory limit and it doesn't matter how much it
5 costs. You have to comply with it. I'm talking about
6 ALARA, even though the dose is far below --
7 MR. KERR: I am, too, and I'm referring back to
8 Appendix I, which, as far as I know, if one meets these
9 criteria, the numerical criteria for exposure with the
10 public, you are considered to be in compliance with "as low
11 as reasonably achieveable" for each missions from plants.
12 The same thing will occur if you become more specific and
13 more mandatory.
14 MR. ALEXANDER: I don't think so. May I try to
15 justify that answer?
16 MR. KERR: Yes, you may.
17 MR. ALEXANDER: In the case of protection of the
18 public from radioactive material and effluents, the EPA and
19 the NRC and others --
20 MR. KERR: You don't have to explain Appendix I
21 to me. I think I know how it came into existence. I
22 simply referred to it as an example of how ALARA was
23 determined -- which to me is no longer ALARA. You are
24 going to tell me how your proposal is different, I think.
25 I am interested in that.

1 MR. ALEXANDER: Simply for the record, may I say
2 that the limits in Appendix I in 40 CFR 190, in the Clean
3 Air Act, are based on technology -- available technology
4 considerations.
5 In the applications of the occupational ALARA
6 concept, the use of available technology is not the
7 principal consideration. The principal consideration is,
8 in the case of nuclear power plants, the design and age of
9 the plant. What is reasonable at one plant may be
10 completely out of reason in terms of cost, for another.
11 MR. KERR: I guess I didn't make my comment
12 specific enough to be clear. My point was that for each
13 plant there probably will be numerical goals. As you point
14 out, they are likely to be different for each plant. It
15 seems to me that once you put down numerical guidelines,
16 you go into the technical specifications. You no longer
17 have something which is an effort to achieve. It now
18 becomes a regulation for that plant.
19 MR. ALEXANDER: I don't anticipate ALARA numbers
20 being placed in any technical specifications. What will be
21 required is that the -- each plant have an occupational
22 ALARA program. They might have all kinds of goals in their
23 individual program and whatever goals they have written
24 there will become regulatory limits, as far as their
25 inspector is concerned. And, so, as far as the individual

1 plants setting a limit, that may very well be done. But
2 the Federal Government will not be setting numbers for
3 these plants the way it has for effluent control.
4 MR. KERR: So when the Federal Government forces
5 a plant to set a limit, that's not the Federal Government
6 setting the limit it's the plant doing it voluntarily?
7 MR. ALEXANDER: Not voluntarily, involuntarily,
8 but picking their own number.
9 MR. WARD: Mr. Reed?
10 MR. REED: I guess what you said is ALARA may
11 now become mandatory rather than the way it has been in the
12 past. Again, it seems to me that the ACRS, as a body, and
13 I in particular, are interested in the overall, total best
14 safety that we can achieve in the operation of nuclear
15 piles important for power. Okay.
16 I think perhaps you are pursuing your discipline
17 and looking at your discipline. This body wants to look at
18 the total.
19 Now, there are other bodies in the regulatory,
20 that pursue their discipline. Security is an example. I
21 have had a lot of problems with the way security has been
22 pursuing its discipline, as aggressively, I guess, as they
23 should. But we find situations where safety is harmed by
24 the pursuit of discipline. The Davis-Besse incident is an
25 example where they felt it was vital to saving something

1 that was locked up and could not be made available for use.
2 What I'm worried about, in pursuit of your
3 discipline, that you may lock up some safety and put it
4 away.
5 ALARA made mandatory -- I understand what that
6 means -- it means that now inspectors can go to the
7 facilities and issue citations and fines can be collected.
8 It puts that whole process into operation. And what that
9 will mean, because this is the way utilities respond, is
10 they will immediately pump up their ALARA program to try to
11 prevent a citation which looks bad in the newspaper and so
12 on and so forth -- they'll pump it up more.
13 But will that mean that getting a crew ready to
14 do a job that's quite important, the ALARA aspects will
15 take precedence over the time aspects, when a stitch in
16 time might save nine lives?
17 I'm not sure that you are considering these
18 kinds of things in the total safety issue.
19 MR. ALEXANDER: Most of these programs have
20 Britain ALARA criteria. What would happen is that those
21 written documents, which were composed by plant management,
22 would become mandatory documents against which the
23 inspector could inspect and cite.
24 We think that it is very possible that what
25 would happen at some of these plants is exactly as you say,

1 particularly after a plant manager finds that he has been
2 fined for exceeding one of his own limits. He may very
3 well call his own people in and say: Let's make sure that
4 doesn't happen again.
5 MR. REED: And slow down the repair work; yes.
6 MR. ALEXANDER: If that happens, that would be a
7 relaxation of controls and as a relaxation of controls
8 would have no effect on the availability of highly skilled
9 workers, the plant safety problems that you are envisioning
10 I think would be relaxed rather than exacerbated.
11 MR. REED: I didn't follow you at all. I talked
12 about two different things. I talked about quality and the
13 dose, accumulating dose. That's one issue. ALARA is a
14 second issue.
15 ALARA will affect the timeliness, on a mandatory
16 basis, the timeliness of repair. I didn't say dose --
17 accumulated dose or anything like that. It will affect the
18 timeliness of repair and will slow down the process, when
19 it becomes mandatory, because of the threat of fine which
20 utilities take very seriously.
21 MR. ALEXANDER: This is just my opinion. I
22 think what would happen is if a plant has, say, a limit of,
23 say, 10 man-rems for a particular operation on a pump or
24 something like that in their book --
25 MR. REED: You are not getting my point on ALARA.

1 ALARA means that you must really plan to make the exposure
2 for that piece of work -- plan in advance and so on and so
3 forth -- make the exposure for that piece, that work, as
4 low as reasonably achieveable. That is what it means.
5 What I'm saying is if you make this mandatory
6 and citable and fineable, then you will slow down the work
7 because the planning must be better. It will be subject to
8 review and study. The threat is there and therefore the
9 timeliness of achievement will slow down.
10 MR. ALEXANDER: I think I understand exactly
11 what you are saying, but I don't agree. And let me say why.
12 Consider this: Plants already have, in general,
13 occupational ALARA programs in effect. I feel that as a
14 result of this regulation that those plans will not be
15 tightened. If they are not tightened, then work in the
16 plant will continue just as it has in the past, without
17 change.
18 If any changes are made in those plans it will
19 be to relax them. If they are relaxed in order to avoid
20 citations from an inspector -- you see, if they are relaxed
21 in order to avoid citations from an inspector, any effects
22 on work going on on the floor of the plant would be -- the
23 restrictions would be removed, not tightened.
24 MR. REED: We are not communicating.
25 MR. SHEWMON: May I suggest this is all settled

1 and clear now and go on to the next topic?

2 MR. MOELLER: We'd better move along, if we can.

3 I know we are interrupting you, Bob, but you still have

4 Alan to speak.

5 MR. ALEXANDER: I think the exchange is good.

6 It will help -- one thing like these questions that you are

7 raising will help us handle the public comments.

8 The second safety improvement that I see --

9 MR. WARD: Carson was waiting in line with a

10 comment.

11 MR. MARK: No, it's merely a very simple

12 question. I am sure you get data from the UK, France,

13 Germany, Japan, you mentioned the Iron Curtain. Do you get

14 any credible data from Russia?

15 MR. ALEXANDER: Nothing.

16 MR. MARK: It would perhaps be a question you

17 might wonder about. They are certainly members of the ICRP,

18 and it would be a kind of fascinating bit of information,

19 were it possible to smoke out.

20 MR. ALEXANDER: At the ACRS recommendation, at

21 least of Dr. Moeller's subcommittee we contacted with

22 Brookhaven National Laboratory to get all of the nuclear

23 power plant occupational dose data they could. A very

24 extensive, comprehensive effort was made which has been

25 published and I have provided copies of that to Dr. Moeller

1 which indicate the occupational experience at a large
2 number of countries, but we were able to get nothing from
3 Russia.
4 For the first time we are proposing a limit to
5 protect the embryo fetus, with informed consent. We have a
6 regulatory guide out for the past 10 or 11 years, which
7 tells -- we have the operator in his training program, and
8 copies of an appendix to the regulatory guide, male and
9 female, explaining the risks as we understand them to the
10 embryo and fetus, and then it's left up to the mother to
11 make her own decision.
12 One of the reasons we take that approach is
13 because there's conflicting legislation in that area. Most
14 of the steps that can be taken to protect the embryo and
15 fetus involve either invasion of privacy or equal
16 employment opportunity or both. So the informed consent
17 has been the position we've taken in the past.
18 The new guidance for the President's signature
19 in the area of occupational radiation protection has been
20 developed by the effective agencies under EPA leadership
21 and has been approved by all of these agencies and has been
22 submitted to the Office of Management and Budget for their
23 comments and approval. It's expected to go to the
24 President for signature within a month.
25 A 500 millirem limit for the embryo fetus is

1 specified in that guidance. This would be to a woman who
2 declares herself to be pregnant. And our regulation is
3 tracking this recommendation in the new guidance to federal
4 agencies.
5 The way it will work is if a woman goes to her
6 employer and says: I'm pregnant; then, at that moment the
7 dose limit for the embryo fetus that she is carrying will
8 go into effect and it will be 500 millirems for the entire
9 gestation period.
10 If she does not declare the pregnancy and on her
11 own volition decides to accept the risk to the embryo fetus,
12 that will continue to be her choice, as in the past.
13 MR. KERR: What about a situation in which she
14 discovers later on that she should have done this and says
15 she was not properly informed about the risks and
16 consequences? Is that dealt with in the regulations at all?
17 MR. ALEXANDER: We have a regulatory guide which
18 provides information on this subject. Commissioner
19 Bernthal, during our briefing to the Commission on the
20 subject, raised the question that you are mentioning and
21 told the staff that we should consider making mandatory a
22 presentation to the worker of this risk information, and
23 that is under consideration. It will be resolved by the
24 staff following the public comment period.
25 MR. KERR: The fact that you make a presentation

1 mandatory doesn't mean that the person receiving it
2 understands it. Are you going to make understanding
3 mandatory?
4 MR. ALEXANDER: Can't do that.
5 MR. KERR: You could write into the regulation
6 something that says that the presentation is prima facie
7 evidence, or something or other, of understanding. Perhaps
8 that doesn't have any legal significance --
9 MR. SIESS: Are you speaking as a teacher or a
10 lawyer?
11 MR. KERR: I'm speaking as a lawyer.
12 I would foresee, given this new legislation,
13 people who might later claim, particularly if they have
14 children that are born with some sort of genetic defect,
15 who would claim that sure they knew about this, they just
16 didn't understand the implications, otherwise they would
17 have declared themselves pregnant.
18 MR. ALEXANDER: I guess in response to that line
19 of questioning --
20 MR. SIESS: Win a $1 million lawsuit.
21 MR. ALEXANDER: In making decisions about
22 imposing regulation protection decisions on our licensees,
23 we do not take into consideration the radiation liability
24 question. That is considered to be a problem, if any,
25 between the employer and the worker and we always make our

1 decisions based on control of the risk in the workplace.

2 MR. REED: The limit, you said, was half a rem?

3 MR. ALEXANDER: Yes.

4 MR. REED: Which is what you are trying to

5 achieve for the average worker in the workplace. So, what

6 else is new? It's the same thing. Isn't that what you

7 just said earlier?

8 MR. ALEXANDER: The difference would be that,

9 whereas we would like to achieve that average among the

10 workers -- average -- there are two differences. One is .5

11 for workers in general is an average, it would be -- the

12 distribution of doses includes far below and far above 5;

13 and that is the desire of the staff. What we are talking

14 about here is the regulatory limit so that we are not

15 talking about an average and we are not talking about an

16 excitation, but we are talking about a limit so that it

17 would be violated if an embryo got more than that dose.

18 MR. REED: It seems to me that half a rem is set

19 fairly high, based on the rate of uncertainties,

20 possibilities of litigation and all sorts of things. The

21 other question I asked, does the NRC open itself now up to

22 litigation in case a child -- what, one in 10 births have

23 some sort of problem? Does the NRC now -- let's say a

24 licensee goes by the book. Does the NRC become a

25 co-litigant?

1 MR. ALEXANDER: I don't know. I'm concerned
2 about that. I have raised that question with ELD. I do
3 not have an answer at this time. But that's a very
4 critical point.
5 Let me say this, in answer to that question. If
6 the answer should be yes, so that protection of the
7 government against radiation liability charges should be
8 considered in our regulations, then there would be drastic
9 changes in those regulations. Those regulations are
10 designed to control hazards in the workplace, and many
11 changes would be required to protect this agency against
12 legal charges.
13 MR. KERR: I thought I heard you just say about
14 five minutes ago that what you are really interested in was
15 protection of the individuals. You weren't concerned about
16 litigation.
17 Now I'm hearing that if litigation is something
18 that might damage the government, then you become less
19 concerned -- have a lower priority on worker protection,
20 perhaps, with some concern about litigation.
21 MR. ALEXANDER: What I'm saying is if we were
22 required to protect the government against litigation, that
23 radiation liability considerations would impose more
24 restrictive limits than those necessary for control.
25 Your comment that 500 millirems sounds pretty

1 high, I think, will be shared by many of our commenters.

2 A 1 rem exposure, or twice what we are proposing

3 to impose, could result in a .6 per thousand risk of

4 childhood cancer and 4 per thousand risk of severe

5 retardation. I think that some people would consider that

6 to be high. But the 500 is an NCRP recommendation. The

7 ICRP recommends a number more than twice that high,

8 something above 1 rem for the gestation period. The NCRP,

9 here in our country, recommends 500 millirems, and that's

10 the number that is being adopted.

11 MR. REED: Well, I have a feeling that any

12 utility, responsible utility will certainly not want to

13 flirt with 500. So I'm surprised it's that --

14 MR. ALEXANDER: That, of course, is where the

15 job discrimination enters the picture.

16 MR. CARBON: Dr. Alexander, is this 500 millirem

17 in line with the fact that we're pretty sure they have not

18 yet found any genetic defects from Hiroshima or Nagasaki --

19 does this fit together with that?

20 MR. ALEXANDER: Childhood cancer, and the

21 microcephaly, the small head size and severe retardation,

22 are the controlling factors for the limit for the embryo

23 fetus. Since no genetic effects have been observed among

24 the survivors in Japan, genetic effects don't really

25 influence the 500 millirems number.

1 MR. WARD: Well, let's see, there are no genetic
2 effects observed but there are -- I don't know if I'm using
3 the right terms -- there are congenital effects observed in
4 the Japanese victims; isn't that right? Fetus damage was
5 clinically observed?
6 MR. ALEXANDER: Yes. That's where the 4 in 1000
7 severe retardation per rem comes from, from the Japanese
8 survivors.
9 But what I'm saying is, from observations --
10 that is the effect of irradiation in the womb at the time
11 the bombs were exploded. None of those effects are
12 attributed to or have been found among children who were
13 conceived after the explosion.
14 MR. WARD: So those were the number -- .6 and .4
15 were associated with an acute dose of 1 rem, not with a
16 nine-month dose of 1 rem; is that right?
17 MR. ALEXANDER: I think primarily an acute dose.
18 MR. WARD: Yes.
19 MR. ALEXANDER: The third one I want to mention,
20 and I personally feel very strongly about this one, is that
21 the new part 20 would reduce intake limits for alpha
22 emitters, some by large factors.
23 For example, uranium oxide, which is the form
24 which uranium is worked with in the preparation, primarily
25 in our fuel fabrication plants; that limit would be reduced

1 by a factor of 5, in compliance with new ICRP regulations.
2 The meaning there is that the ICRP feels, since
3 what you are trying to protect from in your limits for
4 airborne uranium and for other alpha emitters is cancer, so
5 the ICRP obviously feels that these compounds are a factor
6 of 5 more carcinogenic than was previously thought back in
7 the late '50s, when those limits were originally
8 established.
9 So I think that it is necessary to reduce the
10 intake limits for these alpha emitters and the new part 20
11 will do that.
12 MR. MARK: It used to be said that the limit was
13 not radiological at all but was chemical poisoning.
14 MR. ALEXANDER: That's true, Dr. Mark, that's
15 why I was careful to talk about uranium oxides, which are
16 not considered to be sufficiently soluble so that kidney
17 damage is the controlling factor.
18 In uranium mills, where they work with yellow
19 cake, and people who are exposed, as in the recent accident,
20 to UO2F2 when UF6 gas is released and hydrolyzed, where
21 retention in the lung is very short, chemical toxicity to
22 the kidney becomes a controlling factor.
23 In that connection I would like to mention that
24 the NRC is very carefully investigating recommendations
25 from the University of Rochester that the intake limits for

1 soluble compounds also be reduced by a factor of 5.

2 These factors are very large factors for someone

3 trying to run a mill or fuel fabrication plant or a

4 conversion plant. These could have serious economic

5 effects. But we feel that, based on evidence that we have

6 about the efficiency of alpha radioactivity in causing

7 cancer, based on the fact that these oxides are now known

8 to remain in the lung much longer than was assumed when the

9 original limits were promulgated, and based on the fact

10 that some of the transfer fractions from one compartment of

11 the body to another are now known to be different, that

12 reducing these limits is justified and necessary.

13 MR. KERR: How close are we getting to the alpha

14 dose to which one would be exposed in a house that had a

15 fairly high radon concentration?

16 MR. ALEXANDER: I believe you are asking about

17 the exposure levels workers endure now?

18 MR. KERR: Yes. Compared to exposure in a house

19 that had a comparatively high radon level, say? I'm trying

20 to get some idea about how much above background we are

21 going.

22 There is a radon background with which one has

23 to deal in most houses and it varies. Are we getting

24 anywhere near that background, in the new rules?

25 MR. ALEXANDER: We think that the maximum

1 routine exposures to uranium --

2 MR. KERR: No, I'm talking about alpha emitters,

3 not uranium per se. In this case, radon.

4 MR. ALEXANDER: The highest exposures we have in

5 our licensed operators is at the uranium fuel plants and

6 uranium mills, from the information we have, that probably

7 conditions very, very rarely, if ever, exceed 25 percent of

8 the present limits.

9 Now, 25 percent of the present limits --

10 MR. KERR: If we go down by a factor of 5, then

11 we are below chronic exposure?

12 MR. ALEXANDER: It should have some effect on

13 some of the licensees. Those that are already below a

14 factor of 5, that will already be below our limits, will

15 not have the cushion that they like to operate with. I

16 think they'll take action also.

17 MR. KERR: So we are in effect reducing the

18 exposures below what one might see in one's household

19 because of radon?

20 MR. ALEXANDER: Radon, I was just looking at

21 that recently -- I'm not an expert on radon exposures, but

22 based on information that has been published, I believe

23 that in the average home, from what we know about the

24 average home, that the risk of lung cancer is on the order

25 of 1 to 2 times 10 to the minus 4, lifetime -- not per year,

1 but lifetime; that's for the average -- where the average
2 concentration is about 1.5 picocuries per liter in the home.
3 And we know in many homes radon levels are far above that,
4 up to 2000 picocuries. And in the homes we have, the risk
5 would be up to 1 third, in getting lung cancer in a home
6 like that.
7 For some homes the worker is better off in the
8 workplace than he would be at home.
9 MR. KERR: I'm trying to get an idea -- and I
10 guess I'm not making my question clear. Given the reduced
11 level, what's the relationship now between the risk of
12 cancer from radon exposure compared to the risk of cancer
13 using the new levels, if one were operating right up to the
14 limit? Are they comparable? Is the radon exposure still
15 down by a factor of a lot? I don't need exact numbers.
16 I'm just trying to get some general feeling.
17 MR. ALEXANDER: I think I can -- I'll try again
18 from some notes I have in my briefcase, to answer that
19 question.
20 You are asking me to compare the risk for a
21 uranium worker to --
22 MR. KERR: Who is exposed to the maximum under
23 the new rule, to, say, an exposure that one would get in an
24 average household, to radon.
25 MR. ALEXANDER: Okay. I can do that. With

1 certain assumptions, as health facilities are prone to make,
2 the risk from compliance with our radon limit, if a person
3 were to actually be exposed for a working lifetime -- I
4 used 40 years -- and came out with about 2 percent for lung
5 cancer incidence. That's the full working level of 12
6 months per year.
7 MR. MARK: I wonder if you could say that
8 differently. The radon exposure is in the range of one or
9 two picocuries per liter. That's what happens.
10 MR. ALEXANDER: The average, yes.
11 MR. MARK: What is your limit for alpha activity
12 under the new regulations?
13 MR. ALEXANDER: 30 picocuries per liter is the
14 concentration.
15 MR. MARK: 30.
16 MR. ALEXANDER: So, to complete my answer, then,
17 2 percent for lifetime lung cancer incidence for a person
18 exposed at the maximum for 40 years that part 20 would
19 allow; and for lifetime exposure, for living in a home at
20 the average radon concentration in an American home, is
21 about 1.5 times 10 to the minus 4, or about .02 percent.
22 MR. SIESS: A factor of 10.
23 MR. ALEXANDER: So I guess there's two orders of
24 magnitude exactly, difference, if you accept my estimates.
25 I'm sure glad I did this on the subway this morning.

1 MR. KERR: Thank you.

2 I want to mention one more reduction, that is
3 for the extremities, the hands and feet. I think
4 particularly the hands, because these workers have their
5 hands right on contaminated materials.

6 That limit is being reduced by one-third as a
7 result of recommendations from the ICRP. We have been
8 allowing 75. We have been allowing 75 rems per year. That
9 has been reduced to 50 rems per year.

10 So, those are the safety features of part 20
11 that I consider to be important. Now, I don't know how
12 much time we have left for Dr. Brodsky's presentation of
13 details.

14 MR. MOELLER: What we have agreed to do, Bob, is
15 to continue this discussion because of the obvious interest
16 on the part of the committee -- continue this until 10:30.
17 But that includes Alan's presentation. We must terminate
18 at that time. So we have about 28 more minutes.

19 Dr. Carbon has a question.

20 MR. CARBON: Yes. When will we come to the
21 point of the discussion of whether these four safety
22 features could be incorporated in the existing part 20, or
23 whether we need a new part 20 and a new rulemaking?

24 MR. ALEXANDER: The answer is obvious. We could
25 incorporate these safety features that I mentioned with a

1 number of changes to the existing part 20.

2 MR. CARBON: Will you be discussing, then, one

3 of you, the need for the modified part 20 in the new

4 rulemaking?

5 MR. ALEXANDER: Well, I should try to answer

6 that. I think basically you are asking: Why did the staff

7 decide to start from scratch and rewrite part 20 as opposed

8 to just making those changes necessary to comply with the

9 new ICRP recommendations and the new guidance to the

10 federal agencies of the president.

11 MR. CARBON: Yes. And the fact that rewriting

12 and the new rulemaking and so on, if I understand correctly,

13 your figures indicate are going to cost something like $30

14 million, $35 million, primarily to the nuclear power

15 industry?

16 MR. ALEXANDER: Yes. The cost we estimated was

17 $33 million for the first year and then $8 million per year

18 thereafter for compliance with this regulation.

19 The reason the complete overhaul was undertaken

20 was because of, in addition to resolving these rather major

21 problems I have mentioned, there are just hundreds of

22 little things that we have learned over the past 25 or 30

23 years that we wanted to attend to.

24 If you do attend to all of those you do have a

25 major rewrite on your hands. If you take that approach, a

1 comprehansive fix of part 20 rather than a piecemeal
2 approach -- it just wouldn't turn out to be a piecemeal
3 approach. You'd have the complete revision anyhow.
4 But it is possible, of course, to overlook many
5 of the problems that we consider to be significant enough
6 to change but that we could live with without changing.
7 That's why I say that it is feasible to do a patch on part
8 20 and just take care of the safety issues and perhaps one
9 or two others that do appear in the federal guidance, which
10 is the Commission's policy, to comply with the federal
11 guidance. That is a question of policy and we'll be very
12 sensitive to the public comments to see if there is a
13 demand for that approach, and I think we'll be very
14 open-minded about that if there is a widespread desire to
15 go about it that way.
16 MR. CARBON: But to achieve the safety that you
17 feel needed, these four points then could be put in the
18 current part 20, at essentially no cost, I gather? And the
19 new part 20 and so on will involve rulemaking and the added
20 things will not really contribute to safety as such?
21 MR. ALEXANDER: Well, not nearly as much.
22 Certainly not nearly as much.
23 If you took out the four items I have mentioned
24 it would be difficult to make a safety argument at all for
25 part 20. But I believe you said that, if I heard you right,

1 you were indicating that most of the cost would be
2 associated with things other than the four safety items? I
3 don't think that's true.
4 MR. CARBON: Most of the cost would be
5 associated with the new part 20 and the plans to comply
6 with it? But, if I understand correctly, your figures
7 indicate that to accommodate the new part 20 is going to
8 cost the $33 million figure, whereas if you go at it in the
9 direction of simply putting these four changes into the
10 current part 20, that particular $33 million would not be
11 expended; is that correct?
12 MR. ALEXANDER: No. I think you'd shave off
13 some of that $33 million, but I believe most of it might
14 still be there. The reason I say that is, according to my
15 information, the two big items of cost here are changing
16 computer software in the control systems at the plants, as
17 set up. It turns out to be very expensive.
18 When we come in with new limits they say that
19 they have to change their computer software and a large
20 amount of the cost comes there.
21 Other costs are going to be, high costs are
22 going to be associated, I think, with the reduced intake
23 limits for alpha emitters. Those changes are still here.
24 So, I'm sure that the $33 million would come
25 down some but it probably would not come down enough to

1 alter conclusions that a person might arrive at about the
2 cost effectiveness of the regulations.
3 MR. CARBON: I'm not sure how to interpret your
4 last sentence. Are you saying that if we put these four
5 changes into the current part 20, that perhaps we would
6 encounter a cost of $20 million instead of $33 million?
7 Just to get some kind of --
8 MR. ALEXANDER: Yes.
9 MR. CARBON: Because this is quite different
10 from what I understood at our subcommittee meeting.
11 MR. ALEXANDER: Is it? Is that the things I
12 said at the subcommittee meeting?
13 MR. CARBON: I don't know. But I came from the
14 subcommittee meeting with the definite understanding that
15 to go through the new rulemaking, the new part 20, and all
16 the changes associated with that, would cost $33 million
17 more than simply putting these four changes into the
18 present part 20.
19 MR. ALEXANDER: If that impression was left I
20 have to apologize. The $33 million, and $8 million per
21 year are based on all of the changes that are being
22 recommended for part 20.
23 MR. KERR: You said the two big items were
24 software -- what was the other one?
25 MR. ALEXANDER: Reduction in the intake limits

1 for alpha emitters. Those are the two big ones.

2 We have these covered item by item in the

3 regulatory analysis which has been supplied to the

4 committee.

5 MR. CARBON: I don't see that -- perhaps I

6 misunderstood.

7 I would comment to the committee, then, that the

8 notice sent today does not check with what Dr. Alexander is

9 saying at the moment.

10 MR. MOELLER: Bob, one other quick one. If you

11 took the band-aid approach versus the complete rewrite,

12 which you have done, how different is it in terms of public

13 comment and so forth? Can you do the band-aid approach

14 without public comment? You cannot?

15 MR. ALEXANDER: No. Any change we make in part

16 20 would have associated with it a public comment period.

17 MR. MOELLER: Okay. Forest has a question.

18 MR. REMICK: How many total licensees would be

19 affected by a change in part 20?

20 MR. ALEXANDER: All of the licensees of the

21 Commission and eventually all of the licensees of the

22 agreement states would be affected.

23 MR. REMICK: How many are those, roughly?

24 MR. ALEXANDER: You are talking probably 21,000

25 licensees all together.

1 MR. REMICK: That's what makes me question this
2 $33 million. I haven't gone through your impact statement
3 but I would get -- just takes one licensee, and reading the
4 new part 20, just to read that you would probably come
5 pretty close to $20 or $30 million; if you assume every one
6 of those 20- or 21,000 read that draft and decide to make
7 comments, you have consumed somewhere around $20 or $30
8 million. If you take the manpower just to read it,
9 comprehend it, a typical licensee -- the number of people,
10 health personnel, administrative personnel and so forth --
11 the $33 million to me just seems unreasonably low.
12 I'm not against the idea of changing part 20.
13 It just seems inconceivable to me such a small cost for
14 21,000 licensees, and presumably every one of them will be
15 affected in some way.
16 MR. ALEXANDER: They will all be subject to it
17 but many of them won't be affected. Many of them are small
18 operations that wouldn't be affected.
19 MR. KERR: He's assuming they won't read it.
20 MR. REMICK: They are going to have to read it
21 and modify some of the procedures.
22 MR. ALEXANDER: Actually, what we want to do,
23 Dr. Remick, as the IRS, every year, when they change our
24 taxes on us, they put on the cover sheet what important
25 changes are made so you know whether or not you have to

1 read the whole thing or not. We plan to do something like
2 that so that most of our licensees, when they would get
3 this new document, most of them could look at this summary
4 and just set it aside and say: I just have a gauge over
5 here. I'm not affected by this.
6 MR. REMICK: But there's a lot of them if the
7 health physics staff didn't read this I wouldn't think
8 they'd be a very professional health physics staff.
9 MR. ALEXANDER: There are a lot of them that
10 will, but that may measure in the hundreds as opposed to
11 the thousands.
12 MR. MOELLER: Will there be a lot of benefits on
13 education so we should put that on the plus side? It's
14 good for those people to have something new to read and
15 learn.
16 MR. REMICK: I'm not against the concept, I just
17 think it's an underestimation.
18 MR. ALEXANDER: The science of cost estimation
19 is a very interesting one. The staff people that you
20 assign to do these things, after a year or two of working
21 on it, become very much caught up in it and they really
22 want to see that regulation go through. So at the time
23 they do their cost estimates they tend to minimize them.
24 On the other hand, the fellow out in the public
25 who is going to have to implement these and pay for them,

1 when he does his cost analysis, he's very likely to up them
2 a little bit. In fact he's very likely to up them about an
3 order of magnitude.
4 So, somewhere in between the staff's cost and
5 the effective licensee cost is the truth, and actually we
6 never know what that is.
7 MR. CARBON: Does something like this proposed
8 change go through CRGR?
9 MR. ALEXANDER: Oh, yes. You bet.
10 MR. CARBON: Have they approved it as being
11 cost-effective?
12 MR. ALEXANDER: No, sir. The approval of the
13 CRGR -- as a matter of fact, the recommendation of the
14 staff was not justified on a cost/benefit ratio. It was
15 justified on a scientific updating rationale.
16 We feel, the staff really feels, that the places
17 we have licensed are safe places to work in. We do not
18 feel a strong need to revise these regulations to provide
19 additional safety, even though we are not entirely
20 satisfied in every case. But the principal justification
21 for this change is scientific updating. Most regulations
22 are based on concepts that existed prior to 1957 that are
23 just no longer scientifically valid.
24 MR. CARBON: But you do feel that the current
25 situation is safe?

1 MR. ALEXANDER: Yes. For example, in the
2 nuclear power industry, where more than half of the
3 collective dose received by -- in our licensed operations
4 occurs, the 0.65 rems per year is a number that to me would
5 be perfectly acceptable without change. The reason we
6 would like to see it get down to .5 is because of ICRP
7 recommendations.
8 MR. KERR: What is scientific about going from
9 .65 to .5; I don't understand that? Because the ICRP, when
10 it comes to the .5, is not scientific. ICRP is very
11 scientific when it tries to decide what is the appropriate
12 biological effect of radiation. But when they make a
13 recommendation on an appropriate level they are using value
14 judgments such as you or I. They may have a better basis
15 for it but it's not based on logic, it's just a judgment
16 call.
17 MR. ALEXANDER: That's very, very true. 10 to
18 the minus 4 criterion is a judgment call, pure and simple.
19 It is not scientifically based; it is a judgment call and
20 everything springs from that, so I have to agree with what
21 you said.
22 MR. KERR: So let's not call it scientific, in
23 the sense that some new science has led to this. It may be
24 a good thing to do, but because a group has gotten it
25 together and made a value judgment that this is what it

1 thinks ought to be.

2 MR. ALEXANDER: Well, health physicists have

3 been accused of always suffering from what is called

4 monomania. Monomania is a mental disease, a disorder in

5 which the victim is only wrong on one point, but that one

6 point is the basis for everything else he does and it's all

7 perfectly logical.

8 MR. MOELLER: Have you covered all your --

9 MR. WARD: I think that's rampant throughout

10 many of our activities.

11 MR. CARBON: One more question. I'm still

12 concerned about the cost effectiveness and so on. The

13 ALARA principle, $1000 per man-rem turns out to be

14 something like $10 million per life, say. That's late in a

15 person's lifetime. And it's generally accepted that that

16 surely is conservative, and maybe it's $30 or $40 million

17 per life -- how does this compare with that? How does this

18 fit in with the ALARA $1000?

19 MR. ALEXANDER: I haven't looked at that

20 specifically, but I think I can give you an answer that

21 will be acceptable to you. I believe, if you were to

22 examine this rule change, particularly on the basis of

23 dollars per man-rem criteria, that is dollars per man-rem

24 saved criterion, by this regulation, that the regulation

25 could not be justified by the number achieved. It would be

1 a very small number -- I mean a very large number and could
2 not be justified on that basis. That's the primary reason
3 why we have not tried to justify it on that basis.
4 The number of man-rems saved would be small.
5 The cost would be large, and a very, very high cost/benefit
6 ratio, and we are not trying to justify our proposal on
7 that basis and I do not believe it could be justified.
8 Thank you very much. I'm sure that no one is as
9 happy about the time we've consumed as Dr. Brodsky, who has
10 the remaining time, but he has some viewgraphs now and can
11 get into the details of what we are proposing. There are
12 only a few minutes left.
13 MR. MOELLER: Excuse me, I do not want this to
14 run past 10:30, so keep that in mind.
15 MR. BRODSKY: It's a pleasure to be speaking to
16 this distinguished group of people. I am particularly
17 happy that I had the opportunity to see Bob Alexander go
18 through an oral examination. I don't think any of my 125
19 graduate students did as well as Bob Alexander in his
20 examination.
21 MR. ALEXANDER: Do I get an honorary degree?
22 MR. BRODSKY: You, I think, should get an
23 honorary degree. Answered every question.
24 In the remaining time available -- as you can
25 see, this is a complex subject; all I can do is give you a

1 broad brush and we'll do that with the help of Bob
2 Alexander's own slides. That way he keeps me on track.
3 First of all, the name of the branch has been
4 changed recently, Health Effects and Occupational Radiation
5 Protections Office, to Radiation Risk Assessment and
6 Management Branch. So we now go back again to cutting
7 across the occupational and environmental borders in the
8 assessment of risk and the management of risk.
9 The reasons for revising part 20 Bob covered in
10 context; the items on here update the present part 20. It
11 was promulgated in '57 and revised finally in '61. There
12 had been an earlier part which was quickly revised because
13 of changes in the NCRP recommendations which brought 15
14 rems down to 5 average, by the 5(N minus 18) formula
15 allowing up to 12 a year.
16 Bob has mentioned we have now proposed to get in
17 line with the OSHA recommendation, 1977 -- and keep the 5
18 rem per year as a limit each year, not allow people to
19 average, but allow people to get up to 3 rem in the quarter.
20 There is actually some comment that we should
21 have done this back in 1961, and we went back and forth
22 with this. We almost made it simpler back in 1961.
23 (Slide.)
24 It's sometimes hard to make things simple.
25 Needed improvements in the present part 20. Well, Bob has

1 mentioned this pretty well already. Thank you, Bob.
2 5(N minus 18) rule, and also we are now providing explicit
3 dose limits for members of the public.
4 One item of confusion -- and we almost purposely
5 confused the public in the 1961 part 20 -- was the
6 admonition that: Thou shalt not either expose anyone to
7 more than 2 millirem in any one hour, or 100 millirem in
8 any consecutive seven days. It was not written in the best
9 English, it should have been written neither nor -- you
10 have to recognize that larger levels for practical purposes
11 in a generic regulation, knowing that people are not going
12 to stand still for any one year right next to your wall.
13 It's not clear still, but we intend that nobody
14 in the public should get more than .5 rem in a year. I
15 think everybody's designs go along with this, we can live
16 with that pretty well. And there's providing the cutoff on
17 the collective dose rems -- I don't think I did mention
18 that. That's an important consideration in this regulation
19 we want to call to your attention, and we want comments on
20 how should we handle this "de minimis" concept, so-called,
21 that we have been talking about for the last five years in
22 terms of de minimis.
23 So far, in this regulation the only thing that
24 has been put in there is a proposed 1 millirem cutoff in
25 the calculation of the population -- collective doses when

1 you are calculating doses -- environmental doses to the
2 populations at large. There are no de minimis cutoff
3 levels in regard to occupational exposure except that --
4 unless you consider some of the action levels in there
5 about which you are required to provide personal monitoring
6 is at 10 percent of the permitted dose limits, and when you
7 are required to do evaluations of your internal doses,
8 perhaps at 30 percent of your limits of intake for one year.
9 I have five more minutes left. I'm going to try
10 to hit what occurs to me as the highlights as we go along.
11 That's about all we can do.
12 (Slide.)
13 This slide basically says -- the main thing in
14 this slide is that we are dealing with the formulation --
15 in the ICRP 26, which makes things complex, and also in
16 ICRP 30, used the concept to calculate the ALIs, the limits
17 to some degree. Basically what we are doing now in the new
18 proposed regulation is adopting the formalism of ICRP 30,
19 insofar as requiring people to add up the external doses,
20 whole body dose and add up the fractional amounts of the
21 annual limits of intake of each radionuclide to which they
22 may be exposed and keep the total fraction less than 1.
23 In cases where external radiation may be
24 nonuniform, to certain parts of the body, there are
25 weighting factors that can be used in the ICRP 26, to

1 calculate an effective whole body dose. And I believe the
2 way our proposed regulations read, one could interpret, now,
3 that kind of weighting to be done in estimating the
4 external effect of exposures, also.
5 So all this makes the calculation of dose a
6 little bit more complicated but, again, logically it
7 relates your limit to both external and internal exposure
8 of the same individual to some estimated risk, estimated by
9 these bodies that have studied the radiobiological
10 literature of the last few decades.
11 Some people think, well, that that is a benefit,
12 a regulation that you can trace back through logic back to
13 some estimate of risk by some independent international
14 bodies of experts. Tough people, like you, but including
15 people in the fields of radiobiology and medicine.
16 MR. MOELLER: I think, Allen, in view of the
17 time, we do have your handout and we -- the committee
18 members can look at that. There was, or there were many
19 members of the committee at the subcommittee meeting. In
20 view of the time, I think that we better wrap it up at this
21 point, and let me do so by thanking both you and Bob for
22 coming down and for giving us your time and the opportunity
23 to interact and to really address what were the major
24 questions and concerns of the committee.
25 Thank you.

1 MR. BRODSKY: Thank you. I think you picked a
2 good point there.
3 MR. MOELLER: That's it, Mr. Chairman.
4 MR. WARD: Thank you very much. Let's take a
5 15-minute break.
6 (Recess.)
7 MR. WARD: Let's go to the report of the Reactor
8 Operations Subcommittee, the Perry Nuclear Power Station.
9 Jesse?
10 MR. EBERSOLE: I'm just going to -- I want to
11 briefly say that the subcommittee heard the presentation
12 substantially the way you heard it today, in its entirety
13 yesterday; and because of the general interest in it we are
14 having it at the full committee meeting.
15 Our impression was that the earthquake at Perry
16 produced no effect whatsoever on the plant except some
17 unexpected acceleration in containment -- essentially, in
18 its final analysis it has been extremely carefully
19 investigated, and in fact nothing has happened to be
20 concerned about.
21 I'm going to just simply turn it over to the
22 participants here. I believe Mr. Stefano is the lead
23 presenter and we'll go through it much the same way we did
24 yesterday, but in far less time.
25 MR. STEFANO: Thank you, gentlemen. My name is

1 John Stefano, I'm the project manager for Perry.

2 MR. WARD: Would you mind coming over here?

3 MR. STEFANO: I'm John Stefano, the project

4 manager for NRC at Perry. We would like to begin this

5 presentation to you by having the vice-president of the

6 nuclear group from Cleveland Electric, Mr. Murray Edelman,

7 start off by summarizing and giving you a briefing as to

8 what their report contains.

9 I assume that all of you gentlemen have a copy

10 of the report; is that true?

11 MR. SHEWMON: What report?

12 MR. WARD: The answer is we don't.

13 MR. STEFANO: You do not. There were 30 copies

14 yesterday left here, and I guess these reports develop feet.

15 We do have extra copies which I can give out right now if

16 you desire, sir.

17 MR. KERR: There were some in brown and some --

18 MR. WARD: Obviously we don't. We need copies.

19 MR. STEFANO: Do you need them now or after the

20 presentation?

21 MR. WARD: Do you have additional copies? We'd

22 be delighted to have them.

23 Who needs one? Raise your hand. Thank you.

24 MR. STEFANO: With your permission, gentlemen, I

25 would like to introduce to you Mr. Murray Edelman,

1 vice-president, nuclear group, Cleveland Electric, and we
2 will brief you on our action when he is finished.
3 MR. EDELMAN: Good morning, my name is Murray
4 Edelman; I'm vice-president of the nuclear group for the
5 Cleveland Electric Company in charge of the Perry plant.
6 What I would like to do this morning is present
7 an overview and introduction of the technical presentation
8 we made to the NRC staff and their consultants at the Perry
9 site on the 11th, a slightly abbreviated version that we
10 gave to the subcommittee last night.
11 (Slide.)
12 What I have on the board is an agenda. I gave
13 the introduction and overview; we had our plant supervisor
14 in charge of the plant review status. We had Dick Holt of
15 Western Geophysics talk about the geology of the plant, and
16 Dr. Chen talk about the seismic design.
17 For brevity I will cover the first three items
18 and then turn it over to Dr. Chen to talk about the seismic
19 design.
20 The event took place at Perry on January 31, at
21 about 11:45, registered about 4.96 on the Richter scale.
22 Even though we are not an operating nuclear plant, although
23 we are 10 days away from fuel load, we implemented our
24 emergency procedures and activated all centers, Technical
25 Support Center, notified the state, county and NRC and

1 evacuated the prime area of the site, specifically for
2 safety, to account for all the people who were in the plant.
3 It demonstrated that our emergency plan and procedures work;
4 everyone was accounted for. And we then did a plant
5 walkdown with our operating people, allowing us after
6 several hours to downgrade the emergency and eventually,
7 with the consent of the NRC and the state, terminated the
8 emergency, at which point in time we went into the recovery
9 mode.
10 During -- following, several hours later, we did
11 numerous plant walkdowns with a number of our people and
12 determined there was no damage to the plant. We
13 immediately called in our consultant team, composed of
14 recognized experts in the areas of geology and seismology
15 from Gilbert Commonwealth Geophysics, the two instrument
16 people that provide our instrumentation to the plant,
17 Dr. Hall, Dr. Stevenson and a number of other consultants.
18 We also did close examination of the nearby field
19 conditions to compare them to our PSAR.
20 We did note that during the recording of our
21 instruments we had high frequencies at the -- high
22 acceleration and high frequencies, which Dr. Chen will
23 cover, but this high frequency acceleration we considered
24 after extensive engineering review as a nonengineering
25 problem because they are low energy, very short duration

1 and low velocity. We are also aware that the phenomena of
2 high frequency accelerations is a generic subject that the
3 NRC staff has addressed at other sites.
4 We also analyzed and reviewed the plant systems,
5 structures and equipment, considering the conservatism
6 built into this equipment and determined that there was no
7 damage, as well as physical inspection to the equipment.
8 At the time of the event we had a number of our
9 safety systems in operation.
10 (Slide.)
11 We were ongoing testing with diesel generators;
12 a number of our systems were energized and were in the
13 operational mode. We were preparing that afternoon to run
14 one of our response time tests in our diesel generators.
15 The start-up sources were the upper pole and they were not
16 moved, and we had detailed reports on a number of systems
17 energized. All these systems worked before, during and
18 after the earthquake.
19 We saw no change in the status of our control
20 room or our control room paper, annunciators didn't come on,
21 relays didn't trip. We looked over all -- about 47,000
22 relays were energized and none of the safety systems did
23 trip. Two items did, our instrument air compressor trip,
24 which was a nonsafety system, on vibration; we lost the
25 heating boiler for the plant, tripped, and it came back on.

1 And those are the only two actual items that tripped. Yes,
2 sir?
3 MR. SHEWMON: There is an older scale, I think,
4 called a Mercalli, which talks about did old chimneys fall
5 down, did things fall off tables, did people realize what
6 was going on because they could feel something? Would you
7 say something of what people noticed?
8 MR. EDELMAN: Yes, I'll talk about that in a
9 little later slide. We did record as a Mercalli 6,
10 compared to our design of 7, but I'll get to that later.
11 (Slide.)
12 After we terminated the emergency we went into
13 our procedures -- into the recovery organization, where I
14 become the recovery manager. We did additional walkdowns
15 with 65 of our engineers and technicians that Friday night
16 and walked through the plant, identified anything that was
17 not right including any chipped paint, broken light bulbs,
18 anything in the plant that may or may not have been caused
19 because of the earthquake, and went back and analyzed all
20 those things. We did additional site surveys. We normally
21 take survey settlement measurements once a month. We went
22 out and did it once again then, saw no differences there.
23 We looked at our cooling tower, there were no problems.
24 There were still about 20 items we were clearing
25 up, what we called seismic clearance violations the plant

1 identified in the final walkthrough. We went back and
2 looked at all those and nothing moved, changed or was
3 impacted by those. We studied all our energized electrical
4 equipment and none of that was a problem.
5 The NRC staff arrived in several days. First
6 they came on Saturday with an augmented inspection team
7 from Region 3, and a confirmatory action letter which we
8 agreed to Friday night. They came and inspected the plant
9 in detail Saturday and Sunday. Based on their inspections
10 we were released by Monday to go back and continue our
11 testing activities. The NRC then sent additional teams in
12 from both Washington and Chicago to look at equipment
13 qualifications, walkdown systems, and checked it; and based
14 on their preliminary results, again confirmed our results:
15 there was no damage to any plant systems or structures in
16 the plant.
17 (Slide.)
18 What I have on the board now are -- on the slide
19 now is the USGS recordings for the actual earthquake: time,
20 11:47: where it was located; how deep based on 64 stations
21 worldwide; it was about a 4.96 on the Richter scale;
22 located at Geauga County, about 11 miles from the plant
23 site.
24 (Slide.)
25 This is a map of the plant site on top with a

1 5-mile radius and a 10-mile radius. You can see the
2 epicenter occurred here, magnitude of about 4.9. There
3 have been several aftershocks recorded with our
4 instrumentations. We had EPRI, a number of people come out
5 and put instrumentations out that same day in the field.
6 They have been recorded about 3 miles west of the epicenter.
7 The highest recording was about 2.4, and we have not
8 recorded any indications of the aftershocks at any of our
9 plant instrumentation.
10 (Slide.)
11 Now I'm going to sound like a geologist but I'm
12 really not, I'm a mechanical engineer. The Perry site is
13 located on the undifferentiated Paleozoic sedimentary rocks,
14 which I know as shale. It's about two kilometers deep.
15 Below that is the Precambrian rock. This is kind of a
16 skewed slide but shows the focus of the epicenter, about
17 30,000 feet below surface. The initial PSAR found glacial
18 anomalies in the site where we had foldings of the glacial
19 scale which, when we excavated, got back to horizontal
20 bedding planes. That was fully investigated. This, based
21 on our analysis, did not break the surface, it occurred in
22 the Paleozoic rock.
23 The conclusions of our geologists and
24 seismologists, which -- I have to say I'm not one of those
25 -- the tectonic province approach we used in our PSAR and

1 FSAR stage is still valid. There is no fault and there is
2 no tectonic structure. Our safe shutdown earthquake, in
3 response to your question, is a Mercalli 7. What the
4 geologists and seismologists are calling this in terms of
5 what they have seen is a 6 at the most.
6 We have a site-specific spectrum of 5 plus or
7 minus a half on the Richter scale versus the 4.96 we saw on
8 this earthquake. The staff also asked us to use a
9 site-specific earthquake of 5.5 plus or minus a third on
10 the Richter scale for our analysis of our plant.
11 We did have exceedance in one area that Dr. Chen
12 will cover in detail, short duration, high frequency of
13 about 20 hertz. It was above the designed spectrum out of
14 Reg Guide 160, of the 84 percent smoothing curve, and
15 Dr. Chen will discuss that in detail with you now.
16 (Slide.)
17 I would like to turn the meeting over to
18 Dr. Chen and I will come up at the end of his presentation
19 and present our brief overview of conclusions which are
20 presented in our detailed report.
21 DR. CHEN: Good morning, my name is Chang Chen.
22 I'm the manager of surface -- chief structural engineer of
23 Gilbert Commonwealth, Inc.
24 Before we go into the detailed comparison of the
25 recorded event versus the design, I would like to summarize

1 the nature of the recorded event, in comparison with the
2 design earthquake. This recorded 1986 Ohio earthquake has
3 short duration, high frequencies and low velocity, low
4 displacement and low energy.
5 (Slide.)
6 The design earthquake we used for the Perry
7 power plant design has broad band frequency content, high
8 velocity, high displacement, and high energy.
9 (Slide.)
10 Now, I show you how did we reach this conclusion.
11 The way we did it was to compare the recorded time history
12 versus the time history we used in design.
13 The first comparison is in the north-south
14 direction of the reactor building, foundation.
15 (Slide.)
16 The time history at the top is the one we used
17 in the design. The one at the bottom is recorded. Same
18 elevation, same direction.
19 As you can see, the design time history is about
20 22 seconds. Yes?
21 MR. SHEWMON: Did you say this is the motion of
22 the mat itself?
23 DR. CHEN: Yes, sir. The record is on the top
24 of the mat so what we compare with here is also the
25 calculated response at the top of the mat; so it's a 1:1

1 correspondence.

2 The design event has a strong-motion phase of

3 about 22 seconds. The recorded event at the bottom, with

4 the strong-motion part of the record, less than 1 second

5 duration. Also, you can see the one at the top has

6 multiple frequency content and the one at the bottom has

7 high frequency content, especially around two hertz.

8 A similar comparison, I will just go through

9 them very quickly --

10 (Slide.)

11 This is in the east-west direction of the

12 foundation mat. The recorded event is much -- not only

13 shorter but also the acceleration is much slower.

14 (Slide.)

15 The next is a comparison in the vertical

16 direction, with the same conclusion. Vertical direction of

17 the foundation mat.

18 (Slide.)

19 The next is at the upper elevation of the

20 containment vessel, elevation 686, in the north south

21 direction.

22 (Slide.)

23 This one also shows some similarity, except with

24 slightly higher acceleration here. And that is in the 20

25 hertz region, and we will come back to a discussion of that

1 later.

2 (Slide.)

3 The next one is still at the same elevation, the
4 east-west direction. In this direction the design time
5 history, acceleration, exceeds the recorded acceleration.

6 MR. SHEWMON: Sir?

7 DR. CHEN: Yes, sir?

8 MR. SHEWMON: Is it to be expected that if the
9 epicenter is south of the plant, that the main acceleration
10 will be in the north-south direction?

11 DR. CHEN: That is correct. But it also depends
12 on the nature of the focal mechanism. What we recorded on
13 the site is really a combination of three parameters. One
14 is the source mechanism, another one is traveling path, and
15 the third one is the local soil condition. So, as to
16 whether north-south should be stronger than east-west or
17 not depends on the focal mechanism. But based on the
18 record here, we do derive the kind of conclusion that
19 north-south is stronger than east-west.

20 MR. SHEWMON: Okay.

21 DR. CHEN: This one is the same elevation on the
22 containment vessel in the vertical direction.

23 (Slide.)

24 The acceleration is similar but the duration is
25 much shorter.

1 Just because of this comparison we reached a
2 conclusion, as we showed earlier, that this is a short
3 duration, high frequency, low displacement and small
4 displacement, small velocity earthquake.
5 (Slide.)
6 We'll show you the Perry design basis. We used
7 broad band frequency response spectra, which correspond to
8 a smoothed, 84 percent percentile, typical -- and the
9 composite time history we use has long duration and high
10 energy.
11 (Slide.)
12 The next one, everybody has seen this 100 times.
13 I'll just show you briefly, the 160 spectrum.
14 (Slide.)
15 The next one will show you briefly the location
16 of those instruments, the types of the instruments.
17 (Slide.)
18 The two at the top are the Kinemetrics, SMF-3,
19 triaxial time/history recorder. They are located at a
20 foundation mat at the reactor building and elevation 686 of
21 the containment vessel.
22 These two instruments recorded those time
23 histories which we just compared it with earlier, and the
24 three at the bottom are the Engdahl instruments, which
25 record the peak acceleration.

1 The fourth one from the top was in the middle of
2 calibration, that's why there was no record.
3 We have four more instruments on the next
4 viewgraph.
5 (Slide.)
6 They are also Engdahl instruments, that record
7 the response spectrum at four different locations. Each
8 response spectra recorded two discrete frequency points.
9 (Slide.)
10 The next one shows you briefly the location of
11 those instruments in the plane view. They are in the
12 reactor building and in the aux building.
13 (Slide.)
14 The next one shows you the location of the
15 elevation. This -- those six time/history comparisons
16 which we just saw were from this one and this one, one at
17 the foundation mat and the other one at an elevation 686 of
18 the containment vessel. The rest were just the Engdahl
19 recorder.
20 (Slide.)
21 Now we make some detailed, ZPA comparisons.
22 That's zero period acceleration.
23 (Slide.)
24 The summary of the comparison is as follows:
25 The recorded ZPA value varied from far below OBE values to

1 74 percent of SSE values, except at containment vessel
2 elevation 686, also only in the north-south direction, not
3 even east-west or vertical direction.
4 But at this location, this high acceleration, at
5 20 hertz, corresponding to very small relative displacement,
6 and correspondingly with low stress. The reason for that
7 was because of high frequency acceleration.
8 (Slide.)
9 Now we see the detailed comparison in the
10 following table. This table has five columns. It
11 indicates the five locations with available ZPA readings.
12 The first column is at the foundation mat of the aux
13 building; second column is at the foundation mat of the
14 reactor building; and the third one is at the top of the
15 recirculation pump. The fourth column is elevation 630,
16 inside the reactor building, and the fifth column is the
17 one which we mentioned earlier, the containment vessel 686.
18 Now we'll look at this horizontally. The first
19 row here is the north/south component; the second row is
20 east-west; the third row is vertical direction.
21 The first one is recorded; the second one is SSE;
22 the third one is OBE.
23 In the design we assumed three components input
24 simultaneously and the result is the so-called square root
25 of the sum of squares. This is the bottom line comparison.

1 If you look at the first column, here the recorded value is
2 somewhat close to OBE value. In the second column, the
3 recorded value is somewhere between SSE and OBE. In the
4 third column, the recorded value is way below OBE value.
5 And the fourth column, the SRSS value is not available
6 because the vertical ZPA could not be determined. But if
7 we look at the individual components in the north-south and
8 east-west, the recorded values are also way below OBE
9 values.
10 Now, let's look at the last column. At
11 elevation 686 of the containment vessel, the SS -- SRSS
12 value in this row is somewhat, I would say, less than 4
13 percent higher than the SSE value. Now let's concentrate
14 on this location. This exceedance is, again, in the
15 north-south direction.
16 (Slide.)
17 Let's concentrate at this location. By looking
18 into the relative displacement, again, it's north-south,
19 east-west, vertical, and SRSS. The first column is the
20 displacement at the foundation level; second column is at
21 elevation 686.
22 Now let's look at the third column, the bottom
23 line numbers we want to look at. It's the relative
24 displacement which is corresponding to the distortion of
25 the shale as well as to induce the stress.

1 If we look at a comparison of the recorded value
2 versus SSE and OBE, the recorded value is corresponding to
3 less than one-half of OBE value.
4 From this -- we think this high acceleration,
5 high frequency is of no engineering significance because of
6 the small displacement.
7 In addition to the ZPA comparison, we also went
8 one step further to compare the response spectra.
9 (Slide.)
10 The conclusion of the comparison is as follows:
11 The Perry design response spectra are far above the
12 recorded spectra in the frequency range below 14 hertz.
13 For certain recorded response spectra, the
14 design spectra was exceeded in the frequency range of 20
15 hertz. However, those exceedances correspond to a small
16 displacement.
17 For example, at the top of the map the maximum
18 calculated -- or from the recorded data -- displacement is
19 then than 3/100 inch. Because of this we considered that
20 as having no engineering significance, and also the
21 recorded velocity spectra show much less energy than our
22 design spectra.
23 MR. OKRENT: Before you take that off, the term
24 "no engineering significance" has no qualifications on it.
25 Have you systematically -- and I mean systematically and

1 comprehensively -- considered each and every item that
2 could bear either directly on safety or bear indirectly,
3 because it would lead to a transient that could cause a
4 later trip, and judged that there are no things of some
5 degree of criticalness, that have some unique electronic
6 characteristics, that might not have been mounted such that
7 this kind of motion could cause jiggling that could cause
8 damage? I'm trying to understand what the -- how I should
9 really interpret your term "no engineering significance."
10 I'm not talking about the containment building, now. You
11 understand?
12 DR. CHEN: Yes, sir. I think I can answer your
13 question in two ways.
14 I suppose you are more concerned about the
15 active components -- concerned about their function. And
16 another one was what does this small displacement mean?
17 Usually when we install equipment like piping
18 systems, there's always a gap somewhere like 1/16 of an
19 inch. When we say "no engineering significance," this kind
20 of displacement was not high enough to close the gap.
21 MR. OKRENT: I doubt that piping is affected.
22 DR. CHEN: Okay.
23 MR. OKRENT: I'm thinking, really, of smaller
24 things that in fact might -- well, for example, let me just
25 mention something I heard Bob Kennedy say on Wednesday of

1 last week. He said that batteries are usually qualified
2 and tested in the laboratory with shims between components,
3 but it is not uncommon for him to visit an actual plant and
4 find that the shims are not there. So the situation could
5 be different than the qualification.
6 DR. CHEN: Yes.
7 MR. OKRENT: I don't know that it's batteries
8 that are of interest, but I'm trying to understand if,
9 instead of your being in a position of trying to assure us
10 there's just no engineering significance, you were given
11 the job by the chairman of the board: Tell me the one
12 thing for which there is most likely to have been
13 engineering significance --
14 DR. CHEN: Okay.
15 MR. OKRENT: -- what would you say?
16 DR. CHEN: Okay. Let me try to answer your
17 question about a battery rack first.
18 Even if Bob Kennedy's observation is on some
19 plant, it doesn't mean it is applicable to Perry. There
20 were so many numerous inspections on Perry by CEI engineers
21 and by us and also on inspections by NRC who didn't see
22 this kind of anomaly.
23 MR. OKRENT: Excuse me. From now on, on
24 batteries, people have heard enough about it. I hope they
25 have gone back and checked.

1 DR. CHEN: This is the first time I heard it.
2 Okay?
3 MR. OKRENT: I don't know what there is that's
4 the equivalent of the battery.
5 DR. CHEN: Well, some of this discussion was
6 discussed last night. I think everybody here more or less
7 agreed for this kind of small displacement, high frequency,
8 high acceleration earthquake, anybody having any concern,
9 it would be most probably with some type of active
10 component. For example, really -- Professor Siess -- he's
11 not here. He just left. For example, what is the nature
12 of the relay and would a relay be sensitive to this type of
13 high frequency, high acceleration relay. Oh, he just came
14 back.
15 The answer to Professor Siess' question last
16 night is: Nobody knows what was the natural frequency of
17 relays. Fortunately this morning we just found some
18 relevant data. It came from Dr. Arnold Leave, NRC staff.
19 He pointed out in the NUREG CR 3558, in the
20 study performed by NRC, there is natural frequency
21 information on relays here. It happened to be page F-16 of
22 this report. It indicates here the natural frequency of
23 the relays to be 5 to 10 hertz.
24 From here we believe we really have no concern
25 here because -- I should have shown you the spectrum

1 comparison, later on, to show you when the natural
2 frequency is from 5 to 10 hertz, you don't really have this
3 type of concern.
4 MR. WARD: Dr. Chen, let me interrupt you for a
5 moment here. I think this is a line of questioning which
6 eventually we may want to pursue, but we really just have a
7 very few more minutes to spend with this.
8 We want you to wrap up. We want to hear again
9 from Mr. Edelman and again from the staff.
10 I want to call the committee's attention to the
11 fact that we do have a request from two Congressmen in Ohio
12 for the ACRS to make an independent review of the seismic
13 data and its implications regarding -- the seismic data
14 from this event and its implications. So, if we are going
15 to react to this request, I think this type of questioning
16 and review is going to be necessary. But we intend in
17 today's presentation for you just to give us a quick
18 picture of the situation.
19 DR. CHEN: Okay. Let me continue.
20 MR. OKRENT: Is there any chance of him
21 answering the question, what would be his most suspect
22 component? If he has an answer.
23 DR. CHEN: Me?
24 MR. OKRENT: Yes.
25 DR. CHEN: Yes. I think my opinion is the same

1 as the majority of the people in this room. If I have any
2 concern at all it's those active components like relays.
3 And fortunately we found some information about the natural
4 frequency of the relays. Okay?
5 MR. WARD: That's good. Thank you. Can you
6 wrap up in just a minute?
7 DR. CHEN: I'll just show you one more slide and
8 I can wrap up.
9 (Slide.)
10 This is the comparison of the spectrum here. As
11 we see, the design spectrum exceeds the recorded spectrum
12 below 14 hertz and the relay frequencies from 5 to 10,
13 which is in this region, which is far exceeded by the
14 design spectrum.
15 For this reason, we still think -- reached the
16 same conclusion, of no engineering significance. The only
17 exceedance is the 20 hertz peak. But if you look at 20
18 hertz here, the displacement value is about .02 inch. In
19 other words, it's 2/100 of an inch. That's why we say it
20 is not significant.
21 With this I would like to make the conclusion
22 here, again, the 1986 Ohio earthquake is short duration,
23 high frequency, small displacement, low energy in
24 comparison with our design time history. This Ohio 1986
25 earthquake has no engineering significance.

1 Thank you.

2 MR. WARD: Thank you very much, Dr. Chen.

3 Mr. Edelman?

4 MR. EDELMAN: I'll see if I can briefly

5 summarize what we have gone through.

6 (Slide.)

7 We are the first utility to run a proof test on

8 our seismic design and we've shown our plant did withstand

9 this earthquake. We have not changed any of our

10 conclusions that were presented in our FSAR, or our PSAR

11 with respect to the magnitude and the energy of the

12 earthquake used in our design. As has been said a number

13 of times, it's a short duration, high frequency earthquake.

14 In addition to that we have a number of other conservatisms

15 built into the overall design.

16 In addition, we at CEI are monitoring this on an

17 industry basis and dealing with EPRI on the seismicity

18 group. They were at the plant. Their team looked at the

19 plant and reviewed it and are reviewing it along with the

20 staff and the NRC. We have provided copies to you and with

21 the staff. I'll turn the meeting over to John Stefano.

22 MR. WARD: Thank you very much. John?

23 MR. STEFANO: Gentlemen, what I'm really going

24 to do right now in a very few brief minutes is to try to

25 give you a feel of what the NRC staff's schedule is for

1 addressing this matter. I'm not going to repeat anything
2 that Mr. Edelman said.
3 I guess the only thing I can say is I personally
4 was at the plant site, I personally walked down the plant
5 along with other members of NRR and also with the regional
6 staff and I can confirm personally that we saw no
7 significant damage to this plant, at least didn't observe
8 anything.
9 In terms of what our actions are, we have just
10 received this report which you have gotten a copy of. We
11 really don't have any feeling as to what that means yet or
12 any position to give you at this point in time. As I said,
13 we are just starting that.
14 Basically what we are going to be doing is
15 characterizing, as we state here, the earthquake, comparing
16 it against the assumptions used in the design. We are also
17 going to be looking at the structural design, the equipment
18 and components as I have outlined them here.
19 In terms of our scheduled actions, we are
20 convening a meeting among the NRR staff and their
21 consultants next Friday, a week from tomorrow, the 21st, at
22 which point in time we should have a pretty good feel as to
23 where we are in terms of our conclusions on the safety of
24 the plant and the significance of this earthquake on the
25 plant design.

1 At the present time, we are hoping, at that
2 point in time, to develop a draft SSER, which we are going
3 to be submitting to the ACRS and are going to be seeking
4 your comments and opinions on that before we finalize that.
5 My schedule at the present time, and everything
6 beyond the 21st, is strictly just a target at this point.
7 I'm not even sure we can meet those.
8 We are looking at the 7th as a final SSER issue
9 date. So we are looking at a firm time frame here --
10 rather, I'm sorry -- a real quick time frame. And we would
11 appreciate your assistance in this regard.
12 We have or are in the process, tomorrow, of
13 responding to the Congressional delegation who have written
14 you and those are specifically Congressmen Eckert and
15 Seiberling from Ohio and also Congressman Markey. Those
16 letters should be hand-carried to the chairman, probably
17 tomorrow morning as our goal.
18 I can tell you that I have gotten some
19 preliminary input back from the staff on reviewing this
20 report that's before you. I know there are going to be
21 many, many questions. One of the questions that has come
22 up is the computer code used in doing this analysis
23 compared to what was used in documenting the analysis in
24 the FSAR. I am told by the utility this morning -- we
25 haven't checked this out -- that they have really updated

1 what they did in the FSAR as a result of an independent
2 design inspection which was done of the structural as-built
3 design; that we had an IE team and consultants go out and
4 do an independent design review of the plant beginning late
5 in '83, concluded around the middle of '84. They have
6 gotten a good write-off on that from that time.
7 The analysis -- the code that they used, that is
8 the code that they used to confirm the as-built design of
9 the plant. We will be getting a copy of that code, I
10 understand, and that will be cleared up as to how they used
11 that, whatever they staff needs in that regard, to complete
12 our assessment.
13 With that, gentlemen, I really don't have
14 anything more to report at this time. Are there any
15 questions?
16 MR. REMICK: What does the "licensing target"
17 mean?
18 MR. STEFANO: Let me explain one thing here.
19 What I mean by a licensing target is issuance of a
20 low-power operating license.
21 Part of what we are going to try to do on the 21st,
22 we recognize there's great interest from a generic
23 standpoint on what happened at Perry and what this report
24 says. One of the things we are going to try to do is
25 separate out those things with the kind of information we

1 need to make an assessment on adequacy of plant design.

2 Assuming all of these things go well and we have

3 no complications -- and at this point in time we don't have

4 a whole mass of questions which is going to hold back on

5 our completing our reviews -- that's the target, I think,

6 as the earliest we can see of licensing this plant.

7 MR. REMICK: They have a license which enabled

8 them to load fuel currently?

9 MR. STEFANO: Yes, sir -- no, no. They have a

10 license to store the fuel and move the fuel to the upper

11 fuel pool, not to load the fuel.

12 This license will operate them to load fuel and

13 operate up to 5 percent. Are there any other questions?

14 MR. WARD: Thank you, Mr. Stefano.

15 MR. STEFANO: Thank you.

16 MR. WARD: Jesse, let me make one comment. We

17 do have the letter from the two Congressmen. I think the

18 ACRS will need to respond to these. The request is really

19 asking us to make a substantial review of the situation.

20 We'll assign that to the Extreme External Phenomena

21 Subcommittee. I think at this meeting we need to write a

22 bread-and-butter letter, at least, back to these

23 Congressmen.

24 Dave, I'd ask you to take a look at the letter

25 so that you can make sure -- I mean if we are going to

1 commit to some sort of schedule or something in the
2 bread-and-butter letter, you need to agree with it.
3 MR. STEFANO: Bob Bernero I think has a comment
4 on that.
5 MR. BERNERO: Can I interject, I'm Bob Bernero,
6 division of Boiling Water Reactor Licensing. I had the
7 benefit of conversation with both of the Congressmen last
8 week. I briefed them on information we had available at
9 the time. And it was coincident that at the same time that
10 they were writing their letter to the committee, I had
11 already spoken to Dr. Okrent and was taking steps to get
12 this on the agenda. And I explained to the Congressmen
13 that, in my view, it has the character of a unique but in
14 some ways routine licensing matter, because we feel
15 compelled to generate a supplementary safety evaluation
16 report. And we would, because of the nature or character
17 of the thing, solicit the ACRS' reaction or advice on that
18 supplementary safety evaluation report, either through the
19 conventional Subcommittee for the Plant License or, because
20 of this one, through the Extreme Natural Phenomena.
21 I think, if I can speak for the two Congressmen,
22 I think they are prepared to get their response either way,
23 either as a direct ACRS report or as access to, what shall
24 I call it, a conventional ACRS comment on licensing matters.
25 MR. SHEWMON: When you say a "conventional

1 comment on licensing matters," we have written our Perry
2 letter, haven't we?
3 MR. BERNERO: Oh, yes. Quite a long time ago.
4 I can't remember the date.
5 MR. STEFANO: July '82.
6 MR. WARD: Response to the SSER.
7 MR. BERNERO: It will be a supplementary letter,
8 responding to the SSER.
9 MR. SHEWMON: But we had an earthquake which
10 everybody agrees was less than the OBE. The plant
11 performed the way we would expect it to have performed when
12 subjected to this. I hope we don't make too much of a
13 production out of writing a letter, saying those things.
14 MR. WARD: Well, I think both of the points you
15 made are probably true but a little presumptive, I think.
16 MR. SHEWMON: We can send one more. Time to
17 walk through -- there's apparently three times there
18 already, or maybe four.
19 MR. SIESS: I want to go see those cracks.
20 MR. WARD: We are stuck with this specific
21 request. The committee is going to have to deal with that.
22 MR. SHEWMON: We don't have to make a production
23 about it.
24 MR. OKRENT: In view of the weather in Perry
25 this time of year, can the subcommittee meet in Tahiti?

1 (Laughter.)

2 MR. SHEWMON: I was there last week, the weather

3 is not bad. A little brisk.

4 MR. STEFANO: Would the chairman care to see the

5 letter we are going to send tomorrow?

6 MR. WARD: Oh, yes.

7 MR. KERR: But not before lunch.

8 MR. WARD: Before we adjourn, Mr. Ebersole has

9 one comment.

10 MR. EBERSOLE: In order to keep the committee up

11 to date it was a report sent to me a little while ago.

12 "Oconee earthquake: At 6:36 EST this morning

13 a minor earthquake of magnitude 3.5/Mblg occurred about 15

14 kilometers northwest of the Oconee plant. It was felt at

15 the Oconee station. There was no damage and no alarms at

16 the plant. The strong-motion instruments in the switchyard

17 and powerhouse deck did not trigger. The utility notified

18 the response center at about 7:30 this morning and declared

19 an unusual event. They declared it over at 10:30 a.m.

20 This event may be associated with the Keoween and Jocassee

21 reservoirs, which have been the site of the

22 reservoir-induced seismicity contact. Signed Robert

23 Rothman, 492-9434."

24 MR. SHEWMON: You said 3.5?

25 MR. EBERSOLE: Yes.

1 MR. WARD: Let's recess and come back at 1:30.
2 (Whereupon, at 12:40 p.m., the meeting was
3 recessed, to reconvene at 1:30 p.m., this same day.)
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1 AFTERNOON SESSION (3:40 p.m.)

2 MR. WARD: Our next topic is the report of the
3 ACRS Subcommittee regarding protection against criticality
4 during defueling of TMI-2. Dr. Kerr?

5 MR. KERR: Mr. Chairman, on the 29th of January
6 the Subcommittee on Core Performance met with
7 representatives of GPU Nuclear and the NRC to review
8 measures being proposed for preventing criticality during
9 the defueling of the TMI-2 core. This review was taken at
10 the specific request of the Commission, if you'll remember.

11 The report of the Subcommittee meeting is in tab
12 2, under tab 6. The Licensee and Staff described what the
13 Licensee representative represented to us, described a
14 number of analytical studies of critical system
15 calculational methods and the application of these methods
16 to the TMI core and the defueling operation of the core
17 that is being proposed.

18 Based on these calculations, and on a
19 conservative evaluation of the results, Licensee concluded
20 that by maintaining a sufficient and readily achieveable
21 level of boron concentration in the water in the vessel,
22 criticality can be avoided during defueling process.

23 The NRC staff has reviewed the Licensee's work
24 and agrees with GPU's conclusions.

25 The Subcommittee concluded that the study had

1 been appropriate and saw no reason to question the
2 Licensee's conclusions.
3 We asked for a presentation which would include
4 some television-type pictures, taken within the vessel,
5 which I think we have not yet seen. And, although I think
6 this has little bearing on the question at issue, it seems
7 to me it is of enough general interest that I believe it is
8 part of the presentation.
9 I have no further comments. There were other
10 members of the Subcommittee present. Do any of you have
11 comments?
12 MR. REED: We are talking 5000 ppm boron?
13 MR. KERR: We are talking about what they
14 proposed. To my memory, I thought it was about 4300 ppm,
15 but that will be part of the presentation.
16 MR. REED: Still well within the soluble range?
17 MR. KERR: We were assured that it was, yes.
18 Any other questions? Comments? Then I shall turn things
19 over to Mr. Standerfer.
20 MR. STANDERFER: Mr. Chairman, committee members,
21 I'm Frank Standerfer from GPU, director of the TMI-2
22 cleanup.
23 We appreciated the opportunity to meet with your
24 Subcommittee and again appreciate the opportunity to meet
25 with the full committee today. It will give us a chance to

1 give you a status report on the defueling and cleanup and
2 more specifically discuss our program to assure
3 subcriticality of the damaged core during defueling.
4 With me today are members of the GPU Nuclear and
5 Bechtel staffs. We have an integrated team reporting to me
6 made up of Bechtel and GPU people, primarily, and we are
7 supported by Dr. Westphal of the Oak Ridge National
8 Laboratory who played a key role in the analytical process
9 which formed the basis for our program, and Dr. Bill
10 Stratton, a member of our TMI-2 safety advisory board, who
11 is also here today and did perform an independent
12 assessment of our programmatic approach to the maintenance
13 of subcriticality during defueling.
14 A number of important milestones in the TMI-2
15 program have been accomplished since the head was removed
16 from the reactor vessel in July of 1984. The reactor
17 vessel plenum was removed in May of 1985. A system for
18 removing the damaged fuel debris was designed, tested and
19 installed over the open reactor vessel and defueling
20 operations commenced on October 30, 1985.
21 Throughout this period, considerable progress
22 has been made in obtaining and further understanding the
23 conditions of the damaged core and in planning an increased
24 effort to decontaminate the TMI-2 buildings and systems.
25 At this point I would like to show about a

1 six-minute videotape, to show you what the condition of the
2 core looks like and some of the early defueling steps.
3 (Videotape shown.)
4 MR. SHEWMON: (During viewing of tape.) What's
5 the diameter of a canister?
6 MR. STANDERFER: 14 inches.
7 (Tape continues.)
8 MR. STANDERFER: There was a question about
9 canisters. There are three types of canisters the fuel was
10 put into.
11 The fuel canisters would have a small insert,
12 about the size of a fuel element. All the canisters have.
13 So that's the type of canister you saw load here which will
14 take fuel element or fuel element-shaped pieces.
15 There's a second type canister, which is a
16 canister associated with the vacuum system which is called
17 a knock-out canister, and debris is sucked through that and
18 falls out in there. That has boron carbide poison rods in
19 it.
20 The third type of canister is the filter
21 canister in our water purification system, which also has
22 poison rods in it.
23 We have added a new type of canister since we
24 started fueling, in that a number of these end fittings
25 were welded together or so deformed we couldn't get them

1 into the mouth of the fuel canister. We built some
2 canisters which have no internal in them. We are allowed
3 to put end fittings in those canisters as long as we put no
4 fuel in those canisters and those end fittings will
5 probably be cut in half and loaded later.
6 MR. SHEWMON: Will you drain the waters from
7 these, weld them and ship them to a depositor?
8 MR. STANDERFER: The fuel canisters have a head,
9 a bolt-on head with an O ring. They have relief valves.
10 MR. SHEWMON: They have what valves?
11 MR. STANDERFER: Relief valves.
12 The knock-out canisters and the filter canisters
13 have welded on heads with a pipe, joint-type connections.
14 The canisters will be dewatered prior to
15 shipment, weighed, and there will be a small amount of
16 water in the bottom of the shipment but they are basically
17 filled with argon, and there are catalysts in the top and
18 bottom to recombine any hydrogen formed.
19 MR. SHEWMON: What's the activity level up at
20 the place where those suited men were working?
21 MR. STANDERFER: We were shooting for a general
22 radiation level of about 15 mr per hour, running today 3 to
23 4 mr at that location. They walk through a higher dose
24 area getting to that location. That's the lowest dose area
25 in the building, and exit back through higher areas, up to

1 30 or 40 mr.

2 We have now defueled about 20,000 man-hours, and

3 the overall average for the -- for that defueling in the

4 shifts are between, in a 3 and 4 hour shift, is 8.3 mr per

5 hour.

6 MR. MOELLER: A suggestion on that, the bulk of

7 the data I receive, say as an ACRS member, in terms of the

8 defueling and your decontamination operations at TMI-2,

9 comes through the NRC weekly information reports which, you

10 know, has a section on TMI-2.

11 MR. STANDERFER: Yes.

12 MR. MOELLER: And one of the problems with that

13 that I find, which Dr. Shewmon's questions raise, is that

14 it will say the current dose rate on such a such a floor at

15 such and such an elevation is 6 to 7 mr per hour, and it

16 never says this compares to the number three months ago or

17 last week or anything, so one never knows whether the rate

18 at which things -- the rate at which things are improving.

19 It would really help if a table could be put in there

20 showing the change.

21 MR. STANDERFER: We briefed the NRC Commission

22 on the 7th of February -- 7th of November, '84, and again

23 in January, on the 14th of January '85. And both of those

24 presentations included bar charts that showed the radiation

25 and the 347 level, both '83, '84, '85, projected '86 at 305

1 level, and we did include the defueling platform.

2 Just to review that, the basement has not been

3 entered by people and has radiation levels between 10 and

4 1000 R/hour, so we use robots.

5 On the 305 level, which is the level of the

6 outside ground, the radiation levels in '83 were as high as

7 300 to 400 mr/hour. That has been decreased each year so

8 it now averages about 80 mr/hour, although there are hot

9 spots. That's the average. Those are gotten by people who

10 work on those levels. That's not a composite of rad-con

11 measurements, it's the average working on that level. The

12 347 level, the next level higher, started at levels around

13 150 mr/hour -- I'm trying to pull these out of my memory --

14 and the average is now around 45 mr/hour, at that location.

15 At this defueling platform it's running less

16 than 10 mr/hour.

17 MR. MOELLER: Thank you.

18 MR. STANDERFER: We continue to expect our

19 decontamination efforts to continue to bring those numbers

20 down.

21 MR. MARK: Without decontamination, what's the

22 sort of half-life of the radiation?

23 MR. STANDERFER: Well, it's cesium, which is 30

24 years. There's considerable strontium present also, so we

25 often wear respirators and often have people clothed in

1 wetsuits, depending on what they want to do because the
2 surface contamination levels with strontium are very high.
3 MR. REED: Does this kind of activity come under
4 the new proposed NCR 20 rules and the targets for new
5 proposed?
6 MR. MOELLER: Yes. And, although this is cleanup,
7 it's considered -- I'm fairly certain of this -- as a
8 routine. I mean you are complying with all the --
9 MR. STANDERFER: Yes.
10 MR. REED: Are you familiar with the new
11 proposed rules out for comments?
12 MR. STANDERFER: I am not personally familiar
13 with them. I have a member of my licensing staff that I'm
14 sure is.
15 MR. REED: It seems to me that the development
16 to do this kind of work takes quite a long time and the
17 more experienced the person becomes in fishing and
18 retrieving, the better you are -- the more you would like
19 to keep him in that activity without disallowing him for
20 radiation exposure.
21 MR. STANDERFER: We do not expect any of these
22 workers to be restricted throughout the year within our
23 guidelines of no more than 1 rem per quarter. So --
24 MR. KERR: Gentlemen, I'm not opposed to this
25 line of investigation and the Subcommittee has heard this

1 presentation, and it's okay with me, but if you want to
2 hear about the criticalities of the situation, it seems we
3 probably should go to the presentation.
4 MR. STANDERFER: It is interesting work, I know.
5 As indicated, we started defueling three months
6 ago. That effort has been gaining momentum as we've
7 learned how to use the tools and modify the tools. Almost
8 every tool has had to be modified to make it more
9 functional. We expect to start shipping fuel off the site,
10 that's delivering fuel to the Department of Energy -- who
11 will ship fuel to Idaho, this spring, in May. We expect to
12 complete end reactor defueling by mid-1987, about a year
13 and a half from now, and complete decontamination in July
14 of 1988, and complete the cleanup in September of 1988.
15 That was the schedule laid out two years ago and we still
16 are tracking on that schedule.
17 Our budget project planning, schedule and
18 planning are generally consistent with the information
19 provided to the Nuclear Regulatory Commission over the last
20 three years.
21 Throughout the TMI-2 cleanup, we have continued
22 to utilize the technical advisory group, which is a group
23 funded by the Department of Energy and sponsored by GPU
24 Nuclear, DOE, EPRI, and the NRC, and has members from DOE
25 National Labs and consultants. It meets monthly for two

1 days at TMI to render its best judgment and advice on major
2 technical program areas, and to a large degree, to help the
3 Department of Energy direct their research program.
4 Further, our safety advisory board continues to
5 meet on a quarterly basis to review all effects of the
6 project -- all aspects of the project from the standpoint
7 of public and worker safety.
8 We are taking all practical steps to go forward
9 with the cleanup as quickly as practical, and consistent
10 with safety to the public and workers.
11 The programmatic approach to criticality safety
12 in defueling the TMI-2 reactor vessel has a two-fold
13 purpose: First, the soluble poison concentration was
14 established, that will assure subcriticality for all
15 credible fuel configurations. Then, controls are put in
16 place to maintain the poison concentration above this
17 minimum level. We will describe that in much more detail
18 in the next two speakers' comments.
19 With that, I would like to introduce Dan
20 Williams, who will give the next presentation.
21 (Slide.)
22 MR. WILLIAMS: I'm Dan Williams. My
23 presentation today will be a summary of the criticality
24 analyses which were done to support the TMI-2 defueling.
25 Specifically I will summarize the analysis performed for

1 the reactor coolant system, which established the boron
2 concentration at 4350 ppm. As part of that discussion I
3 will also summarize the benchmarking exercise which was
4 done for that analysis. I will then summarize an analysis
5 which was done as part of the canister design program.
6 A few years ago, when GPUN Nuclear was
7 considering how to address the issue of criticality for the
8 remainder of defueling, a number of approaches were
9 considered. The approach selected was termed the infinite
10 poison approach, infinite in the sense that a high boron
11 concentration would be defined which would keep all
12 credible core configurations subcritical.
13 In taking that approach there were a number of
14 advantages. For one, we eliminated the need, or
15 possibility, I should say, of frequent boron concentration
16 increases. Additionally, it eliminated the need to have
17 detailed information on the physical configuration of the
18 core, and it also offered minimal effects on the operation --
19 or defueling. There would be minimal restrictions.
20 I would first like to give an idea of what we
21 knew about the core at the time this analysis was performed.
22 (Slide.)
23 Mr. Standerfer gave you an update as to what we
24 know today, and also showed a videotape. But at the time
25 this analysis was performed, we didn't have that

1 information.

2 There were a few things that we did know. We
3 knew, for instance, that the core obviously had been
4 damaged and the top portion was mixing and collapsed to
5 about half of its original height.

6 We knew, also, there was probably some fuel in
7 the lower head. That was about it.

8 So, to develop the model for our infinite poison
9 approach, we relied on the results of previous analyses.
10 We knew some things about general physics that would help
11 us, and we also knew that putting the fuel in a circular
12 geometry, a circle geometry, would tend to increase K.

13 We also knew, for instance, that putting the
14 highest enrichment fuel in the center of our model would
15 increase K.

16 Now, the model actually selected was termed the
17 lenticular model, and this is what we ended up with. As
18 you can see, it doesn't really resemble the vessel I showed
19 you earlier, but it does incorporate the conservatism,
20 namely putting the highest enrichment fuel that was
21 initially charged in the reactor in the center; that
22 surrounded by the remaining batches, 1 and 2 fuel; and then
23 surrounded by steel which we knew would reduce leakage.

24 To give some physical bearing on our design
25 basis model compared with the vessel, we have that model

1 shown again.

2 (Slide.)

3 The reactor vessel is overlaid.

4 You can see, for instance, that the fuel

5 material does come into the area which is really occupied

6 by some of the vessel internals, but in our model that

7 material was ignored. It would tend to decouple, but it

8 was ignored.

9 I also point out that these analyses, our

10 lenticular model specifically, was analyzed by Oak Ridge

11 National Laboratory, Dr. Westphal, and he is with us today.

12 (Slide.)

13 I would now like to move on and review a few of

14 the conservatisms associated with our model, the first

15 being the configuration of the fuel, placing the highest

16 enrichment fuel in the center, having a geometry that does

17 tend to minimize leakage but at the same time not going to

18 an unrealistic fuel. Our plan does conform with the

19 configuration of the vessel.

20 Credit was taken -- our burn-up was limited to

21 the batch refuel only and credit was taken only for the

22 isotopes. I discussed the fact of the steel reflecter

23 already.

24 As far as the poison material, it was ignored in

25 our analysis. We knew control rod material was there

1 somewhere, but it was ignored.

2 The fuel-to-moderator ratio was optimized.

3 Unlike undamaged cores where we know the physical

4 arrangement of the fuel in terms of the amount of fuel per

5 unit of moderator, we did not know that in this case so we

6 chose to optimize, which meant we took the volume of fuel

7 and water that would maximize K infinity.

8 Additionally, we had done analyses which

9 indicated K would vary inversely with temperature, so the

10 analysis assumed the fuel and moderator temperature was at

11 50 degrees, which is the lowest temperature allowed by the

12 TMI-2 tech specs.

13 In doing this analysis we had, I guess, two

14 concerns. One was to develop a conservative model. But we

15 also had a concern with regard to the bias of the Keno code.

16 Keno had been benchmarked earlier but those benchmarks were

17 primarily for solutions, for highly enriched fuel, and, for

18 the most part, were for arrangements that had little boron.

19 So, to address this issue we developed a

20 benchmarking program. That effort was headed by Dr.

21 Raymond Murray, who could not be with us today.

22 (Slide.)

23 In doing that, we establi...ed a set of criteria

24 that we would use to select our experiments. The first was

25 a dry lattice and that is not dry to the extent all water

1 was removed but we knew the fuel at TMI-2 was closely
2 packed so we wanted criticals that had little moderation.
3 Additionally, we wanted criticals that had boron
4 concentrations in the range of 5000 ppm, which is close to
5 the TMI-2 situation.
6 We were after critical systems, because that
7 would minimize concern with the actual experiment in terms
8 of really saying what decay effect it was. And, since we
9 are out to determine bias for our analysis, we did want to
10 have experiments that themselves had a larger error
11 associated with them.
12 The low enrichment, we wanted something in the
13 range of 3 percent, like the TMI-2 core. And, ideally we
14 wanted to incorporate several organizations as opposed to
15 depending on one, just so there would not be a common mode
16 error in the experiments. And also see if there were any
17 trends that we could examine.
18 As a result of this criteria, we then went
19 through an extensive literature search. We examined
20 publications put out by the National Criticality
21 Information Service as part of Lawrence Livermore.
22 Additionally looked at such publications as the
23 Transactions of the American Nuclear Society, and based on
24 that we came up with 10 experiments that fell into our
25 criteria. These 10 experiments were spread across three

1 organizations -- two, I should say. They fell into three
2 groups to evaluate.
3 (Slide.)
4 These experiments -- I think you can see this at
5 this point in your handout -- but what we used was the
6 moderating ratio, to compare our experiments against. And
7 you can see the open circles are representing the range for
8 the TMI core, of course damaged versus undamaged. The
9 closed are the experiments. This was more to demonstrate
10 that we have selected experiments that we feel came into
11 the range of the TMI-2 fuel.
12 These experiments, as with the lenticular model,
13 were analyzed by Dr. Westphal, and because of the limited
14 number of the experiments, namely 10, we did not apply any
15 statistical methods to the results but, instead, decided to
16 conservatively go with worst case.
17 I have the results plotted here on a bar chart.
18 (Slide.)
19 What you can see is most of them do fall -- unity
20 is here. Most of the results did come within about a 1
21 percent delta of critical.
22 We decided to go with the worst case here;
23 there's only one of the 10 that fell into that category.
24 But from those results decided to apply a K effective --
25 excuse me, a bias of 2.5 percent.

1 That 2.5 percent was to not only account for the
2 code uncertainty, but also the standard deviation, which
3 was typically about a quarter.
4 (Slide.)
5 Now, moving onto results of this analysis, what
6 are shown here are the results for the lenticular model as
7 a function of boron concentration. The numbers with the
8 2-1/2 percent delta K uncertainty add, at 4350 ppm, we have
9 a K effective of 99, which was our criteria. Granted our
10 criteria is somewhat high, it could be lower, but what we
11 felt was, given the highly conservative model, that that,
12 coupled with a K effective of .99 as a criteria, would
13 establish a safe boron concentration.
14 I would also like to point out that the
15 administrative procedure for the plant calls for a boron
16 concentration of at least 4950. You can see from that, it
17 would be up -- down in the .98 range, almost. The plant
18 today is in excess of 5200 ppm, which says even for our
19 conservative model we would be below .97.
20 MR. MARK: Are those K infinity?
21 MR. WILLIAMS: No. These are actually K
22 effective for the model.
23 MR. MARK: For a 14-inch?
24 MR. WILLIAMS: No, sir. This is for our
25 lenticular model.

1 MR. MARK: Okay. The one you showed us.

2 MR. WILLIAMS: For geometry, it's roughly 12

3 foot -- diameter, I should say.

4 In summarizing this portion of the analysis for

5 the record, we can say we have decided to go with soluble

6 neutron poison as a method for ensuring subcriticality. We

7 believe this offers a number of advantages.

8 One, it's something that we are familiar with.

9 It's available, also tryable, and that will be discussed

10 later by me. It's compatible with plant chemistry, and

11 also offers operational flexibility.

12 (Slide.)

13 What I mean by that is we wanted to have a

14 situation where we could let the defueling operators go in

15 and remove fuel without putting restrictions on, one, how

16 they remove it; or what they have to work around. Soluble

17 poison offers that.

18 We believe that our model balances all credible

19 configurations and as long as we keep a boron concentration

20 of 4350, K effective will not exceed .99. I also would

21 like to point out that that .99 is for our design basis

22 model. We believe that in reality, the system today is

23 highly subcritical.

24 That concludes my presentation on the reactor

25 cooling system. I would now like to move on to the

1 canister analysis.

2 MR. KERR: Let me ask a few questions.

3 MR. REMICK: I have one question, the criteria

4 of K effective being .99. How did you arrive at that

5 versus some other number? I think that's what one uses in

6 spent fuel pools, .99, but why here?

7 MR. WILLIAMS: I guess the first thing that got

8 us going on that was the tech specs for recovery at TMI-2,

9 where the criteria was to remain subcritical.

10 Typically for operating reactors, subcritical

11 was .99. That was our starting point then from -- that we

12 felt, well, we do want to have some degree of conservatism,

13 all right? And we figured the criterion, coupled with,

14 like I say, this highly conservative model, would give us

15 the boron concentration. And even considering

16 realistically the boron concentration only, still coupled

17 with our model, we are even lower; say, less than .97.

18 MR. ETHERINGTON: How do you feel your boron

19 concentration penetration is --

20 MR. WILLIAMS: That's a good question and that

21 will be addressed as part of a later presentation by

22 Dr. Knief, if you could hold that question until then. Is

23 that it on this portion?

24 MR. KERR: Yes, sir. Please proceed.

25 (Slide.)

1 As Mr. Standerfer was saying earlier, we have
2 three canisters that we are using for the defueling program.
3 First is the fuel canister. That was the one you saw being
4 loaded, with an end fitting, in the film. This is loaded
5 manually. This head is removed. The system is kept
6 subcritical by a boron shroud around the perimeter of the
7 loading volume. It is used to load all fuel degrees.
8 Sized to take things as large up to a fuel assembly.
9 The second canister is a knock-out canister.
10 This is used with our vacuum system as you saw in the video.
11 Again, the shutdown mechanisms here are the five control
12 rods -- shouldn't say control rods -- the poison tubes that
13 go pretty much the length of the canister.
14 The third is the filter canister. This is used
15 as part of the vacuum system, but also with the water cleanup
16 system. It will contain primarily fine -- fine material.
17 And, again, it has one poison tube that is centered,
18 approximately 2 inches in diameter.
19 MR. MOELLER: To help me, on the filter canister,
20 in other words, when you use the vacuum machine it
21 automatically goes through this?
22 MR. WILLIAMS: When it's part of the vacuum
23 system, the first canister is a knock-out canister where
24 the larger degree, pellet-sized material will be knocked
25 out by centrifugal force. What we'll get out are the finer

1 material that can go along, say, suspended in the flow.
2 That will be trapped in the filter canister.
3 MR. MOELLER: Thank you.
4 MR. REMICK: What considerations determine the
5 diameter of the knock-out canister and filter canister?
6 MR. WILLIAMS: I think the first thing which
7 drove the size of these was the desire to be able to load
8 fuel assemblies. Okay?
9 From the fabrication point of view the shells
10 for all of these canisters are the same. All right? I
11 guess, also, as part of this -- which is kind of out of my
12 area -- was the thermal hydraulics, okay? -- of getting
13 the fuel material to fall out of the flow stream when
14 operational.
15 MR. REMICK: But basically they are the same
16 size as the fuel canister?
17 MR. WILLIAMS: Yes. The shells for all of the
18 canisters are the same.
19 (Slide.)
20 I will now discuss some of the conservatisms
21 associated with the canister analysis. I also point out
22 these canisters were designed by B&W; and the analyses, too,
23 were performed by them.
24 The first is the fact that we only use the batch
25 3 fuel; the fuel configuration -- you saw the video scene.

1 The damaged fuel -- however, these analyses assumed that
2 you still had stacked pellets and that was for all three
3 canisters.
4 The third was the omission of any structural
5 material, some of which we know will get into the canisters.
6 We did not take credit for the soluble boron present in the
7 reactor coolant system, but instead assumed the moderator
8 was unborated water in these analyses. And, with the RCS,
9 we did go through and determine what the optimum moderation
10 was for our fuel.
11 The canisters do have a weight restriction from
12 a loading perspective. However, in the analysis that
13 restriction was ignored and we put optimized fuel into all
14 of them.
15 As with the RCR analysis, we did go with a
16 minimum temperature of 50 degrees.
17 The 95/95 confidence really applies to the
18 fabrication of the poison. We wanted to make sure that the
19 poison did meet the specs defined in the analysis.
20 MR. REMICK: What kind of assumptions are
21 eventually grouping these canisters somewhere down the road
22 and being put together? Is there any requirement that they
23 be a certain distance from one another or anything else?
24 MR. WILLIAMS: We do have a requirement at TMI-2,
25 in terms of the loading into the racks, and that dimension

is a minimum of 17.3. However, I'll point out that was developed, again, ignoring the presence of borated water. It's assumed they were unborated.

MR. MOELLER: Can you say again what it meant "accountable for the fixed poison only," your item 4 or so?

MR. WILLIAMS: What I meant was, when the canisters are at the plant, they will be loaded at, for instance, a reactor coolant system which is borated, soluble boron at 4250; when they are stored in the pool, that, too, is borated. But we did not take any credit for that and just assumed the moderator was unborated water.

MR. MOELLER: When you say "fixed poison," that's the poison in the wall of the fuel canister?

MR. WILLIAMS: In the wall and the poison tubes in the others.

MR. REMICK: You mentioned the 17 inches in the pool. How about shipment? Is there any chance of reflooding with unborated water and you have to worry about canisters --

MR. WILLIAMS: Yes. And that was addressed. However I don't plan to address that here today. The shipping end of things.

MR. STANDERFER: The Department of Energy is buying the shipping casks, licensing them, and they take responsibility. They will be stored in poison racks in

1 pools in Idaho and we weren't going to cover that, although
2 they have done fairly extensive analysis to assure they
3 meet all normal regulations.
4 MR. REMICK: NRC is going to license the
5 canisters?
6 MR. STANDERFER: Yes, they are in the final
7 stages of licensing those two canisters.
8 MR. WILLIAMS: But they did address the question.
9 (Slide.)
10 Okay. The criteria that we apply to the
11 canister program was a K effective of 0.95. For the fuel
12 canister we had a standard configuration that was shown in
13 the picture. K was, and again this is the maximum K, of
14 0.857. And for the array, with the 17.3 inch spacing, was
15 roughly 0.88. This is a minimum. The nominal is closer to
16 18.
17 When I say the K effective, that's the
18 calculated factor coming out of Keno, plus the two-signal
19 standard variations, in addition to a code bias of 2
20 percent for single canisters and 2.3 for arrays of the
21 filter canister.
22 (Slide.)
23 Again, I'll just point out the maximum number
24 which was for a damaged condition and a K of 0.892. The
25 damaged condition here was to cover, say, combining

horizontal and vertical, that was to cover all possible drop orientations.

(Slide.)

Last is the knock-out canister, where the worst case was for the array that had a K effective of 0.915.

MR. ETHERINGTON: These numbers are for the poison?

MR. WILLIAMS: Yes, they are. For the fixed poison.

(Slide.)

In summarizing, we can say that single canisters are designed such that K effective is less than 0.95. If they are moderated with unborated water. For canisters stored in the racks at TMI-2 where the minimum spacing is 13.3, again with unborated water, the K effective in all cases was less than 0.95.

Also, as part of defueling, these canisters will be handled with shields. We had a concern as far as, given the shields which are there to reduce dose but would also minimize neutron leakage, we wanted to make sure we would not exceed our K effective criterion for these canisters.

(Slide.)

Just to briefly give you a perspective of what these shields look like, I know this is a little busy, but the canister would be here, we have a couple of shields

which are laminated steel and lead. In this particular one the lead thickness ranges from about 2 inches to about 6 here. This is our canister transfer shield which is used in containment and to also place canisters in racks in the handling building.

MR. MOELLER: How much does a shield weigh?

MR. WILLIAMS: I'm not sure of that number.

MR. MOELLER: How long are the canisters? You gave us the diameters.

MR. WILLIAMS: The canisters are roughly 150 inches in length.

The other shield which is used is termed the transfer cask. This is used to take single canisters from the spent fuel pool over to the tipping cask. It has a thickness of 4.5 inches of lead in the center, sandwiched between two one-inch steel plates.

MR. SHEWMON: They weigh a lot.

(Slide.)

MR. WILLIAMS: Using the canister that was most reactive, as you can see even with uncertainties the K effective, the worst case being with a fuel transfer cask, the worst case was 0.95.

In summarizing, I'll say that we have defined a boron concentration of 4350. That will keep the reactor coolant system subcritical for all credible configurations

1 on the defueling process.

2 Additionally we have designed canisters that

3 will maintain the fuel in them at a K of less than 0.95 for

4 all credible circumstances.

5 MR. REMICK: Going back to your assumptions, you

6 might have answered this and I might have missed it, but

7 you indicated that you assumed an optimum moderator-to-fuel

8 ratio.

9 MR. WILLIAMS: Yes.

10 MR. REMICK: But the fuel in what configuration?

11 Is this homogeneous? Assuming they were still in rods?

12 What's the most conservative assumption?

13 MR. WILLIAMS: What we did, in the case of

14 canisters, we did assume that the fuel is still in rod form,

15 stacked pellets. So, it was optimum moderation given you

16 had, you know, rods of fuel. And the reactor coolant, the

17 lenticular model, our concern there, we wanted to be

18 conservative but at the same time show some realism; in

19 which case we knew the fuel was primarily in a rubble

20 fashion, closely packed. So we had a lattice structure

21 that had -- call it dodecahedron -- 12 spherical balls;

22 okay? But, again, they were arranged -- how am I going to

23 say it -- in an optimum fashion in terms of the quantity of

24 borated water between those clumps of fuel.

25 MR. REMICK: To assume this was a homogeneous

1 mixture, that would be, I assume, your least conservative
2 assumption?
3 MR. WILLIAMS: Yes. Yes.
4 MR. REMICK: So you took the least conservative
5 either in the shape of rods or in some in-between stages?
6 MR. WILLIAMS: Yes. We did work with discrete
7 clumps of fuel.
8 MR. KERR: Another question?
9 MR. WILLIAMS: I will now turn the discussion
10 over to Dr. Knief, who will talk about the operational
11 aspect of maintaining subcriticality.
12 MR. KNIEF: While Dan was talking, I was
13 provided with at least a partial answer to one of the
14 questions. We actually have three separate bells that we
15 use for transfer of the shipping casks. One of them weighs
16 about eight tons and the other weighs about 10 tons. The
17 other weighs about one ton; is that right, Jim?
18 MR. BURNS: I'm Jim Burns, manager, licensing,
19 TMI-2. There's three shields. The one in the reactor
20 building is a little -- there are two similar shields, one
21 in the reactor building and one in the fuel handling
22 building. One was about 10 tons, the other weighs about
23 eight tons.
24 The transfer cask to transfer the fuel from the
25 fuel pool to truck bay to the basement of the cask weighs

about 17.5 tons.

(Slide.)

MR. KNIEF: Having examined the analytical aspects, it's now appropriate to take a look at some of the facets of operational criticality safety during TMI-2 defueling.

(Slide.)

With the poison concept that Dan was talking about firmly established, and 4350 ppm of soluble boron designated as a minimum level, the prospect for boron dilution from the current level of something in excess of our administrative limit of 50/50 ppm, the boron dilution becomes a key operational concern and, in fact, the subject has been addressed in some detail from the following three perspectives.

First of all, prevention, that is using available methods to prevent unborated, or at least underborated water from entering the reactor vessel.

Secondly, we looked at it in terms of detection, or using redundant means to identify system changes that might signify the onset of dilution.

And then finally we looked at a termination phase which would, essentially, include preplanning to respond to an event which might lead to boron dilution.

Each of the aspects will now be considered in

1 just a little bit more detail.

2 (Slide.)

3 The approach to dilution prevention has been
4 based largely on a detailed hazards analysis, looking at
5 some of the complexities that we have in the TMI-2 system
6 as it exists right now, and then taking into account that,
7 while, yes, we want to isolate the level in terms of
8 potential boron dilution, we still have a need to conduct
9 ongoing operations, including processing of water to reduce
10 the levels of radioactivity. And, so, we took our approach
11 by focusing on isolation of potential inlet points when
12 they were not being used for some other purpose, and using
13 a double-barrier concept, whereby at least two barriers to
14 flow are present in each identified pathway, we proceeded
15 to use some methods, including the following that are shown
16 here.

17 One of the most direct methods, of course, would
18 merely be to close existing valves in the system.

19 A somewhat more positive preventative measure
20 would be to remove spool pieces or, equivalently,
21 essentially remove pieces of pipe so that flow is not
22 possible between two particular locations.

23 And, then, finally there are a number of places
24 where we can take advantage of differential heights, which
25 would prevent flow from going from one source of water to

1 an inlet that happened to be located at a higher level.

2 MR. EBERSOLE: I would like to ask a question.

3 There's been an incident at one of the plants recently

4 where workmen elected to go in and do some pipefitting and

5 heavy maintenance and they used the valve isolation concept.

6 However, they did not recognize that that valve could be --

7 it was still energized. And by spurious operation, someone

8 unfortunately opened the pipe on the far side and they had

9 a terrible flooding event; the reason being that they did

10 not deenergize, disconnect or remove the guts of the valve

11 operating system. Have you taken steps to prevent such

12 from happening?

13 MR. KNIEF: I don't know. Did you hear the

14 question, Adam? Adam Miller is our manager of operations.

15 He may be in a little better --

16 MR. EBERSOLE: I pointed out what happened to

17 them. It would be worse for you.

18 MR. KERR: Would you identify yourself and use

19 the mike, please?

20 MR. MILLER: My name is Adam Miller, manager of

21 plant operations at TMI-2.

22 In cases where it's practical those valves have

23 been deactivated and appropriately tagged.

24 MR. EBERSOLE: What do you mean "deactivated"?

25 MR. MILLER: Deenergized or locked closed.

1 MR. EBERSOLE: These were deenergized but they
2 still had the handles on them.
3 MR. MILLER: Some of them, also a locking
4 technique with a chain and lock.
5 MR. EBERSOLE: Whatever it takes. That's all.
6 I just wanted to comment on what happened to them.
7 MR. KNIEF: Then the intent is that the
8 isolation would be maintained by administrative control,
9 some of which were just mentioned in Adam's comments.
10 (Slide.)
11 In addition, then, we have rather extensive
12 checklists and do require a periodic verification that,
13 indeed, the double barriers are maintained in the manner
14 intended to be able to facilitate our dilution prevention.
15 (Slide.)
16 Dilution detection is based on redundant
17 monitoring. When water processing is not in progress,
18 water-level monitoring turns out to be one of the easiest
19 and one of the most direct ways to monitor against
20 potential dilution of the system. And, in fact, we have,
21 then, redundant level sensors and have readouts both in the
22 control room and locally, in the vicinity of the operations.
23 When routine processing is under way, as you
24 might imagine, we would expect there to be some normal
25 level fluctuation, and at that point monitoring boron

1 concentration becomes our primary method, although we use
2 some of that -- in fact, with boronometers, we do that
3 continuously, even during the static conditions.
4 Right now we have a boronometer which samples
5 the water, essentially continuously. It is located in the
6 vicinity of the top of the vessel and has both control room
7 and local readouts.
8 In addition, a second boronometer is planned for
9 the defueling water cleanup system, or DWCS, which is
10 presently under construction.
11 In addition to this, we have the ability, and
12 routinely make use of grab samples, which can be taken
13 either in the central region of the vessel above the core
14 debris area, or we can also sample in the annulus, between
15 the core form and the reactor vessel.
16 MR. REMICK: Is there any need for temperature
17 control because of the precipitation? At these
18 concentrations you might.
19 MR. KNIEF: I don't know. Does anybody have a
20 handle on that? The normal temperature right now is in the
21 80 to 85 degree range. The total heat load is only about
22 12 kilowatts, so don't -- we don't experience significant
23 temperature increase.
24 MR. BURNS: This is Jim Burns, manager of
25 licensing, again. When we raised the boron concentration

1 from 3500 up to the range of 4350, we changed our technical
2 specification requirements to put an upper limit of 6000
3 ppm boron on the reactor vessel.
4 In doing that we did some experiments with
5 borated water -- the type of system we have, we have boric
6 acid with a cesium hydroxide buffer to maintain pH around
7 7.5. In that range we lowered the water temperature to
8 freezing, basically, and the boron did not precipitate out
9 at 6000 ppm, so there's no concern with boron
10 concentrations.
11 MR. SHEWMON: Do you know whether that's an
12 equilibrium value or whether it was one that just happened
13 to work at the rate at which you happened to cool the
14 solution?
15 MR. BURNS: I really can't answer that question.
16 I don't know the specifics on the analysis that was taken.
17 But, B&W helped us perform that operation and looking, just
18 doing that operation, also experimental course that B&W had
19 -- they maintained we could maintain boron in solution at
20 that concentration at the temperature range we have at
21 TMI-2. We also have a lower temperature range on the
22 system at 50 degrees, so we are pretty positive that that
23 temperature will keep boron in solution.
24 MR. KNIEF: The dilution termination strategy is
25 to isolate the vessel inlets prior to any event which could

1 drop the concentration to 4350 ppm or lower; that having
2 been established through our calculations as the minimum
3 value.
4 (Slide.)
5 In fact, this whole idea of termination serves
6 as one of the important bases for setting 50/50 ppm as the
7 administrative limit and then also establishing a sampling
8 and process matrix so that the processing and sampling
9 rates would allow for a timely response; that is, the more
10 water we are processing the more frequently we do the
11 sampling. This is somewhat independent of the fact that we
12 now have the boronometer present to essentially do
13 continuous sampling.
14 In addition, then, procedures have been
15 developed with the idea of supporting this termination.
16 Procedure actions, then, include securing
17 operations, which encompass activities such as shutting off
18 pumps, closing valves, and so forth, much of which can be
19 done in the control room.
20 If you take a look at any operations that are
21 involved in moving water around, and any that were to be
22 happening at such a time that the boron samples would
23 indicate a potential problem, the first step would be to
24 shut everything down and then reexamine the situation. Do
25 additional checks on boron concentration, review the valves

1 and the valve lineups, close additional valves as
2 appropriate. The most important ones, it turns out, can be
3 closed from the control room.
4 And then, finally, we have the ability to make
5 up with borated water if we decide that that's necessary.
6 Are there questions related to boron dilution before I move
7 on?
8 (Slide.)
9 The other subject that bears some relationship
10 to operations is that of neutron monitoring. There are our
11 nuclear instruments, which are BF-3 detectors. They are
12 located outside of the vessel on the inside of the
13 surrounding concrete shield wall. Their operability for
14 detection of neutrons has been verified by some tests that
15 we ran, one within a year or two after the accident, and
16 then again we did a check just within the last three or
17 four months. In fact, just prior to defueling operation.
18 And then, in addition we have procedural
19 requirements to take these instruments into account and, in
20 fact, have requirements to take action if we were to see a
21 sustained increase in the level indicated by these
22 detectors.
23 MR. REMICK: Are those instruments indicating on
24 scale?
25 MR. KNIEF: Yes, they are indicating a little

1 over one count per second with the circuitry that we have
2 with it. We've taken a look at the pulses, and this is
3 definitely controlling -- pulse shape versus other types of
4 radiation.
5 MR. REMICK: This is without any external source?
6 MR. KNIEF: That's correct. The neutron source
7 that we are relying on, in fact, is the one that comes
8 right now from spontaneous fission, and alpha, n reaction.
9 (Slide.)
10 I have got this backwards. The n, gamma turns
11 out to be essentially negligible because we have don't have
12 any that give off radiation in excess of 2.5, so you'll
13 make that correction in your notes. Switch things around.
14 It makes a lot more sense. (Indicating.)
15 MR. REMICK: On the discussion early on whether
16 you need neutron detectors, why was that discussed if these
17 treatments were working?
18 MR. KNIEF: Let me proceed just a little bit
19 further. We come to that.
20 One of the issues that we've taken a look at is
21 whether or not we can do subcritical multiplication
22 measurements with what we have in place or whether we could
23 do meaningful measurements in some other way.
24 And, in looking at this we immediately identify
25 that we have some rather severe limitations. First of all,

1 we have an unknown geometry; other than the general
2 boundaries of the core region we know very little about the
3 internal geometry.
4 We have a very high boron concentration, which
5 makes it extremely difficult for neutrons to get out. And
6 then we have a very low neutron source. In fact, these two
7 turn out to be roughly the same. We've got some
8 spontaneous fission from plutonium isotopes. The alpha, n
9 is primarily from oxygen.
10 We did some evaluation. In fact Tony Baratta,
11 from Penn State, has done some calculations for us looking
12 at the response of the neutron detectors to various changes
13 in boron concentration in the core and then also to
14 geometry changes. And the conclusion that he came to is
15 that these detectors would -- really don't see any of that.
16 We'd have to postulate some extremely great changes, like
17 almost deborating completely, before being able to see that.
18 Part of the problem is that the spontaneous
19 fission and alpha, n source are located in the lower vessel
20 head with the fuel that is accumulated down there. Those
21 neutrons only need to go through the vessel to get out and
22 be detected.
23 The neutrons that are produced in the core need
24 to travel through the core former, and then an annular of
25 now borated water and then out through the vessel. So it's

1 about 100 to 1000 times more likely to see the neutrons
2 that are in the lower head region.
3 The conclusion, then, from these calculations,
4 is that we would not see anything useful on these detectors,
5 in terms of -- well, in terms of boron concentration change
6 or approach to critical.
7 However, we are convinced that if the system
8 were to go critical and the neutron population go up, then
9 -- many orders of magnitude -- that these detectors would
10 be available for that purpose, although, as you have seen
11 earlier, we certainly have taken every measure that we
12 could to make sure that we never have to use them for that
13 purpose.
14 MR. MARK: Roughly what is the DOT calculation?
15 It's not Department of Transportation?
16 MR. KNIEF: True. Does the acronym mean
17 something?
18 FROM THE FLOOR: Discrete Ordinance Transport.
19 MR. KNIEF: Discrete Ordinance Transport. It's
20 a transport code that is for modeling neutronic systems.
21 MR. KERR: I think it's a two-dimensional
22 conversion.
23 MR. MARK: It's generally used irrespective of
24 code; it's not just from Penn State?
25 MR. KERR: Especially with specs.

1 MR. REMICK: That decreases your confidence; is
2 that right?
3 (Laughter.)
4 MR. KNIEF: One other thing we have done in this
5 area, we have taken advantage of several workshops, one
6 held in 1983 by the Department of Energy, focusing on TMI-2,
7 getting together a number of experts in the field looking
8 at the prospects of doing subcritical reactivity. And then,
9 in addition, another workshop was held -- I don't know
10 fortuitously or otherwise -- in August of 1985. And we had
11 an opportunity, then, to sort of take a final look at the
12 state of the art and to have a chance to talk to some of
13 the practitioners.
14 The conclusions that we drew from each of these
15 sessions seemed to be essentially the same as we had come
16 to here.
17 MR. REMICK: Basically you have no neutron
18 detectors to detect a change in neutron density. You see
19 criticality, of course, assuming this doesn't work -- was
20 any thought given to another neutron detector put in the
21 core, let's say?
22 MR. KNIEF: We did consider putting neutron
23 detectors either in the vicinity of the core or perhaps
24 right next to the bore barrel, and we decided that would
25 not provide us any information that would be usable to us

1 in defueling activities. We had some difficulty being able
2 to interpret the data that we got from it, primarily
3 because we still have these same problems of unknown
4 geometry, high boron concentration, and low neutron source.
5 Had we been able to put neutron sources in the
6 core conveniently without hampering operations, then this
7 might have had some more appeal despite the limitations.
8 MR. KERR: Have you thought about getting class
9 1 seismically qualified, QA experimental neutron setups?
10 MR. REMICK: No. I'm thinking of a nice little
11 swimming pool where you put a tube down, put a detector
12 down, do it simply with confidence.
13 MR. KNIEF: That is exactly what Bob Long and I,
14 as former professors, thought of.
15 Are there any other questions?
16 (Slide.)
17 Then, just as a wrap-up of our entire
18 presentation, our approach to criticality safety is
19 essentially that of bounding poison concentration in the
20 reactor vessel; of a combination of geometry and poisons in
21 the canister. There, with that I think rather substantial
22 conservatism of considering them to be filled with
23 unborated water. And then these being translated over to
24 operating constraints where we have done all we can to
25 maintain the poison concentration and also to monitor, to

1 assure that, indeed, we have maintained that poison.

2 MR. KERR: Are there further questions? Thank

3 you.

4 MR. KERR: Would the staff be willing to comment

5 on your review?

6 MR. TRAVERS: I'm Bill Travers and I'm director

7 of TMI-2 cleanup directorate. As Dr. Kerr mentioned at the

8 beginning of the meeting, and something I'll reiterate, the

9 NRC Staff has taken a very detailed look at the entire

10 measure of the steps that are employed currently and

11 successfully, I might add, to prevent against criticality.

12 And we agree that they are reasonably conservative and

13 bound the issue at hand.

14 MR. KERR: Thank you.

15 MR. TRAVERS: We have looked at essentially all

16 the things you've already heard about.

17 MR. WARD: Thank you very much. We'll turn to

18 our next topic, then, agenda item 7. It's the report of

19 the subcommittee on the San Onofre Unit 1 incident.

20 MR. EBERSOLE: Topic 7 here is schedule for 5:45

21 p.m., but I doubt it will take that long because we found

22 in the course of our Subcommittee yesterday -- that we were

23 by no means in a position to do the things that were stated

24 to be the purpose of the February 12th meeting. That was --

25 that is all of them -- to develop an understanding of the

1 event, possible site-specific and generic implications.

2 We did develop an understanding of the event to

3 degree, to be briefed by the staff at Southern California

4 Edison, to be briefed on the staff actions and

5 recommendations. In that context we are almost midstream

6 in this investigation and I call your attention to what is

7 the most substantial action I can point out. It's a memo

8 from Vic Stello to Denton, Taylor and Martin, subject "staff

9 action resulting from the investigation of the November 21

10 San Onofre Nuclear Generating Station Unit 1 event, NUREG

11 1190, which we can get to you. I suggest you not use the

12 draft because the pictures in there are not good.

13 And then the third purpose was to develop the

14 recommended course of action for ACRS to be presented at

15 this meeting. That cannot be done because we simply don't

16 have enough information yet. You might even notice that we

17 don't even have -- is Ron here?

18 MR. MOELLER: He's in the hall. Everyone has

19 gone back to work, so to speak, on the San Onofre job. We

20 did have a number of members at the meeting yesterday, but

21 I can just, in a very few moments, tell you what the

22 framework of the event was.

23 Through some curious electrical arrangements

24 resulting in some problems with the electrical system, the

25 feeds to the two springs of the main feedwater pumps such

1 that one pump had access to a power supply that the other
2 one -- quite different from the other one. They normally
3 lined up to an off-site power source, which is a prudent
4 design rather than putting them on turbine output.
5 They got it into this configuration and then the
6 original electrical fault showed itself by actual failure
7 of the transformer and this caused -- I forget whether it
8 was east or west; it doesn't matter -- one of the main feed
9 pumps to suddenly go dead.
10 But the other one kept on running, and at that
11 point they discovered a shocking fact that had been
12 materializing, apparently inside the last year, not any
13 longer and perhaps shorter than that.
14 They had five swing-type reversible check valves
15 on the main feed lines which, in essence, didn't have any
16 guts. And this automatically led to the water shocking --
17 and I mean shocking event -- of water flowing backwards in
18 the line with the dead pump, leading to splits in the
19 heater shells and a natural ferocious water hammer which
20 led, then, to subsequent evolutions that finally culminated
21 in a second water shock due to condensation knock sometime
22 later in the event.
23 But the crux of the whole thing is this: They
24 had a highly unanticipated failure coincidentally, or
25 rather over a time interval for which it was designed, with

1 5 swing jacks. They were gone. And they had a transient
2 which in ordinary circumstances would have amounted to
3 virtually nothing but in these specific circumstances
4 resulted in the waterhammer and damage to the equipment.
5 MR. SHEWMON: Were the check valves not there in
6 the first place or not functioning?
7 MR. EBERSOLE: As far as I'm concerned, this
8 bears heavily on the considerations at the Palo Verde plant,
9 and one type of failure of aux feedwater not currently in
10 the PRAs, or other anticipated failures which would kill
11 the aux feedwater system.
12 These valves had been -- this is an old plant,
13 had been running since '67, I think, or thereabouts. It
14 had in recent months pulled back 10 or 15 percent on power
15 because of troubles with the steam generators. That
16 resulted in a curious effect on the check valves, in that
17 they no longer were held driven up against the stops in
18 their pipes, or their bodies against the stop, but rather
19 they fell down a few degrees and began to flap. And that
20 reduction in flow, and the momentum applied to the valves,
21 caused an oscillatory action to begin to beat the retaining
22 bolts and nuts on the back end of the disc, against the
23 stop, the pipes, and eventually apparently hammered and
24 banged around to the point where the nuts came off and, in
25 essence, the discs came off, and they had no guts in the

1 valves and no evident signals came out of the system to
2 tell them that it was the case.
3 Apparently they simply degenerated these valves
4 due to this part-flow condition, as a result of this, the
5 cycle.
6 MR. SHEWMON: The valve flapper should have been
7 running around the system someplace?
8 MR. EBERSOLE: It fell down or lodged up.
9 Didn't go back into the control valves.
10 MR. KERR: There were some that moved away from
11 the position.
12 MR. EBERSOLE: They did not do what they could
13 have done, which was run downstream and eventually take out
14 the valves that were used to stop the flow. They didn't do
15 that. They could have.
16 We asked the representatives there what were the
17 ultimate implications of this in the context that they
18 stood a fair chance of never having recovered aux feedwater
19 as well as main feedwater and in the context of could they
20 feed/bleed -- which, of course, Palo Verde could not -- and
21 they assured us they had it in their pocket if they had to
22 use it and I found out they couldn't.
23 I'm sure we have a generic problem now unearthed
24 in check valve design, dynamics of check valves --
25 MR. SHEWMON: This check valve is presumably a

1 flapper that comes against some seat?

2 MR. EBERSOLE: It comes up in the pipe and

3 strikes a mechanical stop with a bolt that is used to

4 fasten the disc to the swing hinge.

5 MR. SHEWMON: But it comes against a restriction

6 in the pipe?

7 MR. EBERSOLE: Comes into the socket.

8 MR. SHEWMON: Their thesis might be that no

9 matter how this thing comes through it can't go through

10 that constriction, because it is big enough -- it is bigger

11 than the diameter of the constriction so it has to stay on

12 one side.

13 MR. EBERSOLE: Unless it breaks up as a result

14 of severe impact. That would be the case for a

15 hypothetical pipe break on the downstream side of it -- on

16 the upstream side of it.

17 MR. KERR: None of these did break up?

18 MR. EBERSOLE: None of these loads were ever

19 carried by the valve. That's just continued cyclic action.

20 MR. SHEWMON: That's presumably a rock product

21 that wouldn't break up.

22 MR. EBERSOLE: You can't tell, Paul. We in fact

23 learned there are not in the specs any spec on the function

24 of these valves' closure if you have an abrupt upstream

25 break from them. That's been an issue for some time, 10,

1 15 years.

2 MR. SHEWMON: You don't know if it's white cast

3 iron or stainless steel?

4 MR. KERR: He said there were no specifications

5 on the behavior. He didn't say there were none on the

6 material.

7 MR. EBERSOLE: None on the velocity of impact.

8 MR. WARD: Jesse, are you going to have any

9 presentation by the staff or anything?

10 MR. EBERSOLE: Where is Ron Herndon? He offered

11 to come.

12 Okay, I'm it.

13 MR. WARD: Forest, are you going to have any

14 presentation? Okay. We are finished with the record today,

15 then.

16 (Whereupon, at 5:07 p.m., the record was

17 concluded.)

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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
310TH GENERAL MEETING

DOCKET NO.:

PLACE: WASHINGTON, D. C.

DATE: THURSDAY, FEBRUARY 13, 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) Joel Breitner
(TYPED)
JOEL BREITNER
Official Reporter
ACE-FEDERAL REPORTERS, INC.
Reporter's Affiliation

TMI-2 NUCLEAR CRITICALITY SAFETY

(13 February 1986 ACRS Meeting

Introduction and Overview	F. R. Standerfer
Criticality Analyses	D. Williams
Operational Considerations	Dr. R. A. Knief

February 13, 1986
STATEMENT OF GPU NUCLEAR CORPORATION
ON TMI-2 CLEANUP
BEFORE THE ADVISORY COMMITTEE ON REACTOR SAFETY

Mr. Chairman and Committee Members:

I am Frank Standerfer, Vice President-Director of TMI-2. I appreciate the opportunity to meet with you today to provide a status report on the TMI-2 Cleanup and, more specifically, to discuss our program to ensure sub-criticality of the damaged core during defueling.

With me today are members of the GPU Nuclear and Bechtel staffs who have played key roles in the development and implementation of our criticality prevention program. They are supported by Dr. Westphal of the Oak Ridge National Laboratories who played a key role in the analytical processes which formed the basis of our program. Dr. William Stratton, Member of the GPUN TMI-2 Safety Advisory Board (SAB) who also is here today, performed an independent assessment of our programmatic approach to the maintenance of sub-criticality during defueling.

A number of important milestones in the TMI-2 program have been accomplished since the head was removed from the reactor vessel in July 1984. The reactor vessel plenum has been removed. A system for removing damaged fuel debris has been designed, tested and installed over the open reactor vessel. Defueling operations commenced October 30, 1985. Throughout this period, considerable progress has been made in obtaining

further understanding of the conditions of the damaged core and in planning an increased effort to decontaminate the TMI-2 buildings and systems.

SHOW VIDEOTAPE

As indicated, we are three months into Defueling Operations. That effort is gaining momentum and we expect to ship the first fuel off site under our contract with the DOE this Spring. We expect to complete in-reactor vessel defueling by mid 1987, complete decontamination by July 1988, and complete the Cleanup Project in September 1988. Our budget, project manning, schedule and planning are generally consistent with the information provided to the Nuclear Regulatory Commission in November 1984.

Throughout the TMI-2 Cleanup Project, we have continued to utilize the Technical Assistance and Advisory Group (TAAG), which is funded by DOE and directed by GPUN, DOE, and NRC. It meets monthly for two days at TMI to render its best judgment and advice on the major technical program areas in what is, to a large degree, a Research and Development Program. Further, our Safety Advisory Board (SAB) continues to meet on a quarterly basis to review all aspects of the Project from the standpoint of public and worker safety.

We are taking all practical steps to go forward with the TMI-2 Cleanup as quickly as practicable, consistent with the safety of the public and workers.

The programmatic approach to criticality safety in defueling the TMI-2 reactor vessel has been two-fold. First, a soluble poison concentration was determined that will assure sub-criticality for all credible fuel configurations. Then controls were put in place to maintain the poison concentration above this established minimum value.

Major participants in the criticality safety analysis efforts have included not only the integrated TMI-2 project organization of GPU Nuclear and Bechtel, but also Oak Ridge National Laboratory, a private consultant (Dr. Raymond Murray), Babcock and Wilcox, and Pennsylvania State University. Reviews by the Safety Advisory Board (SAB) and the Technical Advisory and Assistance Group (TAAG) have been supplemented by efforts of our own long-standing General Office Review Board (GORB). Interactions with the NRC Staff have provided additional review which in some cases have included independent confirmatory calculations.

During the presentations to follow, members of the TMI-2 Project Staff will discuss our overall program for preventing criticality while the damage core remains in the reactor vessel and during transfer to the fuel pool for preparation and shipment. The scope of our presentation does not include the fuel shipping program which is the responsibility of the Department of Energy. The damaged core packaged in canisters will be turned over to the Department of Energy at TMI for shipment to the Idaho National Engineering Laboratory for long term storage and disposition.

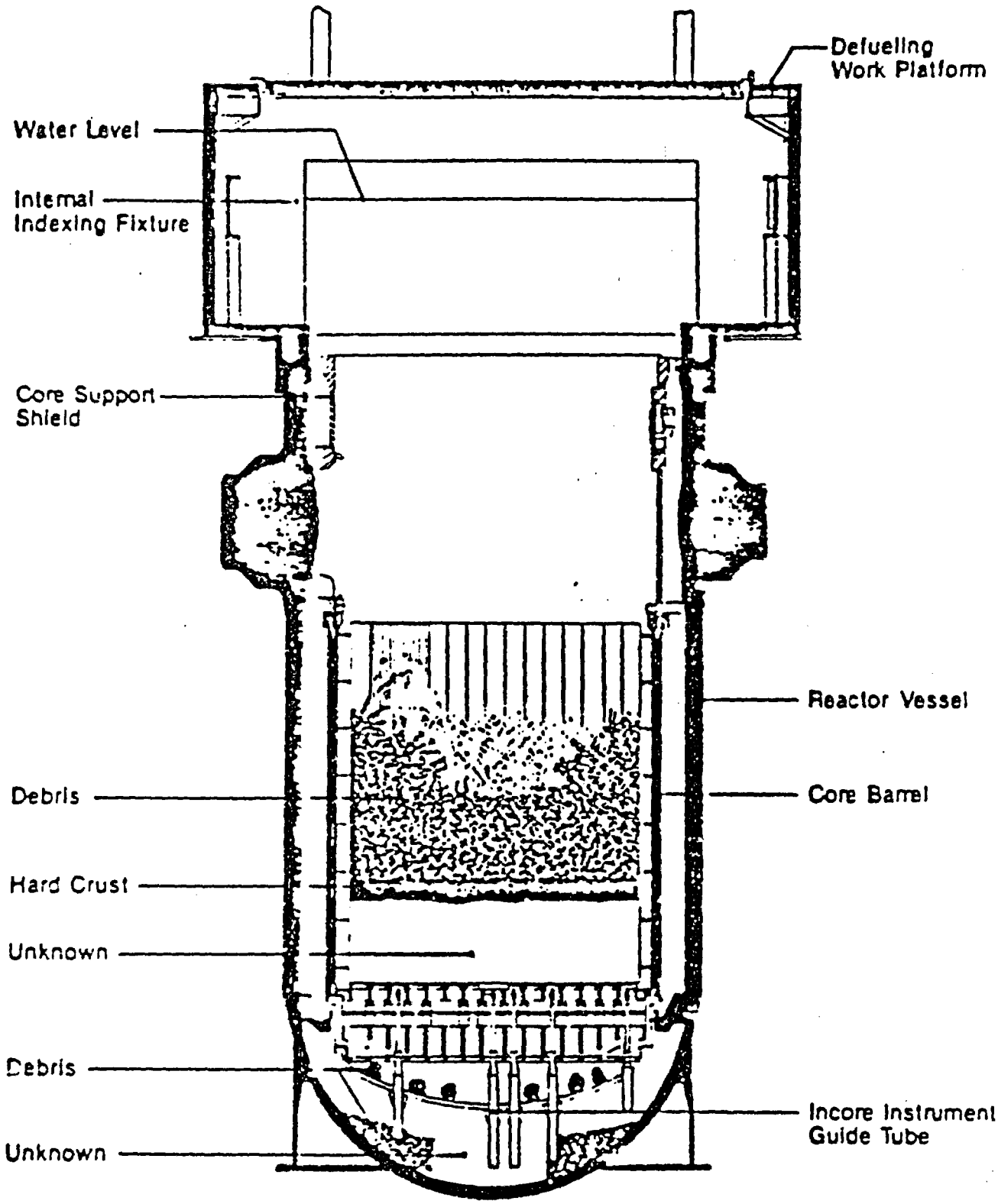
At this time, I would like to introduce Mr. Daniel Williams who will provide an overview of the analytical methods applied to prevention of criticality. He will be followed by Dr. Ronald Knief who will summarize some of the operational aspects of criticality control. Since the two speakers each are presenting some material covered by others at the earlier subcommittee meeting, they may direct some of your questions to appropriate subject matter experts.

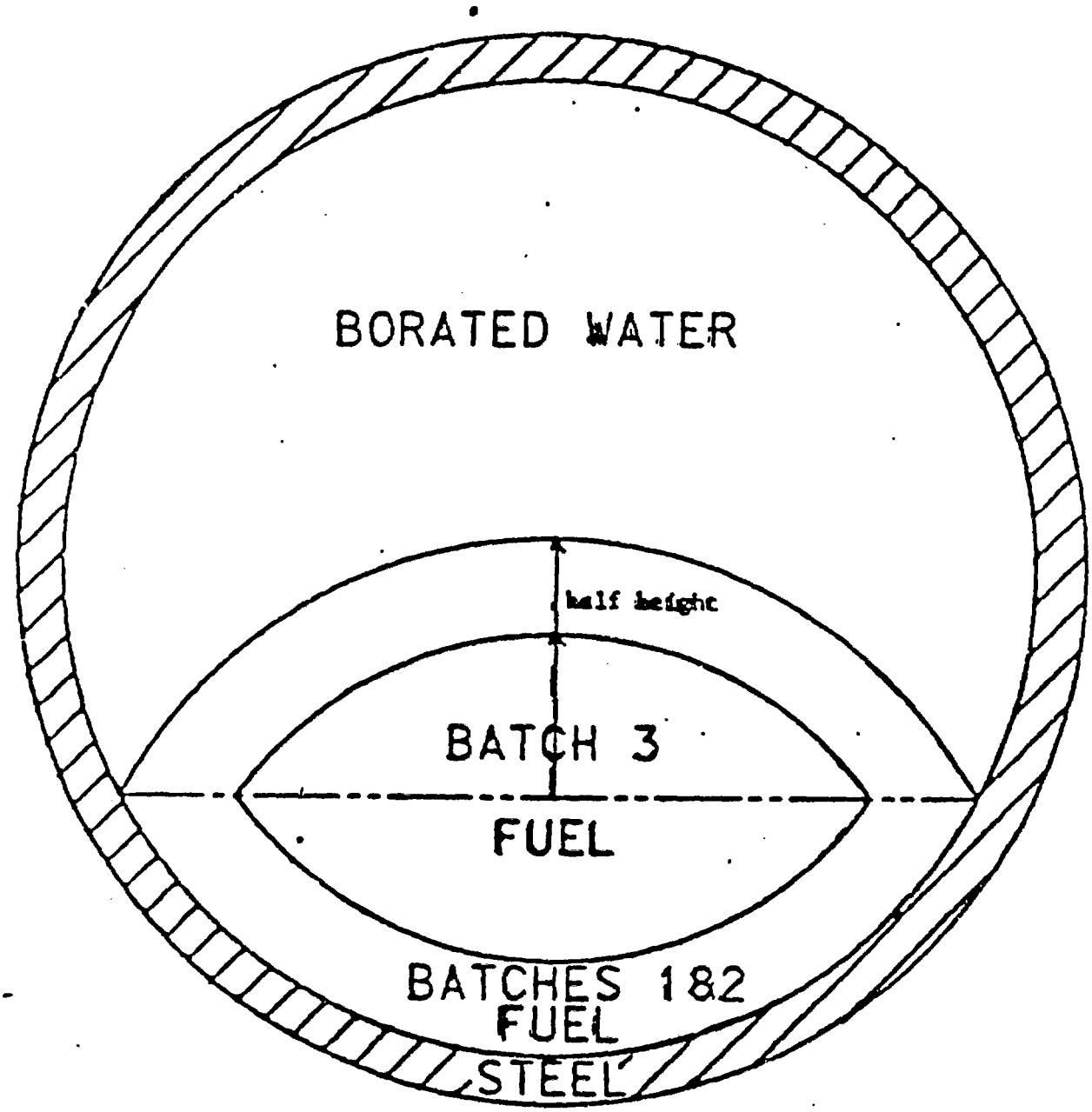
PRESENTATION TO THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

OVERVIEW OF THE TMI-2 CRITICALITY EVALUATIONS TO SUPPORT DEFUELING

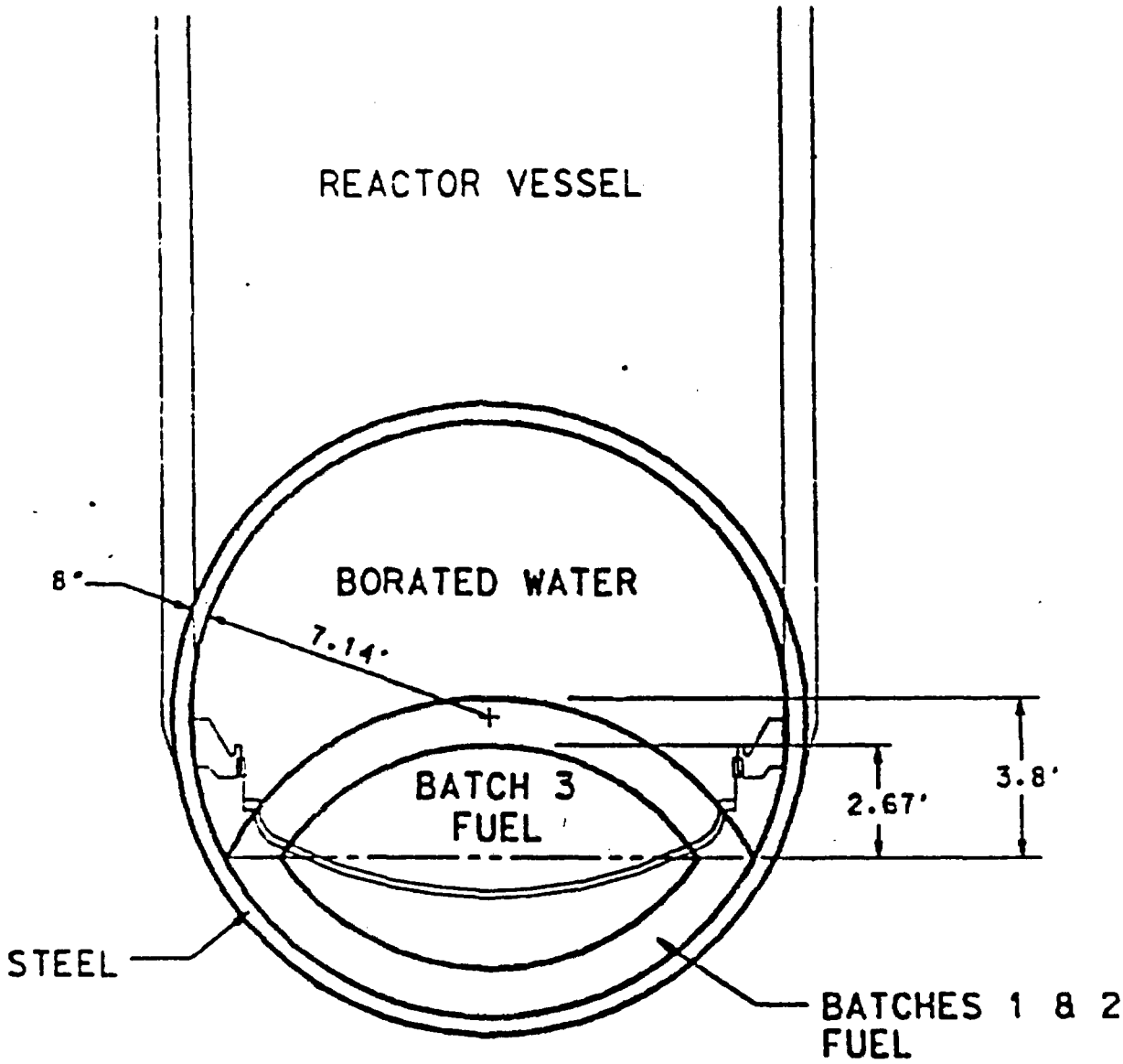
PRESENTED D.S. WILLIAMS

FEBRUARY 13, 1986


Defueling
Work Platform
Water Level
Internal
Indexing Fixture
Core Support
Shield
Reactor Vessel
Debris
Core Barrel
Hard Crust
Unknown
Debris
Incore Instrument
Guide Tube
Unknown



DESIGN BASIS MODEL



LENTICULAR MODEL WITH REACTOR VESSEL

TECHNICAL ISSUES

FUEL COMPOSITION

"FUSED" MASSES

REFLECTION (STEEL)

FUEL BURNUP

COMPUTER CODE BENCHMARKING

RCS MODEL CONSERVATISMS

FUEL CONFIGURATION/ENRICHMENT

MINIMAL BURNUP CREDIT

STEEL REFLECTOR

NO STRUCTURAL OR SOLID POISON MATERIAL

OPTIMIZED FUEL TO MODERATOR RATIO

MINIMUM CREDIBLE FUEL AND MODERATOR TEMPERATURE

CRITERIA FOR SELECTION OF CRITICAL EXPERIMENTS

1. LATTICES DRY
2. BORON UP TO 5000 PPM
3. CRITICAL SYSTEMS
4. LOW EXPERIMENTAL UNCERTAINTY
5. LOW-ENRICHMENT UO_2 FUEL
6. SEVERAL ORGANIZATIONS
7. TRENDS AVAILABLE

MODERATION AND ABSORPTION

$$MR = \frac{\xi \Sigma_s}{\Sigma_a}$$

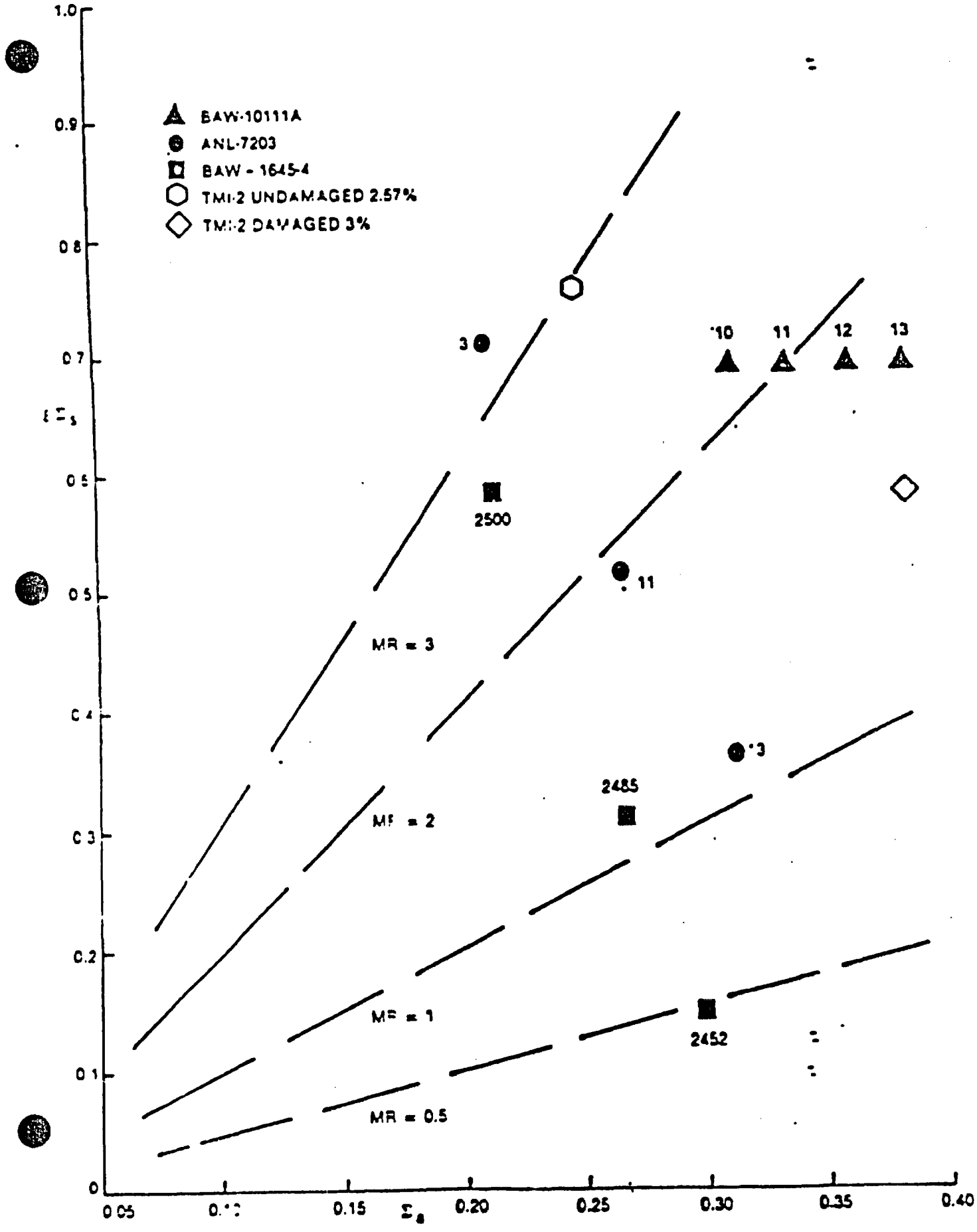
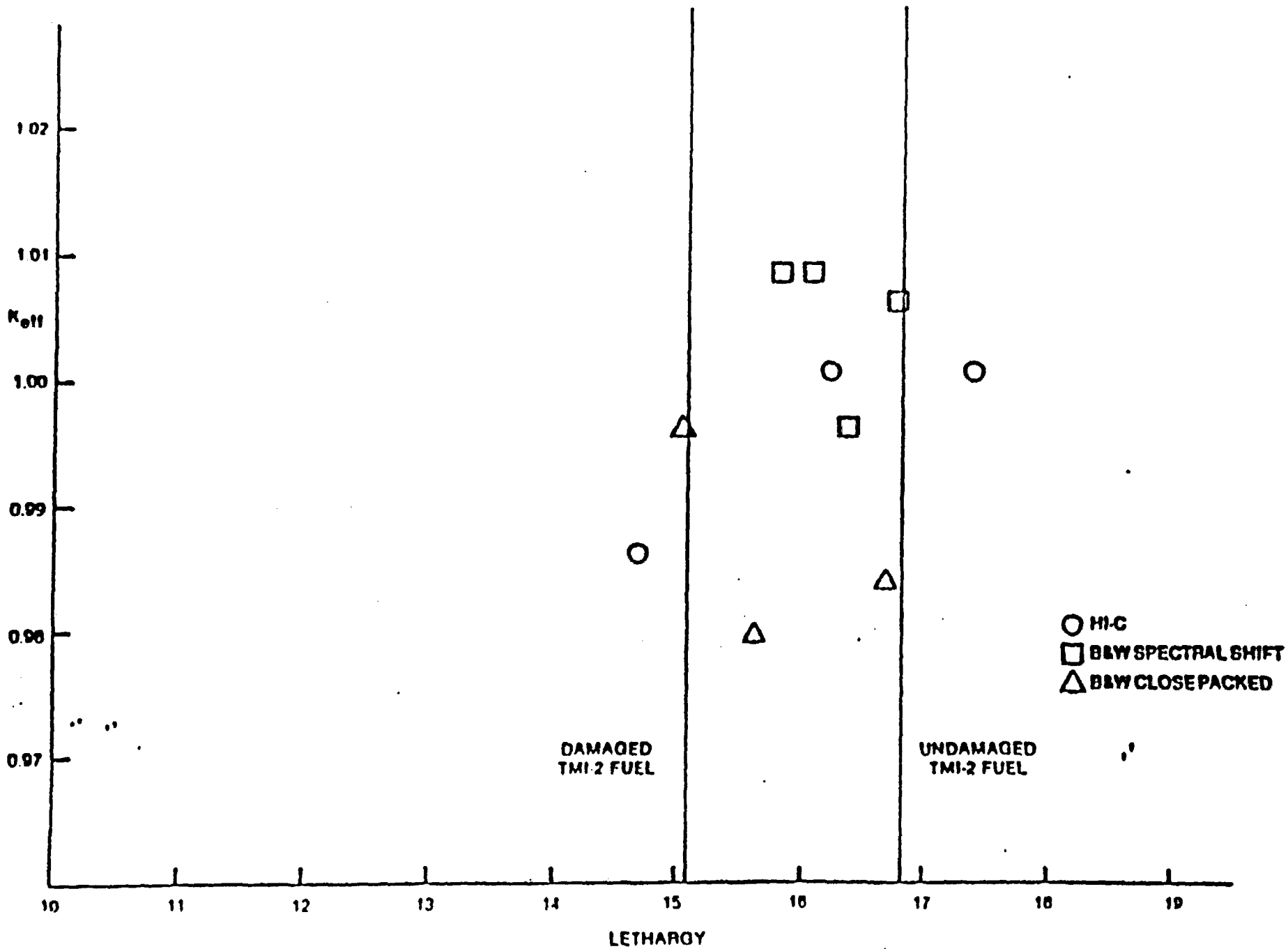


Figure C-4

RELATION OF K_{eff} TO AVERAGE LETHARGY FOR FISSION (KENO)



RCS ANALYSIS RESULTS

BORON CONCENTRATION (PPM)	CALC. K_{EFF}	CALC. K_{EFF} + 2.5% ΔK CODE UNCERTAINTY
4200	0.9688	0.9938
4350	0.9646	0.9896
4750	0.9520	0.9770
5000		<0.970 (ESTIMATED)

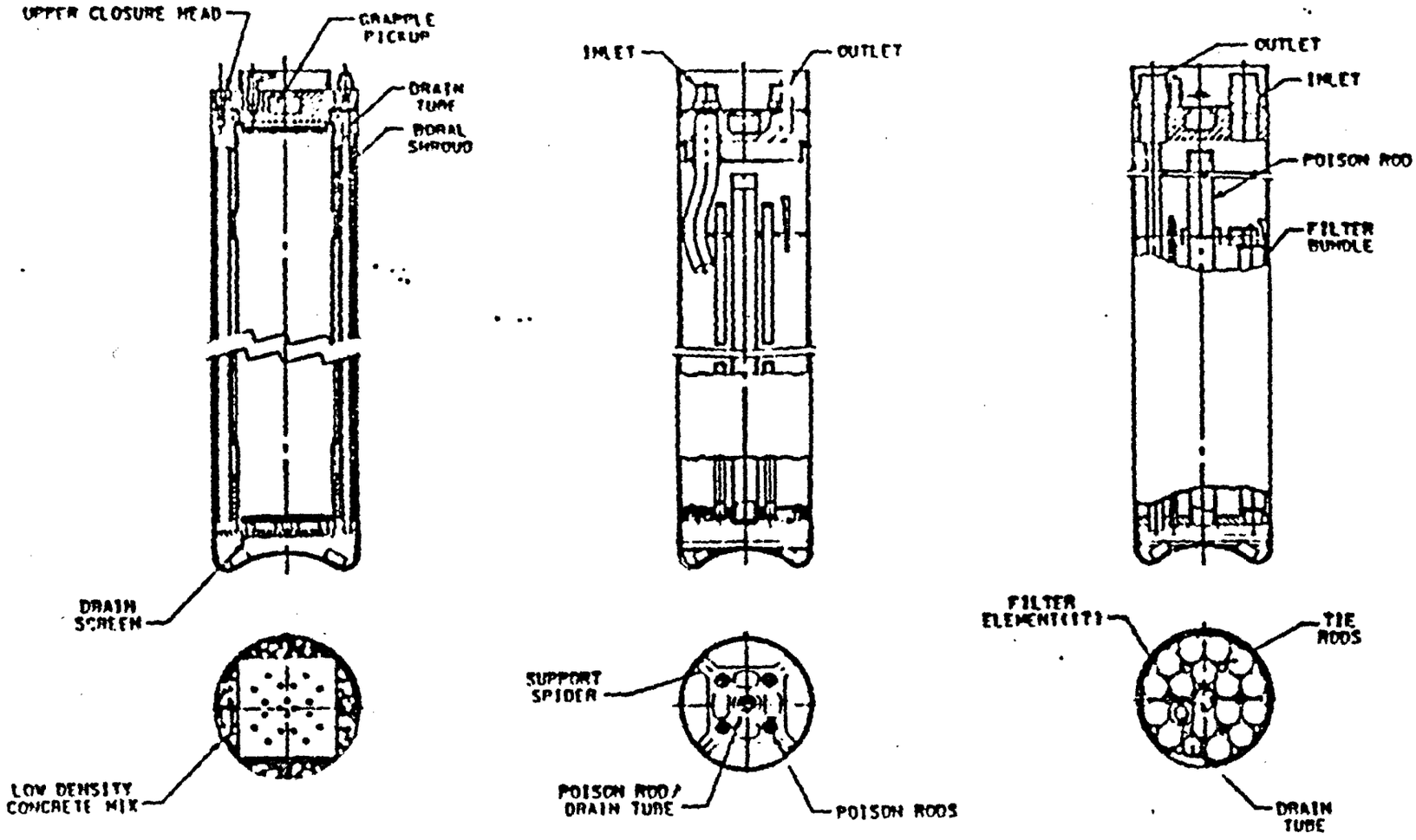
RCS CRITICALITY CONTROL SUMMARY

- SOLUBLE NEUTRON POISON CHOSEN FOR:
  - FAMILIARITY
  - AVAILABILITY
  - VERIFIABILITY
  - COMPATIBILITY WITH PLANT CHEMISTRY
  - OPERATIONAL FLEXIBILITY

- REQUIRED CONCENTRATION DEVELOPED BASED ON HIGHLY CONSERVATIVE MODEL - BOUNDS ALL CREDIBLE FUEL CONFIGURATIONS

- REQUIRED BORON CONCENTRATION - 4350 PPM

  ENSURES $K_{EFF} \leq 0.99$



FUEL CANISTER

KNOCKOUT CANISTER

FILTER CANISTER

TMI-2 DEFUELING CANISTERS

REF DWG- DWG- 11 X 17 "B" SIZE
.DGN 162.07ICANDETAIL.DGN PLOT SCALE =

CANISTER MODEL CONSERVATISMS

1. BATCH 3 FUEL ONLY
2. FUEL CONFIGURATION
3. OMITTED STRUCTURAL MATERIAL
4. CREDIT FOR FIXED POISON ONLY
5. OPTIMAL MODERATION
6. NO WEIGHT RESTRICTIONS ON FUEL QUANTITY
7. 50°F TEMPERATURE
8. 95/95 CONFIDENCE ON B-10 CONTENT OF FIXED POISON

RESULTS OF KENO CRITICALITY CALCULATIONS FOR THE FUEL CANISTER

DESCRIPTION	MAXIMUM K_{EFF}
FUEL CANISTER	
SINGLE, STANDARD CONFIGURATION	0.857
17.3" ARRAY, STANDARD CONFIGURATION	0.877

RESULTS OF KENO CRITICALITY CALCULATIONS FOR THE FILTER CANISTER

DESCRIPTION	MAXIMUM K_{EFF}
FILTER CANISTER	
SINGLE, RUPTURED FILTERS	0.839
17.3" ARRAY, RUPTURED FILTERS	0.867
COMBINED HORIZONTAL/VERTICAL DROP, RUPTURED, WITHOUT SCREENS	0.892

RESULTS OF KENO CRITICALITY CALCULATIONS FOR THE KNOCKOUT CANISTER

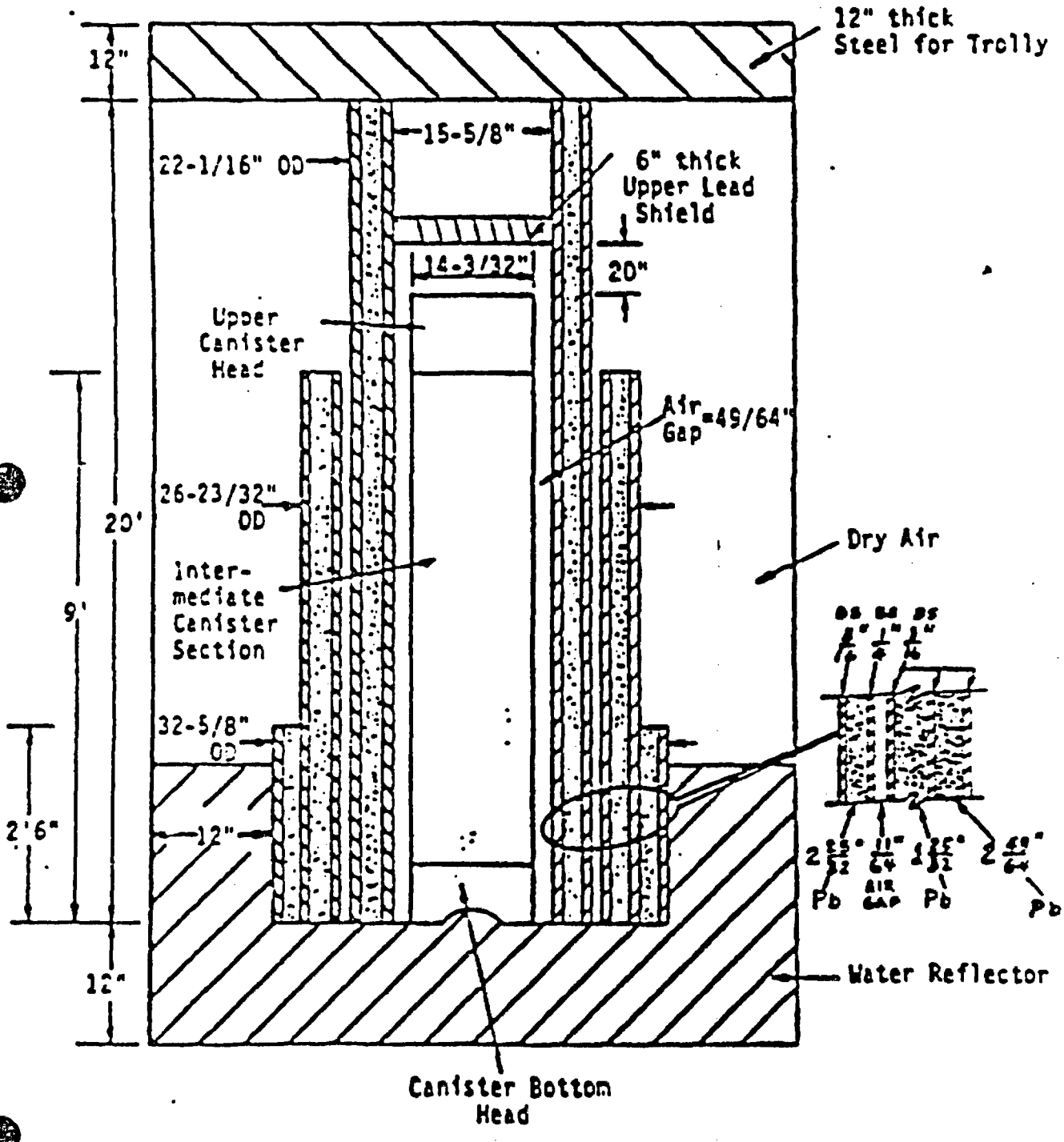
DESCRIPTION	MAXIMUM K_{EFF}
KNOCKOUT CANISTER	
SINGLE, STANDARD CONFIGURATION	0.873
17.3" ARRAY, STANDARD CONFIGURATION	0.915
COMBINED HORIZONTAL/VERTICAL DROP, SINGLE	0.887

CANISTER SUMMARY

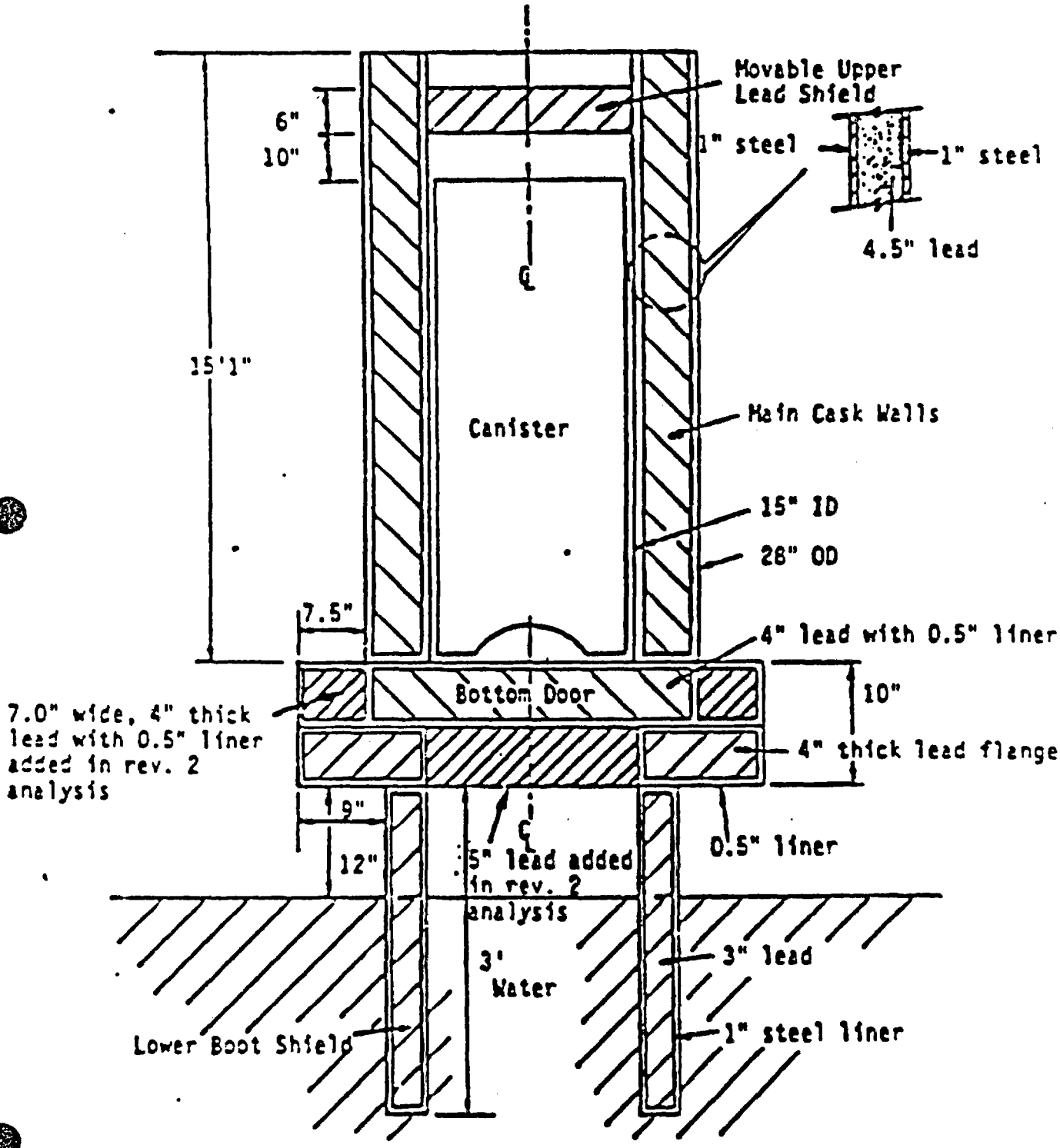
- SINGLE CANISTERS (DAMAGED OR UNDAMAGED) IN UNBORATED WATER - $K_{EFF} < 0.95$

- CANISTER IN STORAGE RACKS WITH MINIMUM CENTER-TO-CENTER SPACING OF 17.3" - $K_{EFF} < 0.95$

Transfer Shield Model



Transfer Cask Model



CTS/FTC RESULTS

CANISTER TRANSFER SHIELD MAX. K_{EFF} = 0.909

FUEL TRANSFER CASK MAX. K_{EFF} = 0.931

BORON DILUTION

PREVENTION

DETECTION

TERMINATION

DILUTION PREVENTION

- Isolate Potential Inlet Points
- Double Barrier Concept
  - Valves
  - Removed Spoolpieces
  - Differential Height
- Administrative Controls

DILUTION DETECTION

- Water Level
  - Control Room
  - Local

- Boron Concentration
  - Boronometers
  - Grab Samples

DILUTION TERMINATION

Isolate Prior to 4350 ppm
- Sample/Process
- Procedures

Procedure Actions
- "Secure" Operations
- Check Boron Concentration
- Review Valves
- Close Valves
- Make-Up with Borated Water

NEUTRON MONITORING

NUCLEAR INSTRUMENTS

- BF3 Detectors
- Ex-Vessel Location
- Operability Verification
- Procedural Requirements

SUBCRITICAL MULTIPLICATION MEASUREMENTS

- Limitations
  - Unknown Geometry
  - High Boron Concentration
  - Low Neutron Source
    - Spont Fiss
    - alpha,n
    - n,gamma
- Evaluation
  - DOT Calculations
  - Workshop
- Conclusions
  - Not Sensitive to Boron Change
  - Don't See Approach-to-Critical
  - Criticality Would Be Observed

SUMMARY

CRITICALITY SAFETY APPROACH

Bounding Poison in Vessel

Geometry/Poison in Canisters

Operating Constraints
- Maintain Poison
- Monitor Poison

P-1 of 6

Fwd to J. B. Martin 6 pages

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

FEB 04 1986

MEMORANDUM FOR: Harold R. Denton, Director, NRR
James M. Taylor, Director, IE
John B. Martin, Regional Administrator, Region V

FROM: Victor Stello, Jr.
Acting Executive Director
of Operations

SUBJECT: STAFF ACTIONS RESULTING FROM THE INVESTIGATION OF THE NOVEMBER 21 SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1 EVENT (NUREG-1190)

An advance copy of the subject report was transmitted to you by memorandum dated January 20, 1986 from the San Onofre Team Leader, Thomas T. Martin. The report documents the Team's efforts in identifying the circumstances and causes of the November 21, 1985 event, together with findings and conclusions which form the basis for identifying follow-on actions.

You will note from the report that the licensee has not completed troubleshooting and the determination of root causes for all equipment failures or malfunctions. Consequently, the results of future troubleshooting or analysis activities may form the basis for additional follow-on actions. The identification of these additional actions is a responsibility of the normal program office. The responsibility for the followup and reporting on the licensee's continued troubleshooting and determination of root cause for equipment failures is Region V.

The purpose of this memorandum is to identify and assign responsibility for generic and plant-specific actions resulting from the investigation of the San Onofre event (documented in NUREG-1190). In this regard, you are requested to review the enclosure which specifies staff actions resulting from the investigation of the San Onofre event. You are requested to determine the actions necessary to resolve each of the items in your area of responsibility and, where appropriate, identify additional staff actions or revisions as our review and understanding of this event are refined. Plant-specific actions required for plant restart should receive priority attention.

In view of the importance of this subject, I intend to closely monitor the resolution of these items. By March 1, 1986, please provide a written summary of the schedule and status of each item within your responsibility listed in the enclosure or that you have identified. Further, I request that you prepare a written status report on the disposition of your items (and anticipated actions for uncompleted items) within three to six months. Every effort should be made to dispose of these items promptly.

The enclosure is based on the Team's report and its presentation to the Commission on January 22, 1986. Accordingly, it does not include all licensee actions, nor does it cover NRC staff activities associated with normal event followup such as authorization for restart, plant inspections, or possible enforcement items. These items are expected to be defined and implemented in a routine manner. Overall lead responsibility for staff actions relating to facility restart is separate from this effort and rests with Region V. Additionally, RV is responsible for coordinating and promptly communicating the staff's requirements which must be resolved before operations at San Onofre may be resumed.

Victor Stello, Jr.
Acting Executive Director
for Operations

Enclosure:
As stated

cc w/enclosure:
J. Davis, NMSS
T. Murley, RI
J. N. Grace, RII
J. Keppler, RIII
R. Martin, RIV

STAFF ACTIONS RESULTING FROM THE INVESTIGATION OF THE NOVEMBER 21 SONGS-1 EVENT

(Reference: NUREG-1190)

1. Item: Adequacy of feedwater check valves to perform safety function. (References: Commission briefing, Sections 6.2.4, 6.4, 6.7, and Principal Finding)

Action	Responsible Office	Category
Implement and coordinate the staff and industry actions necessary to assure the reliability of safety-related check valves. Other offices to assist as requested. Areas to be evaluated include: - licensee's engineering report on root cause analysis and proposed corrective actions - adequacy of check valve design for this application - adequacy of Inservice Testing (IST) Program and procedures to detect degraded and failed valves - adequacy of check valves (and related testing programs) in other systems such as RHR system	IE	Plant-specific Generic

2. Item: Completeness of resolved USI A-1, "Water Hammer". (References: Finding numbers 1, 2, 3, 8 and 9)

Action	Responsible Office	Category
Assess the need to re-evaluate USI A-1 to specifically address the potential for and prevention of condensation-induced water hammers in feedwater piping (assume the issue concerning check valve integrity will be resolved in item 1).	NRR	Generic

3. Item: Adequacy of San Onofre Unit 1 design.
(Commission briefing, Finding numbers 11 and 13)

Action	Responsible Office	Category
Implement and coordinate the staff's actions to re-evaluate the following San Onofre design features:	NRR	Plant-specific
- manual loading of the diesel generators following a loss of power event		
- manual actuation of steam line isolation valves and assurance of steam generator availability to remove decay heat		
- lack of steam generator blowdown status in control room		
- adequacy of the licensee's design change to eliminate spurious SI indication on loss of power		

4. Item: Adequacy of post-trip review.
(References: Sections 6.6 and 7.2.2.4 and Finding number 17)

Action	Responsible Office	Category
a. Evaluate NRC requirements for ensuring that sufficient event data are retrievable to accurately reconstruct the event following a loss of offsite power.	NRR	Generic
b. Evaluate the licensee's process for post-trip review and evaluation, including the thoroughness of review and oversight provided by the onsite and offsite nuclear safety review groups.	Region V	Plant-specific

5. Item: Adequacy of licensee's recordkeeping practices.
(References: Section 6.5 and Finding number 20)

Action	Responsible Office	Category
Evaluate the adequacy of the licensee's maintenance records.	Region V	Plant-specific

6. Item: Adequacy of operator training and/or procedures.
(References: Section 7 and Finding numbers 14, 15 and 16)

Action	Responsible Office	Category
Review the implementation of the training program regarding operator understanding and actions in the area of electrical systems, and invoking technical specification action statements.	Region V	Plant-specific

7. Item: Adequacy of emergency notifications and NRC response.
(References: Section 7.3 and Finding number 22)

Action	Responsible Office	Category
a. Verify the adequacy of the licensee's procedures and training for reporting of events to NRC Operations Center.	Region V	Plant-specific
b. Evaluate the need for changes in NRC policy or guidance regarding: the use of the ENS line; the use of NRC personnel as ENS communicators; and possible approaches to improve the ability to determine the overall plant status.	IE	Generic

8. Item: Significance of backlog of license amendments. (Reference: Commission briefing)

Action	Responsible Office	Category
Evaluate whether a backlog of license amendments and technical specification changes contributed to delays in approving the licensee's IST program.	NRR	Plant-specific

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

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MURRAY R. EDELMAN
VICE PRESIDENT
NUCLEAR

February 12, 1986
PY-CEI/NRR-0437 L

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
Seismic Event Evaluation Report

Dear Mr. Denton:

My letter to you dated February 5, 1986, committed to provide a report on our response and activities related to the earthquake which occurred in the vicinity of the Perry Nuclear Power Plant. Enclosed is the Cleveland Electric Illuminating Company (CEI) report titled "January 31, 1986 Earthquake-Seismic Event Evaluation" for the Perry Nuclear Power Plant. This document has been prepared by CEI and our consultants following a thorough and detailed assessment of the plant response to the January 31, 1986 earthquake.

This report demonstrates the appropriateness of the seismic design for the Perry Nuclear Power Plant. Although this recent event provides an additional "data point" for historical seismic event activity, it will not alter any of the design criteria or licensing basis.

We believe that this report provides the information necessary to support the staff's review and we are available to meet with your staff as necessary. Should you or your staff have any questions please feel free to call.

Very truly yours,

Murray R. Edelman

Murray R. Edelman
Vice President
Nuclear Group

MRE:L

cc: Jay Silberg, Esquire
John Stefano
J. Grobe
D. Eisenhut
R. Bernero
W. Butler
G. Lainas
J. Keppler
C. Norelius
C. Paperiello
R. Knop

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

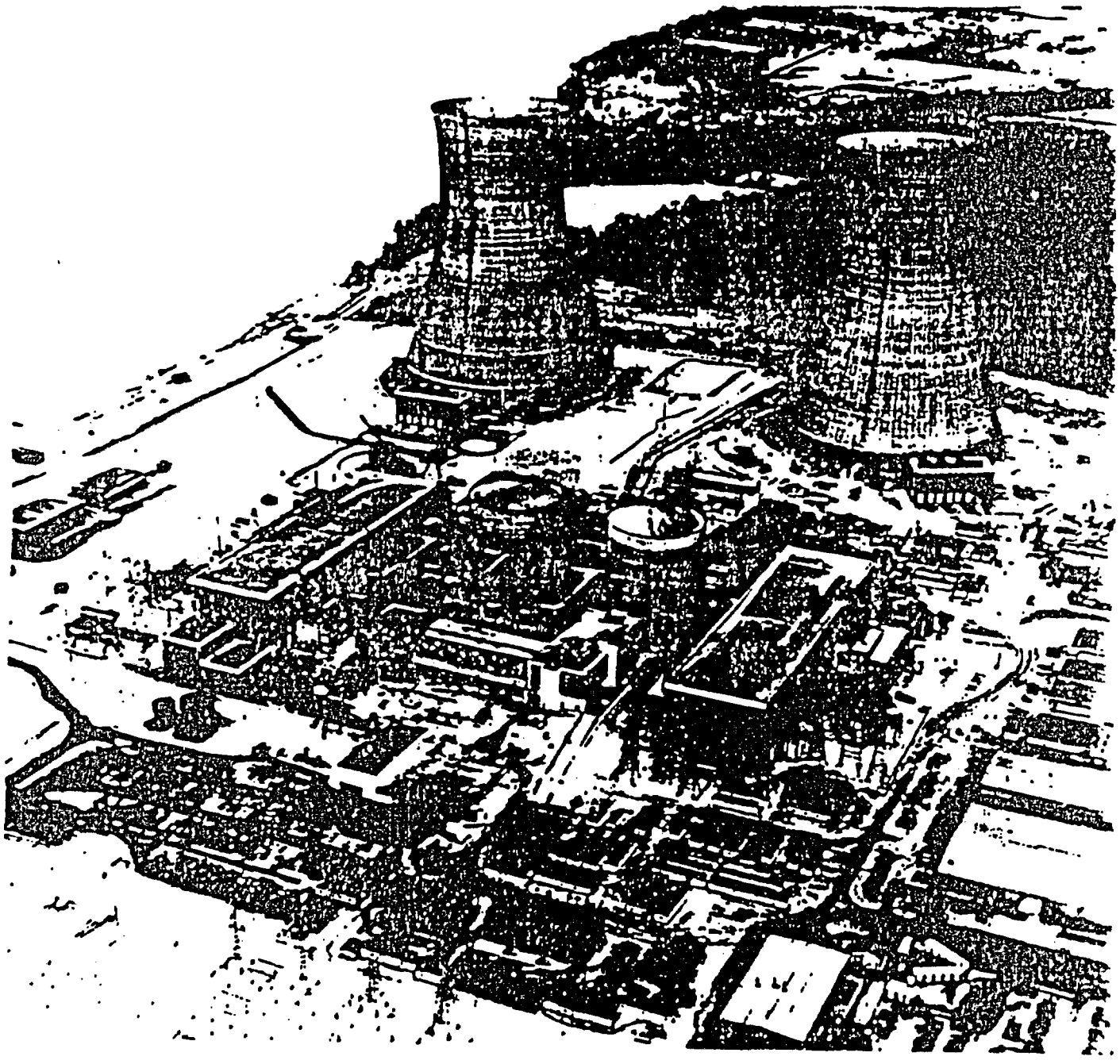
  - NRC STAFF/CONSULTANTS PRELIM. FINDINGS/DATA NEEDS 2/21/86

    - PERRY SPECIFIC DESIGN BASIS

    - OTHER RECOMMENDED GENERIC ACTIVITIES

  - SSER ISSUED 3/7/86

  - PERRY 1 LICENSING TARGET 3/14/86



PERRY POWER PLANT

JANUARY 31, 1986 EARTHQUAKE
SEISMIC EVENT EVALUATION

SEISMIC EVENT EVALUATION

REPORT

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 which 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.

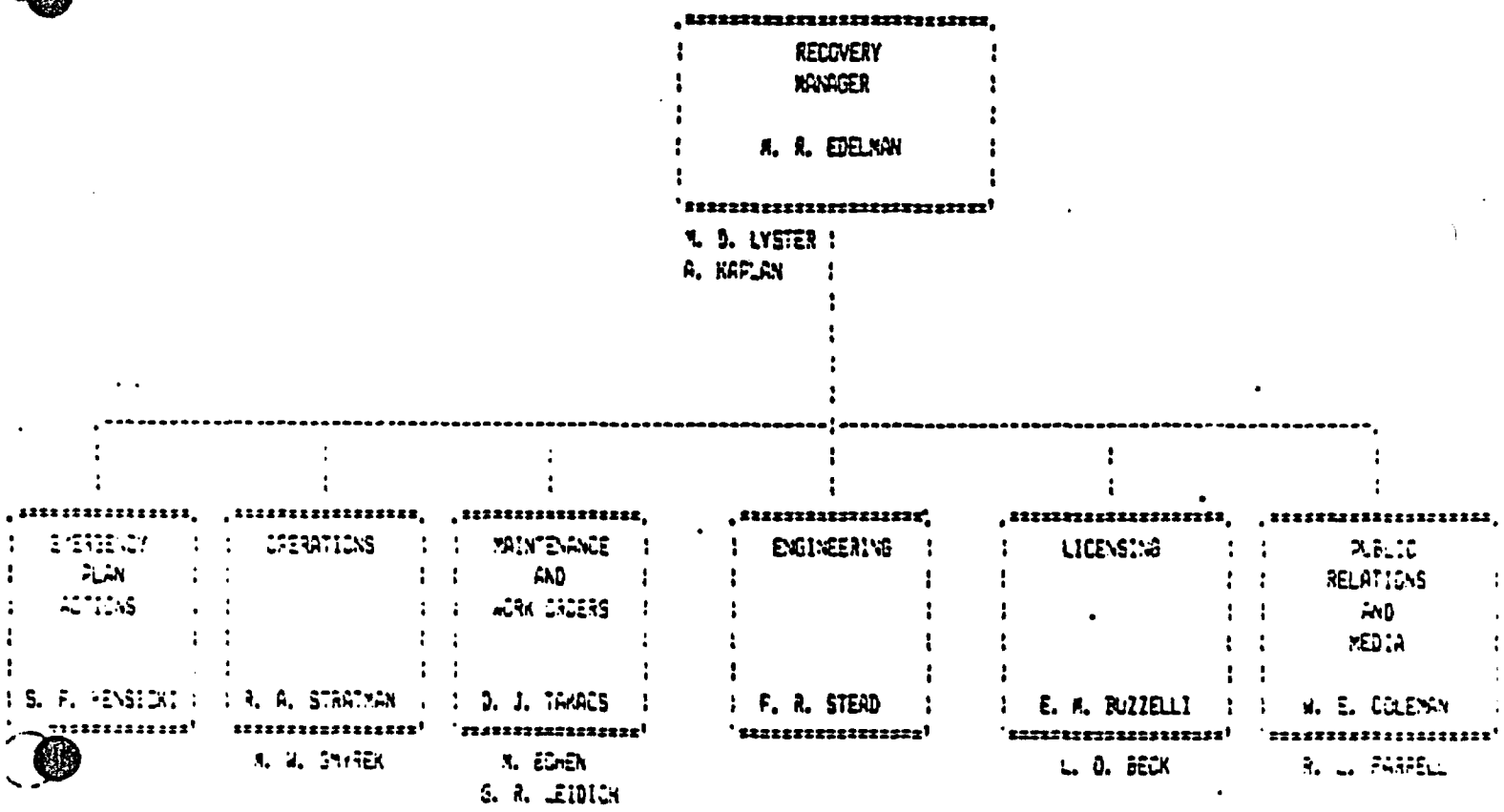
January 31, 1986 Earthquake

Chronological Summary of Events

Time of Occurrence	Event
11.46:42.3 (USGS data)	Seismic event occurs
1148	Control room reports noise & 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

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° 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:

- o Chatham sag faults
- o Peck fault, Howell-Northville anticline faults
- o Bowling Green fault
- o Anna Ohio faults
- o Cincinnati arch faults
- o Eastern Ohio faults
- o Western New York faults
- o 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:

- o test borings (maximum depth 730')
- o 42" drilled exploratory shafts
- o in-site testing, plate load tests
- o permeability determinations
- o piezometer installations
- o seismic analyses
- o seismic refraction and seismic shear wave determinations
- o 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:

- detailed mapping of tunnel walls
- reconnaissance of lake bottom
- lake shore reconnaissance
- exploratory borings
- borehole logging, offshore and onshore magnetic surveys
- review of existing geophysical data
- isotopic analyses of Lake Erie and fault seepage water

Studies to date fault included:

- x-ray diffraction
- clay mineralogic analysis
- microcrack
- consolidation of gouge

Miscellaneous studies included:

- o borehole stress
- o structure contour maps
- o 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:

- o faults are genetically related; same fault or an echelon
- o faults confined in Chagrin shale; limited lateral and vertical extent
- o date of last motion is Pleistocene or older
- o motion sense indicates faults originated in northwest directed stress field, approximately 90° from present stress field
- o possible mechanisms of nontectonic glacial origin include ice sheet traction, differential downwarp, differential rebound, surficial stress relief ("pop up")
- o 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°51.24'	81°11.46'	5	$2.7M_{bLg}$ (2)
31-JANUARY-1986	16:46:42.3	41°38.84'	81°05.30'	10	$4.9\ M_b$ (3)
01-FEBRUARY-1986	18:54:49.7	41°38.39'	81°09.99'	3.1	--
02-FEBRUARY-1986	03:22:49.1	41°38.37'	81°09.81'	2.3	--
03-FEBRUARY-1986	19:47:19.6	41°39.19'	81°10.27'	9	--
05-FEBRUARY-1986	06:34:02.4	41°39.93'	81°09.11'	6	--
06-FEBRUARY-1986	18:36:22.6	41°38.66'	81°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)

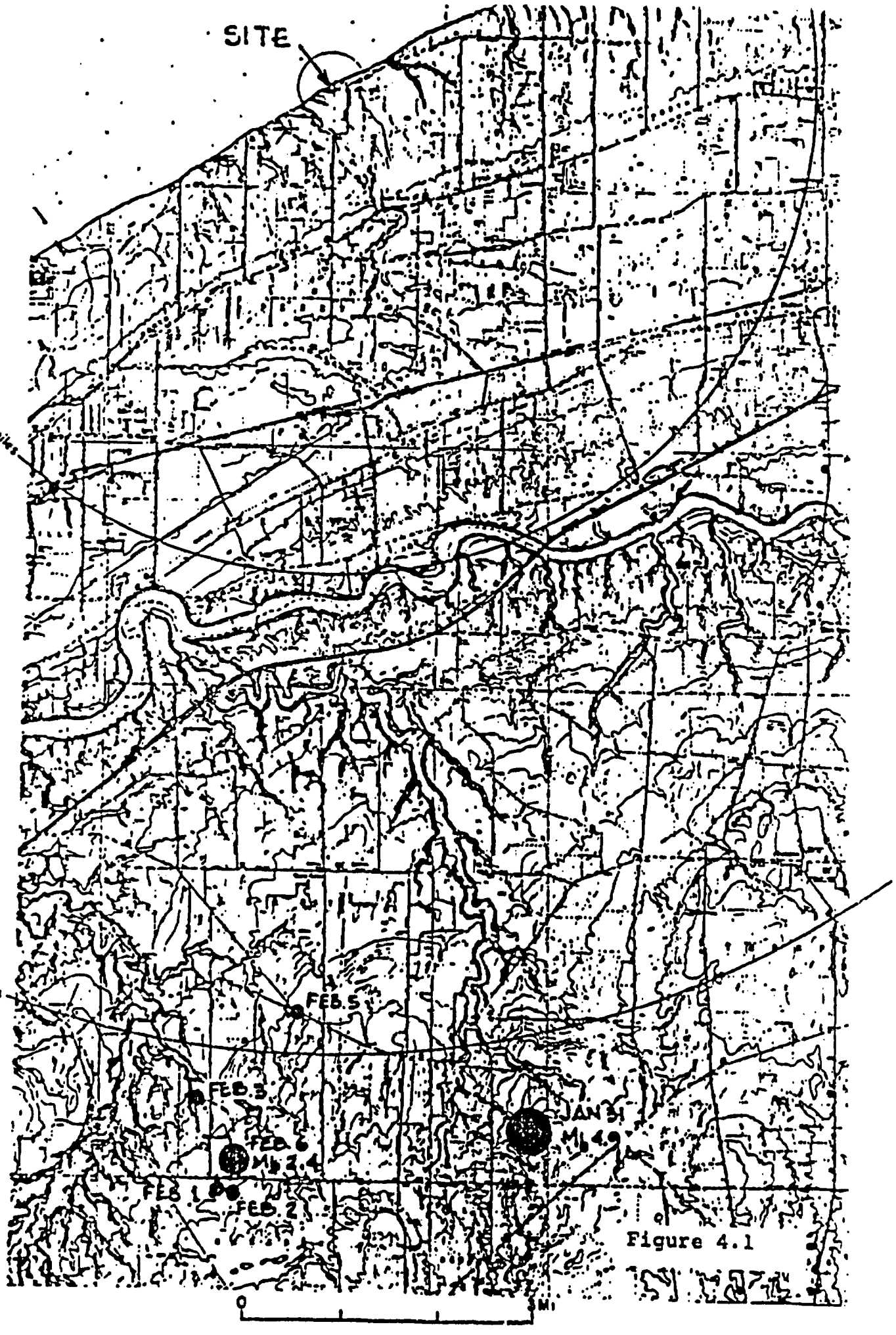
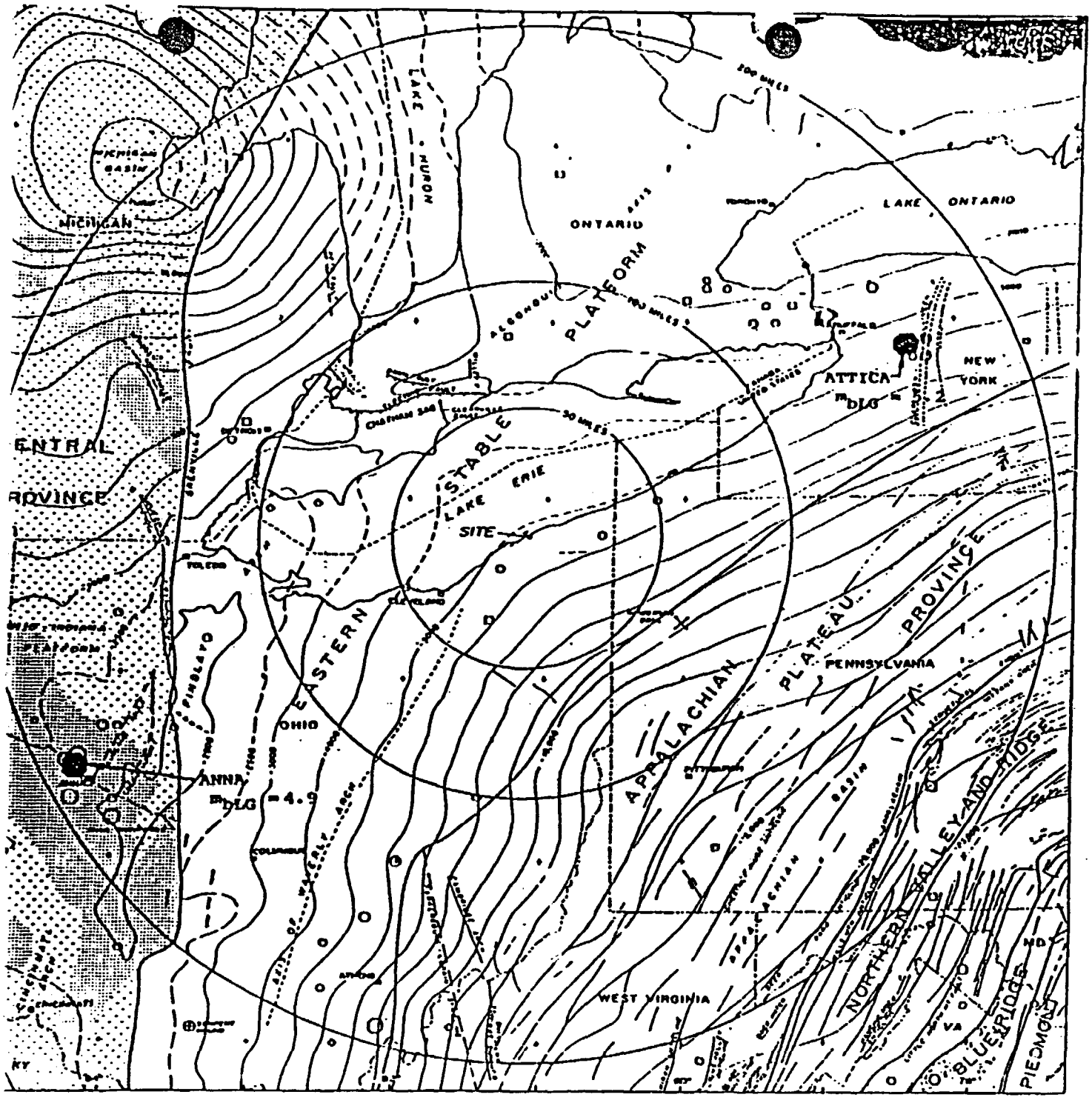
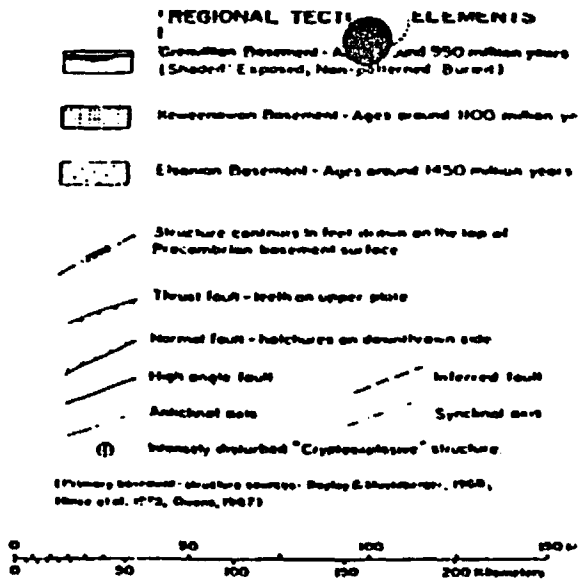

SITE
5 miles
10 miles
FEB.5
FEB.3
FEB.6
Mb 2.4
FEB.1
FEB.2
JAN 31
Mb 4.9
0
Mi

Figure 4.1





REGIONAL TECTONIC PROVINCES

PROVINCE BOUNDARY

EARTHQUAKES

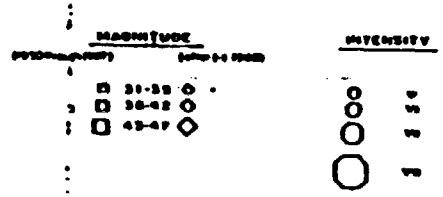


Figure 4.2

Am. 10 (11-29-82

PERRY NUCLEAR POW
THE CLEVELAND EL
ILLUMINATING CO

Regional Tectonics -
Earthquake Tectonic Prov

Figure 2.5-59

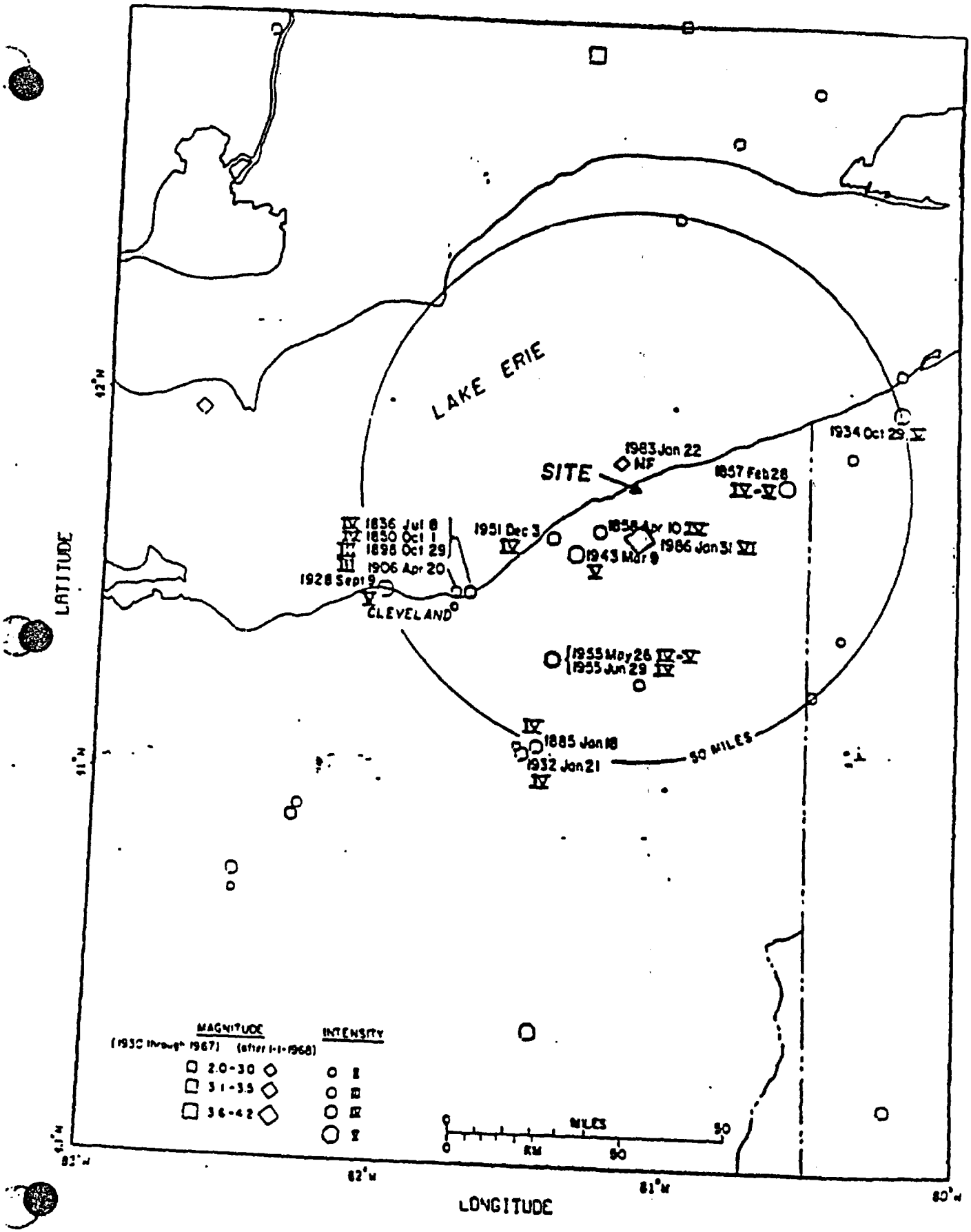

LAKE ERIE
SITE
1983 Jan 22
NF
1857 Feb 28
IV-V
1934 Oct 29
V
1836 Jul 8
1850 Oct 1
1898 Oct 29
1906 Apr 20
1928 Sept 9
1951 Dec 3
IV
1943 Mar 9
V
1986 Jan 31
VI
CLEVELAND
1955 May 26
IV-V
1955 Jun 29
IV
1885 Jan 18
1932 Jan 21
50 MILES
MAGNITUDE
(1930 through 1967)
(after 1-1-1968)
2.0-3.0
3.1-3.5
3.6-4.2
INTENSITY
MILES
KM
LATITUDE
LONGITUDE
83°W
82°W
81°W
80°W

Figure 4.3

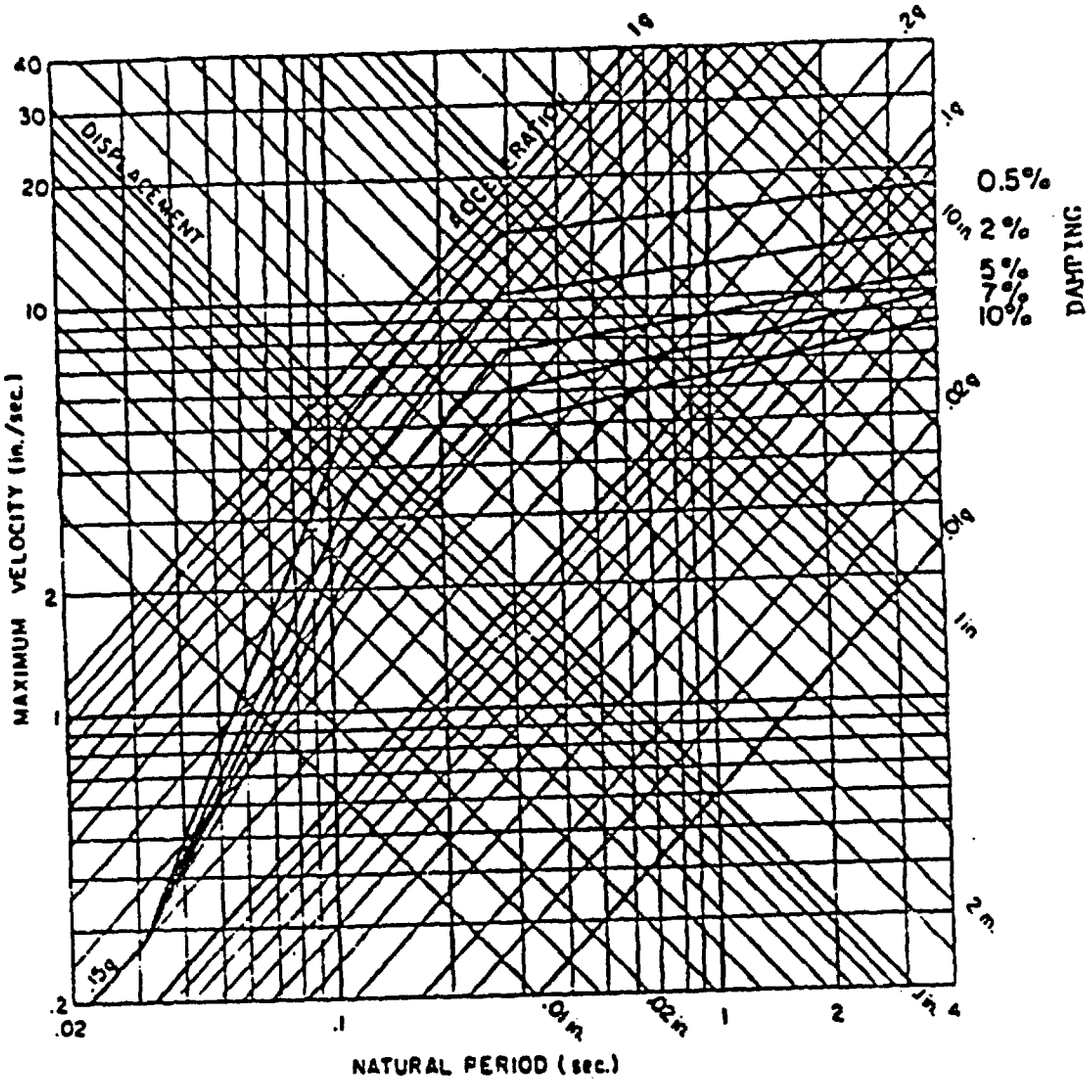


Figure 4.4

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Safe Shutdown Earthquake
Design Response Spectra -
Vertical Motion

Figure 3.7-2

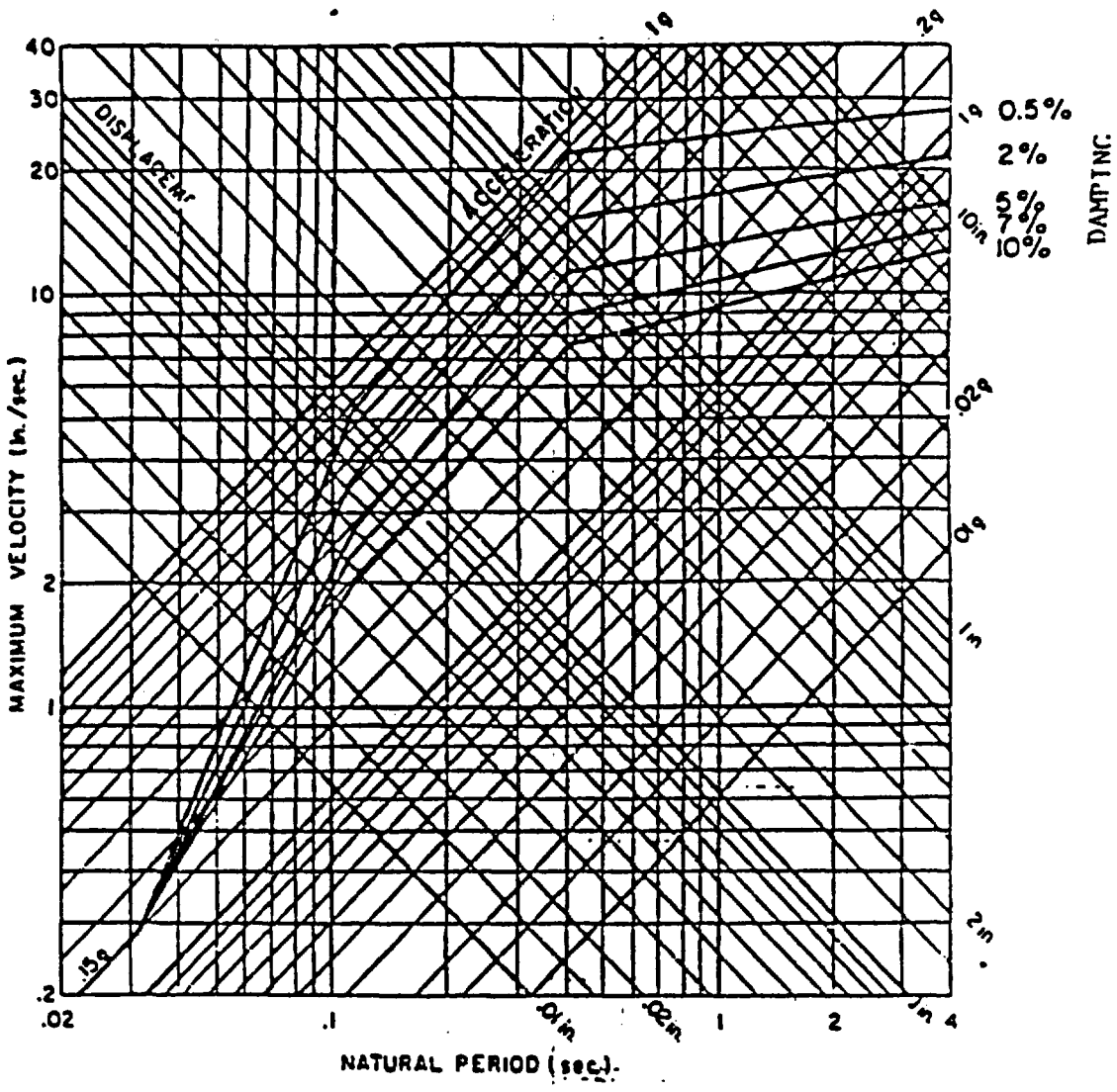
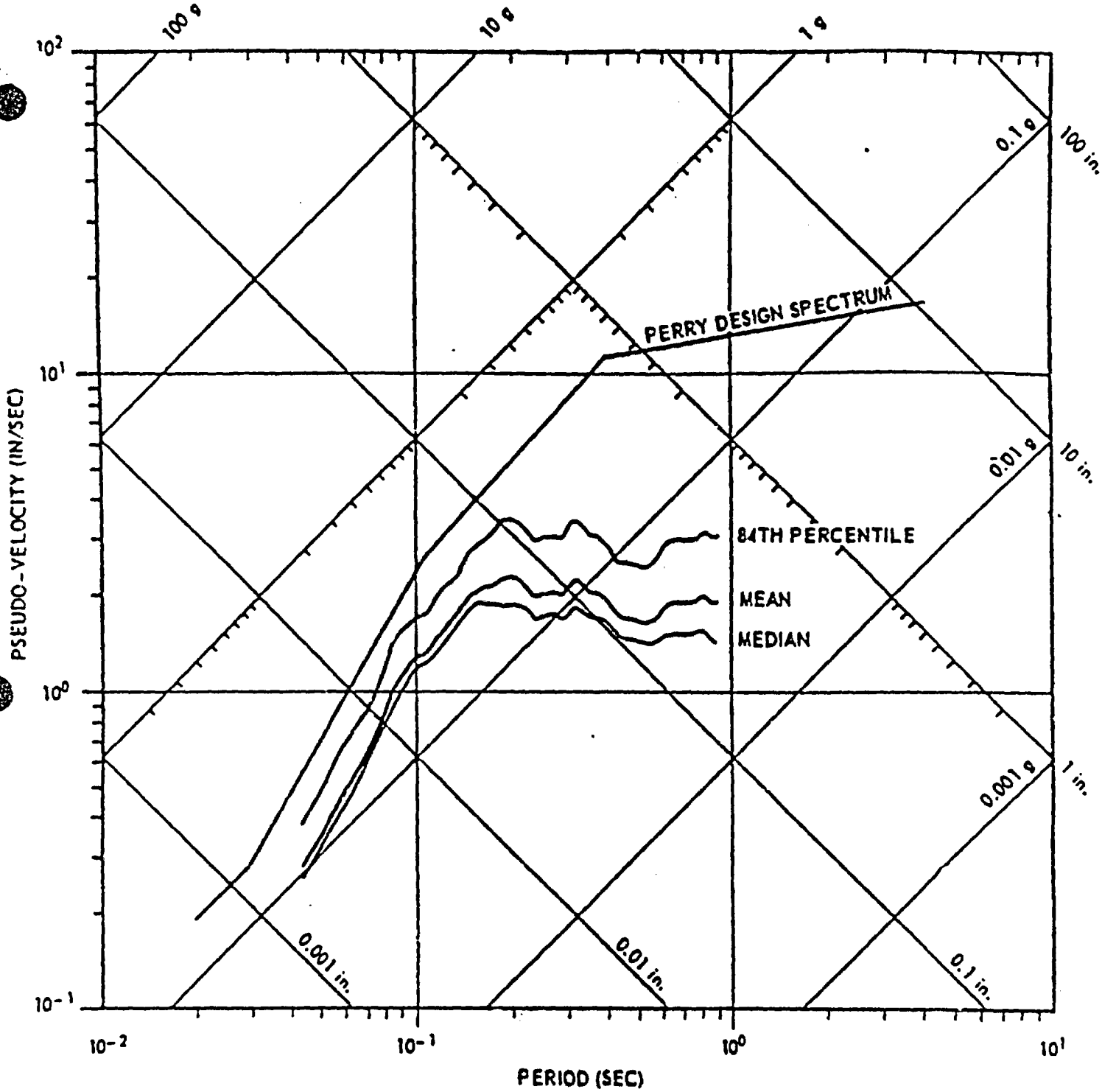


Figure 4.5

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)

$m_{bLG} = 5.3 \pm .5$

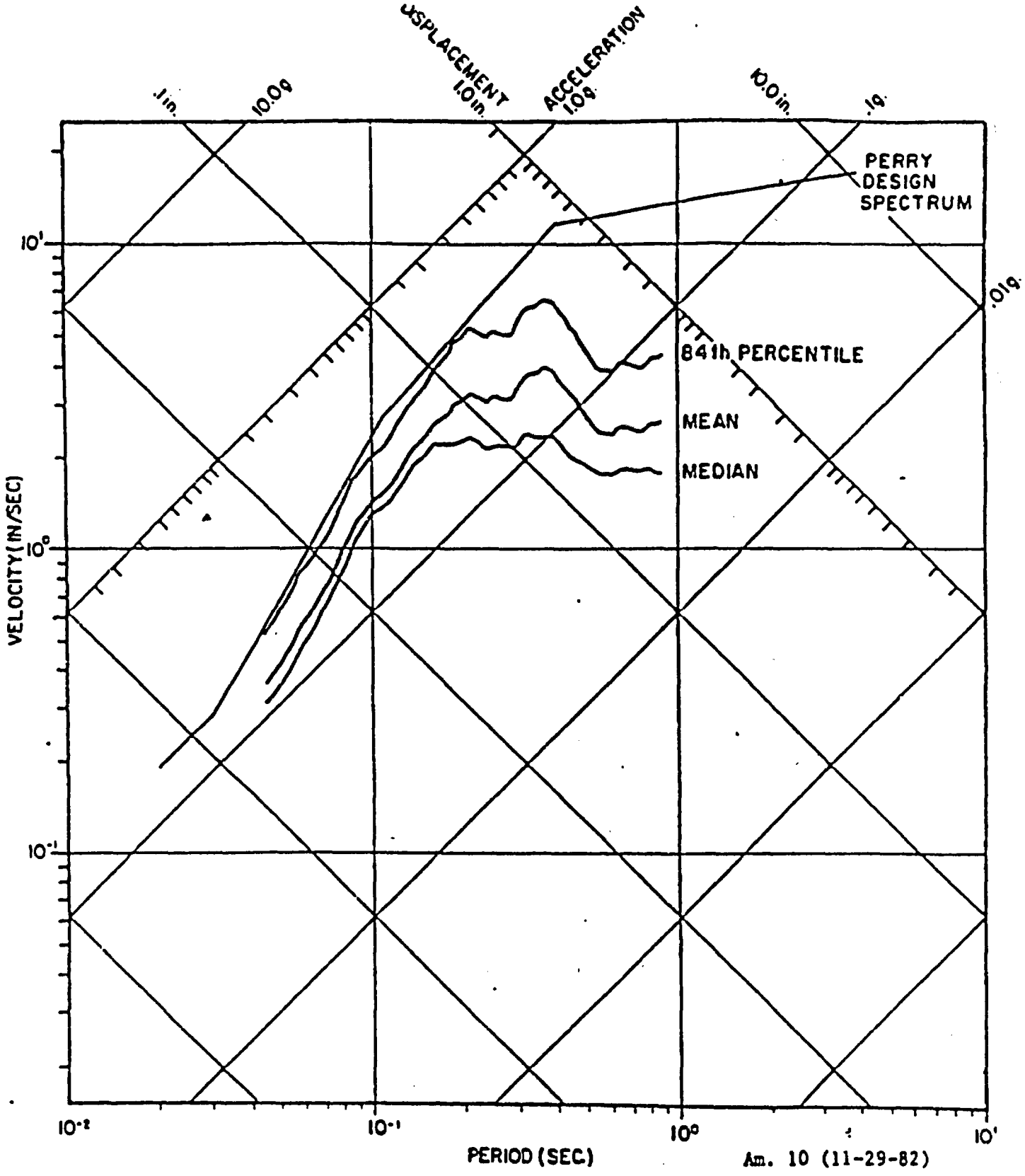
Am. 10 (11-29-82)

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Figure 230.6-2

Q&R 2.5-37

Figure 4.6



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

Figure 4.7

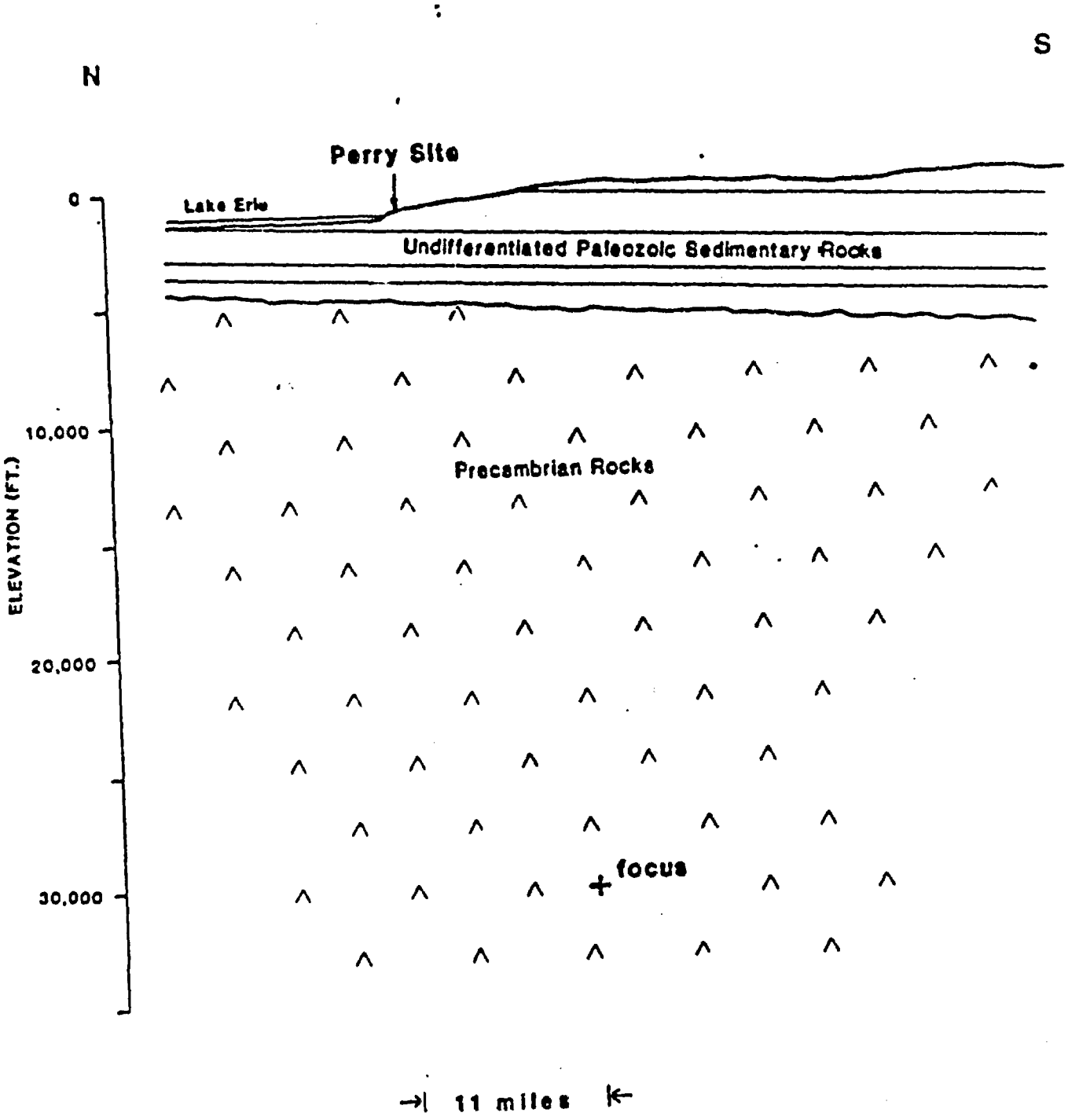

N
S
Perry Site
Lake Erie
Undifferentiated Paleozoic Sedimentary Rocks
Precambrian Rocks
focus
0
10,000
20,000
30,000
ELEVATION (FT.)
11 miles

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 acceleration (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 51

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

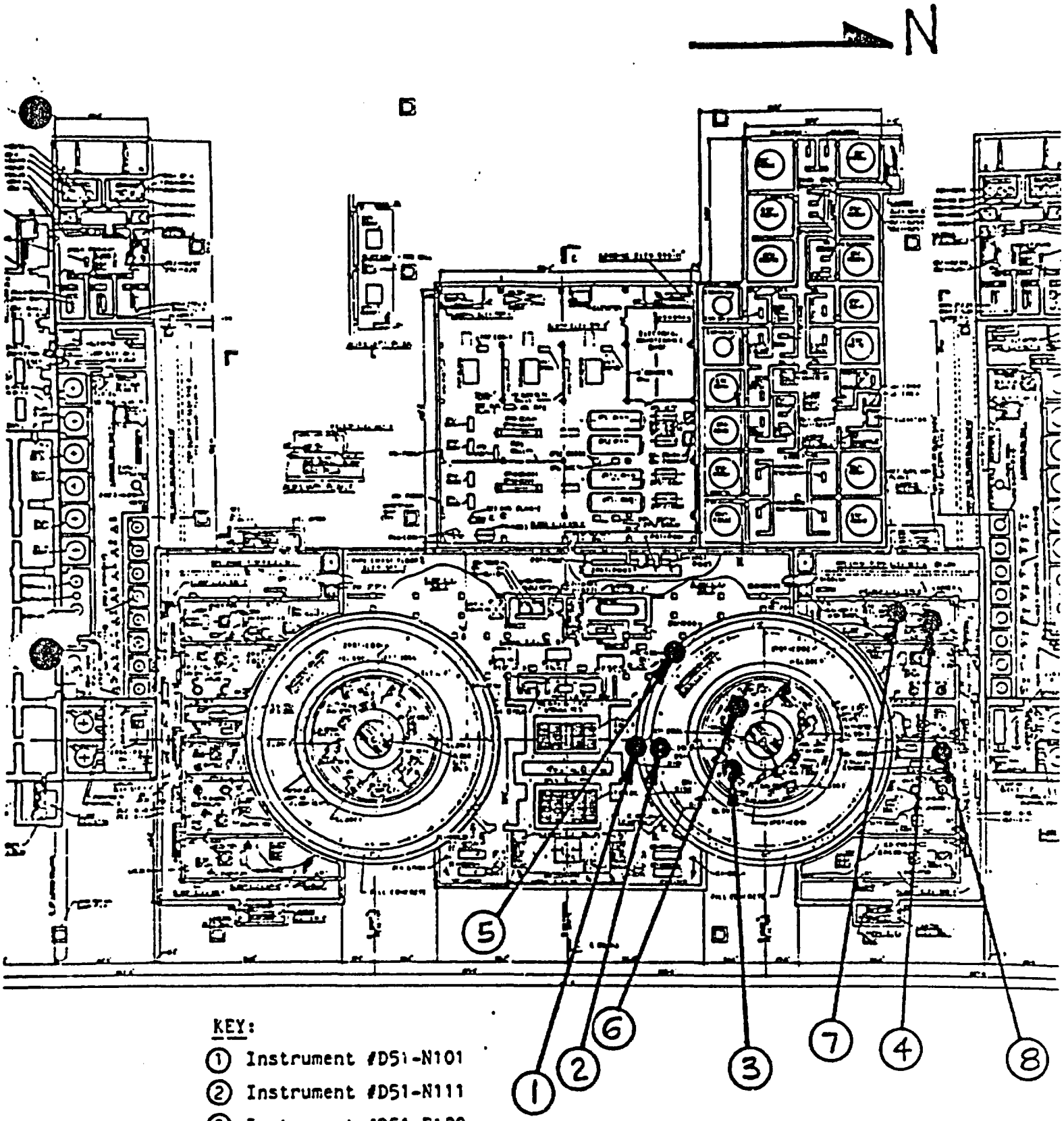
1 Triaxial Time-History Accelerograph
2 Triaxial Peak Accelerograph
3 Triaxial Response Spectrum Recorder

PERRY NUCLEAR POWER PLANT UNIT NO. 1
SEISMIC MONITORING INSTRUMENTATION
TABLE 51

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

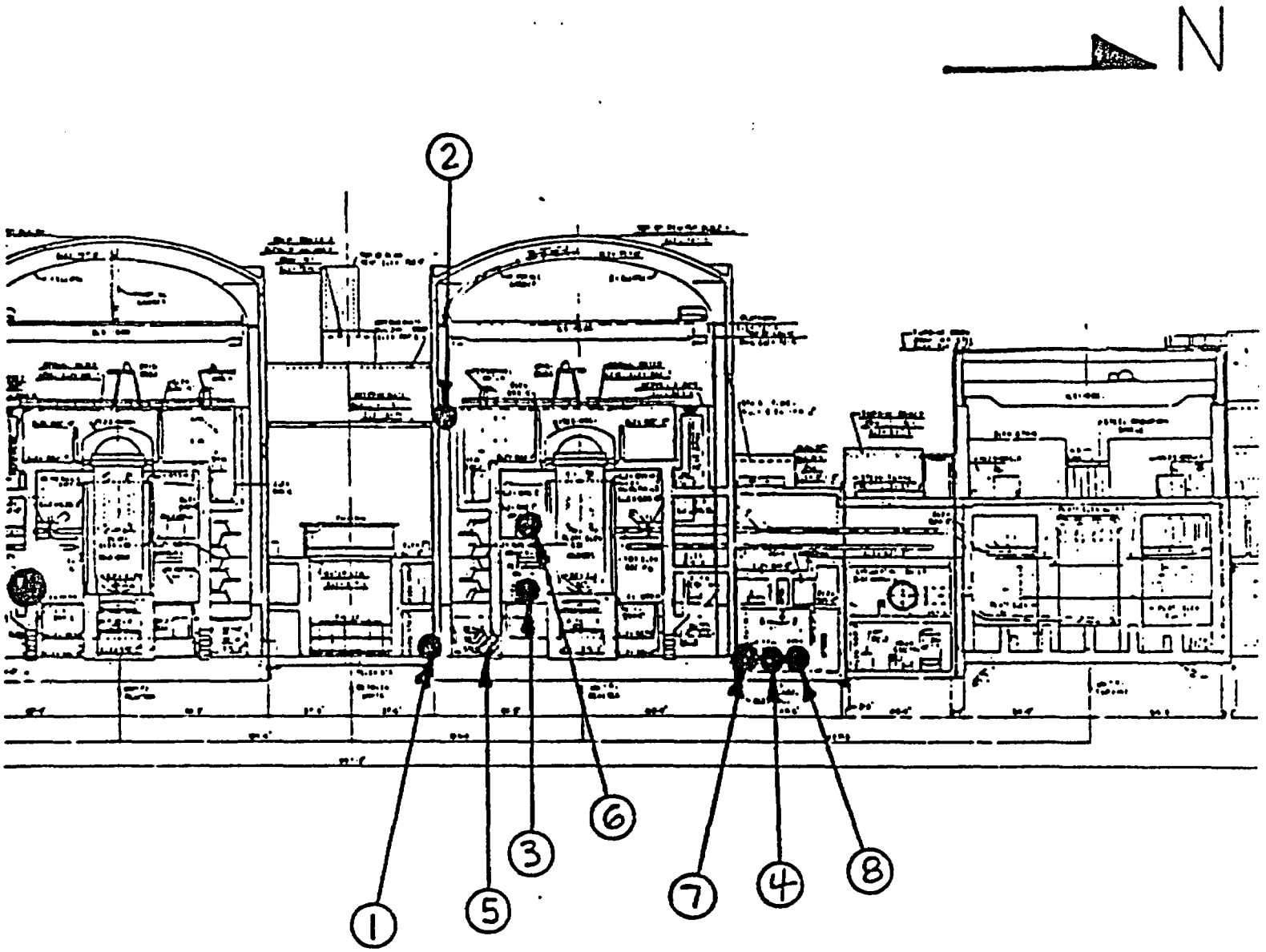
FIGURE A
Sheet 1 of 2



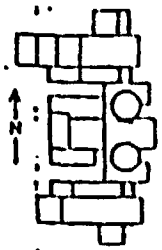
KEY:

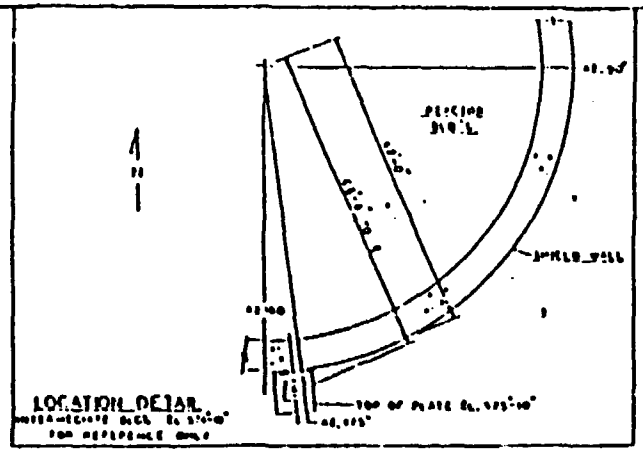
1. Instrument #D51-N101
2. Instrument #D51-N111
3. Instrument #D51-R120
4. Instrument #D51-R140
5. Instrument #D51-R160
6. Instrument #D51-R170
7. Instrument #D51-R180
8. Instrument #D51-R190 .

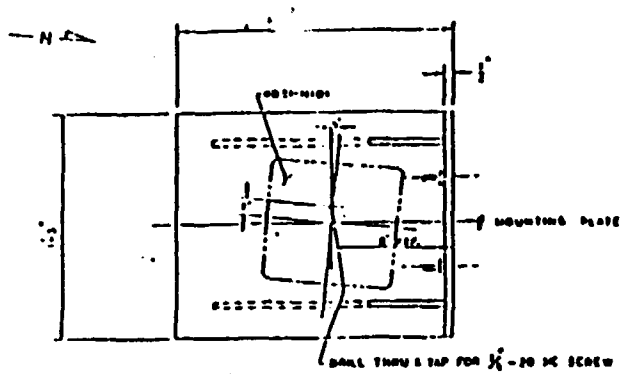
FIGURE A
Sheet 2 of 2



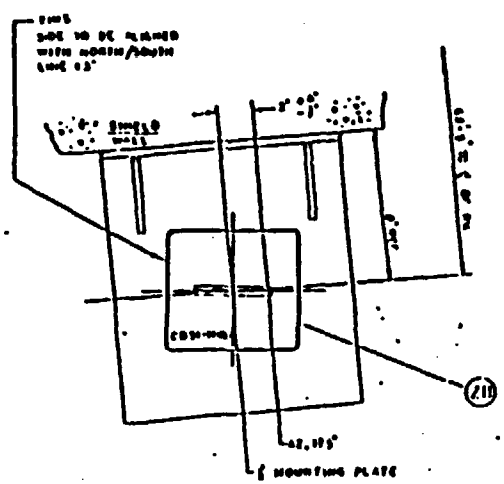
1. #D51-N101 R/B Foundation Mat, El. 575', Az. 175°
2. #D51-N111 R/B Containment Vessel, El. 686', Az. 174°
3. #D51-R120 Reactor Recirc Pump, El. 605', Az. 145°
4. #D51-R140 A/B Foundation Mat, El. 568'
5. #D51-R160 R/B Foundation Mat, El. 574' Az. 225°
6. #D51-R170 R/B Platform, El. 630' Az. 238°
7. #D51-R180 A/B Foundation Mat, El. 568'
8. #D51-R190 A/B Foundation Mat, El. 568'



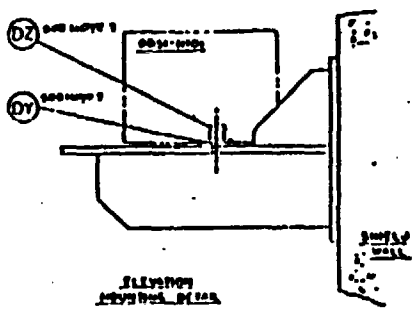




PLAN VIEW
MOUNTING & MODIFICATION DETAIL



PLAN VIEW
INSTRUMENT MOUNTING DETAIL



ELEVATION
MOUNTING DETAIL

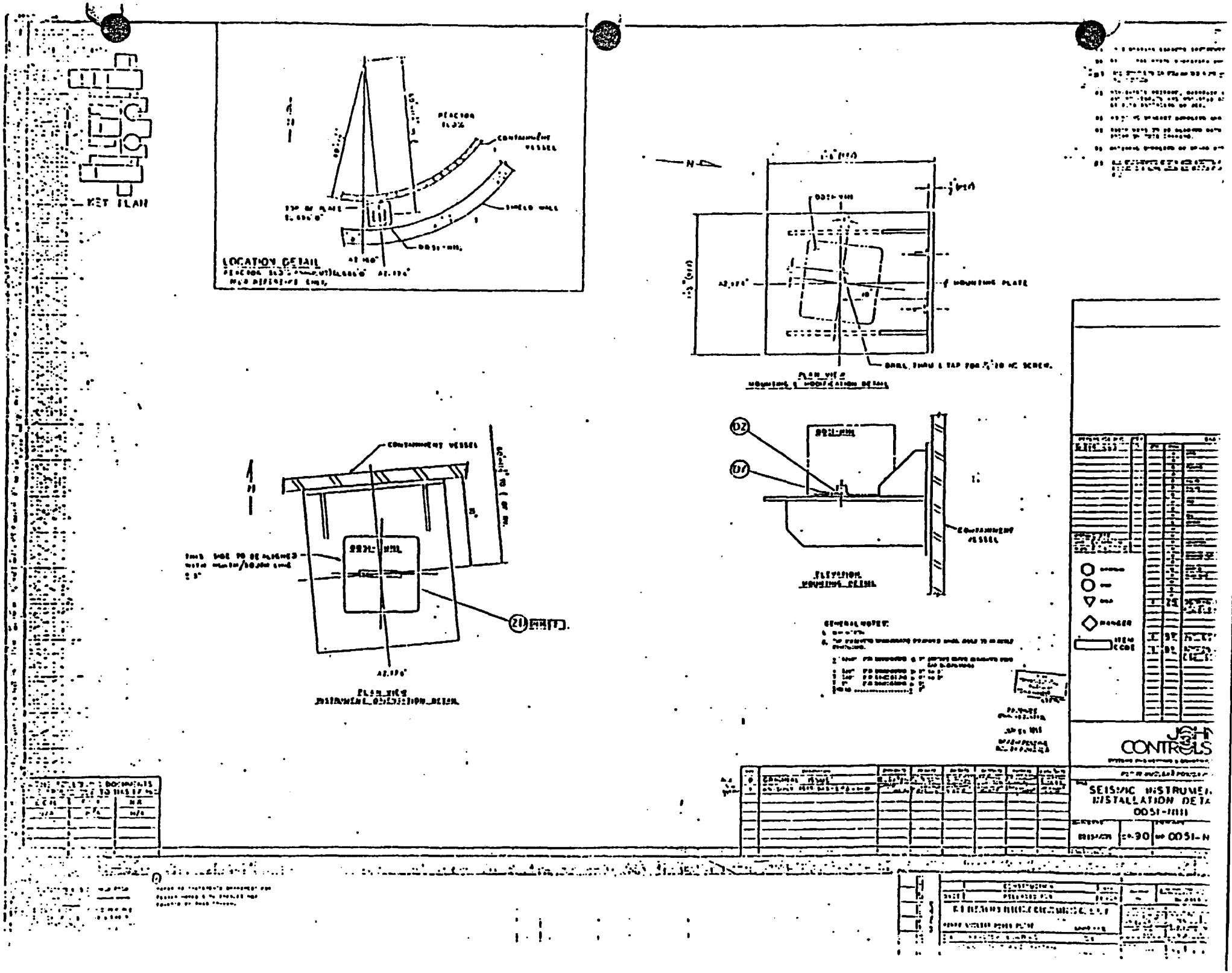
THE FOLLOWING DOCUMENTS ARE APPLICABLE TO THIS SHEET		
ECN	EVA	NR

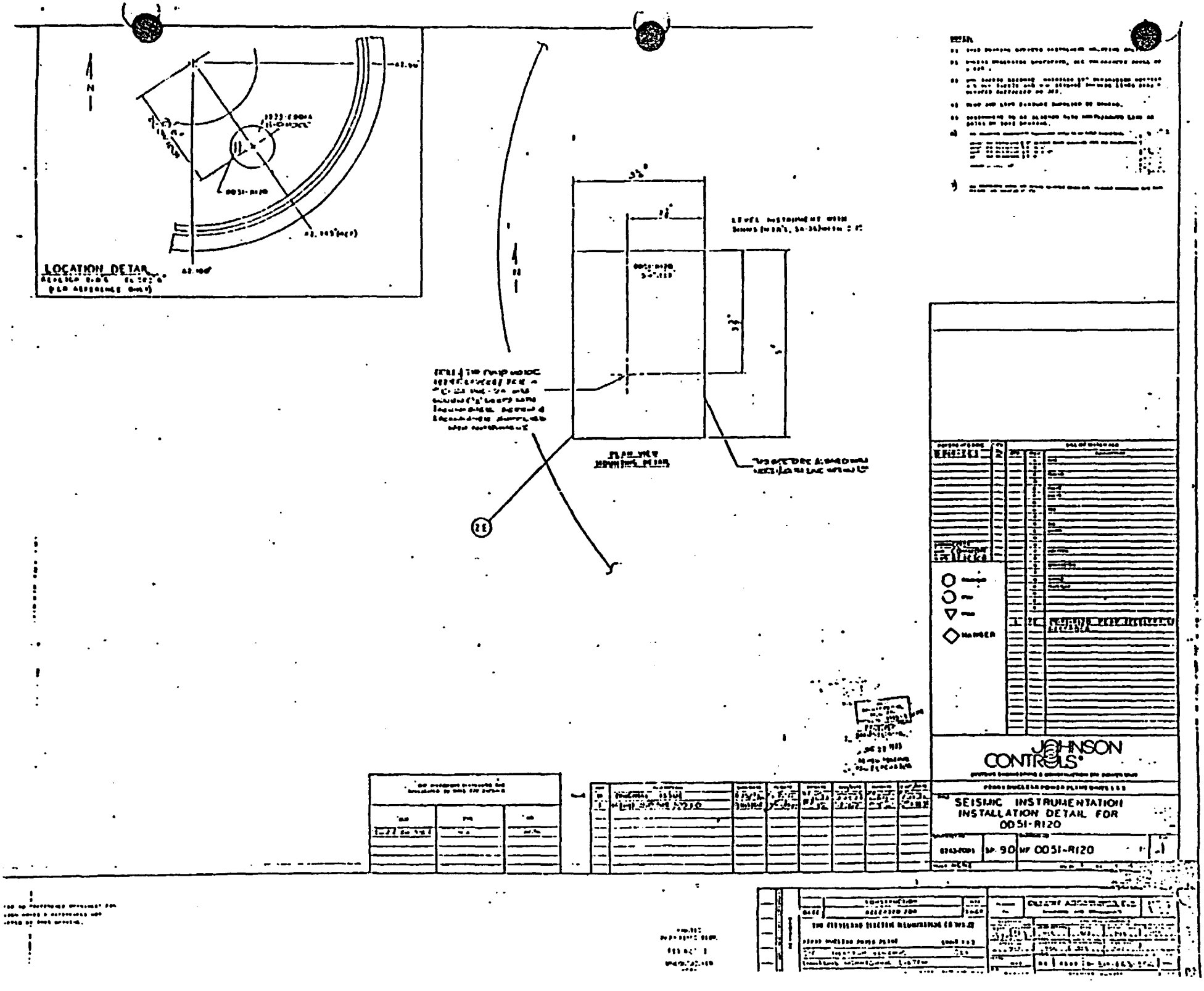
JOHNSON CONTROLS

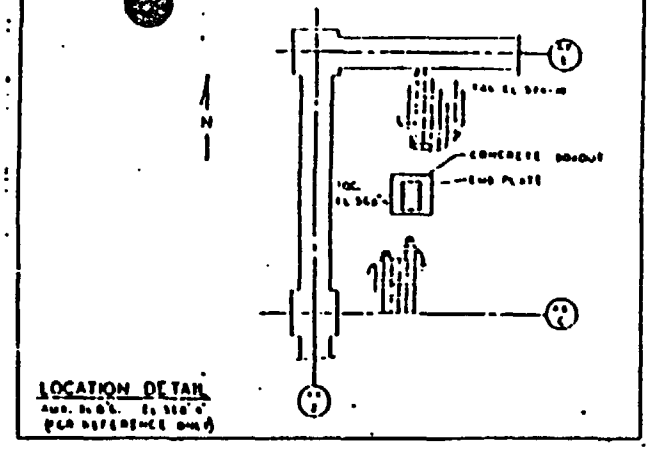
SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR 0051-H101

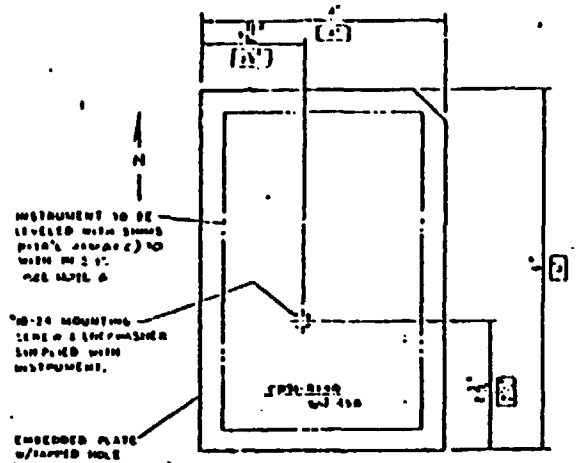
SP-90

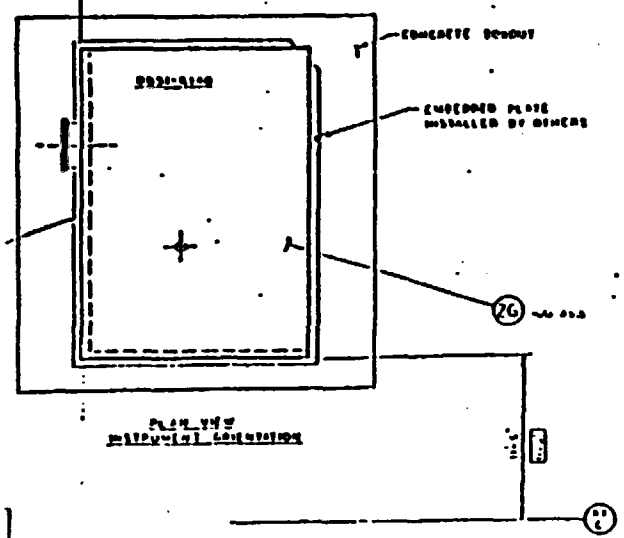
0051-H101


KEY PLAN
LOCATION DETAIL
CONTAINMENT VESSEL
SHIELD WALL
PLAN VIEW
MOUNTING & MODIFICATION DETAIL
MOUNTING PLATE
CONTAINMENT VESSEL
ELEVATION
MOUNTING DETAIL
CONTAINMENT VESSEL
PLAN VIEW
INSTRUMENT INSTALLATION DETAIL
GENERAL NOTES:
CONTROLS
SEISMIC INSTRUMENT
INSTALLATION DETAIL
0051-1111


LOCATION DETAIL
N
PLAN VIEW
22
JOHNSON CONTROLS
SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR 0051-R120
SP. 90
WP 0051-R120



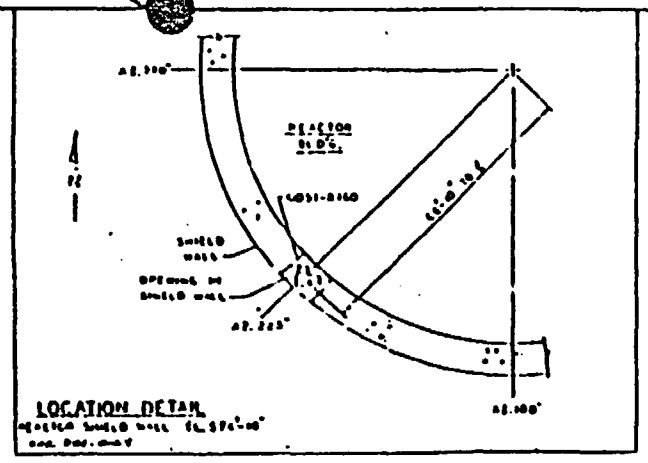


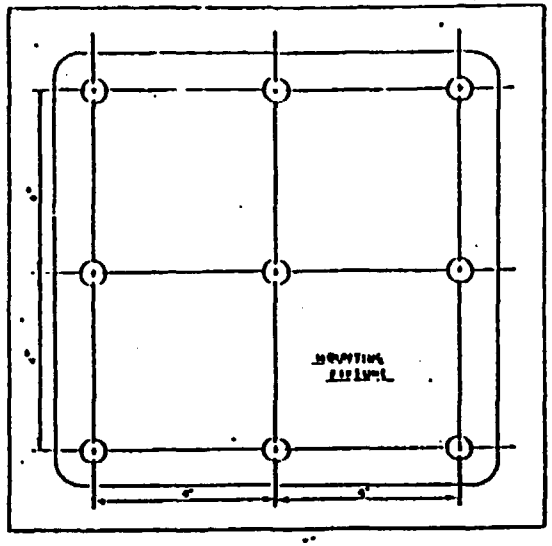


JOHNSON CONTROLS

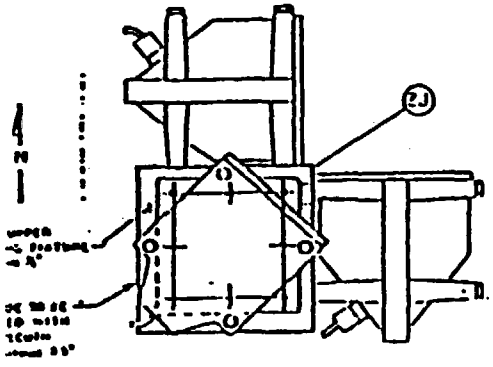
SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR OD51-R140

SP-90 OD51-R140

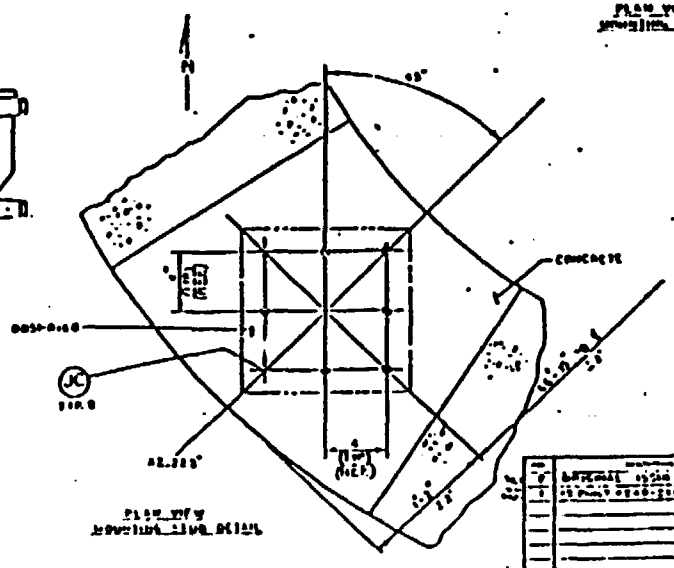




PLAN VIEW



PLAN VIEW

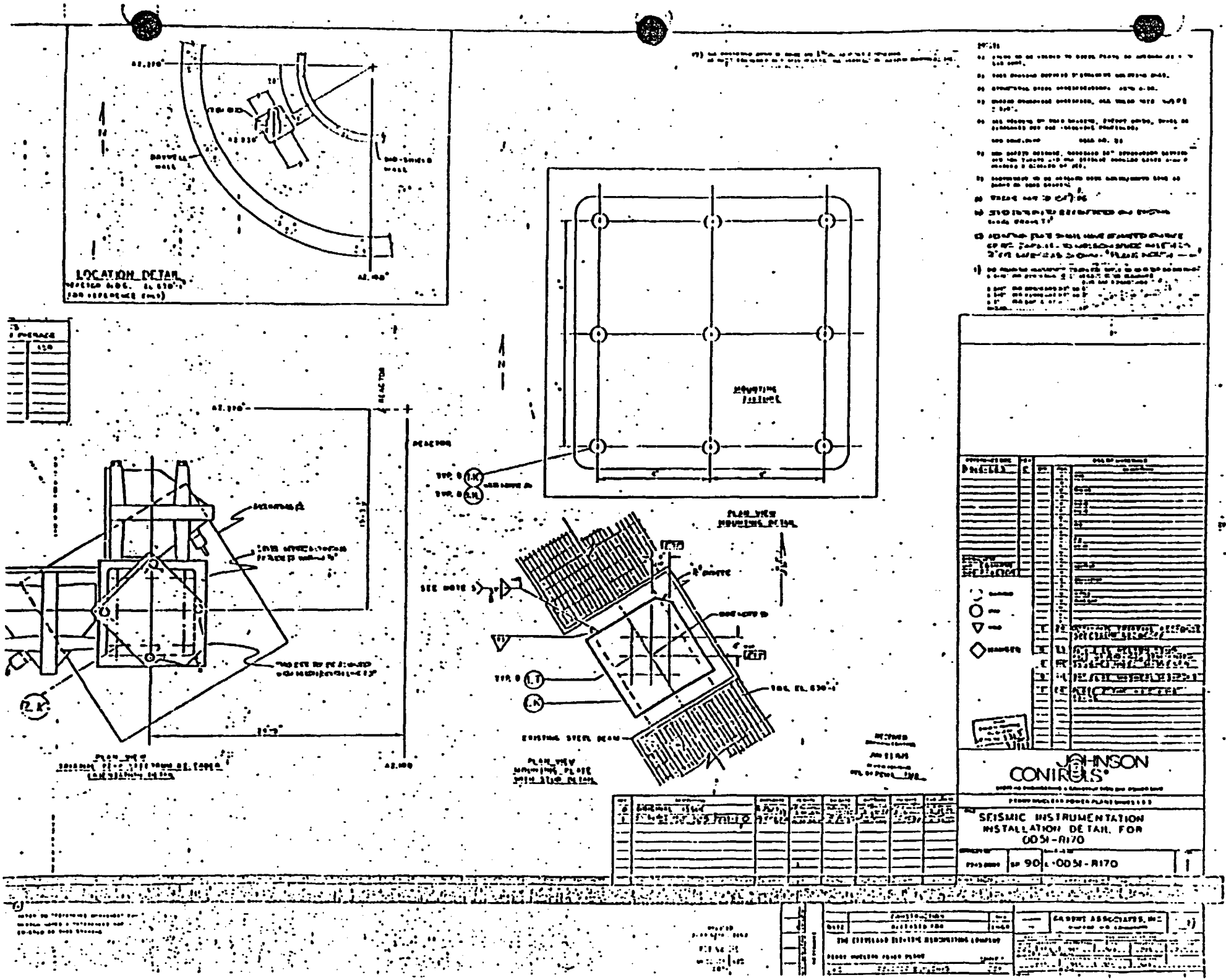


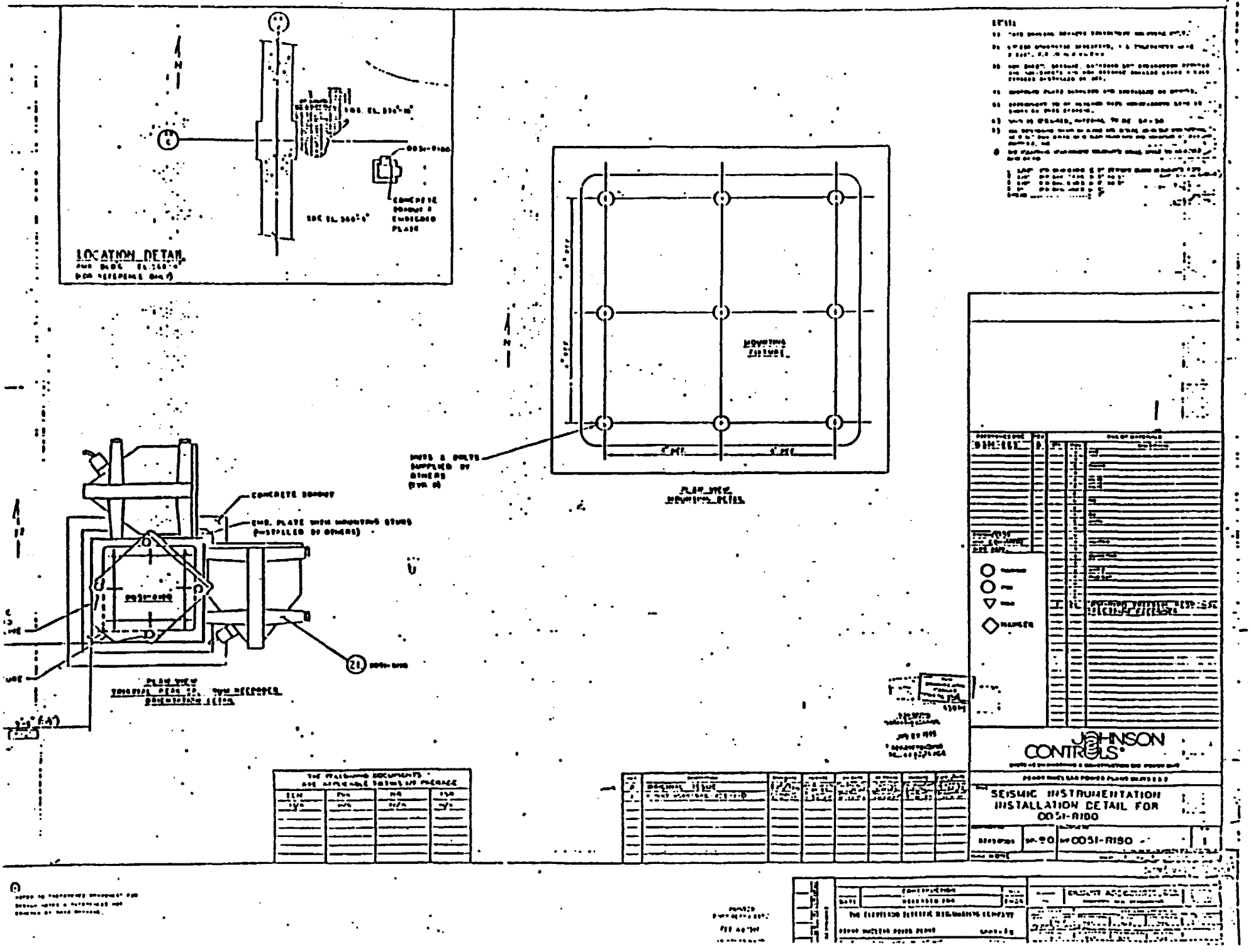
PLAN VIEW

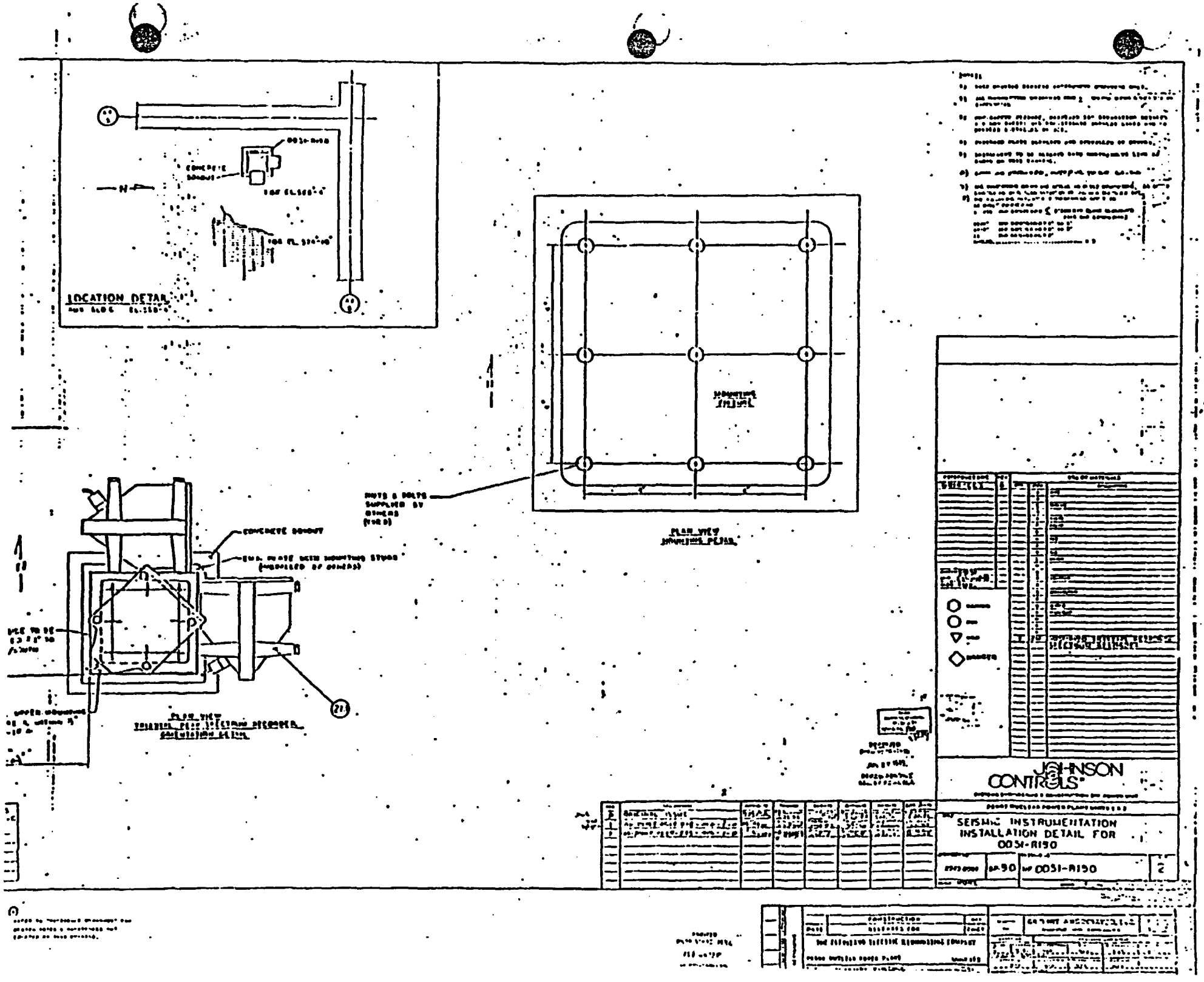
JOHNSON CONTROLS®

SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR 0051-R160

SA-90 0051-R160 1


LOCATION DETAIL
PLAN VIEW
JOHNSON CONTROLS
SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR 0051-R170


LOCATION DETAIL
CONCRETE DONOUT
END. PLATE WITH MOUNTING STUDS
(INSTALLED BY OTHERS)
MOUNTING FIXTURE
JOHNSON CONTROLS
SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR 0051-R100


LOCATION DETAIL
CONCRETE DONOUT
PLAN VIEW
CONCRETE DONOUT
JOHNSON CONTROLS
SEISMIC INSTRUMENTATION INSTALLATION DETAIL FOR 0051-R190
0051-R190

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 tha 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	

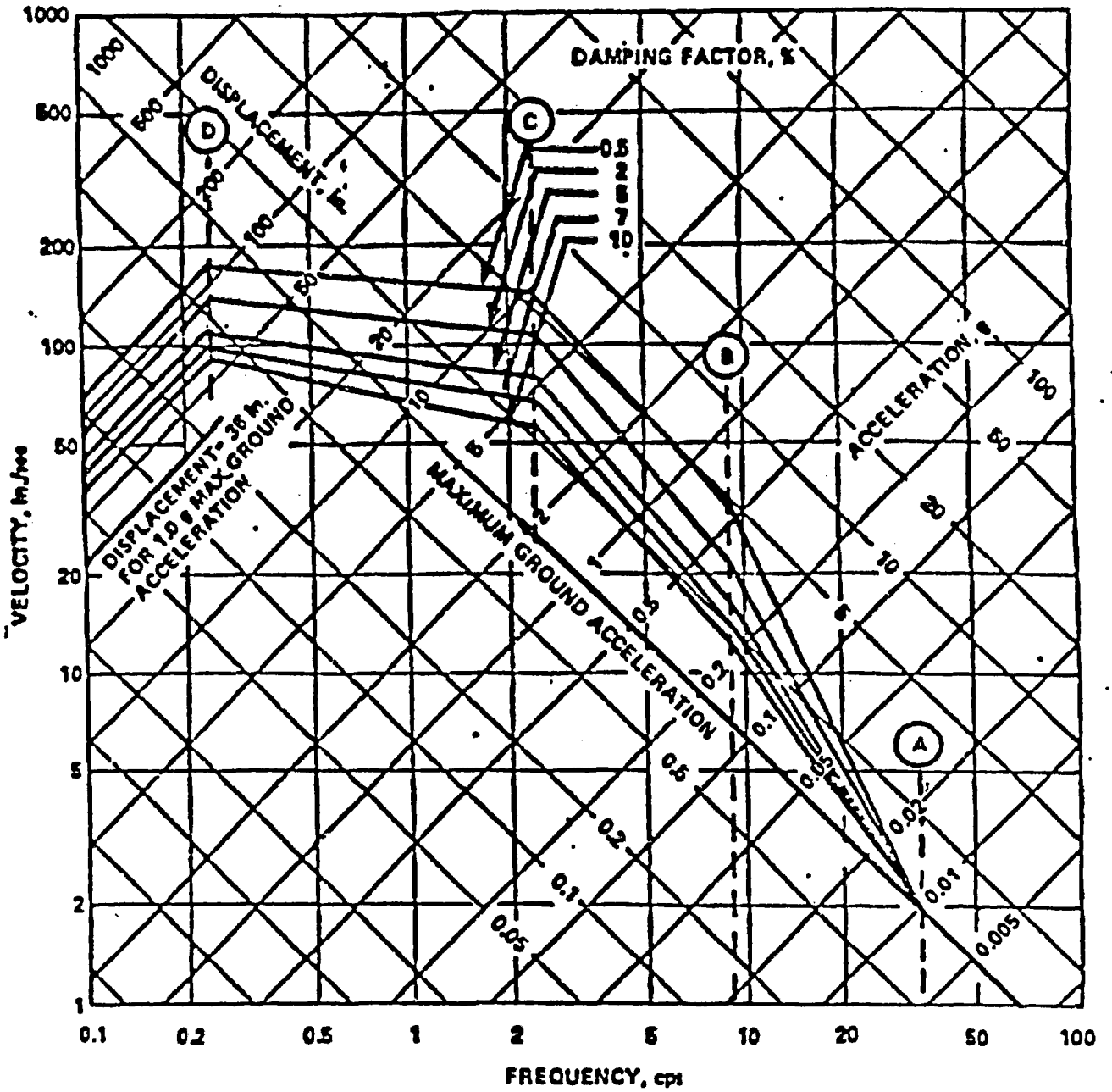
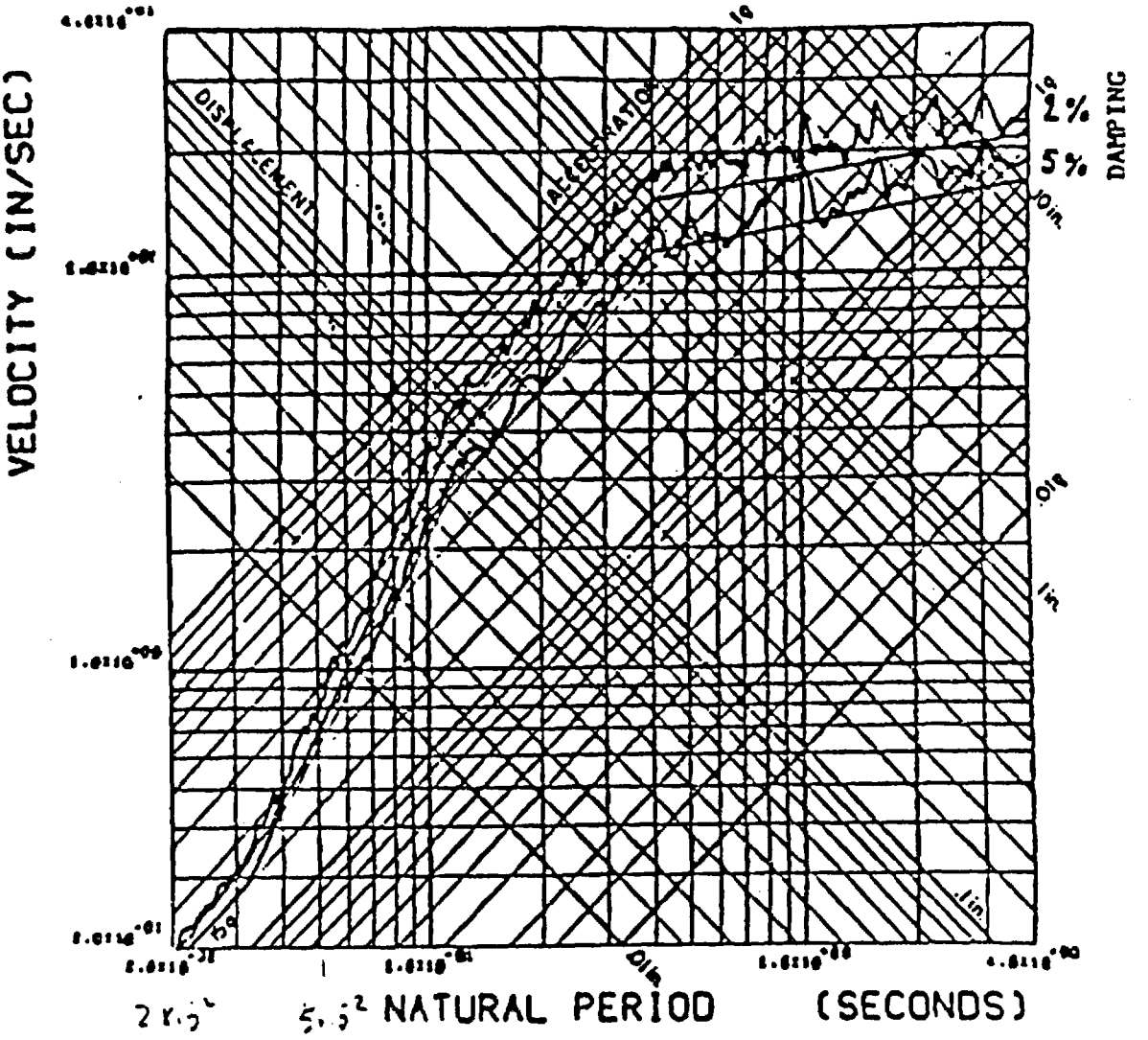


FIGURE 1. HORIZONTAL DESIGN RESPONSE SPECTRA – SCALED TO 1g HORIZONTAL GROUND ACCELERATION

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

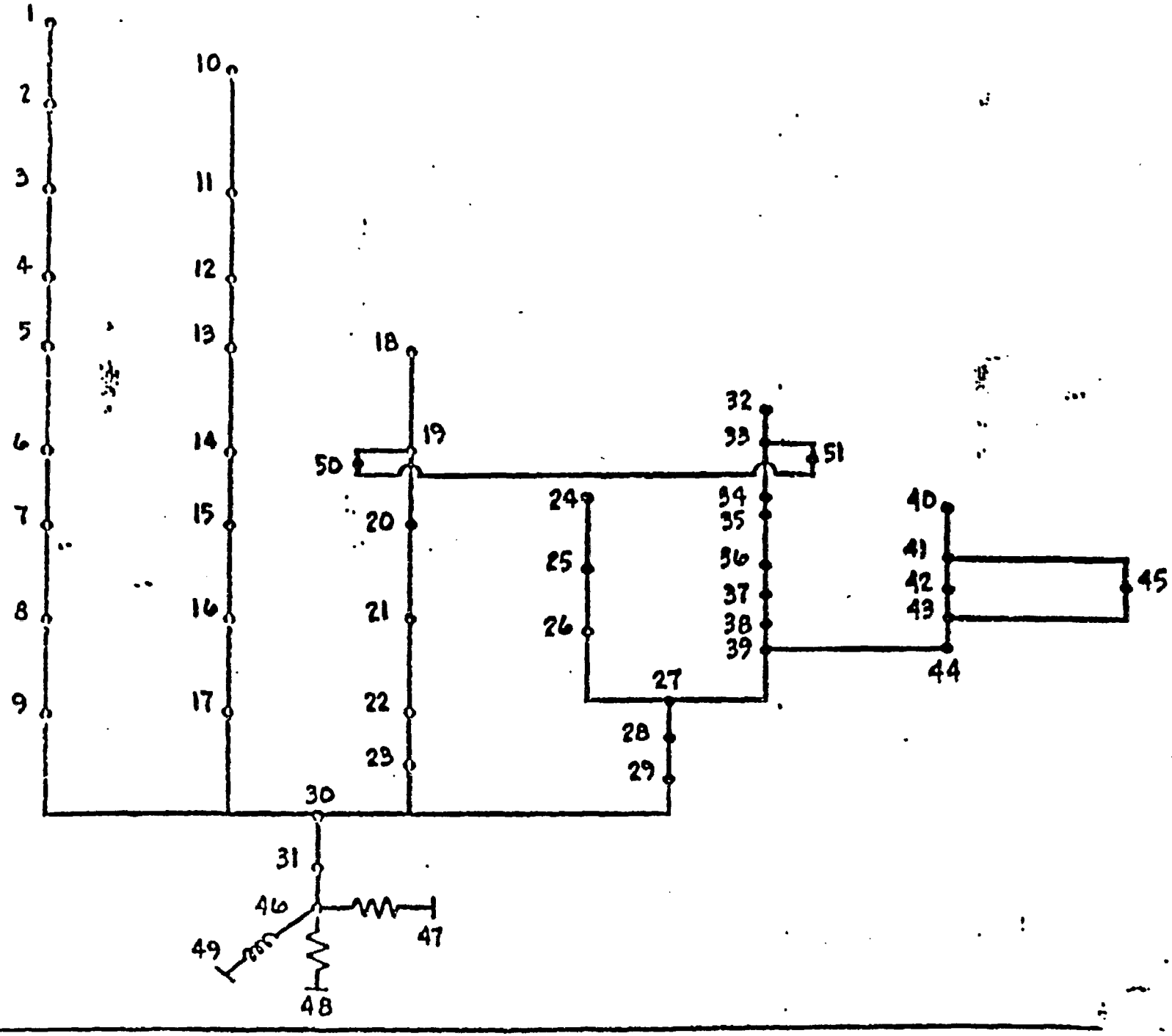

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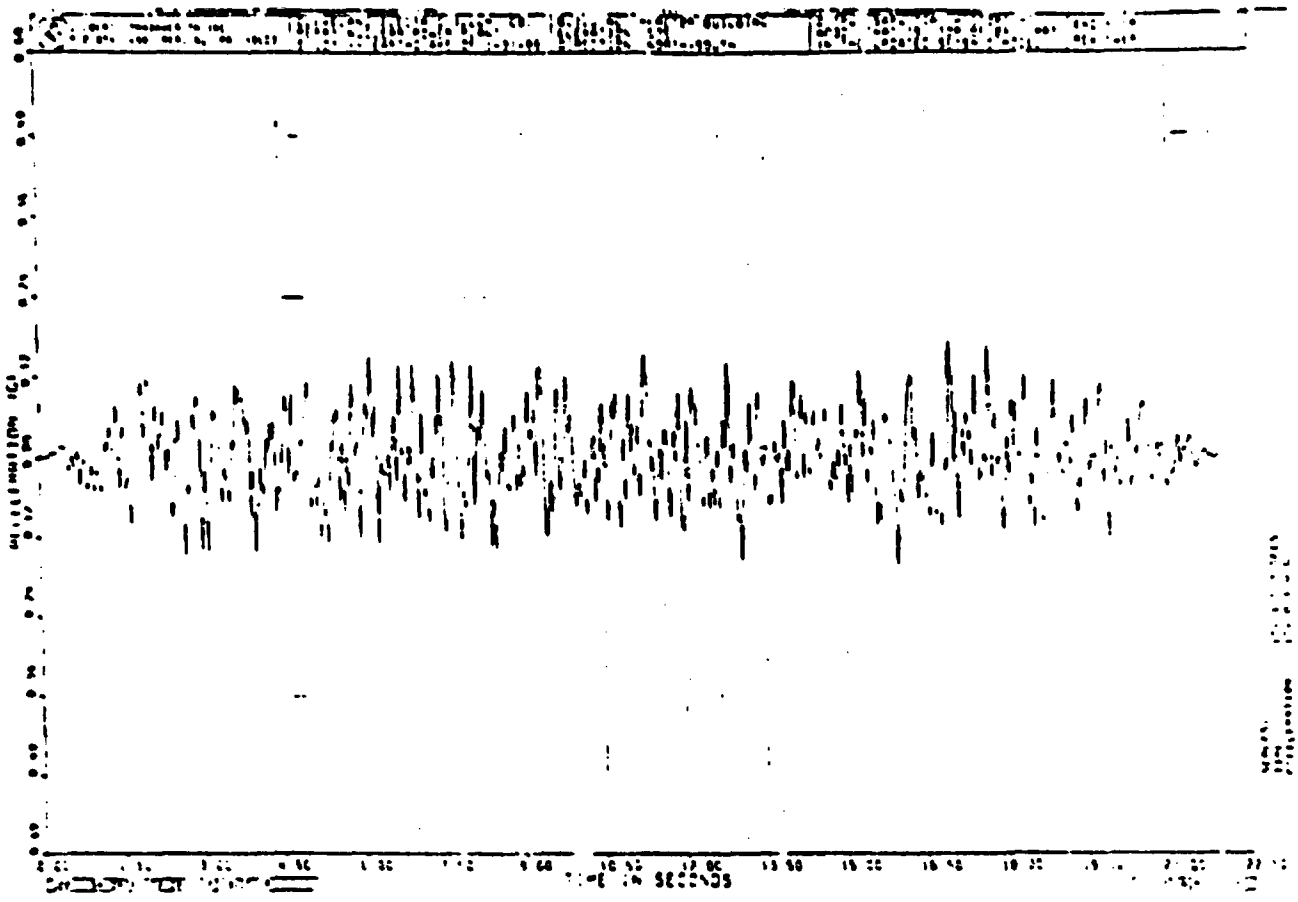
Figure 6.3



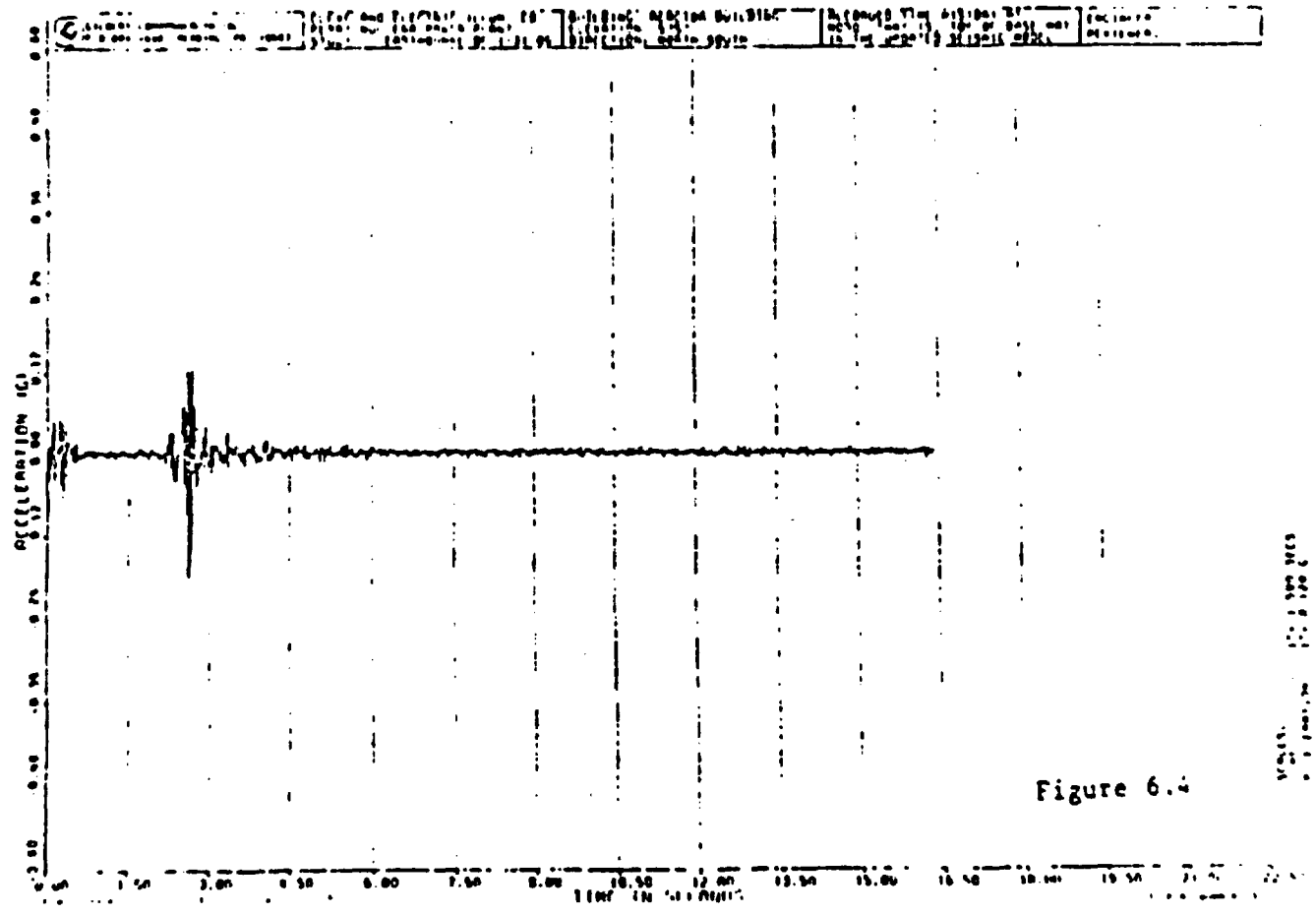


Figure 6.4

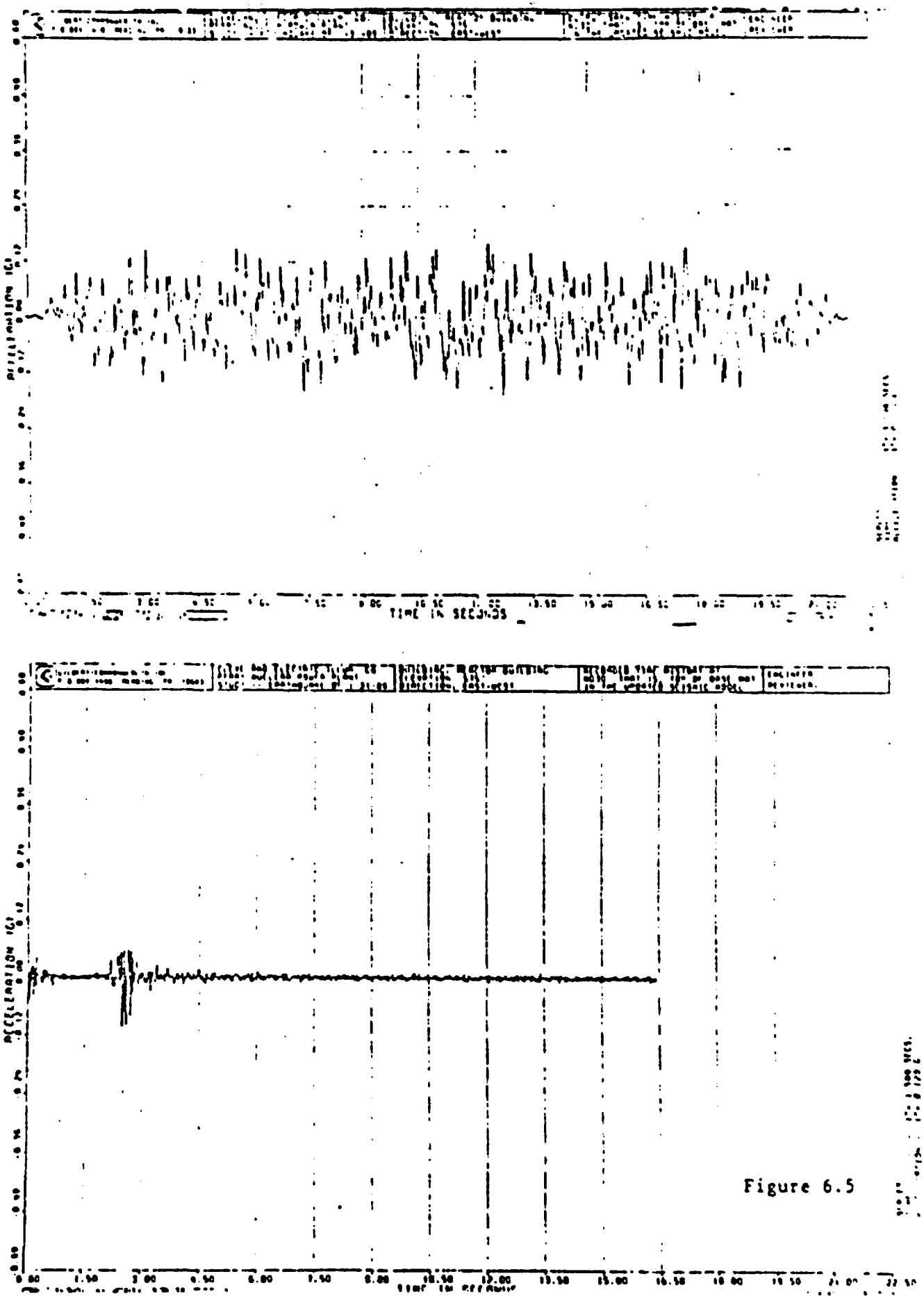
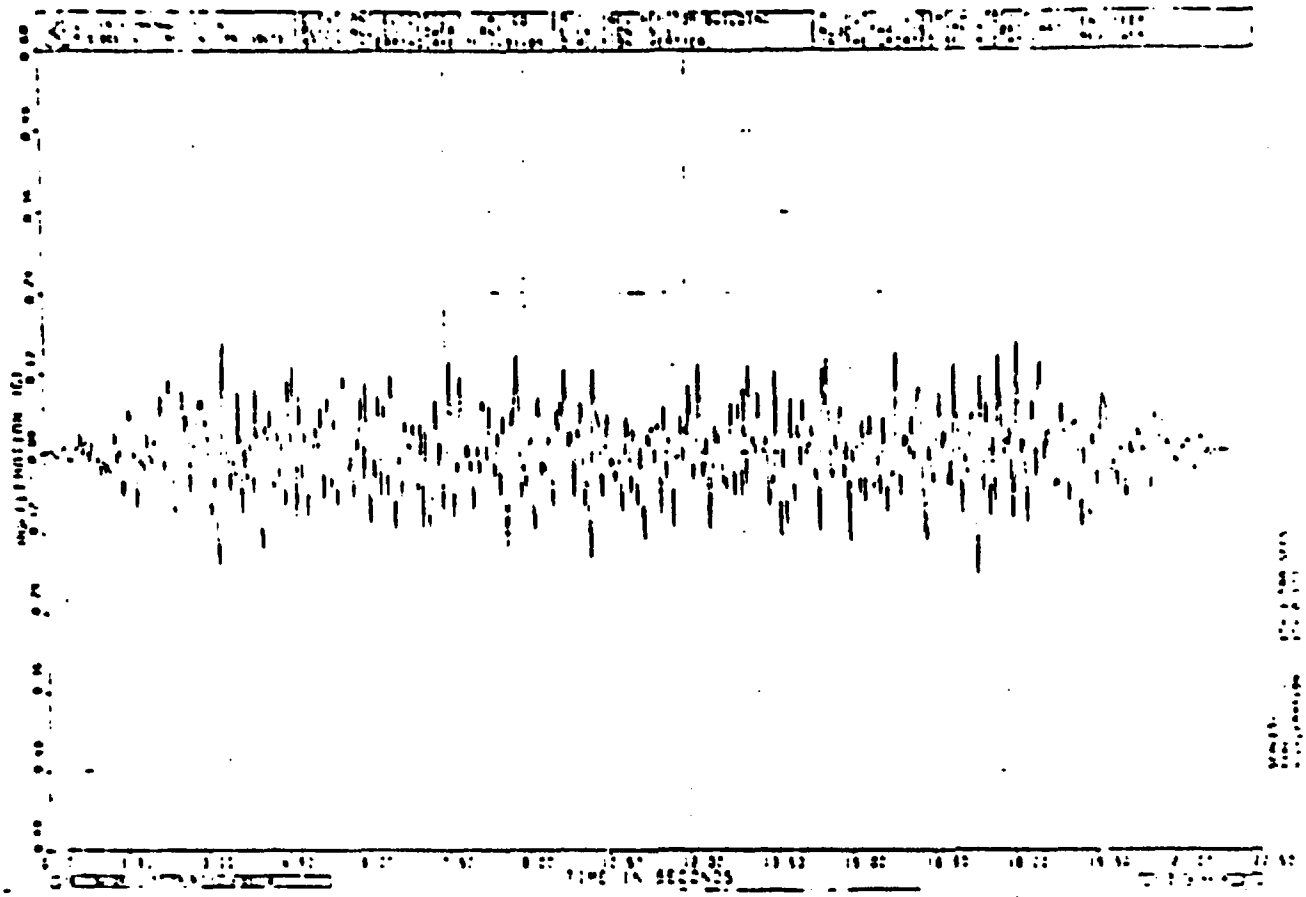
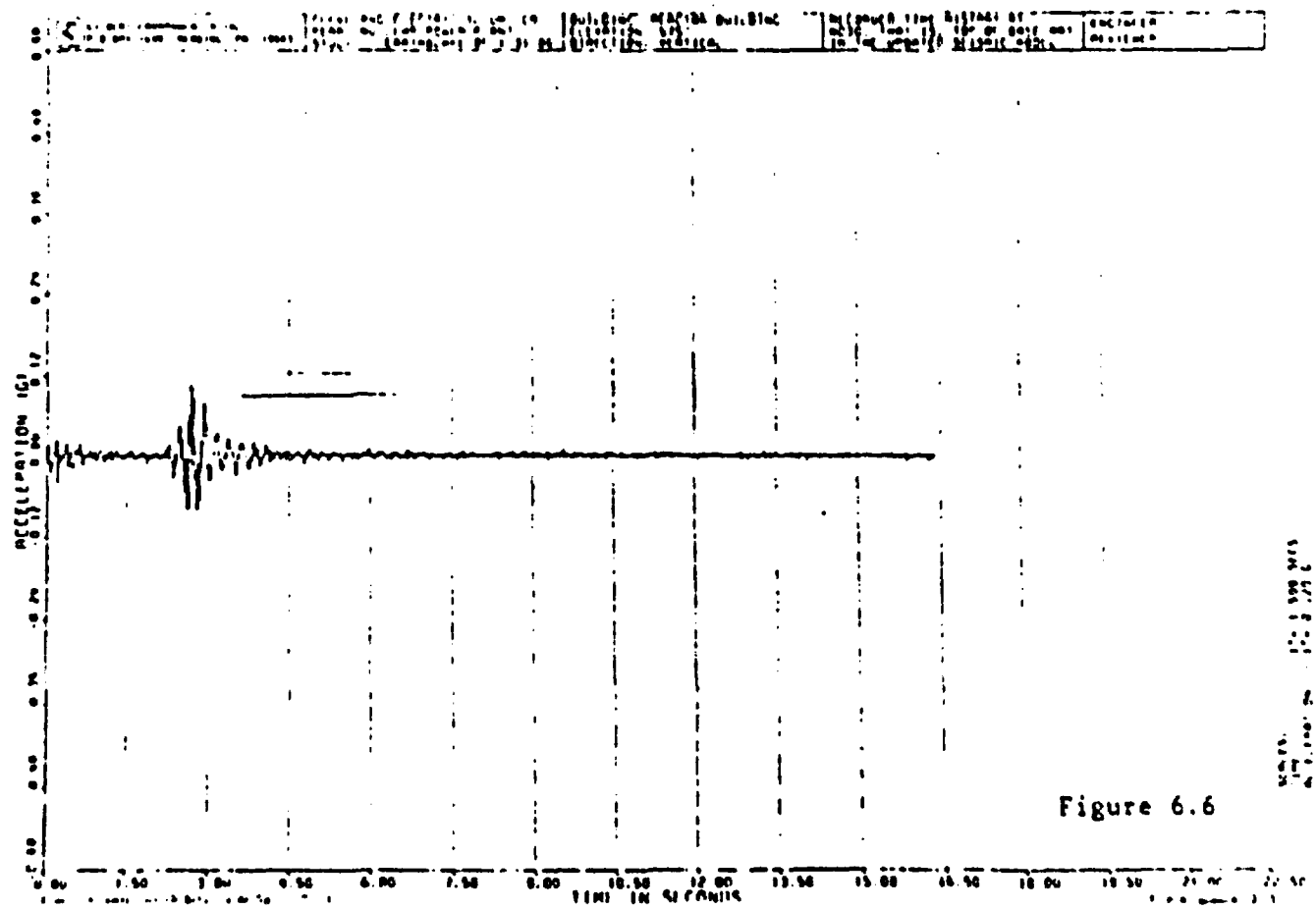


Figure 6.5





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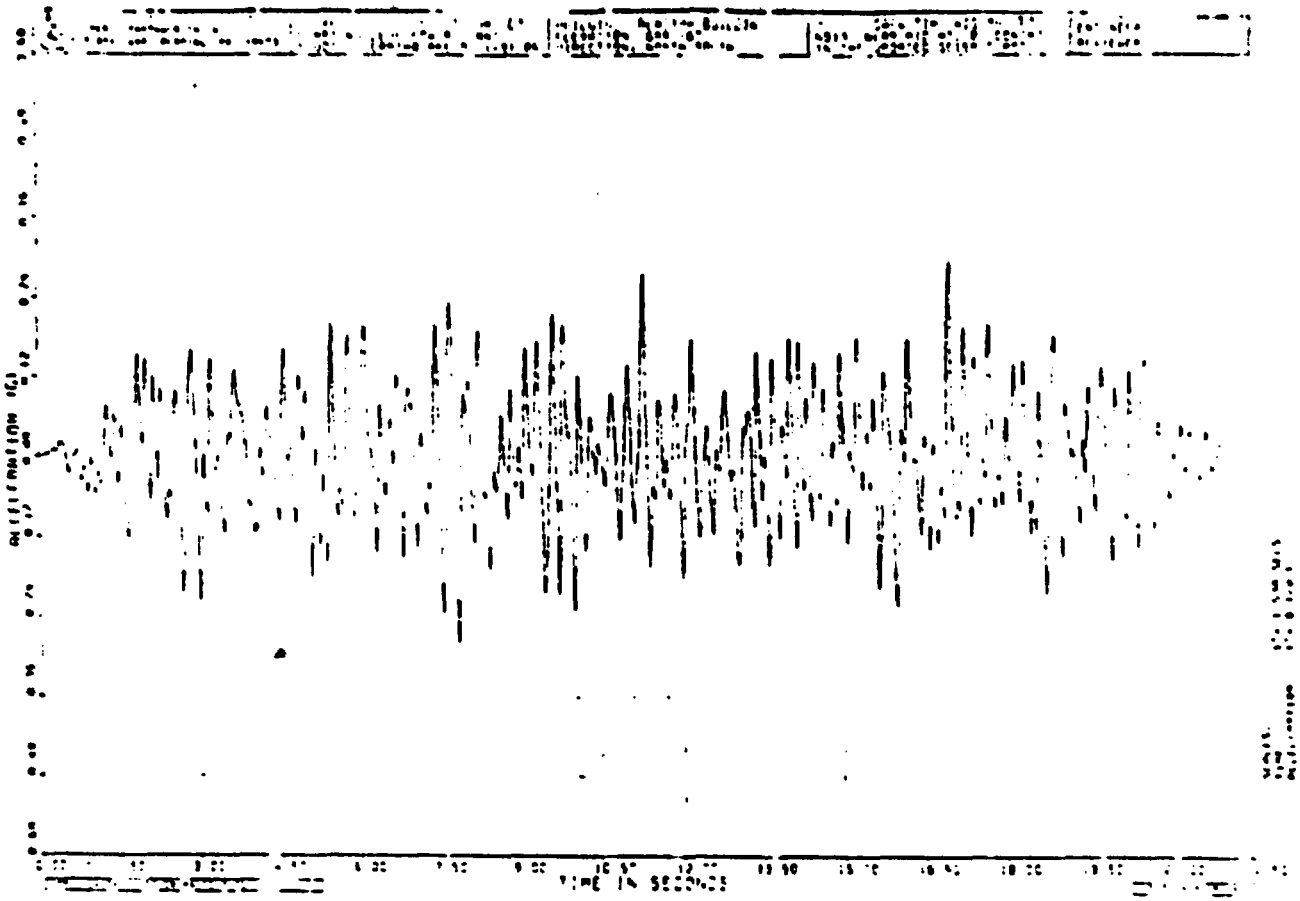


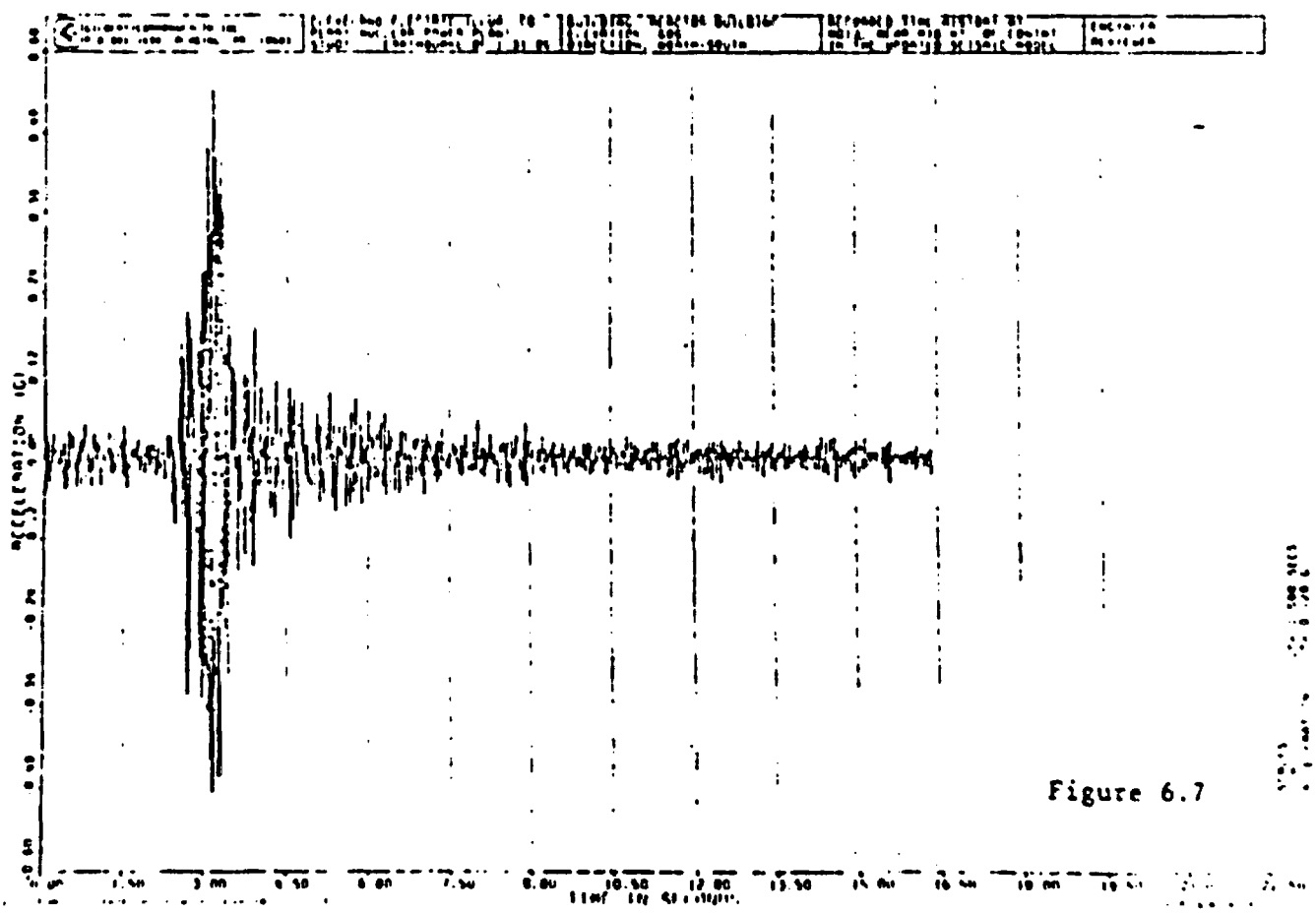
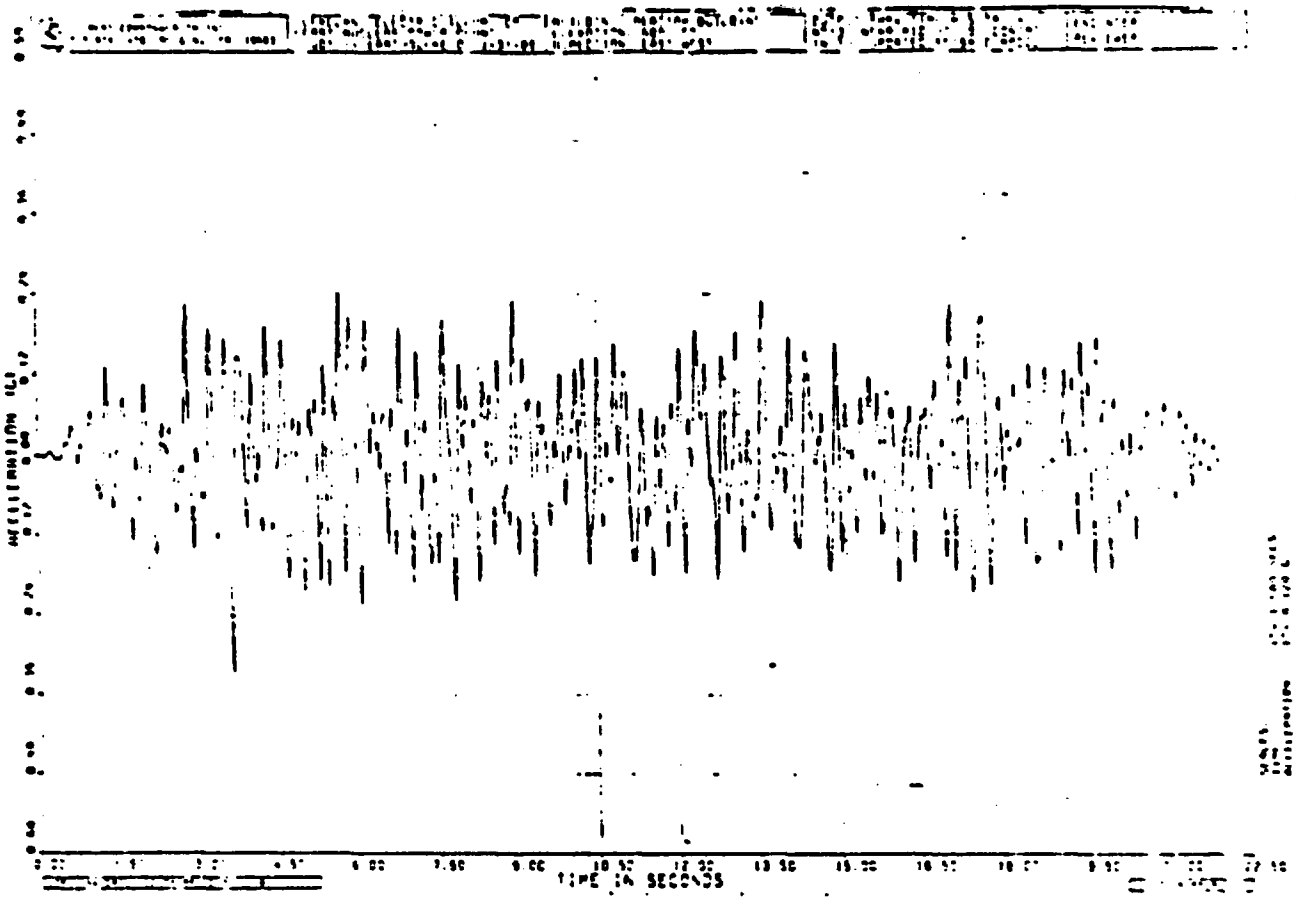


Figure 6.7



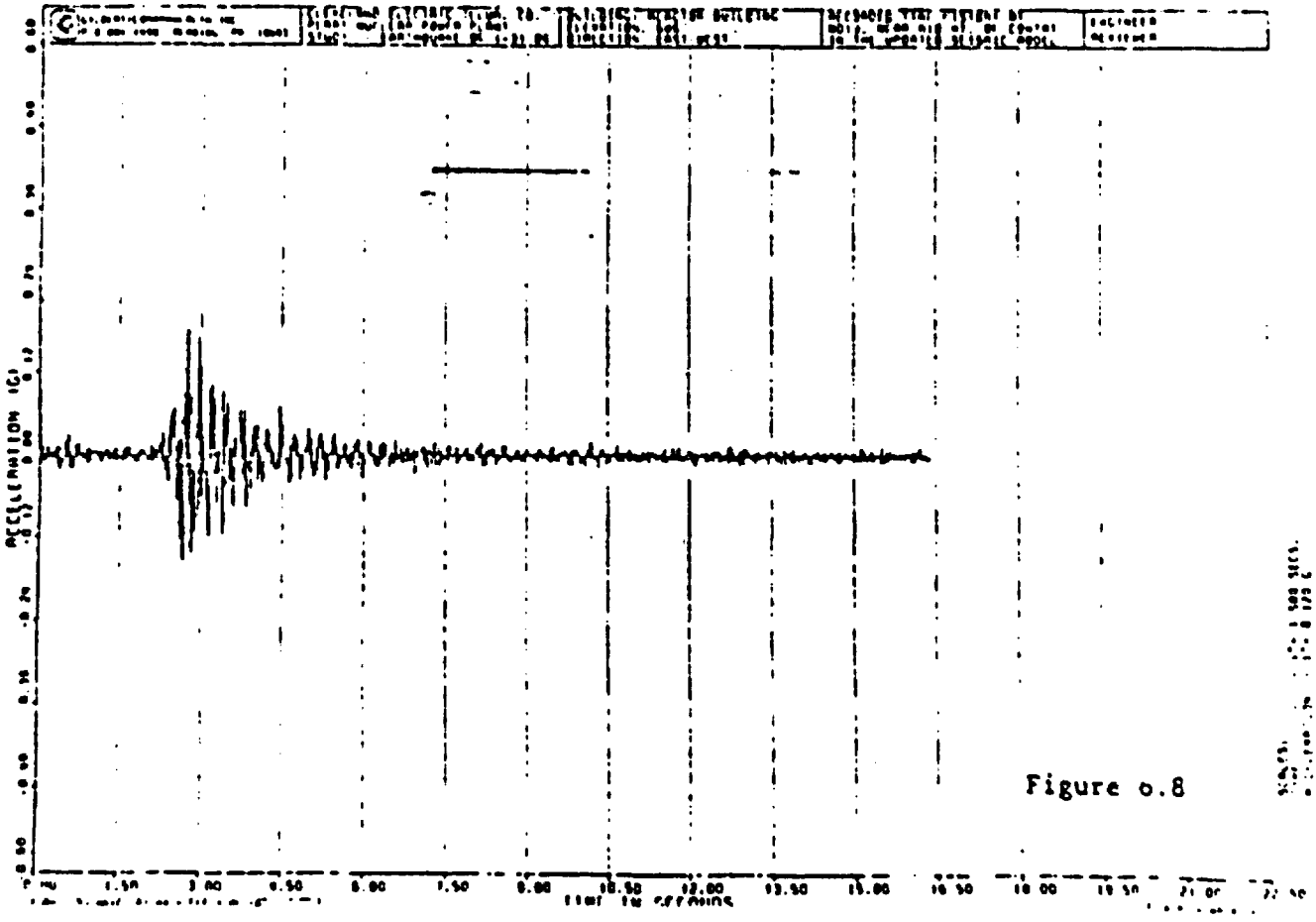
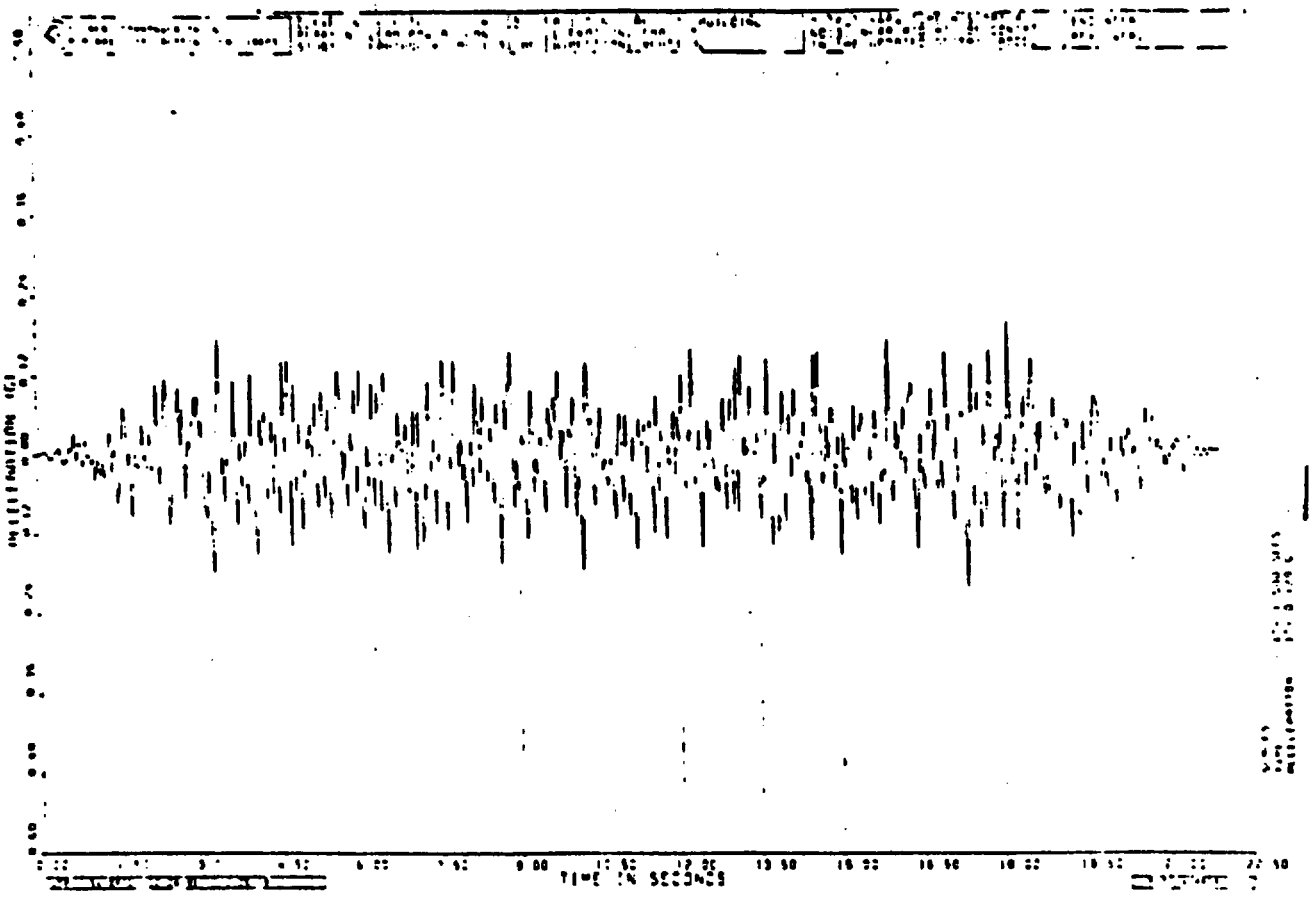


Figure o.8



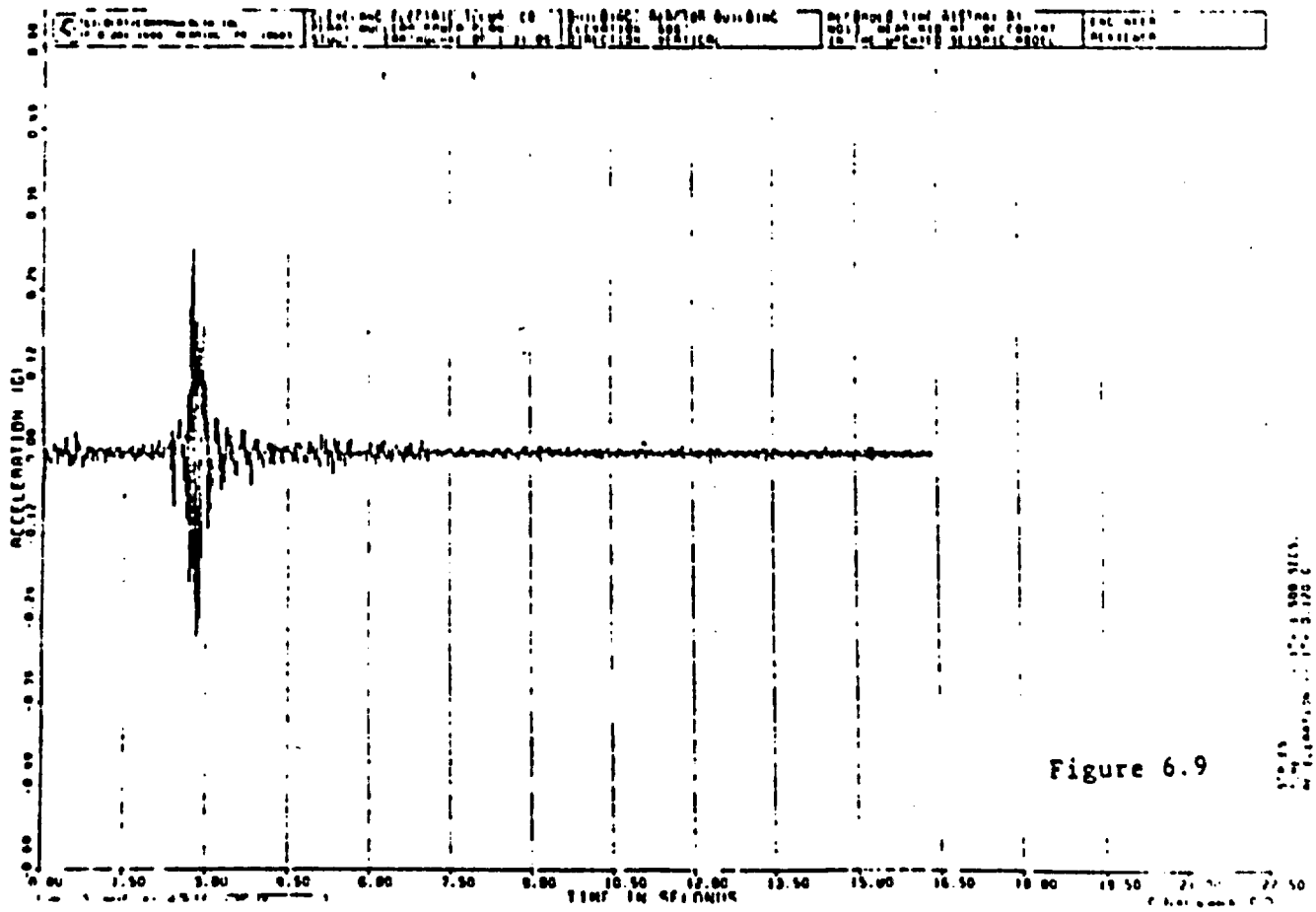


Figure 6.9

ML 5.0 EARTHQUAKE JANUARY 31, 1980

11A8001 PERRY NUCLEAR POWER PLANT COMP WEST SMA35/N 165-1

DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL

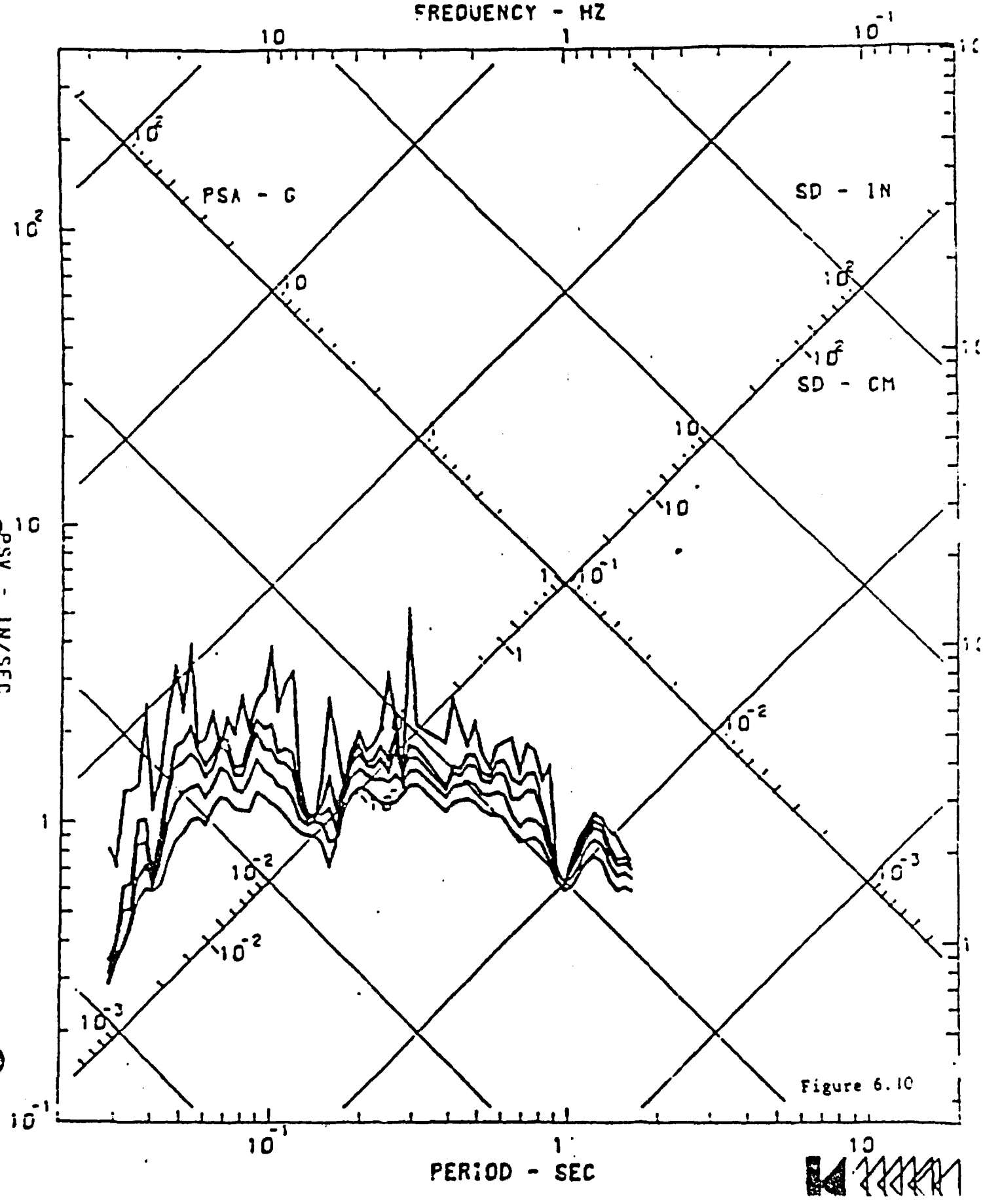
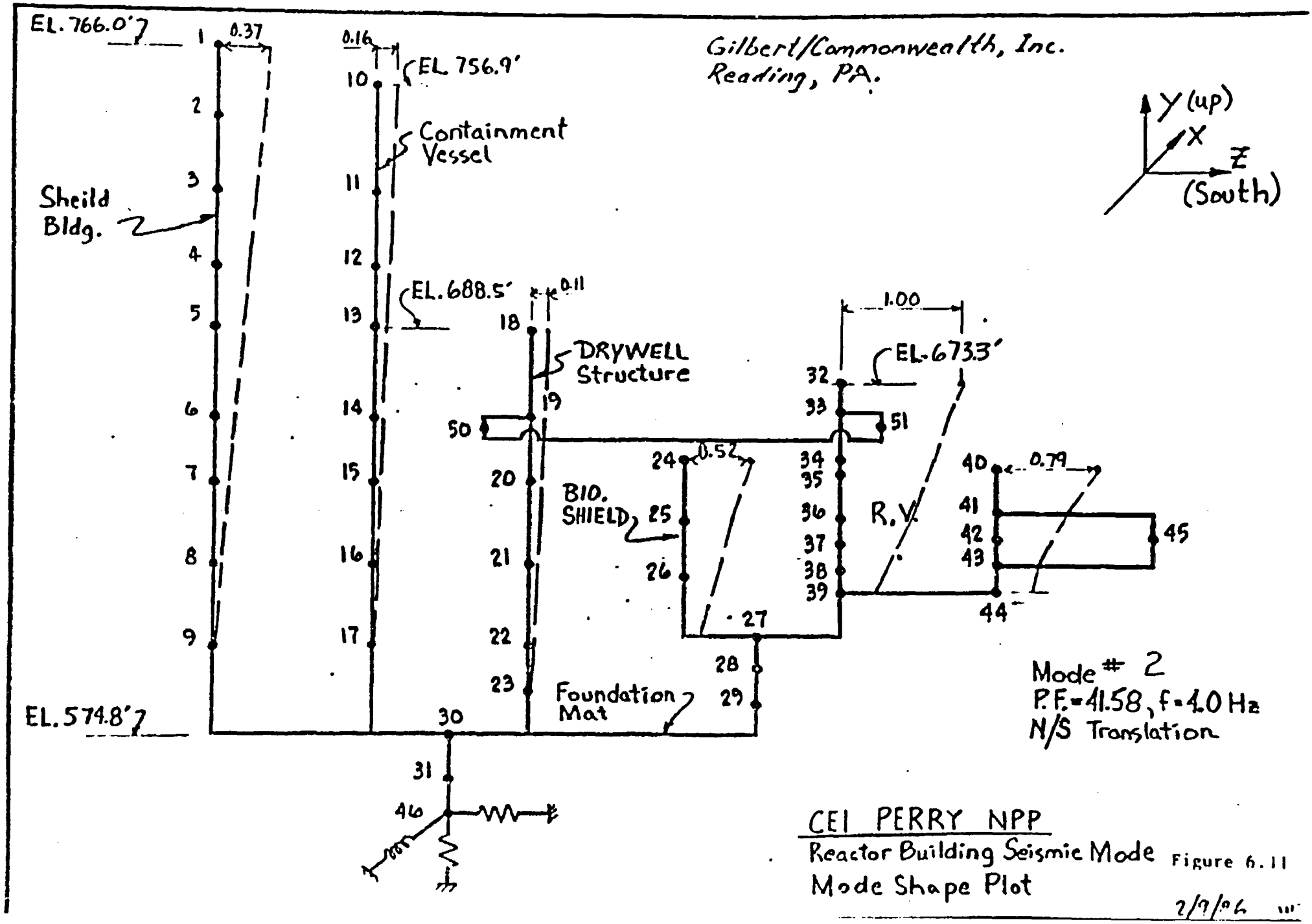


Figure 6.10

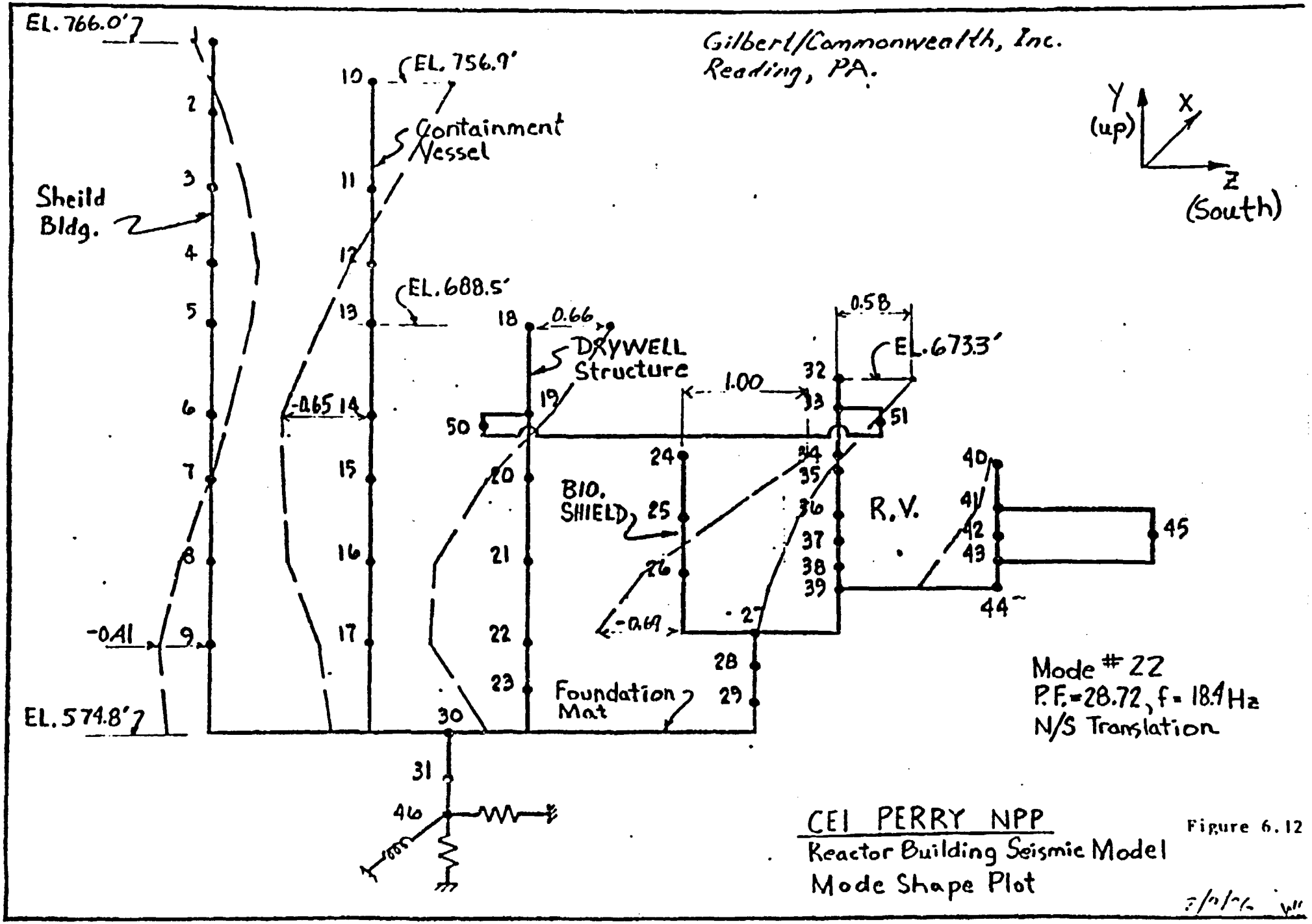


CEI PERRY NPP

Reactor Building Seismic Mode Mode Shape Plot

Figure 6.11

2/9/86

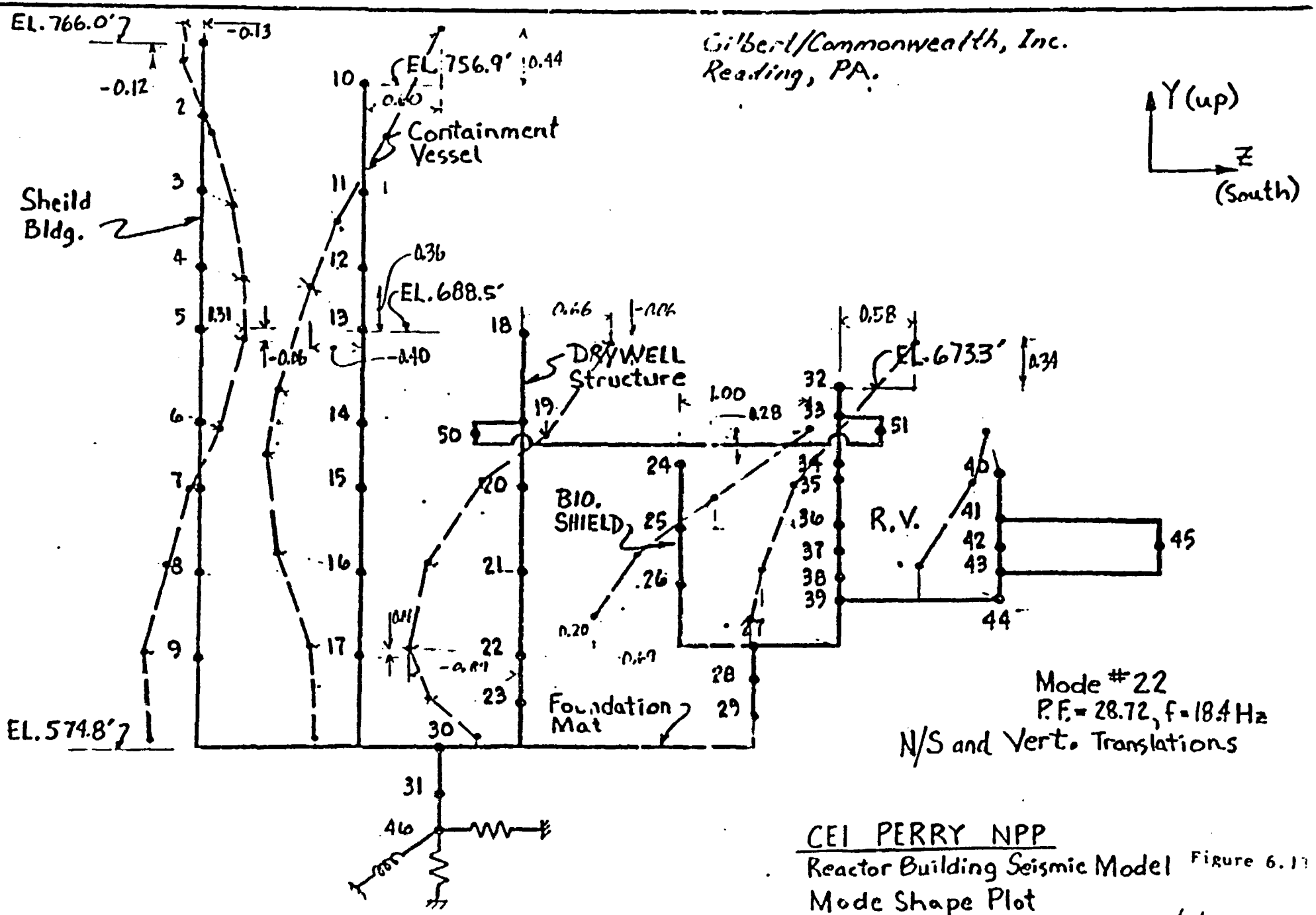


CEI PERRY NPP

Reactor Building Seismic Model

Mode Shape Plot

Figure 6.12



CEI PERRY NPP

Reactor Building Seismic Model

Mode Shape Plot

Figure 6.1?

2/?/86

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

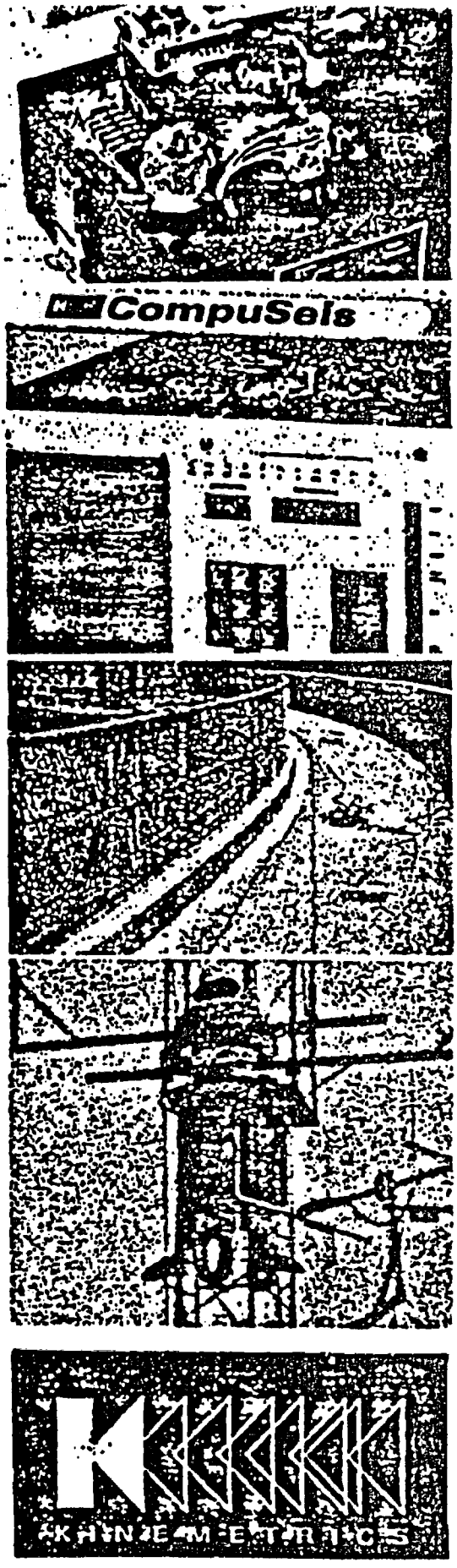
STRONG-MOTION DATA FROM THE PERRY NUCLEAR POWER PLANT SEISMIC INSTRUMENTATION KINEMETRICS

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 Compensator, 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.

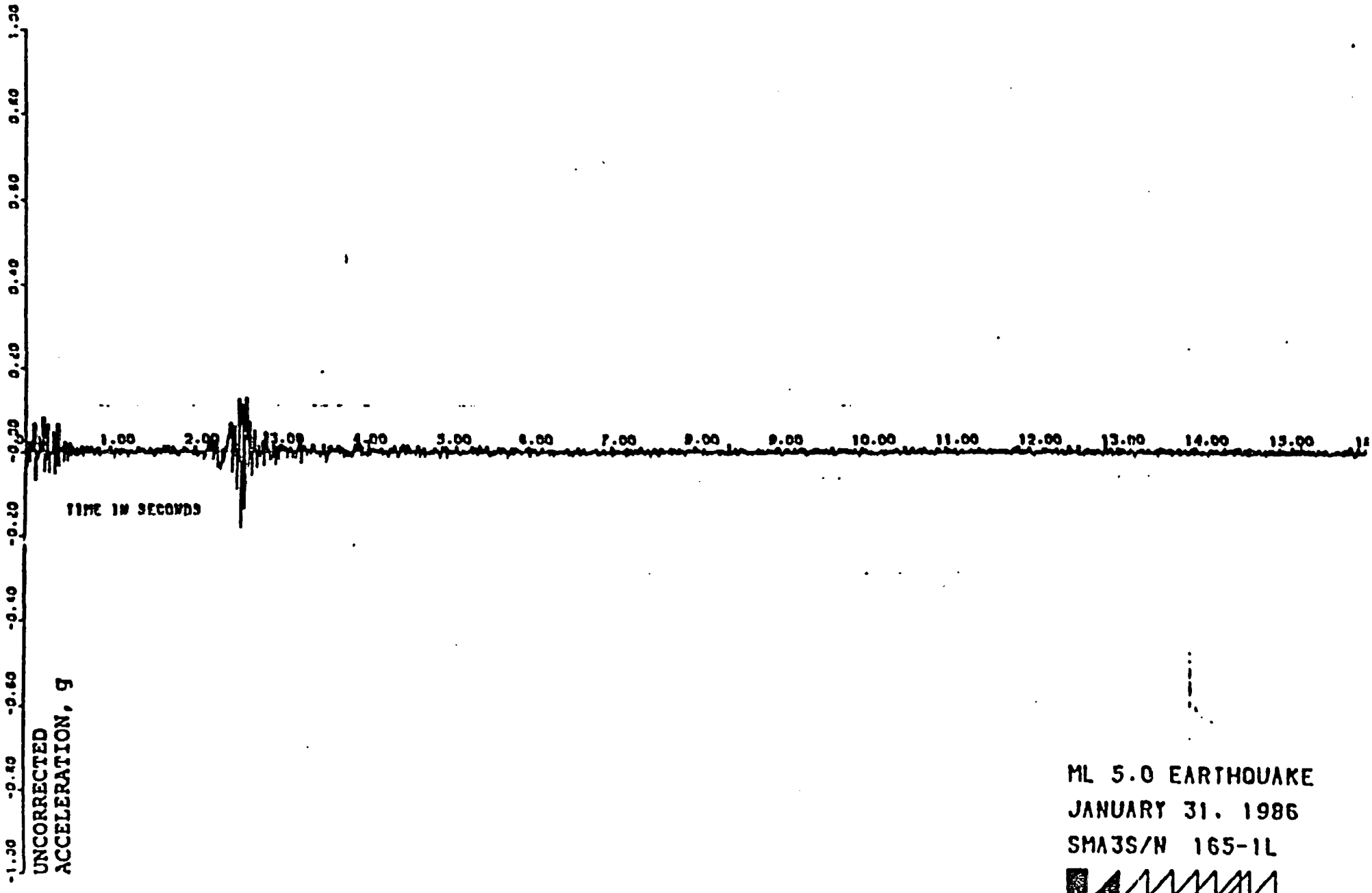
3.4 VOL3 Processing

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.

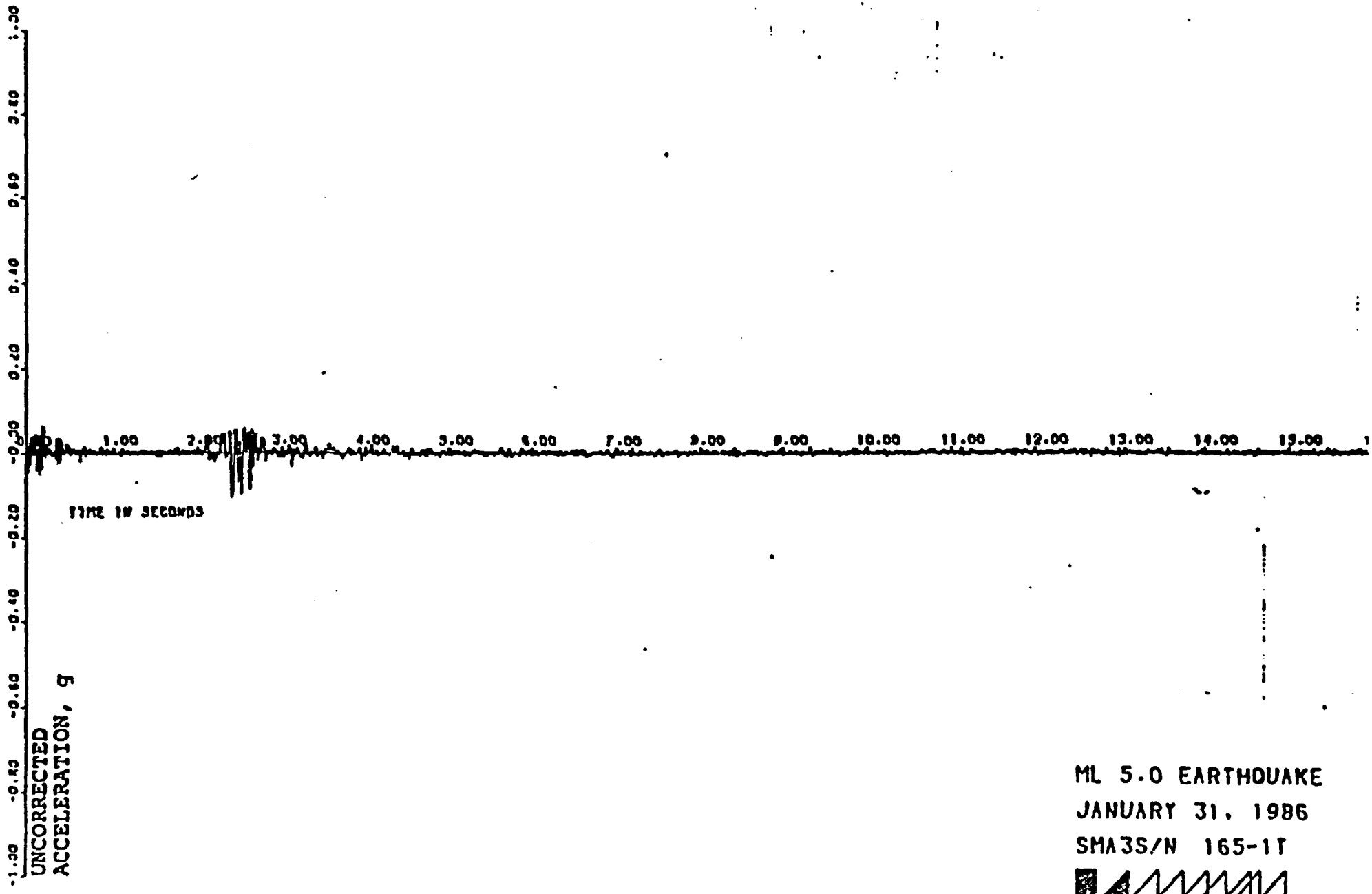
Two types of plots were produced and are included in the data section of this report. The first type is the traditional tripartite log-log plot of pseudo-velocity vs. period. The second is a linear plot of velocity response and Fourier spectrum vs. frequency.

Reactor Building Foundation, Elevation 575 Ft.

SMA-3 Serial Number 165-1
Tag Number D51-H101
Longitudinal Channel - South Orientation
Transverse Channel - West Orientation
Vertical Channel - Up Orientation

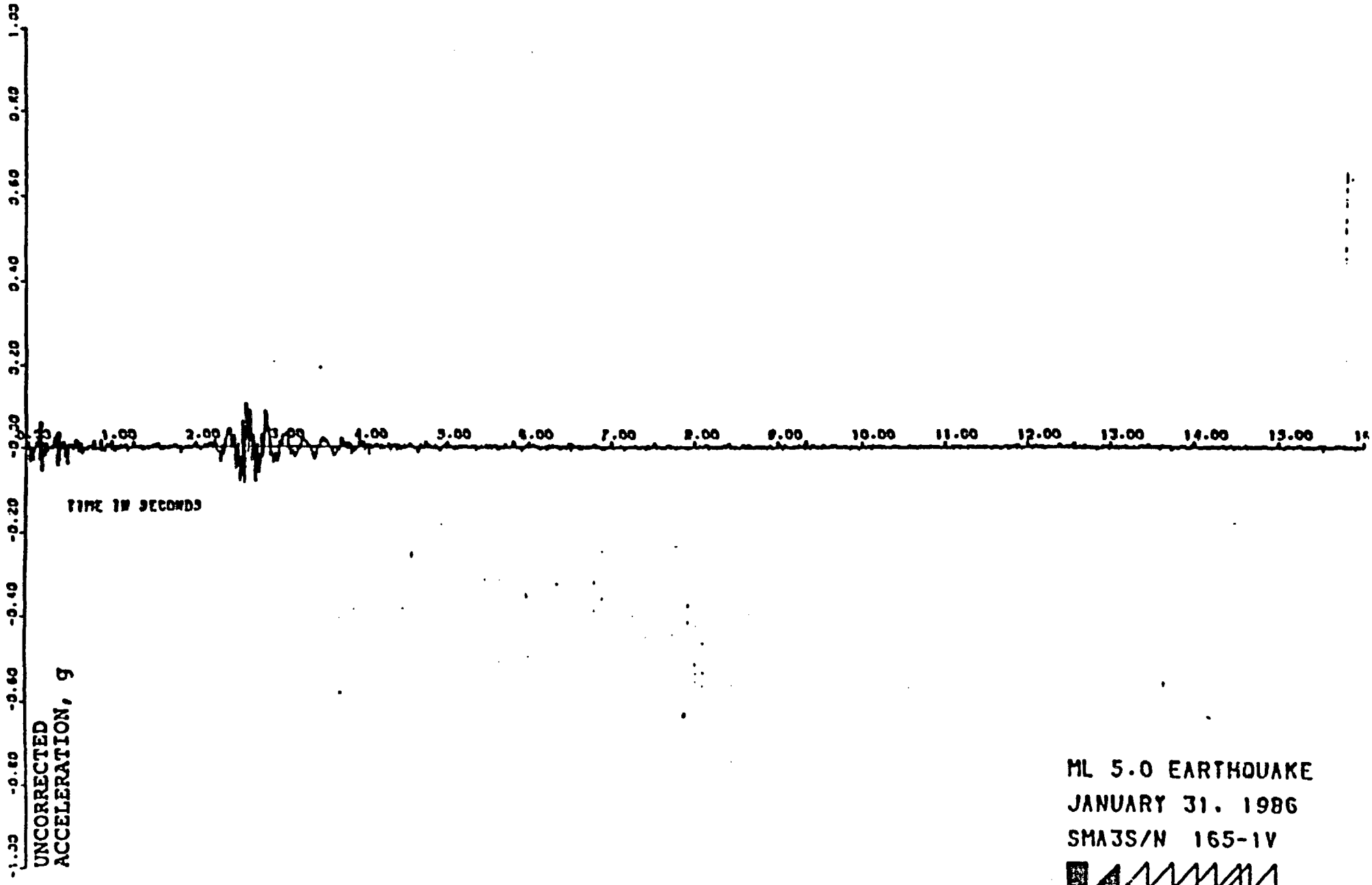


ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA3S/N 165-1L



ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA3S/N 165-1T

KINEMETRICS



ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA3S/N 165-1V

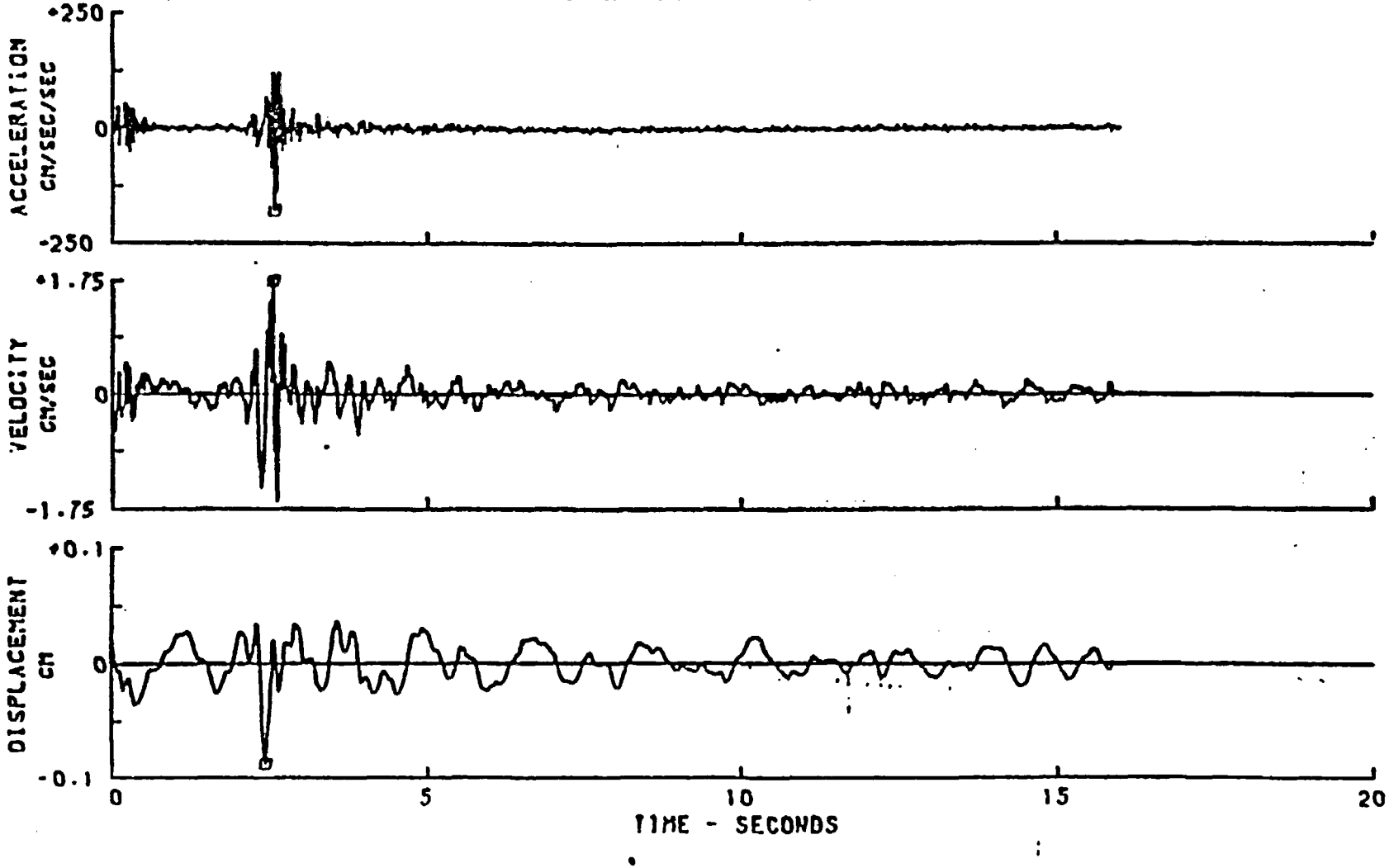
KINEMETRI

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

PEAK VALUES: ACCEL= -177.21 CM/SEC/SEC VEL= +1.74 CM/SEC DISPL=-0.09 CM



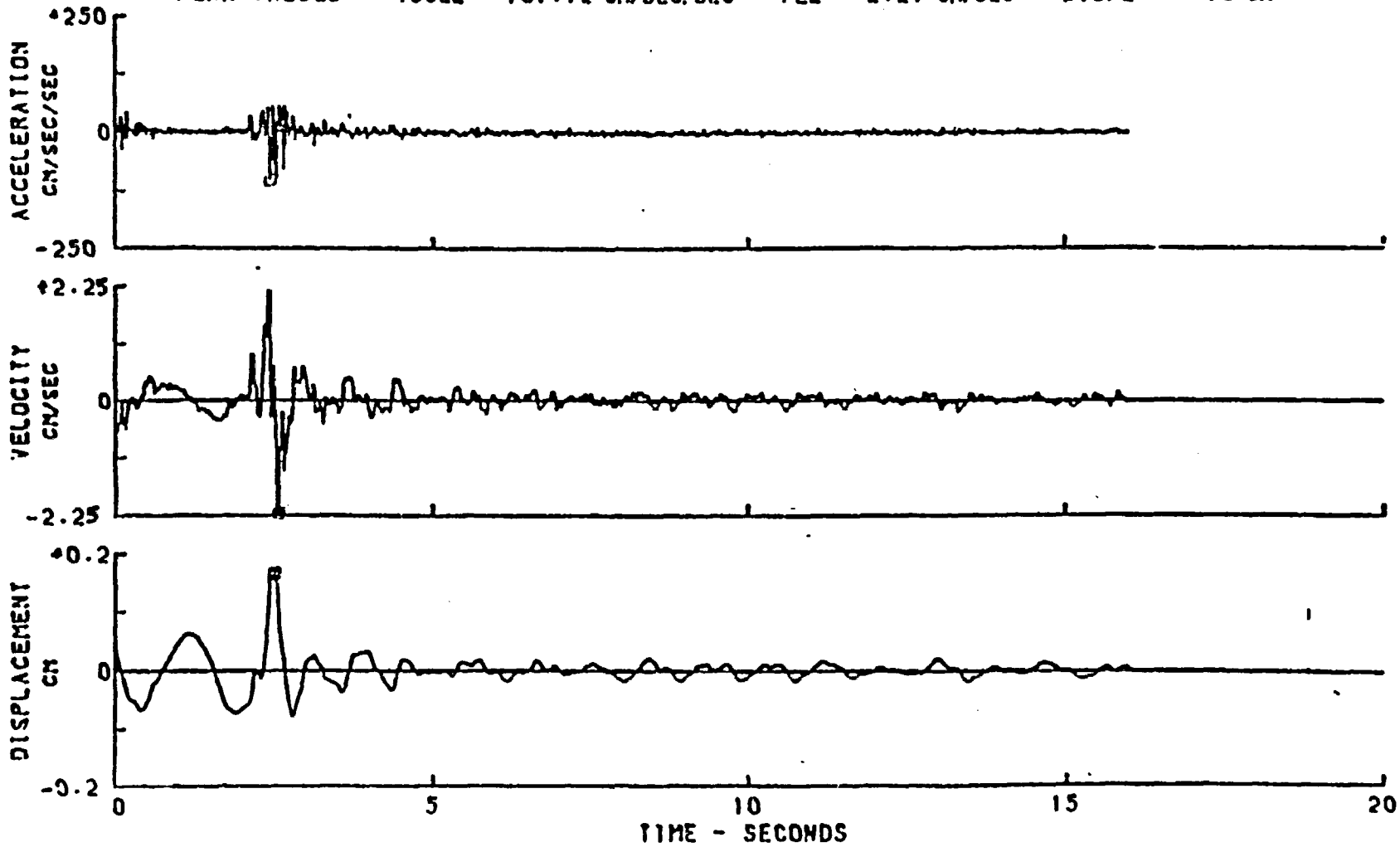
KINEMETRIC

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8001 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-11

ACCELEROGRAM IS BAND-PASS FILTERED BETWEEN 0.400- 0.625 AND 35.00- 40.00 HERTZ

PEAK VALUES: ACCEL= -101.12 CM/SEC/SEC VEL= -2.21 CM/SEC DISPL= +.16 CM



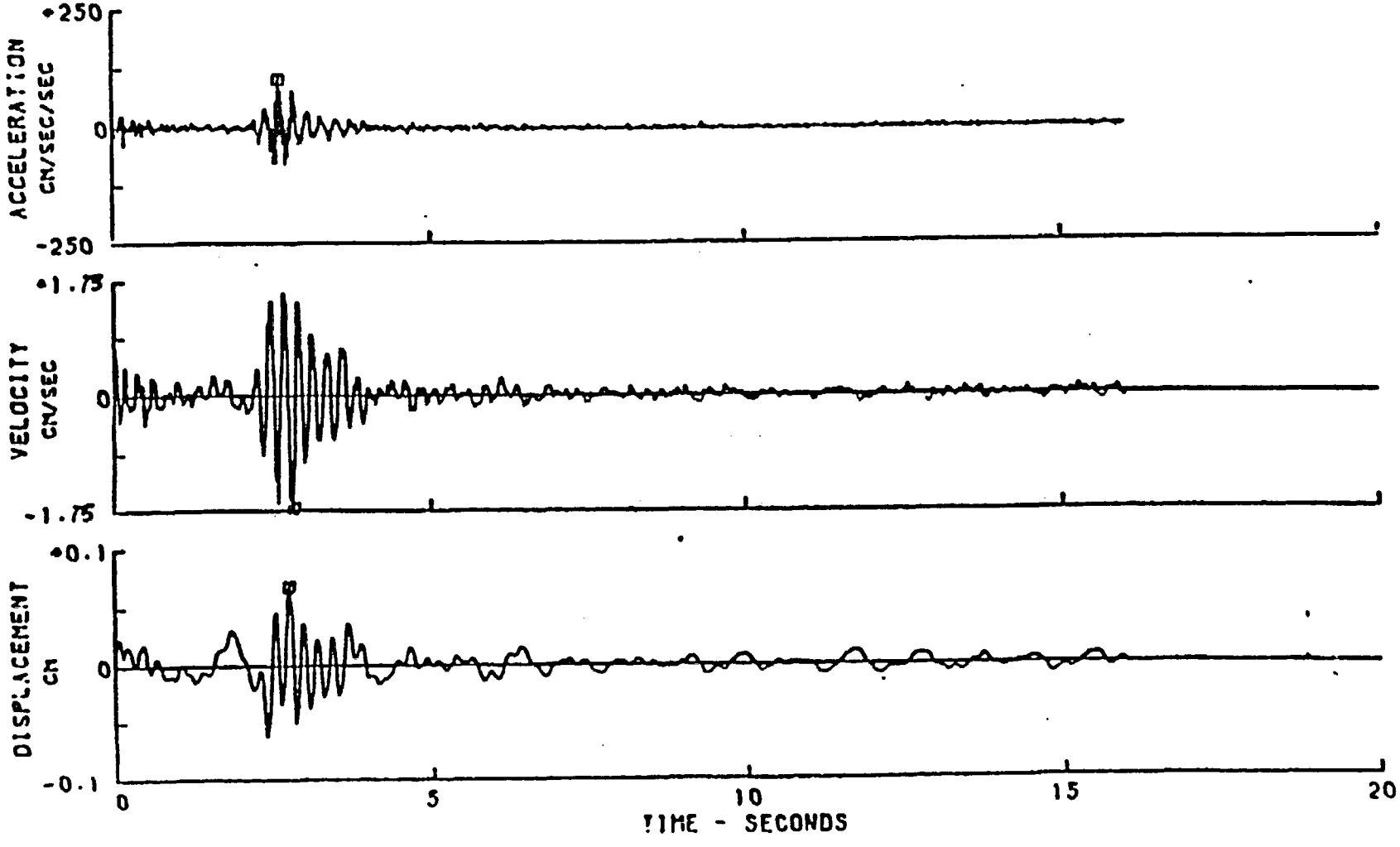
KINEMETRICS

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8001 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 165-1V

ACCELEROGRAM IS BAND-PASS FILTERED BETWEEN 0.400- 0.625 AND 35.00- 40.00 HERTZ

⊡ PEAK VALUES: ACCEL= +103.46 CM/SEC/SEC VEL= -1.71 CM/SEC DISPL= +.07 CM

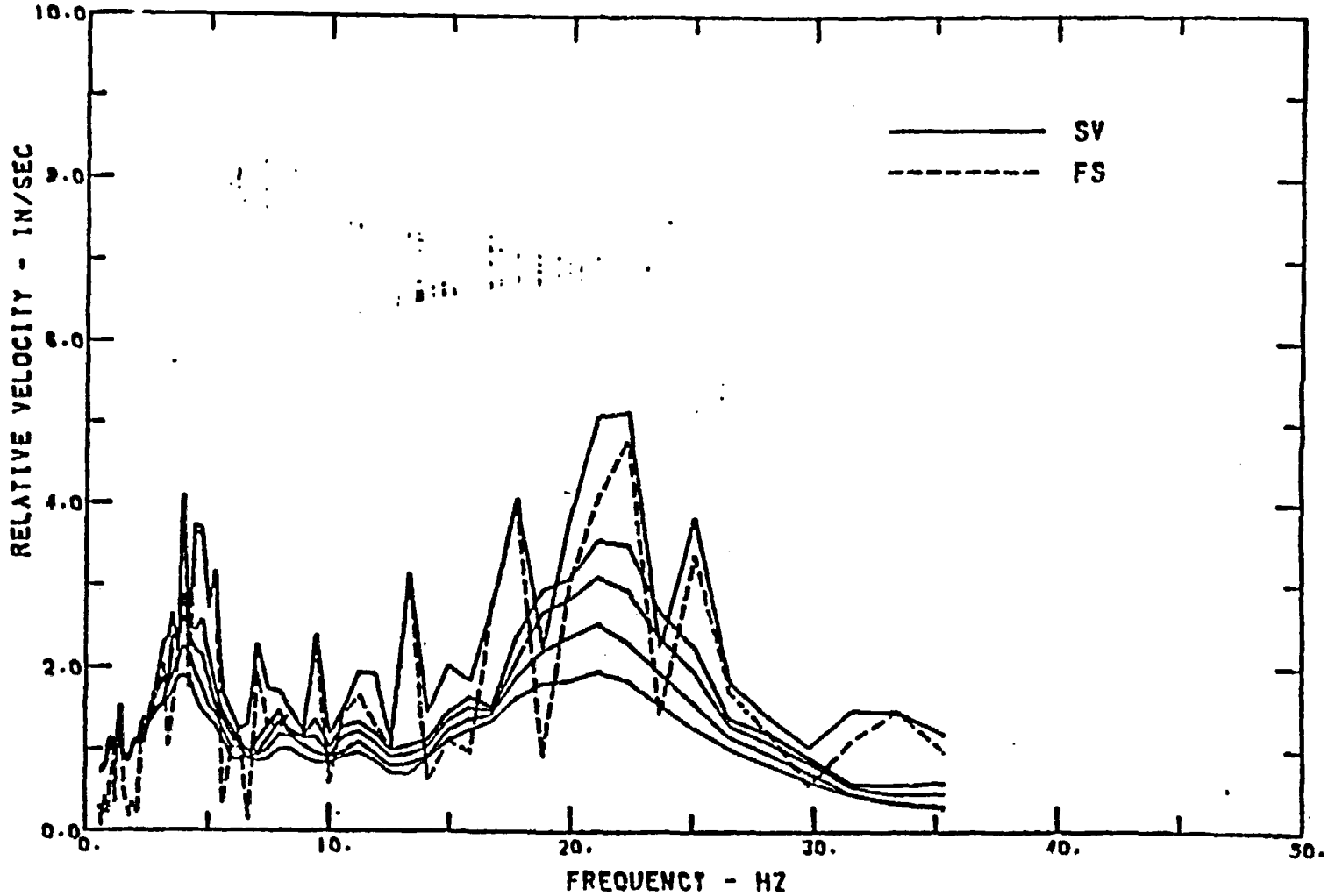


RELATIVE VELOCITY RESPONSE SPECTRUM

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8001 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-1L

DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



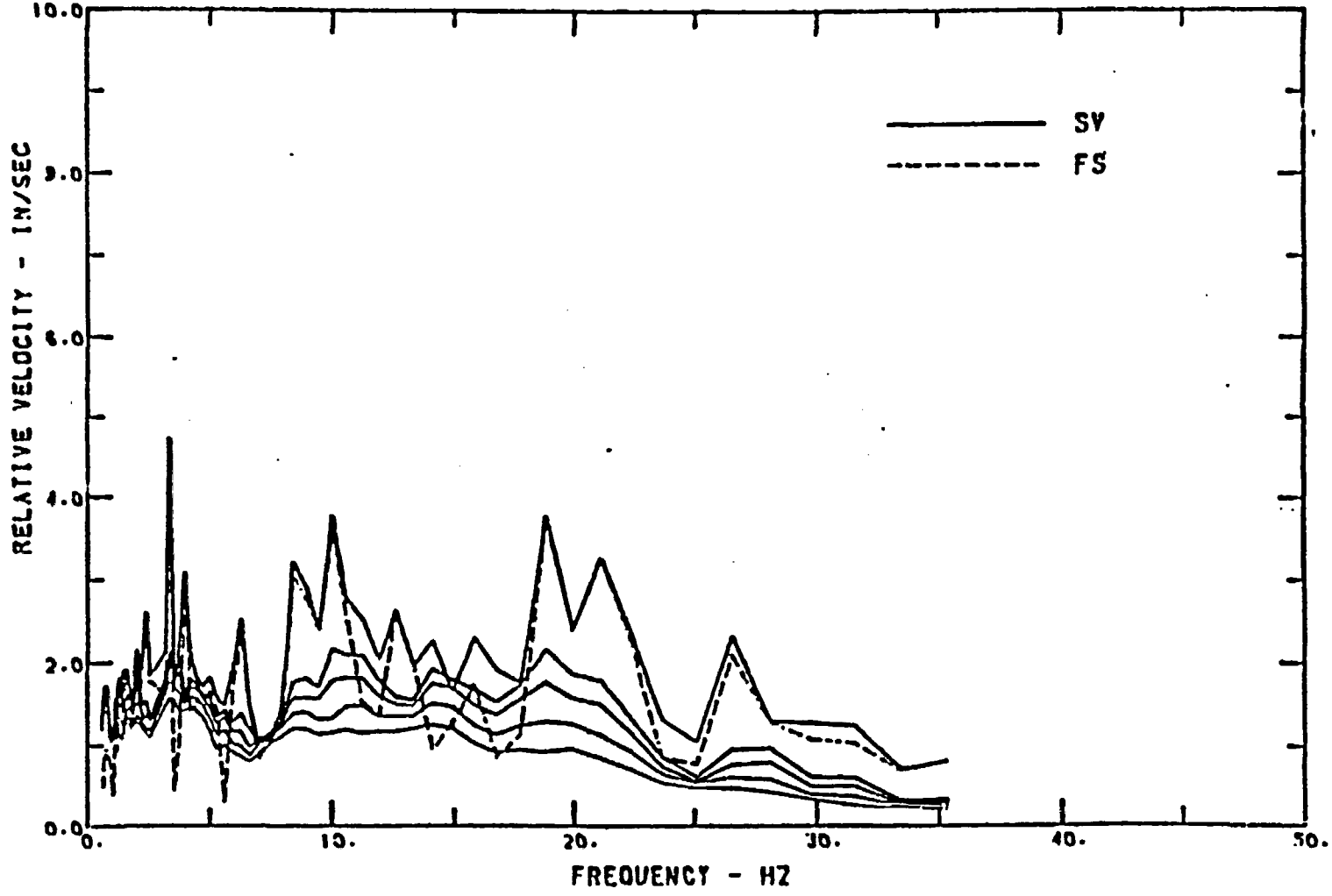
KINEMETRICS

RELATIVE VELOCITY RESPONSE SPECTRUM

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8001 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-1T

DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



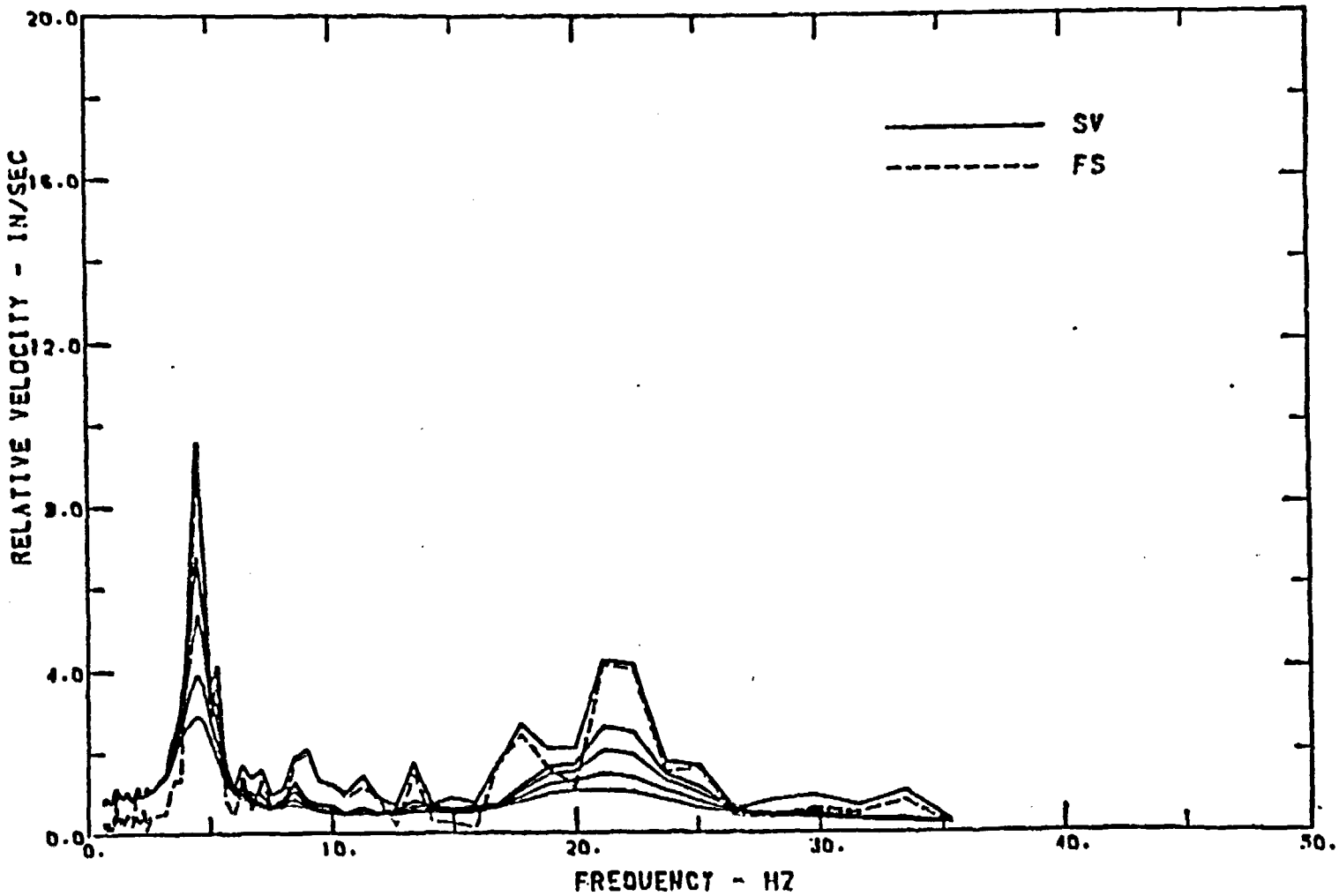
KINEMETRICS

RELATIVE VELOCITY RESPONSE SPECTRUM

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8001 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 165-1V

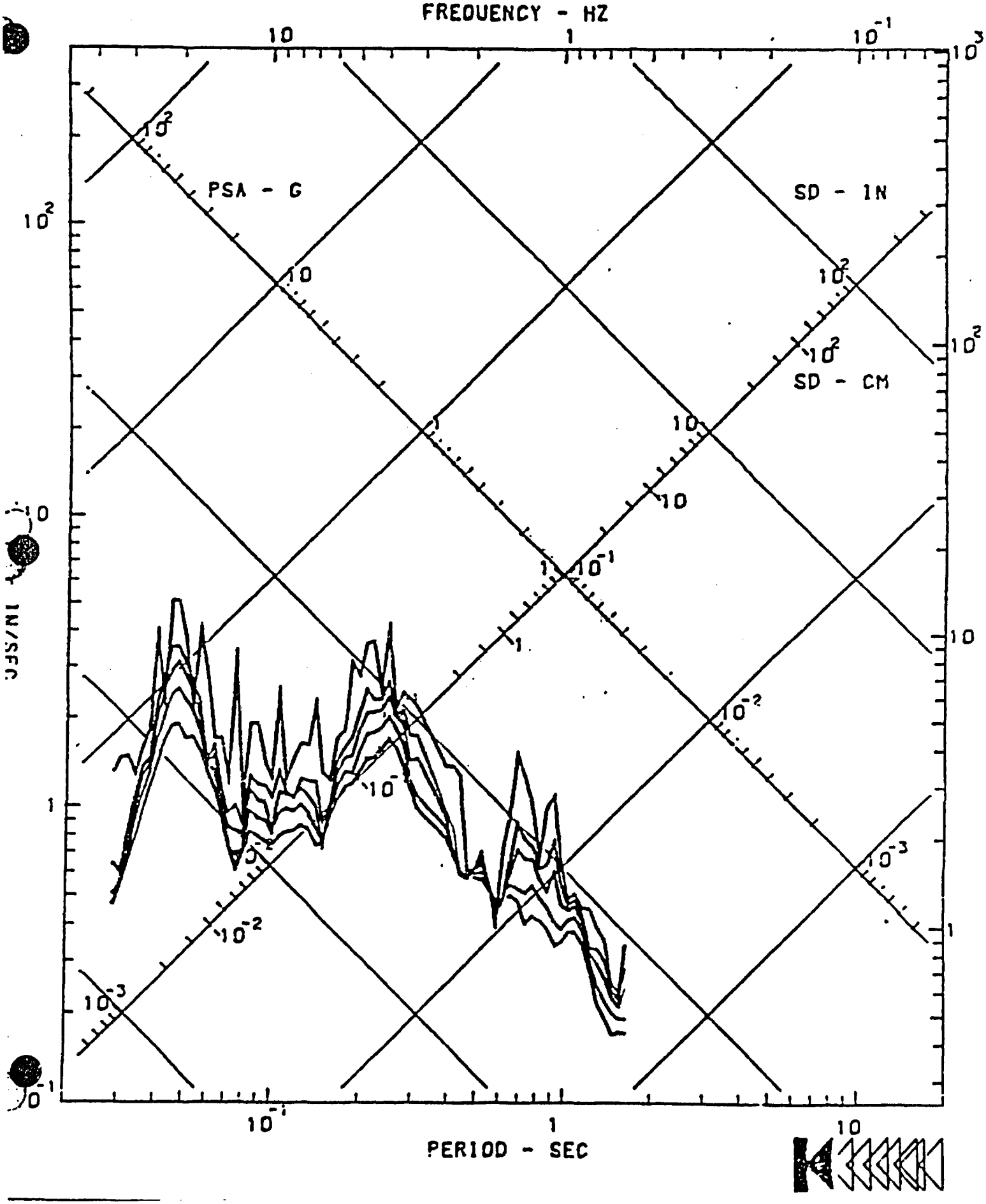
DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



KINEMETRICS

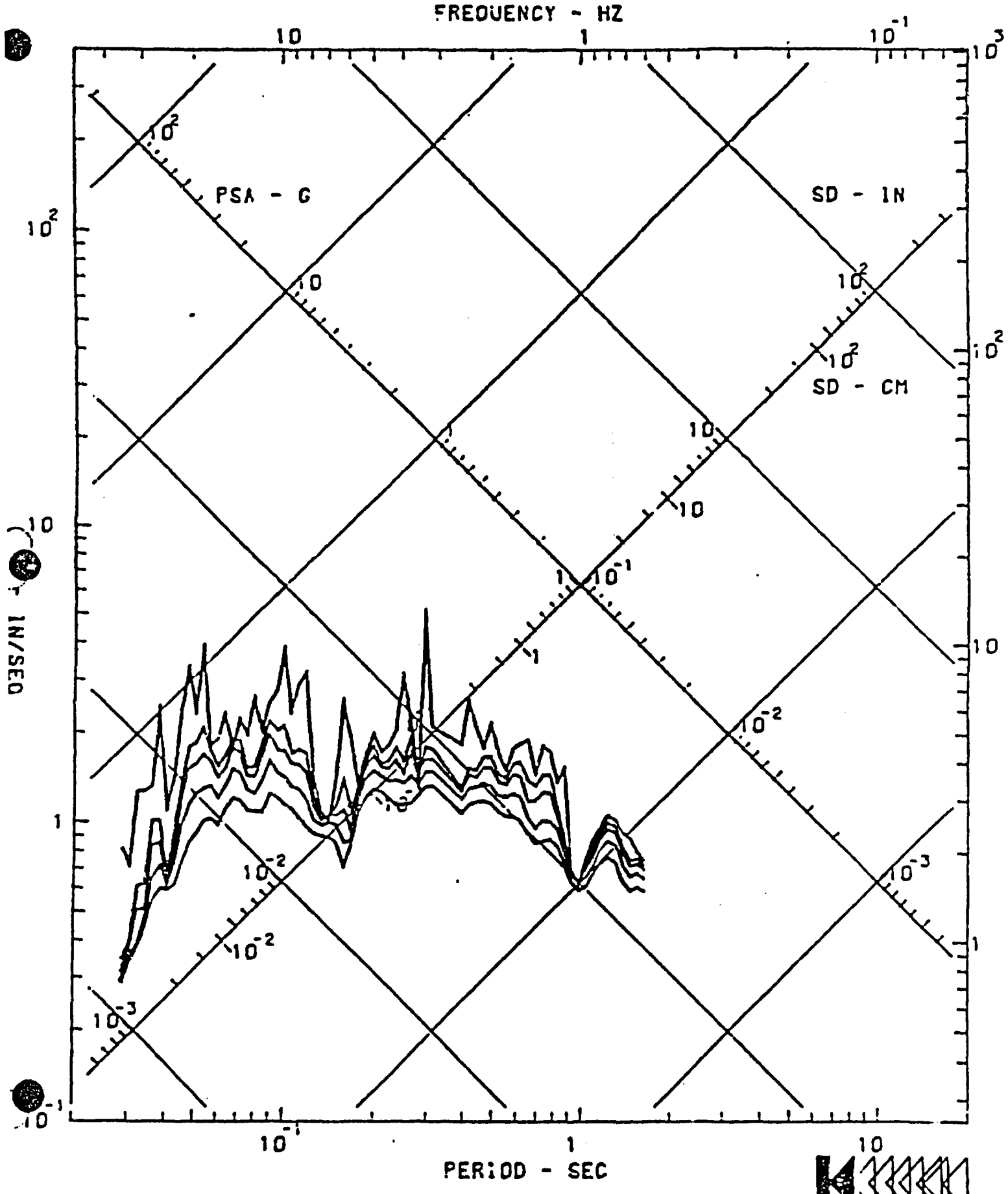
11A8001 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-1L

DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



11A6001 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-1T

DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL

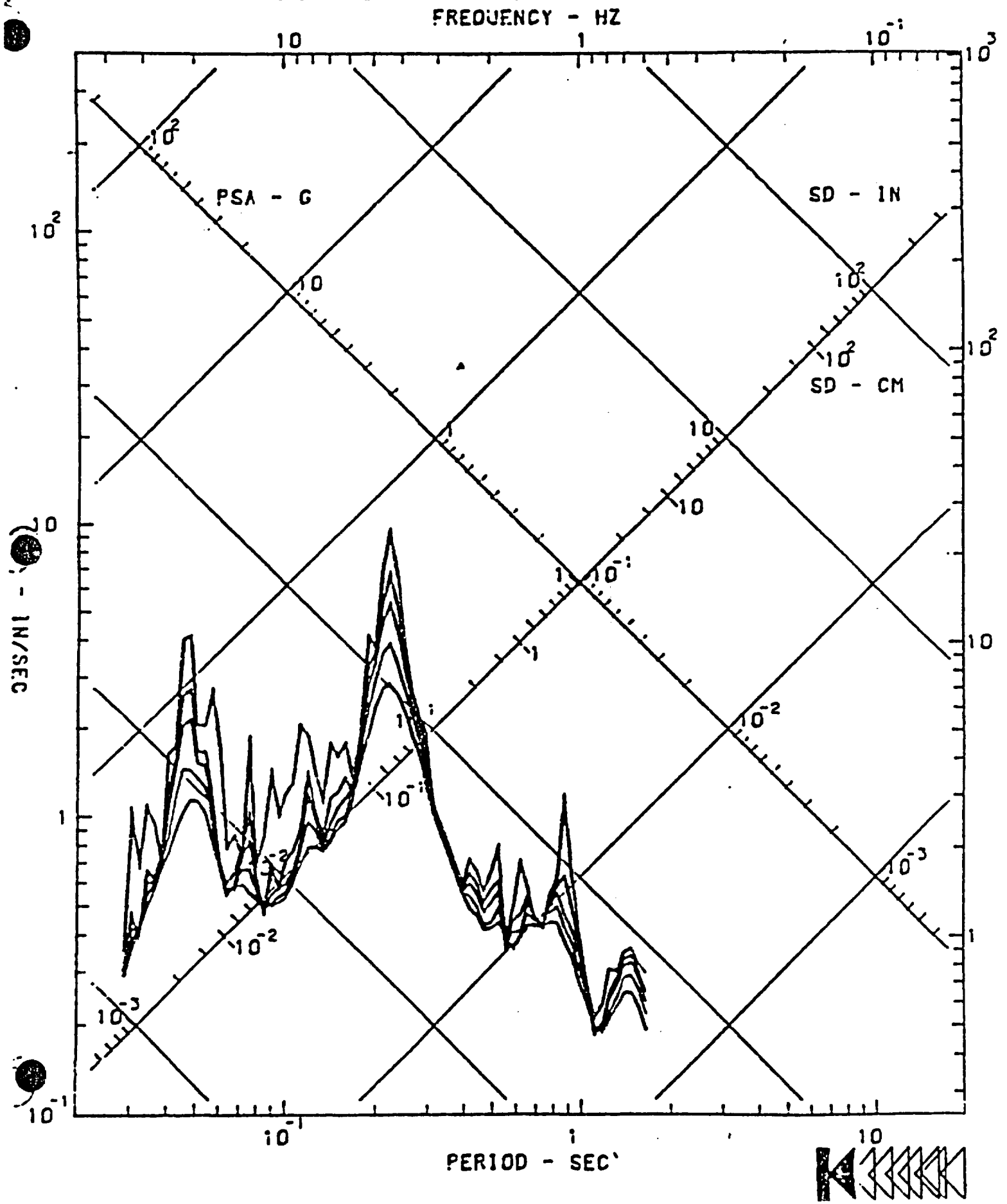


KINEMETRICS

ML 5.0 EARTHQUAKE JANUARY 31, 1986

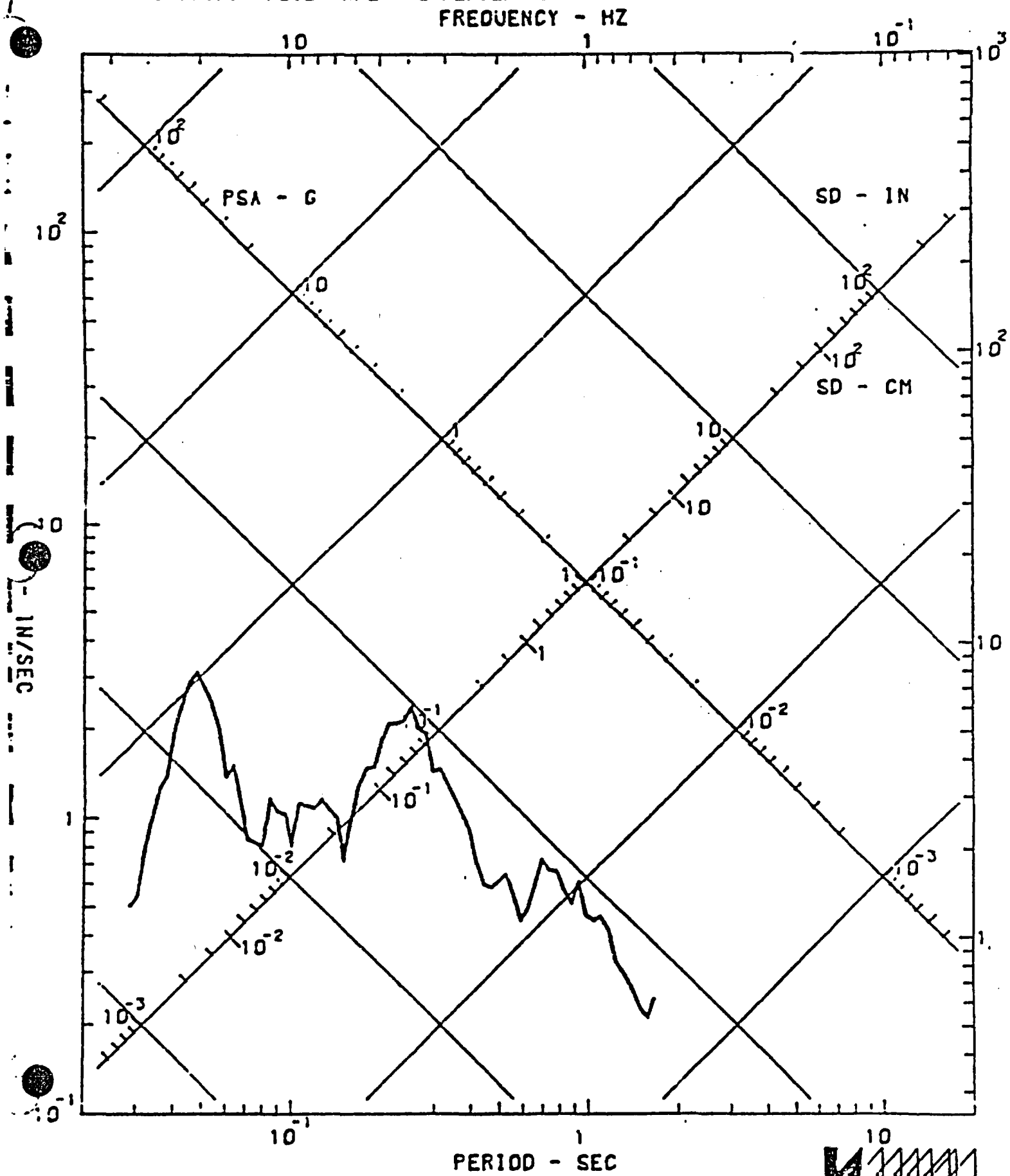
11AB001 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 165-1V

DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



11A8001 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-1L

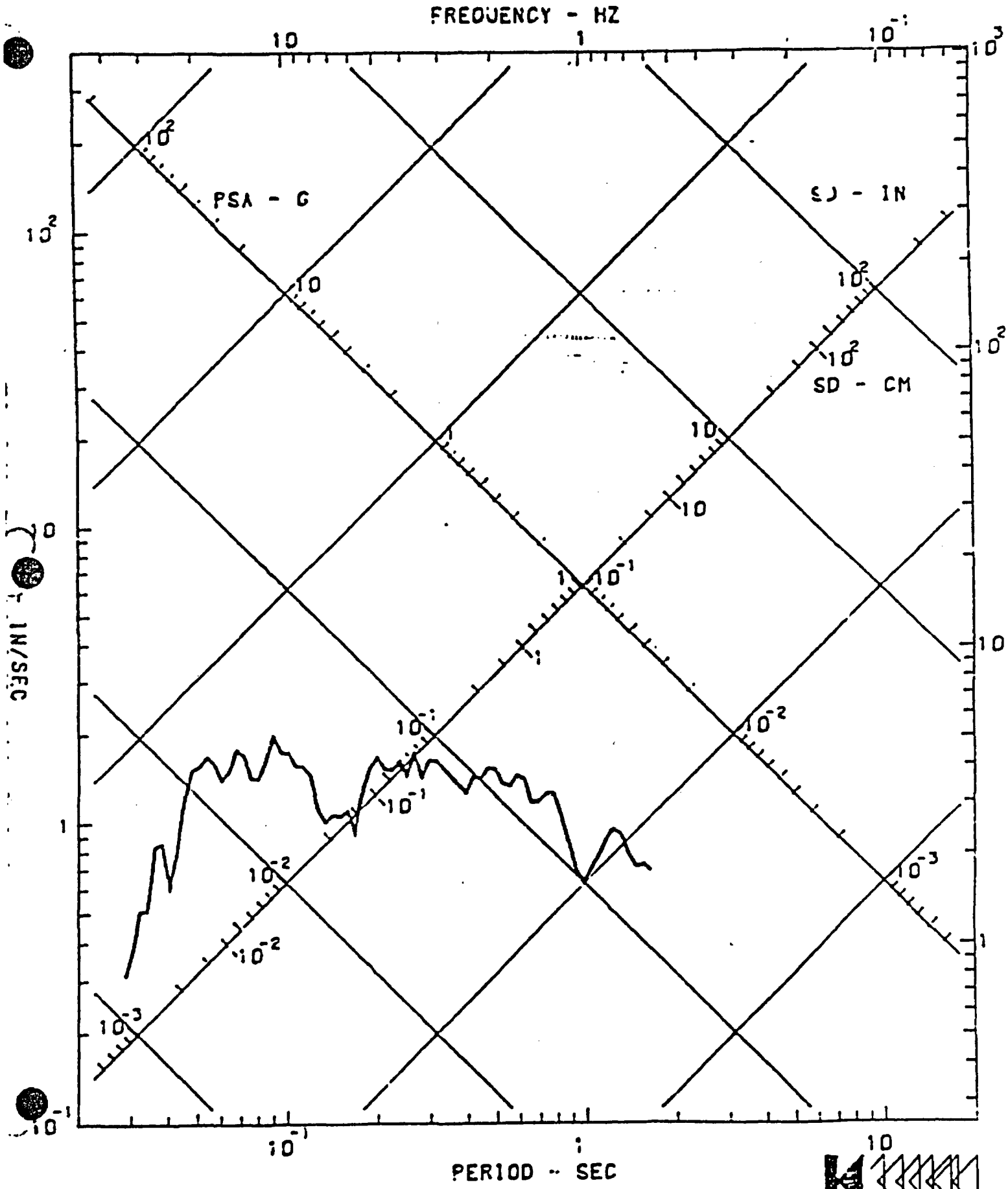
DAMPING VALUES ARE 2 PERCENT OF CRITICAL



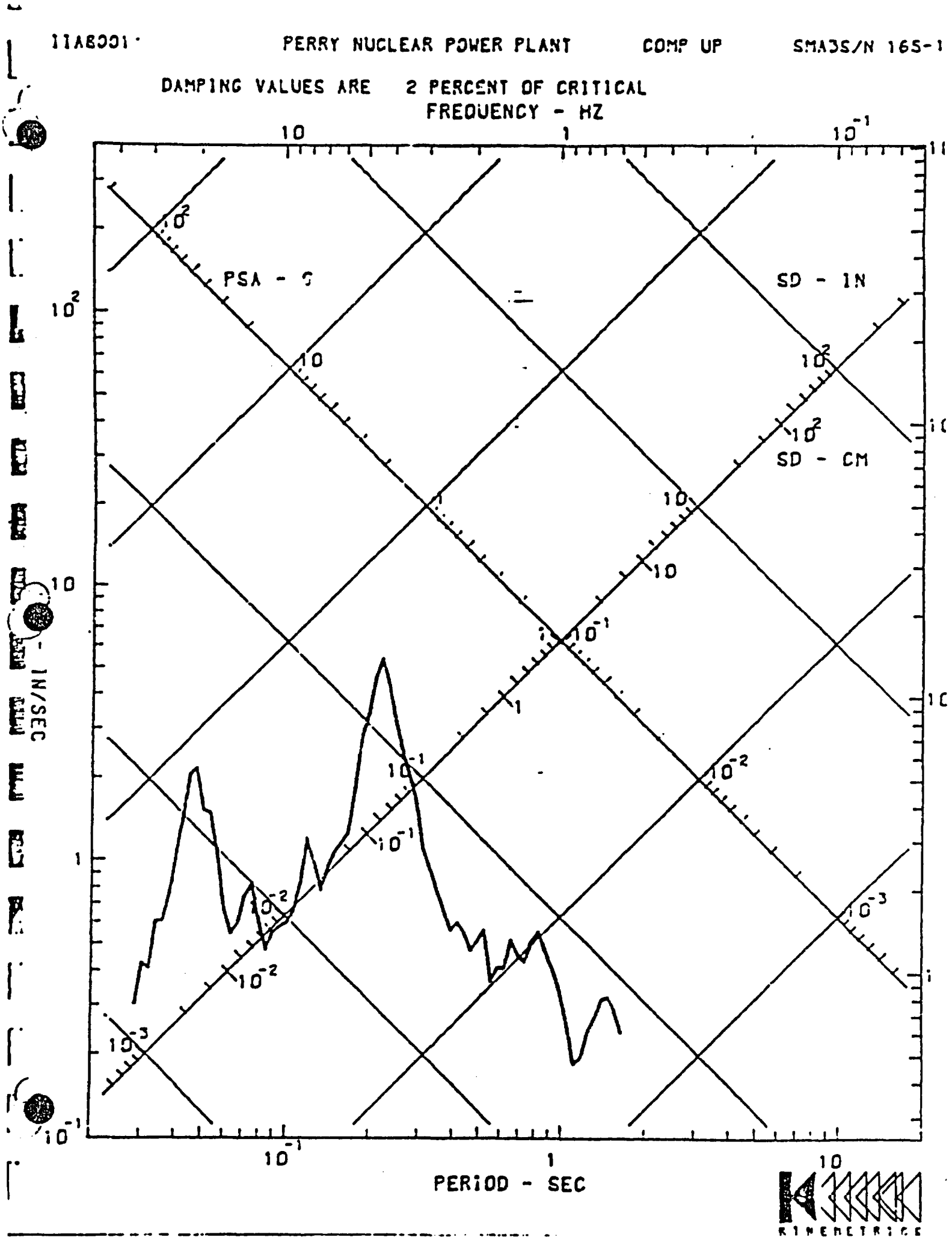
KINEMETRICS

11AB001 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-1T

DAMPING VALUES ARE 2 PERCENT OF CRITICAL

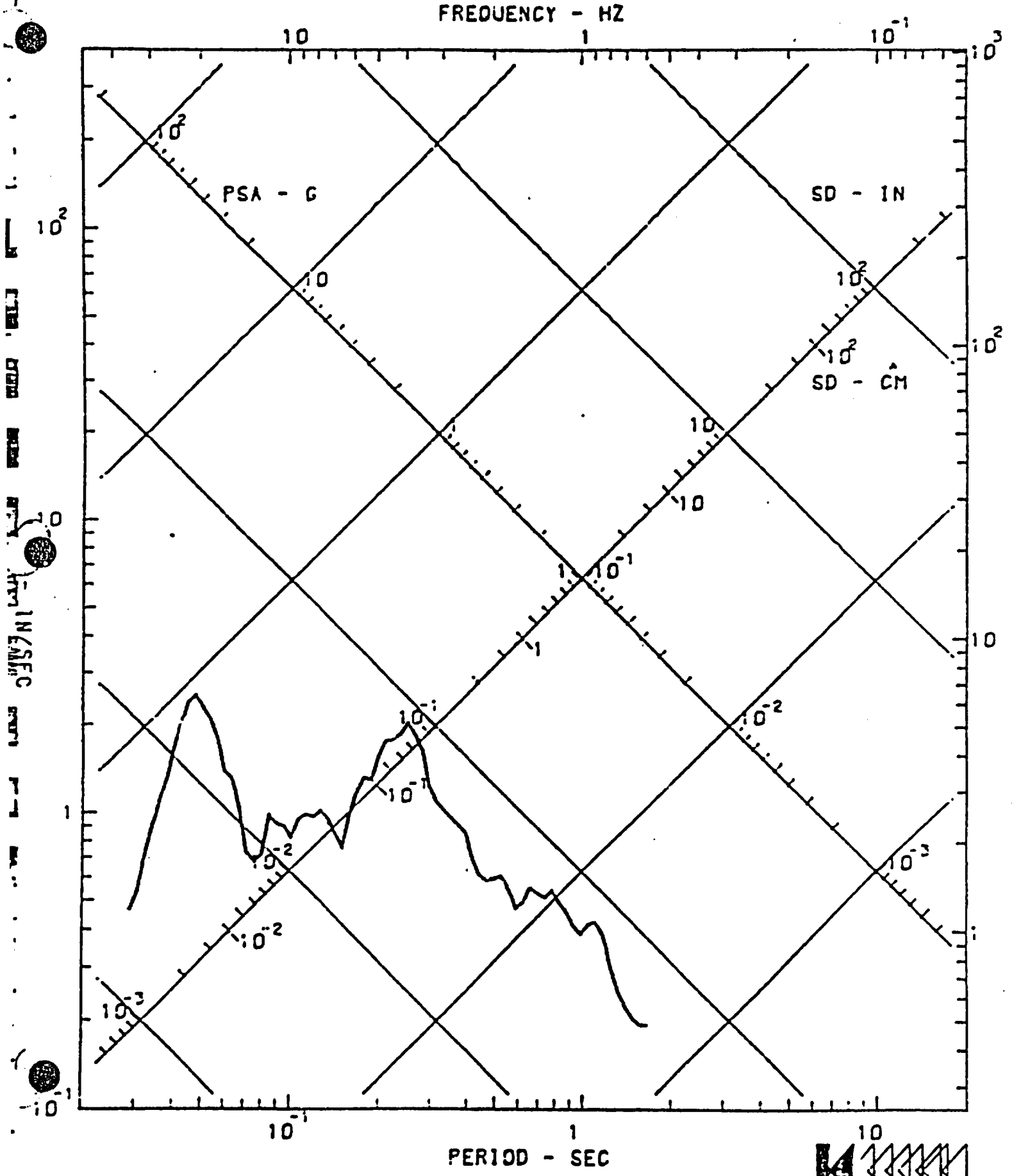


KINEMETRICS


11A8001
PERRY NUCLEAR POWER PLANT
COMP UP
SMA3S/N 165-1
DAMPING VALUES ARE 2 PERCENT OF CRITICAL
FREQUENCY - HZ
PSA - G
SD - IN
SD - CM
IN/SEC
PERIOD - SEC
KINEMETRICS

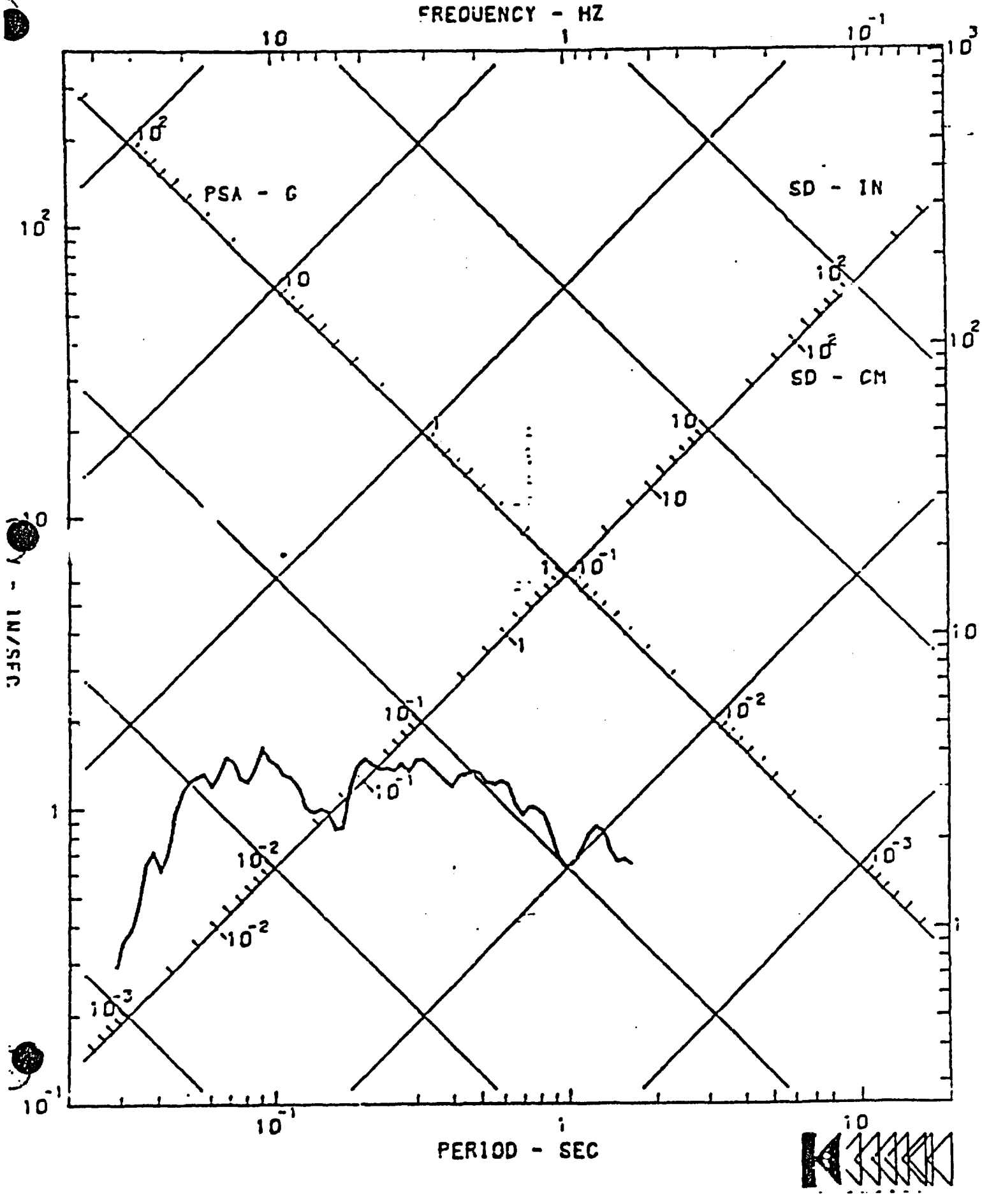
11A6001 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-1L

DAMPING VALUES ARE 0 PERCENT OF CRITICAL



11A6001 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-1T

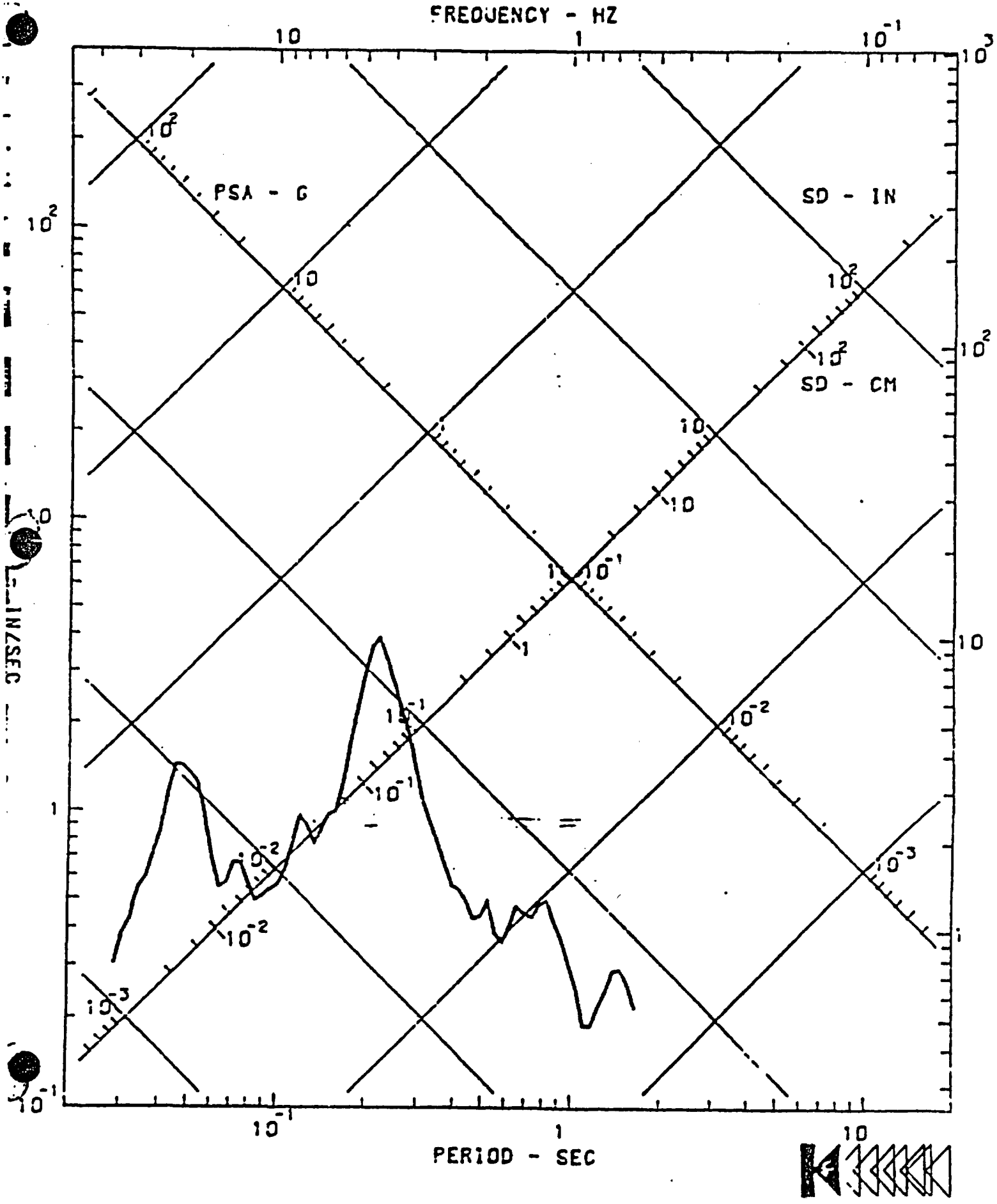
DAMPING VALUES ARE 4 PERCENT OF CRITICAL



HL 5.0 EARTHQUAKE JANUARY 31, 1986

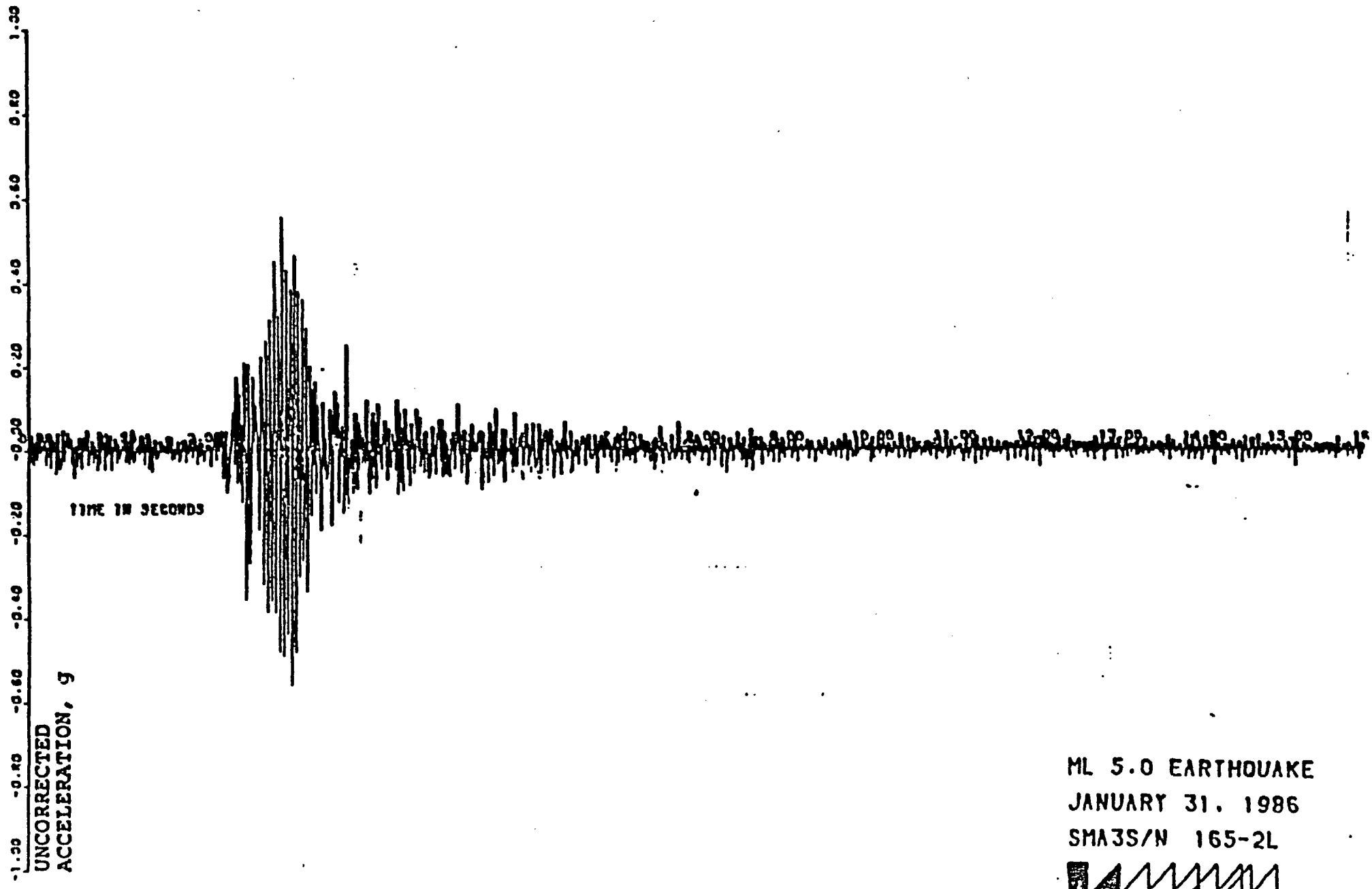
11AB001 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 165-1V

DAMPING VALUES ARE 4 PERCENT OF CRITICAL

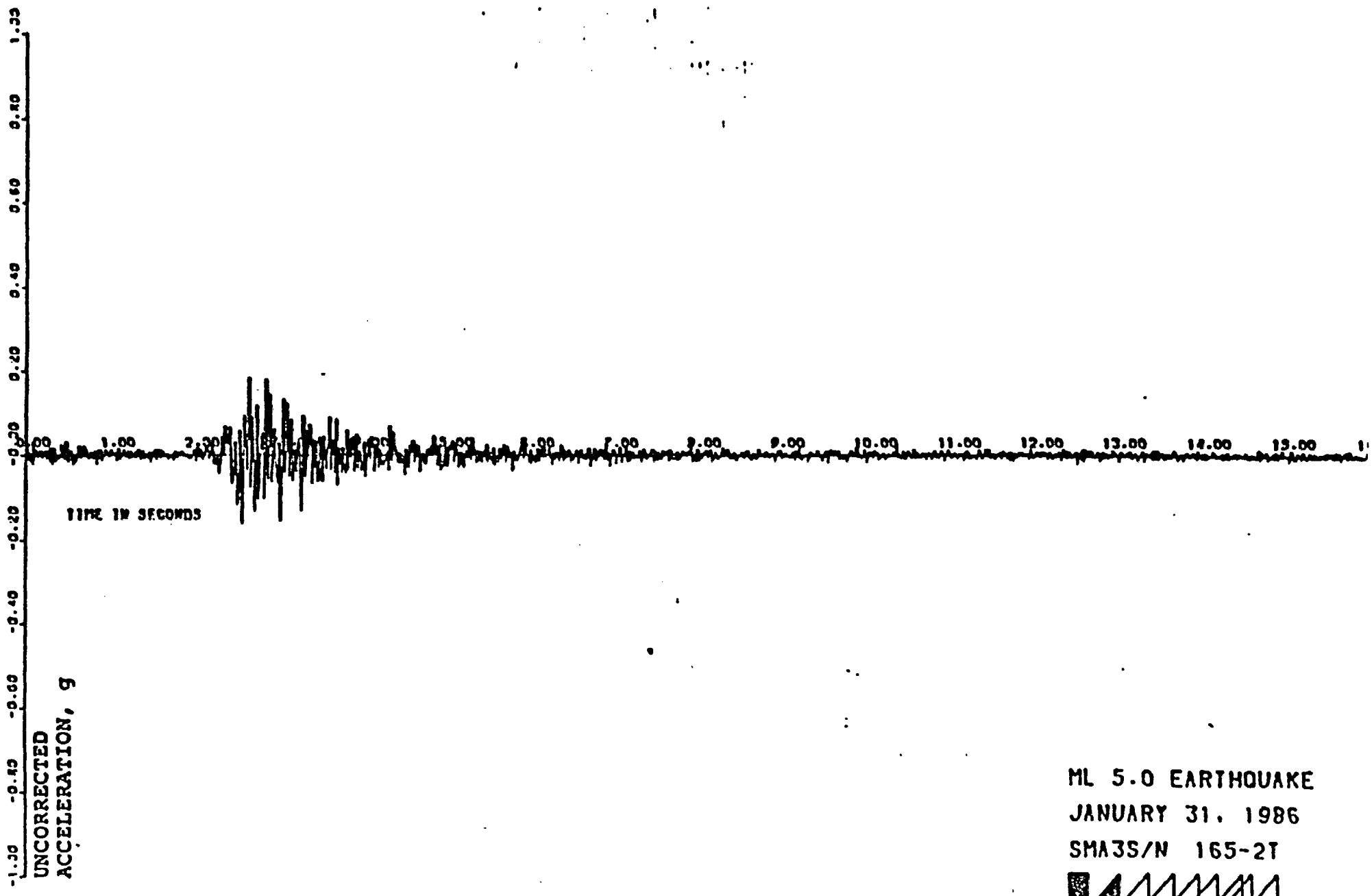


Containment Vessel Annulus, Elevation 682 Ft.

SMA-3 Serial Number 165-2
Tag Number D51-N111
Longitudinal Channel - South Orientation
Transverse Channel - West Orientation
Vertical Channel - Up Orientation

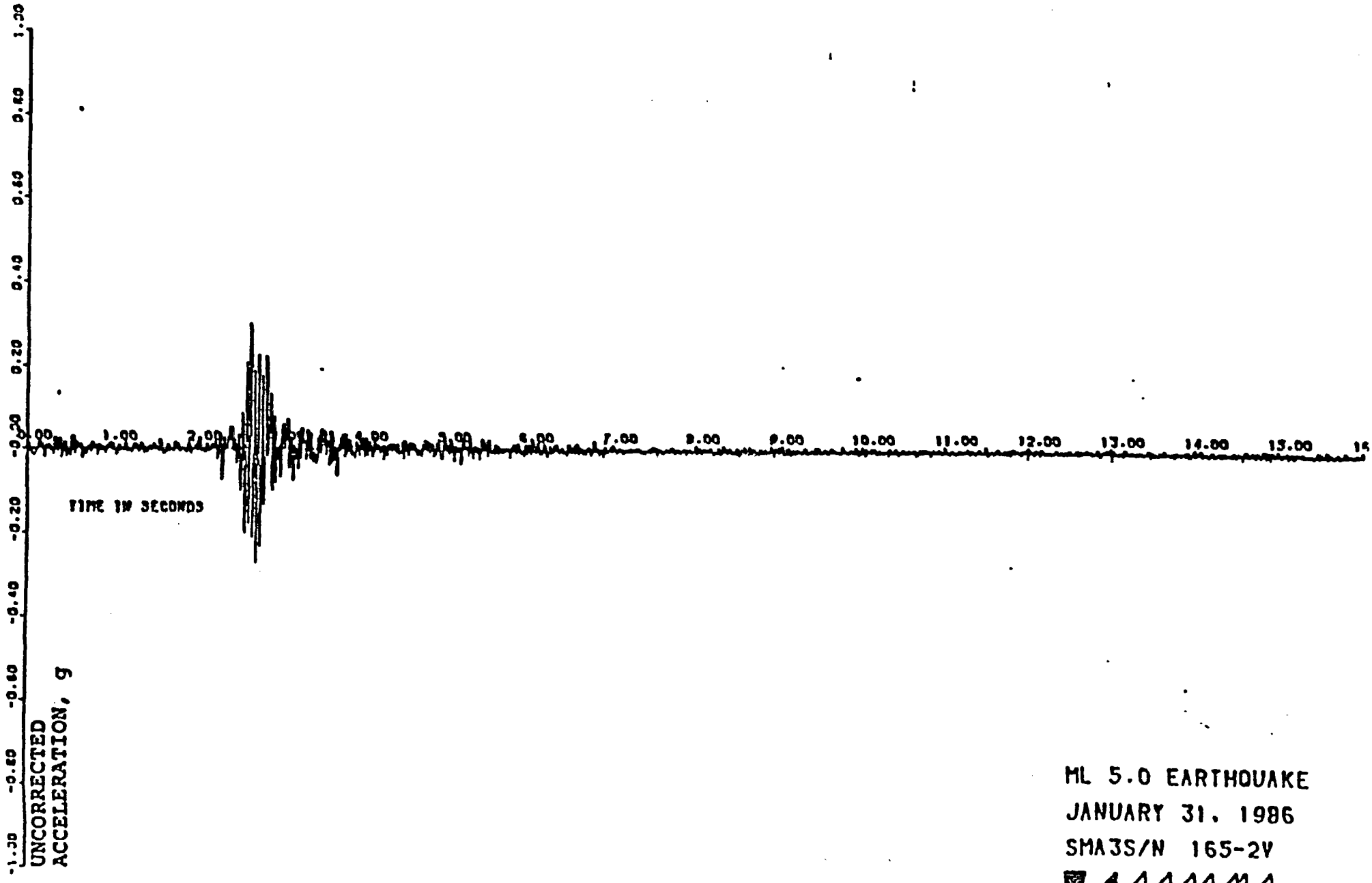


ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA3S/N 165-2L



ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA3S/N 165-2T

KINEMETRICS



ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA3S/N 165-2V

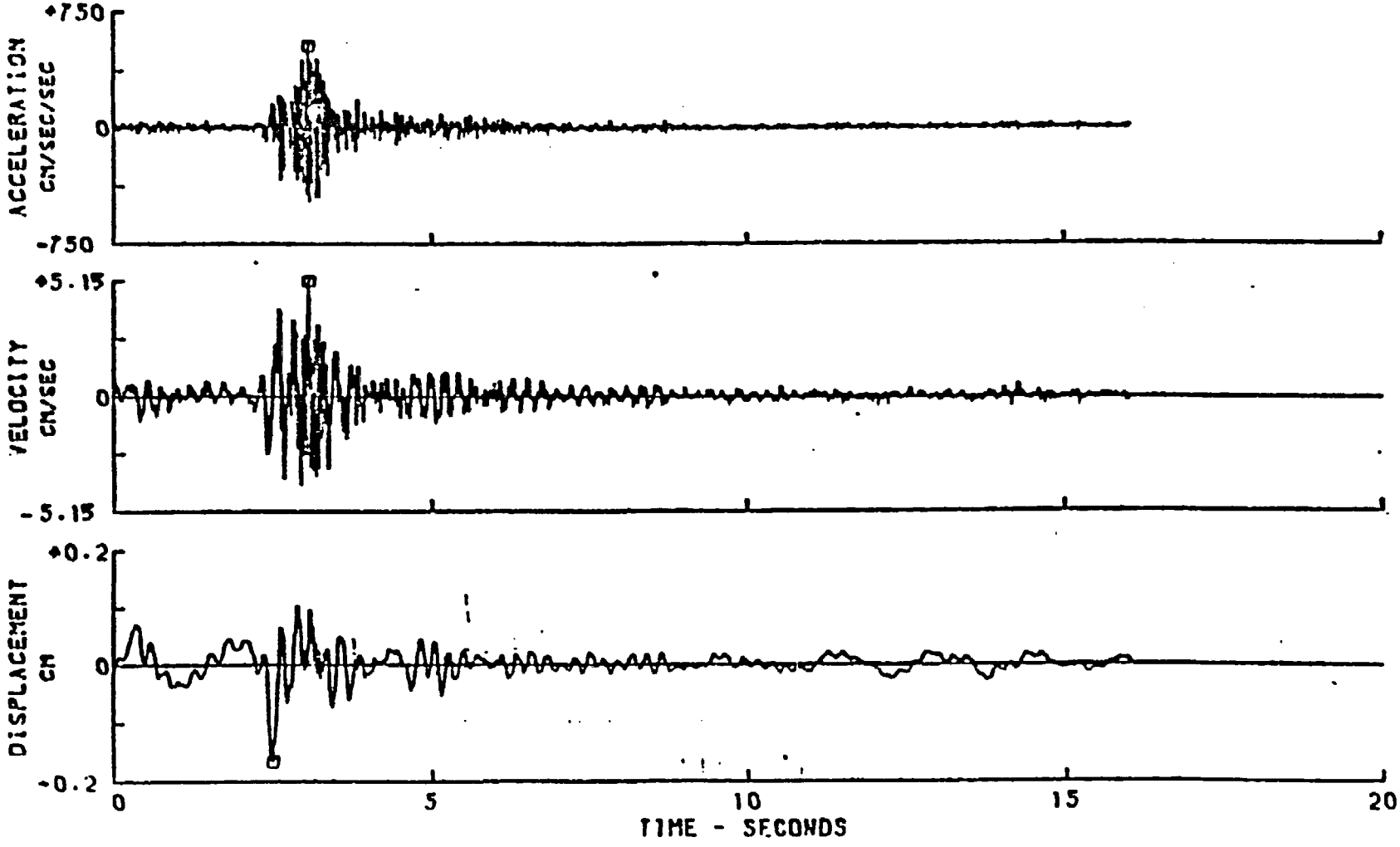
KINEMETRIC

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8002 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-2L

ACCELEROGRAM IS BAND-PASS FILTERED BETWEEN 0.400- 0.625 AND 35.00- 40.00 HERTZ

⊡ PEAK VALUES: ACCEL= +535.17 CM/SEC/SEC VEL= +5.13 CM/SEC DISPL=-0.17 CM



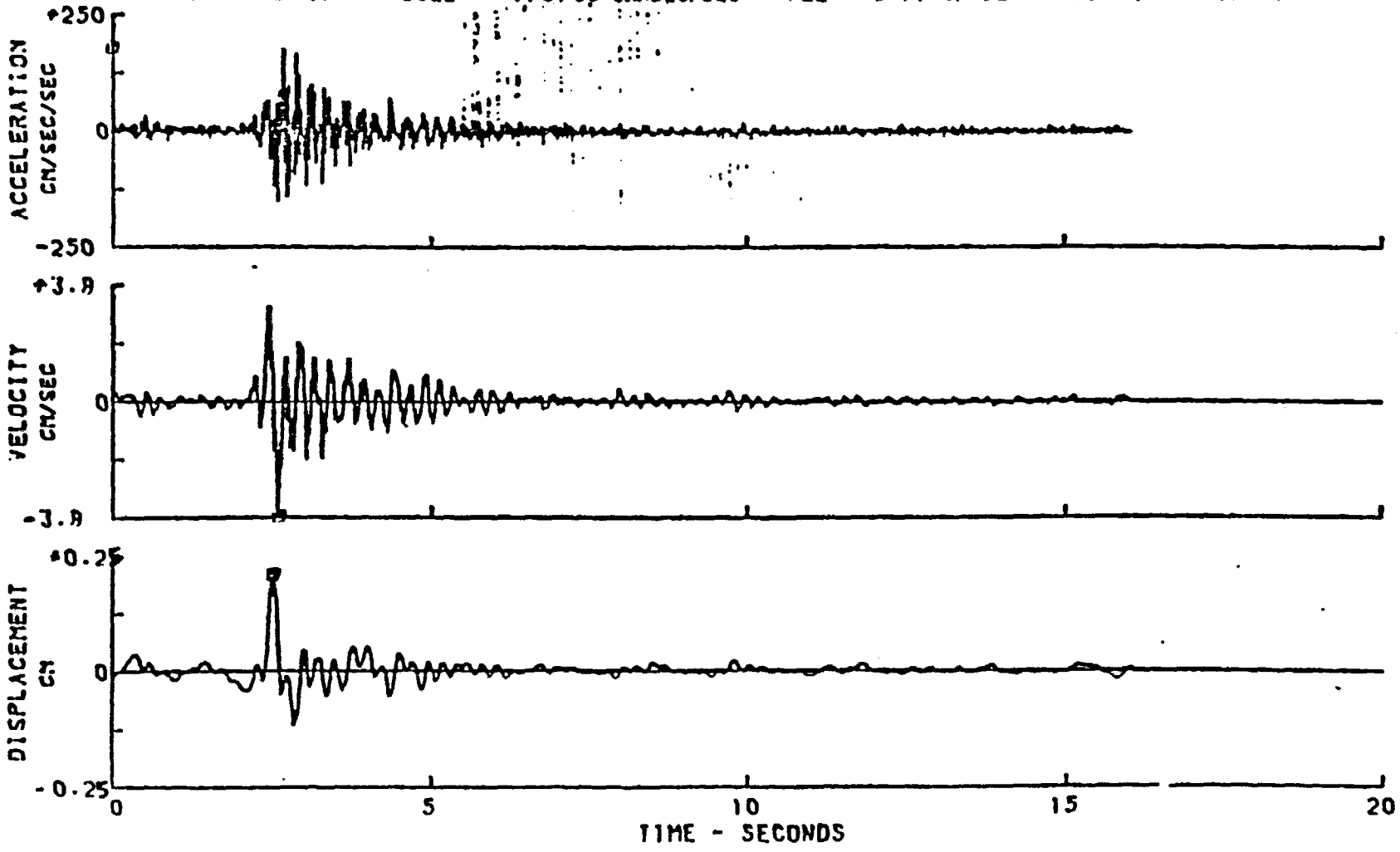
KINEMETRIC

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8002 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-2T

ACCELEROGRAM IS BAND-PASS FILTERED BETWEEN 0.400- 0.625 AND 35.00- 40.00 HERTZ

PEAK VALUES: ACCEL= +178.35 CM/SEC/SEC VEL= -3.77 CM/SEC DISPL= +.21 CM



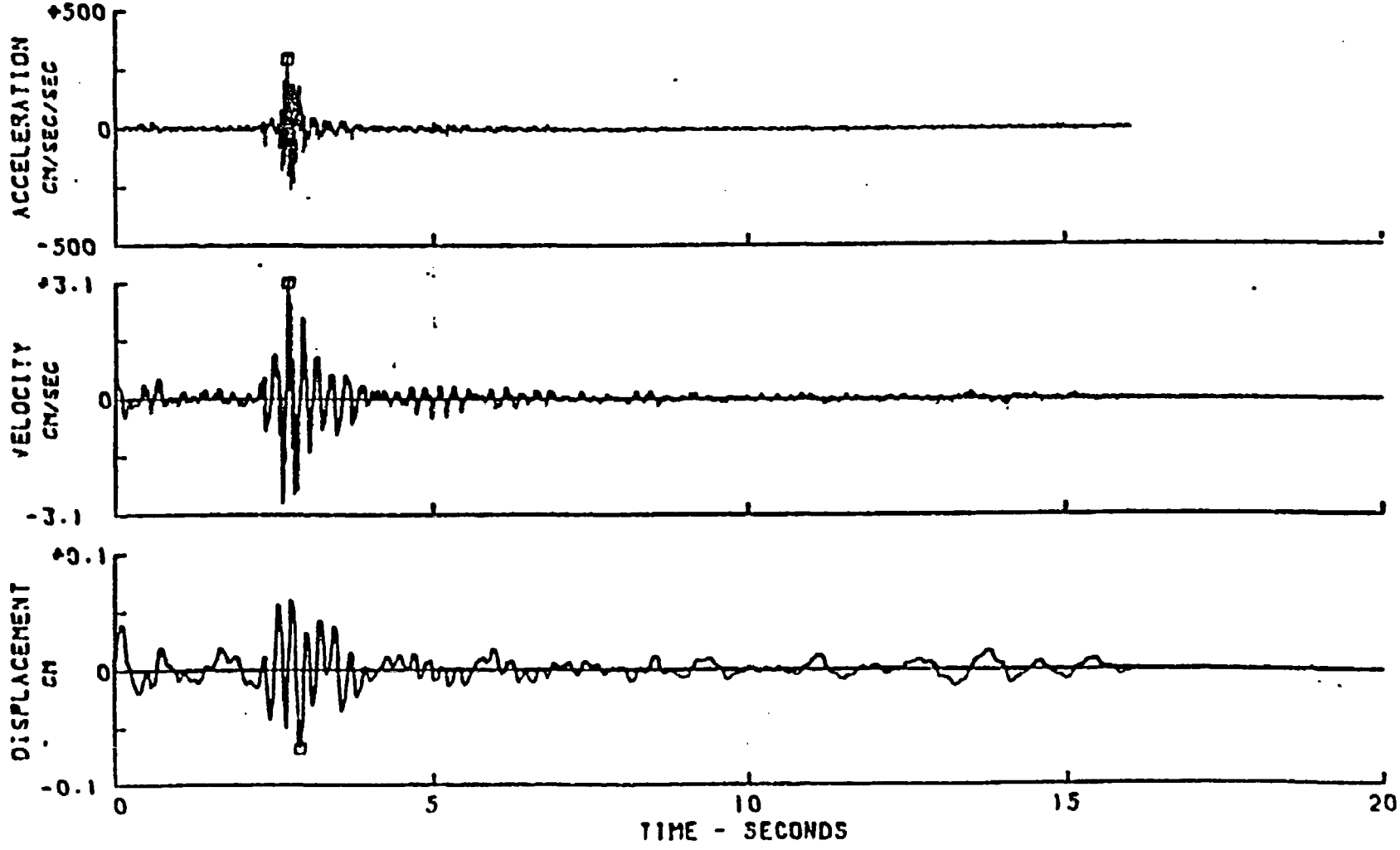
KINEMETRICS

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8002 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 165-2V

ACCELEROGRAM IS BAND-PASS FILTERED BETWEEN 0.400- 0.625 AND 35.00- 40.00 HERTZ

⊡ PEAK VALUES: ACCEL= +297.21 CM/SEC/SEC VEL= +3.09 CM/SEC DISPL=0.07 CM

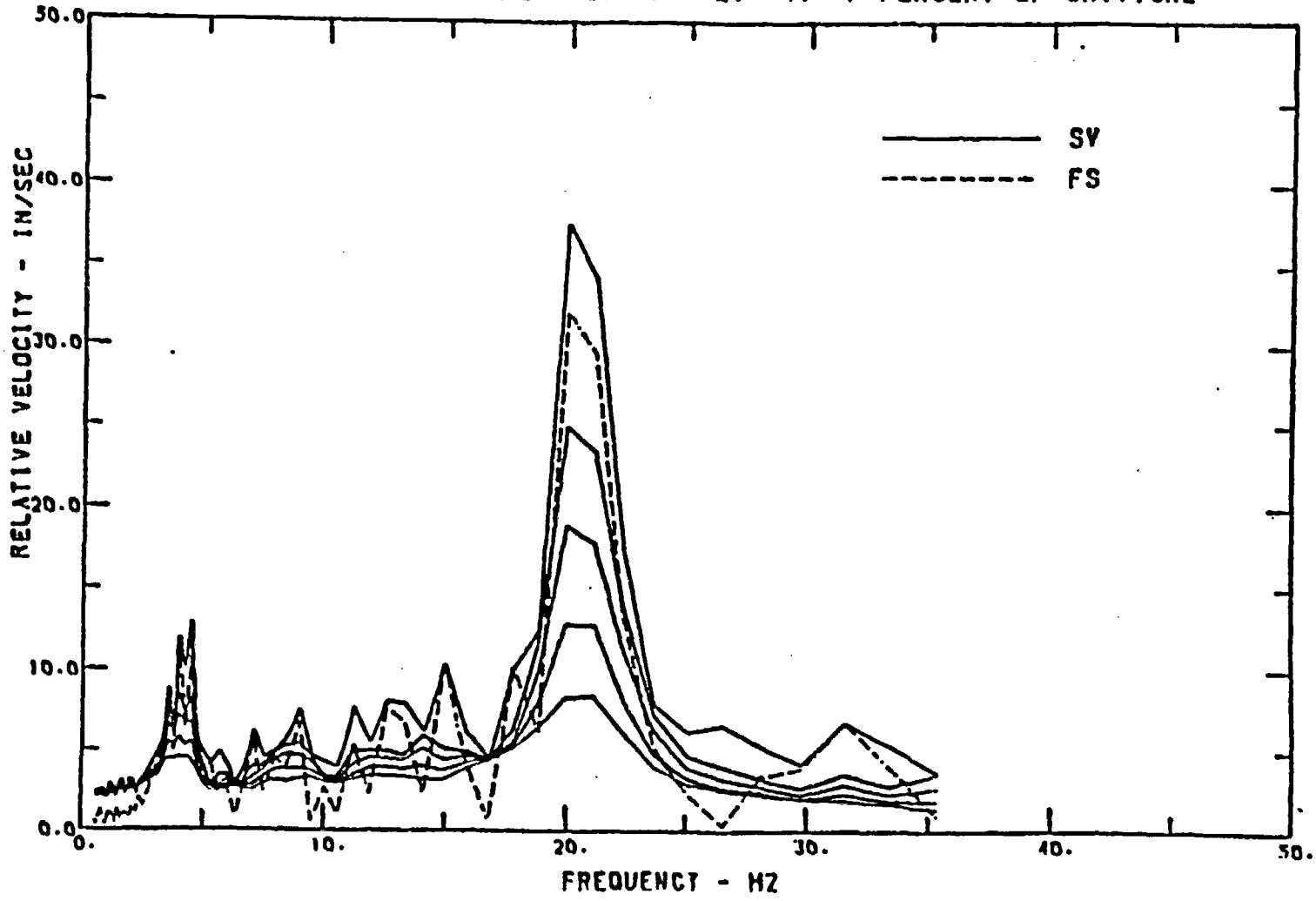


RELATIVE VELOCITY RESPONSE SPECTRUM

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A8002 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-2L

DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



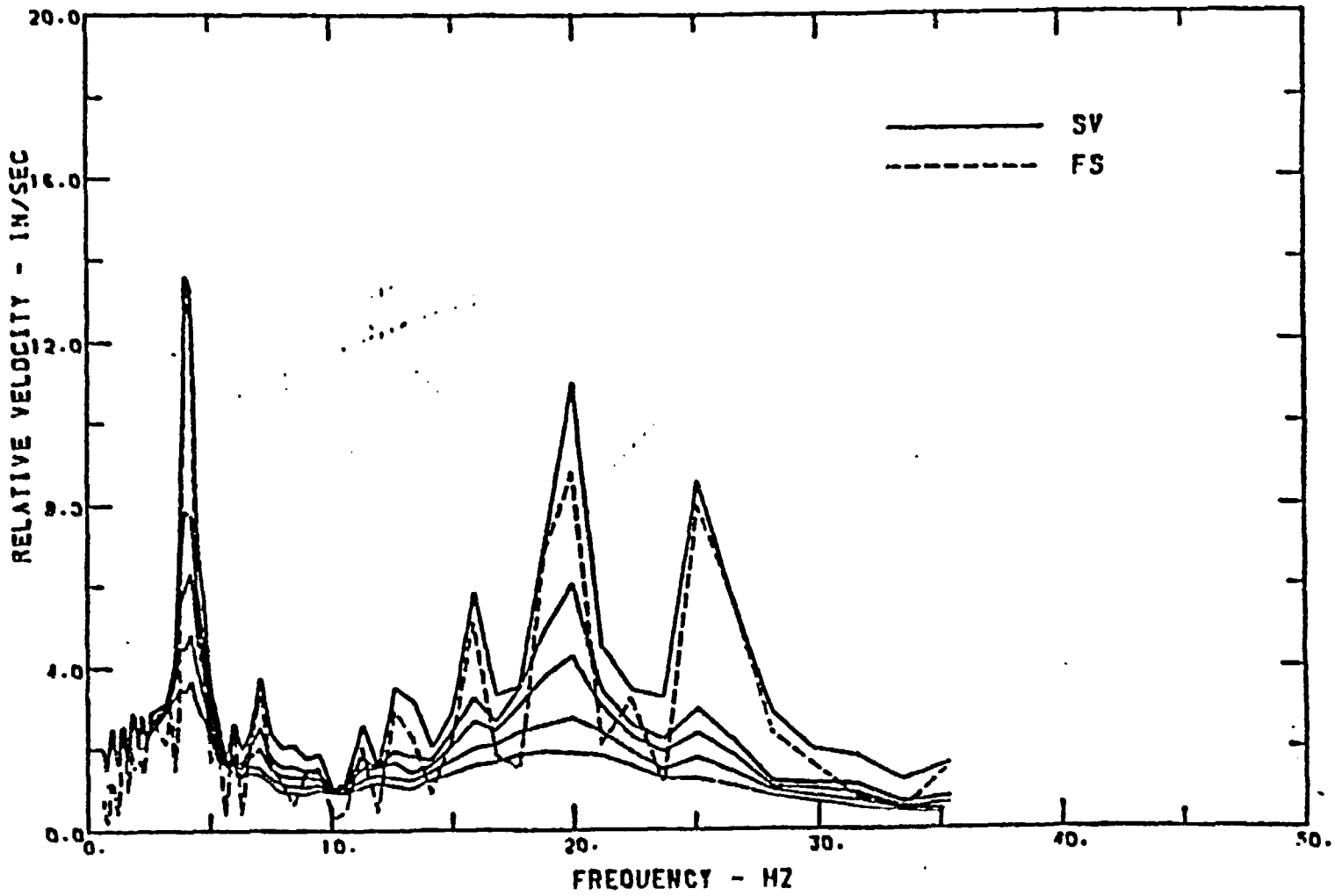
KINEMETRICS

RELATIVE VELOCITY RESPONSE SPECTRUM

ML 5.0 EARTHQUAKE JANUARY 31. 1986

11A8002 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-2T

DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



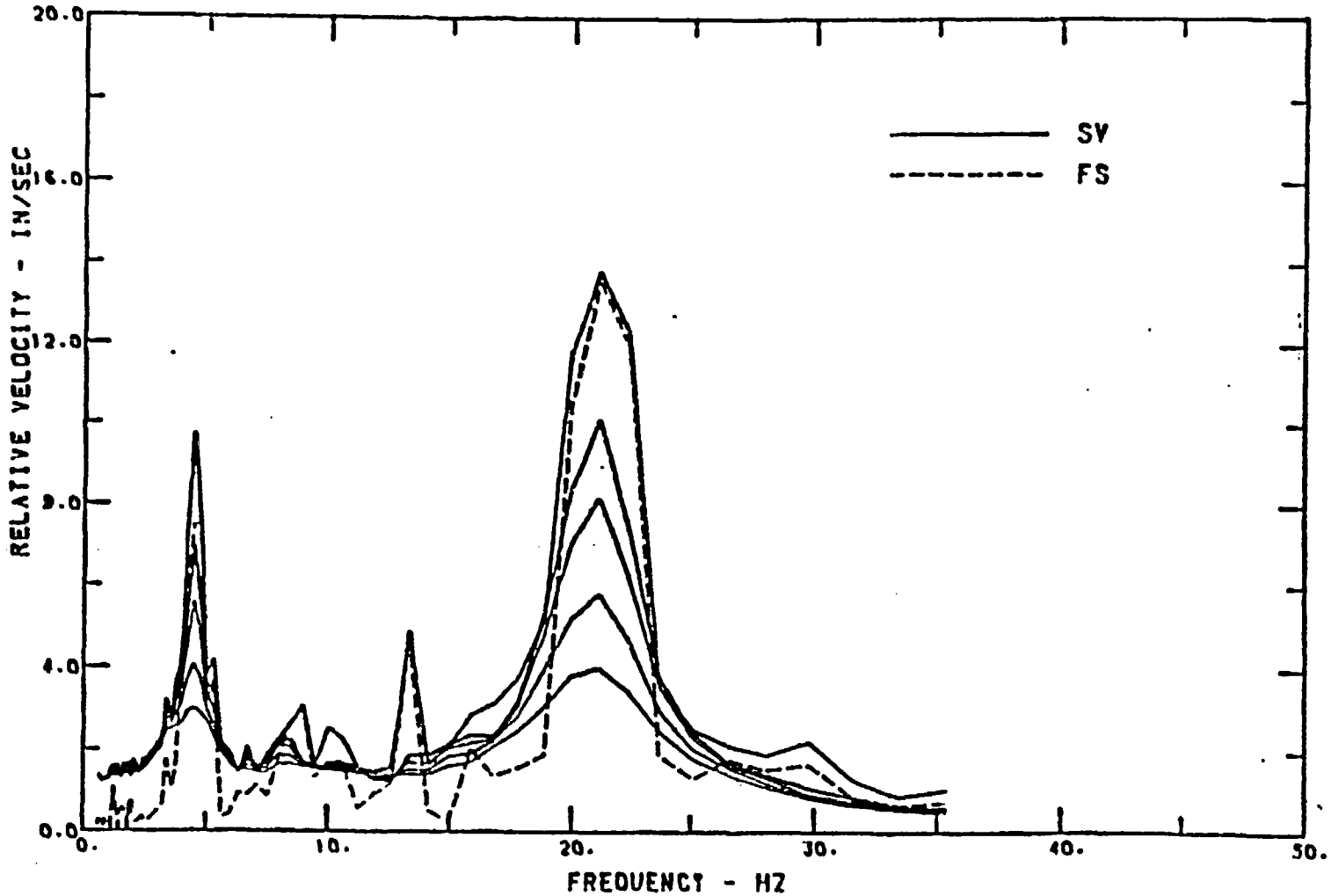
KINEMETRICS

RELATIVE VELOCITY RESPONSE SPECTRUM

ML 5.0 EARTHQUAKE JANUARY 31, 1986

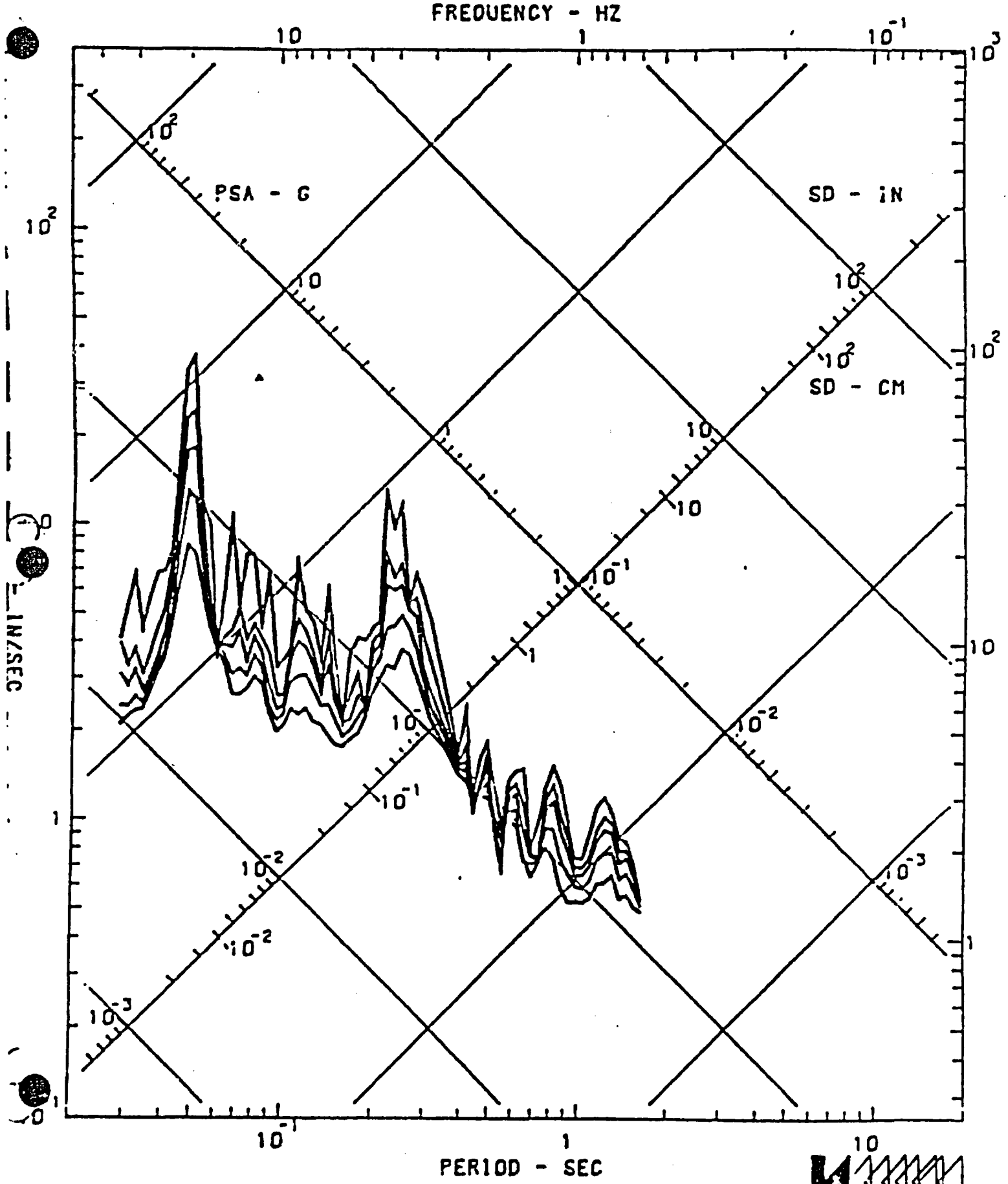
11A8002 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 165-2V

DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



11A6002 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 165-2L

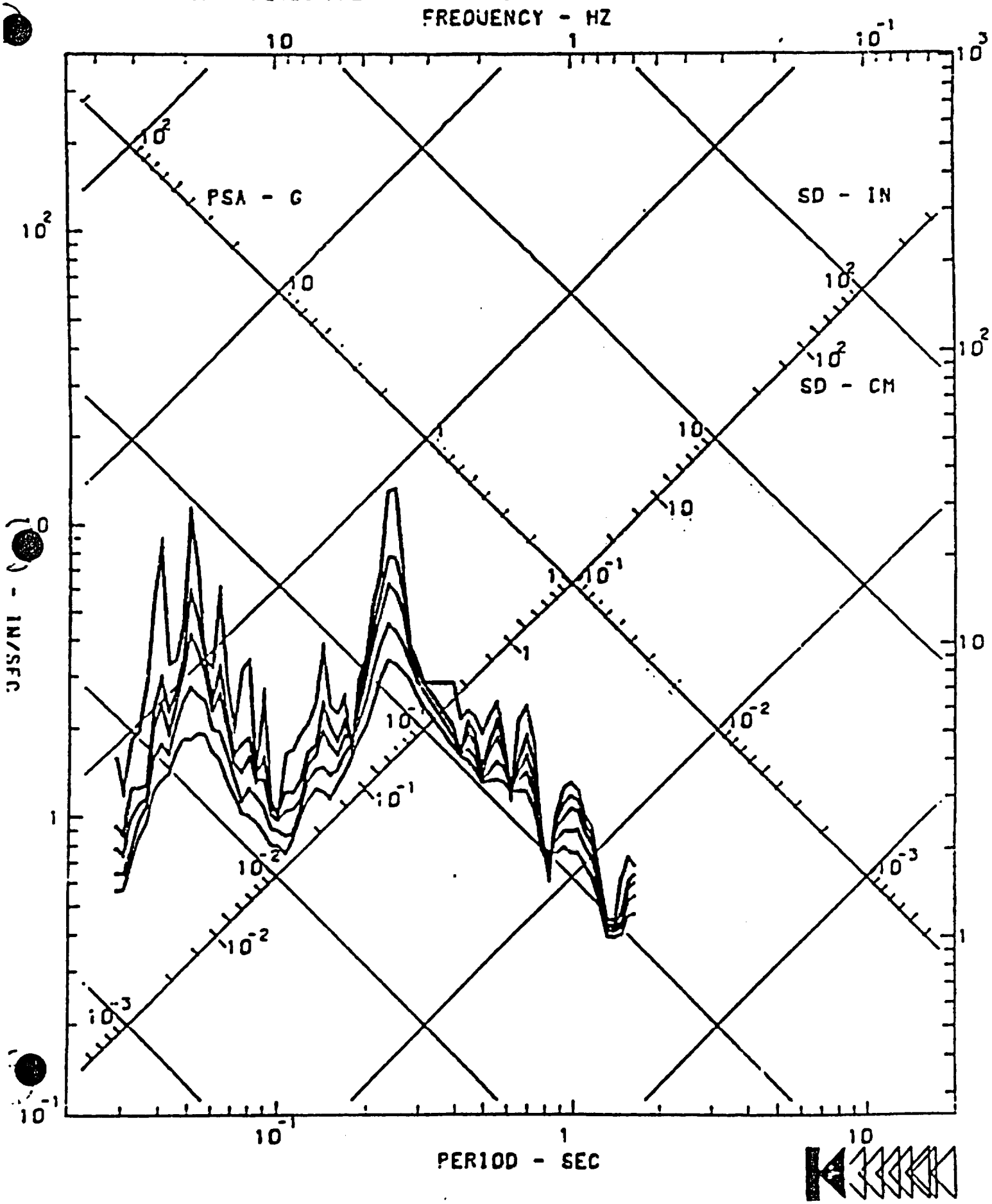
DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



HL 3.0 EARTHQUAKE JANUARY 31, 1980

11A8002 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-2T

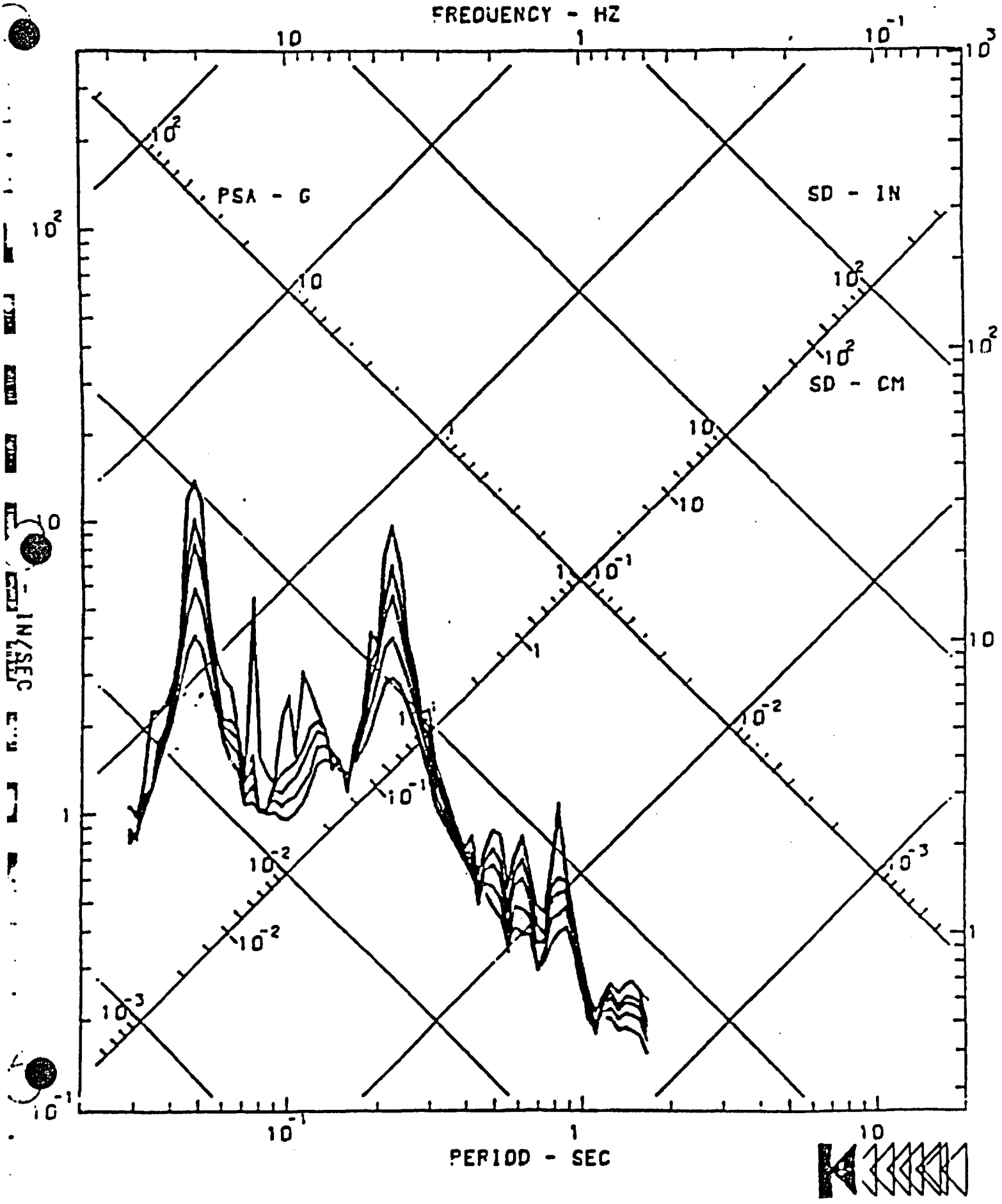
DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



THE 3.0 EARTHQUAKE JANUARY 31, 1980

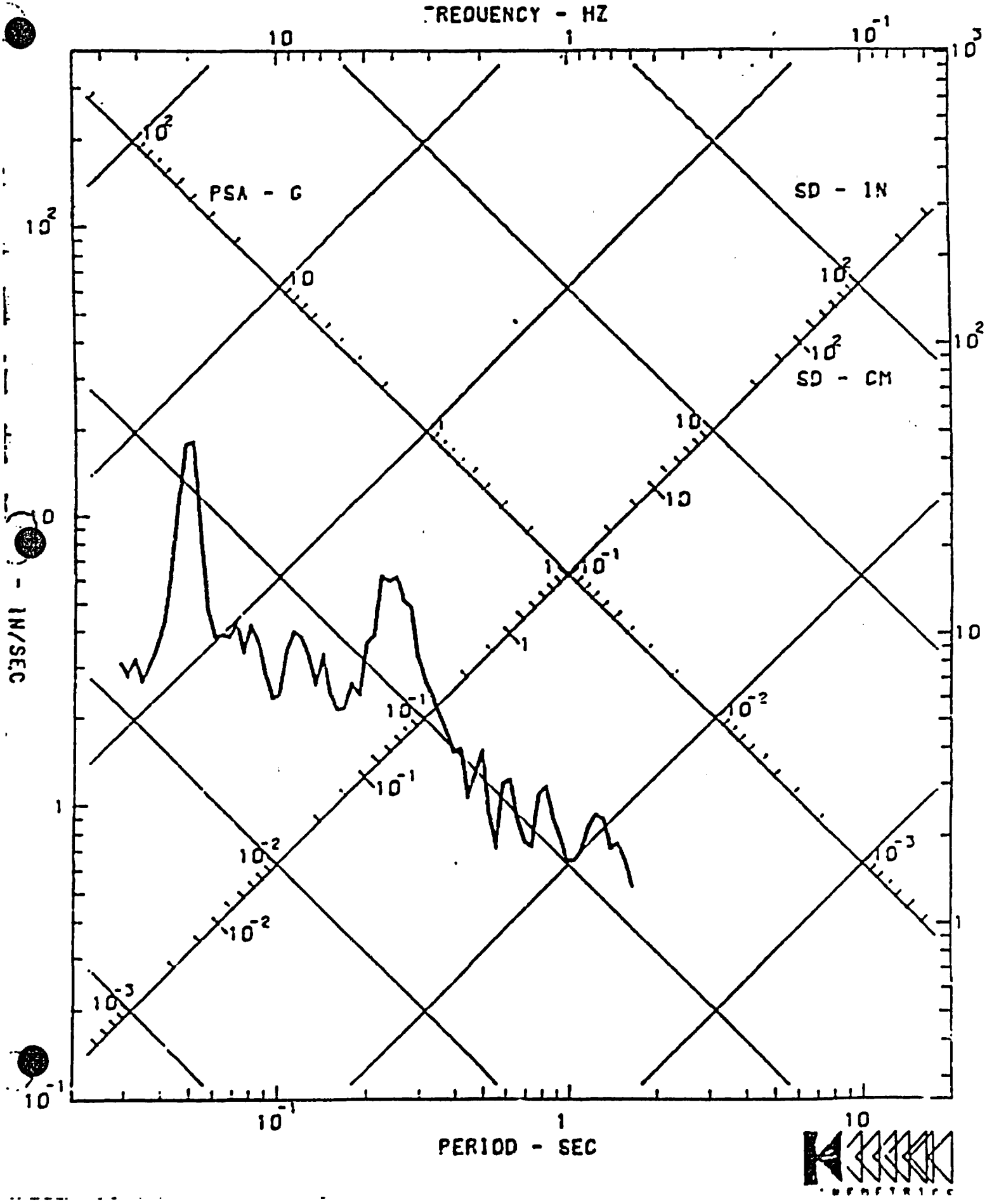
11A8002 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 16S-2V

DAMPING VALUES ARE 0. 1. 2. 4. 7 PERCENT OF CRITICAL



11A6002 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N :6S-2L

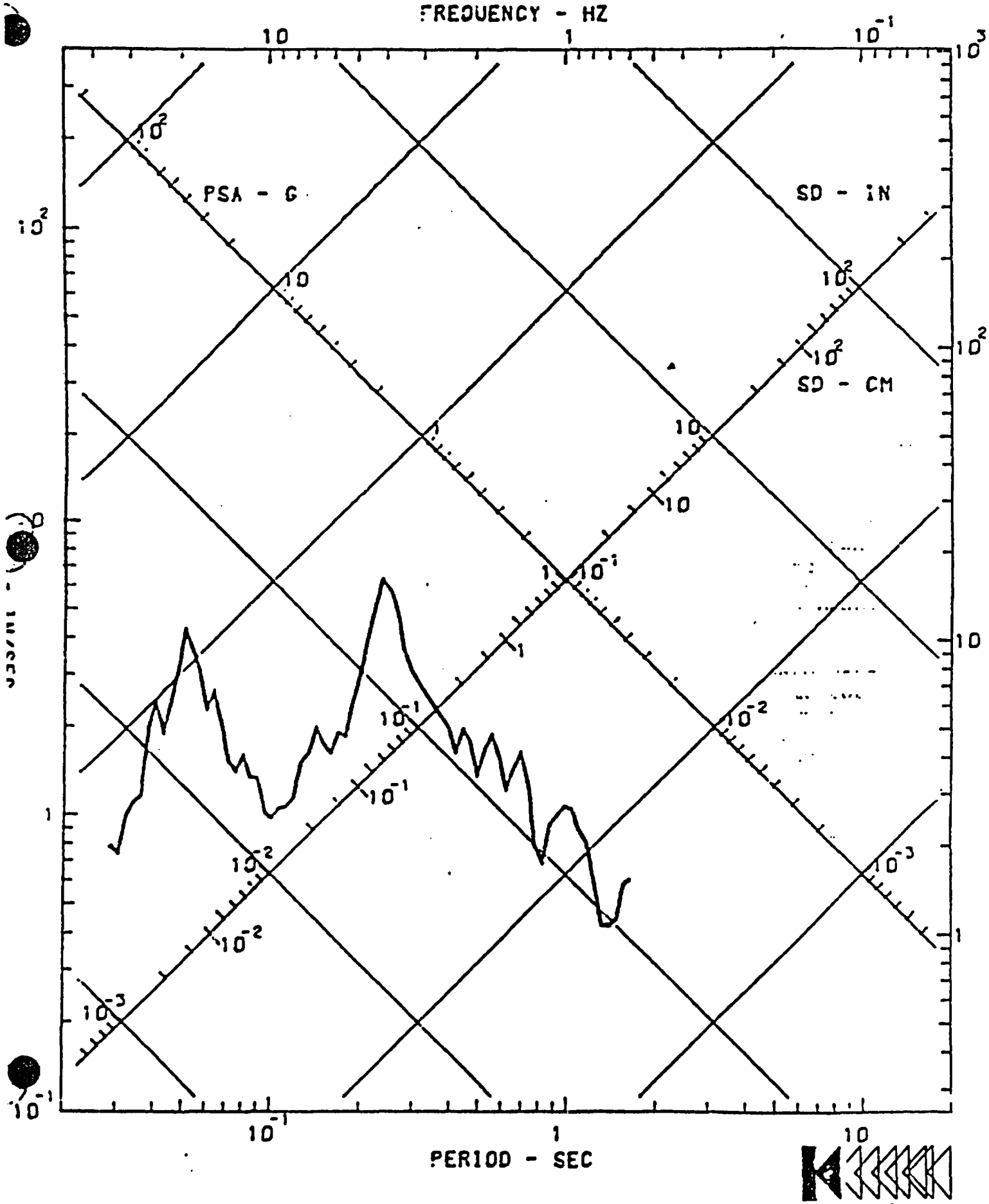
DAMPING VALUES ARE 2 PERCENT OF CRITICAL



HL 3.0 EARTHQUAKE JANUARY 31, 1986

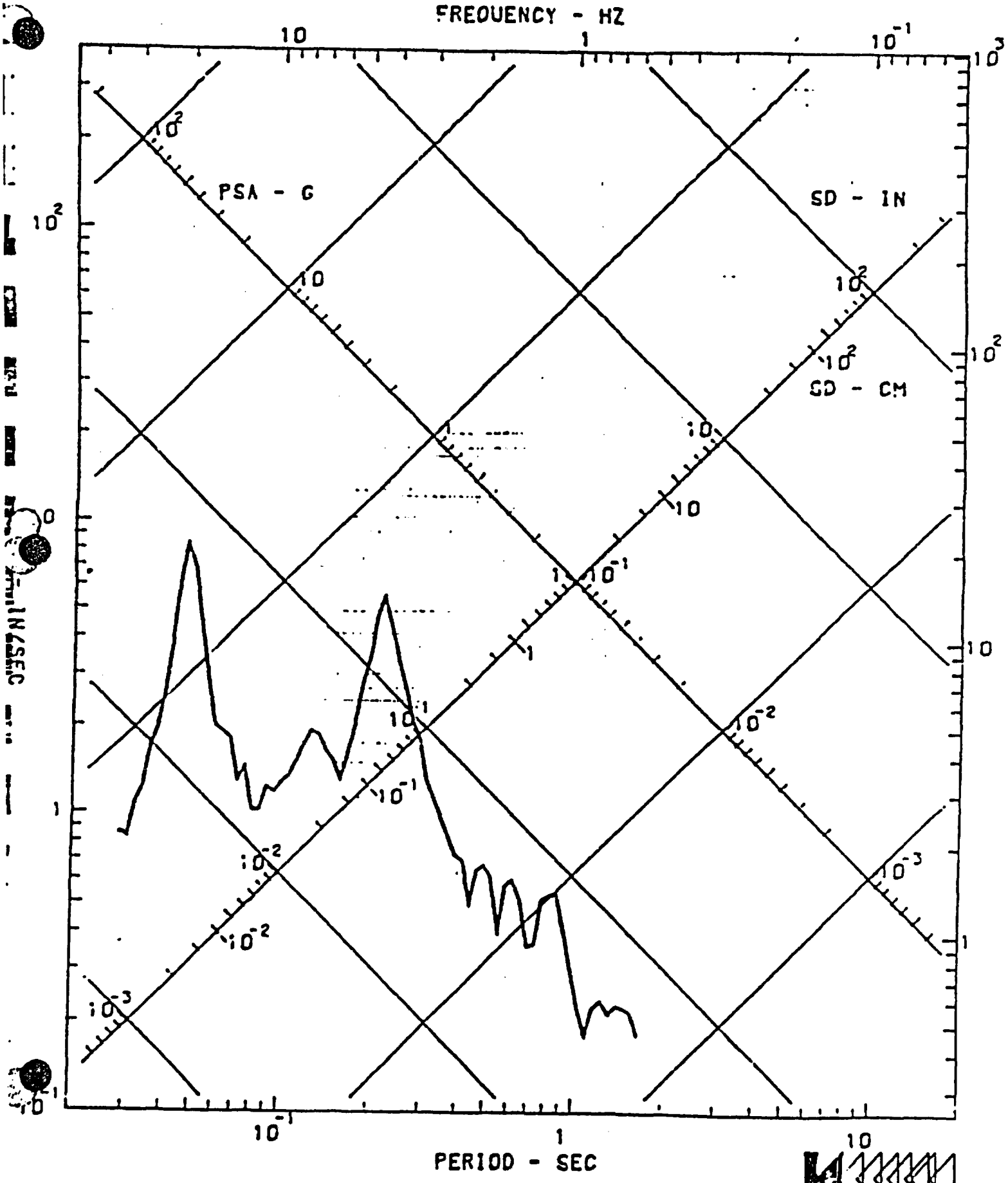
11A6002 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-2T

DAMPING VALUES ARE 2 PERCENT OF CRITICAL



11A6002 PERRY NUCLEAR POWER PLANT COMP UP SMA3S/N 16S-2V

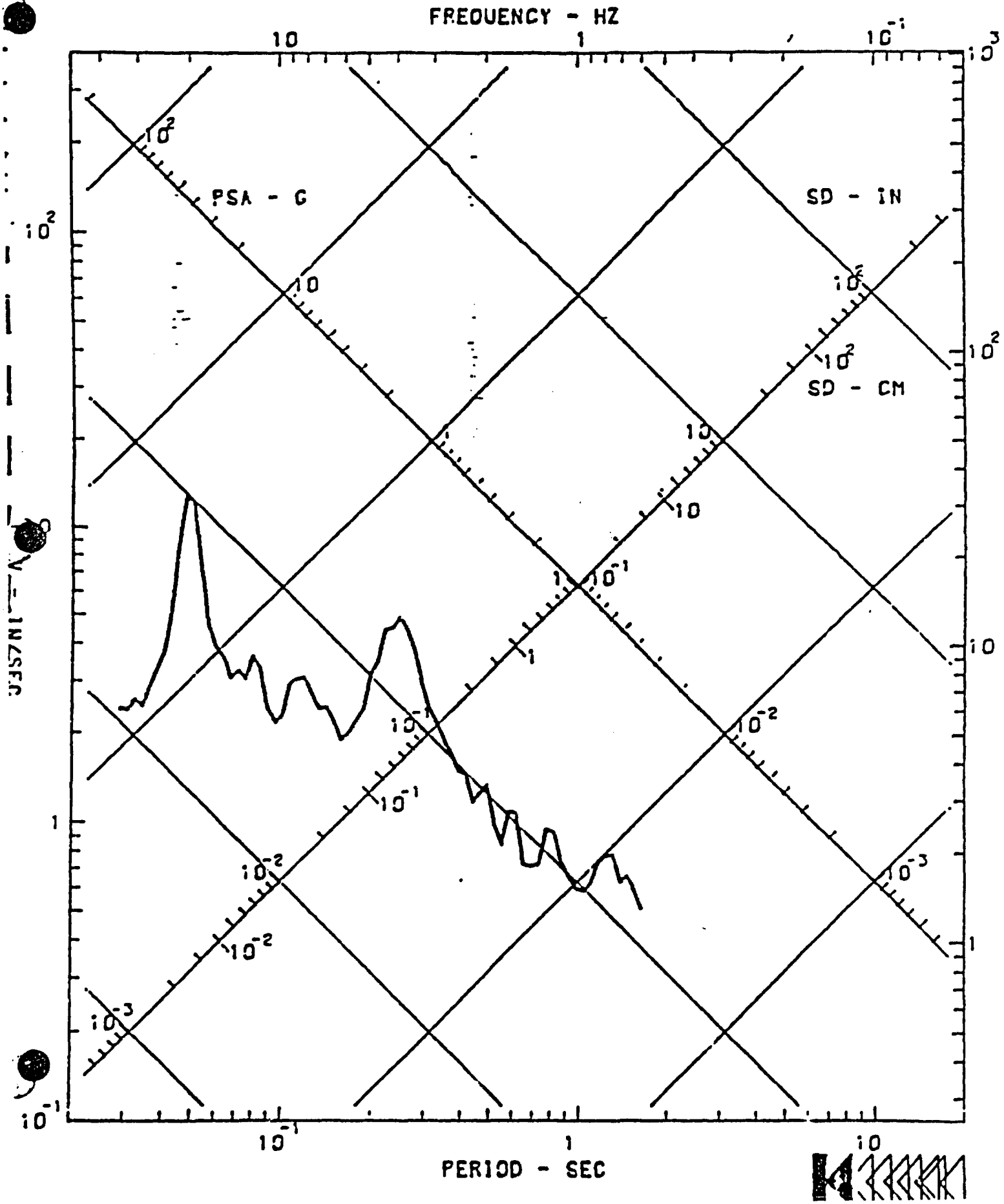
DAMPING VALUES ARE 2 PERCENT OF CRITICAL



ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A6002 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 16S-2L

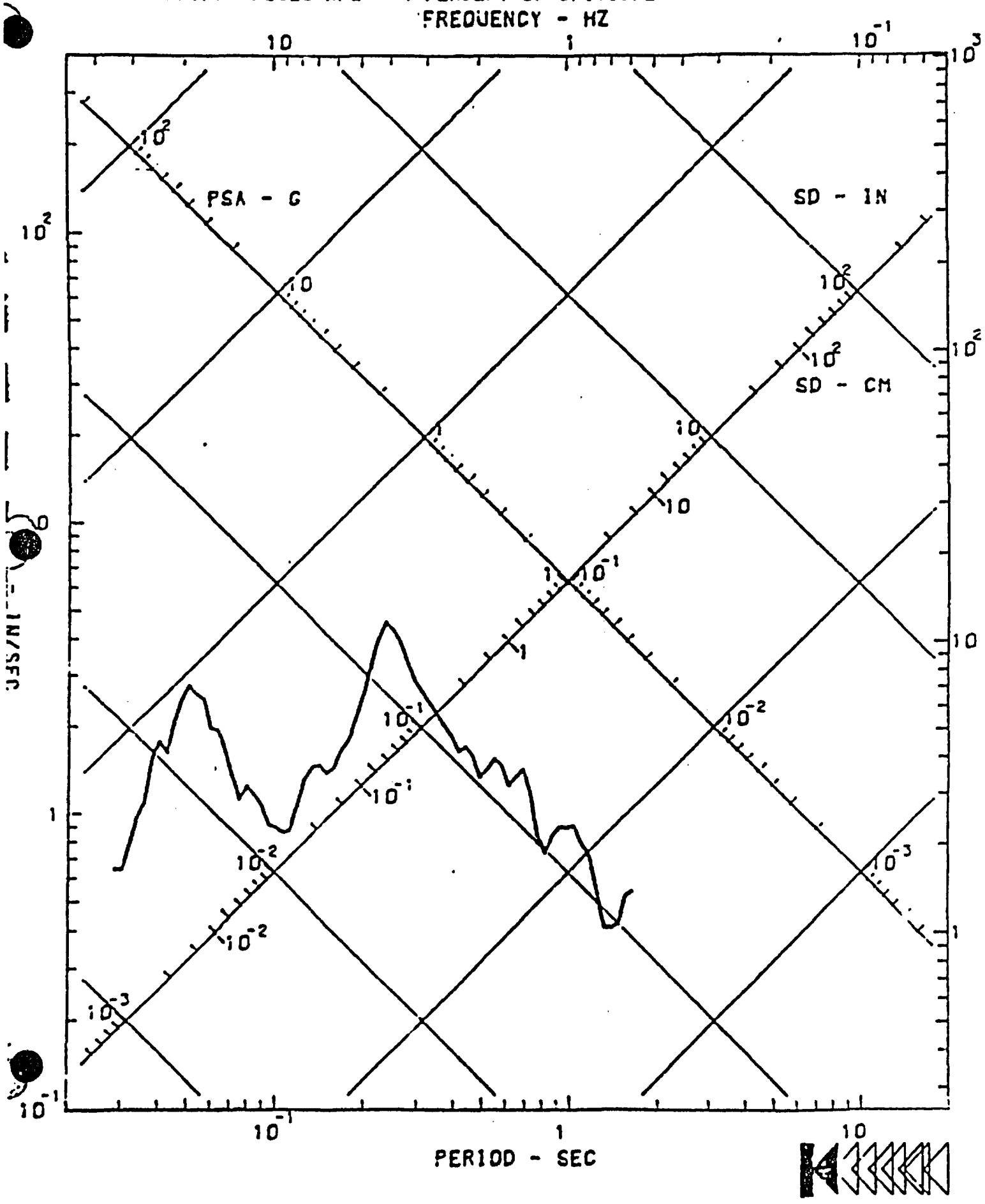
DAMPING VALUES ARE 4 PERCENT OF CRITICAL



HL 3.0 EARTHQUAKE JANUARY 31, 1980

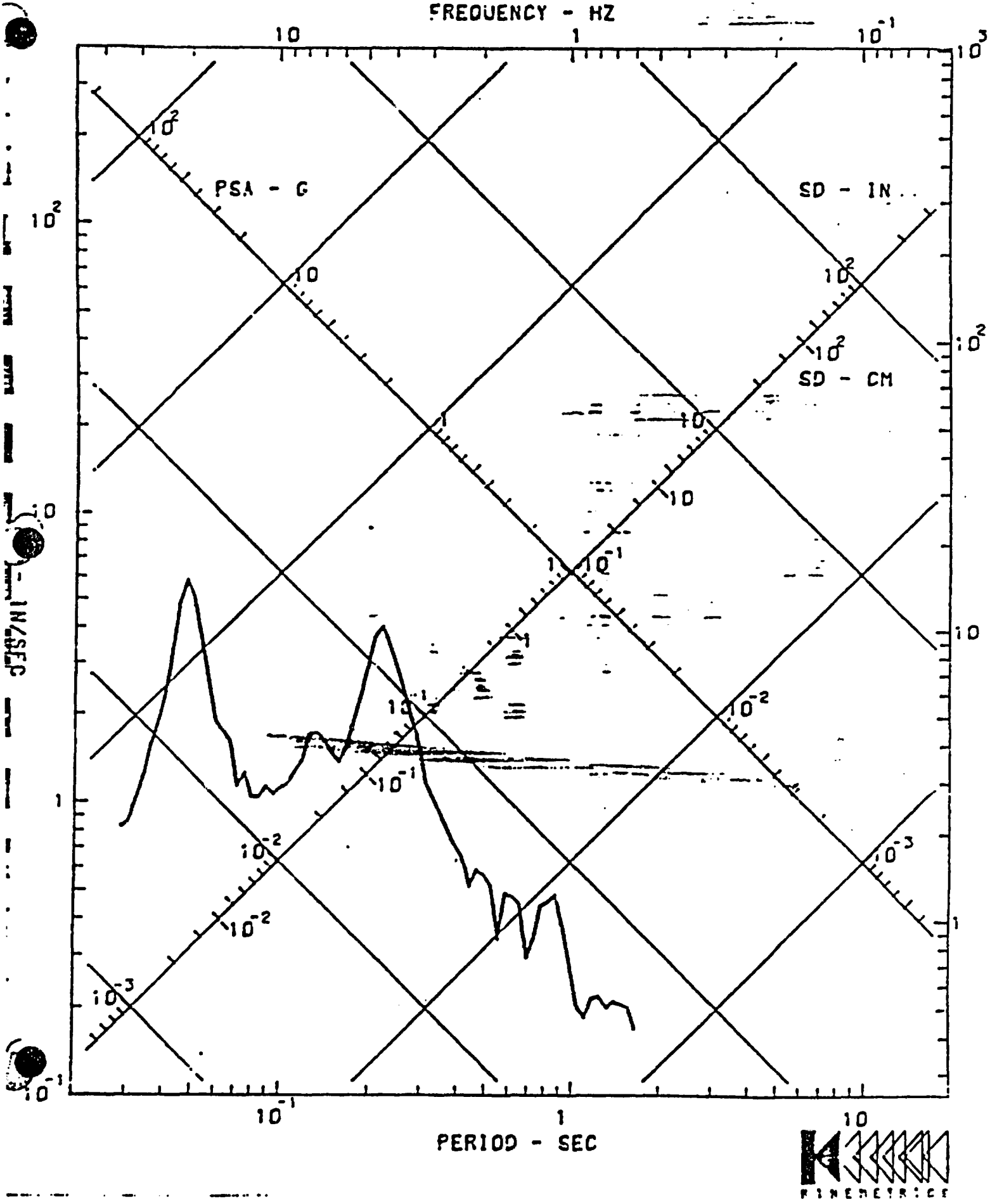
11A8002 PERRY NUCLEAR POWER PLANT COMP WEST SMA3S/N 165-27

DAMPING VALUES ARE 4 PERCENT OF CRITICAL



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

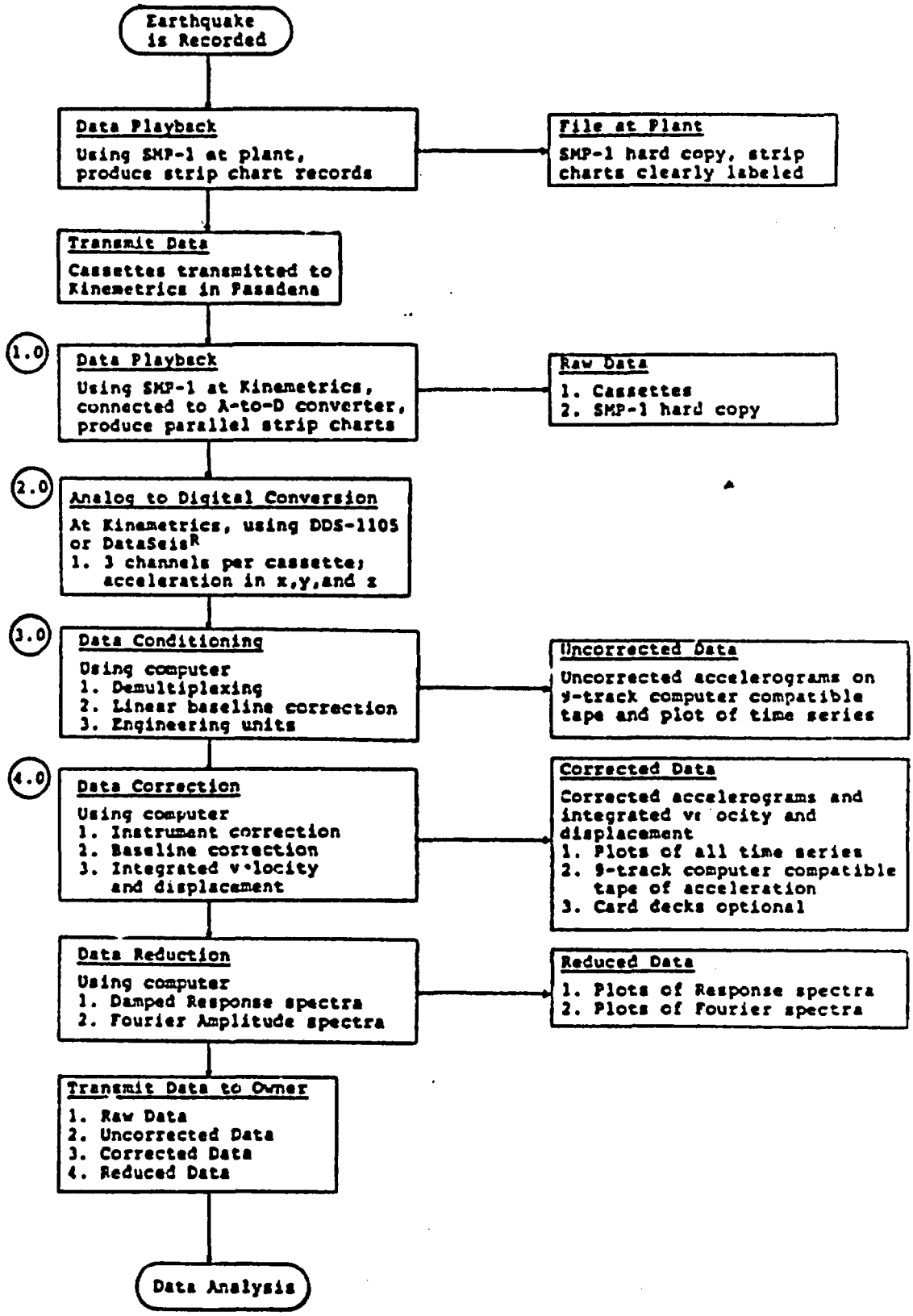
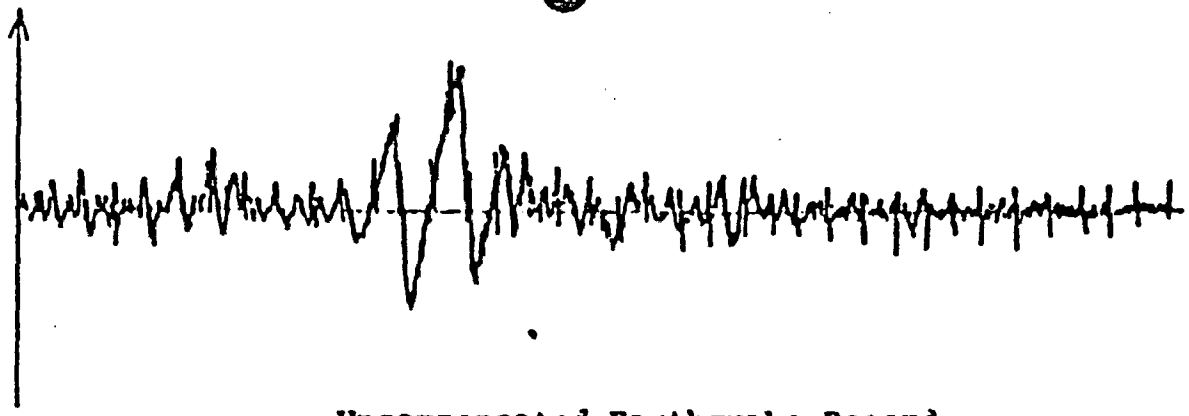


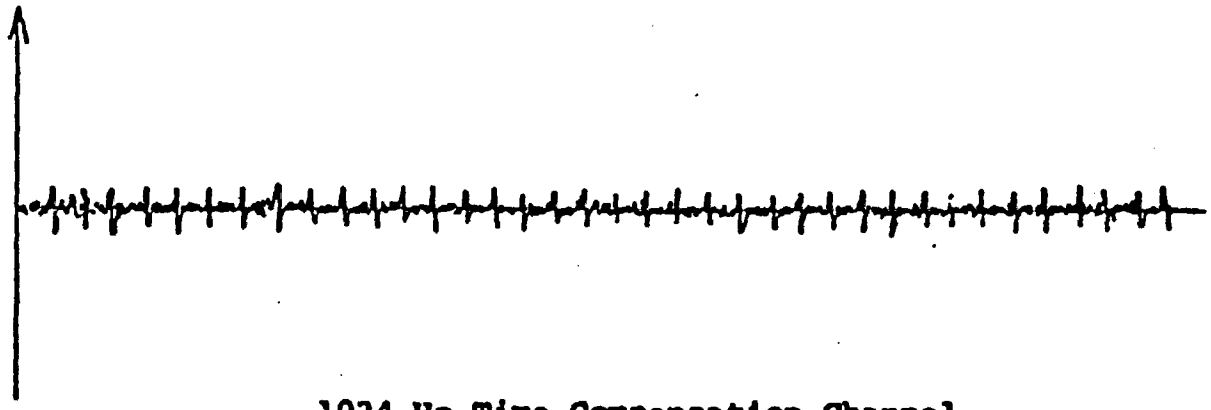
FIGURE 1 Flow Diagram for Kinemetrics E.D.R.S. (Earthquake Data Reduction Sequence)

Channel 1
(see Figure 4)



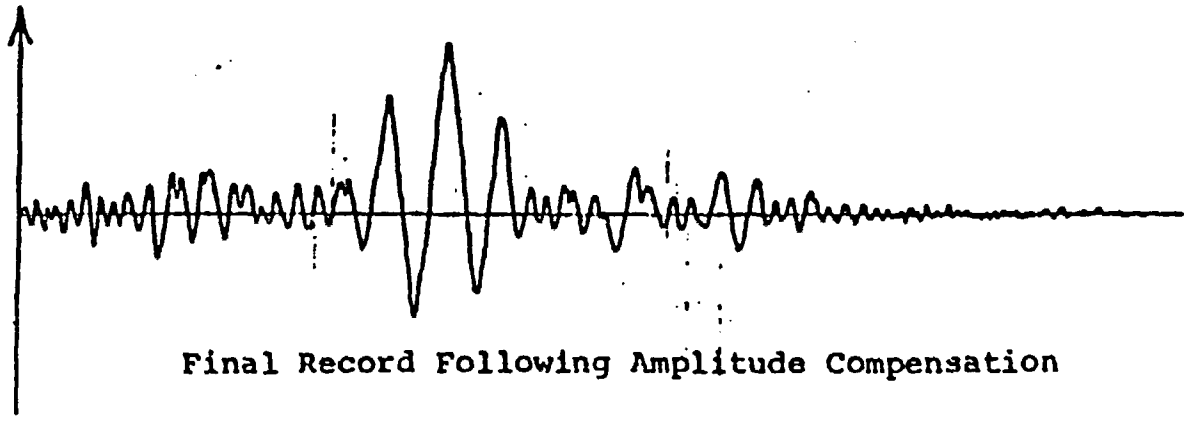
Uncompensated Earthquake Record

Channel 4
(see Figure 4)



1024 Hz Time Compensation Channel

Channel 4
subtracted from
Channel 1



Final Record Following Amplitude Compensation

FIGURE 2 Amplitude Compensation

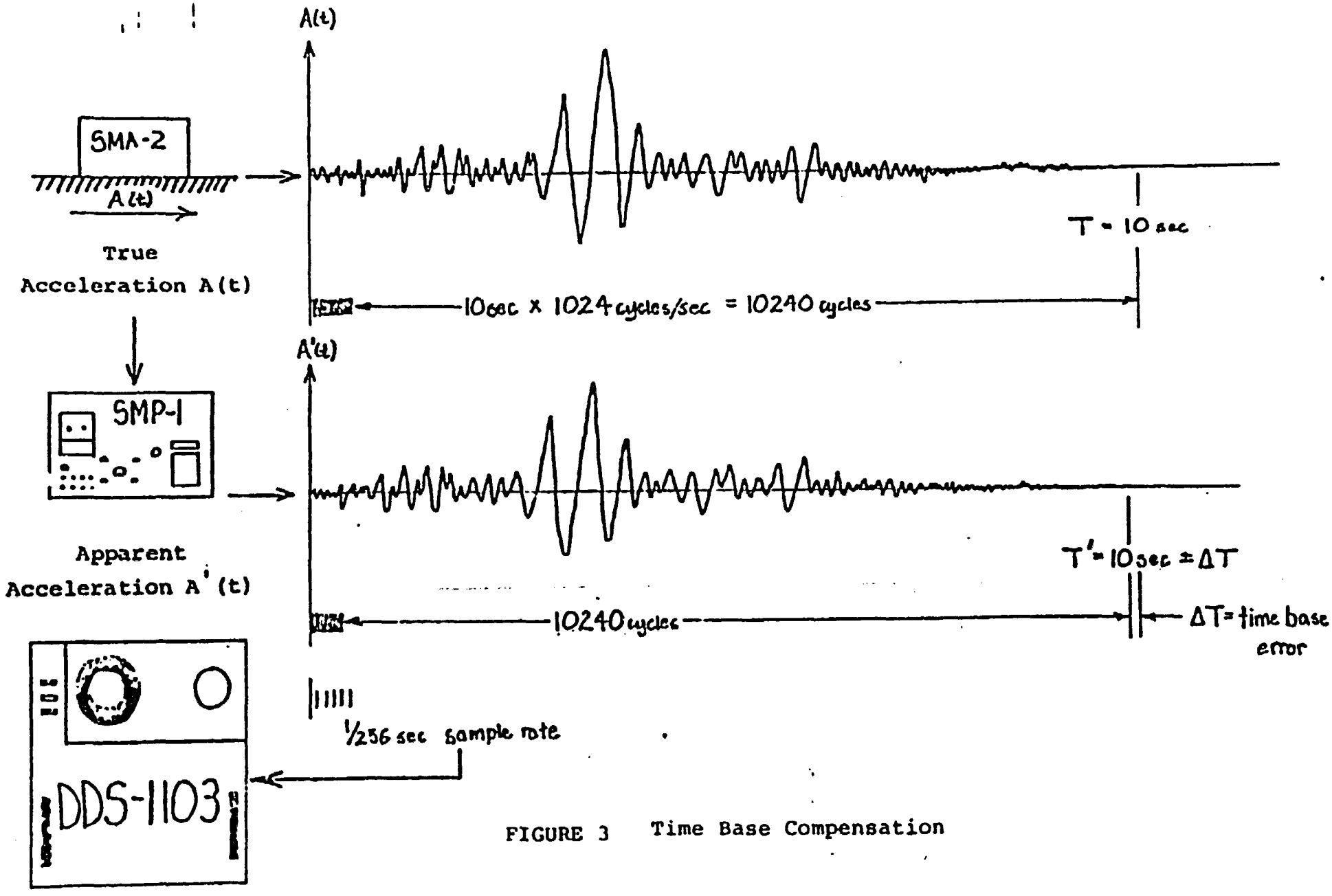
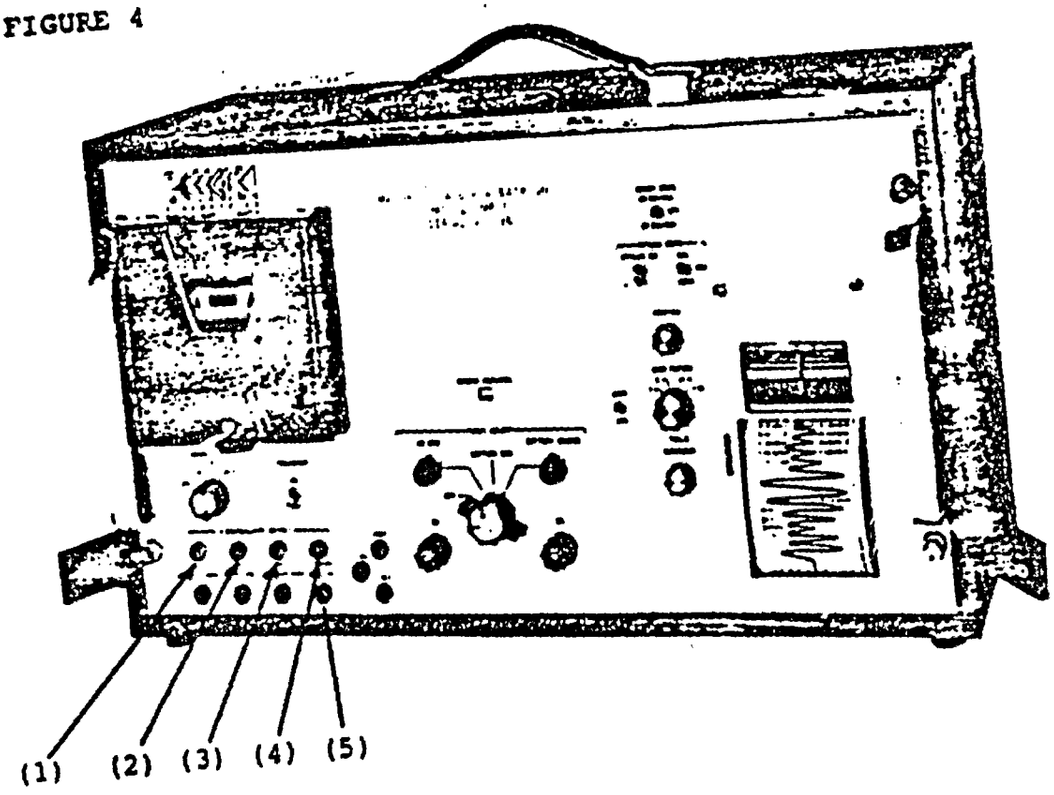


FIGURE 3 Time Base Compensation

SMP-1

Magnetic Tape Playback System

FIGURE 4



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.

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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.

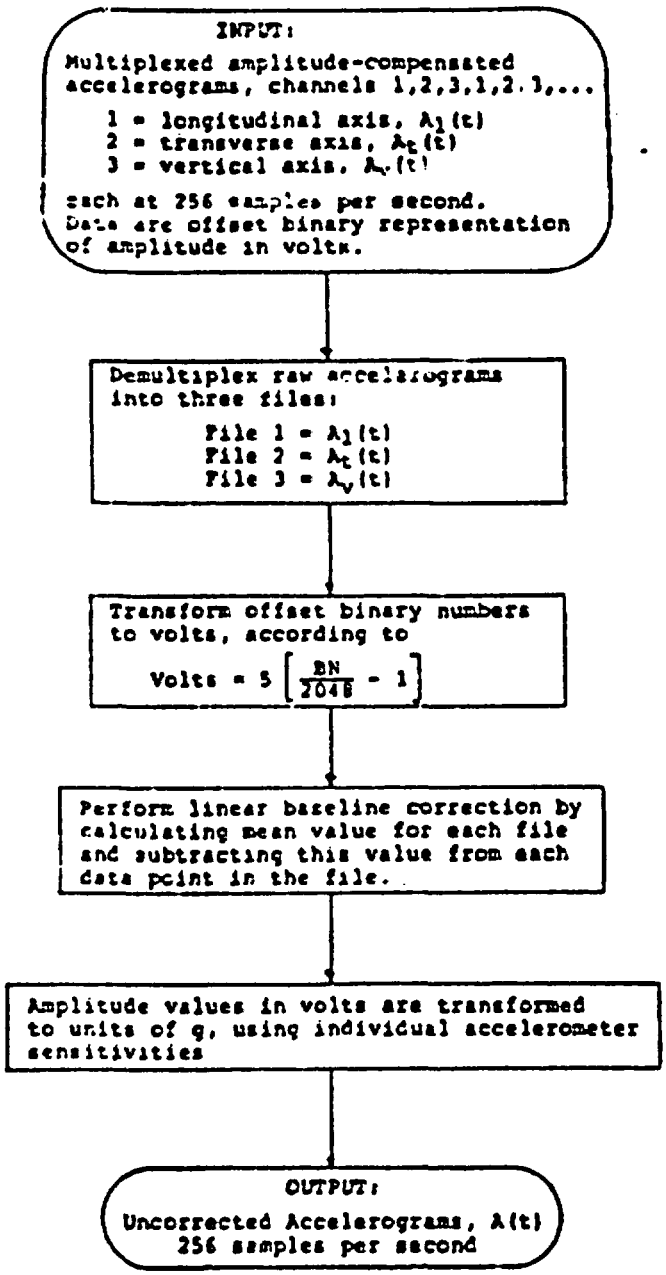


FIGURE 5

Data Conditioning, E.D.R.S.

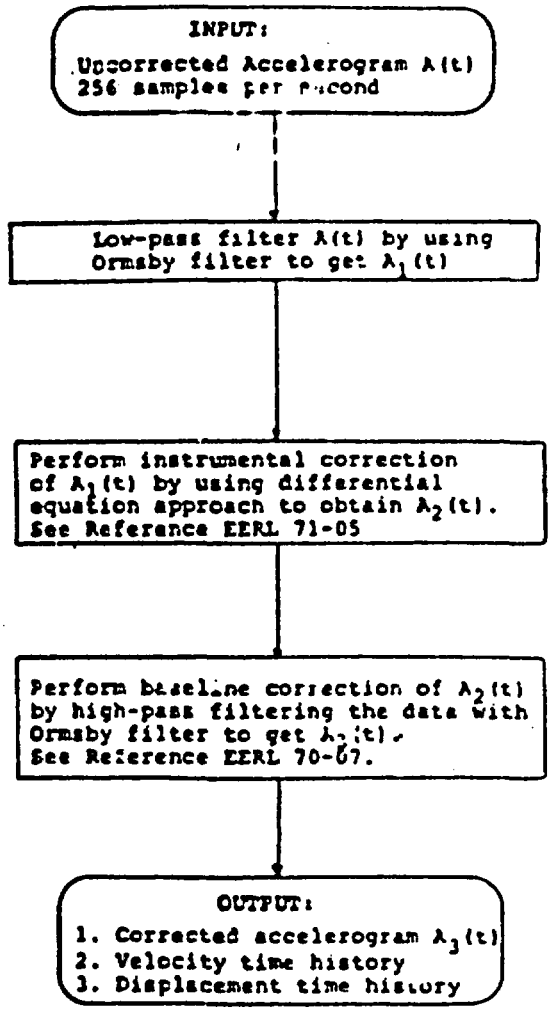
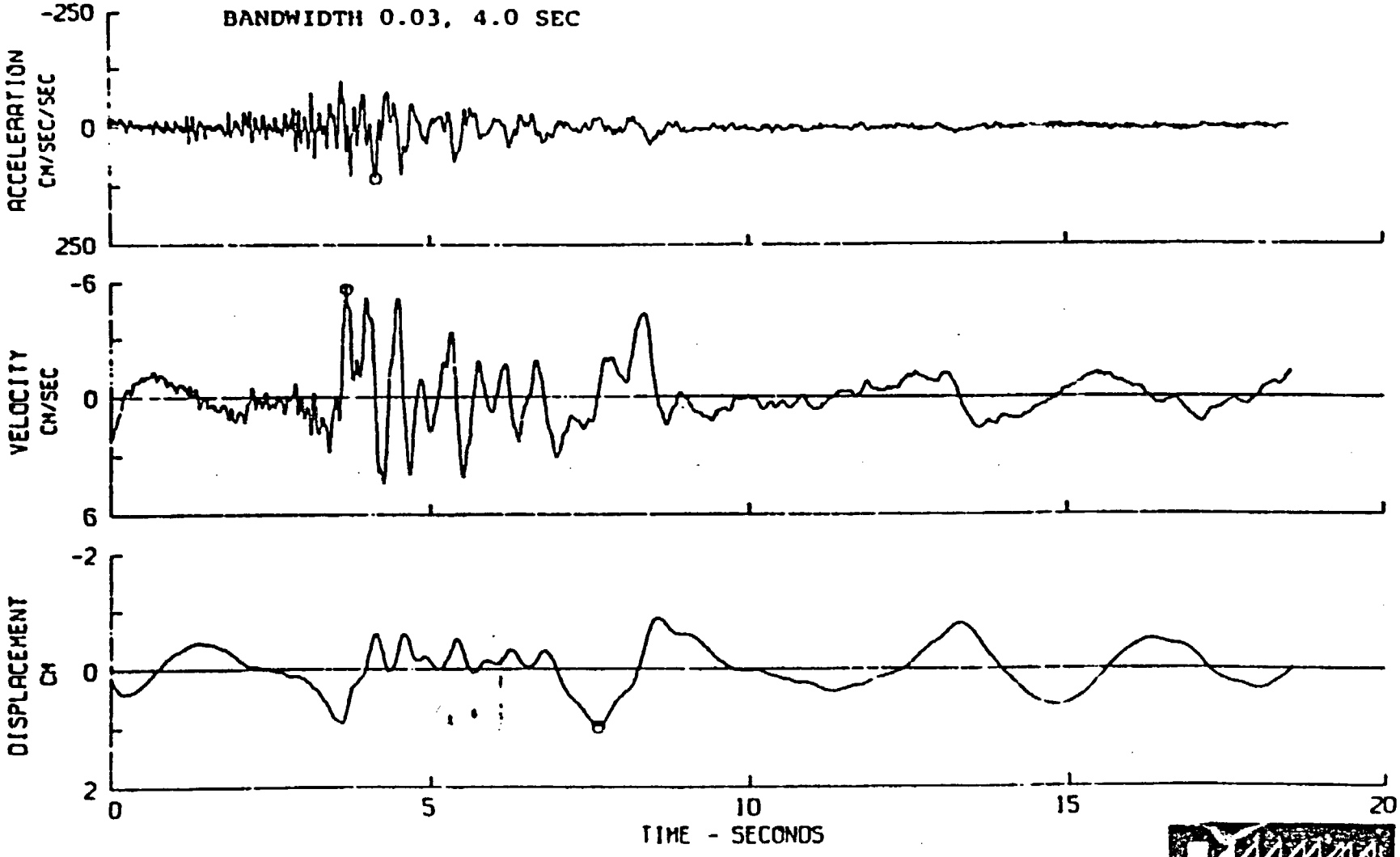


FIGURE 6

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

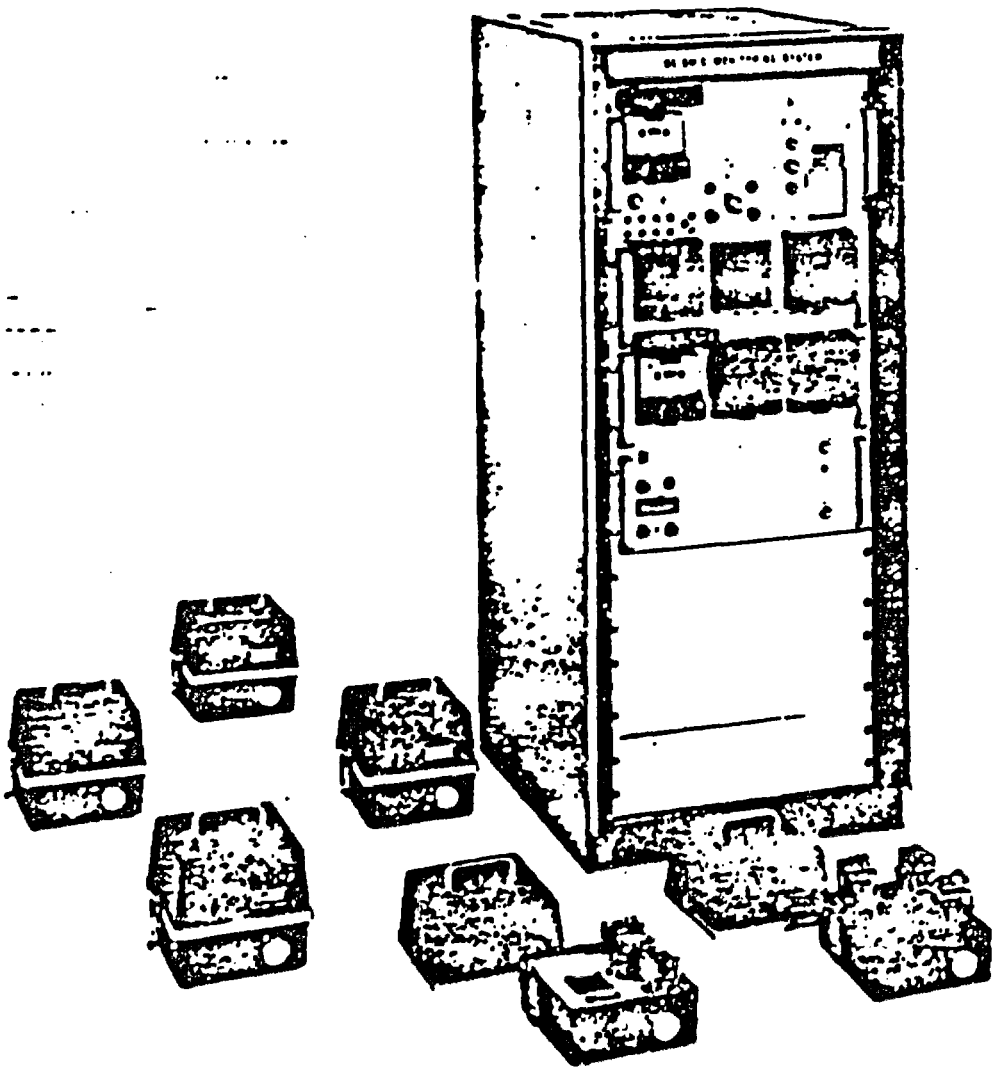
Trifunac, M. D., F. E. Udwadia and A. G. Brady (1971). High Frequency Errors and Instrument Corrections of Strong-Motion Accelerograms, Earthquake Engineering Research Laboratory, EERL 71-05, pgs. 33-47, California Institute of Technology, Pasadena

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

Ed. II--September 1984

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.

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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

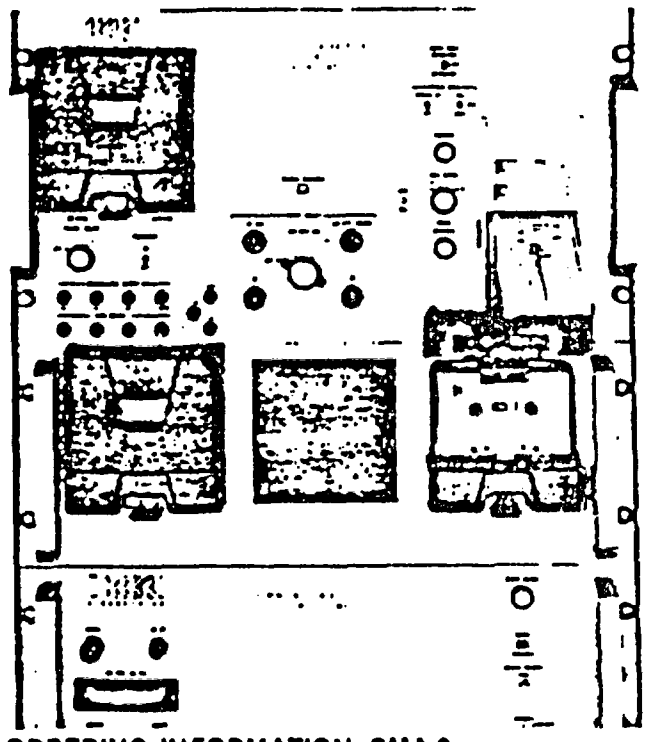
Type: Frequency modulation
Tape: Four track magnetic tape cassette
Tape Speed: 1-7/8" per second
Recording Time: 30 minutes
Bandwidth: 0 to 50 Hz
Dynamic Range: 40 dB from 15° to 35° C (with SMP-1)
Modulation Frequency: 1000 Hz ± 50% modulation
Timing Frequency: 1024 Hz ± 0.2%
System Accuracy (with SMP-1): ± 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 ±10 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

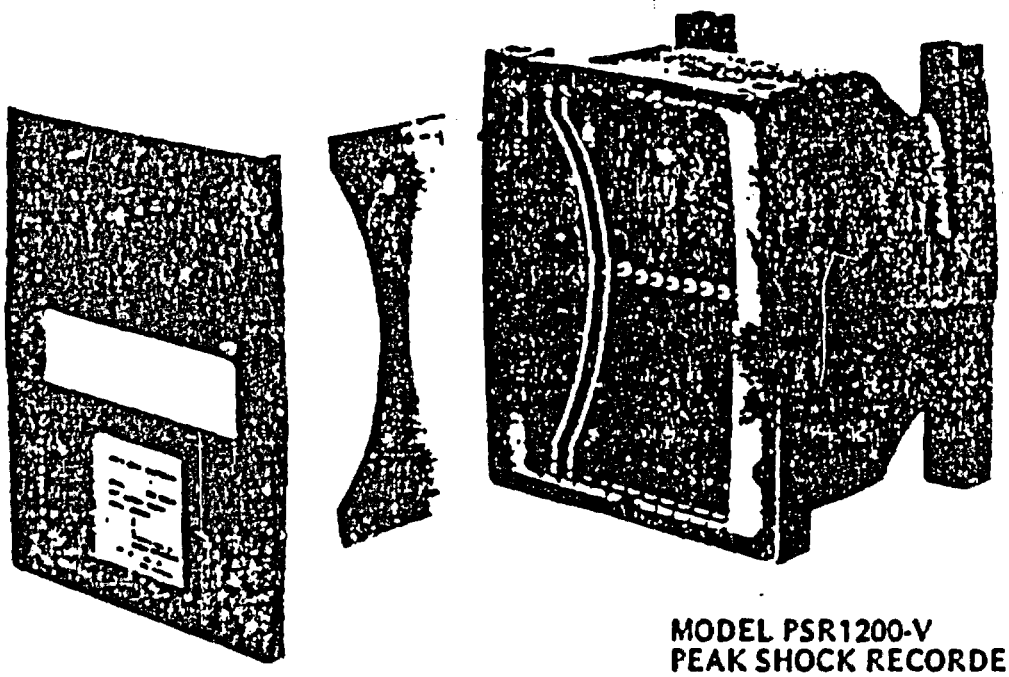
(1) Mathematical error corrected. Originally reported acceleration 0.198.
* Unable to read due to corrosion
** Unreadable

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 twelve 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.

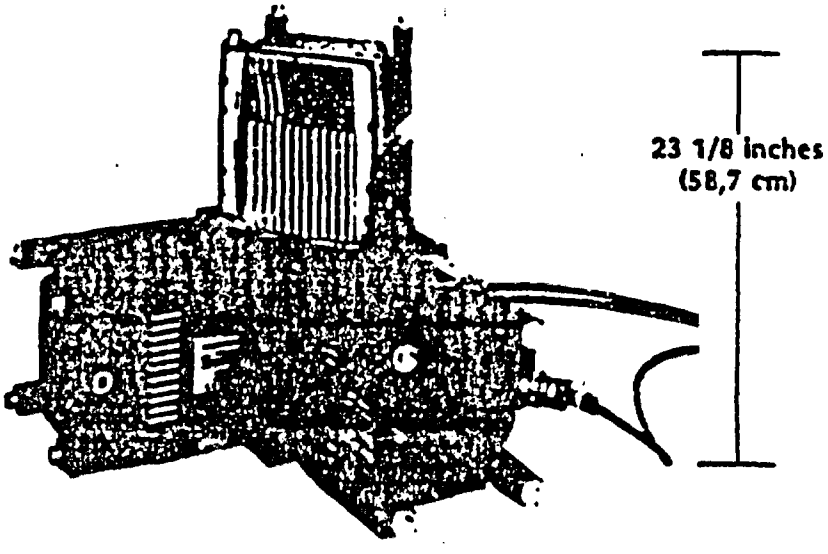
To retain the basic reliability of the PSR1200, no batteries, motors, or bearings have been added. Electrical power is provided from the Peak Shock Annunciator™.

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)

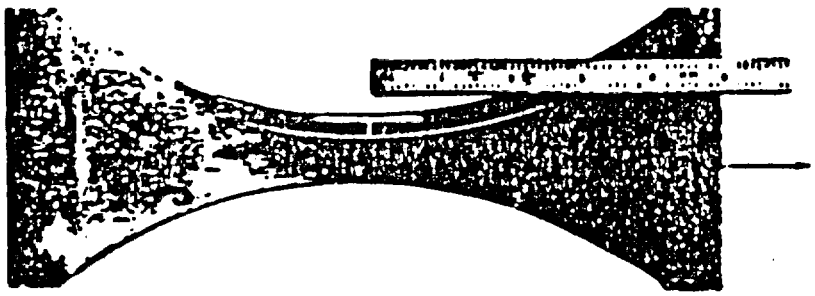
The switch settings are permanently set to positions required by the customer's application. The —4A allows four dual settings, that is, the customer selects four frequencies to be monitored between 2 and 25 Hertz. Two acceleration levels can be selected for switch contacts for each reed frequency, e.g., .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.

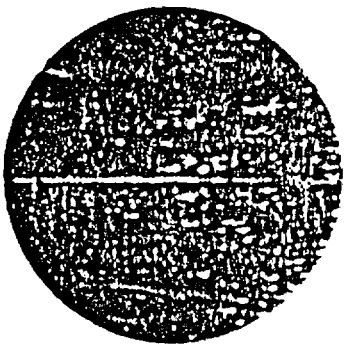
Data Reduction

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).

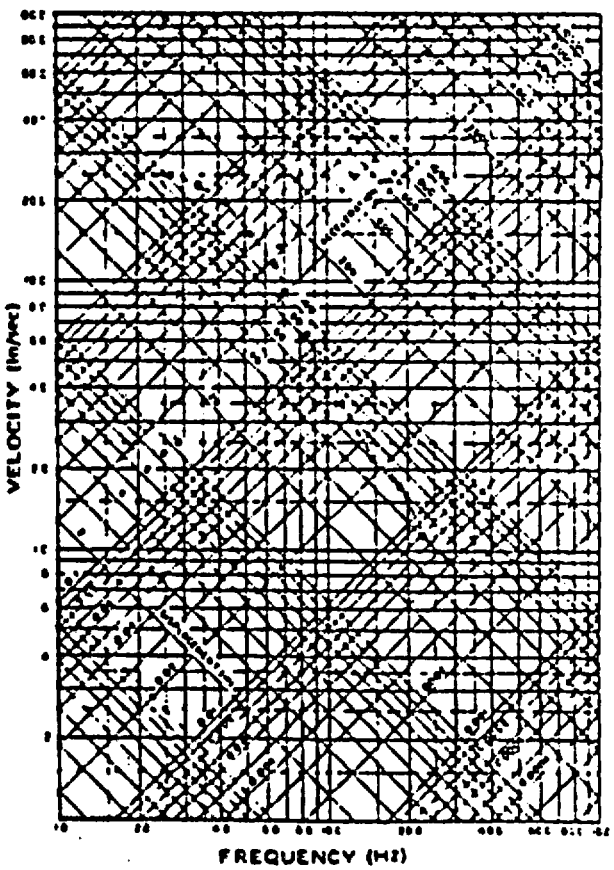




CALIBRATION SHEET AND TEST DATA

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	.56	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	.076	.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

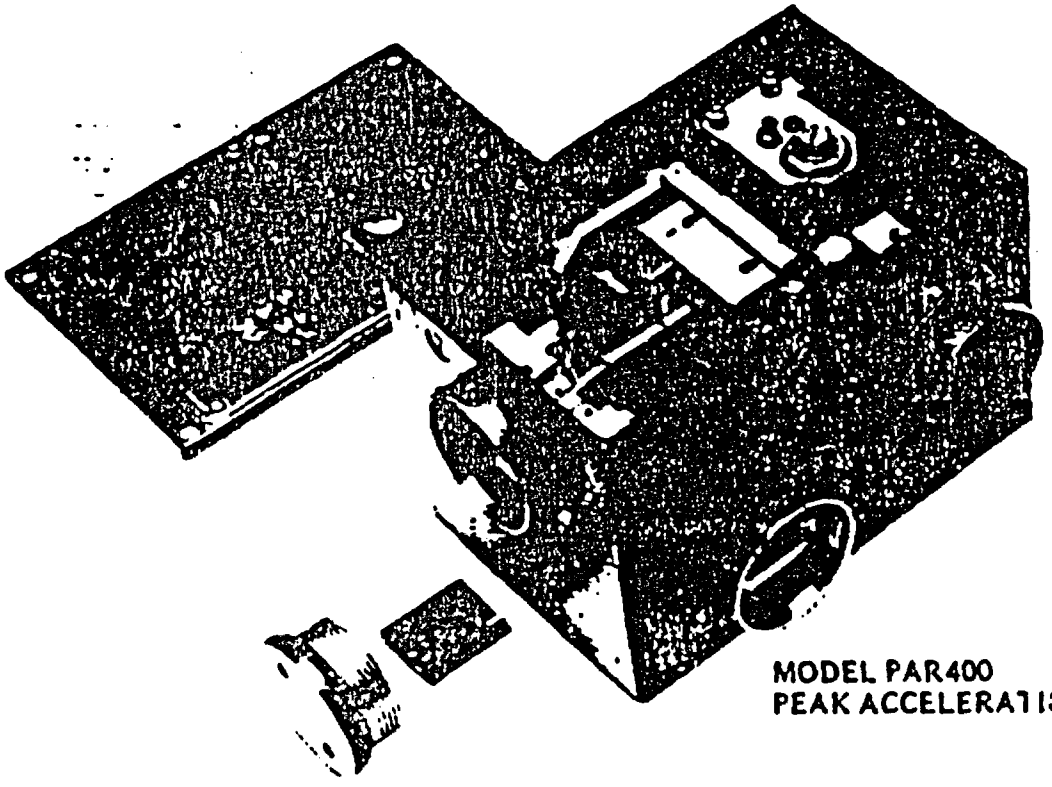
FREQUENCY, RANGE, and SWITCH SETTING LIMITS OF SENSING ELEMENTS

Reed Number	Nominal* Resonant Frequency (Hertz)	Full Scale Acceleration (g)	Dynamic Range		Switch Setting Limits** (g)	
			(db)	Ratio	Minimum*** (Accuracy ±100%)	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	.0[illegible]	4
12	25.4	6	49.5	300:1	.0[illegible]	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 wherever 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.

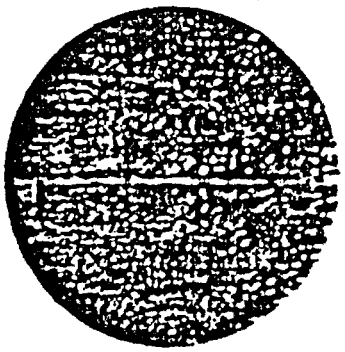
Data reduction is very simple requiring only one measurement and one multiplication for each of the three record plates to obtain its maximum acceleration.

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.

MAGNIFIED RECORD
USING A RETICLE



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.

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	

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

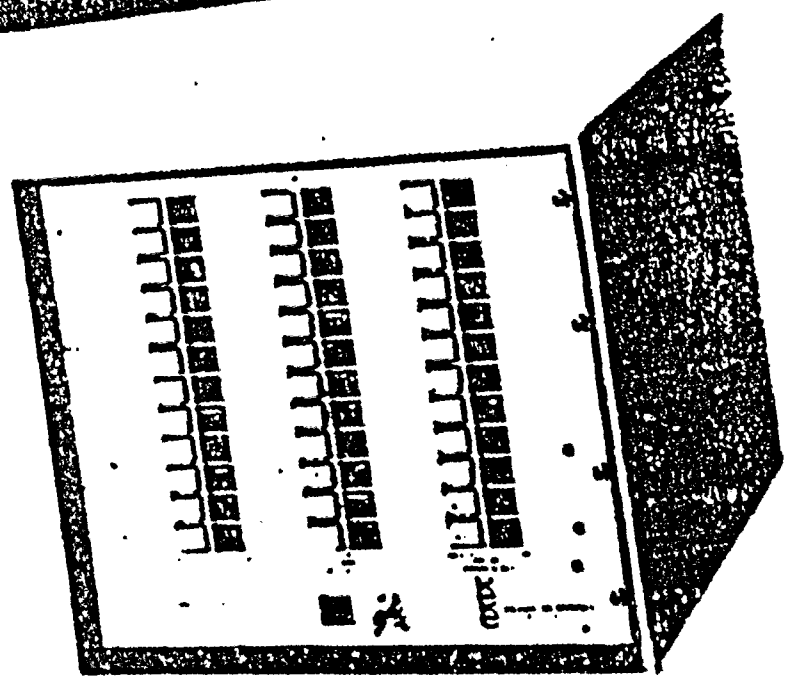
Description	Material	Stress Level	Approx. Stability[2] (RAD)
Paint	Epoxy	Low	1×10^8
Adhesive	Epoxy	Low	1×10^8
Adhesive	Anaerobic	Low	2×10^8
Adhesive	Cyanocrylate	Low	2×10^8
Gaskets	EPDM	Low	1×10^8
Piston	Graphite	Low	2×10^8
Cylinder	Pyrex	Low	2×10^8
Scriber	Diamond	Low	2×10^8

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:

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

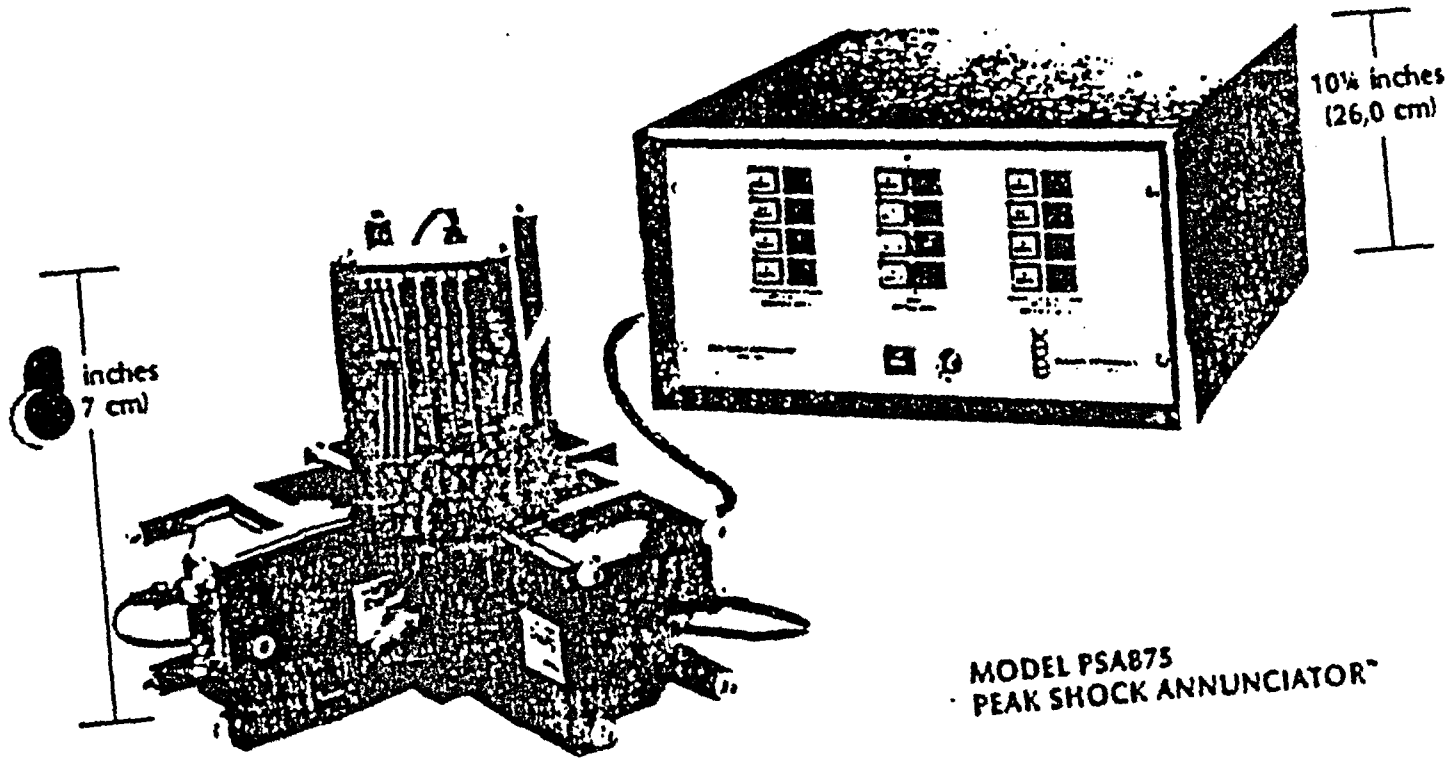
The "AC Power" indicator lamp is fed from the DC power on the printed circuit boards and shows that the incoming power line and the regulated DC supply are both operating normally.

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:

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:

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9217 Midwest Avenue
Cleveland, Ohio 44125
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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. Reactor Building Foundation: Elevation 574', Reactor Building foundation mat, Azimuth 210°. This recorder was most recently calibrated on January 14, 1985.

2. Reactor Building Drywell Platform: 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. HPCS Pump Base Mat: 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. RCIC Pump Base Mat: 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

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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:

$$K_p = \sqrt{2 \ln (2T_D/T_o)} \geq 2.0$$

T_D = 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.

To = Predominant period of motion (sec.)

rms = $\sqrt{P}$

P = $E(T)/T_D$ = earthquake power (average rate of energy input)

$E_T = \int_{t_o}^{t_o + T_D} a^2(t)dt$ = total energy fed into the structure between times to and to + 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 these 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 (i.e., 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	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	.045	.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	.134	.091	.165	.158	1.7	1.54

(*) Unreadable

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
NS	Recorded	.17	.18	.32	.55	.09
	SSE	.17	.18	1.06	.40	.48
	OBE	.10	.10	.86	.24	.40
EW	Recorded	.06	.10	.11	.18	.16
	SSE	.20	.18	1.06	.40	.48
	OBE	.10	.10	.86	.24	.40
VERT	Recorded	.03	.11	.05	.30	Note 2
	SSE	.20	.18	.47	.24	.28
	OBE	.10	.10	.38	.15	.16
SRSS(3)	Recorded	.18	.23	.34	.65	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

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
Ohio - 1986	4.9	17	(*)	0.75	.434(Average --

(*) 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 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

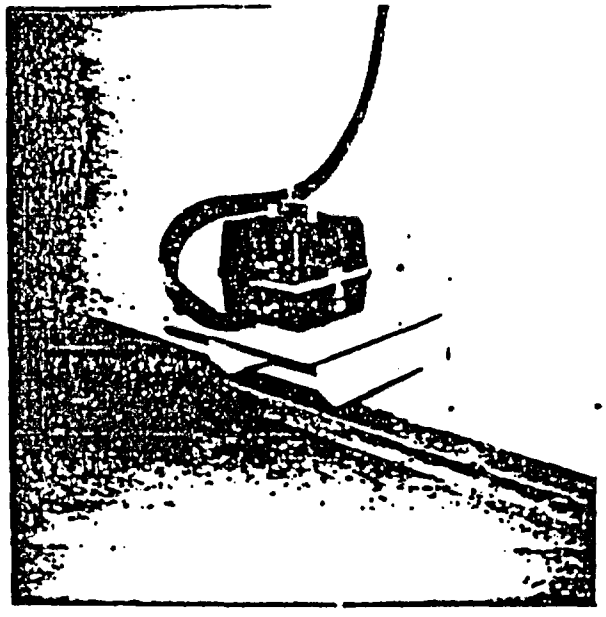

FIGURE 1 - KINEMETRICS SMA-1
ACCELERATION TIME HISTORY RECORDER
SERIAL NO. 165-1, TAG NO. [illegible]
LOCATED AT BASE OF CONTAINMENT [illegible]
ELEVATION 575.

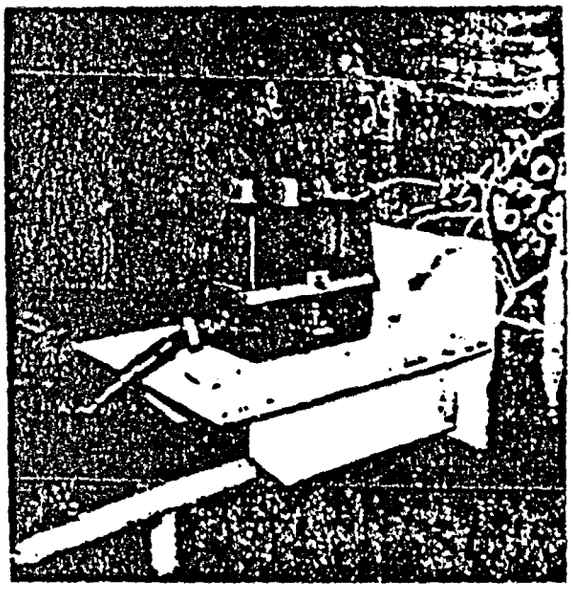

FIGURE 2 - KINEMETRICS SMA-1
ACCELERATION TIME HISTORY [illegible]
SERIAL NO. 165-2, TAG NO. [illegible]
LOCATED ON THE STEEL CONTAI[illegible]
SHELL AT NO. 6[illegible].

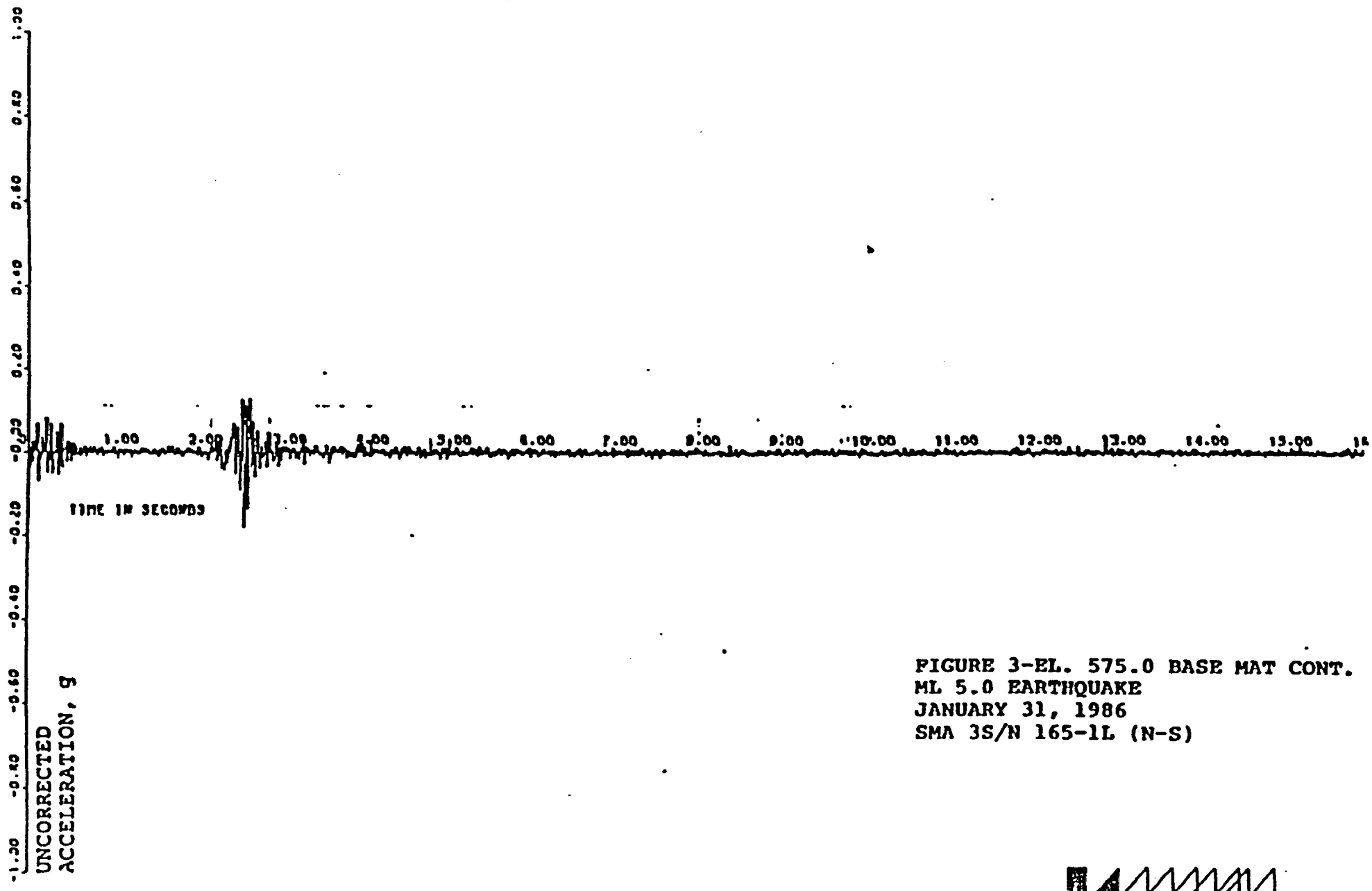

FIGURE 3-EL. 575.0 BASE MAT CONT.
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-1L (N-S)

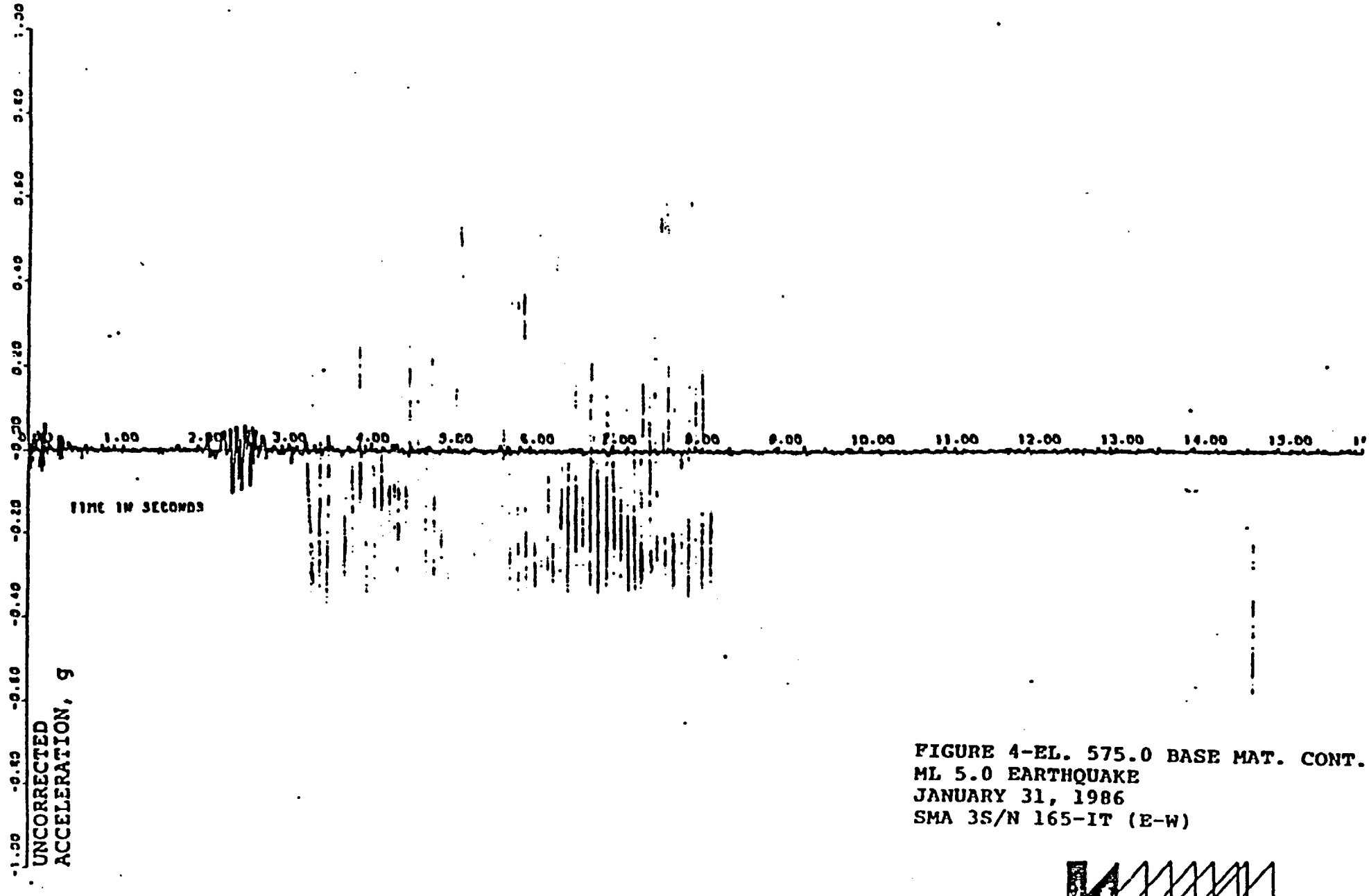

FIGURE 4-EL. 575.0 BASE MAT. CONT.
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-IT (E-W)

KINEMETRICS

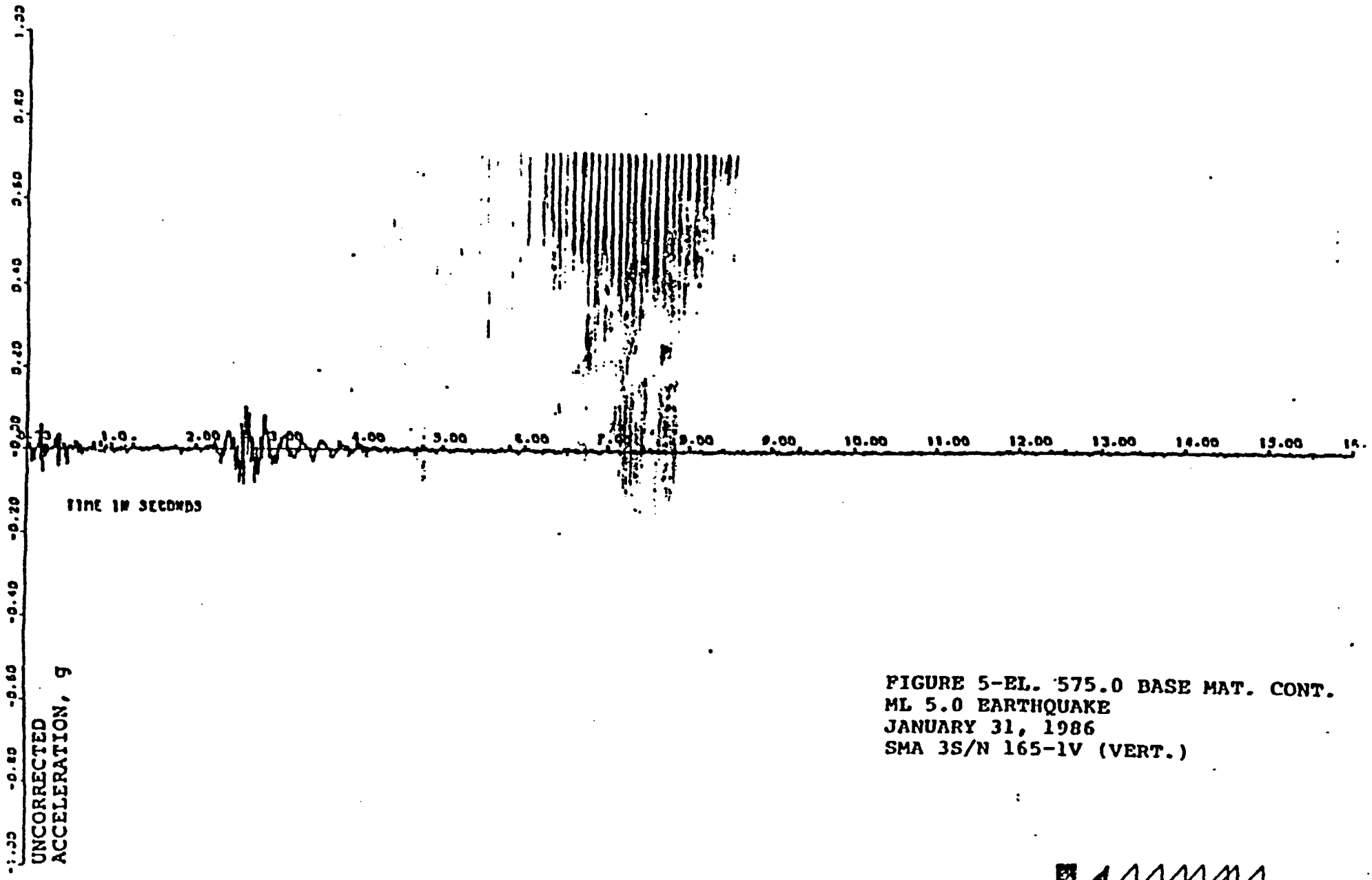

FIGURE 5-EL. 575.0 BASE MAT. CONT.
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-1V (VERT.)

KINEMETRICS

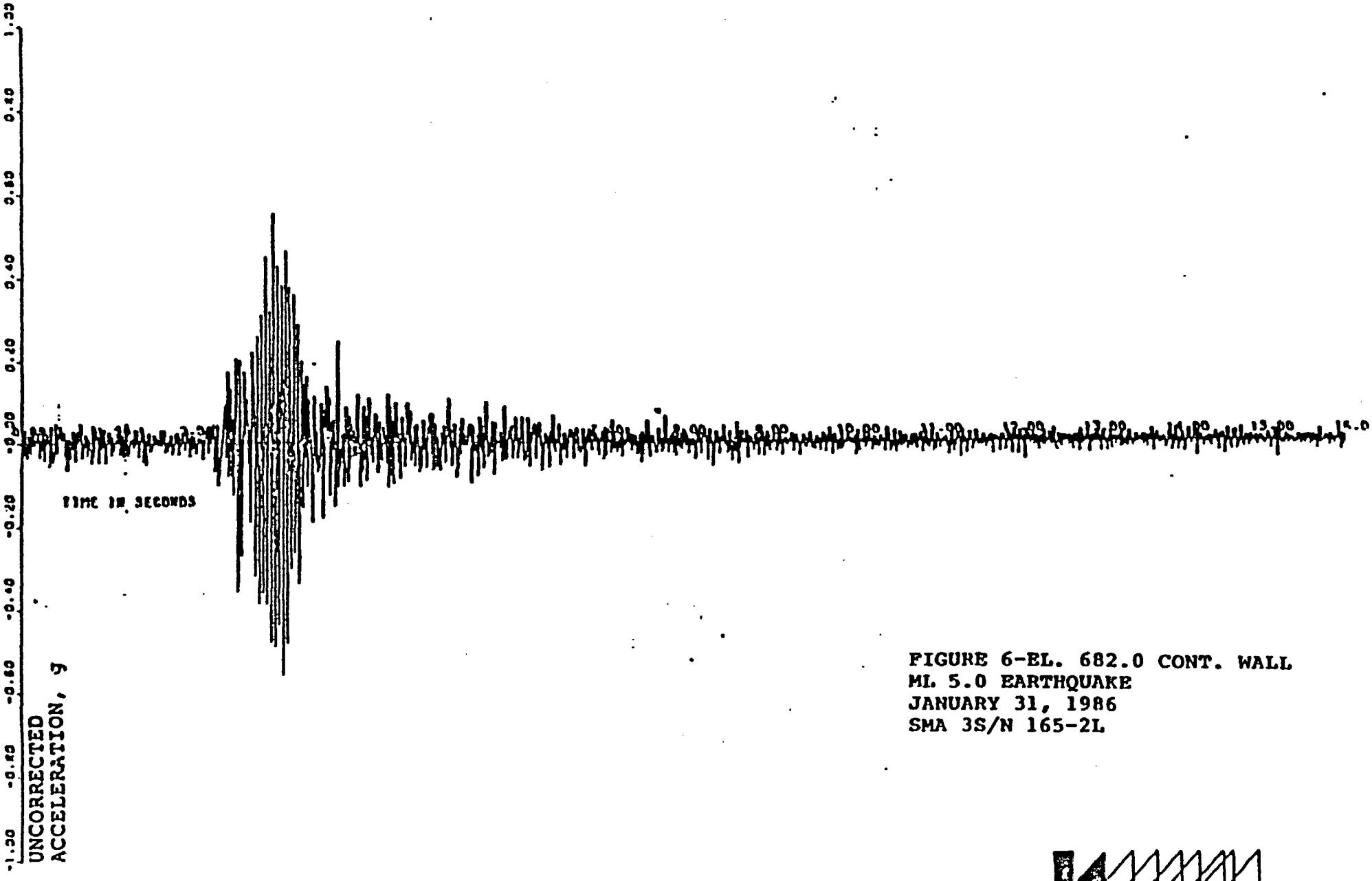

FIGURE 6-EL. 682.0 CONT. WALL
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-2L

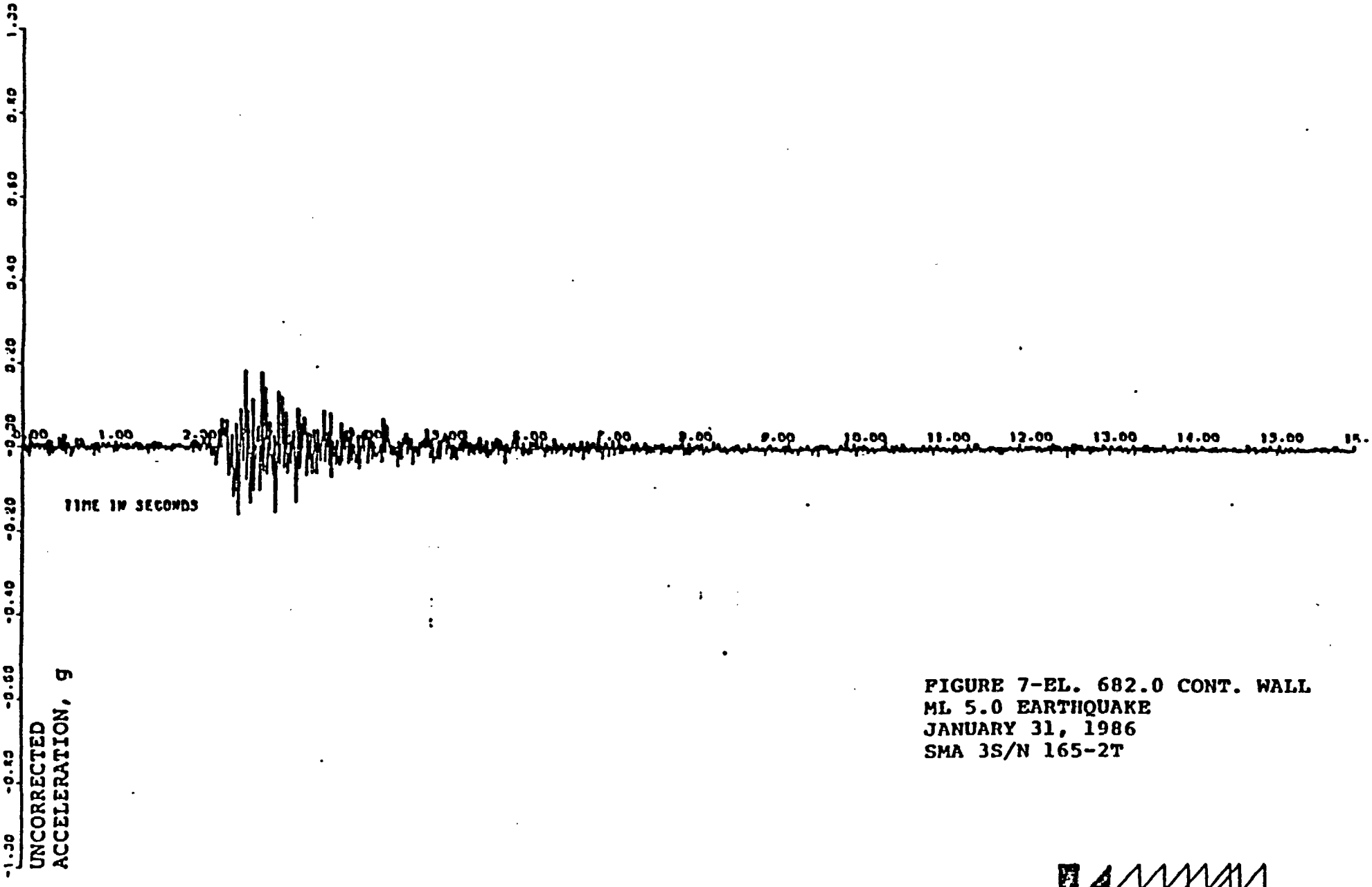

FIGURE 7-EL. 682.0 CONT. WALL
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-2T

KINEMETRICS

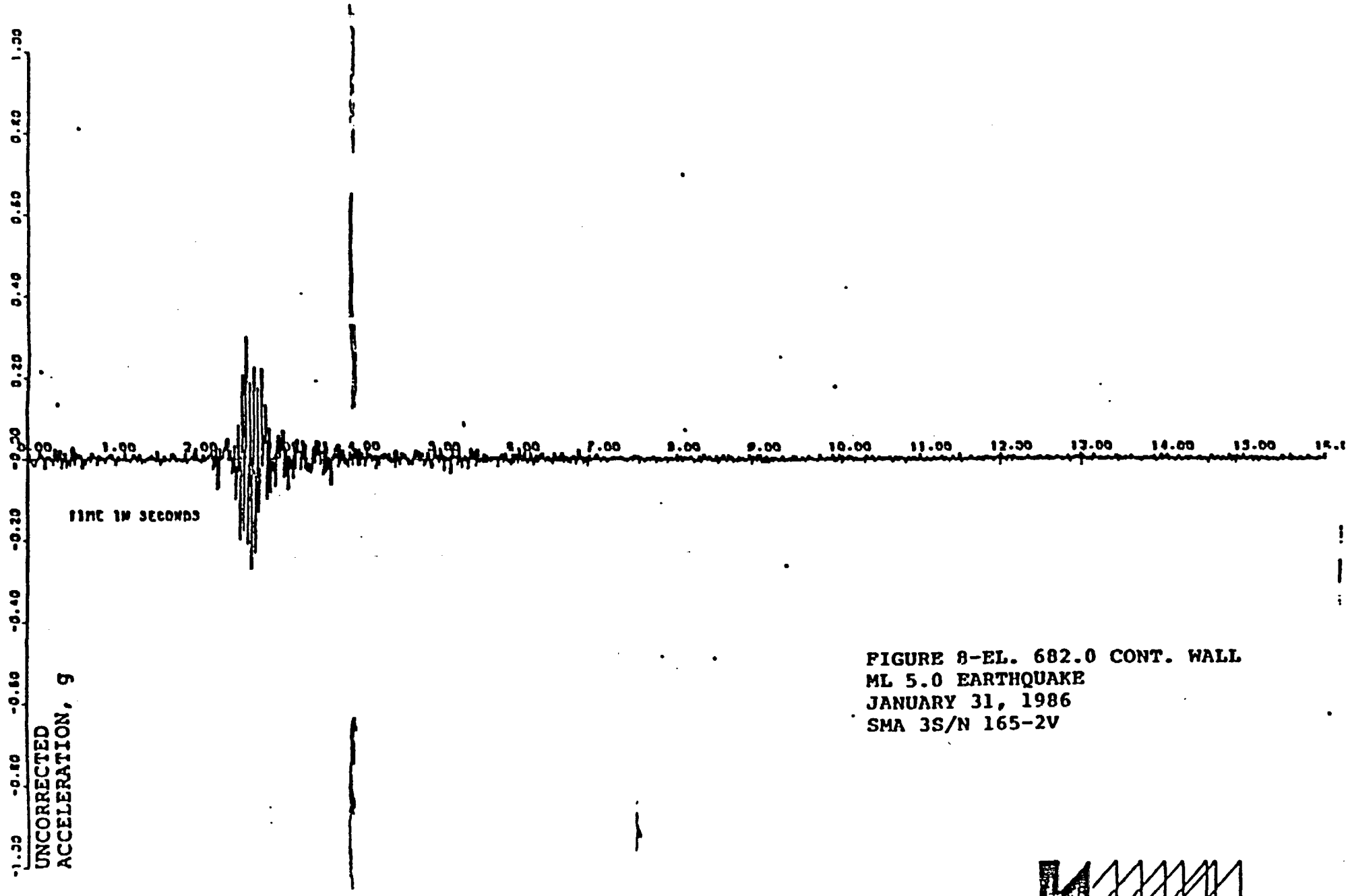

FIGURE 8-EL. 682.0 CONT. WALL
ML 5.0 EARTHQUAKE
JANUARY 31, 1986
SMA 3S/N 165-2V

KINEMETRI

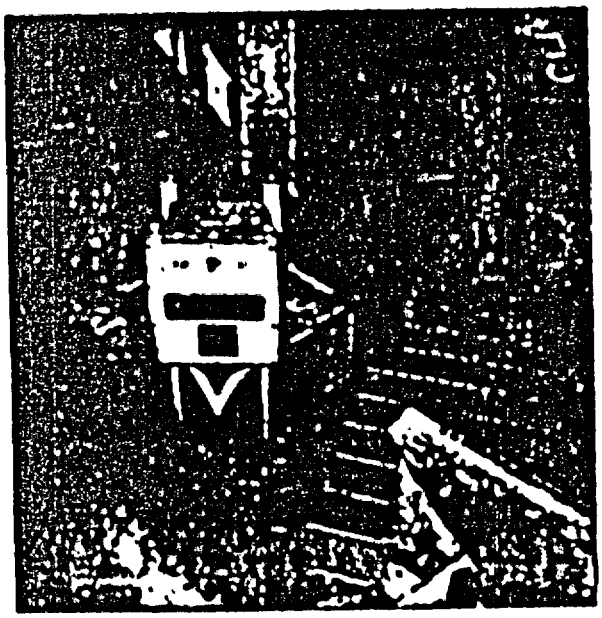

FIGURE 9 - MOUNTING OF
ENGDAHL PSR 1200-H/V
RECORDER ON TWV NO. 43
SHIPWELL DIAGRAM.

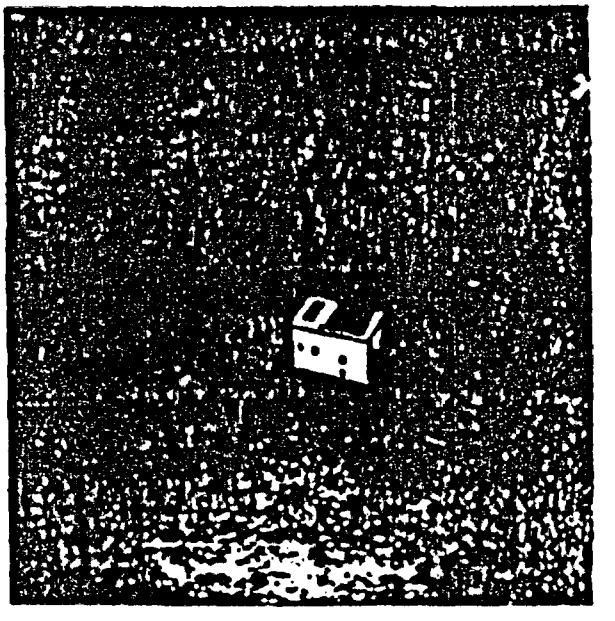

FIGURE 10 - MOUNTING OF
ENGBARL TYPE PAV
PECORBIT ON
BUILDING
NO. 555.

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(11) Kennedy, R.P., Tong, W.H., and Short, S.A., "Earthquake Design Ground Acceleration Versus Instrumental Peak Ground Acceleration", SMA 12501.01, Structural Mechanics Associates, Newport Beach, California, October 1980.

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(17) U.S. Nuclear Regulatory Commission, Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants" (Rev. 1, December 1973).

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.

2. Moses, F., and Stevenson, J.D. "Reliability Based Structural Design," Journal of the Structural Division, ASCE, Vol. 96, No. ST 2, Proc. Paper 7072, February 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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22. Stevenson, J.D. Chairman, Editing Board, Structural Analysis and Design of Nuclear Plant Facilities, ASCE Manuals and Reports on Engineering Practice - No. 58, American Society of Civil Engineers, August 1980.

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.

25. Stevenson, J.D. and Thomas, F.A., "Selected Review of Foreign Licensing Practices for Nuclear Power Plants," NUREG/CR-2664, U.S. Nuclear Regulatory Commission, April 1982.

26. Stevenson, J.D. and Thomas, F.A., "Selected Review of Regulatory Standards and Licensing Issues for Nuclear Power Plants," NUREG/CR-3020, U.S. Nuclear Regulatory Commission, November 1982.

27. Stevenson, J.D. and Thomas, F.A., "Selected Review of Foreign Safety Research for Nuclear Power Plants," NUREG/CR-3040, U.S. Nuclear Regulatory Commission, November 1982.

28. Stevenson, J.D., "Use of the Delphi Approach in Seismic Qualification of Existing Electrical and Mechanical Equipment and Distribution Systems in Industrial Facilities," Proceedings of the 6th Japan Earthquake Engineering Symposium - 1982, Tokyo, Japan, December 1982.

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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

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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

RESPONSE SPECTRA (SSE)
N/S DIRECTION
ELEVATION 568'-4"

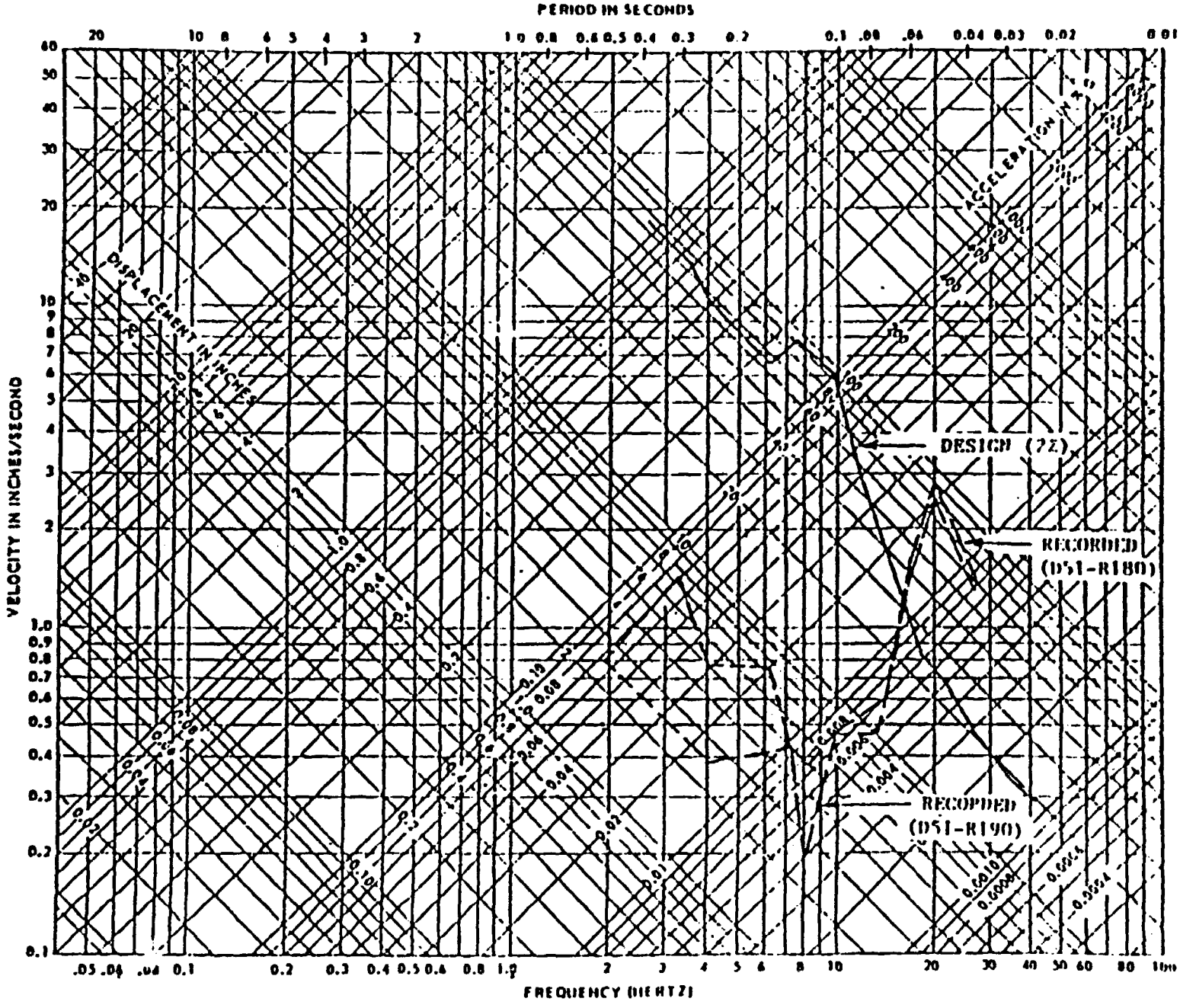
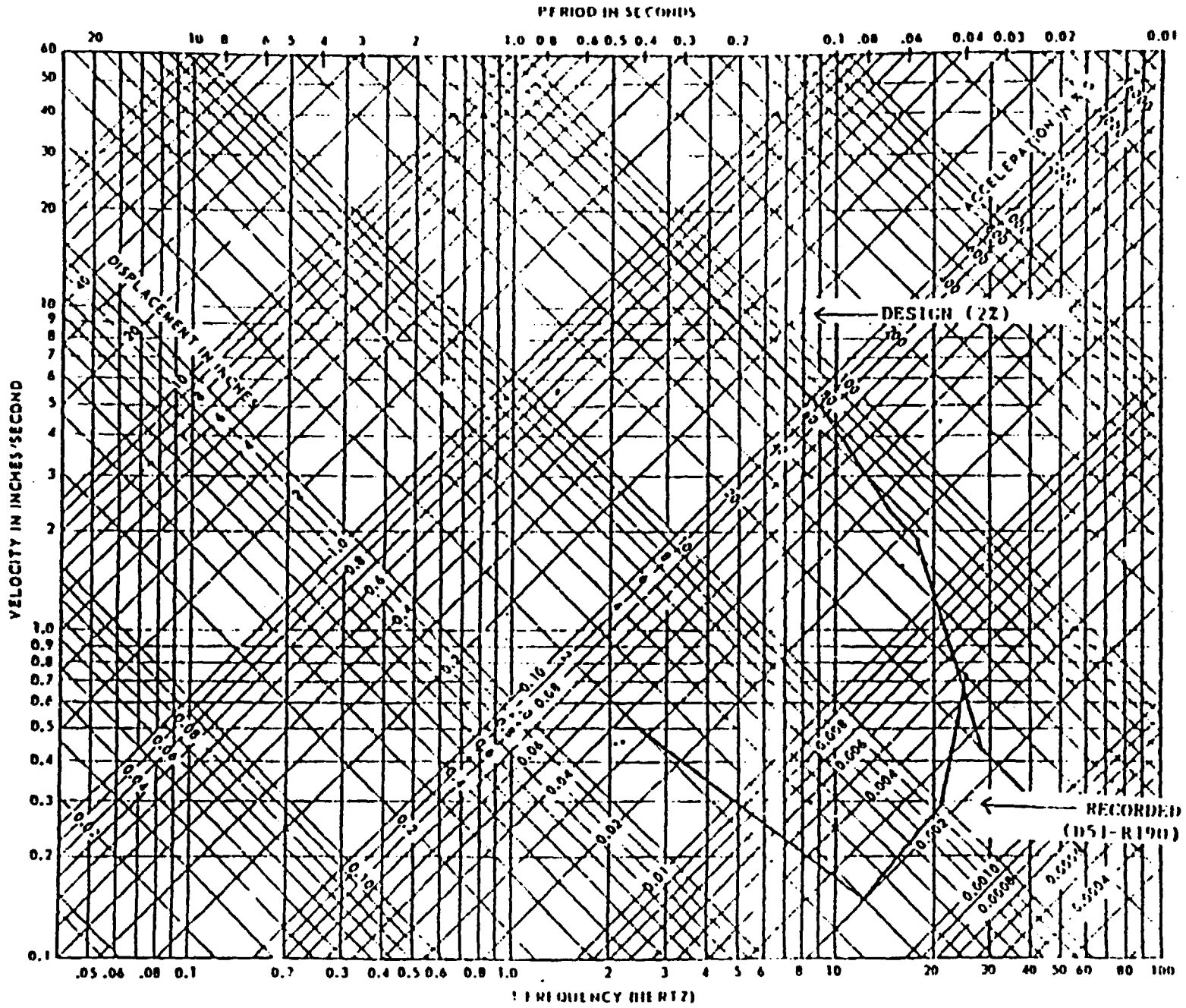


FIGURE D-1

PSPP UNIT NO. 1
AUXILIARY BUILDING
RESPONSE SPECTRA (SSE)
VERTICAL
ELEVATION 568'-4"



NOTE: D51-R180
(Vertical) Out of [illegible]
Due to Recalibration

FIGURE D-3

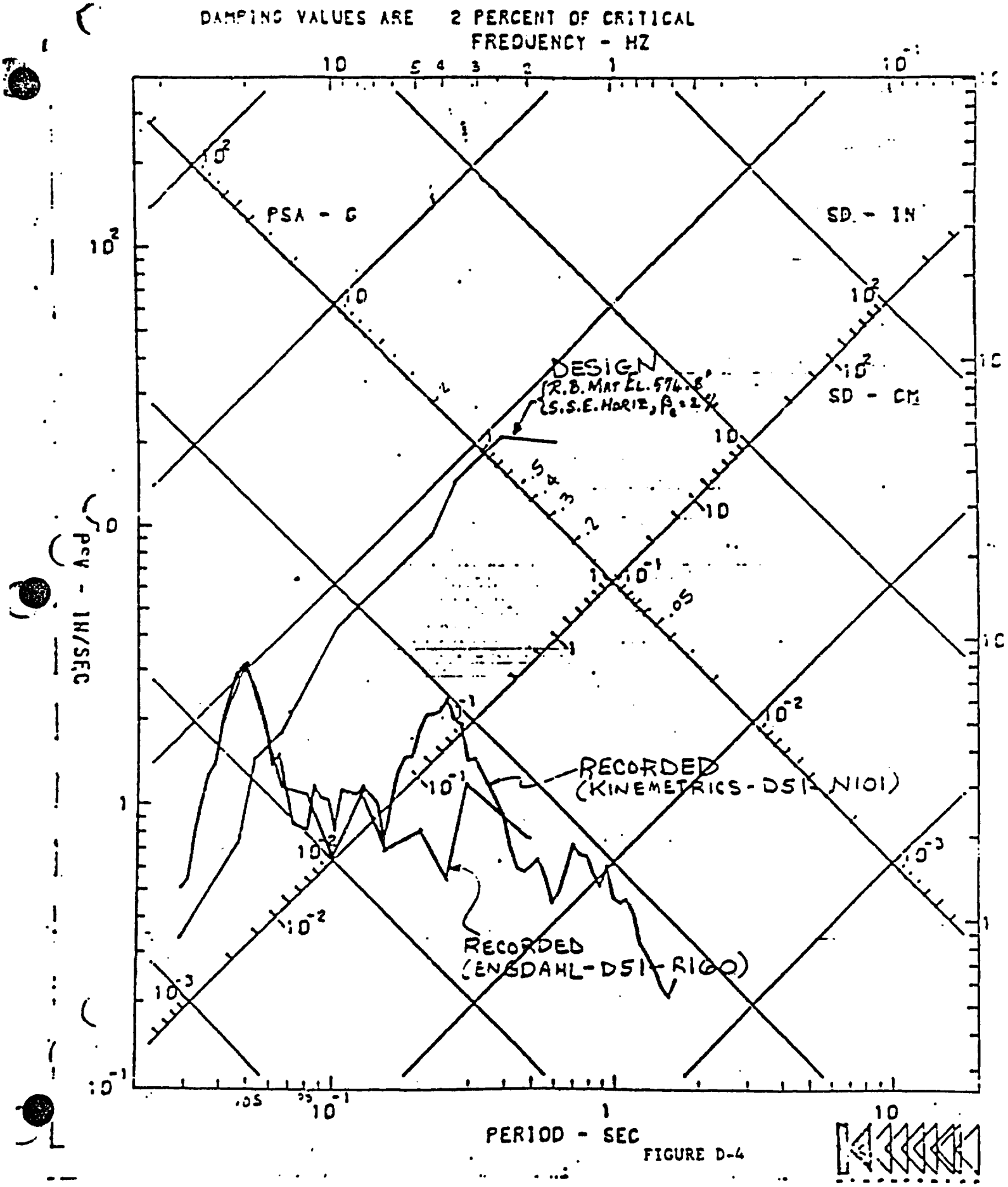

DAMPING VALUES ARE 2 PERCENT OF CRITICAL
FREQUENCY - HZ
PSA - G
SD - IN
SD - CM
DESIGN
(R.B. MAT EL. 574.8'
(S.S.E. HORIZ, β = 2%
RECORDED
(KINEMETRICS - DS1 - N101)
RECORDED
(ENSDAHL - DS1 - R100)
PSV - IN/SEC
PERIOD - SEC

FIGURE D-4

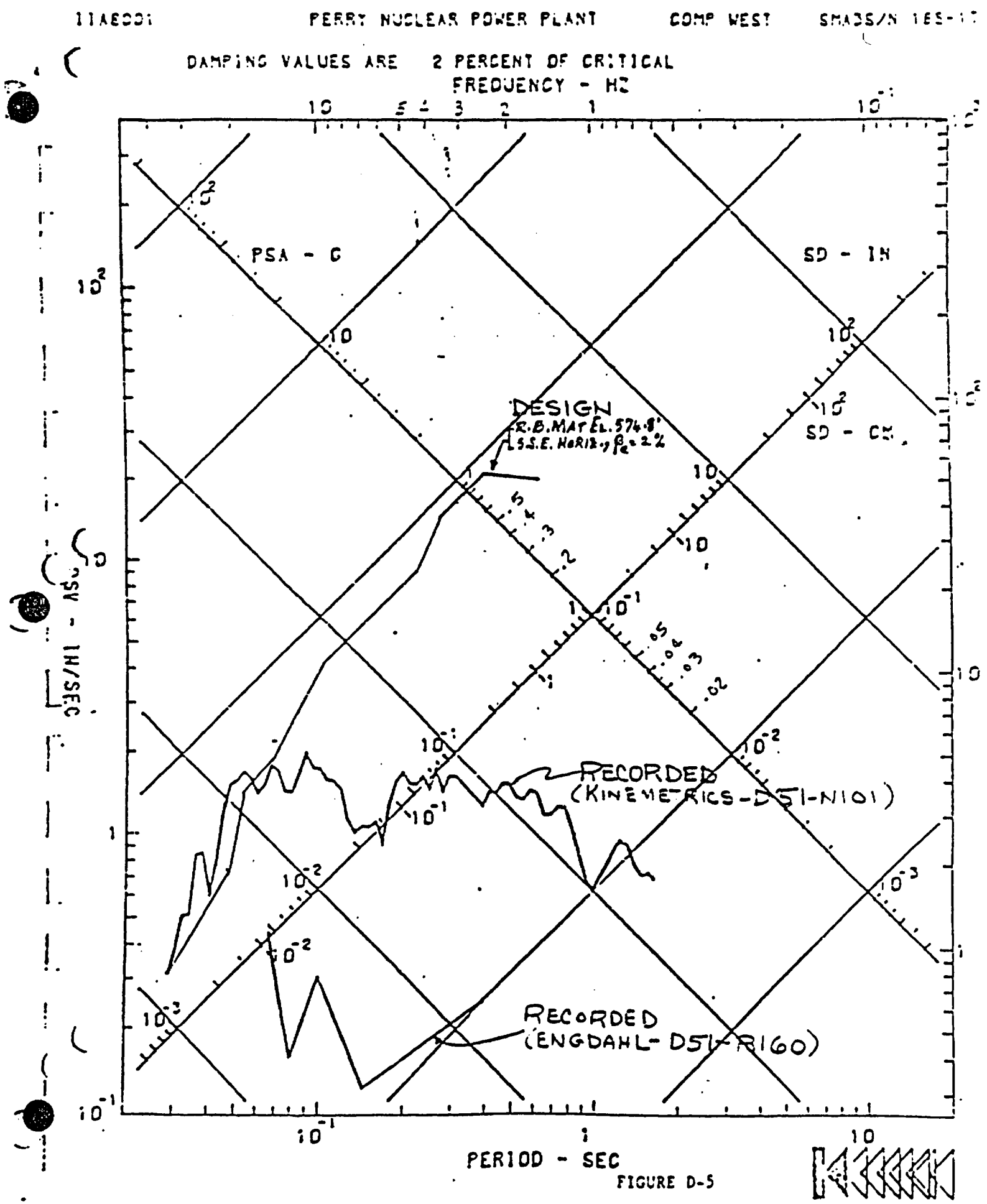

PERRY NUCLEAR POWER PLANT
COMP WEST
DAMPING VALUES ARE 2 PERCENT OF CRITICAL
FREQUENCY - HZ
PSA - G
SD - IN
SD - CM
DESIGN
R.B. MAT EL. 574.8'
S.S.E. HORIZ., β_c = 2%
RECORDED
(KINEMETRICS-D51-N101)
RECORDED
(ENGDAHL-D51-R160)
PERIOD - SEC

FIGURE D-5

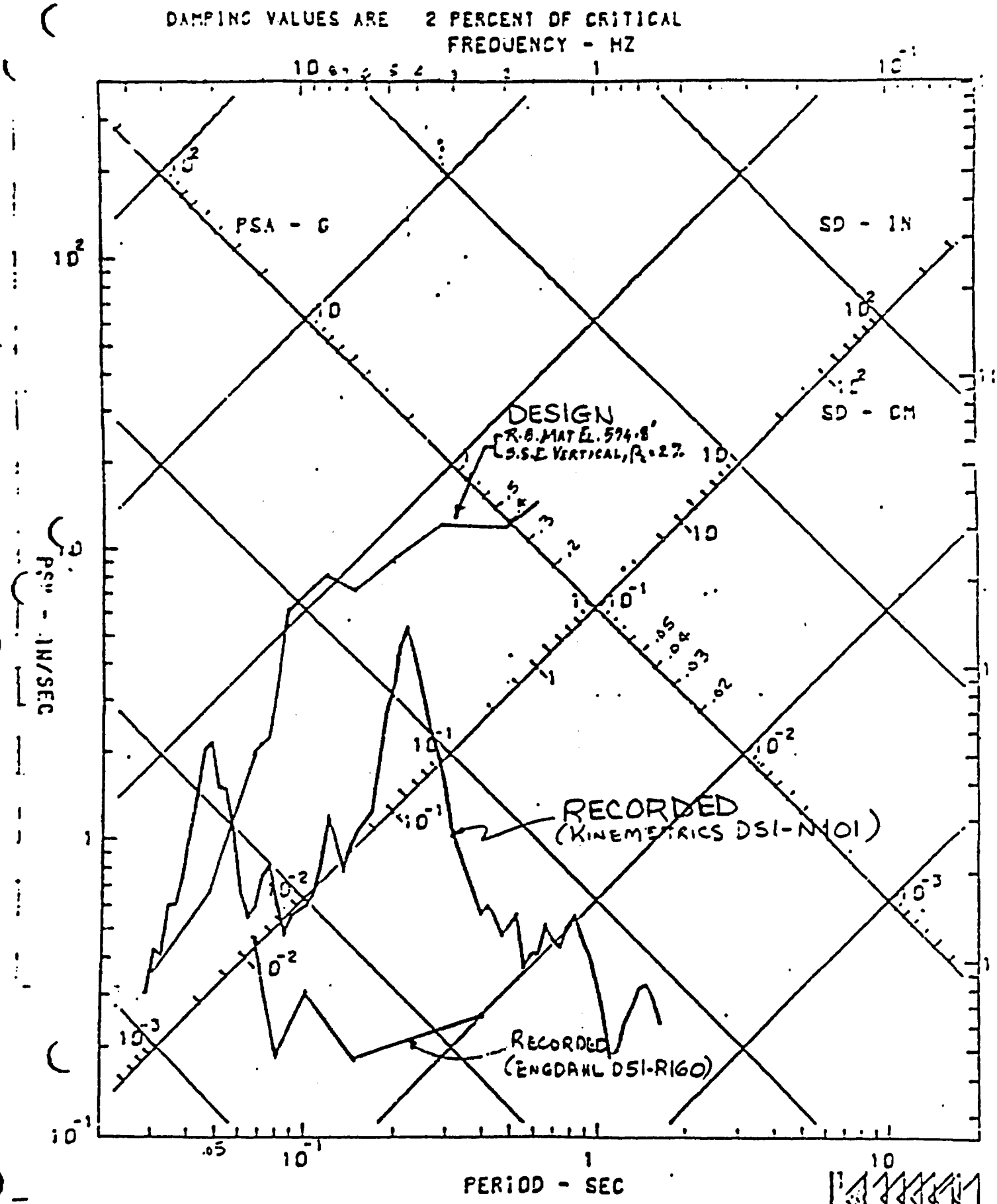
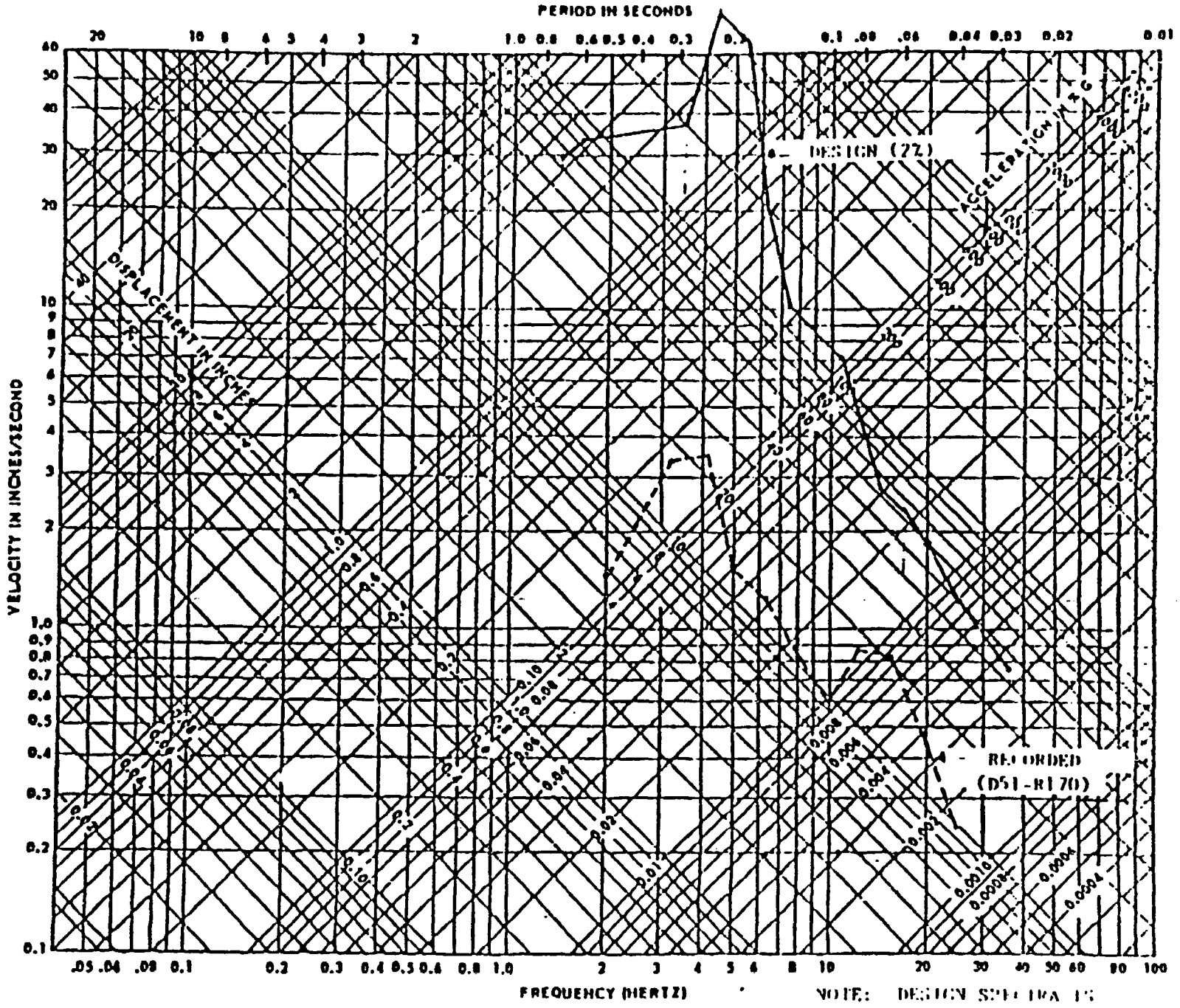

DAMPING VALUES ARE 2 PERCENT OF CRITICAL
FREQUENCY - HZ
PSA - G
SD - IN
SD - CM
DESIGN
R.B. MAT EL. 574.8'
S.S.E. VERTICAL, β_R = 2%
RECORDED
(KINEMETRICS DS1-N101)
RECORDED
(ENGDAHL DS1-R160)
PSV - IN/SEC
PERIOD - SEC

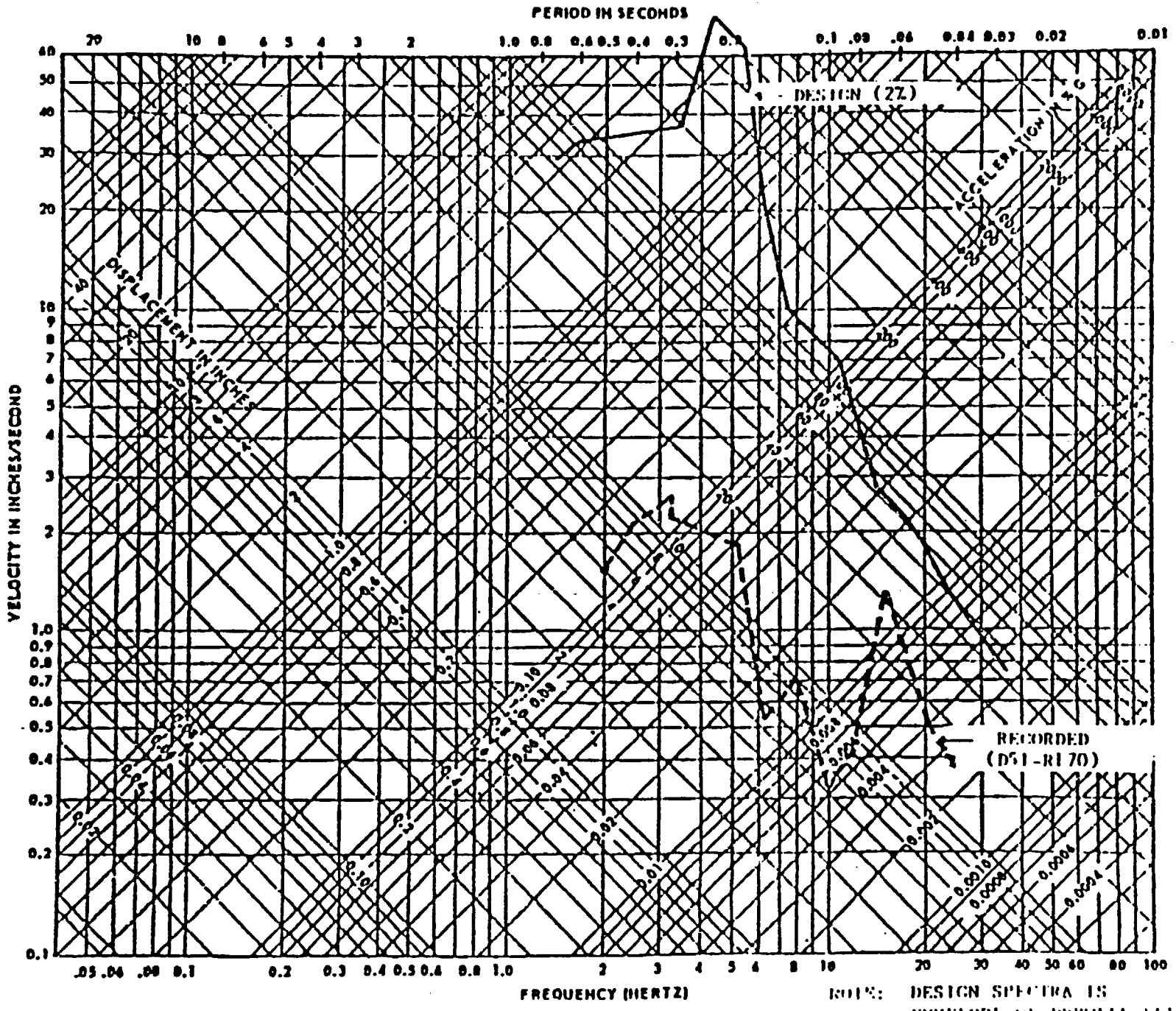
FIGURE D-6



NOTE: DESIGN SPECTRA IS ENVELOPE OF DRYWELL [illegible] AND BIOLOGICAL WALL [illegible] SPECTRA

REACTOR BUILDING
RESPONSE SPECTRA
NORTH-SOUTH
EL [illegible] PLATFORM

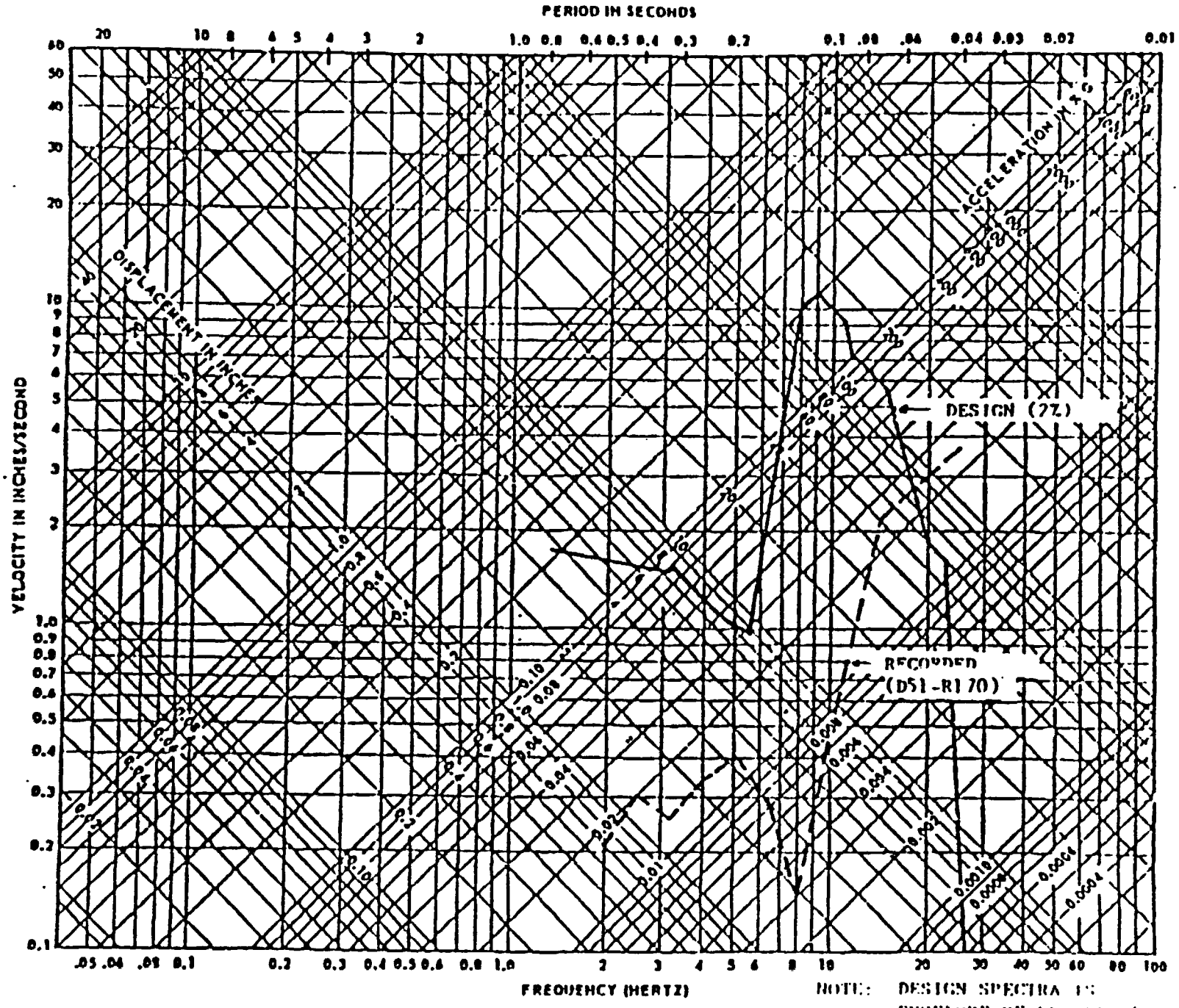
FIGURE D-7



NOTE: DESIGN SPECTRA IS ENVELOPE OF DRYWELL [illegible] AND MIDDLE [illegible]

BWRP UNIT NO. 1
REACTOR BUILDING
RESPONSE SPECTRA [illegible]
EAST-WEST
EL 631' PLATFORM

FIGURE D-8



NOTE: DESIGN SPECTRA IS
ENVELOPE OF DROOPY [illegible]
AND BIOLOGICAL [illegible]
SPECTRA

[illegible] NO. 1
REACTOR BUILDING
RESPONSE SPECTRA /SSE
VERTICAL
EL 831' PLATFORM

FIGURE D-9

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11A6002 PERRY NUCLEAR POWER PLANT COMP SOUTH SMA3S/N 65-...

DAMPING VALUES ARE 2 PERCENT OF CRITICAL

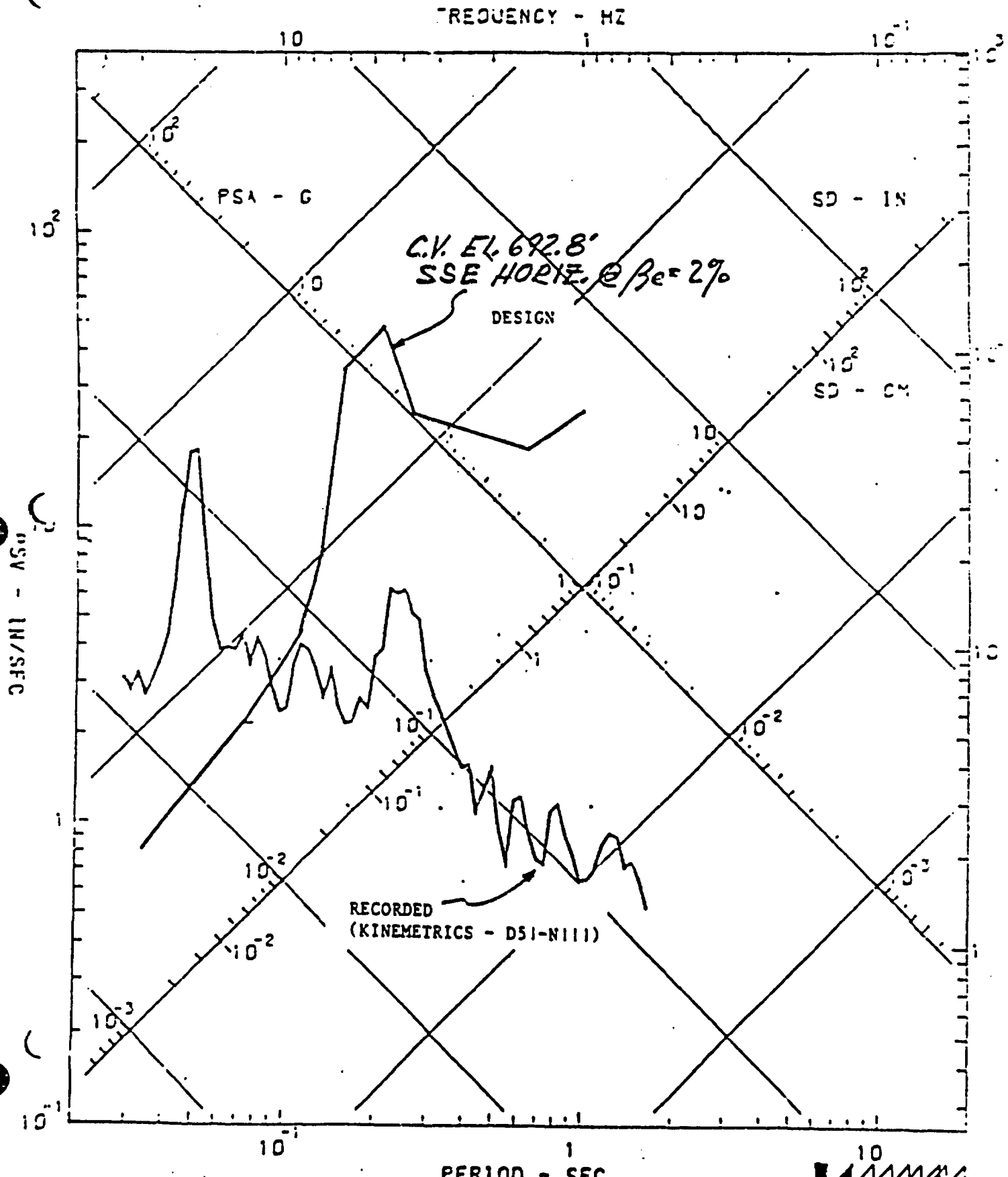


FIGURE D-10

ML 5.0 EARTHQUAKE JANUARY 31, 1986

11AE002 PERRY NUCLEAR POWER PLANT COMP WEST SMAD5/N 165-21

DAMPING VALUES ARE 2 PERCENT OF CRITICAL

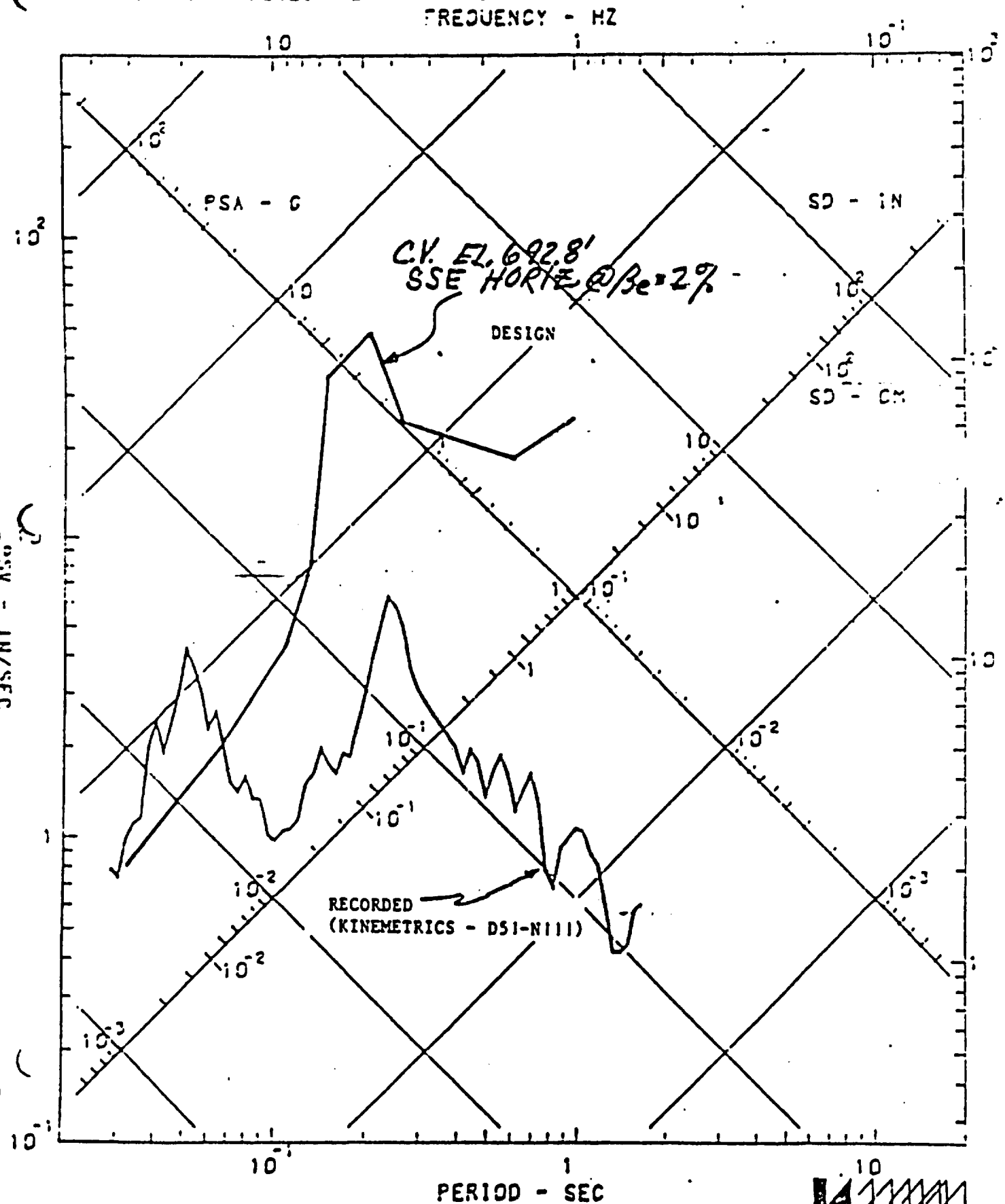


FIGURE D-11

ML 5.0 EARTHQUAKE JANUARY 31, 1986

PERRY NUCLEAR POWER PLANT COMP UP

DAMPING VALUES ARE 2 PERCENT OF CRITICAL

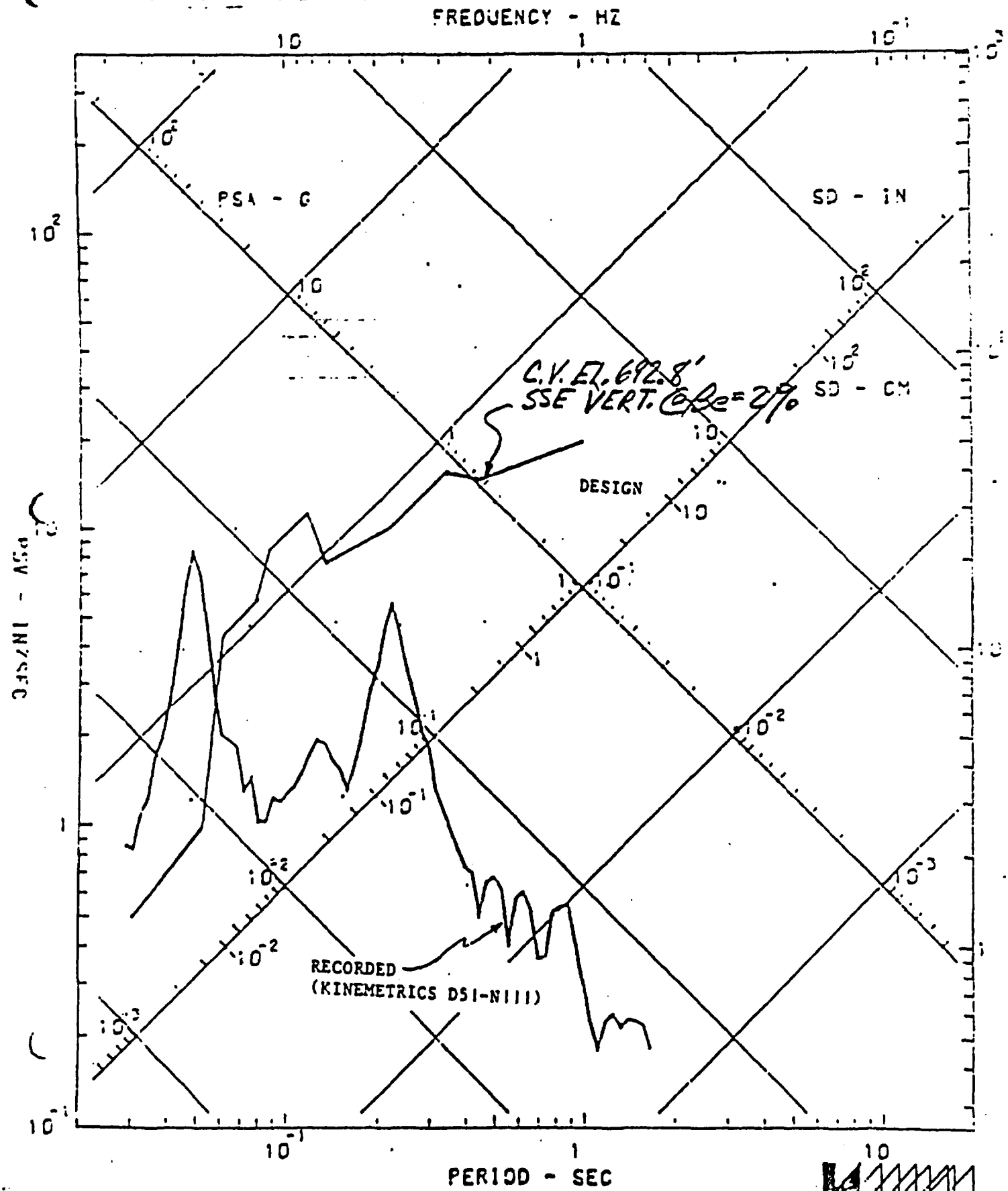


FIGURE D-12

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

MEMORANDUM

☐ I no longer wish to receive this memorandum

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

6-2
REV. 1-82

MEMORANDUM

☐ I no longer wish to receive this material.

O. K. Pech

ROOM W210 FROM M.R. Kritzer DATE 2-6-86
PHONE 6460 ROOM TQ6
SUBJECT PLANT SETTLEMENT READINGS

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	1/15/86	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				
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. SURVEYORS — 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	626.390	626.387	626.388	626.391		
5·D	624.538	624.544	624.544	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								

DAKELL & ABDUL, SURVEYORS — PERRY, OHIO.

SKETCH OF: PLANT SETTLEMENTS

DATE: (M)

SCALE:

FIELD BOOK:

PT.	SEPT 27, 83	OCT 27, 1983	NOV. 23, 83	DEC 21 83	JAN. 5, 1984	FEB 10, 84	MAR. 24, 84
1-F	625.396	625.3[illegible]5	625.402	625.407	625.405	625.406	625.401
2-D	624.332	624.3[illegible]7	624.335	624.338	624.334	HOCKED	624.331
3-D	624.372	624.365	624.365	624.363	624.361	624.362	624.361
4-F	626.389	626.387	626.388	626.388	626.385	626.391	626.381
5-D	624.543	624.517	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	624.33[illegible]	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	626.39[illegible]	626.389	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	624.516	624.543
6-D	622.096	622.094	622.105	622.103	622.106	622.100	622.101	622.108
7-C	[illegible]17	621.296	621.292	621.297	621.297	621.297	621.306	621.299

GARRETT & ASSOC. CIV. ENGINEERS SURVEYORS

PERRY NUCLEAR POWER PLANT
PERRY, OHIO.

SKETCH OF

FIELD BOOK

SCALE

DATE

PERRY TOWER PLANT
FOUNDATION SETTLEMENT CHART

MARK NO	DEC 18, 78	JAN [illegible]	FEB 27	MARCH 20	MAY 6, 1979	MAY 23, 1979	JULY 18, 79	SEPT 10, 79	OCT 5, 79	OCT [illegible]
1-E	625.286	625.274	625.270	625.270	625.270	—	—			
1-F	NEW DISC		→	625.393	625.391	625.395	625.406	625.399	625.410	625.410
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	624.397	624.398	624.395	624.403	624.386
4-C	596.003	COVERED		596.001	COVERED	COVERED	COVERED			
4-D	REMOVED →	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	TEMPORARY COVERED	COVERED	"			
5-D				NEW DISC →	624.561	624.551	624.548	624.548	624.551	624.551
6-C	619.025	COVERED								
6-D	NEW DISC		→	622.086	622.088	622.096	622.095	622.096	622.112	622.107
7-A	COVERED			→	COVERED	COVERED				
7-B	NEW MARK		→	616.733	616.735	616.732	616.737	616.729	616.725	(1)621.[illegible]
4-E	NEW				620.414	620.465	620.462			
4-F							626.412	626.409	NEW 626.407	626.[illegible]

MARK NO	OCT 31, 1979	NOV 17, 1979	JAN 21, 1980	MAR 22, 1980	APR [illegible], 1980	JUNE 23, 1980	AUG 12, 1980	SEPT 19, 1980	OCT 22, 1980	DEC 15, 1980	JAN 21, 1981
1-E	625.418	625.412	625.402	625.414	625.415	625.419	625.407	625.392	625.390	625.400	625.385
2-D	624.346	624.349	624.339	624.353	624.350	624.354	624.344	624.336	624.338	624.336	624.326
3-D	624.396	624.391	624.382	624.386	624.385	624.400	624.385	624.372	Blanked	624.370	624.362
4-F	626.405	626.409	626.370	626.403	626.407	626.409	626.397	626.387	626.390	626.419	626.377
5-D	624.554	624.551	624.540	624.547	624.549	624.549	624.546	624.534	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 (621.296) RECHECKED	621.308	621.311	621.310	621.314	621.369	621.303	621.301

GARRETT & ASSOC. ENG. CONSULTANTS SURVEYORS

PERRY NUCLEAR POWER PLANT
PERRY, OHIO.

SKETCH OF

FIELD BOOK

SCALE

DATE

POINT \ DATE	FEB 9, 1981	[illegible]	APRIL 15 1981	MAY 22 1981	July 2, 1981	[illegible]	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	620.396	625.396	625.391
2-D	624.332	624.325	624.325	624.322	624.332	624.338	624.333	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	626.385
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	622.105	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
	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	

POINT \ DATE	JAN 5 1982	JAN 15 1982	FEB 17, 82	MAR. 11, 1982	APRIL 26, 82	MAY 17 1982	JUNE 22 1982	July 26, 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	625.393	625.394	625.398	625.398
2-D	624.327	624.325	624.322	BLOCKED	624.328	624.333	624.324	624.331	624.330	624.335	624.335
3-D	624.365	624.363	624.352	624.361	624.366	624.365	624.358	BLOCKED	BLOCKED	BLOCKED	BLOCKED
4-F	626.387	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	624.537	624.541	624.538	624.541	624.541	624.544	624.533	624.538
6-D	622.092	622.095	622.096	622.096	622.095	622.096	622.098	622.095	622.098	622.097	622.097
7-C	621.281	621.276	621.276	621.269	621.284	621.289	621.282	621.269	621.290	621.270	621.294

POINT \ DATE	Nov. 30, 82	DEC. 8, 1982	JAN 25, 1983	FEB 16, 83	MAR 30, 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	625.406	625.401	625.402	625.393	625.399	625.403	625.399
2-D	624.334	624.329	624.331	624.328	624.333	624.330	624.332	624.321	624.333	624.339	624.332
3-D	BLOCKED	BLOCKED	BLOCKED	624.367	624.363	624.361	624.368	624.356	624.363	624.374	624.367
4-F	626.387	626.374	626.379	626.382	626.392	626.380	626.394	626.380	626.383	626.394	626.388
5-D	624.539	624.542	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	621.280	621.296	621.288	621.294	621.291	621.286	621.288		621.295	621.289	621.295

PERRY NUCLEAR POWER PLANT
FOUNDATION SETTLEMENT CHART CONTINUED BY GARRETT
SHEET NO. 3-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	
1-B	COVERED	COVERED	WILL BE CLEAR BEFORE NOV	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
3-B	599.046	599.041	599.040	599.044	599.052	COVERED 5' TO 604.ML	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 OVER LINE.	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	"	"			
3-B				598.805	598.794	COVERED	COVERED	"	"			
1-D					NEW. 614.287	COVERED	COVERED	"	"		614.291	614.295
3-C					623.514	623.507	623.506	"	"			
4-C					576.018	COVERED	COVERED	595.997	595.999			596.007
7-A.	MARK	ON WALL	12' W. OF SE COR		NEW 563.148	562.132	562.113	COVERED				563.100
3-D					PERMANENT	DISC. SET.	REACTOR	N°2 (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 →	605.600
												SEE NOTE A

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. 12 77	AUG 19, 77	AUG 29, 77	SEPT. 14, 77	SEPT 26, 77	OCT 10 77	[illegible]
1A	582.090	582.093	582.092	582.095	582.095	582.087	582.092	582.091	COVERED	—	—	
1B	595.619	595.619	595.622	595.625	595.621	595.615	595.624	595.620	595.620	595.619	595.622	595.6[illegible]
2	572.981	572.983	COVERED			572.979	572.984	COVERED	—	—	—	—
2A	576.334	578.333	578.335	COVERED		—	COVERED	COVERED	—	—	—	—
2B	589.980	COVERED	589.981	589.984	589.984	589.978		589.982	589.981	COVERED	—	—
3	572.982	572.982	572.984	572.987	COVERED	572.986	572.990	572.989	572.987	COVERED	—	—
[illegible]	585.491	585.493	585.494	585.493	585.493	585.493	585.493	585.493	585.495	585.494	585.496	585.498
[illegible]	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	587.476	587.477	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							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	COVERED	COVERED	—
• 2-D	624.358	624.358	624.361	624.362	624.358	624.362
3-A	COVERED	—		—	—	—
• 4-A	COVERED BY PEDESTAL	—		—	—	—
• 5	COVERED	COVERED	[illegible]	COVERED	COVERED	—
• 6-A	587.470		587.490	587.460	587.446	587.446
• 1-B	596.943	596.942	596.938	596.940	596.940	596.940
• 2-C	600.142	—	600.143	600.155	600.155	—
• 1-B	602.183	602.180	602.190	602.194	VOID	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

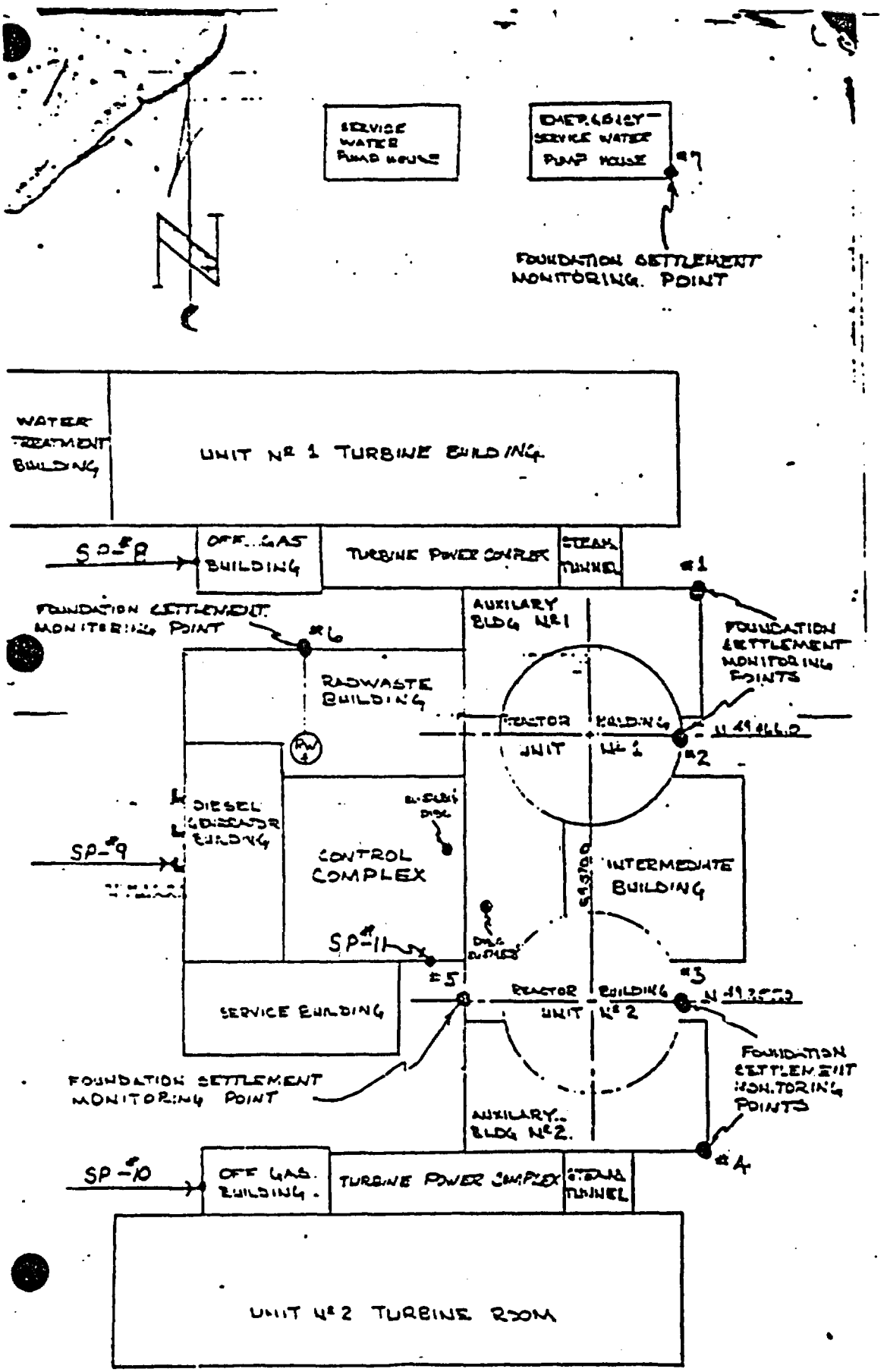
FOUNDATION SETTLEMENT CHART BY GARRETT + ASSOC. INCORPORATED

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
1	568.334	568.346	568.339	568.333	568.334	568.334	568.335	568.333	568.333	568.318	568.333	568.3
2	572.992	572.993	572.990	572.991	572.990	572.988	572.987	572.988	572.988	572.987	572.986	572.9
3	572.998	572.996	572.996	572.996	572.995	572.994	572.992	572.993	572.989	572.986	572.984	572.9
4								SET MARCH 14, 77	568.295	568.291	568.287	568.2
5												
6					SET FEB 14, 77	573.934	573.931	573.936	573.938	573.931	573.930	573.93

MARK Nº	APRIL 14, 77	APRIL 29 77	APRIL 22 77	[illegible]	MAY 12, 77	MAY 19, 77	MAY 27, 77	JUNE 9, 1977	JUNE 16, 77	JUNE 15, 77	JUNE 22 77	JULY 1
1	568.333	568.330	568.331	568.330	568.334	568.335	568.337	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
2	572.986	572.986	572.986	572.986	572.986	572.984	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
3	572.987	572.984	572.987	572.988	572.990	572.986	572.985	572.989	572.988	572.990	572.989	572.984
4	568.287	568.284	568.285	568.287	568.288	568.287	568.291	568.290	568.286	568.285.493	568.493	[illegible]
5		SET MAR 27, 77	572.905	572.905	572.906	572.905	572.904	572.897	572.896	572.905	572.895	572.904
6	573.938	573.936	573.939	573.935	573.935	573.932	573.932	[illegible]	[illegible]	573.493	573.493	[illegible]


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
FOUNDATION SETTLEMENT MONITORING POINT
#6
AUXILARY BLDG Nº1
FOUNDATION SETTLEMENT MONITORING POINTS
RADWASTE BUILDING
REACTOR BUILDING
UNIT Nº 1
#2
DIESEL GENERATOR BUILDING
SP-#9
CONTROL COMPLEX
INTERMEDIATE BUILDING
SP-#11
#5
#3
SERVICE BUILDING
REACTOR BUILDING
UNIT Nº 2
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

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 T06
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 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 ELIA NO DAMAGE ENCOUNTERED.

cc: S. DODEJA - E180

	[signature]	2-3-86
	INSPECTOR	DATE
	[signature]	2-4-86
	REVIEWED - LEAD INSPECTOR	DATE

441/T/1/ba

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		M.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. Dedeja F-180

INSPECTOR 2-4-86 DATE

REVIEWED - LEAD INSPECTOR 2-4-86 DATE

441/T/1/DA

PNPP No. 5978

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 }

S.C.U. # 6639 } 599' Floor elev.
S.C.U. # 6688 }

S.C.U. # 5968 - 574'-~~6~~ 10" Floor elev. (see 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. Dedeja E-180

K. Manno — INSPECTOR — 2-4-86 DATE

J. Messenger — REVIEWED - LEAD INSPECTOR — 2-4-86 DATE

441/T/1/ba

PHPP No. 5978

N/A	N/A	C-23663
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
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 SCU# 6262 has repair work in-progress.

cc: S. Dodge E-180

H Hemm — INSPECTOR — 2-4-86 DATE

J Messenger — REVIEWED - LEAD INSPECTOR — 2-4-86 DATE

441/T/1/bs

Nonsafety Inspection Report (NIR)

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	*	EL. 599'
597	*	
5518	*	
6633		
4727	*	EL. 682'
4729	*	

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. Shane, Jr. 2/4/86
INSPECTOR DATE

John Messenger 2-4-86
REVIEWED - LEAD INSPECTOR DATE

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

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. — INSPECTOR — 2/4/86 DATE

John W. Bruenger — REVIEWED - LEAD INSPECTOR — 2-4-86 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. Shaney, Jr. 2/5/86
INSPECTOR DATE

John W Massinger 2/5/86
REVIEWED - LEAD INSPECTOR DATE

441/T/1/ba

NIR C-23677 SH. 6 OF 6

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

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

C-2
REV. 1-82

MEMORANDUM

"E"SO/-2712

☐ I no longer wish to receive this material.

E. M. Mead — ROOM E280
T. M. Jameson — E260
T. P. Keaveney — E230

FROM I. B. Babiak
PHONE 6699 ROOM E260
SUBJECT N71, Circulating Water System Walkdown to Assess the Intensity of the January 31, 1986 Earthquake

DATE February 3, 1986

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

MEMORANDUM

"C"SO-2773

☐ I no longer wish to receive this memo.

K. R. Pech — ROOM W220

FROM: E. C. Christiansen — 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