Transcript of Proceedings

UNITED STATES OF AMERICA

PRESIDENT'S COMMISSION ON THE ACCIDENT AT
THREE MILE ISLAND

DEPOSITION OF: THOMAS A. IPPOLITO

Bethesda, Maryland

August 9, 1979

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DEPOSITION OF: THOMAS A. IPPOLITO

Old Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland

August 9, 1979
10:25 o'clock, a.m.

APPEARANCES:

On Behalf of the Commission:

KEVIN P. KANE, ESQ.
Deputy Chief Counsel
2100 M Street, N. W.
Washington, D. C.

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

On Behalf of the Deponent:

MARJORIE S. NORDLINGER, Attorney
Office of General Counsel, NRC
1717 H Street, N. W.
Washington, D. C. 20555

I N D E X

E X H I B I T S

P R O C E E D I N G S

Whereupon,

THOMAS A. IPPOLITO

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

EXAMINATION

BY MR. SIDELL:

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

A My name is Thomas A. Ippolito.

Q And your current position with the NRC.

A I am Chief, Operating Reactor Branch, #3, Division of Operating Reactors, Nuclear Reactor Regulation, NRC.

Q And how long have you been in that position?

A Since June of '78.

Q Were you requested to bring a resume to this deposition?

A Yes, I was.

Q Have you brought one?

A I have put together one hurriedly. You can have it. I don't know how many copies you need. There are four or five there.

Q Okay, we will just keep them I think.

A Okay.

Q Is the information contained in the resume that you have just provided me complete and accurate to the best of

knowledge?

A Yes, it is.

MR. SIDELL: Let's mark this as exhibit one to the desposition.

(The document referred to was marked for identification as exhibit one and was received in evidence.)

BY MR. SIDELL:

Q Mr. Ippolito, have you ever had your deposition taken before on any subject?

A Deposition, that is a legal term that I am not familiar with -- I have appeared before investigating bodies, both internally and externally.

Q Let me briefly tell you what we are going to be doing this morning in terms of this particular deposition. First of all, your testimony is sworn and even though we are sitting in your office in one of the NRC buildings in Bethesda, Maryland, it still has the same force and effect as though it were given in a court of law before a judge or a jury.

Therefore, it is necessary that you be as precise and accurate in your responses to my questions as you can be to avoid any confusion or misunderstanding. If you are uncertain at all as to what my question deals with, ask me for clarification; I will try and explain what I am looking for

in the way of a response. In view of the fact that your testimony will be transcribed after it is recorded, it would be helpful for you to avoid inaudible responses to questions such as nods of the head or hand gestures. Obviously microphones can't record those very well.

A I understand.

Q It is also helpful if you would allow me to finish my questions prior to beginning your responses. I will likewise restrain myself from asking a question until you finish your response since the reporter can't easily pick up two people simultaneously talking.

At the conclusion of the deposition, your testimony will be transcribed and presented to you for your review, correction if you find any need for it, and signature. I must advise you that should you feel there are any substantial changes that are necessary, you may of course make those changes; but we in response will be entitled to comment on those changes which may adversely affect your credibility and therefore again the emphasis now in the first instance to be as precise and accurate as you can be in terms of your responses.

Do you have any questions?

A No.

Q Okay. Could you give me a brief description of your current responsibilities within the NRC?

A Well, as stated in the last page of my resume, I am Chief of Operating Reactors Branch #3 and stated maybe a little bit differently than stated in this resume is that I am responsible to assure that the boiling water operating reactor plants are operating within the regulations; that is including their technical specifications. Any changes that are necessary are processed through this branch.

We also review and comment on their operating experience, their availability, ~~their~~ whatever events take place at those facilities, and we are the primary contact with the licensee.

Q Have you previously had any involvement with PWRs as opposed to BWRs?

A Not as in the project function.

Q Can you explain what that means.

A Yes. I as is indicated in my resume again, is that prior to my present position, I was Chief of the Instrumentation and Control Systems Branch, Division of Systems Safety, NRR; and ~~a~~ in that capacity, I was responsible for the review of instrumentation and control systems for PWRs and BWRs. So I am familiar with the PWR design ~~at that~~, if that is part of your point.

Q And that position was held up until June of 1978?

A Correct.

Q In that position, did you have responsibility over

review of electrical evaluations of PWRs by other NRC staff?

A I don't know if you are phrasing that correctly. Let me say this to you. My branch performed the electrical evaluations of the, during the construction permit review and the operating license reviews, correct, until about a year before I left that position; then there was another branch that was set up headed now Mr. Faust Rosa which is called the Power Systems Branch. In other words, my branch was divided into two segments.

Now starting, I guess, if I left in June of '78 and around June of '77, this ~~Electrical~~ Power Systems Branch was developed. ~~Now I don't want to~~, I want you to understand, that electrical sometimes is used too generically. I am using electrical as meaning power, electrical power, as opposed to instrumentation and control.

Q And you were exclusively involved with electrical as it relates to power rather than control?

A No, initially I was responsible for all of that.

Q Okay.

A Both electrical and instrumentation and control up to about June of '77. At that time, there was a reorganization at which the electrical or power systems portion was set up in this new branch.

Q So prior to June of 1978, did Frank Ashe report to you?

A Yes. I am just trying to be real accurate. My branch, the ICSB branch consisted of ~~two~~ three sections. Each section had a section leader. Excuse me. And Frank worked for one of the section leaders.

Q So he reported to you indirectly.

A Indirectly, correct.

Q Were you involved in any way in the final evaluation for an operating license for TMI-II?

A If you can tell me when they got their license, I will be able to answer that.

Q February 8, 1978.

A So I was there.

Q Do you have any independent recollection as to the reviews that went on at that time?

A No, about the only thing I can tell you about TMI-II, it was an extended review. Meaning that it didn't stay on schedule. It just for various reasons which I don't really have recall, it just extended over a number of years before it finally was completed. That is about all I can tell you.

Q Was that a result of actions taken on the part of the licensee or actions taken on the part of the NRC?

A I would have --

Q Or both?

A I would have to say both, and you have to remember

at that time I guess money became very tight and people -- I am not saying that this was the case at TMI-II. But it was in that period of time when money became very tight and people -- were delaying responses and were not in a hurry, you know, particular hurry, to get to do work on their plants. They just simply delayed them. And that might have been one of the reasons, but I am sure that the way all of these operating license review goes, there is equal culpability on both sides of the fence.

Q. What were some of the factors that caused the delay from the NRC's side?

A I really couldn't tell you. I would have to go back and look at that particular file. And I say this in all candor to you. You know I had ~~we have~~ something like 30 to 40 plants, either construction or operating license reviews; and I am sorry. I can't recite specifics You know I just can't do it. If I had the branch files, I might after a few minutes of thumbing through them, I might be able to tell you more precisely. I can't.

Q Are those files readily available?

A I -- All of my files were left with in that branch. I took no files with me.

Q Is that in this building?

A It is in this building. Yes.

Q Could we make an attempt to get the files so you could refresh your recollection?

A You mean right now?

Q Yes.

A Okay.

(Brief recess)

BY MR. SIDELL:

Q Mr. Ippolito, prior to Three Mile Island of March 1979, were you aware of a transient involving a failed open PORV at Davis-Besse in September 1977?

A I just don't remember it. You say it happened in '77?

Q Yes, September 24, 1977, involving a fail to open PORV among other things.

A I would have to say at this point if I did hear of it, it did not register as something significant ~~to~~ my mind.

Q Well, if that event involved an electrical problem, would your office have been involved in that or at least aware of that?

A Okay, what you are saying is that the event took place in '77. Davis-Besse was, I am assuming, an operating plant at the time. And the problem would come in to the Division of Operating Reactors who has a technical staff, okay.

We have -- and I believe that if it had been something of significance that ~~they feel~~ consideration for checking into other plant designs -- then they would have advised me. DOR had even at that time a system of

information memoranda, I believe they are called; whereby information such as this is transmitted to DSS for consideration ~~and~~ in their reviews ~~during~~ of the CP and OLs. So I think I would have heard of it that way.

I don't know. I don't know offhand whether this was even a subject of an information memo.

Q So unless DOR concluded that the problem might have generic consequences, you would not normally have found out about it.

A That is correct.

Q If you know, does DOR -- or at least at the time did DOR -- have a procedure whereby they would be able to make a determination of a problem as being generic as opposed to plant specific? Or was this left up to an individual who may have reviewed an LER and concluded on the basis of what was contained in the report that it might be generic.

A I would have to say that there is the existence of the information memorandum, ~~and a --~~ . When they were created, I would imagine that they had certain procedures by which you make these determinations. I remember I have read a number of these things since I have been here, but actually to go back in the archives and read what these criteria are; I haven't. They don't directly involve projects. They involve primarily the technical arm of DOR.

Q If there had been a problem at Davis-Besse in

September of 1977 involving a situation where with a failure of electricity the PORV failed open. In other words, it was necessary to have a constant source of electricity to keep the PORV closed, and with any electrical failure, they would fail open. Would that have been a subject of interest to you in your usual reviews?

A The answer to that is probably no. And let me explain. The reason for it is that on these relief valves on a PWR were always considered to be not essential to safety.

Q Not safety related.

A Correct. And as such, I make the assumption that our systems people have evaluated the transients and accidents and also agree with the licensee that these are not important to safety. So therefore their failure, whether open or closed, whatever, was not important to safety so therefore it could have failed in either direction.

Q And the fact that the problem may have been instigated by an electrical problem rather than a mechanical or human type problem would again not have been of substantial concern based on its original definition of non-safety related?

A Correct. And just let me add if it is not safety related, it would not be on the safety buses, if you will; therefore, they are designed to less stringent requirements and I wouldn't use the time, the review time, to investigate

it. That is basically it.

Q So in addition to being designed on a less stringent basis, they are also reviewed on a less stringent basis as well.

A Well I will simply say to you that it is probably not reviewed at all. As long as I am told that the analyses do not require it, then I will not spend the precious review time, being understaffed, I would not spend the review time to look at it. It is just that simple.

Q. Well, you just mentioned that I gather you are currently understaffed in your opinion. Or was that in reference to your prior position --

A Prior position in --

Q -- ICSB.

A That is correct, the prior position in ICSB.

Q At the time you were in ICSB from February '73 to June of '78?

A Correct.

Q Did you make your concerns about the staffing inadequacies known to anyone?

A Let me say this to you. You have to divide that period of time into at least two different eras or periods, and when I first came aboard, there was something like eight reviewers. In a period of two years, it grew to, I had roughly, a staff of 30. There was a need --

(Interruption)

BY MR. SIDELL:

Q You are dealing with staff shortages.

A Correct. At that time, the projected work loads were actively understood by upper management and the byword then was hire what you need to do the job, and we did that, and that is how I built up a staff. Then things became more difficult. I would say especially around the time just ~~when~~ prior to and during the time of this split of my branch; and let's see; I actually prepared a manpower study which indicated -- it had to be around '76, '77 era -- at that time that as a very minimum I was understaffed by about 40 percent or about seven people.

Q That is seven people out of the 30?

A No, no. Excuse me. By that time, my branch was down to I would say eight per section, two sections, about 16 people. A number of the responsibilities I had were ~~remote.~~ removed In other words, I no longer had the electrical review responsibilities. So they took people and that responsibility with them.

So with what was left and knowing what the work load was, the projected work loads, I had a rather detailed manpower study prepared and presented it to management. And that was it.

Q Did you get a formal response from management to

your study?

A No.

Q Did the study expressly request additional people?

A Yes.

Q And management failed to respond to that request.

A You asked "formally respond." And the answer is I really don't believe they formally responded to that. I, my recollection is that ~~this~~ they did not. Obviously I didn't let the thing die. I think that the budgetary constraints were such that there were just not enough slots to go around to satisfy what the needs were.

I know that my needs were unequivocally made known, and I was told that, my recollection is that I was told, there is just no room for that type of growth.

Q Do you recall who told you that?

A I believe it was Mr. Heinemann who was director of Systems Safety, Roger Matson's predecessor.

Q Now you mentioned that that was not a formal reply to your request for additional staff. Was this merely in the course of a conversation that he indicated this to you, or was there anything written?

A I don't recall anything ever being written with regard to that study. I remember that we had numerous meetings on how to handle the fact that we were -- not only myself, other branches as well -- felt that they were

understaffed, and there were numerous meetings and discussions on how best to utilize our staff to assure an adequate review.

And there were many, many discussions during that period.

Q Can we get a copy of your staff evaluation?

A I doubt that one even exists anymore.

Q If one were to still exist, where would it be?

A I turned it in to Mr. Heinemann. It wasn't something that I did and with pink, yellow, green, you know, all that stuff. I didn't do that. It was just a note from me to Bob Heinemann through Bob Zecheser saying, "Look, I have done this. This is what I need. And here is the data." It was a report I would say about three-quarters of an inch thick.

Q Did you make a copy of that for yourself, for your own records?

A If there was a copy or anything, I left it in my old branch.

Q Would it be possible --

A If you let me make a phone call, I had -- I directed this to be done, and I outlined what I wanted, and I had -- a fellow by the name of Ray Scholl do the background and writing the report for me, and I edited it and sent it out to Bob Heinemann.

MR. SIDELL: Off the record.

(Brief recess)

BY MR. SIDELL:

Q Now earlier, Mr. Ippolito, you indicated that the PORV was not previously defined as safety related. Is that correct?

A Yes.

Q Do you know why it was not considered to be a safety related piece of the primary system?

A Not specifically. I think that I made the assumptions that a systems evaluation was performed and that its failure was not significant to safety. I didn't personally pursue that analysis or review.

Q If the failure of the PORV was not a concern in a safety sense, would that have been because there was an alternative mechanism that could be employed to serve the same purpose as the PORV, for instance the block valve?

A I would say, ~~you know~~ I don't know how much of this is hindsight, ~~you know~~ because TMI has forced us to focus a little better. But I would guess that we have the pressurizer heaters, okay. We have ~~got~~ the quench tank with its blow-out patch, and ~~I guess in the long~~ -- we have high pressure injection, whether it is for make-up or safety; we had those systems to control the event. So I guess -- I am guessing.

Let me say I would have -- if I were to ~~maybe~~ think back at what the attitude might have been then is that, ~~hey~~ it might have been messy as all hell, ~~but~~ we would have

controlled it one way or another. If not, we could have just turned the block valve or start safety injection or high pressure injection or whatever, however the system is designed. We would have taken care of it.

We would have come down and that was it. I would guess that that was probably the thinking at the time.

Q So you are unaware whether the determination to define a PORV is non-safety related was based on the fact that there was a block valve downstream or at least between the PORV and the pressurizer that was available to essentially serve the same purpose?

A No, that is not what I am saying. I am saying to you had it -- . If I ~~was~~ were a system type and had to perform the analysis on it, I guess what I am saying is that they probably did not care one iota if that thing stuck open or if it failed to open because we had, ~~you know~~ (as I identified to you earlier) two ways or three ways that would ordinarily ~~at that time would~~ have been considered more than enough to take care of it.

Q One of which was the block valve availability.

A Correct, yes. They would have said, "Well, if that didn't work, then this; and if that didn't work, ~~I have had~~ and -- if it went over pressure, I could have -- blown out the patch in the quench tank, you know, so what. If it were relieved that way, it would have been messy, ~~but it wouldn't~~ --

you know, it is something that was not at that time considered to lead to significant hazards concern.

Q So you are concluding that the consequences of a nonfunctional PORV were evaluated and that based on the consequences and the alternative methods by which the plant could be brought to a safe shutdown were such that the PORV was considered relatively inconsequential element of the primary system.

A Let me make it clear. That is my conjecture, my opinion. What I am saying that, you know, is that this could have been what our systems people were thinking and had done at the time. I am not sure.

Q Were you aware pre-TMI-II of another incident at TMI-II on March 29, 1978, almost a year before the accident we are now investigating; of a problem with the electronic curcuitry producing the PORV to fail open with an electrical failure?

(Brief recess)

WITNESS: The answer to this question is I don't recall but remember that it would have involved Frank Ashe and Faust Rosa and not me during that period of time.

So again unless it was something that Faust would have considered important enough for me to know about, I guess he would have done something more positively. But I don't recall a memo from him saying that it had that type of

importance. So how he handled it, I have no idea.

BY MR. SIDELL:

Q If this were the second occurrence of the same problem dealing with an electrical setup where a loss of current would cause the PORV to fail open, would you consider that to be of generic consequences for other B&W reactors?

A I doubt it. Let me tell you why. ~~The plants are --~~ As much as we would like to say that ~~the plants~~ they are identical, the most we can say is that -- the plants are only similar in design. In other words, within any nuclear steam supplier, all you can say is that they are similar. You can never say they are anywhere near identical.

~~And like I said earlier, when you consider something not safety related, okay, I would not I guess since I would have to say that all of these plants -- and there was one other factor. Normally this area is what we call balance of plant design. Usually the architect-engineer designs this portion of the plant, okay?~~

So therefore I would almost have to say that I would not consider it generic. But again it was non-safety, and I don't know whether I would have focused on it. I think I probably would not have focused on it.

Q What about the fact that essentially we have an electrical failure producing a small break LOCA. Would that have attained more significance?

A I believe the answer to that is unless the systems

people were to come to say that we will not, we cannot, tolerate this happening, we are going to recategorize that as a safety system, and we want it to be designed correctly; then the answer is probably no.

In other words, the systems people must recognize that their analysis of both transients and accidents have, they have reconsidered it and reanalyzed it or whatever; and now they are removing this component in system from a non-safety related system to a safety related system and that my reviews should be so modified.

And basically that is it. I think that informally I might pick up the phone and say, talk to the systems branch chief and say, "Hey, do you like this?" I probably would have come from the direction ~~areas~~ that I don't think it is good for plants to go through many transients of this type. It is not good for them. I think that you want to avoid everytime you can to shut these plants down, really.

Q Why?

A Because you don't like to take the plant through transients, and when you scram or something like that, that is a real mild transient. But you know it is just something that you just don't like to see ~~it~~ happen. It would be nice if you never had to scram a plant, if you could always bring it down gently, you know, through, right on through to cold shutdown. That is always the best way to treat any piece of

machinery.

You don't want to abruptly stop something, whether it is mechanical or electrical; you just don't want to ~~tear~~ Put on the brakes just like on a car. You wear out your brakes much faster and things like that. That is all I am saying.

Q So your concern is from a -- (note: Something appears missing)

A Do I think? You would ask do I think that they are too frequent and that they should, we should ask for something more? I might approach it as far as that is concerned, but that is about all.

Q Well, you are concerned with transients at least in terms of interrupting the constant production of power. Do I take it that --

A Only as it relates to safety. I don't -- I hope it is understood in the context I mean it. I really care very little about whether an electron of electricity is produced. I mean when I am wearing my NRC hat, okay? When I ~~relate pair~~ am a rate payer (?), that is something else. But when I wear my NRC hat, that is not my primary concern.

Is it safe enough? Okay? I think it is in that context I am speaking. ~~Not whether -- and~~ What I am saying is that intuitively I feel that these abrupt changes ~~is~~ are not good for the machine. So therefore they should be avoided.

Q Well, when you say not good for the machine, is that not good for the machine in the sense that if you break

a pipe or damage a valve or something during one of these shutdowns, you could produce safety consequences?

A We are approaching an area in which I am not an expertise, so what I am trying to say to you: I know we have Appendix G requirements that effectively say bring the reactor down at 100 degrees per hour, or whatever it is. Why? Because as I understand it, ~~is that what you want, you want to~~; you don't want that metal to go through a severe change quickly. You want to bring it down gently.

In other words, you (don't want to) ~~are~~ stress~~ing~~ the material. And if they bring it down the way it is supposed to, then we have got something that (good for) the life of the plant. ~~If you start~~, If you shock it frequently, you may be going into that margin, okay?

Now that is all. That is the only context in which I mean what I am saying.

Q Well, would you consider the problem where you have a small break LOCA induced by an electrical wiring problem to be something that could be easily correct?

A Oh, no doubt.

Q And therefore you would avoid needless transients.

A Correct. But again, please. The only thing I want to add is that we are regulators and as a regulator, we are to assure public health and safety. And unless it ~~that~~ is determined to be a safety matter, I would think -- and you as

a lawyer probably know better than I -- if I were to say, "Mr. Licensee, improve the design of that non-safety system," you know what he will probably tell me. You know, and I think it is as simple as that.

Q As a regulator?

A Pardon?

Q As a regulator, you would expect a less than --

A No, he could say that.

Q -- polite response from --

A He could say that. In other words, I have absolutely, I have very little to say as far as the balance of plant is concerned.

Q Well, as an alternative --

A And to enforce regulations, that is all I am saying.

Q As an alternative wearing your NRC hat, would it not be possible to reclassify the non-safety related item as safety related should you have a concern that it might possibly in a lesser extent that otherwise produce safety related consequences such as the electrical wiring of a PORV causing a small break LOCA?

A Let me ask. Are you asking should I have had that reclassified?

Q Not you personally but in a general sense as an NRC regulator.

A Yes, I think that if one examines the frequency of

events of some what was considered ~~to~~ have a minor perturbation on the system and finds that either the rate or the severity is different than originally thought, he should reconsider it. Definitely.

And I think I could only think of one example that immediately which comes to mind is the over-pressurization concerns that we had about a year or two, three ago. Where the primary systems was being subjected to over-pressurization particularly during ~~repair~~ shut down periods.

And it happened once, and we said, "Well, it is just a fluke," and it happened a few more times; and we were seeing that there was a history of this type of thing.

There has never been a problem, a safety problem, resulting from these events happening. But you could see the frequency increasing. And again this happened to be a materials problem ~~again;~~ over-pressurizing the vessel ~~it cold,~~ in a cold condition, is not good for the metal.

So I what am saying in that ~~there~~ it was recognized and something was done about it. Okay? And I am saying it has to be that type of thing where one says, "Hey, wait a minute. This is not good. We don't want to do this often."

Q Are the circumstances you just referred to those that produced the NRC response to licensees to avoid going solid in start-up procedures and therefore one of the problems to avoided would be brittle fractures in the primary system?

A Remember where I come from. I think you are saying is this the bulletin or whatever --. Yes, I think that was it.

Q Do you recall how many specific incidents existed prior to the issuance of the notice or bulletin or letter?

A It is not specifically my area. I mean it ~~is~~ wasn't, you know, and I would not have focused on it.

Q Would you know the general number? Are we talking about one or two or three or something on the order of 10 or 15 incidents?

A I will give you a poor guess; I think it is more like the second, 10 to 15 range.

Q So if you had a situation where you had five problems out of 50 --

A I am sorry. Finish.

Q If you had five problems out of 150 uses or occurrences of a particular part or mechanism, you would consider that not to be of a significant nature. Is that correct?

A There are no criteria or a ~~by staple~~ written if you will that says it is okay up to this point; at this point, trip. No.

Again, each failure is looked upon as, you know, so what? So what not in the non-safety thing. We mean safety-wise, so what? You know, let's see what happens ~~ed~~ if it failed? Why type of redundancy do we have? I mean do we

have, if it is a non-safety system (that fails), are safety systems sufficient to be able to mitigate against its consequences. If the answers are "yes," that is it.

And I think that type of thinking involves: ~~Now~~ Not only what is the magic number (as such) I think you have to look at what the consequences are. I think that people looked at the over-pressurization event by saying, "Hey, you know, we don't like what this might be doing to the metal and we don't like blowing out this patch in the pressurizer ~~dump~~ quench tank," and that type of thing, "We believe we have had enough."

If that is the case, then I believe that I would guess that that is the decision that was made. Now exactly what and how and whether 10 is enough or 12 is enough; I think it clearly depends upon what are the consequences to safety.

Q Well, are you aware relative to TMI-II in March 1979 of a PORV failure that went unnoticed for some two hours and 20 minutes?

A Yes.

Q And at least in part the consequences of that failure were of a rather significant nature, were they not?

A The event, the whole sequence of events. You see what I am saying. I am saying that there were prior to this time others, and they were nothing more than a tickle on the plant.

Thank you. I will look at this.

(Brief recess)

(The record was read by the reporter.)

end
tape 1

(Witness continues).

What I mean by "tickle on the system" is that they have happened in the past and the systems that were installed to mitigate against its consequences were available and proved to be satisfactory to make this event uneventful.

There were a number of other things done at TMI-II in the sequence of things that made it come out the way it did.

BY MR. SIDELL:

Q. Well, dealing for the moment just with the PORV problem failing open in TMI-II, and as it existed at a few other plants preceding TMI-II in March of 1979, were the prior incidents that you referred to, just now, properly corrected at least in part on the basis of an operator being able to determine conclusively that he had a problem when this PORV failed to open?

A You are asking a question that I just don't know about. What was actually done on a non-safety system--it probably was not even reported. Probably, I am just guessing now. And truthfully I have absolutely no idea how many of these things there were.

You know, whether just the ones that Davis-Besse and here at TMI, I don't know; I really don't know the frequency of this type of thing happening on a PWR. I just don't know.

Q Well, generally speaking, even if there is a failure of the PORV to operate as it is intended, and based on its

non-safety related definition, classification; it doesn't really provide too great a strain on the system so long as the operator can take the proper reactions to a failed open PORV, does it?

A Again, I would say, in my opinion, the answer is it does not result in a severe strain. Remember, my background is instrumentation and control and not thermohydraulics or metallurgy.

Q Well, the instrumentation aspect is my question. The operator, by being able to properly recognized the PORV failed open situaion, has to rely on indirect or somewhat indirect information to tell him he does have a problem with the PORV; not at the moment dealing with the specific indication, but he has to rely on something to know that he has a problem in order to allow him to take further corrective action.

A To answer your question, the answer is obviously "yes". You left it so broad and general that I could say to you there are probably half a dozen ways he is able to determine that he has got a problem inside containment.

Q With the PORV failed open?

A Yes. I would think he could probably look at his containment pressure read-outs and see them rising. He could probably determine that he has to add feed-water--not feed-water, but make-up on a PWR.

Q After what period of time?

A Whatever. He could probably tell by the noise that he has ~~got~~ a problem inside containment. I am not saying that all of these things will tell him that hey, that particular valve is blowing down. I am not saying that.

But, he knows that he has ~~got~~ a problem in there.

Q Do you know what methods are available for an operator to determine there is a problem with precisely the PORV?

A The answer is ~~is~~ that even for non-safety systems, there would be a position-indicating device of some form on the valves indicating whether it is open or closed. THis would be my immediate response.

Q Would the position indication methods you just referred to be direct position or indirect position, as a general rule?

A Tell me what a direct and indirect position is?

Q For instance, with a PORV, a direct position indicator would indicate the actual position of the valve rather than the indirect indication of merely indicating a signal being sent to an energy source to energize the valve itself. Is the distinction clear?

A Yes. Let me answer it this way. On a non-safety system, you probably will find both methods used.

Q On the same?

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A No, not on the same. I am speaking generally, now. This is simply,in answer to your question, is that probably on a non-safety system that you will find instances where they just monitor the current to a solenoid rather than whether the piston or the gate on the valve--~~its~~ in the open or closed position.

As far as safety systems are concerned, we should be monitoring directly the position of the valve; in other words, ~~from its shaft,~~ knowing what position its shaft is in. THat is the way I would answer it.

For TMI, I take it they had a current monitor, an indirect method, rather than a direct methods.

Q So in any event, you would conclude it would be necessary to have at least, or as a minimum and indirect indication, relative to no indication at all for a non-safety related item?

A Yes, definitely. Again, in this case, on these systems type of valves , since they do relate to the primary ~~contain-~~ ~~ment~~, okay? I would say yes to your question. But I am not saying that all non-safety stuff inside the containment needs to be this good, if you will.

If it is purely economic in nature, I don't care. But as an engineer, it is just a fundamental concept of all instrument engineers that you try to measure"directly"what you want to know about. That is just fundamental. For various reasons, people compromise on that fundamental criteria

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if you will.

Q So if you had a problem with a non-safety related element in the primary system, not some other place in containment, but which did not have any form of indication as to what the position was--no direct, no indirect, nothing. That might provide the operator with a bit of a problem in terms of the types of actions he would take to correct the transient situation.

A Yes, very, very true. And that is why the standard review plan does require that, and a review or attempts to provide and review those instruments which are required by the operator to mitigate against accidents and transients. They are to provide certain information to the operator.

Now again, these are transients and accidents; transients that have been evaluated that need protection, that the protection systems are installed to insure that either fuel-design limits or LOCA's are taken care of.

Your question, though, was much broader than that and I am saying to you again, one looks at the consequences of these things and either recategorizes the safety system, therefore requiring the most stringent design criteria or else he reaffirms that it is not needed.

~~You just don't look at it, and as far as safety is concerned, demand should not be blind. But administratively or power production, whatever have you, if he doesn't provide~~

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~~it, that is where he will suffer.~~

Q Based on TMI-II this year, apparently the conclusion has been reached that an indirect method of indication for the PORV position is unsatisfactory, is that correct?

A ~~The lessons learned,~~ If requirement that is listed in the lessons learned, report ~~that I am saying that probably,~~ I think the Commission will bless that and it probably will be enforced. I remember there are 23 items; whether that one was one--

Q I believe that that was one of the suggestions.

A I would say the likelihood of that being approved is unquestioned.

Q So that looking back with hindsight, it appears as though indirect methods of indications for the PORV are unacceptable, in light of TMI-II of this year?

A I wish, and I don't mean to preach to you, I wish we don't try to take this problem and try to piece it up in a manner where we lose track of the total picture. Now what I am saying is that I don't think there is any doubt, again I have not been directly involved, but there is no doubt in my mind that that man knew he had a hole in his primary system.

This hole, knowing the PWR, the hole was probably the PORV. I think it is the whole sequence of things that made it bad.

Q But at least in part, had the operator been able to

conclusively determine the position of the PORV at an earlier time than it actually happened, it might have mitigated the consequences of the act.

A I just got this on Wednesday.

Q I haven't had the I and E report within a few months.

A This now again goes to the sequence of events. Now it is blessed. This is what we, the NRC, believes were the exact sequence of events. I would say that my opinion before reading this again is: where I come from, I believe that if he hadn't had the aux- feed system (and interfered with Safety Injection) switches off, you would have never known about TMI-II.

I firmly believe that. Now I might change my mind when I read this ~~book~~ I&E report. Does that answer your question? In other words, I think it was not the fact whether or not (as to) he knew, exactly when did he know or whatever have you. I think the system would have responded as designed, had the aux-feed systems (and safety injection) been available, the auxiliary feed-water system.

He could have taken as much time as he wanted to determine where did the hole come from if he were confused as to exactly where the hole was.

Q To the extent that the same problem occurred in other plants pre-TMI-II and an indirect method of indication was felt to be an adequate solution of the problem: with

hindsight, an indirect indication was inadequate, was it not?

A Correct. My whole life I have been an I and C type and yes, I would not personally design it that way because I feel I am cheating whoever I am providing the information to. Probably I am cheating somebody some amount. Now whether that is significant or not I feel--you know, you try not to do it.

Q Let me try it this way. Do you know what is required to put in a direct position indication for a PORV? Are we talking about a relatively minor adjustment or are we talking about a substantial major costly change in the system?

A I guess I could answer that question if I knew exactly how those valves are designed. ~~If it is a~~, just off-hand, I would have to say that I cannot believe that they will add much money to the thing.

I think that it adds, in fact, a very modest amount of money to do it directly.

Q What are the terms of money you are referring to as modest?

A Excuse me, I have been away from buying and purchasing for so long--

Q Give a ballpark figure.

A What I am trying to you is I don't know what a

position indicator

qualified--(qualified meaning one that you are assured can operate at a temperature, pressure and whatever have you of that environment)--~~what it~~ costs today. Just a limit switch, ~~if you will,~~ that you could mount, ~~like~~ on the valve stem, with some fixed, hard limits on it: ~~when it reaches this limit you know it is open; when it reaches that limit you know it is closed.~~

That is direct, because that shaft is moving. It cost ~~Something like that,~~ that limit switch, I don't know, but I can't believe they are more than about a couple of hundred dollars. Then there is the pedigree that goes with it, ~~that~~ the cable, cable routing ~~You have to go~~ through a containment penetration ~~in routing it~~ and up to the control in some way.

I don't know --I don't know what labor costs are; I just don't know anymore.

Q On the order of something less than a few thousand dollars total?

A Yes, I would say so, yes, per valve.

Q Given the fact that we are talking about a relatively inexpensive mechanism to install, to purchase and install, the means of a direct position indicator for PORV, balanced against what we know now are substantial consequences for operator not-knowing about the precise positions of the valve for a period of time, and the potential safety-related consequences; an operator's ignorance as to where the valve

itself is. Would it not be your opinion that it is better to err on the side of completeness and put the actual position indication in?

A Let me answer it this way. I am going to ignore for the moment your qualifier and answer the question "yes", it is best to directly monitor the position of the valve. The qualifier, again, I have to go to my previous answer to that qualifier, and that is it wasn't alone his knowledge of where that valve is, it is the fact that other things had taken place or were in effect at the time that made this thing get away from him.

I am saying ~~within~~, if one wants to look at the little contributors to this whole thing, yes, you could eliminate that contributor; it is cheap to do; do it. That is my recommendation.

Q So then, if there was a prior problem with the PORV, having no indication at all, and therefore the operator having to rely on completely indirect indications, such as reactor temperature and pressure or quench tank indicators to determine the position of the PORV, your solution would have been to put in an actual position rather than an indirect position indication for the PORV, would that be correct?

A Yes.

Q Were you aware in March of 1978 the situation at TMI-II involving loss of pressurizer, low indication off the

low end of the scale? April 23, 1978?

A I don't recall it. Again, it was not in my branch.

Q Were you in any way concerned with the pressurizer level indication as it provided information to the operator about core inventory?

A Okay. You realize the subject of PWR primary vessel water level instrumentation dates back a very, very long time ago ~~again~~. I would say in the late 60's, and early 70's. We were wrestling with the thought: should we or should we not? The reason why I am aware of it is that as an instrument type, I was a very, very interested bystander to this analysis of whether or not it is needed for safety.

My recollection is there were meetings specifically on that point. There has been correspondence with B and W on that point, and there may have been discussions at the ACRS in about that same period of time, ~~that discussed this.~~

But I think yes, we thought about it. All I could tell you is that apparently, since it was never made a requirement to monitor the vessel, a PWR vessel water level directly, that the end result of that is that it was concluded ~~whatever~~, in the early 70's possibly, that it was not necessary.

Q Are you aware of the reliance an operator places on pressurizer level indication in the operation of a reactor, a PWR?

A Yes, I am aware of it. I think that is particularly one of the major reasons why ~~you have~~ a plant designed ~~like, a nuclear steam supplier like~~ by Westinghouse who ~~they~~ think so highly of their control system that it is designed to receive extremely good information, very reliable information, ~~why they use~~ from the same sensors that they use to provide protection system actuation signals, ~~to feed into their control system.~~

It ~~That~~ is their belief that that is how important they are. And to the extent that you can monitor and know the level; you can mitigate, you can mollify, ~~rather,~~ operational transients so that it precludes the necessity to even actuate or bring into force your protection system.

Q So the more accurate the pressurizer level indication, the better the ---

A The more responsive to system performance, and that includes accuracy.

Q Would system performance also include abnormal events as well as normal operating position?

A Yes. Wait a minute, say that again.

Q Would system performance, as you just used it, also include abnormal operating conditions as well as normal operating conditions?

A The system's performance as I meant it was that if you can accurately monitor and control your reactor with highly reliable instruments, you are able to then maintain your

reactor in this norm position and therefore preclude things like abnormal transients, ~~or transients.~~ See, that is the whole objective.

Except for those transients possibly that could result in fuel damage, all of the other, let me call them operational transients, ~~that~~ is the reason why you put in a control system, ~~That is one of the objectives of a control system, is to keep that thing so that you keep it out of this transient condition, because it is a mild upsetter, or something; it affects operability and things like that.~~

~~That is why you want the control system to keep you within a certain operational band so that you don't shoot off into these other areas.~~

Q If we assume, that for whatever reason, the reactor incurs a transient of one form or another, you would want, for instance, a pressurizer level indication that would be able to give you accurate information to get you back on to a normal or steady state.

A Sure, yes.

Q Do you know whether or not pressurizer level indicators, in terms of B&W reactors, are designed to operate in abnormal condition as accurately as normal conditions?

A Let me say it as follows: I am a little fuzzy right now, but I don't believe that the B&W plant has the same design concept as the Westinghouse plant in that they have a separate

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set of level sensors, I believe. You know, right off hand, I wish I had the diagram in front of me.

Let me say this to you, if B&W uses separate sensors for protection and control, then I would say I don't know the pedigree of the control signals to that. If they do as Westinghouse does, it is just fuzzy in my mind, then I would say we would have required, that our requirements are that those sensors be qualified to be able to operate during normal operation, transient, and accident conditions, whatever accuracy is demanded by the analyses.

In other words, they have to have somewhere, either in our shop here in the NRC or in the QC records, a record of its qualification to be able to behave correctly.

Q Alternatively, with a minimum degree of accuracy across the board, regardless of conditions of operations?

A No, you can't, it is impossible to. I know of no level-sensing technique of reactors that will give you the same degree of accuracy at normal operation as it would at cold shut-down, as it would at under a frothing depressurization. It is going to suffer inaccuracy, okay?

Now what is really important is that when you find that you are losing level, that it is the rate at which you lose it and where it is. If you are losing mildly, like just above the capacity of the high head pumps, that instrument is fairly accurate. But if you have a rapid depressurization

like a double ended rupture, it is not going to be as accurate. I don't know. It has been a long time since I looked at a test report, but I would guess 20, 30 percent off. But, hell, that thing is going down so fast that that is not significant.

~~Okay, so I am saying to you, it depends how you are saying. The phenomena is density of water; most of these devices are designed to determine what the density of water is and so indicate. If you are in a rapid transient, it is not going to see that density as fast as it would ordinarily.~~

~~Do I make~~ stat → make myself clear?

Q Yes. Are you aware of the fact that operators at least in the first instance rely primarily on pressurizer low indication to determine core inventory?

A No. I would have guessed--I have not personally reviewed the training of an operator and what he is counted on to use ~~in his,~~ as the prime signal for him to use to perform actions. That I am not aware of, I just don't know. It could very likely be. I would guess in a BWR especially, you know, that would be the case.

But, in the PWR, I am not sure. I would think that in a PWR, pressure and hot and cold leg temperatures would be another very good measure of doing it.

Q Would it be as convenient from an operator's perspective to rely on pressure as it would on pressurizer low indication?

A As convenient?

Q Would it be as easy for the operator to determine what the system pressure was as it would be to rely on pressurizer level indication to determine core inventory?

A Now you added the magic words. No, pressure is not, in my opinion, a good indication of water level, no. All you have to do is just heat up the water a little bit and increase the pressure and you can have a water level way down, ~~something~~ as long as it sees a certain pressure in the primary system, it doesn't know how much water is in there.

Q So, pressurizer level indication is a pretty important method for the operator to determine or assess core inventory in the first inst e?

A Yes, I think that he could, I hate to say this, but he has a whole wealth of other things like leak detectors, leak detection systems and he has an idea about what he is losing because he has to make it up from time to time. ~~That gives him an alive, you might say.~~

He pretty much knows what he is losing because of ~~a seal,~~ a minor leak in a seal ~~here and a minor leak in a seal~~--he pretty much knows what his losses are and I can't help but feel if he sees that rise to any extent, an abrupt change, because you know we can see things down to five GPMs and stuff like that--at least on the boilers we can, he will be able to take corrective action.

The PWRs have a much larger containment, so I am

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not too sure about what their capabilities are in that regard. There is another thing-- if you are perking along like at one or two GPM leakage and all of a sudden this thing goes up five, ten, fifteen, twenty GPM, there is something wrong in there.

You getting to the point where you are going to have to add inventory quickly. YOu know there is something wrong. Now you watch it awhile and determine what to do from that point on. So I am saying that here is another excellent way of monitoring your plant (phonetic) continuously.

Q Do you know whether or not there were any operating conditions informing operators of PWRs, what to do in case they had lost level indication in the pressurizer, in order to assess core inventory?

A I don't know.

Q Were you aware of a November 1978 transient occurring at TMI-II with loss of pressurizer level indication off the low end of the scale?

A You asked me that previously and I said no.

Q This is a new date. November 7, 1978.

A Oh, no I wouldn't know. I was down here at the time.

Q Do you know whether or not TMI-II was eventually grandfathered relative to the standard review plan?

A I don't know. I think really this area right here,

should be addressed to,

I hate to throw another name in the pocket, ~~but~~ either Frank Ashe or Faust Rosa, because these ~~things~~ plants initially came through as Unit I and II, but then the reviews were broken off and Unit I was completed and they got a license and then Unit II perked along after that.

Exactly how much more was done on Unit II than on Unit I, I don't know. But rest assured, sure as I am standing here even though I had nothing to do with it at the time, rest assured that more was done on II than was done on I.

That is just the way it happens to be.

Q Pre TMI-II of this year, were you aware that another Davis-Besse occurred in November of 1977 involving loss of pressurizer level indication off the low end of the scale?

A No.

Q Pre-TMI-II , were you aware of something referred to as the Michelson Report?

A No.

Q Have you heard of it since then?

A Yes.

Q Have you read it?

A No. I haven't seen it.

Q How have you heard about it?

A Through discussions and we recently provided

information to Roger Mattson. I believe, my recollection is that the Roger Mattson has been assigned to respond to the Michelson report. Information was provided to him in the form of a memorandum from Darrel Eisenhut which said that in the preparation of your response on the Michelson report, we offer following.

We talked about ~~the boilers and~~ the boiling water reactors. I believe the memo only addressed the boiling water reactors, I believe.

Q The Michelson memo only addressed--

A No, no, our memo to Mattson.

Q What specifics did you response memo include? Do you have a copy available?

(Pause) Okay.

A Memo to Mattson on Michelson.

Q Why don't we get rid of some paperwork at the time. Your management study, enclosure two, I guess, is the available copy that I have. Somebody is in the process of providing a copy of the first part of the--

A I had not done that. I thought we agreed that we would, that something that thick would have to be sent downstairs. It would take a day or so to get. So I am saying, I have a record that you want this, I have a record that you want copies of this, I guess I will have to get somebody--

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MS. NORDLINGER: Is there a xerox on this floor? The first few pages of this could be xeroxed immediatley so that you could consult them at the same time as Gary asks you questions about them? I would like to proceed--

MR. SIDELL: The management study I merely want for an exhibit at this point. We are done with the questions. The notes with reference to TMI-II were provided as a basis for finding out some more information about TMI-II, which if we tried anything, we may have to have copies of that admitted in terms of exhibits.

At this point, I am merely concerned with getting a copy of this as an exhibit. So, I am not concerned if we have it today or tomorrow. If we don't have it today, we could have a stipulation that would be included as an exhibit to be number two to this deposition. That is fine.

MS. NORDLINGER: The first part of the memo, do you feel the same way about that?

MR. SIDELL: Sure.

MS. NORDLINGER: I thought you wanted to ask some questions about it now.

MR. SIDELL:: Okay, so the manpower study for instrumentation and control systems branch for 1977 and 1978 will be exhibit two to the deposition when we receive the complete copy, which has a cover memorandum for Robert Tedesco, Assistant Director for Plant Systems, DSS, from

Thomas A. Ippolito, Chief Instrumentation and Control Systems Branch, also DSS; Subject: Manpower Requirements for ICSB.

(The document referred to was marked for identification as Exhibit 2 and was received in evidence.)

BY MR. SIDELL:

Q Can you recall whether or not your response to the Michelson Memorandum deals with pressurizer level indication capabilities?

A Let me say, I don't have total recall of everything in that memo. I know it addressed BWRs and BWR level instrumentation because that is what I focused on as BWR branch chief. Now I just don't have recall. We will have that in a few minutes.

Q Let me ask you whether or not BWRs have direct core level indication?

A Yes.

Q Are there any indicators that would show voiding in the core of a BWR?

A That is the way it operates. It is boiling. You will see bubbles going through it. That is what it is supposed to do. ~~It boils in a boiler.~~ It is boiling water in the reactor vessel itself and the steam comes right off the top of it and into steam lines to the ~~condensor.~~ It has no pressurizer; it has no steam generator.

I didn't mean to demean you by describing the system to you.

Q That is perfectly alright. Let me ask you, in terms of voids in the system, I was referring to voids not at the top but within the system itself, below the top, where you might not necessarily want them. Or, is that where you want them?

A Are you speaking of void in the context of the void that we had in the vessel at TMI-II?

Q Yes.

A The answer to that is no, because the vessel--first of all, the steam lines are at the top of the vessel. That is where the natural flow is--out that way. You pour water in, steam is generated, and it goes out. That is just the way the beast behaves. Offhand, I can't conceive of a way in which you could have a void.

MR. SIDELL: Off the record. (Off the record)

end of tape 2

BY MR. SIDELL:

Q During an off the record discussion, we have found out a little bit more about how a BWR plant operates, and I believe you indicated, Mr. Ippolito, that BWR's have direct measurements of core level of water. Is that correct?

A That's correct.

Q And have they had them for some period of time?

A I would say without doubt, since I came aboard, everything from Oyster Creek on, through current designs, all have this method of measurement of liquid in the -- or water in the core.

Q And are you referring --

A In the vessel.

Q Are you referring to 1967, when you began working for --

A Yes.

Q -- the NRC?

A Right. No -- the reason why I'm hesitating -- there are some, what they call old boilers, like Big Rock Point, ~~and things~~ -- I'm not sure what they have in them -- only because I've never had an opportunity to look at those older plants -- designs. That's the only reason why I say it.

Q But generally, direct measurement of core inventory in a BWR has been available for quite some time?

A Correct.

Q Do you know why there is no direct core inventory

measurement in a PWR, comparable to that in a BWR?

A Well, okay -- I see you qualified it. You said -- I guess you mean in a vessel.

Q Direct measurement --

A Yes, of water level in the PWR vessel. I think it was determined that it was not necessary. Now, this is -- I said I think, and I mean it's just conjecture on my part -- is that the primary system of a ~~boiler~~ PWR is a closed loop, and need only to be replenished as -- normally replenished with primary coolant, as leakage, minor leakage, dictates.

So, therefore, being, let's say, a closed system, running with a large volume of water in the pressurizer, that it was thought sufficient to monitor -- I might add, directly monitor pressurizer level, as sufficient to determine primary system inventory.

Q Are you aware of whether or not the decision to have, in PWR's, only an indirect measurement, by way of the pressurizer level indicator, was an NRC decision, or was that a steam supplier decision, if you know?

A I don't know.

Q Are you aware, pre-TMI, of something referred to as a Novak memorandum?

A No.

Q Since TMI, are you aware of the Novak memorandum?

A No.

Q The Israel memorandum?

A No.

Q Did you have any personal involvement after the TMI-2 accident of this year?

A No. Oh, wait a minute -- one phone call.

Q To whom, or from whom?

A I don't recall whether I called Dominic Tondi, or he called me. Mr. Tondi was at the Incident Control Center at the time, and we were speaking of water level instrumentations, and --

Q In the core?

A Water level instrumentation, period. And I think what I was doing ~~is that I~~ was assuring myself that the boilers were okay, in fact, and I couldn't understand why people were saying some of the things I heard. You know, this was a panic situation, and I just provided Dominic with -- we talked a little while, and I gave him my opinion, as to, you know -- but at the time that I talked, I had -- I didn't know that the water level had dropped to where it dropped, okay, so what -- I guess that was it. That was the conversation.

Q Do you recall when this occurred?

A Gee, no, it was --

Q Was it early on in the event?

A I would say early on -- I would say probably within

the first few days.

Q Before the weekend? The event occurred on a Wednesday morning.

A I just don't recall.

Q Okay. Did you find out any information as a result of your phone call as to exactly what the expected or determined water level at TMI was?

A No.

Q Well, you indicated that --

A We were not -- at that point, we were not even sure -- I mean, I was not aware of a void, okay, so therefore I was confused as to, ~~you know~~, what ~~is~~ the problem was with the water level instrumentation? ~~I said -- you know, I said, sometimes this is a did~~ I asked whether they install it incorrectly, ~~did they have a~~ -- didn't they have this, and didn't they have that? Was it far enough away from the heat source, the pressurizer, so that it wasn't measuring accurately ~~directly~~?

I mean, all of these fundamental things that one normally does in hocking up a level instrument -- we were -- I mean, I was trying to get this information, I was trying to understand, at the time. And I would have to say that before we found out there was voiding in the thing -- is the point of where I was asking questions, because I was confused. ~~And the type of thing I did was -- in the boilers.~~ I was trying to find out, ~~because I was trying to do~~ whether I had

a problem with my boilers? And I confirmed with some telephone calls, ~~you know~~, that my recollection was accurate -- there had been test reports indicating, ~~you know~~, of the performance of these level instruments under rapid depressurization, things like that, that had been considered a long time ago -- and I say, in the '67-- '68 area, as far as the boilers were concerned.

So they had run through tests and things like that, and knowing the design as it is today, I just simply satisfied myself that unless something strange that I didn't understand was taking place, that I believed immediately that the boilers were okay.

Q Did you get any information during this phone conversation that, at TMI-2, they had a pressurizer level indication that was off-scale high, in other words, completely full of water?

A What I heard was -- is that it read off-scale, or something like that. You know, down, I guess -- I don't recall. I couldn't -- again, all I can tell you is things were confused at the time. And being -- we didn't have the as-built drawings here, so I mean you could really learn a lot if you had the as-builts here, as far as things like this go. But we didn't have the as-builts.

Q Do you know if anyone did?

A I doubt that anyone did. I understand -- I would

doubt it, but I really don't know.

Q Why would you doubt that anyone had --

A In this building -- excuse me.

Q Okay.

A In this building.

Q My question was directed to anyone, including the site itself, as to whether or not they had the as-built blueprints.

A That would blow my mind, if you'll pardon the expression. I could not understand any ~~IWC~~ I&C-type mechanic, if you will, or professional, not having those type of drawings. I just couldn't understand it.

Q At the site?

A At the site, yes.

Q Do you know whether or not in fact they do have as-built drawings at TMI-2?

A I do not know in fact. I don't know how they would have built the plant without it, but I -- you're asking me the question and I'm answering it -- I don't know, for a fact, ~~especially one that is really not -- I don't even think it passed its 100-hour running, had it? I mean, it was still -- it was still --~~ since it hadn't been transferred here to DCR yet, ~~so~~ I would think that it was just recently completed -- turned over they would have all of that stuff (drawings) there.

Q Could that have possibly been one reason why the

plant was not transferred to DCR, for failure to have as-built blueprints?

A I don't know.

Q Have you ever experienced a plant not having as-built blueprints before?

A No. In fact, when I was Branch Chief, ICSB, I insisted that prior to our final SER writeoff, in my area, my reviewers would actually go to the plant, and as a part of that review, do part of their ~~joined~~ drawing review, at the site ~~okay~~ -- now, wait a minute, I want you to understand, we didn't always -- we didn't do a complete schematic drawing review, no. We didn't have enough people.

We audited certain systems, in fact, sometimes certain portions of systems. But we went down there simply for this -- for that fact. We wanted to determine whether the drawings we had here -- were ~~how~~ current ~~they were~~, because there at the site you see the beginning of what goes through the formal quality assurance chain. It's first done right there, on the as-built, on the site. You know, it's properly noted, it's signed, and then finally become the QA drawings.

So I looked to see how our review of the drawings that we had reviewed here, compared to the final design, and we did that at the plant. So I know they have them at the plant.

Q To your knowledge, is there a specific NRC regulation

or requirement that an as-built blueprint be provided?

A We -- well, okay, when I say we require, I don't know the mechanism. I -- it has been a requirement that three copies of drawings, EI and CSB -- that's electrical, instrumentation, and control -- be provided the NRC. Boy, it's getting foggy now. One copy obviously came up to my branch -- I think one copy went to I and E, and one copy was maintained as a matter of record.

Q Where?

A I think downstairs here, in our files.

Q All the as-built blueprints?

A Correct.

Q Do you have any involvement with ECCS actuation? For instance --

A Yes.

Q -- the Westinghouse facilities?

A I'm aware of how they do it, yes.

Q How is that?

A Coincidence of level and pressure.

Q So is it a situation where you have to have both level and pressure, or either one?

A Both.

Q And does it operate, if you have --

A Excuse me -- I think both until recently. I think the --

Q It's a result of one of the --

A The bulletins, am I right, they had them separated -- or were they thinking about it? I'm not sure whether it's being done, or what. I'm not sure exactly the --

Q There's evidently some change as a result of --

A Right.

Q -- bulletins issued after TMI-2 this year. Prior to TMI-2, in terms of the coincident logic for ECCS actuation, what would happen if you had a situation of increasing pressurizer indication level, and --

A Pressurizer --

Q -- and decreasing pressure? In other words, divergent?

A You said, pressurizer pressure indication. Pressurizer pressure indication?

Q Pressurizer level indication.

A Okay. Increased level --

Q Pressurizer level indication.

A Right, and rising?

Q Right, and a decrease --

A Level rising --

Q Level rising, and a decrease in pressure, in other words, divergent actions? We're assuming coincident logic for ECCS actuation. What would happen in that case? Would he see it come on, or not?

A Well --

Q Automatically?

A I won't say categorically no. I'm saying I'd have to look at what rates these are happening at, okay -- I just don't know. And then I'd have to look at the -- I'd have to go back and look at the mathematics of the circuit, the analysis of the circuit, as to what rates it can do it, you know, stuff like -- to answer your question. But I think just generally speaking the answer is, in this indication, probably, you could not rely on it -- I considered that one signal.

For ECCS actuation, we have another signal, and that's high ~~drywell~~ pressure containment, okay, and I think it's a pretty good, diverse signal. It's not 100 percent diverse, but it's a pretty damn good one, because, gee, ~~it~~ depends upon plant design -- I -- my recollection is that it goes off at around four pounds, or thereabouts, I think.

And then I think ~~what~~ that containment sprays -- go on a little higher, I think 15 pounds or something like that, that's my recollection. So, you know, that's a diverse signal for ECCS.

Q So you would not expect ECCS automatically to come on with divergent situations involving pressure and pressurizer level?

A I could see where it may not come on. Again, as I

said, I'd have to look at -- there may be break sizes or whatever have you -- I don't know, frankly, well --

Q Well, let me put it this way --

A Well, wait a minute, wait a minute. What you would have here -- see, you're totally ignoring the control system, okay? What you would have ~~in a~~ is -- the control system would try to correct this. In other words, the Pressurizer heaters would come on, or go off, ~~depending, you know -- and you would try~~ to control the process, where these are not combined.

Q In other words, they're not going in the directions they should be?

A No, what I'm saying -- in other words, in that level signals would go to the level controller, and it ~~would tell the -- or it~~ would tell it, ~~like, if~~ you're low, that means ~~you might be~~ -- you might want to turn on the heaters, ~~okay,~~ you might want to, you know, to bring level up. ~~But then again, you know, you give me this set of things, and I really -- what you are trying to do -- I would guess that as you decrease pressure, pressure would cause a -- normally cause the liquid inventory to swell, okay.~~

So you would try to control the pressure here by turning on the pressurizer heaters, okay, so that starts heating that thing up, and increase the pressure, therefore, you can maintain control of level that way.

Q But you --

A But I'm saying your control system would try to control you, okay, and, you know, that's a positive action.

Q But you wouldn't necessarily actuate ECCS as a result of those divergent actions?

A Yes -- not necessarily is a good way of saying it.

Q Would you consider divergence of pressurizer level and pressure to be an unusual occurrence, or something out of the ordinary or expected?

A Since you control the plant this way, I would imagine that you would expect this swelling and, you know -- and it's almost done automatically. I don't know whether the guy would even -- the operator would even think about it. Because this is what you normally do, you know. You turn on those heaters and turn them off in order to, you know, keep yourself within those bounds.

I think he would -- I guess -- I don't know, I'm just guessing. I guess within certain bounds, he would probably not be concerned about it, ~~within certain bounds.~~ But I think if he diverged beyond what his figures told him, if anything -- I'm sorry, I just don't know. I would imagine that he's told that you should stay within these bounds, okay -- and probably, if not, he should make sure -- maybe start another bank of heaters, you know.

~~You have certain~~ -- My recollection on the PWR's is that ~~you have certain -- like,~~ you may have five banks of heate

Three of the banks, or two of the banks, are automatic, and the others are manual, okay -- so you might decide to throw on, manually, another bank, to try to pick this thing up or something like that, you know. I'm not saying for a minute that he has been taught to suspect a hole, or suspect something strange ~~in the as~~ is happening within the reactor vessel. I'm not suggesting that. I don't know.

Q If you had rapid rates of change, in the first instance, of decrease in pressure, and on the other hand an increase in pressurizer level, would that be considered a rather exceptional circumstance?

A When you say rapid -- I would normally expect that if I had -- let me speak of the big break. If I would have a big break, the pressure is going to go like this, and that liquid, until it starts pouring out of there, is going to swell quickly. But, you know, that thing -- you know, it's going to go like that.

Now, the ~~immediate~~ instantaneous thing is for it to swell. But then it quickly will come down, and ~~I don't think~~ -- I think that this happens so fast that the operator ~~didn't~~ wouldn't even know what happened. You know, it's that type of thing. And he's immediately on a loss of pressure, loss of liquid, condition.

Now, when you start talking about this small break, you know, this break that ~~hovers about this and that~~ is equivalent to the capacity of the make-up system,

really don't know. I think there you need an analyst to tell you that type of thing. I really don't know.

Q Do you know if there's been an occasion where you've had pressurizer level go off-scale high, and pressure drop, with ECCS coincident actuation, and therefore not have ECCS come on?

A I'm not aware of any, no.

Q Have you been involved at all with interrelationships between various systems in reactors -- how one system interacts with another one?

A Oh, yes, the --

Q A primary system?

A What do you mean by primary system? Are you speaking fluid-wise, or system operational-wise? In other words --

Q Interdependent analyses between the operation of a primary system and, let's say, for instance, operator action or reaction, if you have a problem involved?

A Okay, you've used some new terms on me. Do you mean control systems -- operator interface, that type of thing?

Q Yes.

A In other words, a control system being just one way, the operator should recognize it, and do things, and stuff like that?

Q Yes.

A We have some studies. My philosophy has been, for the last X number of years -- four or five years -- three years, four years, I'm not sure, that we're driving these reactors very hard now, because I believe -- ~~and I think there's a reason for it -- is that~~ we know more about the reactors than we did in years past.

But I believe that we have very closely coupled -- I recall, the primary loop with the secondary loop, and I think it's got to the point of where if you just sneeze into the secondary loop, and it would have an immediate effect on the primary loop, ~~where before~~ in the past you could do a whole lot of things on the secondary side.

And ~~it~~ - the primary side would -- ~~oh, yeah, it~~ would see it, ~~you know~~, but it would kind of lazily follow along and take care of itself. Because of the close coupling of the primary and as a consequence, I was worried about this, and I got some money, and we started a study at Oak Ridge to look into this. And we started to get some reports on it, but unfortunately I left -- I just don't know where that program is now. But I know that, yes, we were worried about this relationship between the secondary and the primary system.

~~And I guess you almost can convert that into saying the control systems, and how they affect the primary system -- okay, that's what I'm saying, and then there have been studies -- Oak Ridge has done the work. In fact, gee,~~

~~what's his name -- well, Oak Ridge has done the work, and has probably written reports on that now. I'm not sure. Well, we started the work, and I'm sure that it -- I'm almost positive it's still supported.~~

Q Would it be possible to get copies of these reports?

A I would -- yes, the answer is yes.

Q Okay.

A But who is to give them to you -- I don't have them. I would guess either of the following people have that, either Mr. -- I don't know who -- in the split between DOR and DSS, I don't know who got what.

Q Is there someone you could get in touch with?

A Either Dominic Tondi or Rod Satterfield should be able to tell me who has those reports.

Q Okay. If you could see about getting a copy -- it's not something we need, obviously, this afternoon --

A You want copies --

Q Of all the reports.

A -- the control system interaction?

Q Right. And if we could have a stipulation that those would be included as the next numbered exhibits, when they're received, to the deposition?

A Okay, that would be number 4. Okay, so -- but, yes, to your question, work has been done and is being done on it.

Exactly where we stand on it, I don't know.

Q Why don't we mark these -- memorandum for Roger Madison from Darrell Eisenhut, dated July 31, 1978, subject: Suggested material for inclusion in your response to C. Michaelson's 6-4-79 memo to D. Okrent, O-K-R-E-N-T, concerning pipe break isolation, marked as Exhibit 5?

A No, I'm sorry -- I thought we called it 3 initially. Check your records.

Q Three, then, the next numbered exhibit. And we'll await the studies on the interaction of the systems --

A Right.

Q -- number 4.

A You want all reports?

Q Right.

(The documents referred to were marked for identification as Exhibits 3 and 4, and were received in evidence.)

MR. SIDELL: Off the record.

(Whereupon, a recess was taken.)

MR. SIDELL: On the record.

BY MR. SIDELL:

Q Mr. Ippolito, have you had an opportunity to review the TMI-2 files to refresh your recollection as to what the types of delays on the part of the NRC were that

you mentioned earlier in the deposition?

A. I've just opened up the files, and I'm thumbing through them, and I come to a memo that I think supports some of my earlier remarks, in that the memo is written from my Assistant Director, Robert L. Tedesco, to -- at that time, to R. C. DeYoung, which effectively says, we've had a meeting with the applicant, and on December 18, he promised to provide information that would help us evaluate the remaining open items on -- during our INC review, and we have not received them yet, and the date of that memo is March 12, '76.

And we continue to say that, unless we get the stuff by -- in April, we effectively cannot meet the overall projected schedule. I think this is indicative of some of the problems, and the long review period, that was necessary to complete Unit 2.

Q. Does that memo refer to a specific completion date for the OL?

A. No, it just says, our final input to the SER, okay -- and what that means, if you know the process, the final SER was supposed to be issued in April of '76, we would require 30 days -- in other words, the ACRS would require 30 days to have that in its hands before we can schedule it for an ACRS meeting. We would have the ACRS meeting some time after that -- a week, or that same week, there would be

a letter from the ~~ACR~~ ACRS saying -- recommending to the Chairman ~~as to~~ whether or not ~~they would~~ the NRC should issue a license to this plant. Assuming that it was an affirmative reply, we would attempt to clean out as many open items as possible, and issue the operating license soon thereafter.

So I'm saying that, if I were to make a guess, at that time, I would say maybe if everything worked according to the clock, September, they should have had their operating license.

Q September of 1976?

A Right, if everything worked real well -- according to the clock.

Q To your knowledge, has the ACRS ever not recommended against the issuance of an OL for a reactor?

A I don't recall -- I know -- no, I don't ever recall -- I know that what they have said, on a number of occasions, is that, we want -- we feel that we want to study this longer -- we suggest that you come back again, either next month or two months from now or three months from now, or whatever -- they've done that. But I do not recall an instance where they recommended to the Chairman, or to the Commission, that an OL not be issued to somebody.

Q And that's going back from your perspective of about 12 years at the NRC -- and at the AEC?

A Yes.

Q Do you know, of the 70-some odd plants currently licensed, how many plants that would have included back in 1967?

A I would think all but about six or seven or eight, somewhere around that number.

Q We're talking about on the order of maybe 60 or 65 plants?

A Yes.

Q Can we have a copy of the memo you referred to from Bob Tedesco, and mark that as the exhibit next in order?

A This happens to have an enclosure to it, listing the --

Q Open items?

A -- open items.

MR. SIDELL: Off the record.

(Whereupon, a recess was taken.)

MR. SIDELL: On the record.

BY MR. SIDELL:

Q Mr. Ippolito, you have provided me with a copy of a memorandum from Robert L. Tedesco, T-E-D-E-S-C-O, Assistant Director for Plant Systems, Division of Systems Safety, dated March 12, 1976, to R. C. DeYoung, Assistant Director for LWR's DPM, entitled, "Unresolved Items in Electrical Instrumentation and Control Systems for Three Mile Island Unit Number 2," which is attached -- which is a two-

page memorandum with, as an enclosure, a list of several items, apparently unresolved, dealing with instrumentation, control, and electrical systems, for TMI-2, which includes a reference to a list entitled, "Pre-Safety Evaluation Report Items of Concern for Three Mile Island Unit Number 2."

Let's mark as exhibit next in order the March 12, 1976 memorandum from Tedesco to DeYoung as Exhibit 5, and as Exhibit 6, a listing entitled "Pre-Safety Evaluation Report, Items of Concern for Three Mile Island Unit 2," which is a nine-page report referred to in Exhibit 5. And for the report, Mr. Ippolito, can you give a description of what this pre-safety evaluation report is?

(The documents referred to were marked for identification as Exhibits 5 and 6, and were received in evidence.)

THE WITNESS: Yes, what is common practice for my reviewers is -- and is for them at certain times in their review to provide a compilation of the items or areas remaining to be resolved with the licensee, or applicant, as the case may be, so that I can best judge of how much more effort is needed to complete the review, and to determine what type of schedule -- manpower schedule, I could meet, and suggest to our management.

This happens to be one of these work items, a

laundry list, if you will, provided to me, and I in turn -- to Tedesco, to the Projects Branch, to inform them of exactly the status of their review, and what remains open. So therefore you might call this a working list of items yet to be completed on that docket, in the instrumentation and control area.

BY MR. SIDELL:

Q Now, although what we have now marked as Exhibit 6 of this deposition might not be a formal document included in the licensee's docket, is there any question that the information contained in it is accurate in terms of the fact that these are the open items, at the time this evaluation report was produced?

A I have no doubt that this was in fact -- ~~this in fact~~ represented the open items yet to be completed on TMI at that time.

Q Let me direct your attention to page 3 of Exhibit 6, item 6, which refers to, quote, "Documentation relating to the decay heat removal (DHR) low pressure to high pressure isolation valve circuity."

A Circuitry -- it's a typographical error. It should read circuitry, valve circuitry.

Q Under "statement of concern," number four, there is a statement, quote, "Suitable valve position indication should be provided for these valves in the control room," end

quote. Can you tell me what that statement refers to?

A I would have to -- the decay heat removal system, as I recall, on the B and W plant, is what we normally refer to, I believe, as the RHR system, or residual heat removal system, or the ultimate shutdown system, if you will. And as a consequence, it has an interface with the primary system, because that is the system which it eventually has to cool down.

And what we are assuring is that this interface between the primary system and this shutdown system, is properly protected, and what this statement of concern is, is that we want two valves, okay, and the valves are to be able to do the items as listed here.

One of the items happens to be that you are to provide -- TMI-2, suitable valve position indication. Now, since -- and since TMI-2 is supposed to have been designed to ~~I Triple E~~ IEEE Standard 279, which is a requirement of them, ~~I Triple E~~ IEEE Standard 279 requires that you provide information to the operator that is not ambiguous, and that all systems should measure directly the parameter, or variable, that it's intended to monitor.

So what I'm saying here is that I would expect, had I been the reviewer, that when they submitted this design to me for these DHR valves, that I would assure myself that these valves were -- that the valve positions were directly

measured, that is, we normally find that a suitable, reasonable way to directly measure these motor-operated valves is normally by limit switches affixed to the stem of the valve. And that's what I would expect -- that I guess would be the position indication, I would expect, on these valves.

Q Would that be a position indication in a direct manner, rather than an indirect manner, as we previously discussed?

A I -- this would be a direct manner.

Q Do you know whether or not these particular valves are classified as safety-related?

A These would be safety-related, because they form the interface between a high-pressure system and a low-pressure system, and also these lines generally penetrate containments, so they would serve both purposes.

Q Do you know whether or not the licensee at TMI-2 did in fact provide actual position indications for these particular valves?

A I don't know.

Q Could this have possibly been one of the remaining open items?

A This was at the time. In other words, this system -- these four requirements were -- had to be satisfied in one way or another for this system. And I -- just generally speaking, you know, the fact that they did get a license from

us, I have to assume that the review progressed to the point of where satisfactory designs were provided for these -- in these areas.

Q So by virtue of the issuance of an operating license, you would conclude that all of these items were properly resolved by the licensee?

A Correct.

MR. SIDELL: Do you have any questions?

MS. NORDLINGER: No questions.

MR. SIDELL: Okay, I have nothing further at this time, Mr. Ippolito. What we have been doing in the past with the depositions -- rather than adjourn them, merely recess them, so that in the event, hopefully unlikely, that we have further questions of you, we can more easily continue on.

THE WITNESS: Okay.

MR. SIDELL: I would represent to you that today we have not recalled anyone to continue a deposition, although we are planning on doing that in a small number of cases. Again, it merely leaves the option open to us.

kk

But at this time, with nothing further, we will recess the deposition, and we certainly thank you for your understanding and your time.

THE WITNESS: You're welcome.

(Whereupon, the deposition was recessed at 1:23 p.m.)

I have read the foregoing pages, 1 through 74, and they are a true and accurate record of my testimony therein recorded.

_____
THOMAS A. IPPOLITO

Subscribed and sworn to before me this_____day of_____, 1979

_____
Notary Public

My Commission Expires:_____

REPORTER'S CERTIFICATE

DOCKET NUMBER:

CASE TITLE: DEPOSITION OF THOMAS A. IPPOLITO

HEARING DATE: August 9, 1979

LOCATION: Bethesda, Maryland

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

Date: August 10, 1979

[redacted]

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

Ex.

Thomas A. Ippolito

PROFESSIONAL QUALIFICATIONS

I. Education and Training

I received a Bachelor of Electrical Engineering degree from Marquette University, Milwaukee, Wisconsin in June 1953. In addition, I have taken courses in Nuclear Physics and Engineering, Engineering Management and Program Management at the Department of the Navy and I completed the Executive Leadership and Management Program at the Federal Executive Institute in 1974.

II. Professional Experience

From July 1953 to May 1955, I served as a junior project engineer responsible for the design and development of electron tubes and transistors for the Department of Navy, Bureau of Ships. I served in the Army Signal Corps from May 1955 to May 1957, where I was responsible for the design of test instrumentation for a large surveillance radar system. In May 1957, I accepted a position in the Nucleonics Staff of the Navy Department, Bureau of Ships. I served as an electronics consultant to the Navy's Nuclear Power Division in the areas of nuclear instrumentation and reactor protection systems. Concurrently, I served as lead project engineer on radiation monitoring systems. I was responsible for the design, development, fabrication, and installation of these systems aboard nuclear powered submarines and surface ships. From May 1966 to July 1967, I served as program manager for short range weapon systems in the Anti-submarine Warfare Project Office, Department of Navy.

I have been with the Regulatory staff since July 1967, during which time I served as a senior reactor engineer (Instrumentation) and have

participated in the review of instrumentation, control and electrical systems of numerous nuclear power stations and in the formulation of related standards and regulatory guides. In February 1973, I was appointed Branch Chief of ICSB.

The responsibilities of the Instrumentation and Control Systems Branch (ICSB) include the performance of technical reviews of the design, fabrication and operation of nuclear power plant electrical power, reactor protection and safety instrumentation, control instrumentation and radiation monitoring systems; reviews the electrical power and control systems for all power reactors for adherence to appropriate codes and standards; and performs a review of reactor electrical and control systems with respect to systems reliability, stability, and transient characteristics. This review encompasses evaluation of applicants' safety analysis reports, generic reports and other related information on the electrical and instrumentation design. Further, the Branch develops the bases for Regulatory acceptance criteria for electrical and instrumentation power systems designs; evaluates experience obtained during the construction and operation of nuclear power plants and relates this information to future evaluations and acceptance criteria; and participates in the development of Regulatory guides and regulations pertaining to electrical and instrumentation power systems. There efforts were used to develop and update the Standard Review Plan (SRP).

I was a member of the IEEE Nuclear Power Engineering Committee of the Power Engineering Society including Subcommittee 1 (SC 1) General Plant Criteria from 1973 through 1978. I have participated in the development of numberous IEEE standards including IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Station", IEEE Std 308, "Standard Criteria for Class IE Power Systems for Nuclear Power Generating Stations" and IEEE Std 323, "Qualifying Class I Electric Equipment for Nuclear Power Generating Stations".

Additionally, I was appointed as a U. S. Delegate to the International Electrotechnical Commission (IEC) to serve as Secretary to Working A3 of Subcommittee 45A. In 1977 I has appointed to serve as Chairman of Subcommittee 45A of the IEC. This Subcommittee prepares international standards, for reactor instrumentation and controls.

My present position is that of Chief, Operating Reactors Branch #3, in the Division of Operating Reactors, Office of Nuclear Reactor Regulation. I was reassigned to this position in June 1978. In this position I am responsible for the supervision of NRC's administration of the Boiling Water Reactor (BWR) facilities licensed for operation, and in particular, the review and processing of applications for license amendments for these power reactors. My responsibilities include the supervision and coordination of such reviews by personnel within the Branch and by technical consultants, including the preparation of safety evaluations, hearing testimony and other reports, as necessary. Additionally, I am responsible for the review of operating history and performance of all assigned operating reactors and for special reviews, determinations regarding back-fit and determinations regarding items of non-compliance and safety.

MEMORANDUM FOR: Robert L. Tedesco, Assistant Director for Plant Systems, DSS

FROM: Thomas A. Ippolito, Chief, Instrumentation and Control Systems Branch, DSS

SUBJECT: MANPOWER REQUIREMENTS FOR ICSB

The enclosed manpower studies of the ICSB are provided for your information. Enclosure 1 is a simplified study which is based on data derived from the more detailed study presented in Enclosure 2.

Enclosure 1 concludes that 11 additional engineers will have to be hired in 1977 and 4 more in 1978. The more detailed study presents a more modest recommendation of 7 engineers and one supervisor in 1977 and none in 1978. I strongly recommend adoption of the recommendations of Enclosure 2 which also quantifies the slippage, in manhours, which will result from a failure to follow the recommended hiring a schedule.

Thomas A. Ippolito, Chief
Instrumentation and Control
Systems Branch
Division of Systems Safety

Enclosure:
1. Simplified calculation of ICSB workload and manpower availability
2. Detailed Manpower Study

cc: C. Miller
M. Srinivasan
R. Scholl
C. Long
J. Glynn

Contact
R. Scholl
X27387

ENCLOSURE 1

ICSB MANPOWER STUDY OF PROFESSIONAL MANPOWER REQUIRED IN THE REVIEW OF LICENSING APPLICATIONS

	CP					
A. Licensing Review	MIN*	NORM	MAX	MIN	NORM	MAX*
	2	2	2	2	4	5
1. Acceptance Review						
2. Application Review						
a. Initial Questions - Q 1	2	16	28	23	30	38
b. Regulatory Positions - Q 2	1	23	52	4	127	249
c. SER	4	14	25	59	128	198
d. Drawing Review/ Site Visit	-	-	-	16	16	16
e. Technical Specifications	-	-	-	16	16	16
f. Supplemental SER	3	3	3	8	8	8
g. Hearings	-	-	-	2	2	2
h. Follow Up	5	5	5	1	1	1
B. Standard Plants		PDA			FDA	
1. Acceptance Review	3	3	3	5	5	5
2. Application Review						
a. Initial Questions Q 1	11	20	28	29	38	47
b. Regulatory Positions Q 2	14	22	29	5	158	312
c. SER	14	30	46	73	160	247
d. Drawing Review/ Site Visit	-	-	-	20	20	20
e. Technical Specifications	-	-	-	20	20	20
f. Supplemental SER	5	5	5	10	10	10
g. Hearings	-	-	-	2	2	2
h. Follow Up	9	9	9	1	1	1

C	TACS	MIN*	NORMAL	MAX*
	R18	1	10	27
	R35	4	4	4
	R37	1	5	13
	R38	1	7	15
	R40	7	7	7
	R51	1	7	16
	R52	6	6	6
	R53.	1	4	7
	R54	1	10	20
	R55	1	5	11
	R56	1	24	83
	R71	1	5	16
	R73	1	7	21
	JNK	1	3	6

* 68% of all work will fall within the range of these minima and maxima.

WORK LOAD

1. Scheduled Cases 1977

A. Construction Permits

1. Acceptance Reviews	3@	2	=	6
2. Q 1	5@	16	=	80
3. Q 2	4@	23	=	92
4. SER	5@	14	=	70
5. SSER	5@	3	=	15

B. Operating Licenses

1. Acceptance Reviews	4@	4	=	16
2. Q 1	8@	30	=	240
3. Q 2	10@	127	=	1270
4. SER	5@	128	=	640
5. SSER	3@	8	=	24
6. Drawing Reviews/ Site Visits	3@	16	=	48
7. Technical Specifications	3@	16	=	48
8. Hearings	3@	2	=	6

C. Standard Plants PDA

1. Acceptance Reviews	1@	3	=	3
2. Q 1	3@	20	=	60
3. Q 2	3@	22	=	66
4. SER	2@	30	=	60
5. SSER	0@	5	=	0

D. Standard Plants FDA

1.	Acceptance Reviews	2@	5	=	10
2.	Q 1	2@	38	=	76
3.	Q 2	2@	158	=	316
4.	SER	2@	160	=	320
5.	SSER	0@		=	0
6.	Drawing Review/ Site Visit	0@		=	0
7.	Technical Specifications	0@		=	0
8.	Hearings	0@		=	0

2. TACS - 1977

R18	11	@	10	=	110
R35	1	@	4	=	4
R37	10	@	5	=	50
R38	9	@	7	=	63
R40	1	@	7	=	7
R51	25	@	7	=	175
R52	2	@	6	=	12
R53	3	@	4	=	12
R54	3	@	10	=	30
R55	13	@	5	=	65
R56	19	@	24	=	456
R71	20	@	5	=	100
R73	22	@	7	=	154
UNK	2	@	3	=	6
		TOTAL		=	4710 (man days)

Available [(52 x 5) - (31)] 10 = 2290 / -2420 (man days)

Need 11 more men in 1977

3. Scheduled Cases 1978

A. Construction Permits					
1. Acceptance Reviews	9	@	2	=	18
2. Q 1	9	@	16	=	144
3. Q 2	4	@	23	=	92
4. SER	5	@	14	=	70
5. SSER	4	@	3	=	12
B. Operating Licenses					
1. Acceptance Reviews	9	@	4	=	36
2. Q 1	8	@	30	=	240
3. Q 2	8	@	127	=	1016
4. SER	9	@	128	=	1152
5. SSER	9	@	8	=	72
6. Drawing Review/ Site Visit	9	@	16	=	144
7. Technical Specifications	9	@	16	=	144
8. Hearings	9	@	2	=	18
C. Standard Plants					
1. Acceptance Reviews	2	@	3	=	6
2. Q 1	2	@	20	=	40
3. Q 2	2	@	22	=	44
4. SER	3	@	30	=	90
5. SSER	3	@	5	=	15

D. Standard Plants FDA

1. Acceptance Reviews	4	@	5	=	20
2. Q 1	4	@	38	=	152
3. Q 2	4	@	158	=	632
4. SER	4	@	160	=	640
5. SSER	4	@	10	=	40
6. Drawing Review	4	@	20	=	80
7. Technical Specifications	4	@	20	=	80
8. Hearings	4	@	2	=	8

4. TACS - 1978

R18	7	@	10	=	70
R35	1	@	4	=	4
R37	9	@	5	=	45
R38	7	@	7	=	49
R40	1	@	7	=	7
R51	16	@	7	=	112
R52	1	@	6	=	6
R53	2	@	4	=	8
R54	1	@	10	=	10
R55	6	@	5	=	30
R56	12	@	24	=	288
R71	15	@	5	=	75
R73	16	@	7	=	112
JNK	2	@	3	=	6
			TOTAL	=	3827
Available [260-31] 10				=	2290
Possible [229] 11				=	2519
			deficit	=	-1018

Need to hire 4 more men in 1978

MANPOWER STUDY FOR INSTRUMENTATION AND CONTROL SYSTEMS BRANCH FOR 1977 AND 1978

Abstract

This report describes the basis, methods and results of a study into the manpower requirements for the Instrumentation and Control Systems Branch (ICSB) for the calendar years 1977 and 1978. This study was necessitated by the recent reorganization of the Electrical, Instrumentation and Control Systems Branch (EICSB) and the increased depth of review as reflected in the Standard Review Plan (NUREG 75/087).

Introduction

The projection of manpower requirements is, of necessity, based on many variables. The resultant forecasts are limited in their accurracy to the same degree as the input data. In order to have confidence in the results of this report, the input data and analysis methods are described and the results are evaluated in the conclusion of this report. Appendix A identifies the assumptions which have been made in this study and references the report section in which they were used.

Data Base

The projected new Construction Permit Applications are presented in Table I. The projected new Operating License Applications are presented in Table II. The projected new case load for standard plants and special reviews for Naval Reactors is presented in Table III. The data in these three tables were extracted from the February 1977, issue of "Status Summary Report" (NR-MI-002-2) and, for the purposes of this report, is assumed to be 100% accurate at the 100% confidence level.

TABLE I

NEW APPLICATION FORECAST - CONSTRUCTION PERMITS

As of Week Ending February 4, 1977

FACILITY NAME	UTILITY	PROJECT NUMBER	APPL. PSAR SUB. DATE
ERIE 1 & 2	OHIO EDISON CO	512	11/30/76C
SONGSERT 1 & 2	SAN DIEGO GAS & ELECTRIC	558	03/00/77
SUMMIT 1 & 2 (REV)	DELMARVA POWER & LIGHT	450	10/00/77
CARROLL 1 & 2	COMMONWEALTH EDISON	559	10/00/77
CENTRAL IOWA NUC. UNIT	IOWA ELECTRIC POWER & LIGHT	532	06/00/78
SAN JOAQUIN 1,2,3,4	L.A. DEPT. OF WATER & POWER	479	02/00/78
NEW YORK 1	NEW YORK STATE ELECTRIC & GAS	507	05/00/78
CAROLINA 1*	CAROLINA POWER & LIGHT	643	06/00/78
FULTON 1 & 2 (REV)	PHILADELPHIA ELECTRIC	463	06/00/78
INDIANA 1*	PUBLIC SERVICE OF INDIANA	647	06/00/78
DETROIT 1*	DETROIT EDISON	649	07/00/78
NEBRASKA 1*	NEBRASKA PUB. POWER DISTRICT	638	07/00/78
STANISLAUS 1 & 2	PACIFIC GAS & ELECTRIC	564	10/00/78

*NOT FIRM

TABLE .

NEW APPLICATION FORECAST - OPERATING LICENSE REVIEWS
(Anticipated OL Tender Dates For All Units With Construction Permits)

As Of Week Ending February 4, 1977

	FACILITY	DOCKET NUMBER	UTILITY	APPLICANT SUBMITS PSAR
	[illegible] 2 & 3	0361E0362	SOUTHERN CALIFORNIA EDISON	11/22/76L
	[illegible] 1	350-00395	SOUTH CAROLINA ELECTRIC & GAS	12/10/76C
	[illegible] NUCLEAR 2	050-00397	WASHINGTON P.P.S.S.	03/15/77
	[illegible] PEAK 1 & 2	0445E0446	TEXAS UTILITIES GENERATING	06/00/77
	[illegible] 1 & 2	0330E0329	CONSUMERS POWER	06/00/77
	[illegible] 1 & 2	0416E0417	MISSISSIPPI POWER & LIGHT	07/00/77
FY 1978	[illegible] 1 & 2	0438E0439	TENNESSEE VALLEY AUTHORITY	12/00/77
	[illegible] 1 & 2	0498E0499	HOUSTON POWER & LIGHT	01/03/78
	[illegible] 1 & 2	0387E0388	PENNSYLVANIA POWER & LIGHT	01/00/78
	[illegible] 1	050-00382	LOUISIANA POWER & LIGHT	03/00/78
	[illegible] 1 & 2	0456E0457	COMMONWEALTH EDISON	06/00/78
	[illegible] 1 & 2	0454E0455	COMMONWEALTH EDISON	06/00/78
	[illegible] 1 & 2	0414E0415	DUKE POWER	06/00/78
	[illegible] 1 & 2	0461E0462	ILLINOIS POWER	06/01/78
FY 1979	[illegible] AREA 3 & 4	0404E0405	VIRGINIA ELECTRIC & POWER	11/00/78
	[illegible] RIVER	050-00363	JERSEY CENTRAL POWER & LIGHT	01/00/79
	[illegible] NUCLEAR 1	050-00460	WASHINGTON P.P.S.S.	03/00/79
	[illegible] 1 & 2	3481E0486	UNION ELECTRIC	03/00/79
	[illegible] 1 & 2	0443E0444	PUBLIC SERVICE OF NEW HAMPSHIRE	08/03/79
FY 1980	[illegible] 3	050-00423	NORTHEAST NUCLEAR ENERGY	11/00/79
	[illegible] VALLEY 2	050-00412	DUQUESNE LIGHT	12/00/79
	[illegible] VERDE 1,2,3	0528-0530	ARIZONA PUBLIC SERVICE	12/30/79
	[illegible] MILE POINT 2	050-00410	NIAGARA MOHAWK POWER	03/00/80
FY 1981	[illegible] 1 & 2	0352E0353	PHILADELPHIA ELECTRIC	10/00/80
	[illegible] CREEK 1 & 2	0354E0355	PUBLIC SERVICE GAS & ELECTRIC	11/03/81
FY 1982	[illegible] 1 & 2	0424-0425	GEORGIA POWER	01/00/82
FY 1983	[illegible] 1 & 4	0434E0435	VIRGINIA ELECTRIC & POWER	12/03/82
	[illegible] 1	350-00367	NORTHERN INDIANA PUBLIC SERVICE	N/S

TABLE III

STANDARDIZATION REVIEWS

PROJECT	PROJ NO.	APPLICANT	TENDER DATE
REPLICATION			
[illegible] RESAR-414	572	WESTINGHOUSE	10/08/76C
[illegible] BOPSAR (W)	575	GIBBS & HILL	10/15/76C
PRELIMINARY			
[illegible]	426	MARITIME ADMINISTRATION	03/00/77
[illegible] GIBBSAR ICE/BCW	649	GIBBS & HILL	06/00/77
[illegible] GESSAR 238 N.I.	447		11/00/77
[illegible] CHIEF*	659	BECHTEL POWER	01/00/78
[illegible] ESSAR-BWR	556	STONE & WEBSTER	01/00/78
[illegible] CESSAR NSSS	470	COMBUSTION ENGINEERING	03/00/78
[illegible] GESSAR-238 NSSS	559	GENERAL ELECTRIC	07/00/78
[illegible] GESSAR-251 NSSS	511	GENERAL ELECTRIC	07/00/78
[illegible] GASSAR-6 REV. NSSS	646	GENERAL ATOMIC	09/00/78

OTHER SPECIAL REVIEWS

PROJECT	PROJ NO.	APPLICANT	TENDER DATE
ESPTS 9 G	667	ERDA NAVAL REACTORS	02/28/77
ESP[illegible] PSAR	426	MARITIME ADMINISTRATION	09/03/77
ESPTS 4 G SHIP	655	ERDA NAVAL REACTORS	10/00/77
ESP[illegible] SUBMARINE CORE	656	ERDA NAVAL REACTORS	01/00/78
ESP[illegible]	657	ERDA	08/00/78

The current work load for the ICSB is presented in Table IV. Tables VA and VB presents the mean time required for an EICSB reviewer to accomplish a case review milestone or other, specified task under the Standard Review Plan.

The data for Table IV Parts A and B was taken from the same source as Table I, II, and III except for Docket No. 50-373 which was estimated by approximating the due dates and averaging the other scheduled review times given in Table IV. The derivation of the data in Table VA is presented in Appendix B. The derivation of Table VB is presented in Appendix C. The remainder of the Table IV (Parts C through I) and Table VI (which presents a listing of Current Technical Activities (TACS) and projected TACS work load) was obtained from the NRC's Manpower System (MPS) and the derivation is given in Appendix C.

The data presented in Table VII is representative of all past EICSB work effort and is assumed to be a valid basis for determining the reduction in ICSB scope of work as a result of the recent reorganization and its effect on the data of Table V.

Methods

The basic methods which were used in this study are:

1. Data Collection
2. Data Correction
3. Data analysis using classical statistical techniques (e.g., determination of the mean and standard deviation.

TABLE IV

CURRENT WORK LOAD

Part A - Construction Permits

Docket Number	Milestones (Scheduled Review Time / Scheduled Date)			
	05	12	24	27
500	C	C	6M/(7/1/77)	1.8M/(12/16/77)
535	C	C	2M/(1/7/77)	1.5M/(6/28/77)
537	C	C	1.5M/(4/15/77)	2.5M/(9/0/77)
548	C	C	C	2.5M/(5/26/77)
549	C	C	C	2M/(6/17/77)
566	2W/(1/19/77)	3W/(2/8/77)	1.5M/(4/8/77)	2M/(8/15/78)
7448	1.5M/(2/15/77)	1.6M/(7/15/77)	2M/(1/27/78)	3M/(8/15/78)

Part B - Operating Licenses

327	C	C	C	1.5M/(4/20/77)
358	C	1M/(4/1/77)	1.5W/(5/20/77)	2M/(9/16/77)
366	C	1.5M/(3/4/77)	2M/(10/7/77)	1.2M/(1/12/78)
368	C	C	C	2M/(8/19/77)
373	2.2M/(5/12/77)	1.5M*/(7/1/77)	1.6M*/(9/25/77)	1.6*/(2/5/780)
390	3M/(3/25/77)	2M/(8/5/77)	2.2M/(1/6/78)	1.5M/(5/12/78)

KEY

* = Estimated

M = Months

W = Weeks

C = Completed

+ = Not included in Figure 2

TABLE IV (Continued)

Part C - Topical Reports (R51)

TACS Number	Due Date	Report Number	Notes
300		WCAP-8587	
472		NEDO-10802	
554		NEDO-10905	
969	4/15/77	BAW-10085	
1075		NEDO-20304	
1408		AI-75-2	
1796		CENPD-170	
3244	5/20/77		See Zimmer
3953	1/21/77	WCAP-8584	

Part D - Standards (R73)

TACS Number	Due Date	Notes
3258		(2 men assigned)
3369		
4223		
4343		(4 men assigned)
4378		(2 men assigned)
4390		

Part E - Standard Review Plan (R56)

TAC Number	Due Date
3297	
3306	
3681	
3721	
3926	
4297	
8300	7/1/78
8365	

TABLE IV (Continued)

Part F - Research (R54)

TACS Number
3436
3446

Part G - Regulatory Guides (R71)

TACS Number	
1623	
3378	(2 men assigned)
3971	
4062	
4385	

Part H - Generic Issues (R55)

TACS Number	Due Date	Issue Number
2129		I.E.A.2
2131	5/20/77	I.E.A.4
2134		I.E.A.7
2135	9/30/77	I.E.A.8
2140		I.E.B.3
2141 2142	9/1/77	I.E.B.4

Part I - Other TACS

TACS Number	Due Date	Notes
965		
2370		(R52)
3256		(R53)
3329		(R18)
3905		(R37)

TABLE IV (Continued)

Part I - Other TACS (Continued)

TACS Number	Due Date	Notes
3916		(R18)
3918		(R18)
3919		(R38)
4049	10/30/77	(3 men assigned)
4196		
4209		
4370		(R18)
4375		
6372		(R38)
JNK		

TABLE V

EICSB AVERAGE TIMES

PART A - CASE WORK

CASE TYPE	CP	OL	SP
Acceptance Review	24.0	35.5	24.0
Milestone 05	196.1	303.2	157.0
12	300.4	1265.0	174.5
24	174.7	1281.0	242.0
27	40	80	40.0
Drawing Review/ Site Visit	NA	160.0	NA
Technical Specifications	NA	160.0	NA
Hearings	NA	16.0	NA
Follow up	68.0	7.0	68.0
Special Projects	131.2		

PART B TACS

TACS	TIME	COUNT	TACS	TIME	COUNT(per Year)
R18	83.2	10	R53	29.3	3
R35	32.8	1	R54	83.6	2
R37	40.8	12	R55	37.7	9
R38	53.9	10	R56	191.4	16
R40	55.4	1	R71	37.3	21
R51	55.4	22	R73	54.6	23
R52	47.0	1	JNK	24.6	3

TABLE VI

PROJECTED TACS WORK LOAD

TACS	DUE DATE	TIME REQUIRED	
		1977	1978
969	4/15/77	55.4	0
2131	5/20/77	37.7	0
2135	9/30/77	37.7	0
2141	9/1/77	37.7	0
3244	5/20/77	15.1	0
3953	1/21/77	55.4	0
8300	7/1/78	470.7	279.3
All other TACS		9627.7	6256.4

4. Performance of a resource balance (e.g., Manhours required less manhours available equals additional manpower requirements

The most controversial of these methods is the correction of the data for the time required for each task in order that these time requirements reflect the division of responsibility between the Instrumentation and Control Systems Branch (ICSB) and the Power Systems Branch (PSB).

The method for determining the reduction in the scope of review was to divide the past questions asked by EICSB, as shown in Table VII, into the questions in the area of the ICSB responsibility. The results of ratioing the first column to the third column for the means of each category are given in the results section under Item 1.

The method for determining the impact of the reorganization on other tasks was to find the mean of the results of result Item 1. This result is presented as result Item 2.

Table VIII presents the data of Table V corrected by results Items 1 and 2 as specified in Conclusion 1. No corrections were made to PDA or FDA(SP) because little electrical power design is presented in these SARs. The data for special reviews was not changed for the same reason. The next step in the analysis was to convert, by estimation, the data of Tables I, II, and III into a form similar to that of Table IV Parts A and B. The new Table of Estimated Case Loads is presented as Table IX Parts A, B, and C.

TABLE VII

	EICSB Questions Not In SAR Sections 3.12, 8, 9, or 10.2	Questions In 3.12, 8, 9, or 10.2	TOTAL
CP			
Duke Project 81	27	23	50
Clinton	48	24	72
	75	47	122
CP Acceptance Review			
Duke Project 81	17	14	31
A. R. Barton	28	12	40
	45	26	71
OL			
Zimmer	195	50	245
Hatch 2	27	13	40
	222	63	285
OL Acceptance Review			
La Salle	69	12	81
Zimmer	10	3	13
Hatch 2	7	5	12
	86	20	106

TABLE VIII

ICSB REVIEW TIMES

PART A - CASE WORK

CASE STEP	TIME (MAN HOURS)			
	CP	OL	PDA	FDA
Acceptance Review	14.9	28.4	24.0	35.5
Q1 (05)	121.6	242.6	157.0	303.2
Q2 (12)	186.2	1012.0	174.5	1265.0
SER (24)	108.3	1024.8	242.0	1281.0
SSER (27)	24.8	64.0	40.0	80.0
Drawing Review/ Site Visit	NA	128.0	NA	160.0
Technical Specs	NA	128.0	NA	160.0
Hearings	NA	12.8	NA	16.0
Follow Up	42.2	5.6	68.0	7.0
Special Reviews	131.2			

PART B TACS

TACS	TIME	COUNT	TACS	TIME	COUNT
R18	83.2	7	R53	29.3	2
R35	32.8	1	R54	83.6	1
R37	40.8	9	R55	37.7	6
R38	53.9	7	R56	191.4	11
R40	55.4	1	R71	37.3	15
R51	55.4	16	R73	54.6	16
R52	47.0	1	JNK	24.6	2

TABLE IX

ESTIMATED CASEWORK

Part A (CP)

Docket Number	01	05	12	24	27
450	1M/(11/77)	2M/(2/78)	4.7W/(5/78)	2.6M/(8/73)	N/A
463	1M/(7/78)	2M/(10/78)	N/A	N/A	N/A
499	1M/(3/78)	2M/(6/78)	4.7W/(9/78)	2.6M/(12/78)	N/A
507	1M/(6/78)	2M/(9/78)	4.7W/(12/78)	N/A	N/A
512	C	2M/(3/1/77)	4.7W/(6/1/77)	2.6M/(9/1/77)	2.2M/(2/15/78)
532	1M/(7/78)	2M/(10/78)	N/A	N/A	N/A
558	1M/(4/77)	2M/(7/77)	4.7W/(10/77)	2.6M/(1/78)	2.2M/(5/15/78)
559	1M/(11/77)	2M/(2/77)	4.7W/(5/78)	2.6M/(8/78)	N/A
[illegible]	1M/(11/78)	N/A	N/A	N/A	N/A
637	1M/(7/78)	2M/(10/78)	N/A	N/A	N/A
638	1M/(8/78)	2M/(11/78)	N/A	N/A	N/A
639	1M/(8/78)	2M/(11/78)	N/A	N/A	N/A
643	1M/(7/78)	2M/(10/78)	N/A	N/A	N/A

KEY

NA = Not Applicable (Falls in 1979)

C = Completed

M = Months

W = Weeks

TABLE IX

ESTIMATED CASEWORK

Part B (OL)

Docket Number	01	05	12	24	27
330	1M/(6/77)	2.6M/(9/77)	1.5M/(11/77)	1.6M/(3/78)	1.6M/(7/78)
631	C	2.6M/(3/77)	1.5M/(6/77)	1.6M/(10/77)	1.6M/(2/78)
382	1M/(4/78)	2.6M/(7/78)	1.5M/(9/78)	1.6M/(12/78)	N/A
387	1M/(2/78)	2.6M/(5/78)	1.5M/(7/78)	1.6M/(11/78)	N/A
395	C	2.6M/(4/77)	1.5M/(7/77)	1.6M/(10/77)	1.6M/(2/78)
397	1M/(5/77)	2.6M/(8/77)	1.5M/(11/77)	1.6M/(2/78)	1.6M/(6/78)
404	1M/(12/78)	N/A	N/A	N/A	N/A
413	1M/(7/78)	2.6M/(2/78)	1.5M/(12/78)	N/A	N/A
	1M/(7/77)	2.6M/(10/77)	1.5M/(12/77)	1.6M/(4/78)	1.6M/(8/78)
438	1M/(1/78)	2.6M/(4/78)	1.5M/(6/78)	1.6M/(10/78)	N/A
445	1M/(6/77)	2.6M/(9/77)	1.5M/(11/77)	1.6M/(3/78)	1.6M/(7/78)
454	1M/(7/78)	2.6M/(10/78)	1.5M/(12/78)	N/A	N/A
456	1M/(7/78)	2.6M/(10/78)	1.5M/(12/78)	N/A	N/A
461	1M/(7/78)	2.6M/(10/78)	1.5M/(12/78)	N/A	N/A
498	1M/(2/78)	2.6M/(5/78)	1.5M/(7/78)	1.6M/(11/78)	N/A

TABLE IX

Table C Standard Plants and Others

Docket Number	01	05	12	24	27
572	C	2.6M/(3/25/77)	1.5M/(6/3/77)	1.6M/(9/9/77)	1.6M/(1/78)
575	C	2.6M/(4/77)	1.5M/(7/77)	1.6M/(10/77)	1.6M/(2/78)
426	1M/(4/77)	2.6M/(7/77)	1.5M/(9/77)	1.6M/(12/77)	1.6M/(4/78)
649	1M/(7/77)	2.6M/(10/77)	1.5M/(12/77)	1.6M/(3/78)	1.6M/(7/78)
447	1M/(12/77)	2.6M/(3/78)	1.5M/(5/78)	1.6M/(8/78)	1.6M/(12/78)
650	1M/(2/78)	2.6M/(5,78)	1.5M/(7/78)	1.6M/(10/78)	N/A
544	1M/(2/78)	2.6M/(5/78)	1.5M/(7/78)	1.6M/(10/78)	N/A
470	1M/(4/78)	2.6M/(7/78)	1.5M/(9/78)	1.6M/(12/78)	N/A
550	1M/(8/78)	2.6M/(11/78)	N/A	N/A	N/A
581	1M/(8/78)	2.6M/(11/78)	N/A	N/A	N/A
548	1M/(10/78)	N/A	N/A	N/A	N/A
607		One Step Review			2M/(5/77)
426		One Step Review			2M/(11/77)
655		One Step Review			2M/(12/77)
656		One Step Review			2M/(3/78)
653		One Step Review			2M/(10/78)

After Table IX was completed, the data from Tables IV, Parts A and B and Table IX were plotted for each milestone and type of case (CP, OL, PDA or FDA). The value of the dependent parameter was taken from Table VIII (manhours) divided by the available review time (days) to give manhours/day. The work required after milestone 27 was divided by the time available until 1979 to determine work rate. This technique is necessary because this work is traditionally unscheduled (and unschedulable) and this technique helps to compensate for the graphical "roll off" which would occur because work efforts which will not be completed until 1979 are not included in this study.

The results are given in the figures which are listed in Table X. The summary of effort required for casework is shown in Figures 11(CP), 12(OL), 13(SP), and 14(Special Reviews). Figure 15 shows the total case work effort. The same techniques were used for TACS. The work load requirements from TABLE VI are shown in Figure 16. The total ICSB workload is shown in Figure 17.

When Figure 17 was completed, the manpower availability was claculated. The first step was to determine the effective number of days which are available in a working year.

The calculation is 52 weeks times 5 days/week, less 7 Federal holidays, less 15 days per year annual leave, less 10 days per year for training.

(52x5) - (31) = 260 - 31 = 229 days/year

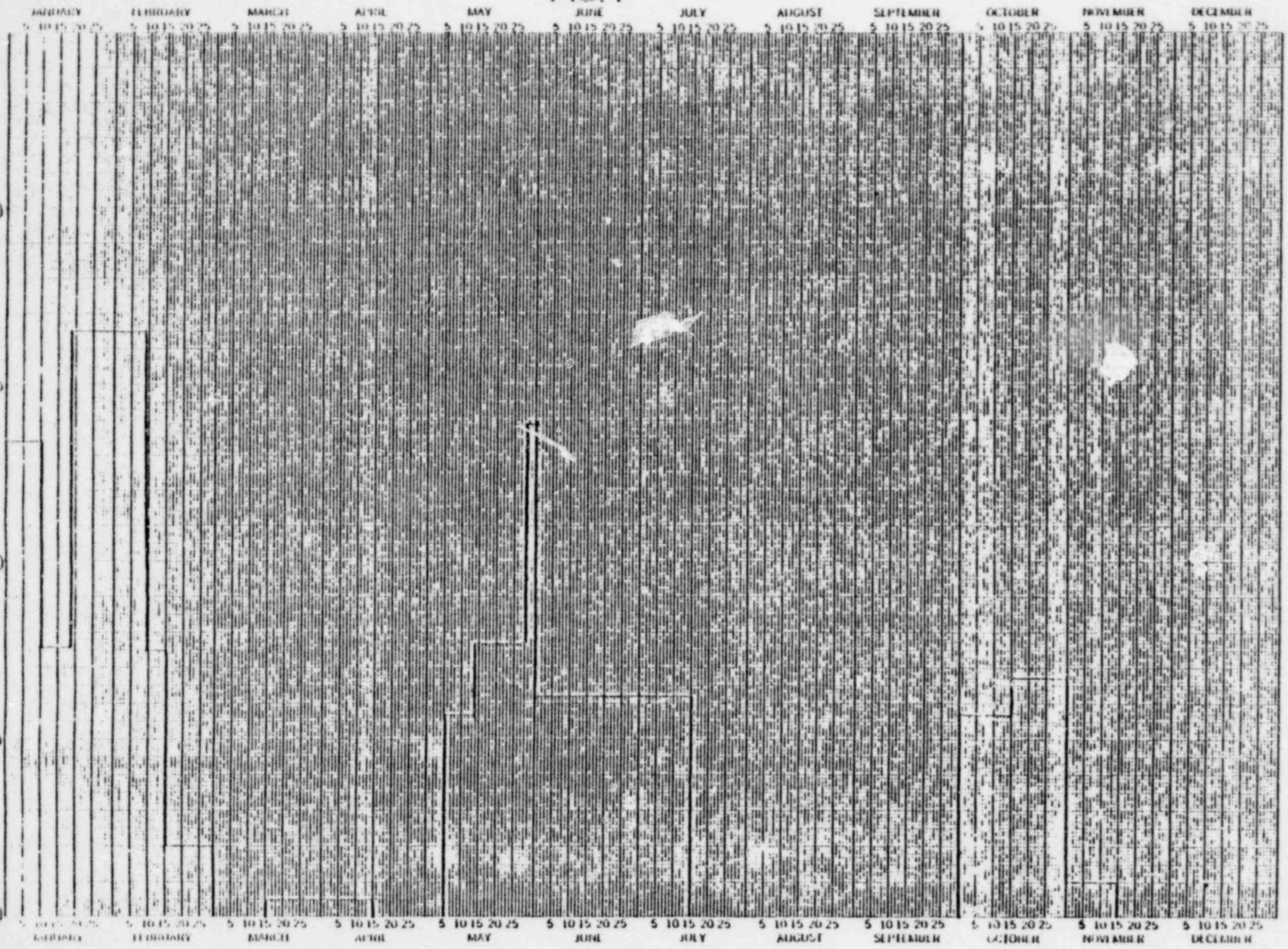
TABLE X

Milestone	CP Figures	OL Figures	PDA Figures	FDA Figures
01	1	3	7	8
05	1	3	7	8
12	1	4	7	9
24	2	5	7	10
27	2	6	7	10
post 27	2	6	7	10

K-E 1 YEAR BY DAYS X 250 DIVISIONS
KEUFFEL & ESSER CO. MADE IN U.S.A.

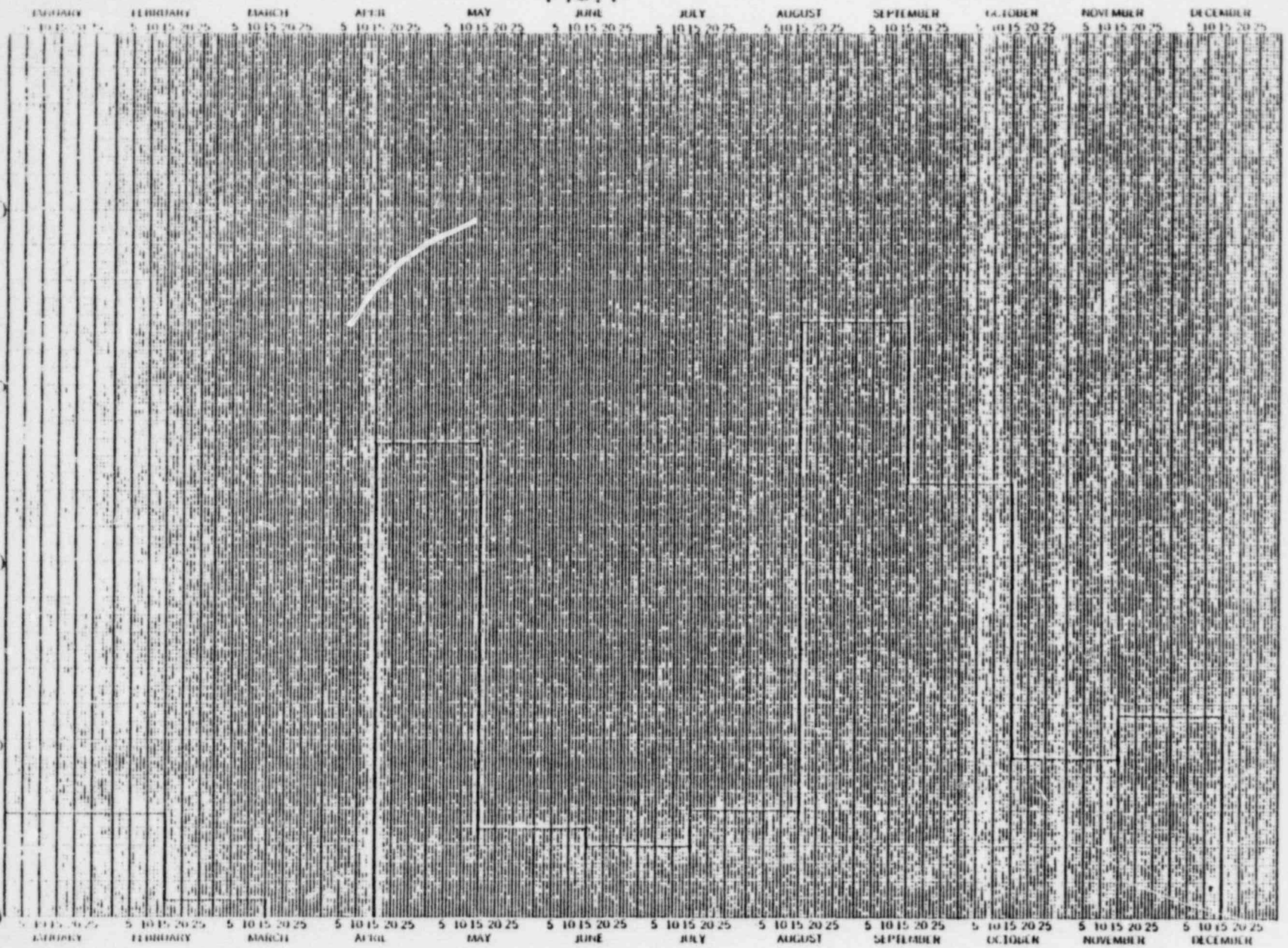
46 2890

FIG. 1



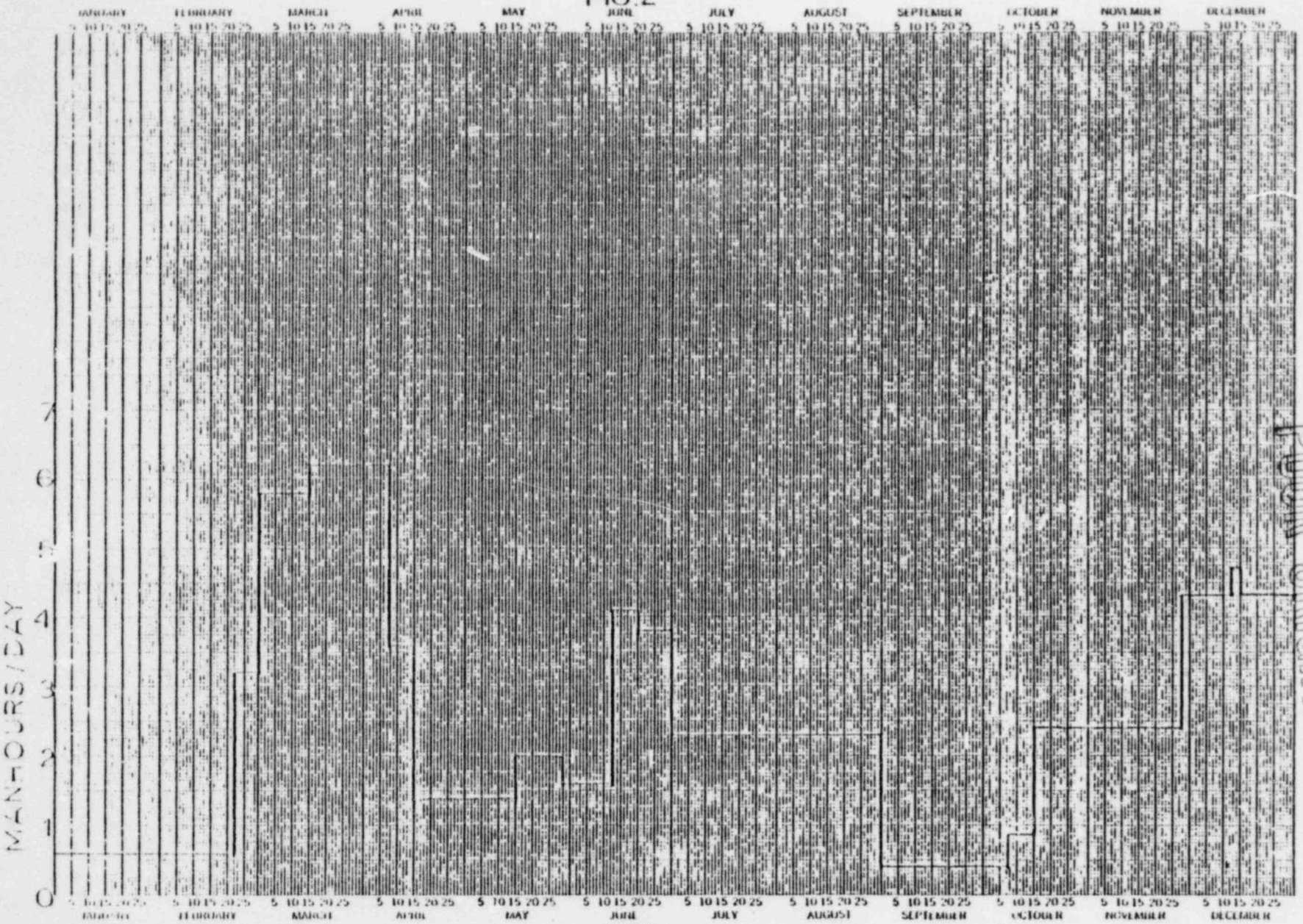
1977

FIG. 1



1978

FIG. 2



1977

K-E 1 YEAR BY DAYS X 250 DIVISIONS
KEUFFEL & ESSER CO. MADE IN U.S.A.

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FIG. 2

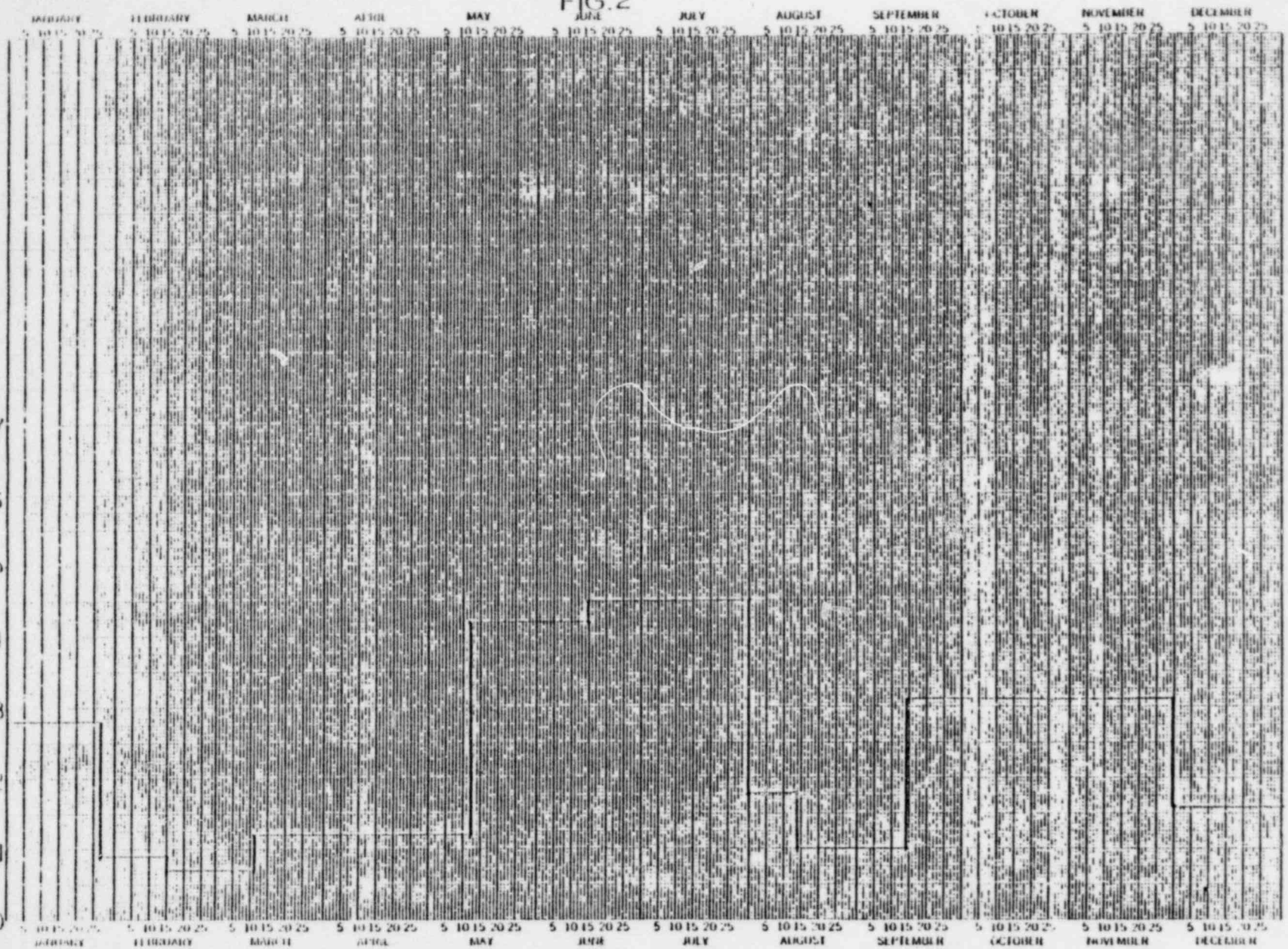


FIG.3

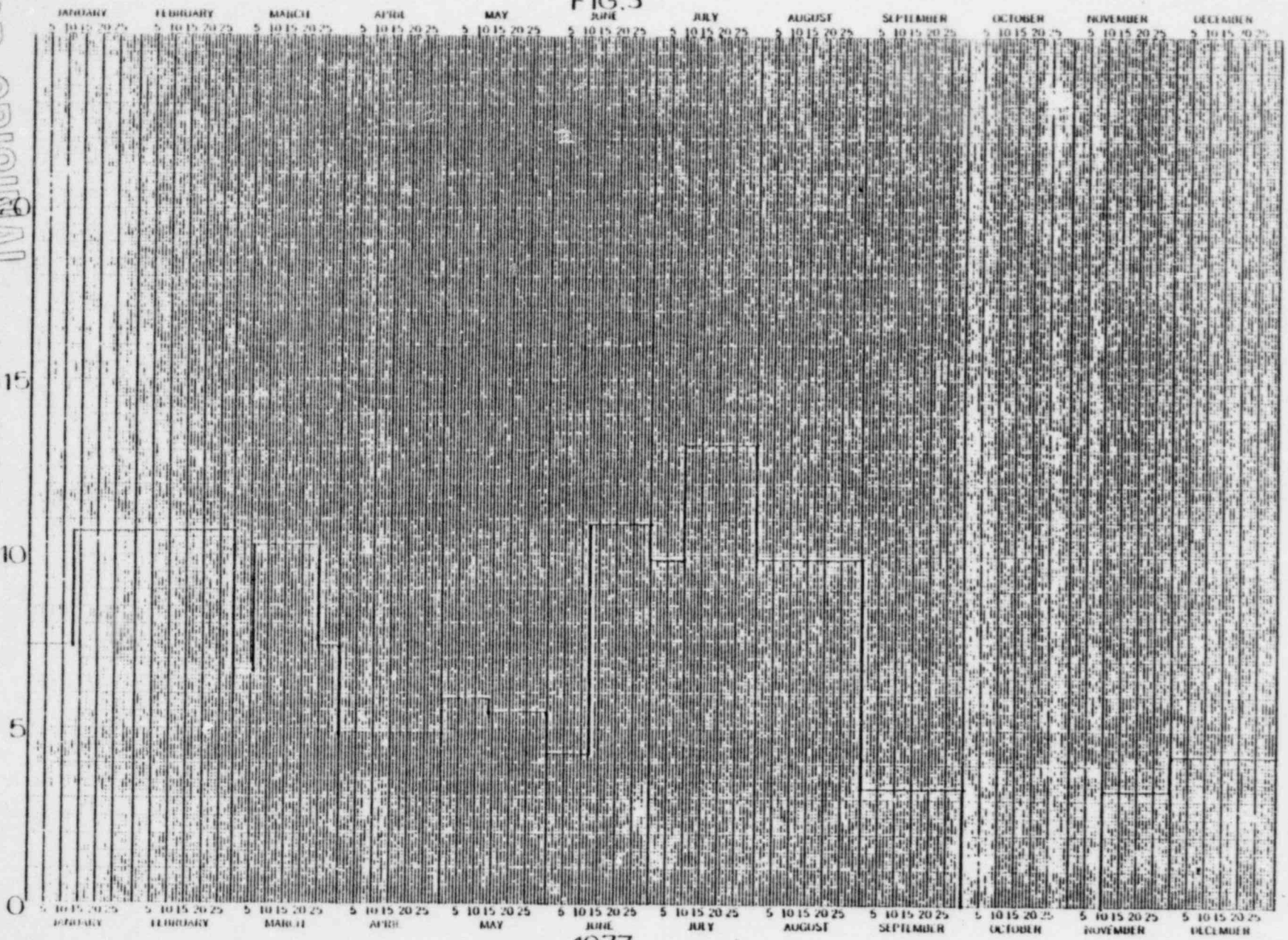
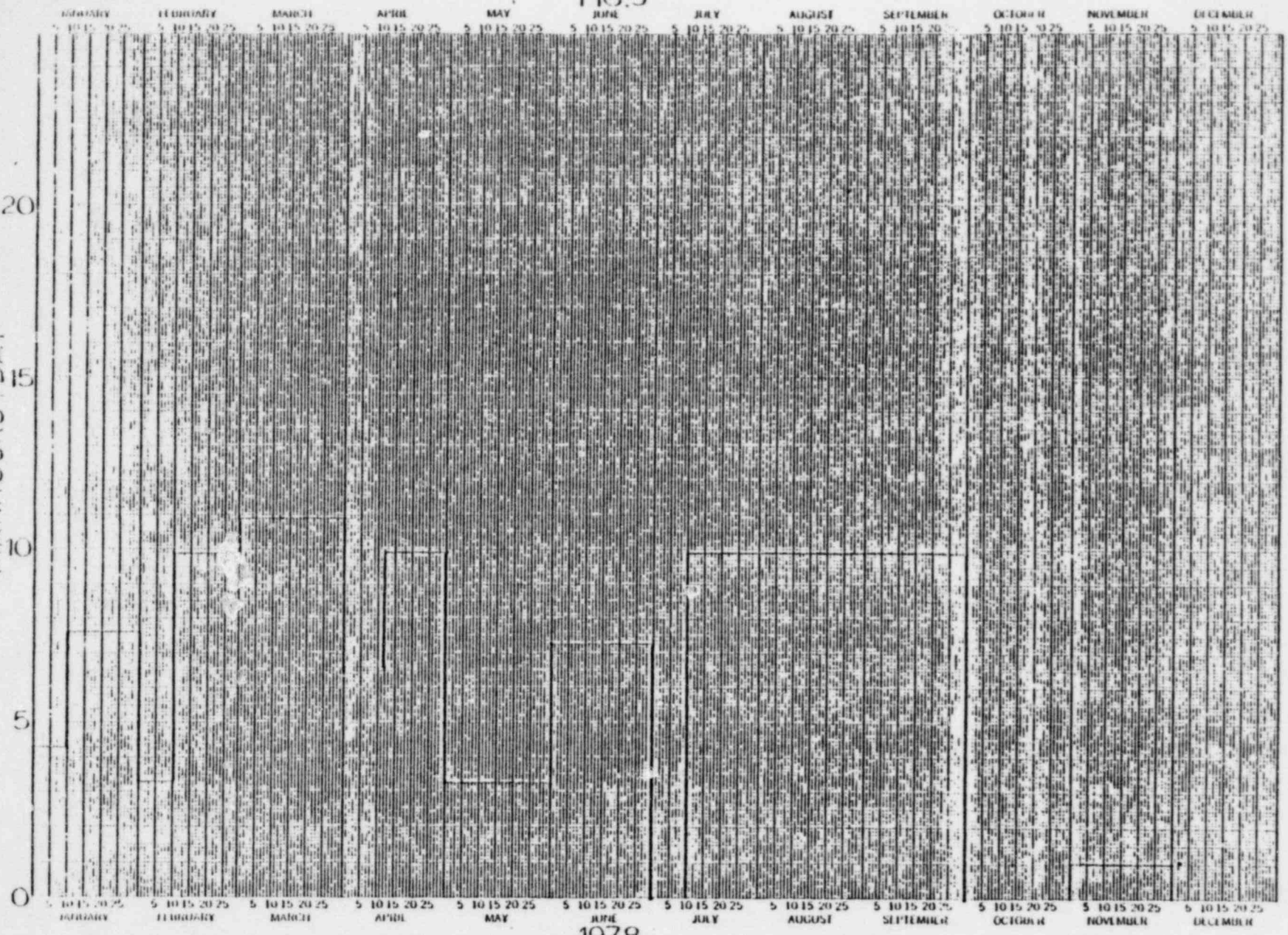


FIG. 3



1978

K-E 1 YEAR BY DAYS X 250 DIVISIONS
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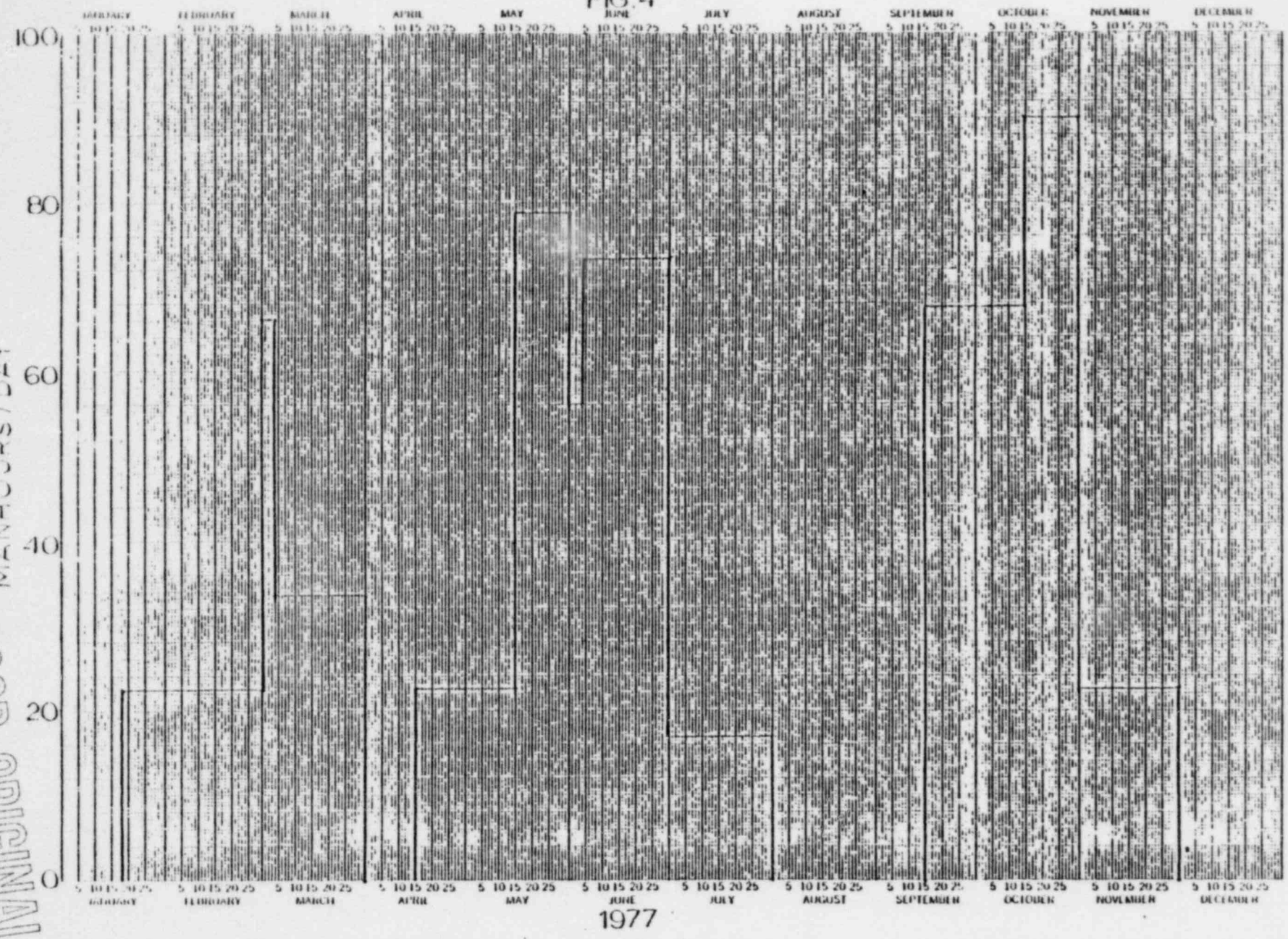
46 2890

FIG. 4



1977

K-E 1 YEAR BY DAYS X 250 DIVISIONS
KEUFFEL & ESSER CO. MADE IN U.S.A.

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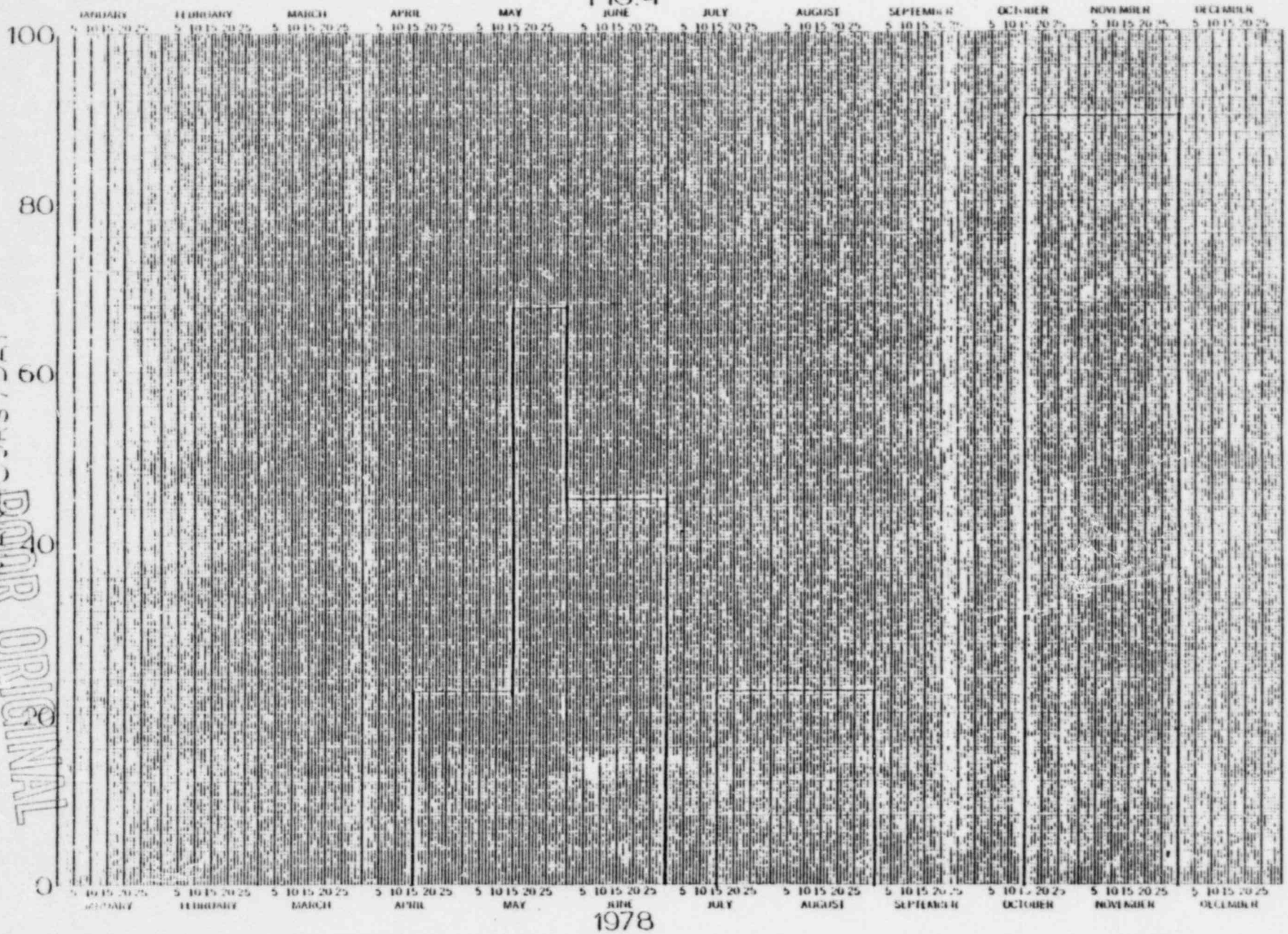
FIG. 4



K-E 1 YEAR BY DAYS X 250 DIVISION.
KEUFFEL & ESSER CO. MADE IN U.S.A.

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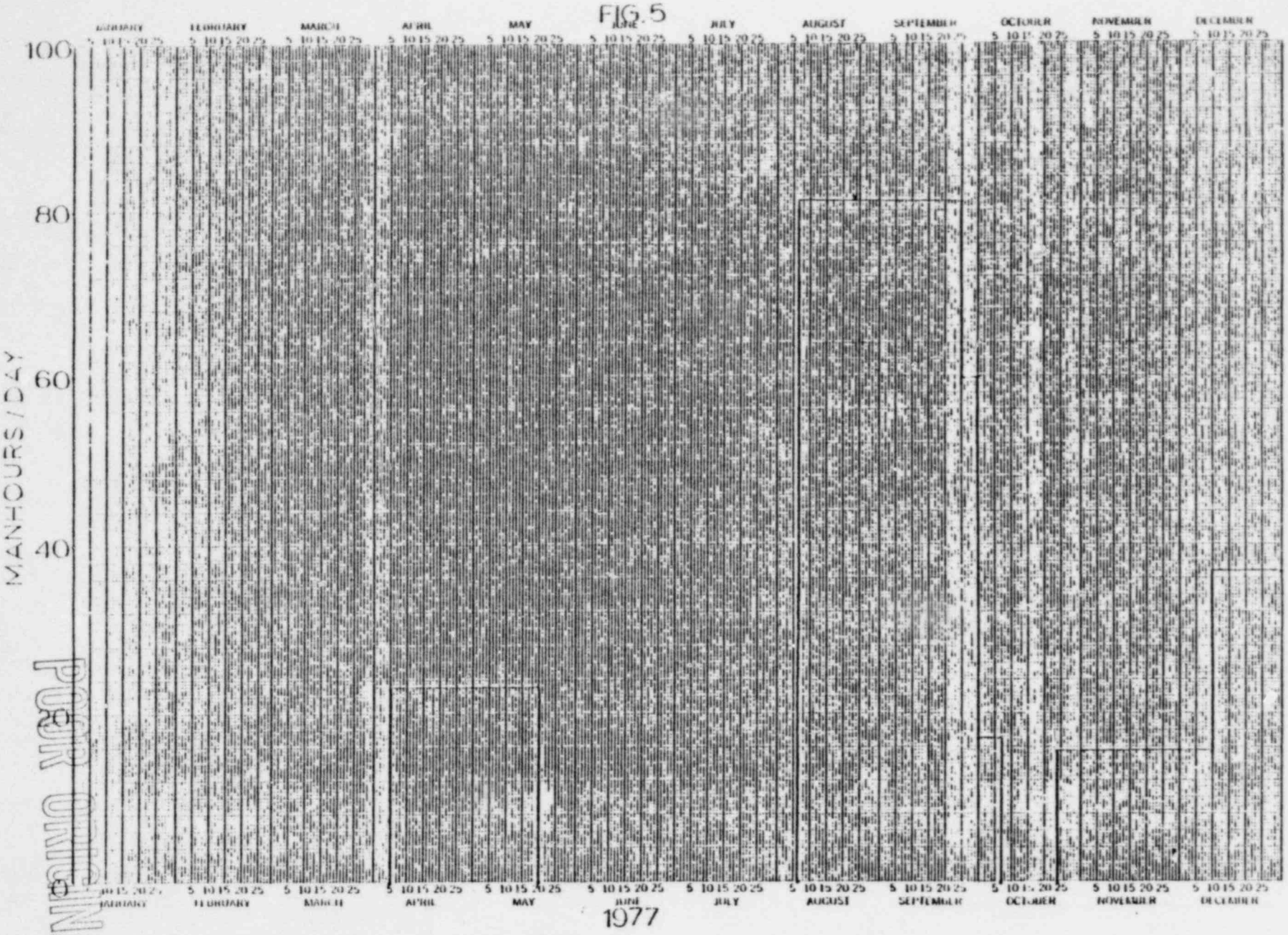
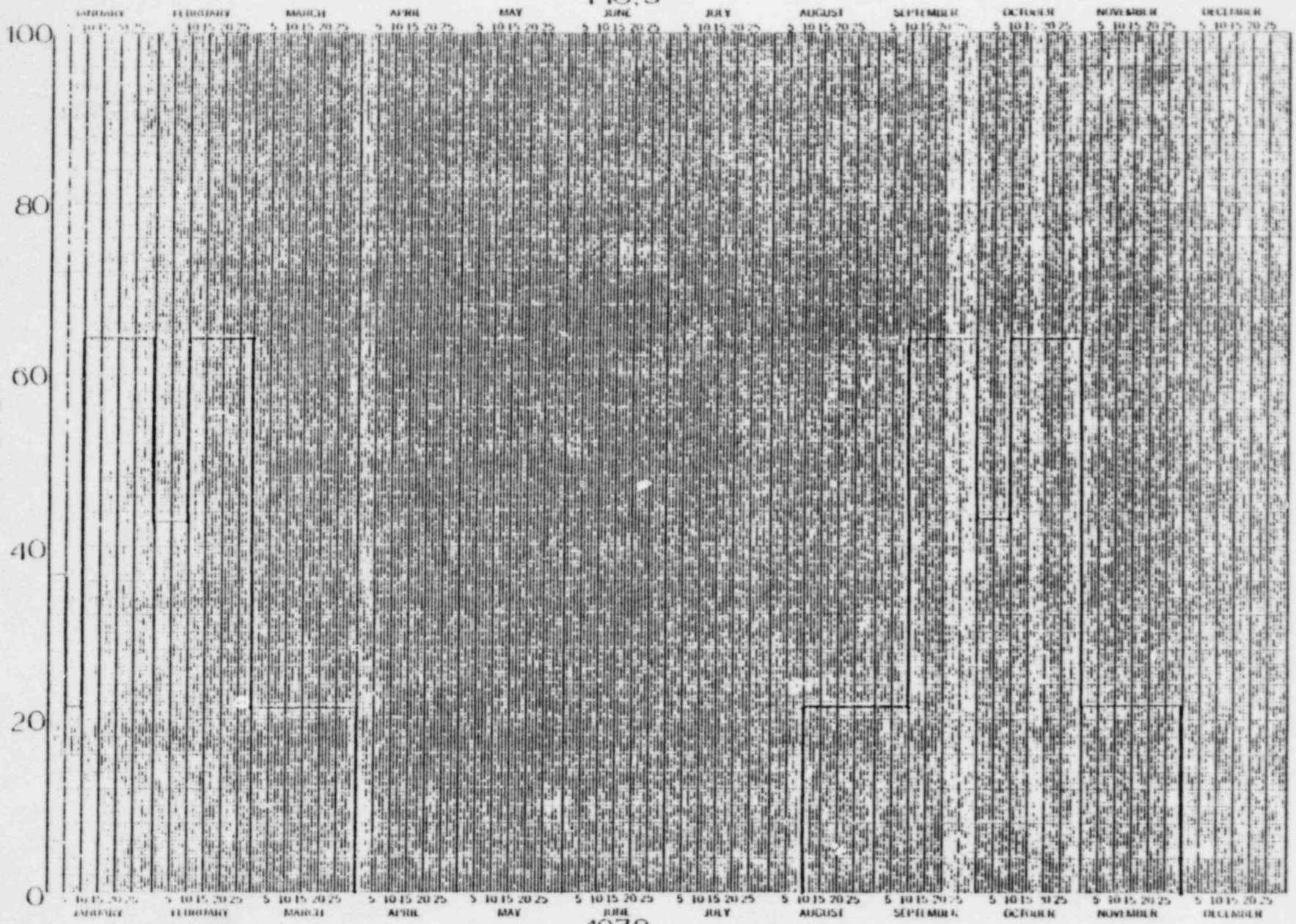
FIG. 5



1977

FIG. 5



1978

K-E 1 YEAR BY DAYS X 250 DIVISIONS
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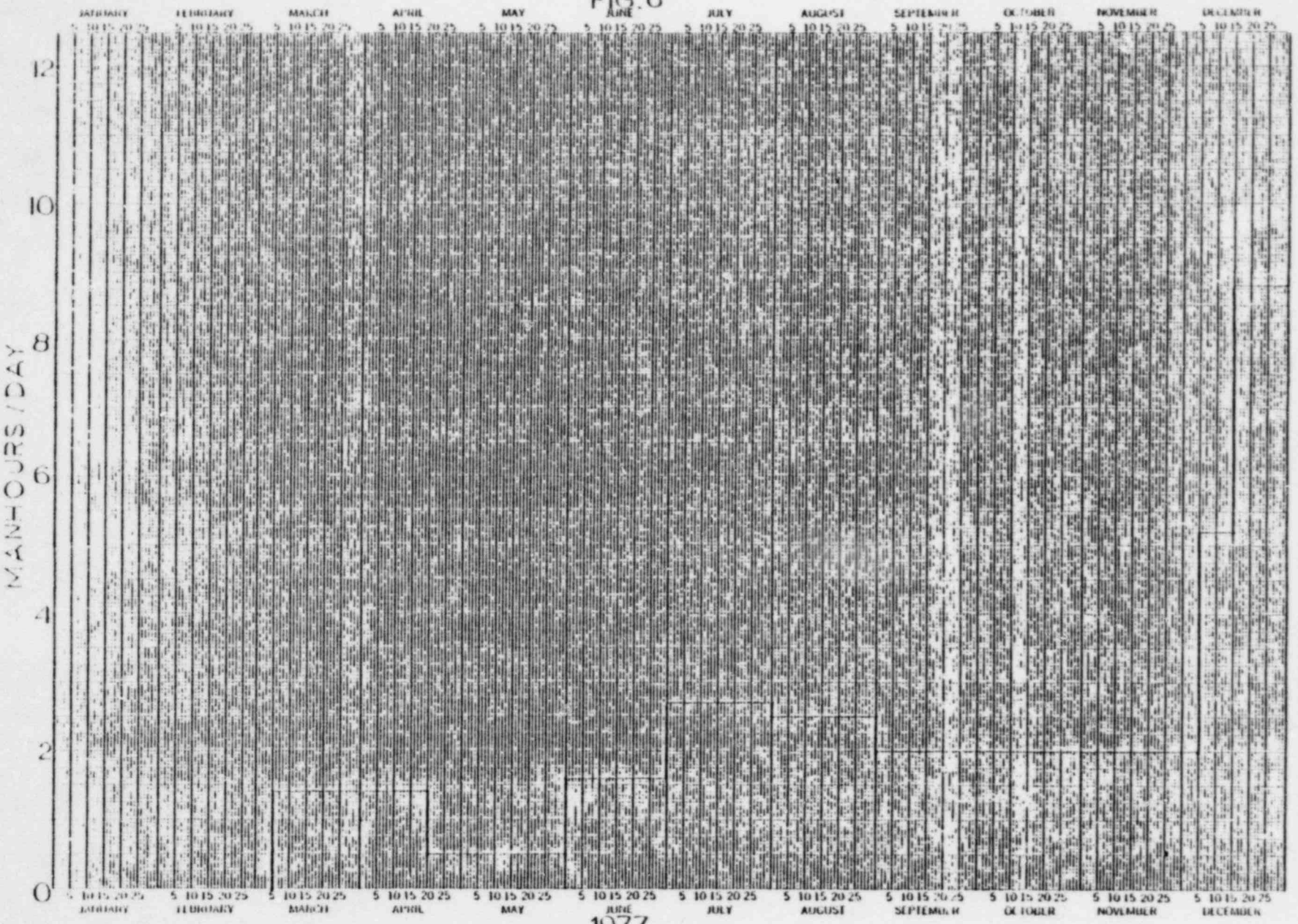
FIG. 6



1977

K&E 1 YEAR BY DAYS X 250 DIVISIONS
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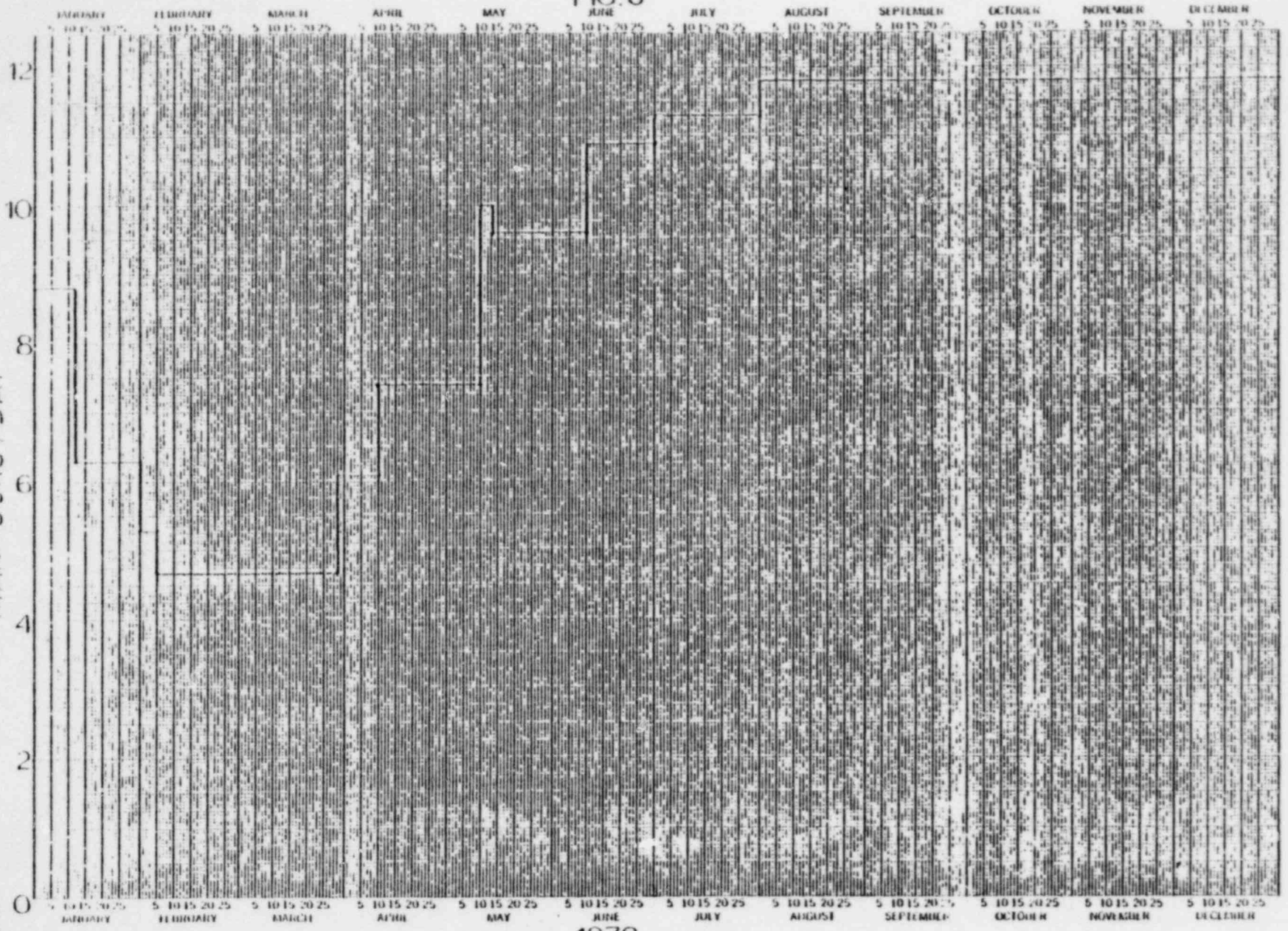
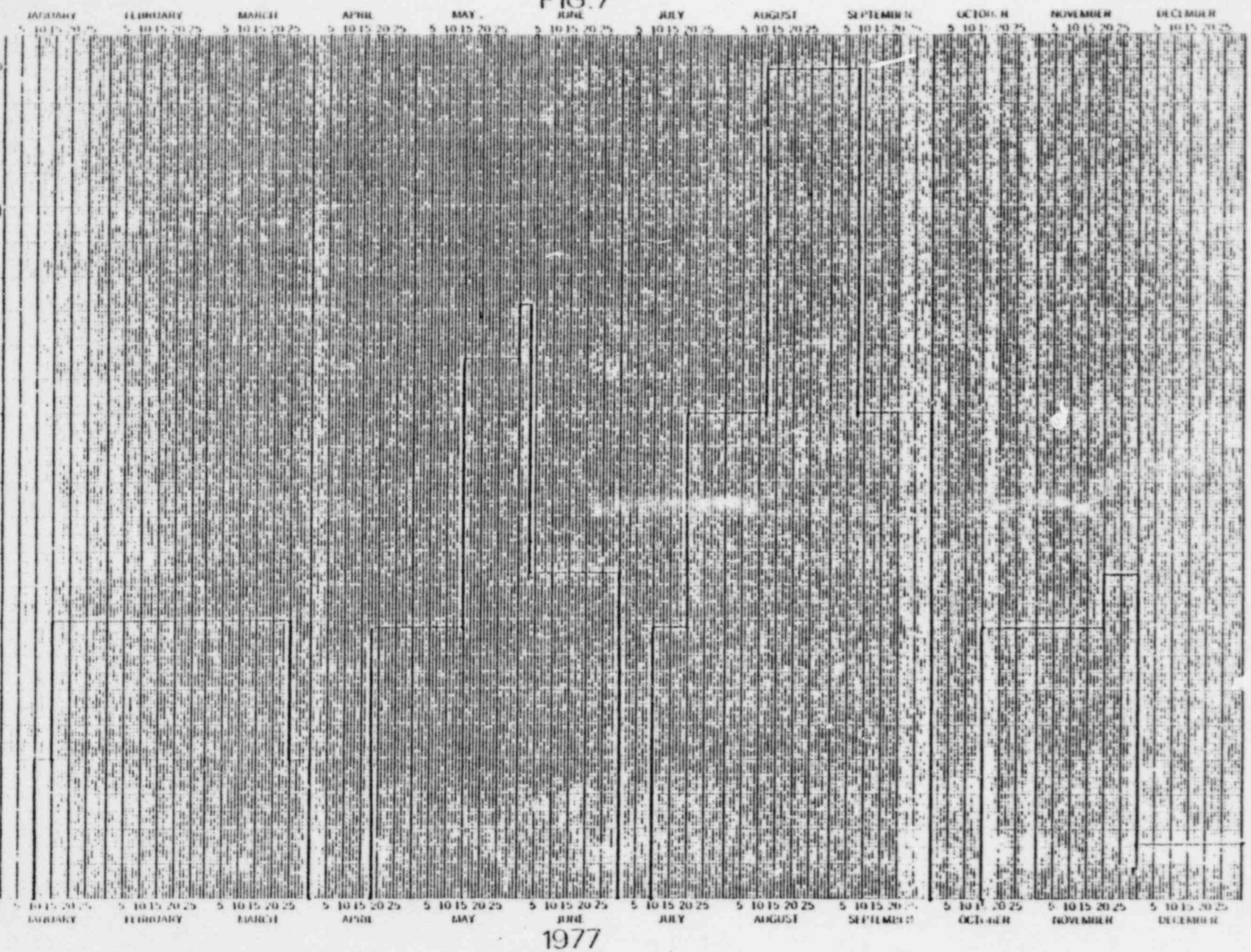
FIG. 6



1978

FIG. 7



K-E 1 YEAR BY DAYS X 250 DIVISIONS
KEUFFEL & ESSER CO.

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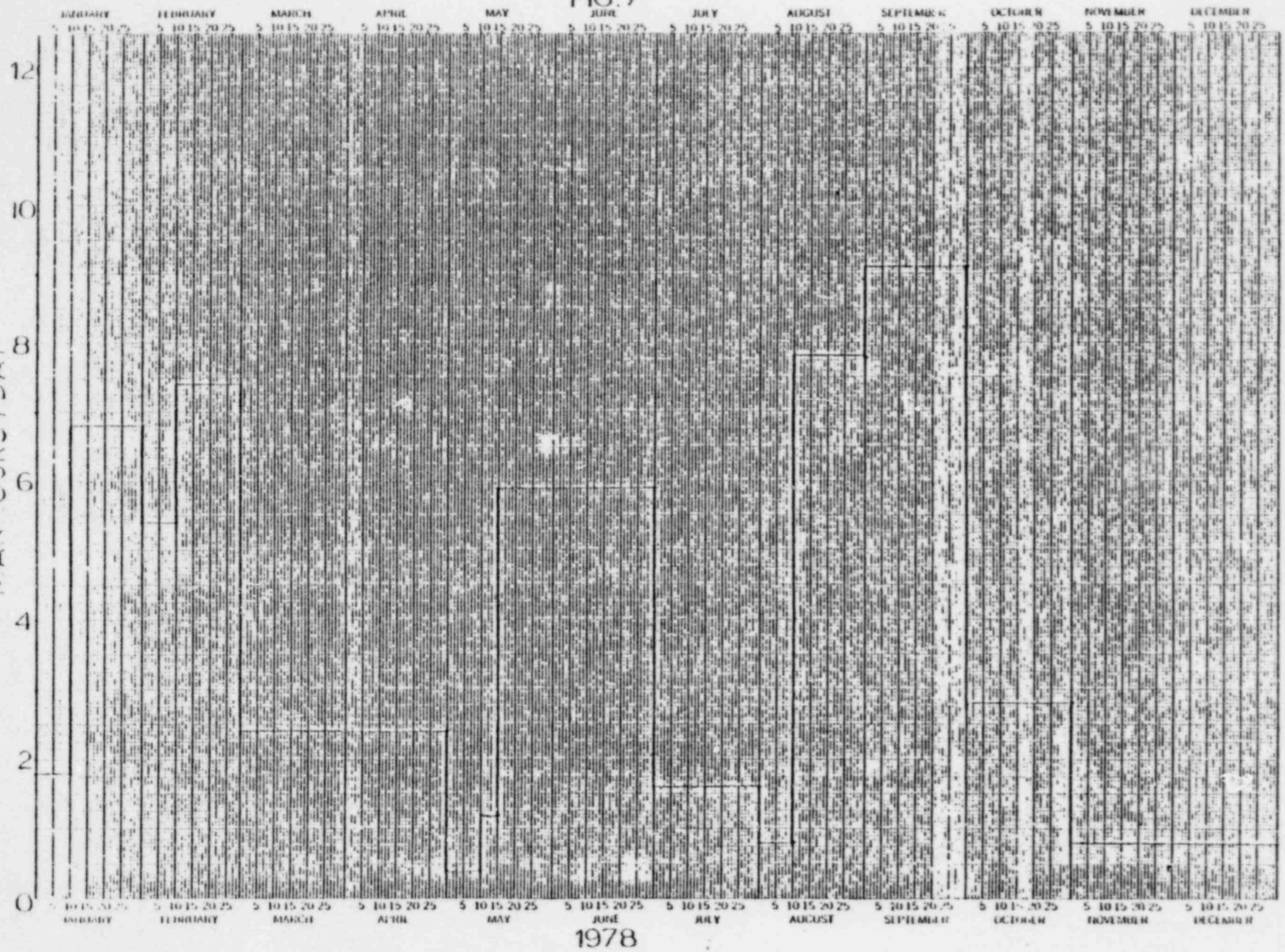
FIG. 7



1978

K-E 1 YEAR BY DAYS X 250 DIVISIONS
KEUFFEL & ESSER CO MADE IN U.S.A.

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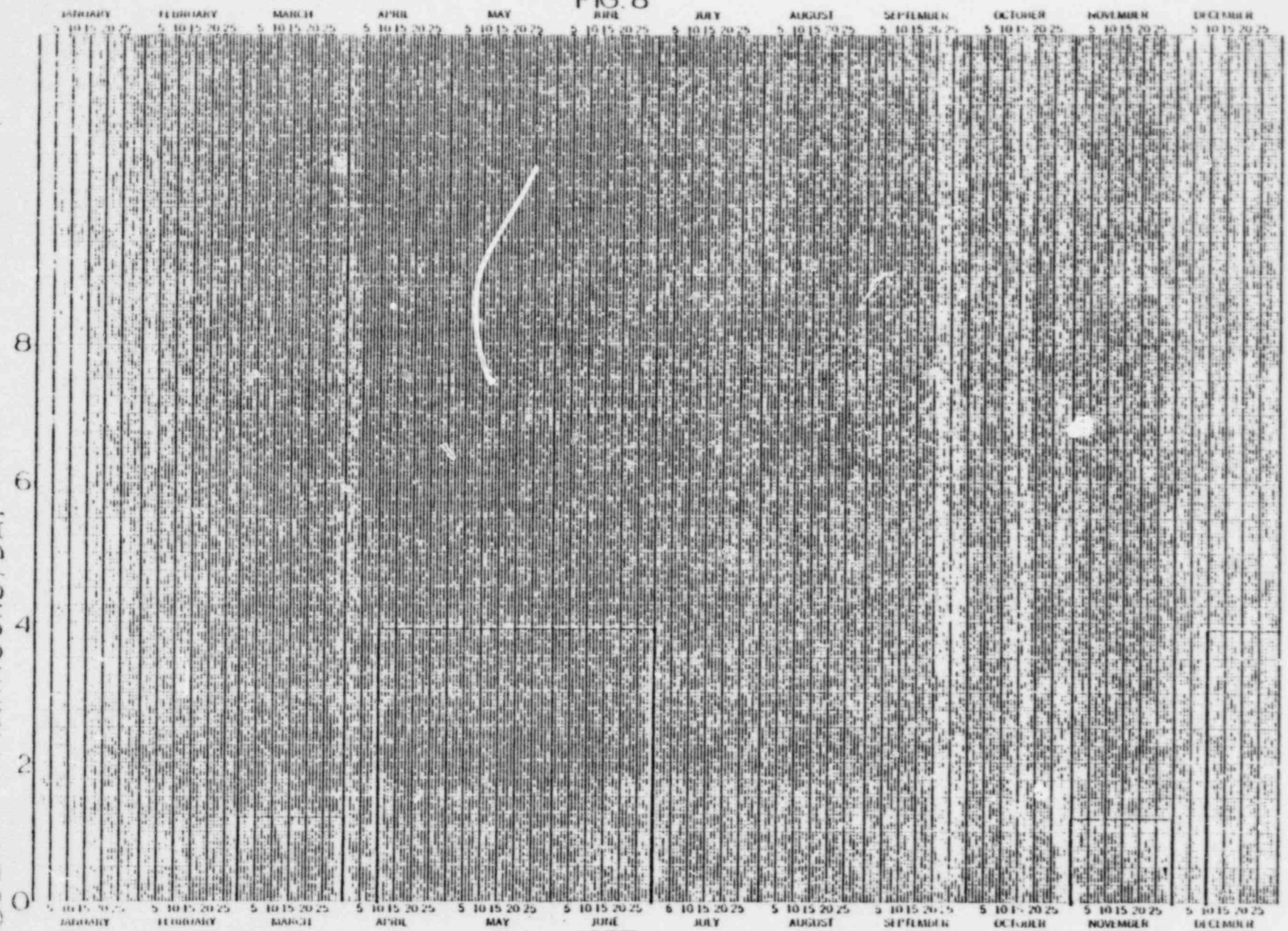
FIG. 8



1977

K&E 1 YEAR BY DAYS X 250 DIVISIONS
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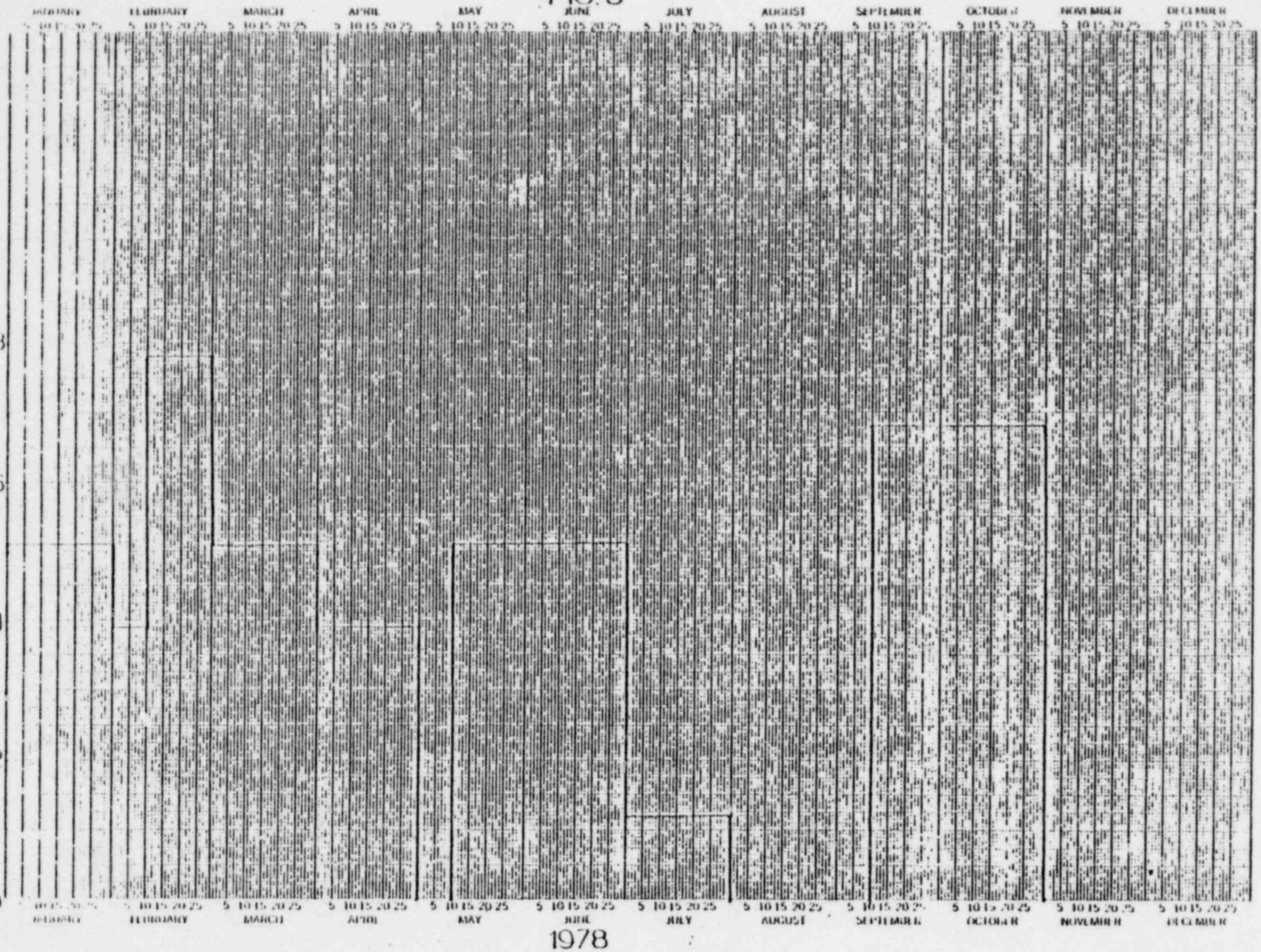
FIG. 8



1 YEAR BY DAYS X 250 DIVISIONS
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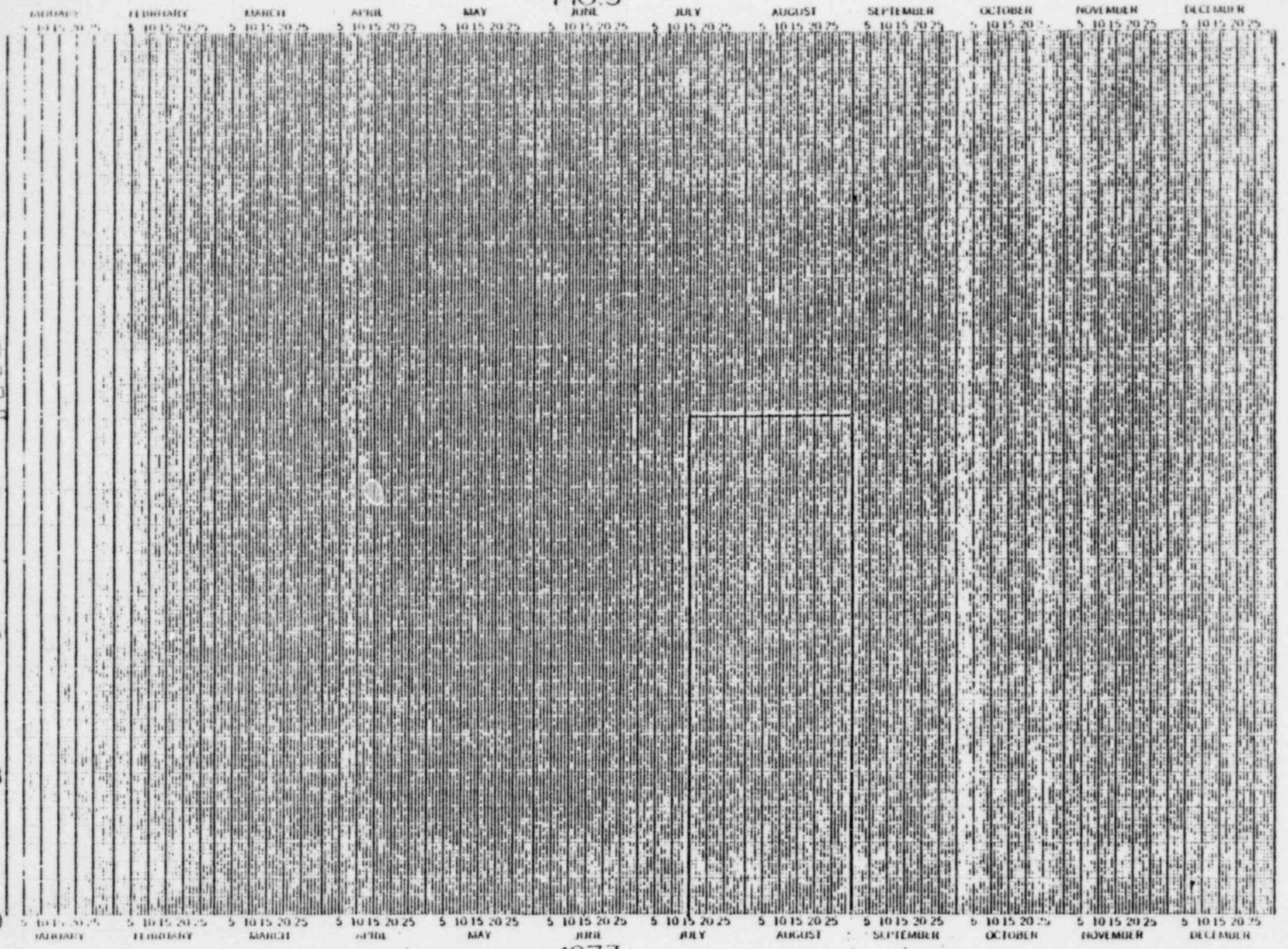
FIG.9



K-E 1 YEAR BY DAYS X 250 DIVISIONS
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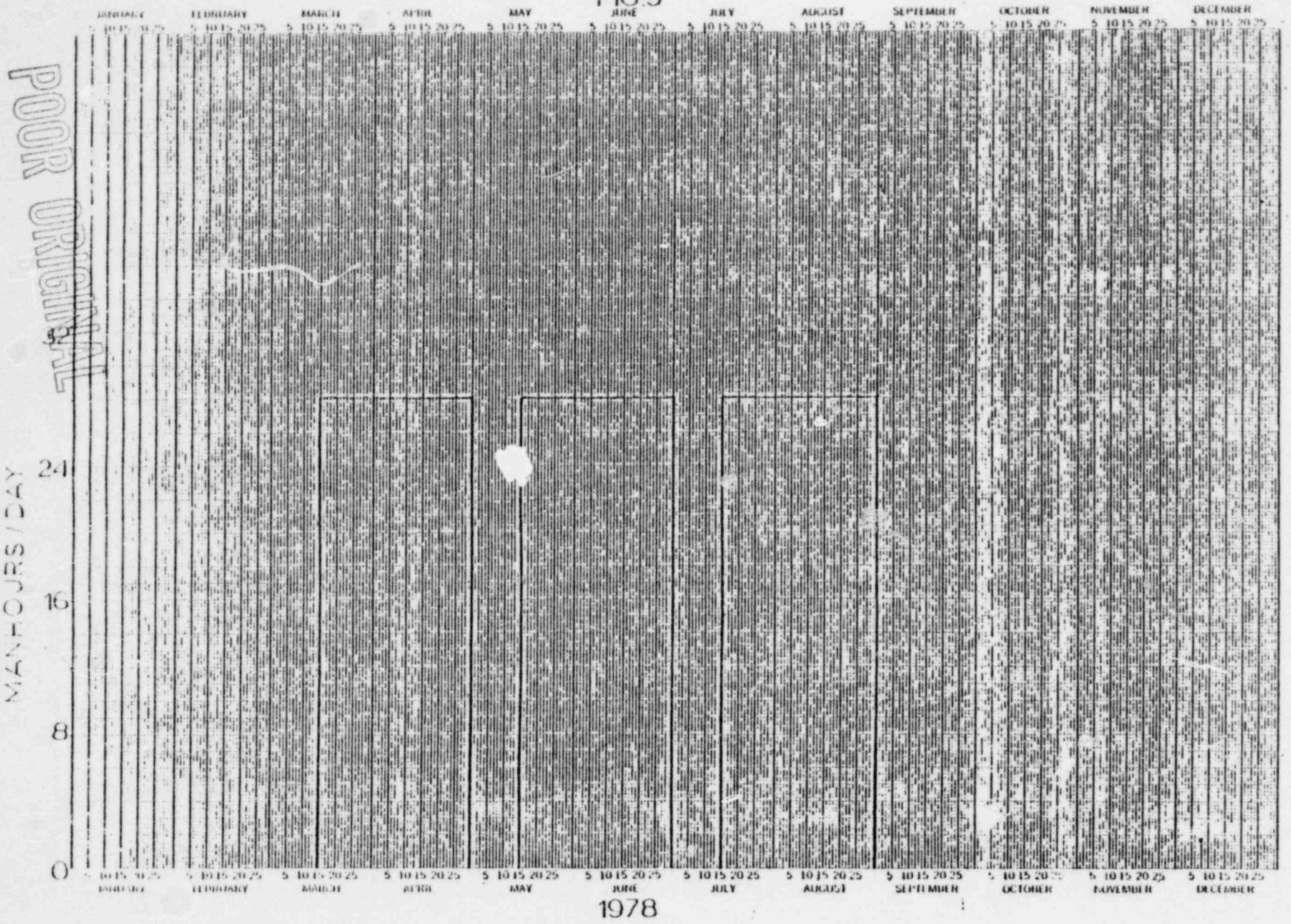
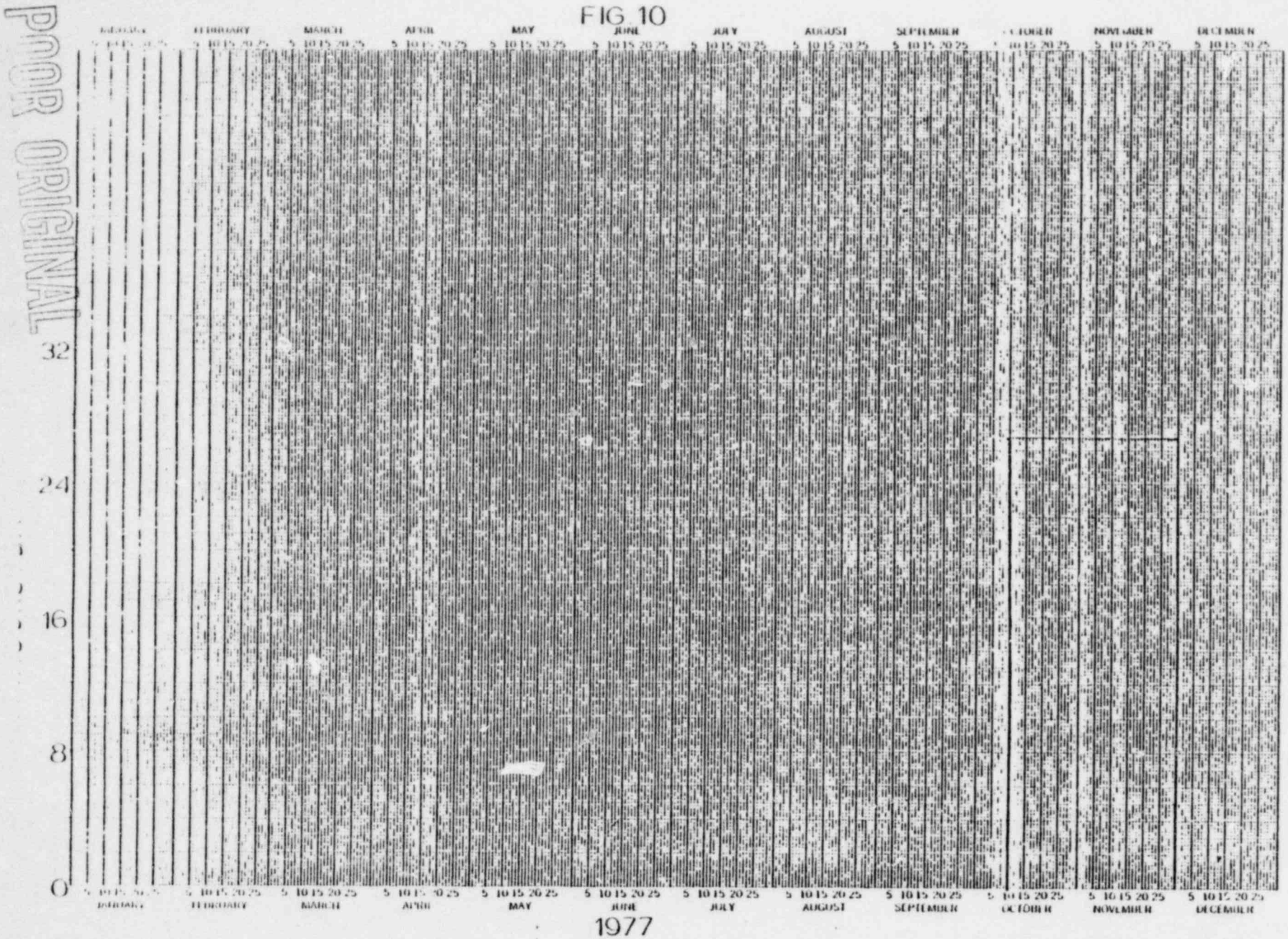
FIG 9



K-E 1 YEAR BY DAYS X 250 DIVISIONS

46 2890

FIG 10



1977

K-E 1 YEAR BY DAYS X 250 DIVISIONS
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FIG. 10

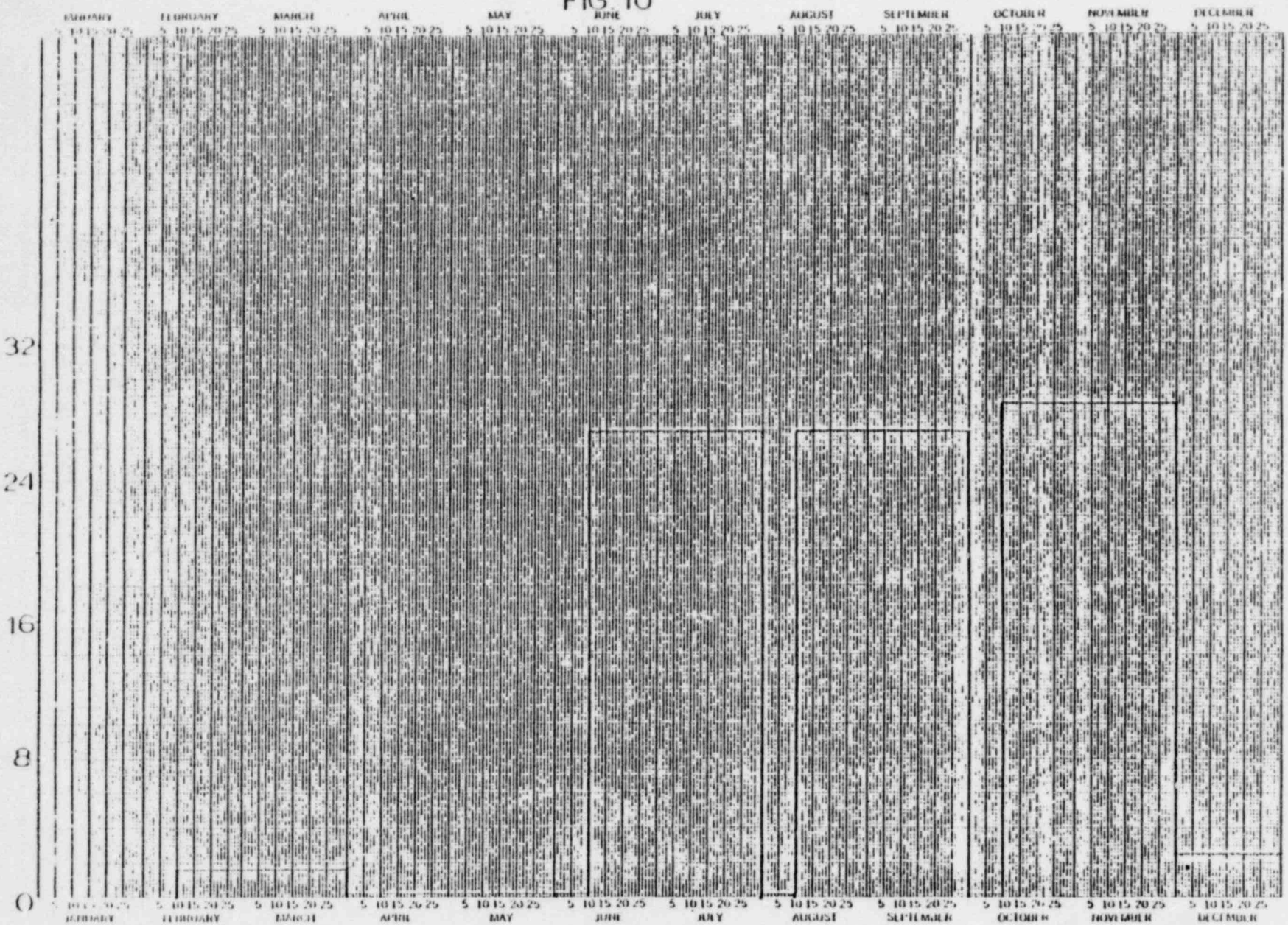


FIG. 11

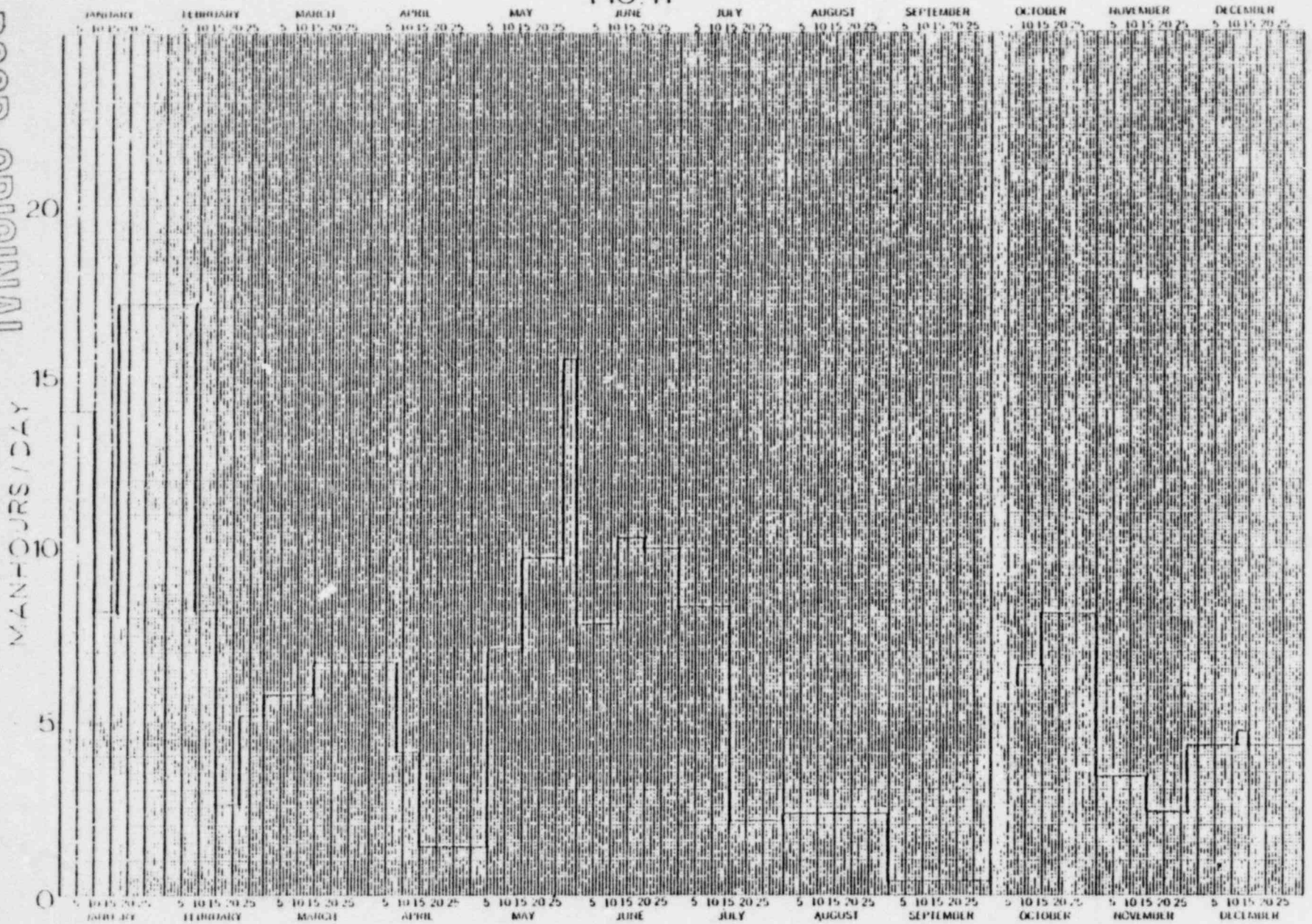
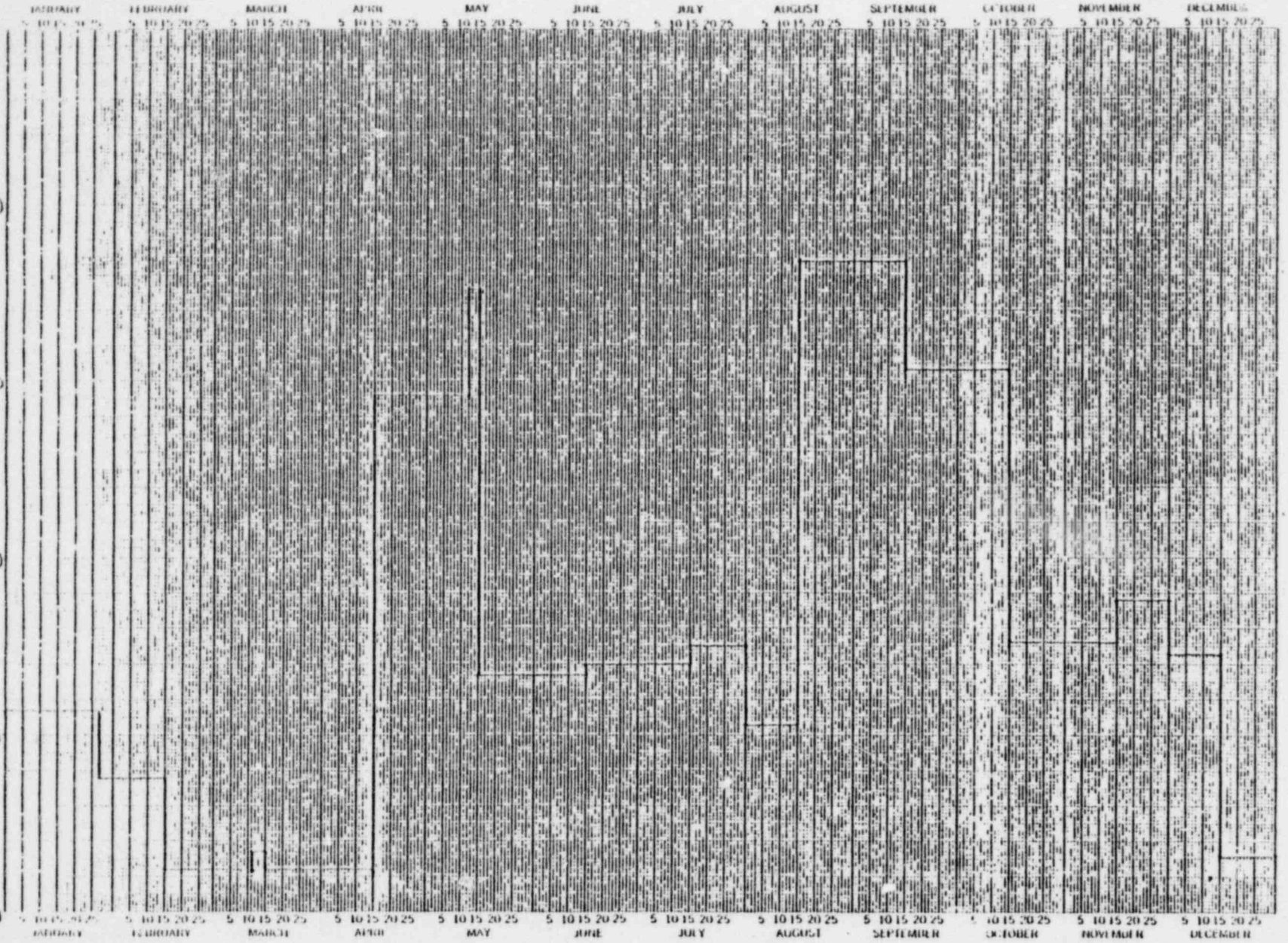


FIG. 11



1978

K-E 1 YEAR BY DAYS X 250 DIVISIONS
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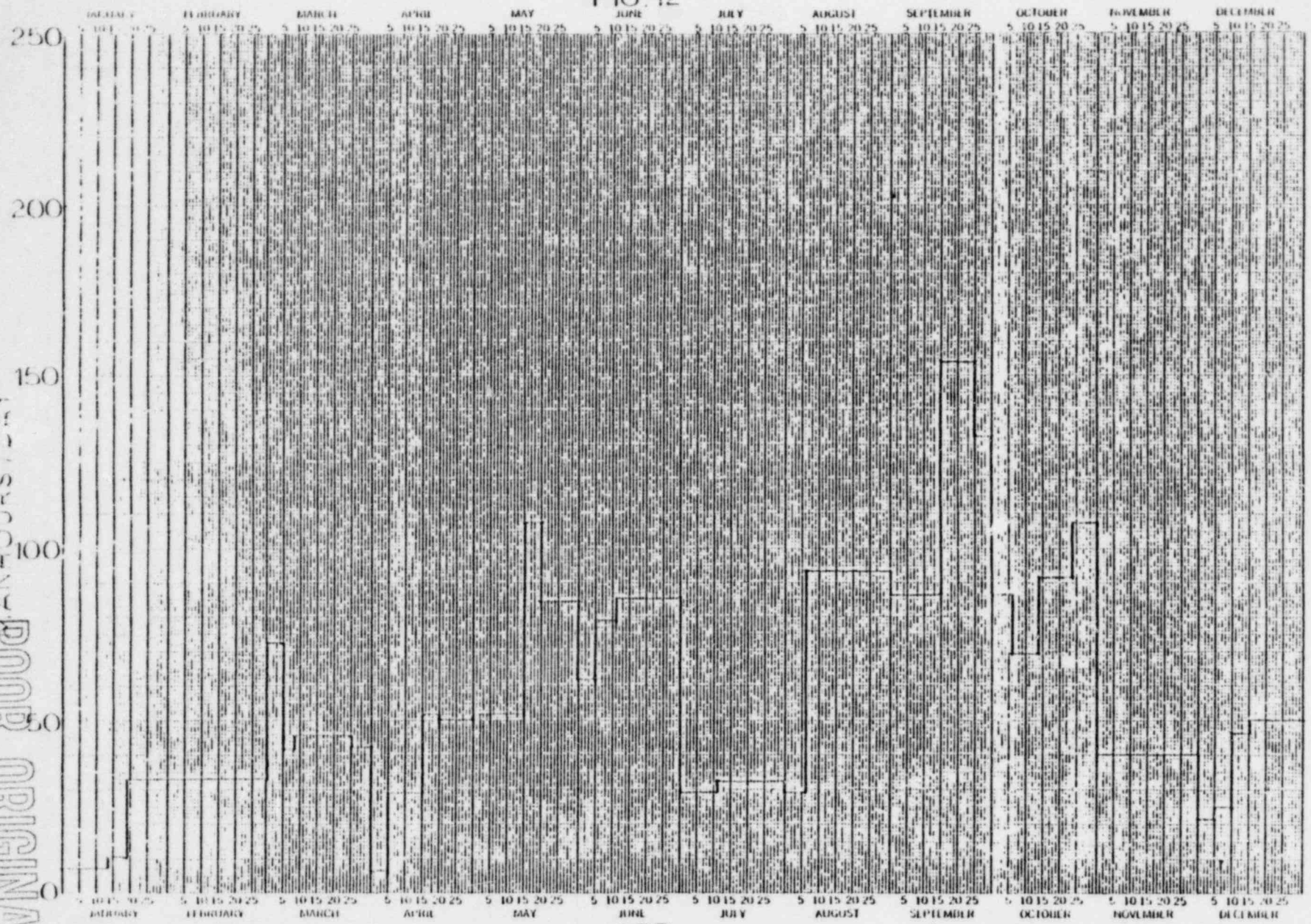
FIG. 12

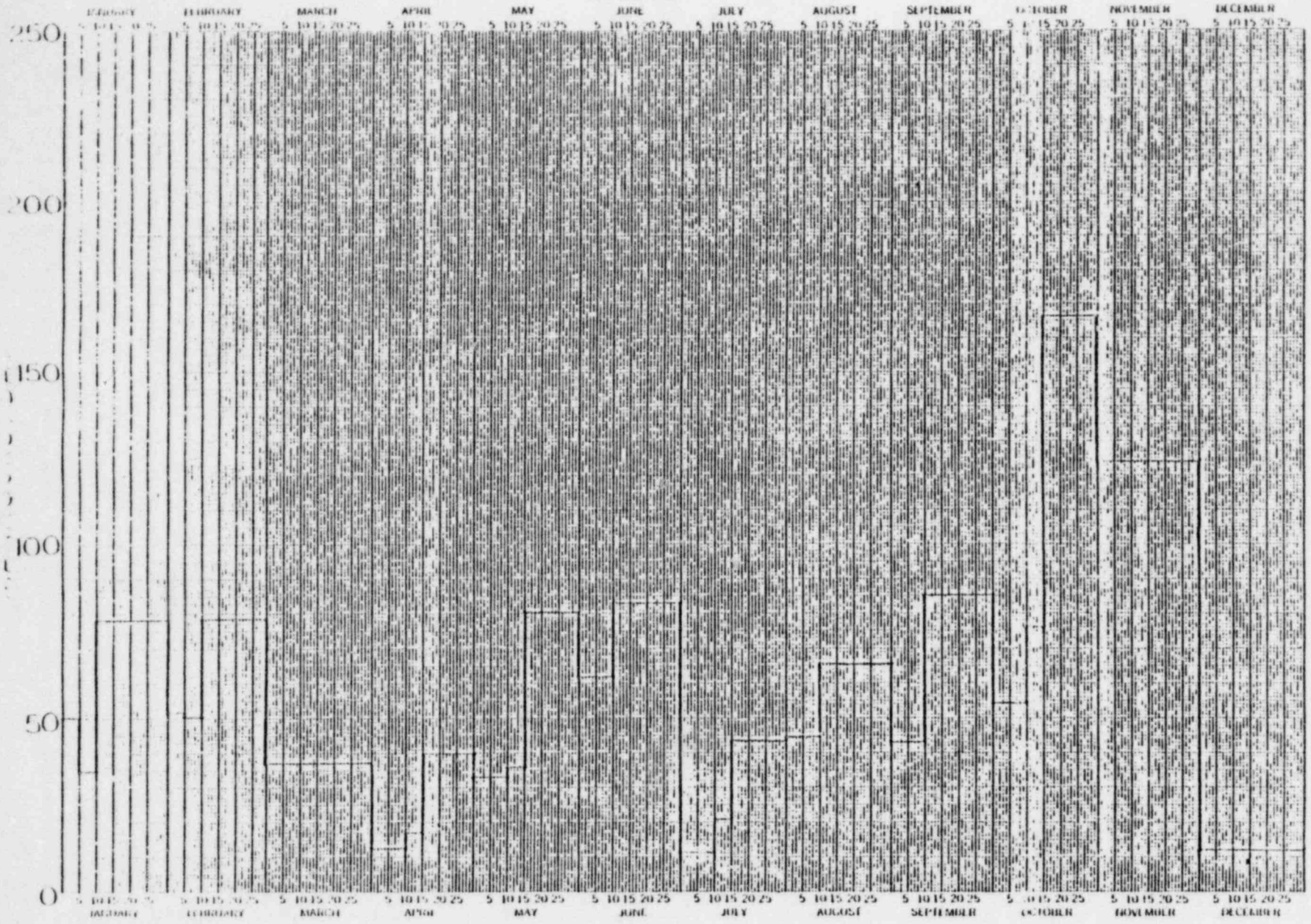


1977

K-Σ 1 YEAR BY DAYS X 250 DIVISIONS

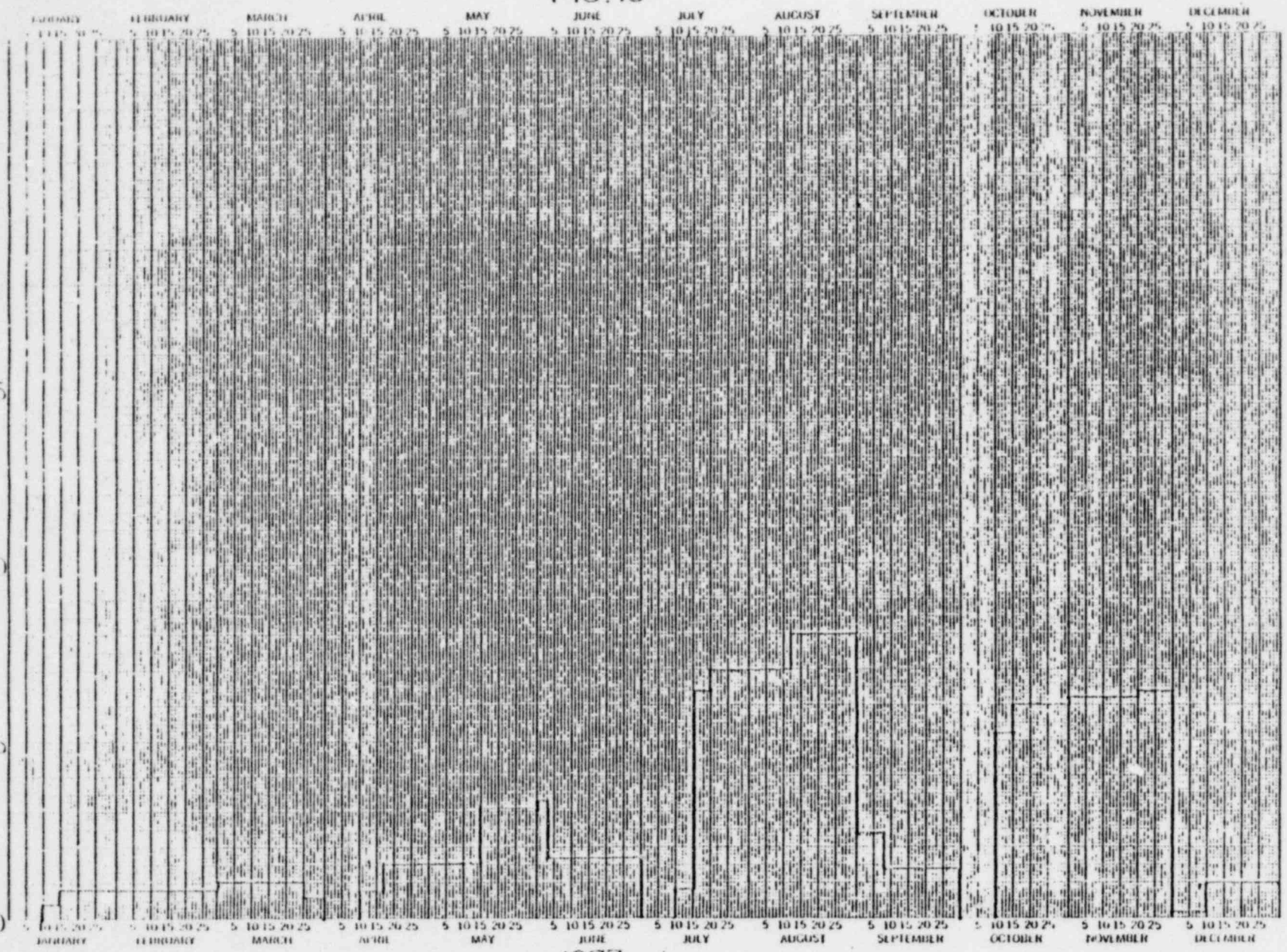
46 2890

FIG. 12



1978

FIG. 13



K-E 1 YEAR BY DAYS X 250 DIVISIONS
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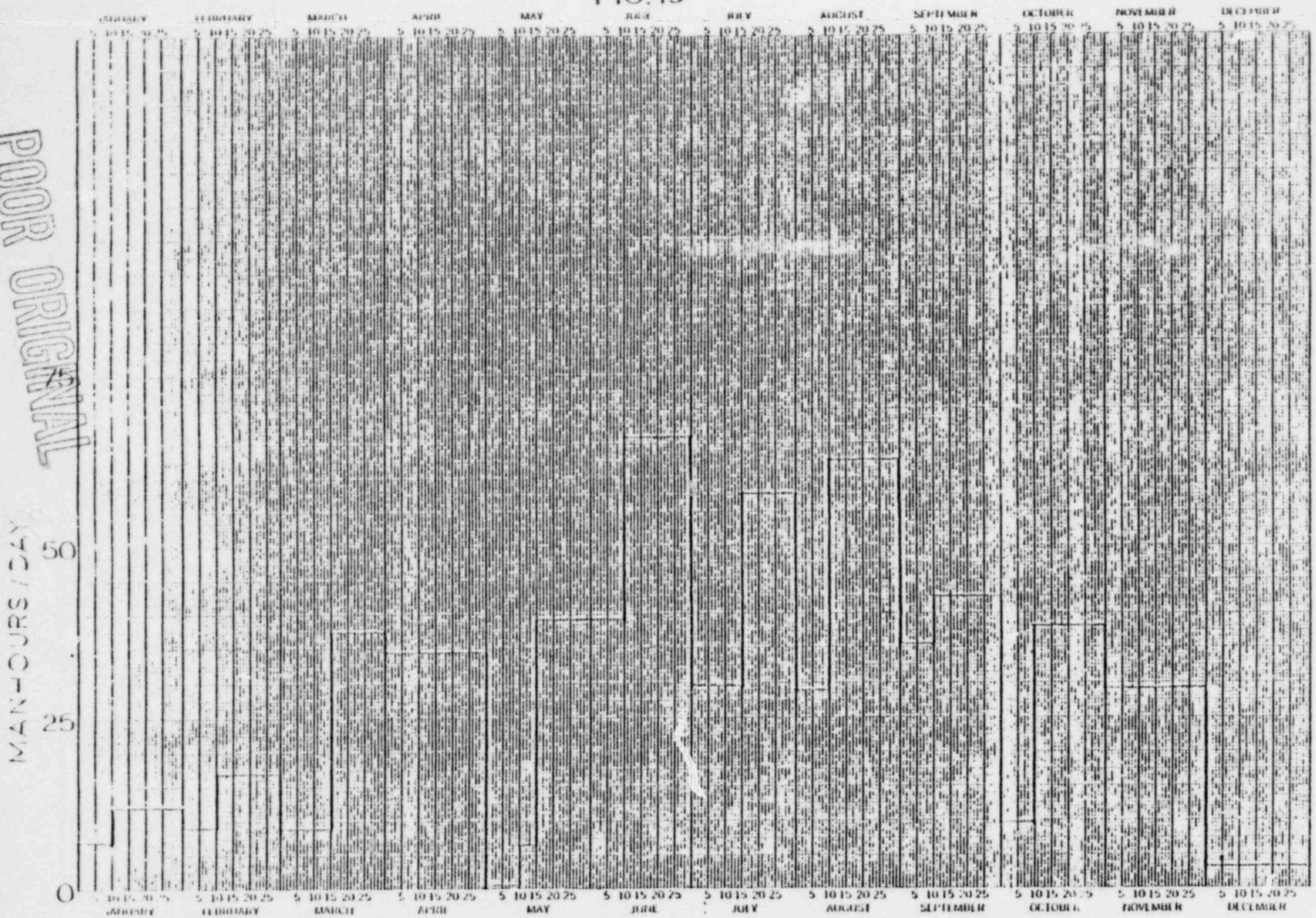
FIG. 13



K-E 1 YEAR BY DAYS X 250 DIVISIONS
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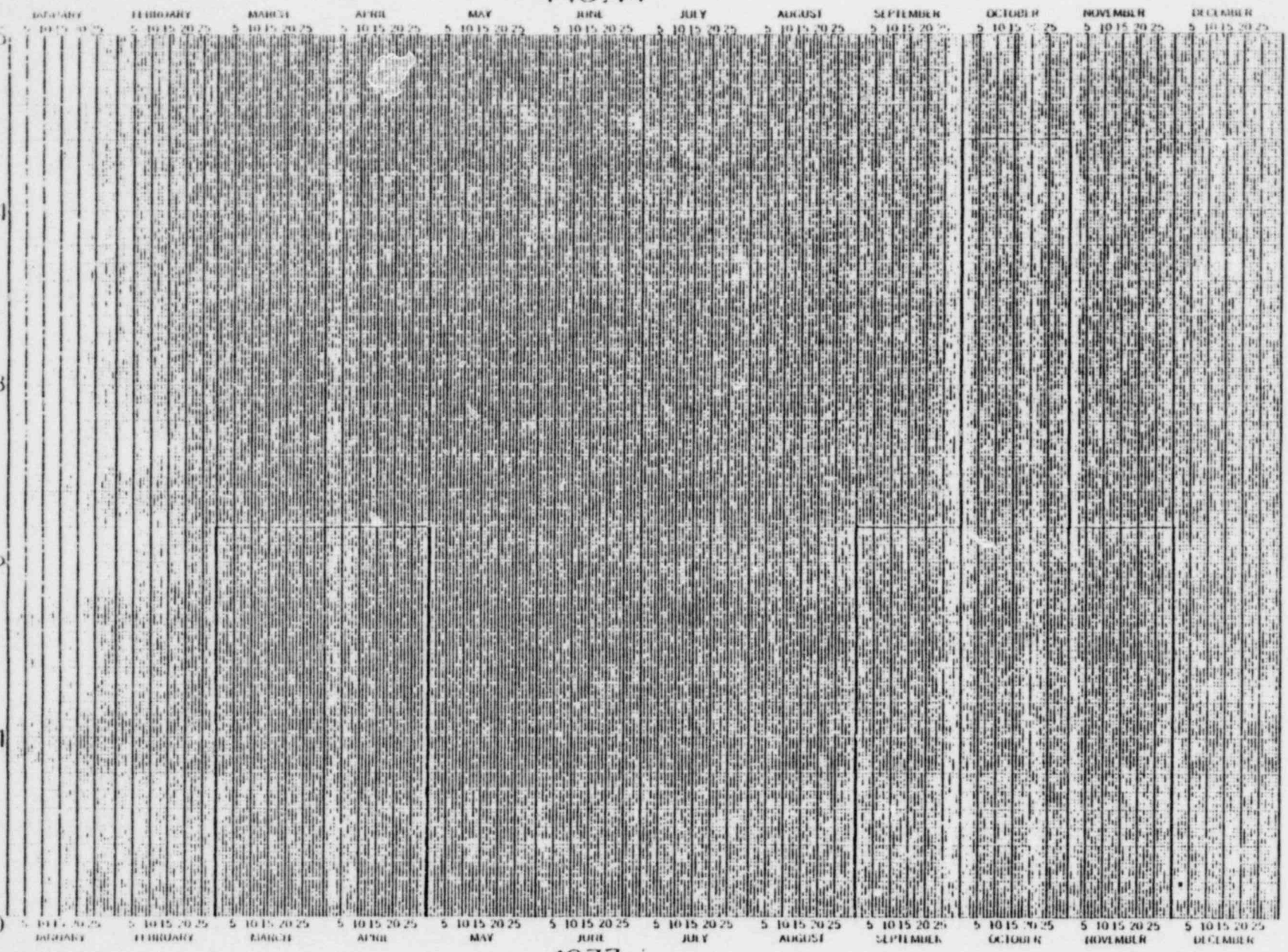
FIG. 14



1977

K&E 1 YEAR BY DAYS X 20 DIVISIONS
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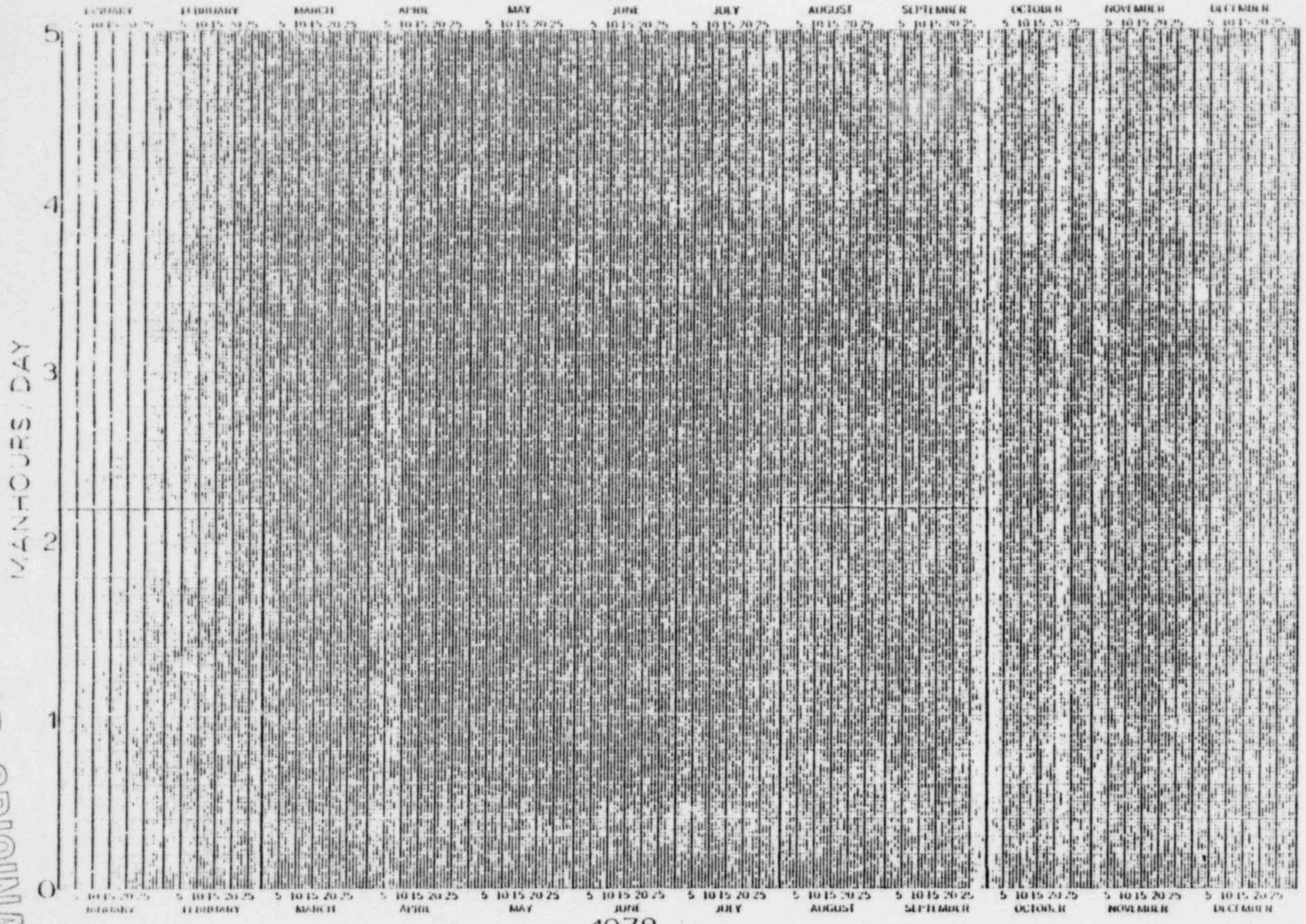
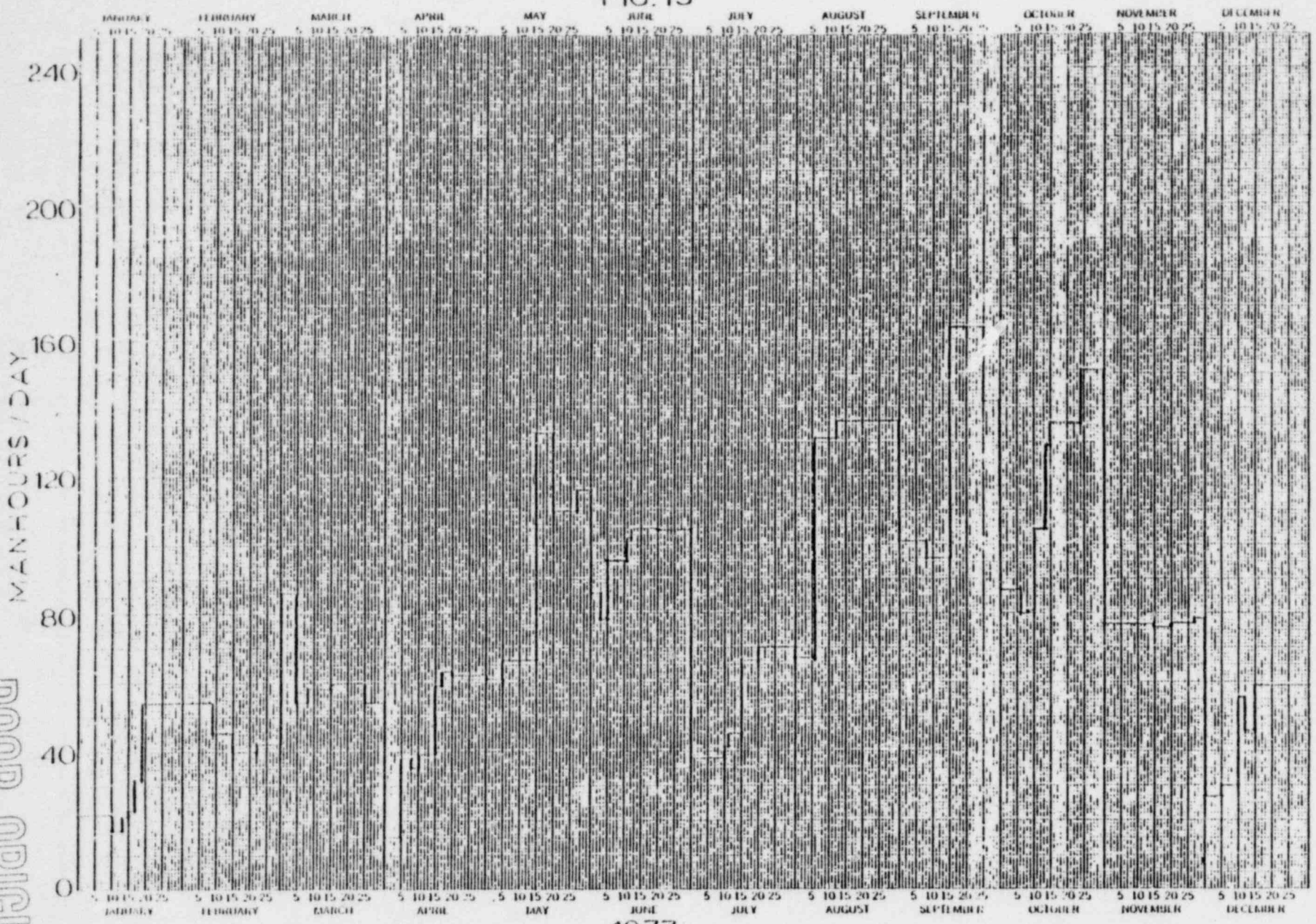
FIG. 14



1978

FIG. 15



K-E 1 YEAR BY DAYS X 250 DIVISIONS
KEUFFEL & ESSER CO. MADE IN U.S.A.

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FIG. 15

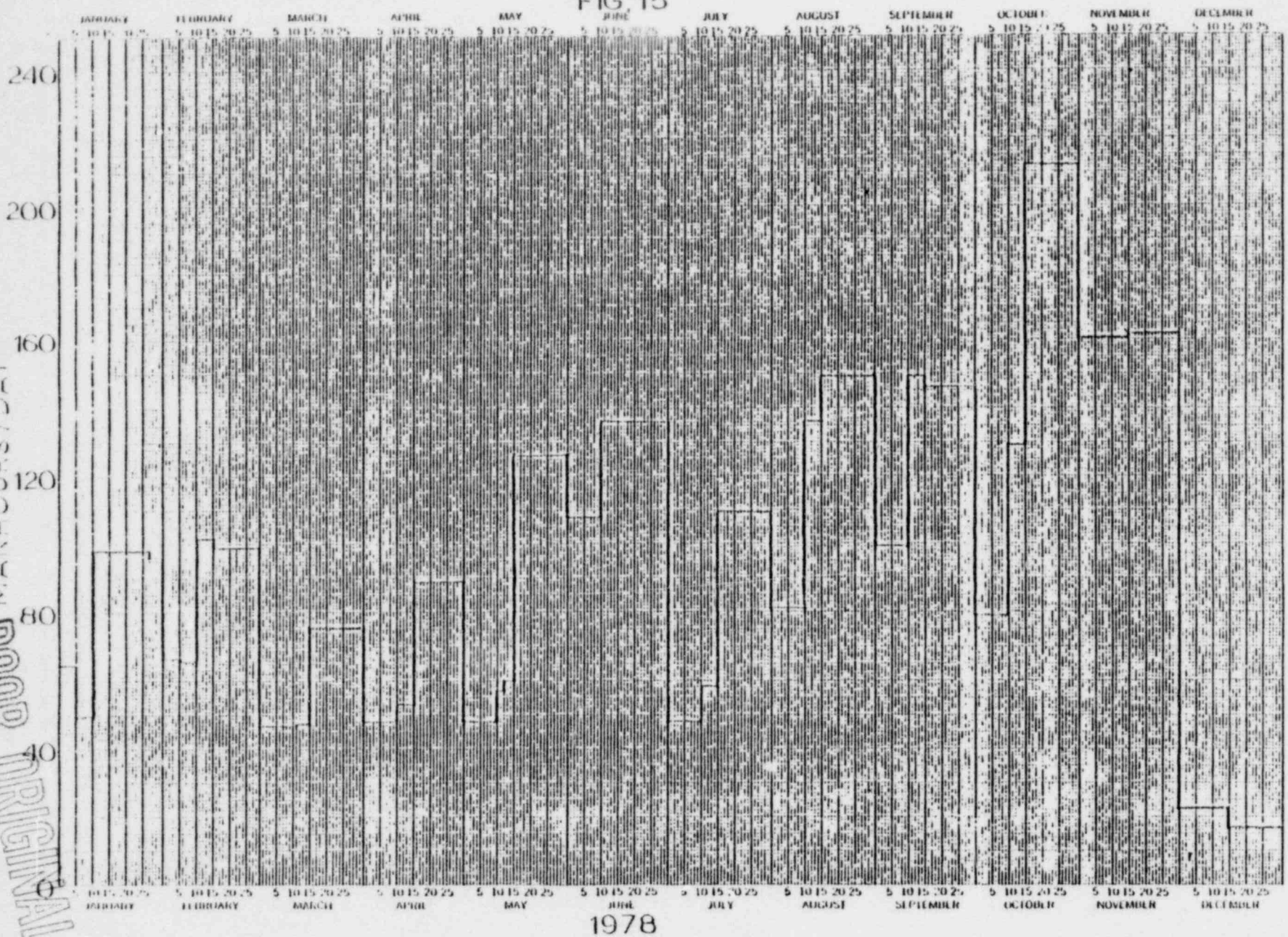


FIG. 16

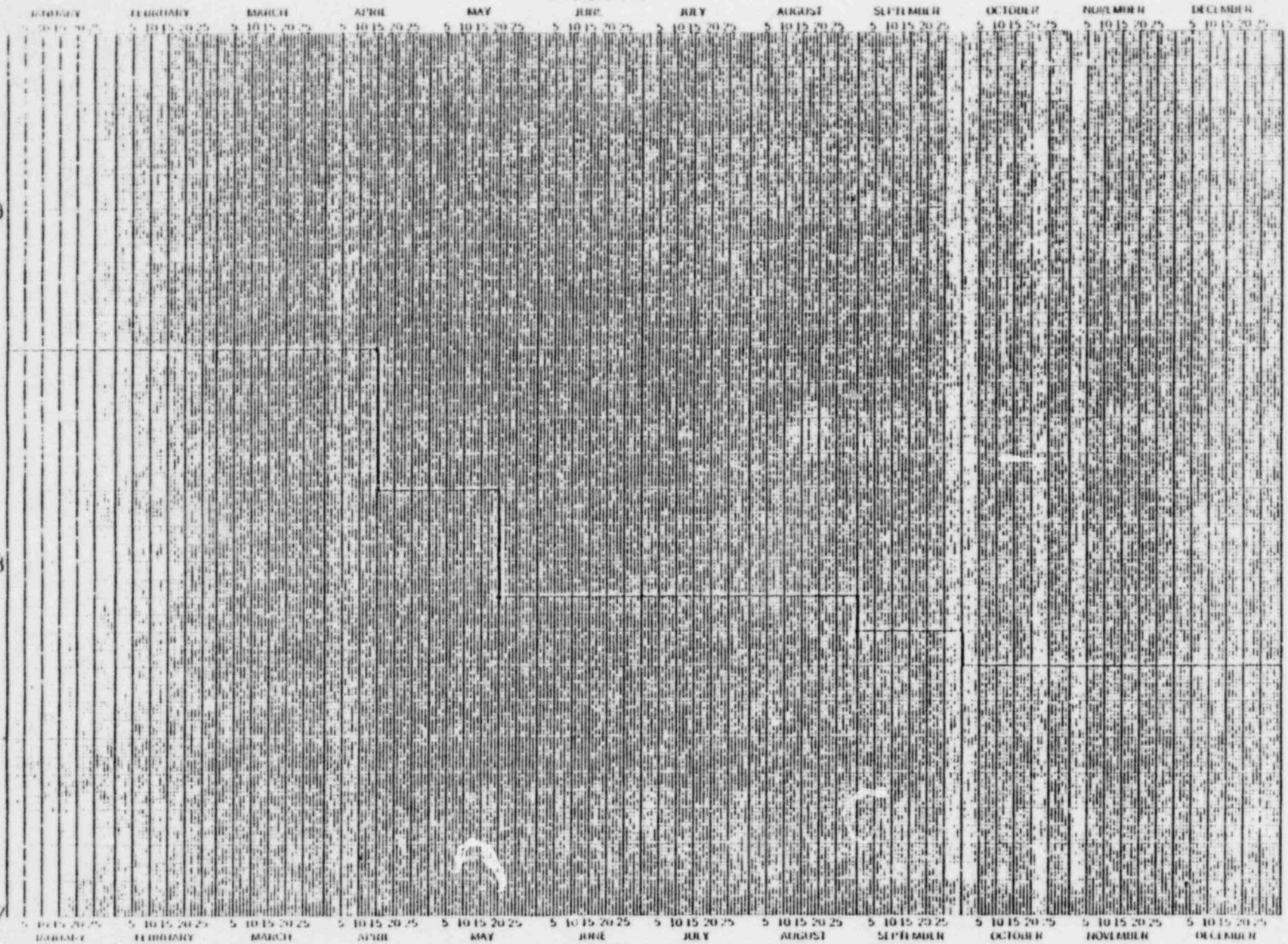
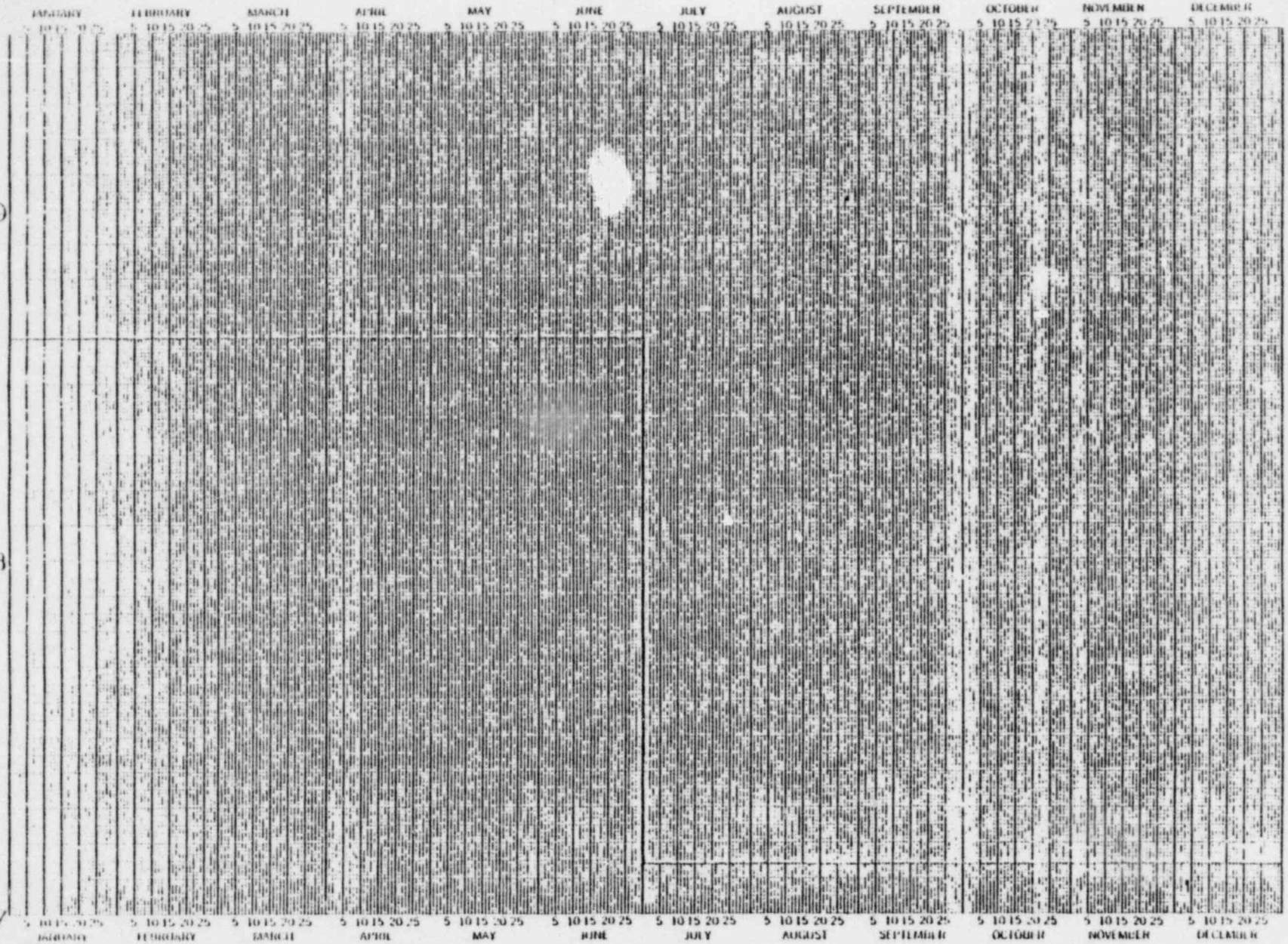


FIG. 16



1978

K-E 1 YEAR BY DAYS X 250 DIVISIONS
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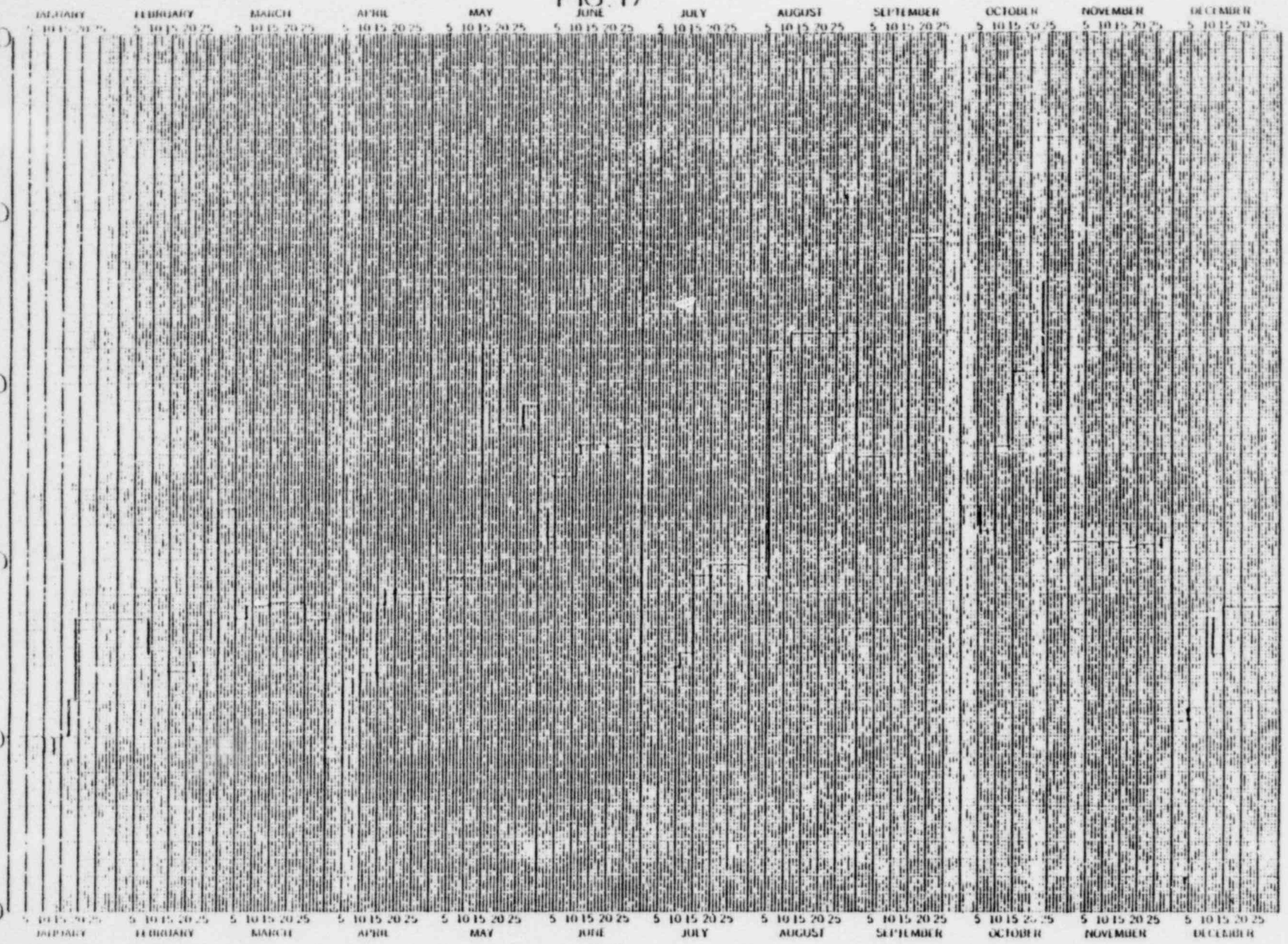
FIG. 17



1977

K-E 1 YEAR BY DAYS X 250 DIVISIONS
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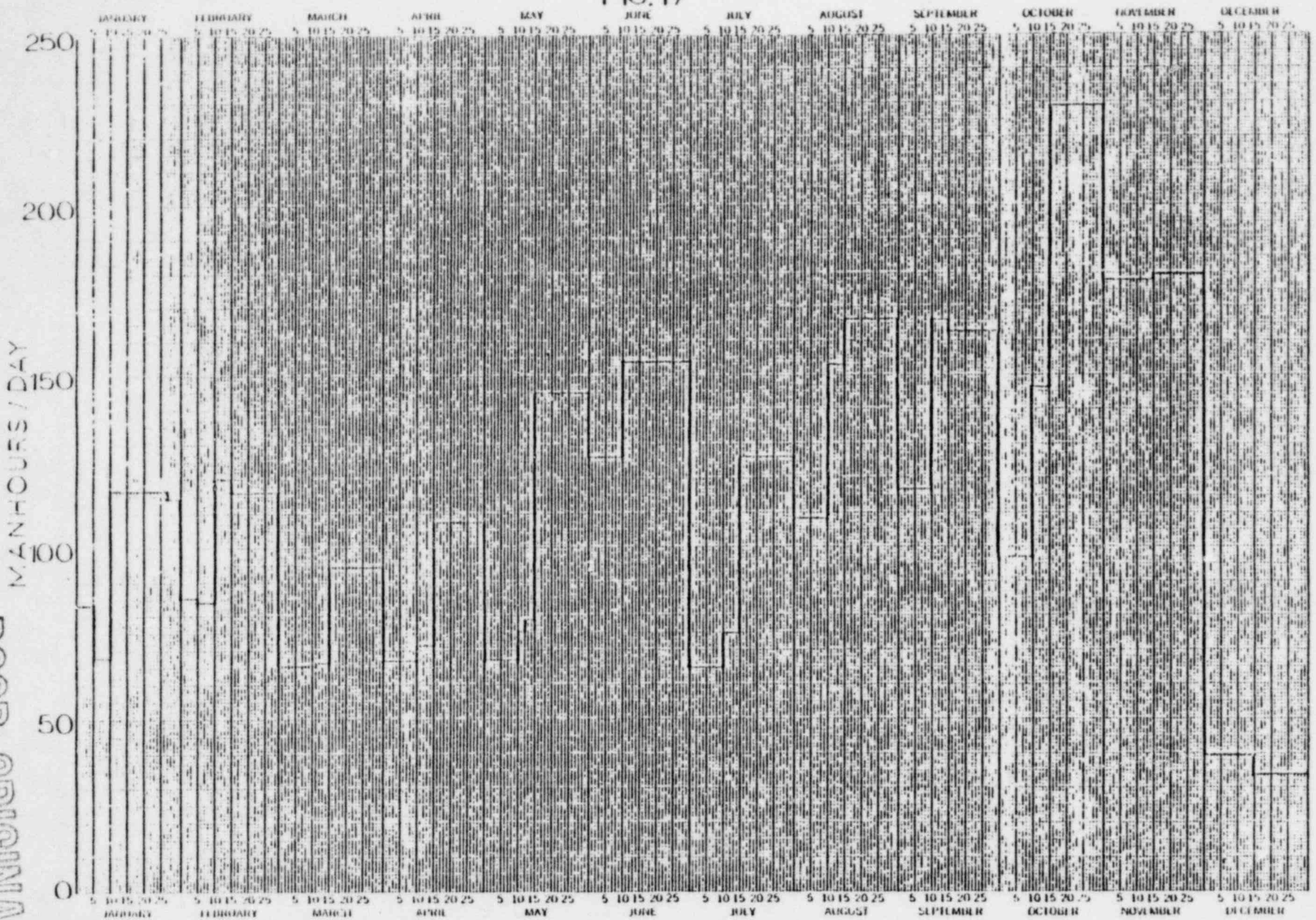
FIG. 17



Because Figures 1 thru 17 were based on a 7 day week, a four week month of 30 days, and a 365 day year, the available man hours per man per day could be calculated as 8 hours times 229 divided by 365. This is equal to 5 hours/day. Note also that no allowance for sick leave has been made. The values used in this study are 38% more than the normal work year of 1800 hours (NRC "Blue Book" assumption) and represents a conservative bias in terms of the man power requirements which result from this study because they are based on a 56 hour per man week. The bases for using this 38% underestimation are:

1. The manpower requirements which result from this study are based on projected case loads. It is assumed that it is better to have too few people to handle the actual work than it is to staff up for work which is not docketed when it is expected.
2. The calculations of the times required to conduct an operating license review (or FDA) are considerably larger than expected, based on past studies of this type. The reason is believed to be the fact that the review of previously unreviewed systems such as the Combustion Engineers CPC and the new General Electric Reactor Manual Control System were differed until a FSAR was docketed. However, the 38% factor is a tacit admission that follow-on reviews of plants similar to Arkansas and Zimmer should take less time.
3. The use of the 38% factor tends to represent a possible reduction in review effort, in the future, for the licensing of standard plants.

After determining that a conservative result would be obtained from calculating the workload on a seven day week, the available manpower was determined. The calculation is simple. There are presently 10 reviewers. The annual turnover rate for NRC employees is 6% per year. In two years we will lose at least one member from the ICSB. (12% of 10 people). Therefore, the average staff is 9.5 men per year.

On this basis, there is a theoretical manhour availability of 76 hours per day with the present staff. This manning level is plotted on Figure 17A as 80 hours in 1977 and 72 hours for 1978.

After Figure 17A was plotted, the net surplus or deficit was determined and plotted in Figure 18 as a function of the day in which it occurs. A "x" is plotted at each point on the curve where the deficit equals 229x8 (=1832) hours. This point represents a man year of 5-day weeks. (A five day week is used in this calculation to avoid using the 37% factor twice.) The resulting manning level is plotted against the total workload in Figure 17B. Figure 18A curve 1 is a plot of net surplus and deficit resulting from the Figure 17B manning levels. Each point where a fifth new employee is obtained is circled to indicate the need to hire an additional supervisor. After Figure 18A curve 1 was plotted, it was noted that advancing the schedule for hiring the first man in 1977 could possibly reduce the deficit in man hours and could eliminate the need for hiring the last of the 15 reviewers in December 1978. Therefore, Figure 17C was prepared on this assumption and Figure 18A curve 2 was prepared to test the result. (The hiring of the first man was advanced to May 20, the first zero crossing in Figure 18A.) This process was repeated several times until, the number of employees was reduced and the surplus deficit was averaged to near zero.

K·E 1 YEAR BY DAYS X 250 DIVISIONS
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FIG. 17A

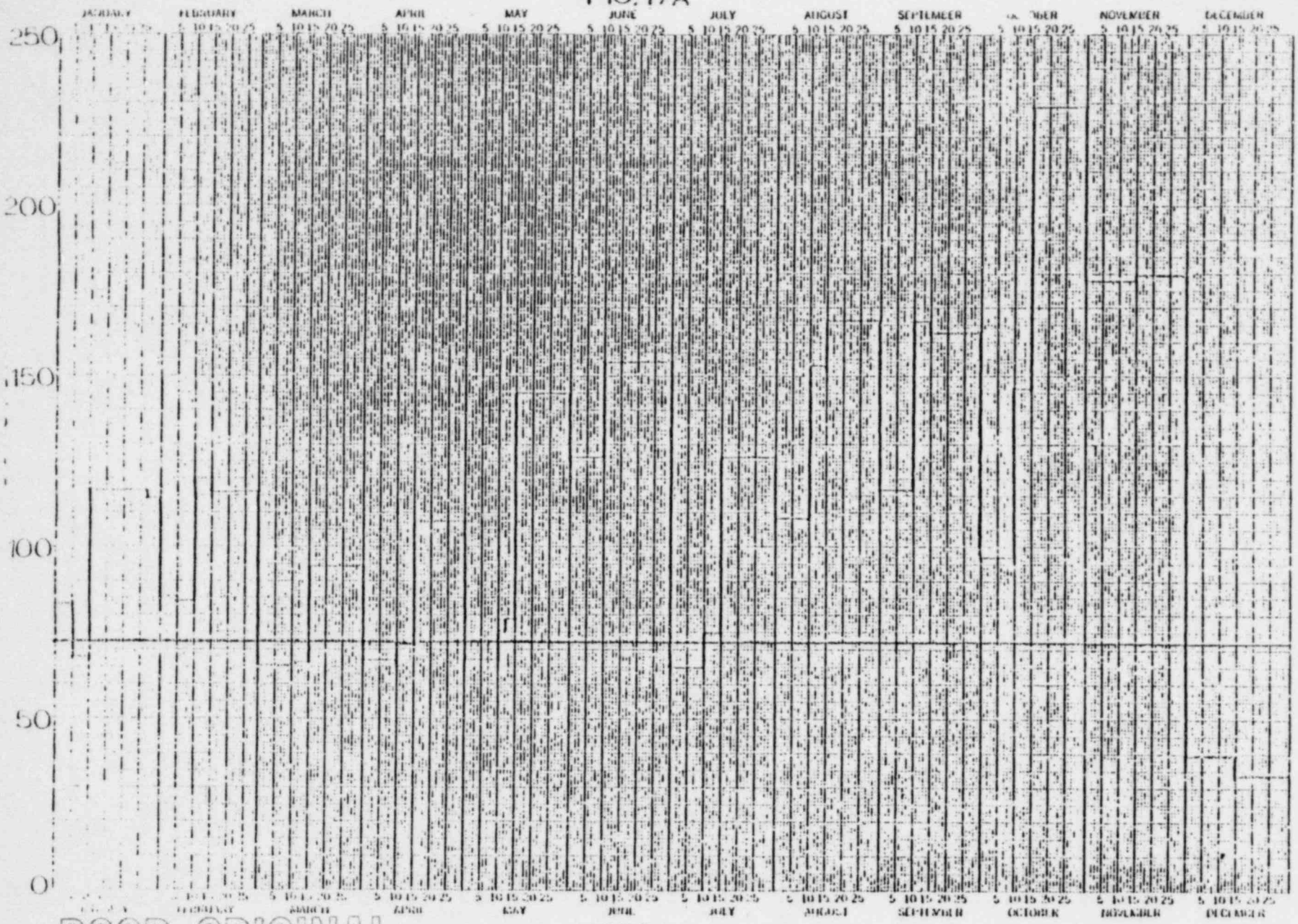


FIG. 17A

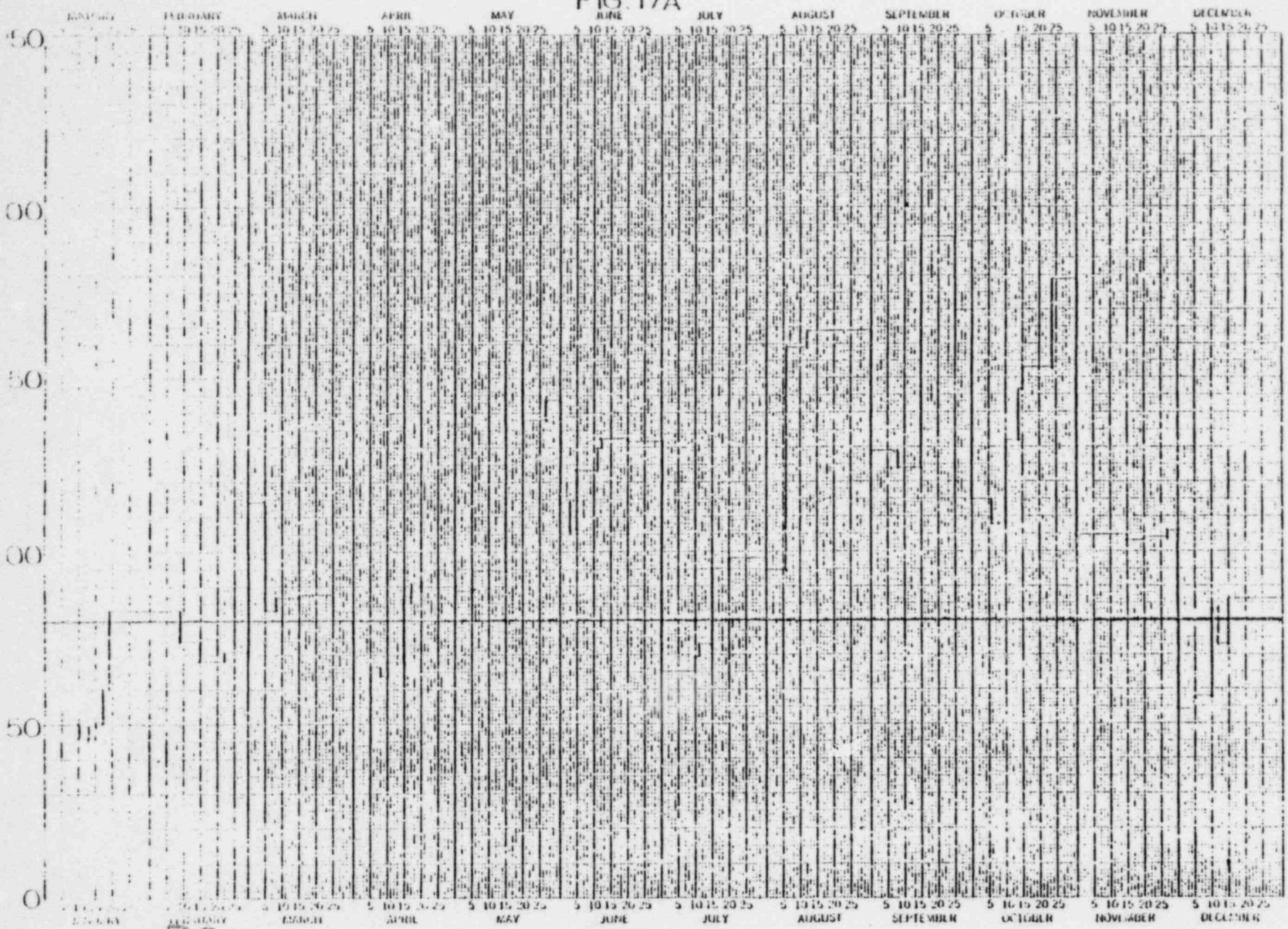
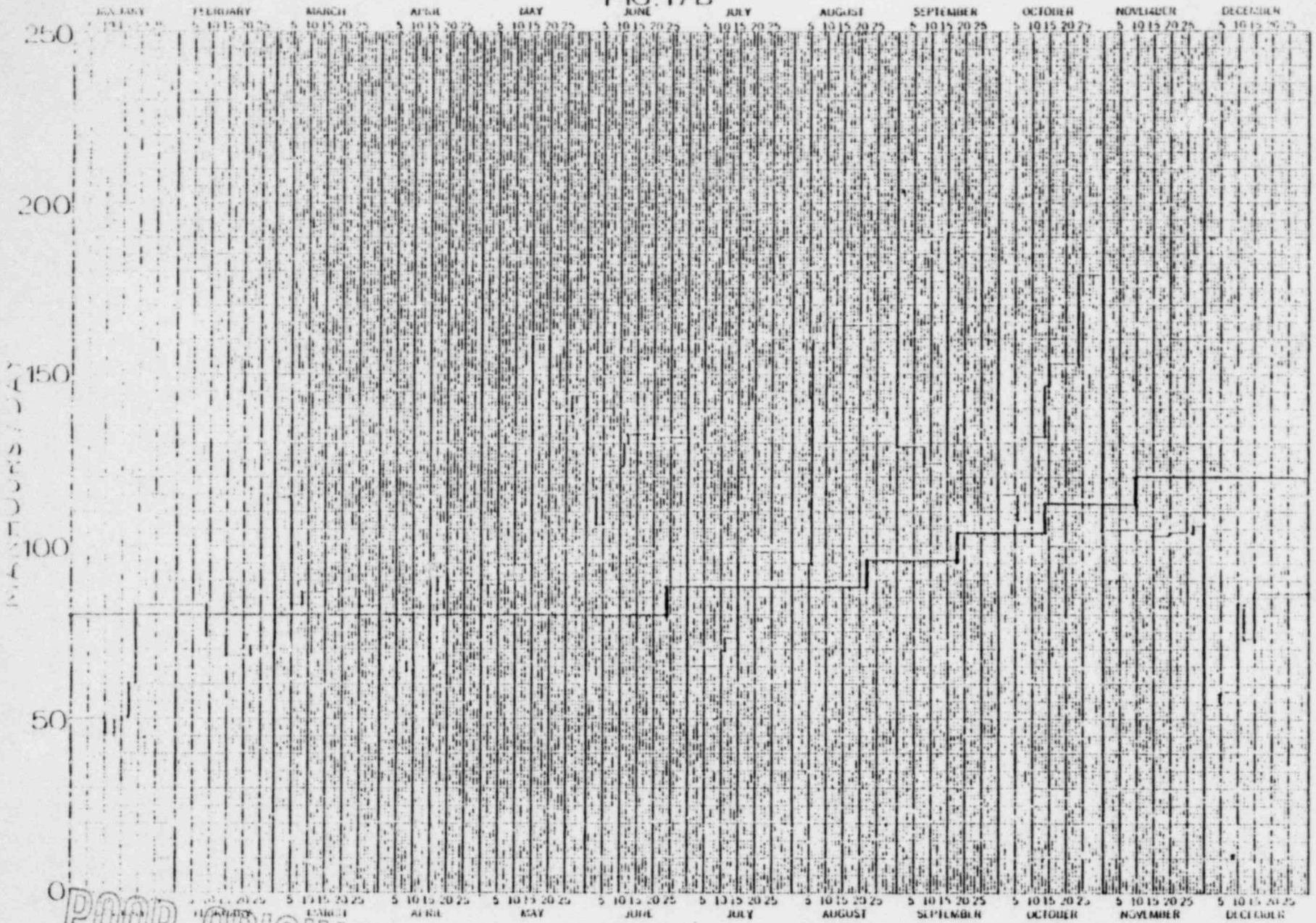


FIG. 17B



K-E 1 YEAR BY DAYS X 250 DIVISIONS
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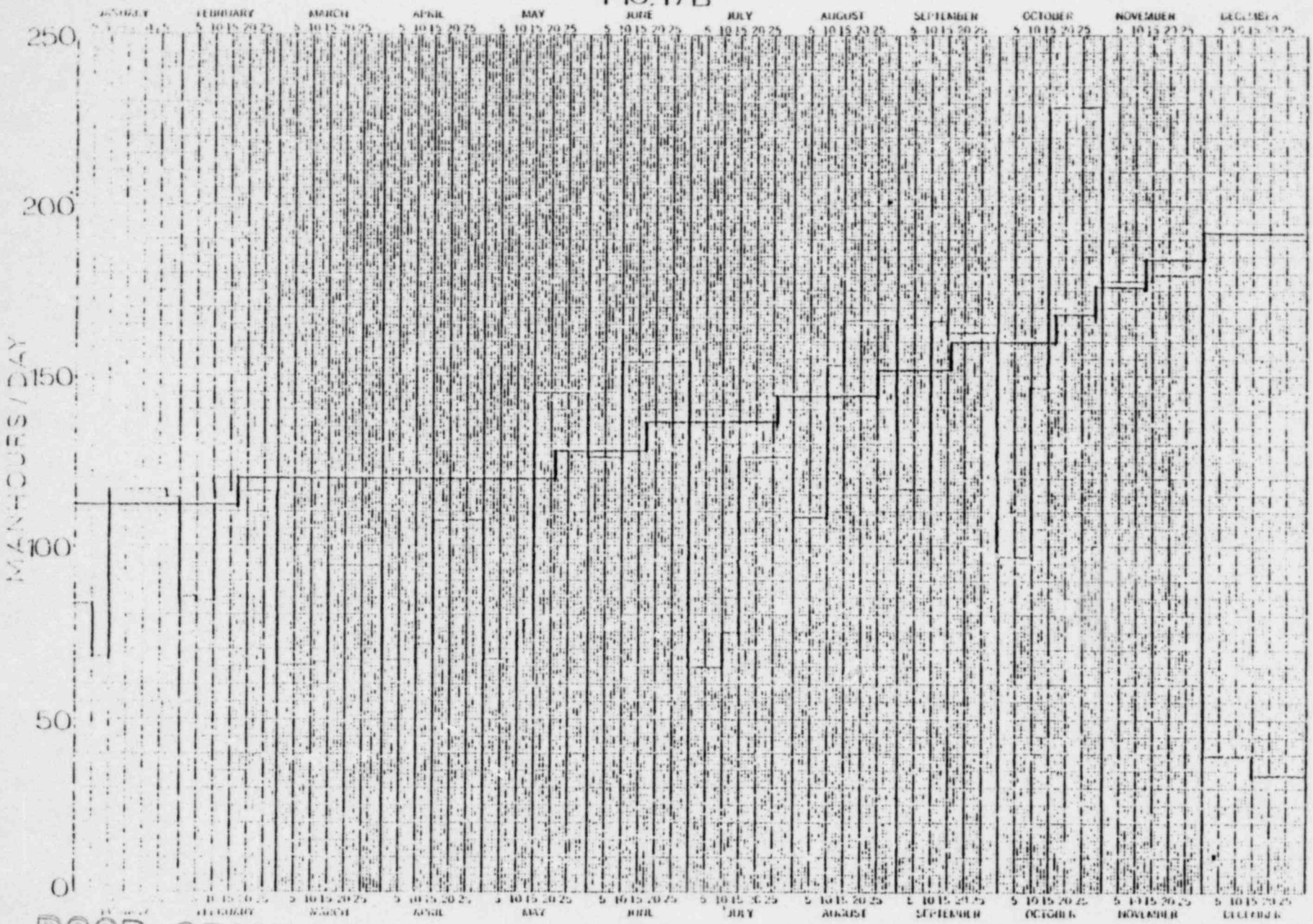
FIG. 17B



1978

FIG. 17C

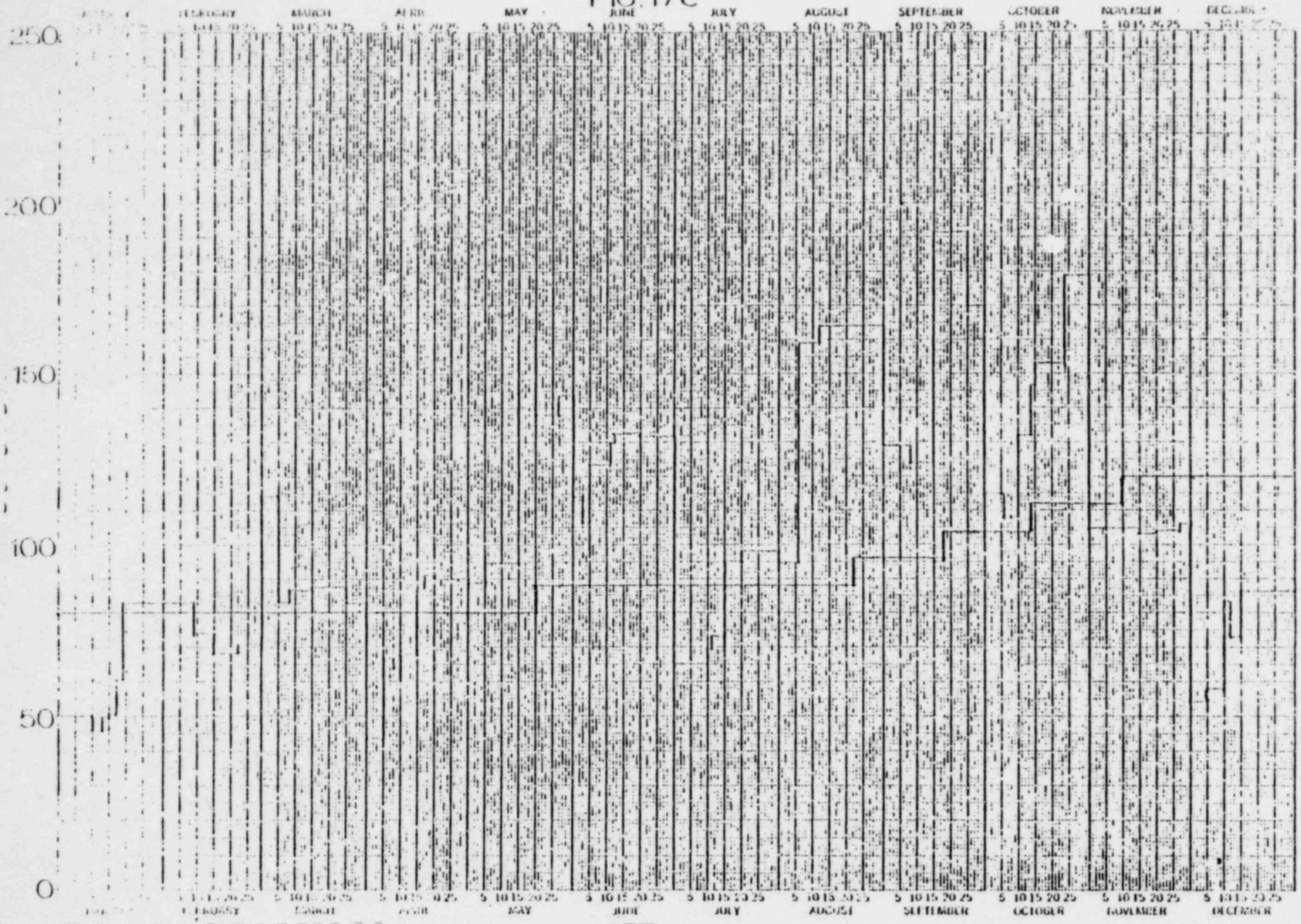
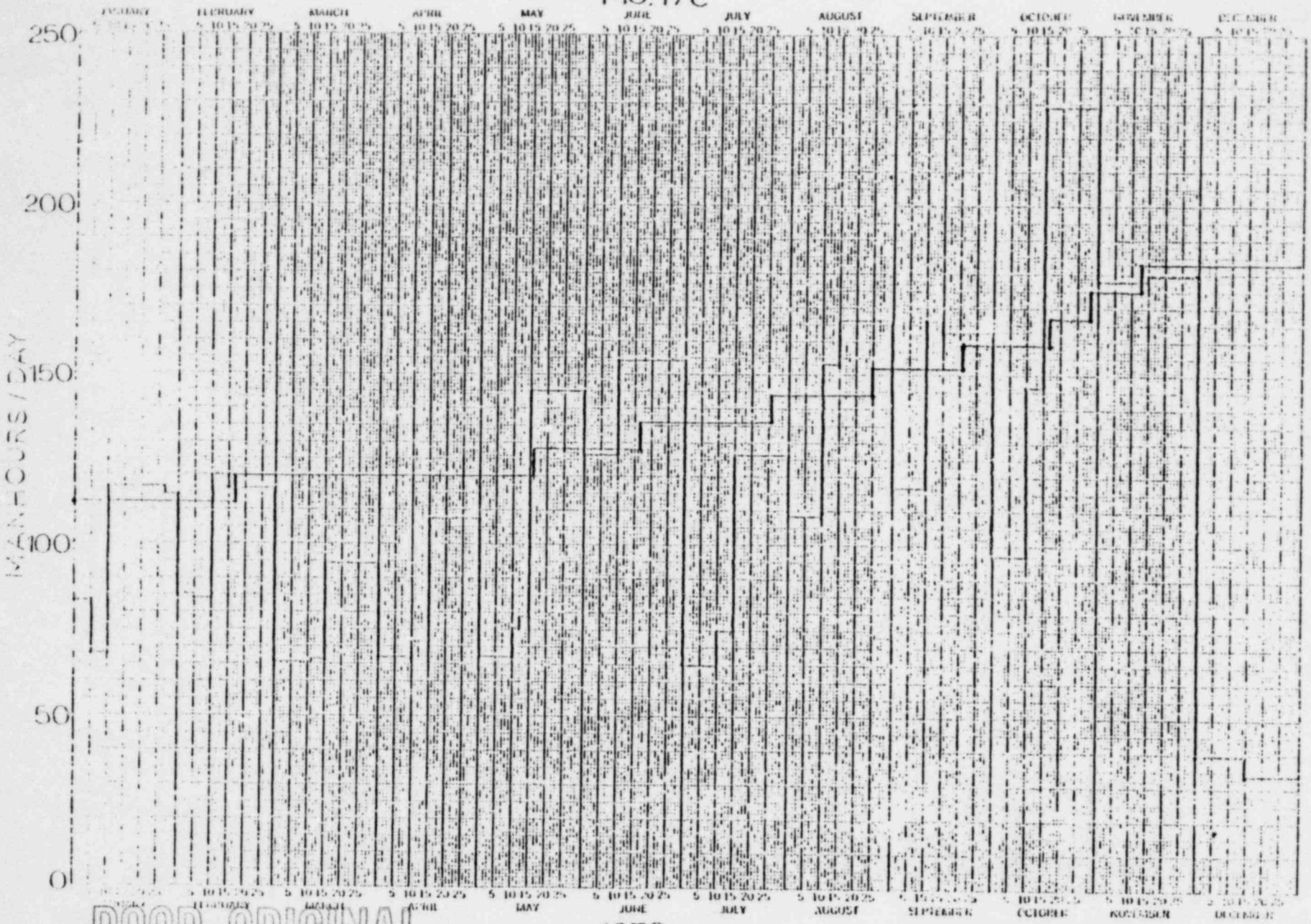


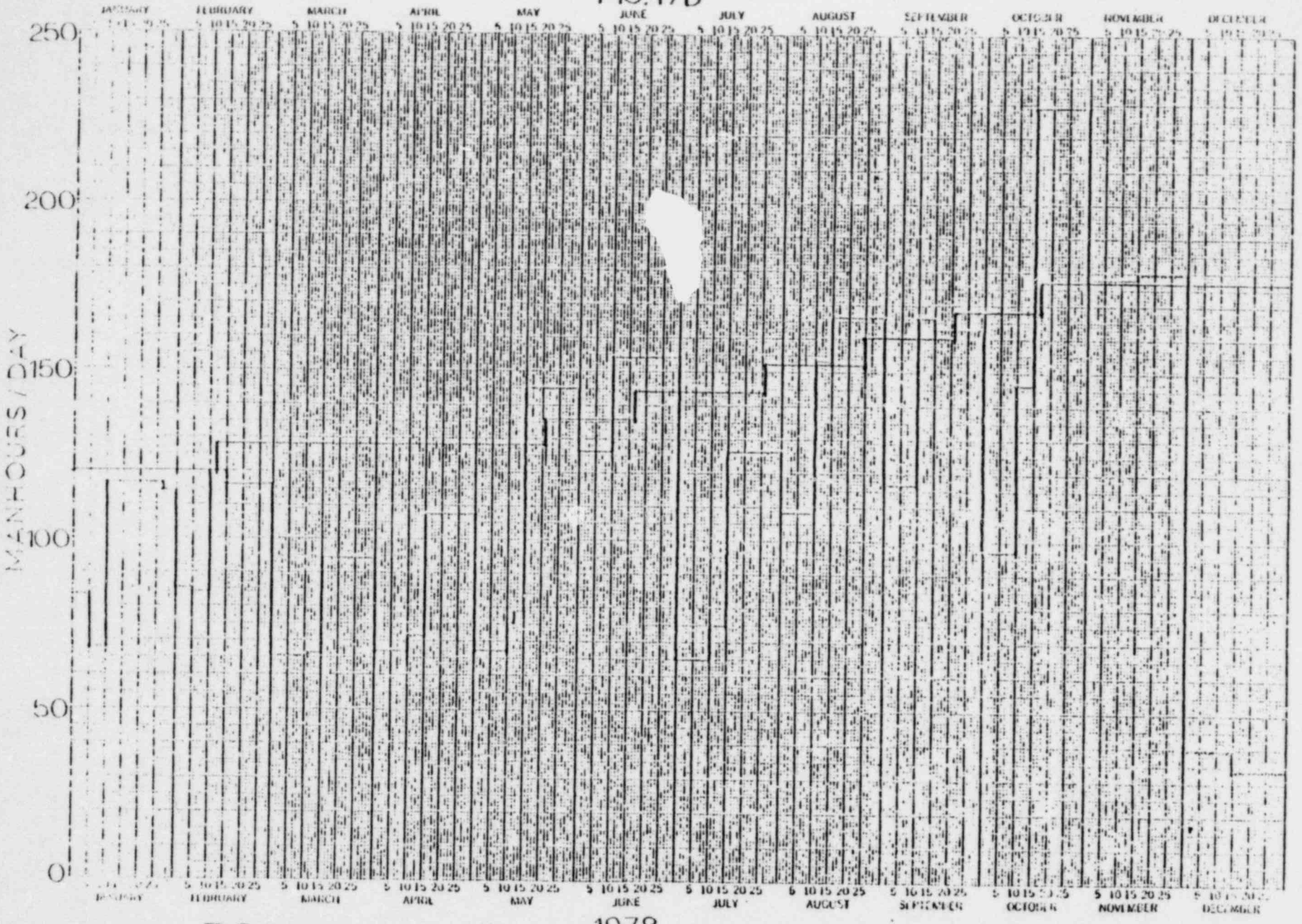
FIG. 17C



K&E 1 YEAR BY DAYS X 250 DIVISIONS
KEUFFEL & ESSER CO

46 2890

FIG. 17D



1978

FIG. 17D

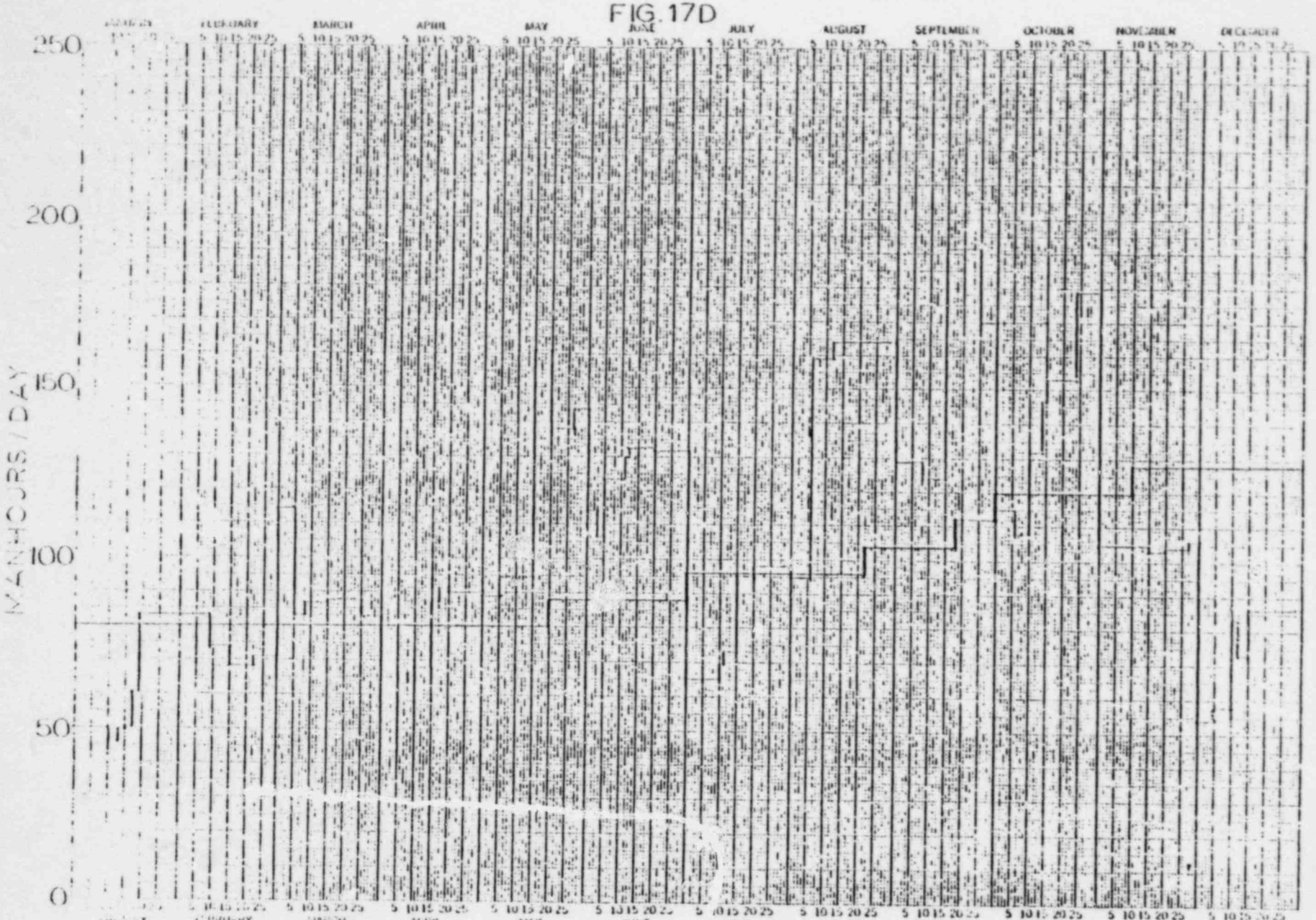


FIG. 17E

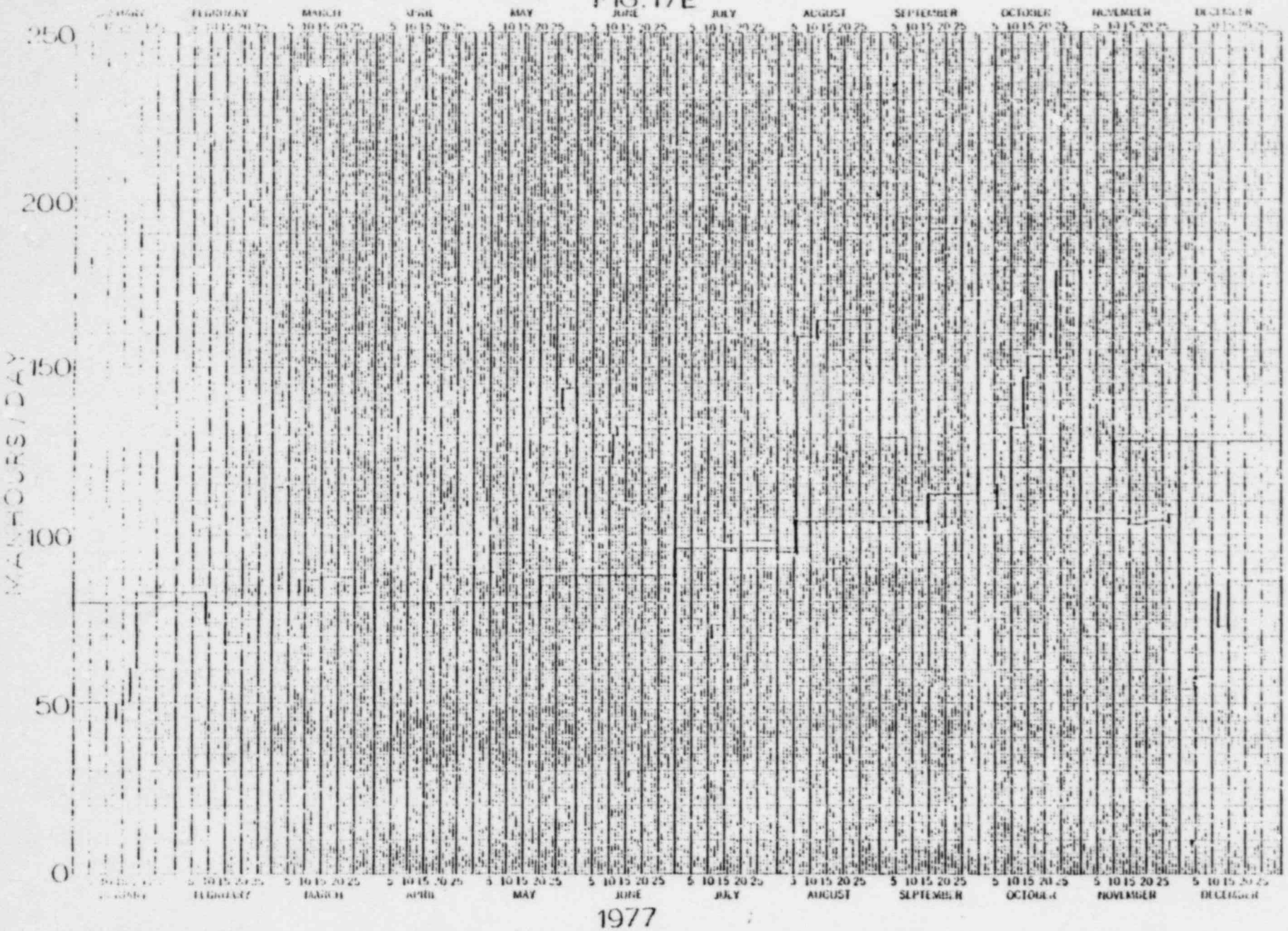
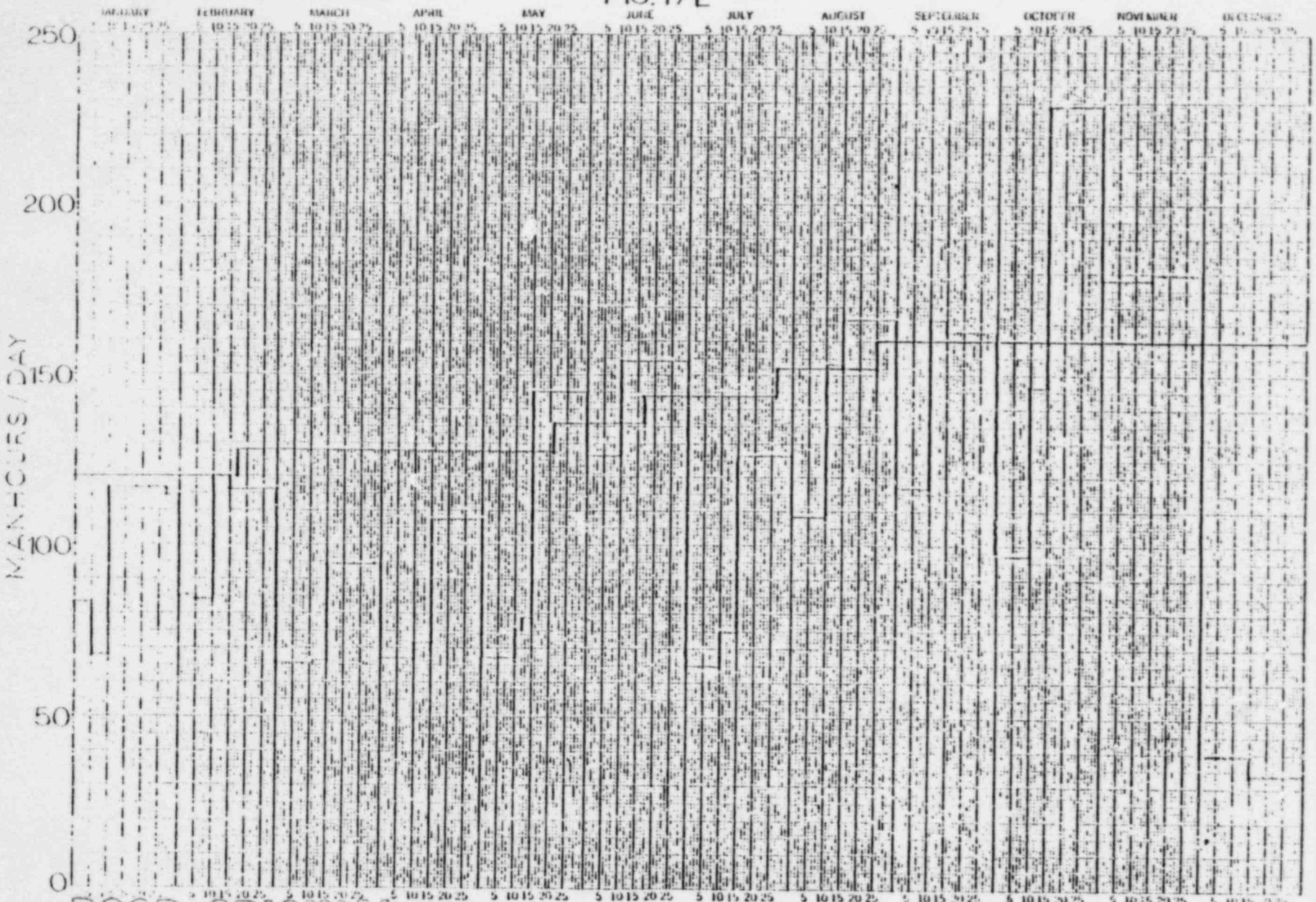


FIG. 17E



1 YEAR BY DAYS X 250 DIVISIONS

46 2890

FIG. 17F

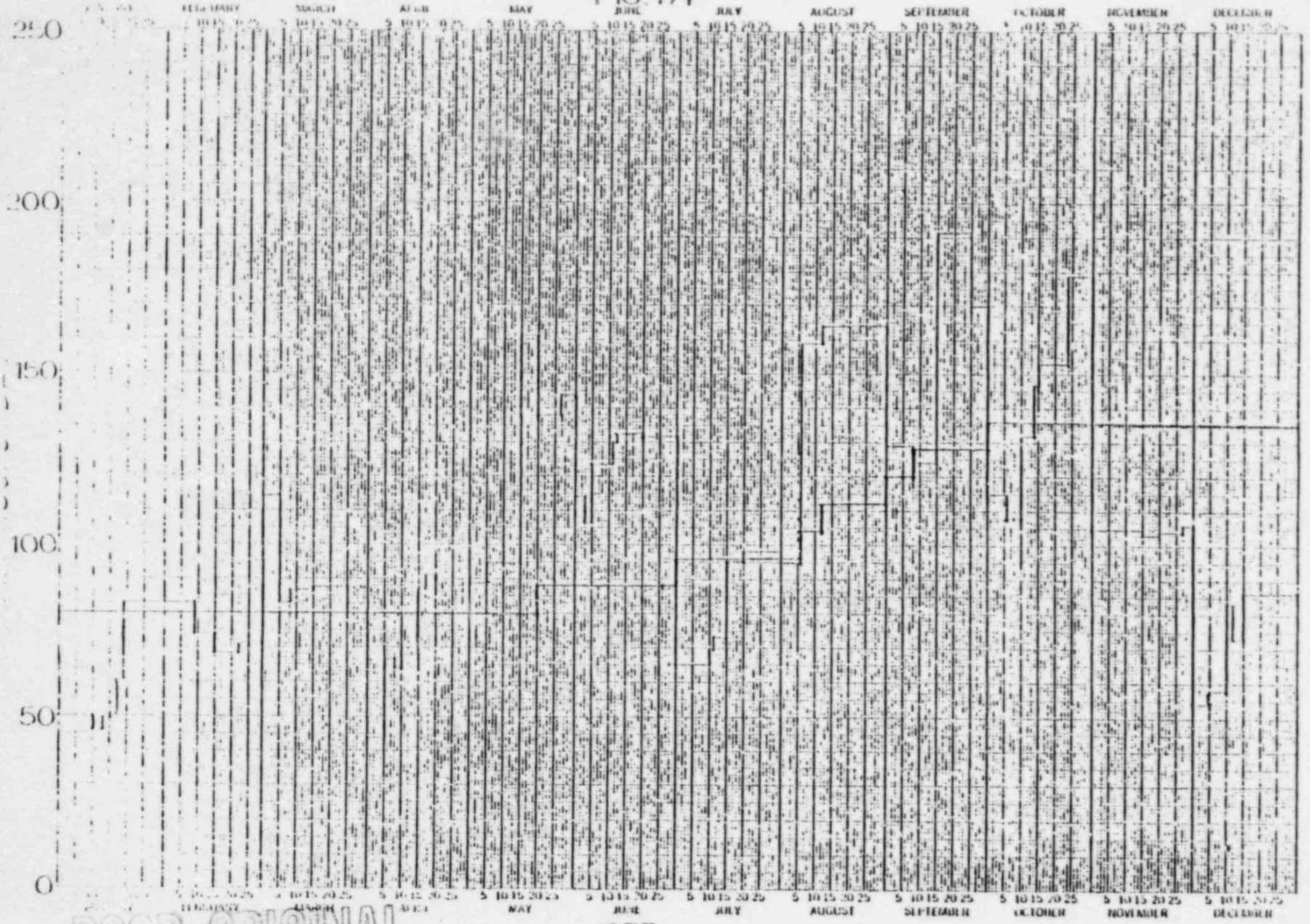
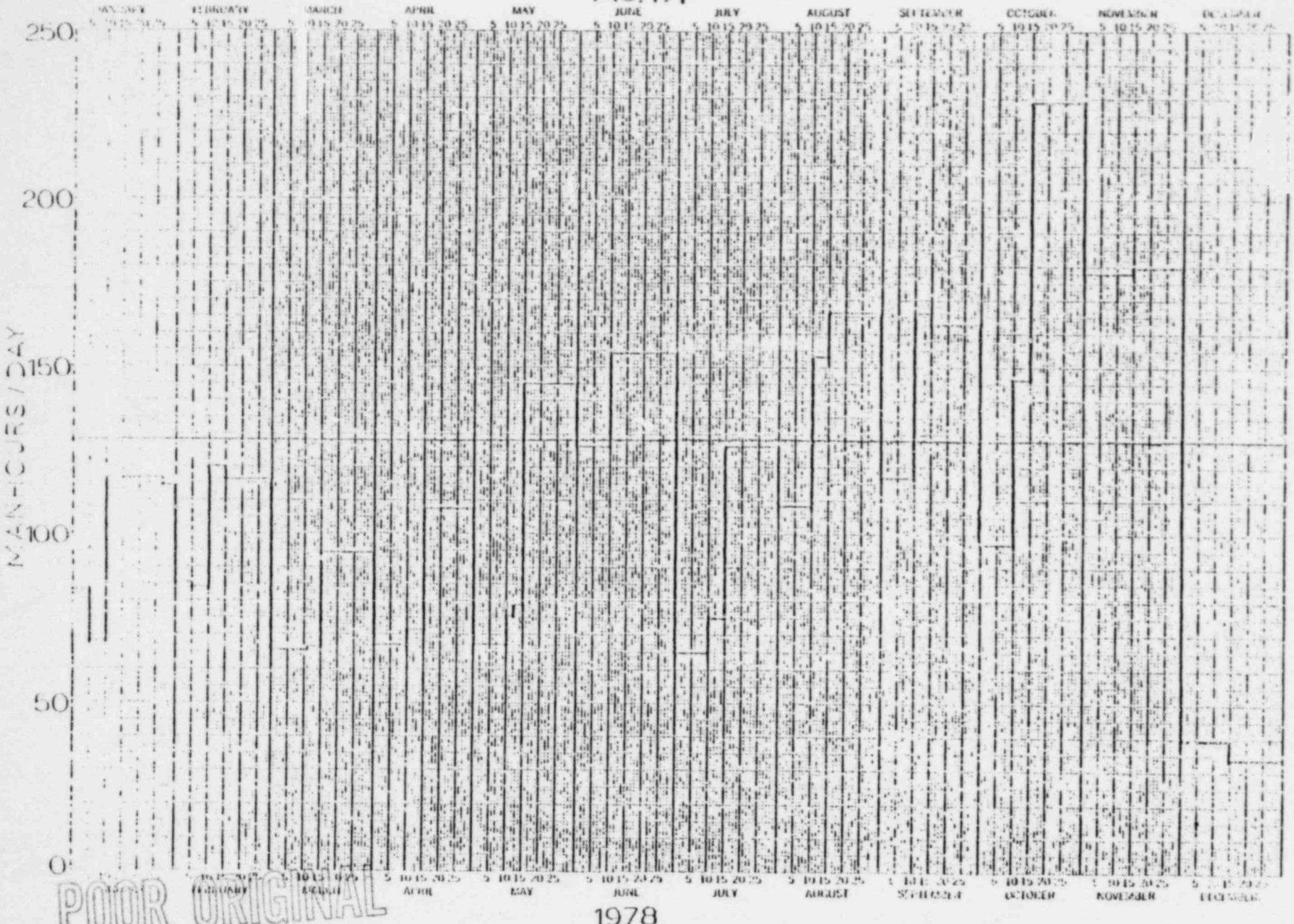


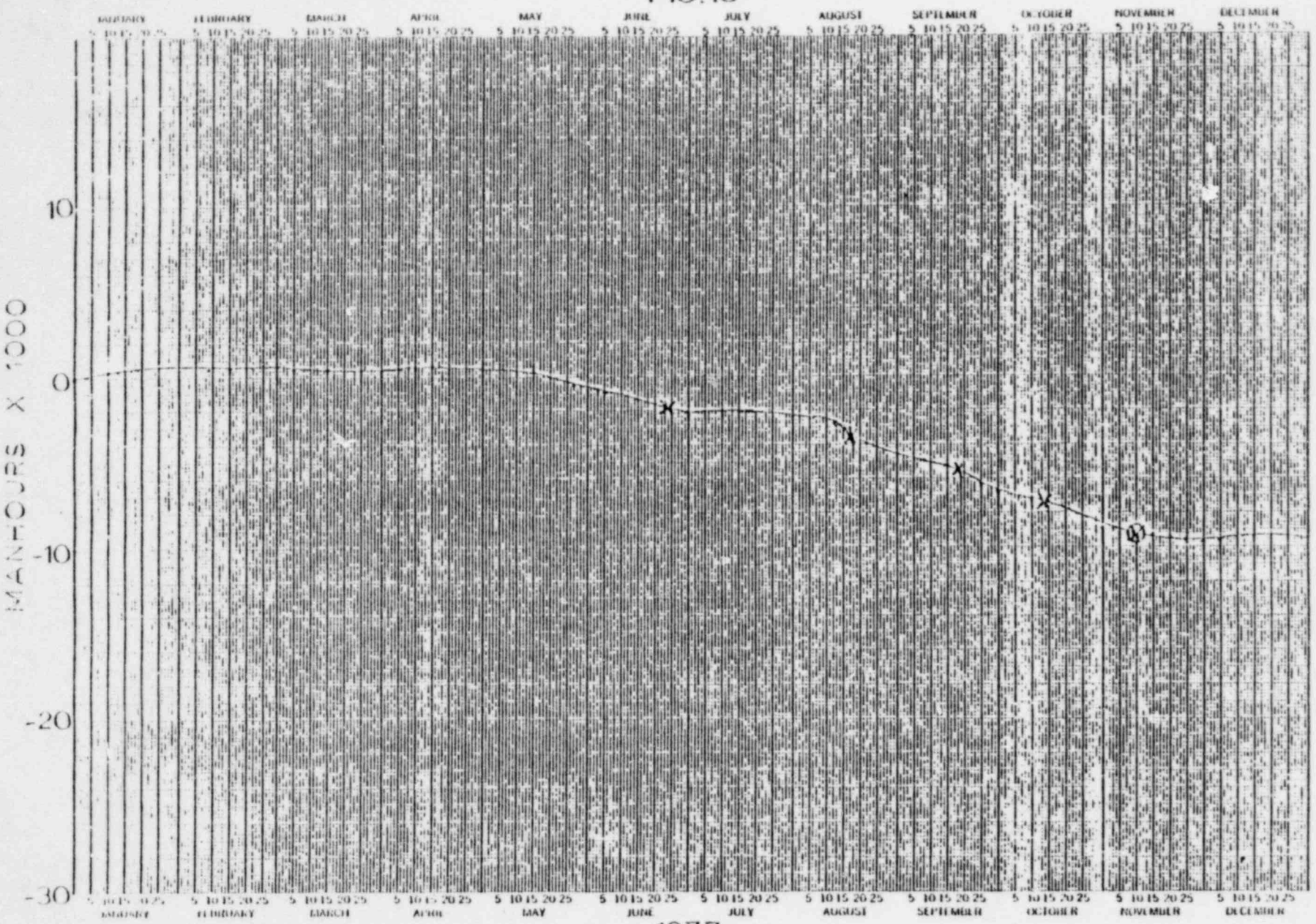
FIG. 17F



K-E 1 YEAR BY DAYS X 250 DIVISIONS
KEUFFEL & ESSER CO. MADE IN U.S.A.

46 2890

FIG. 18



K&E 1 YEAR BY DAYS X 250 DIVISIONS
KEUFFEL & ESSER CO.

46 2890

FIG. 18

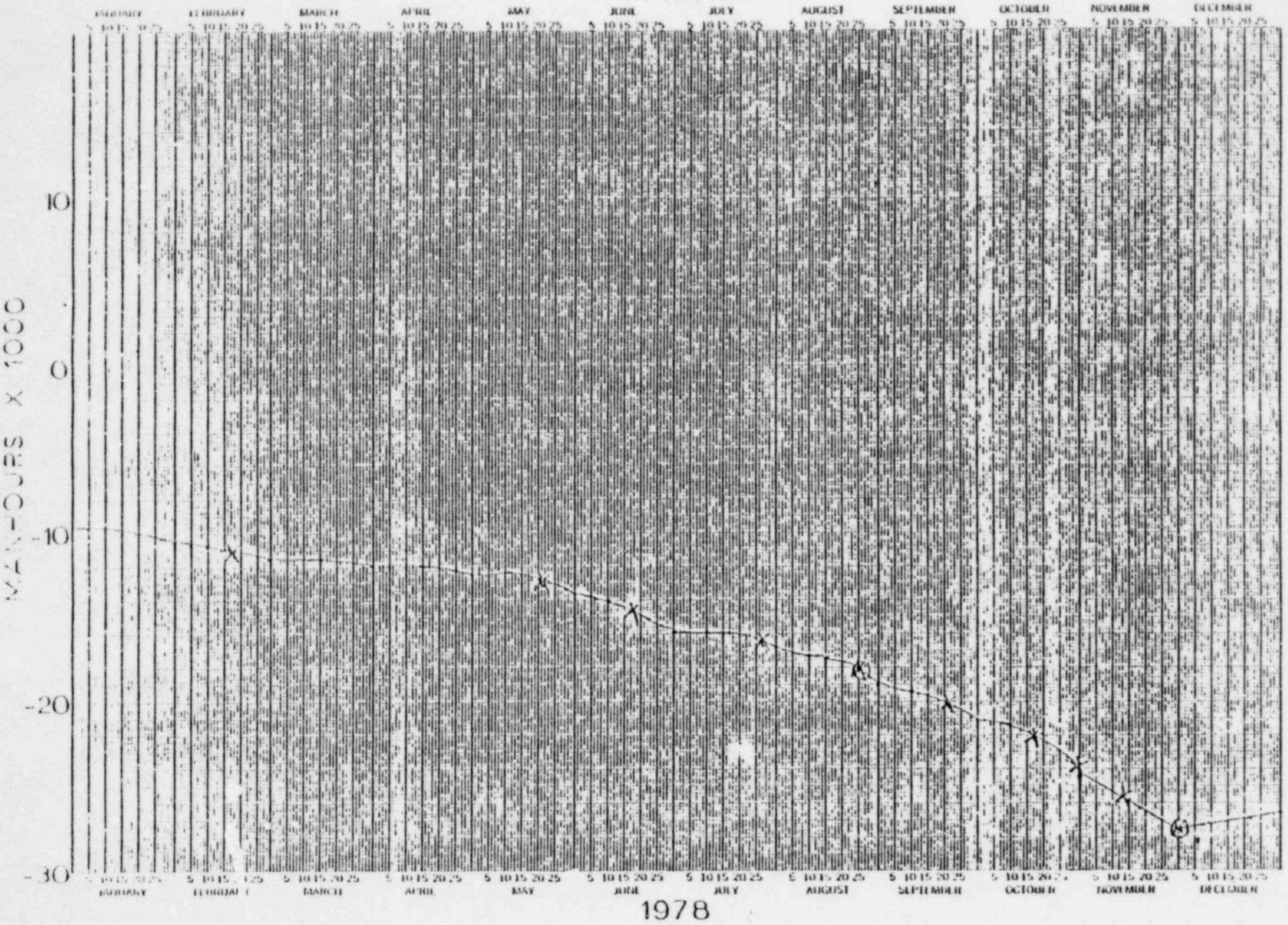


FIG. 18A

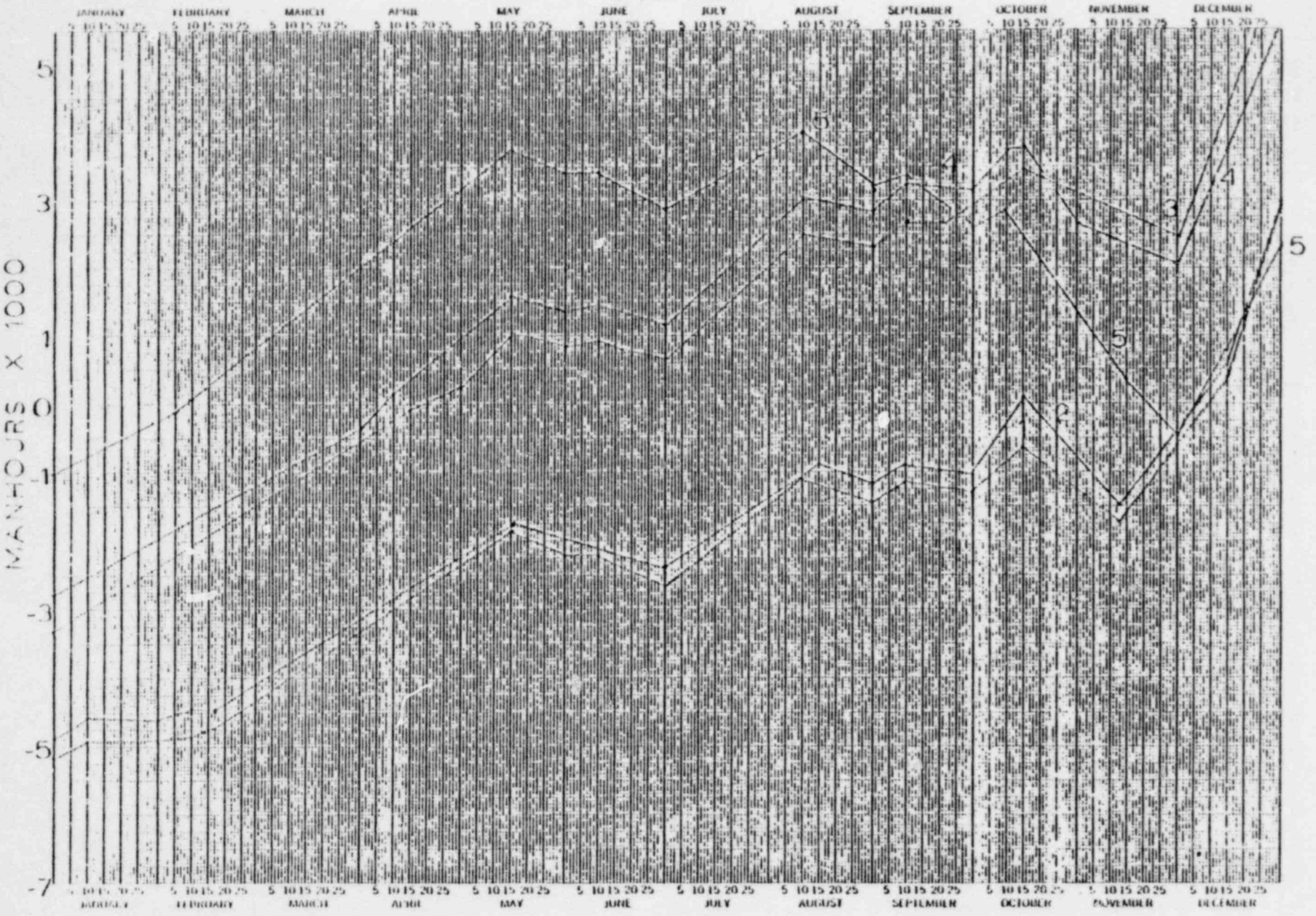
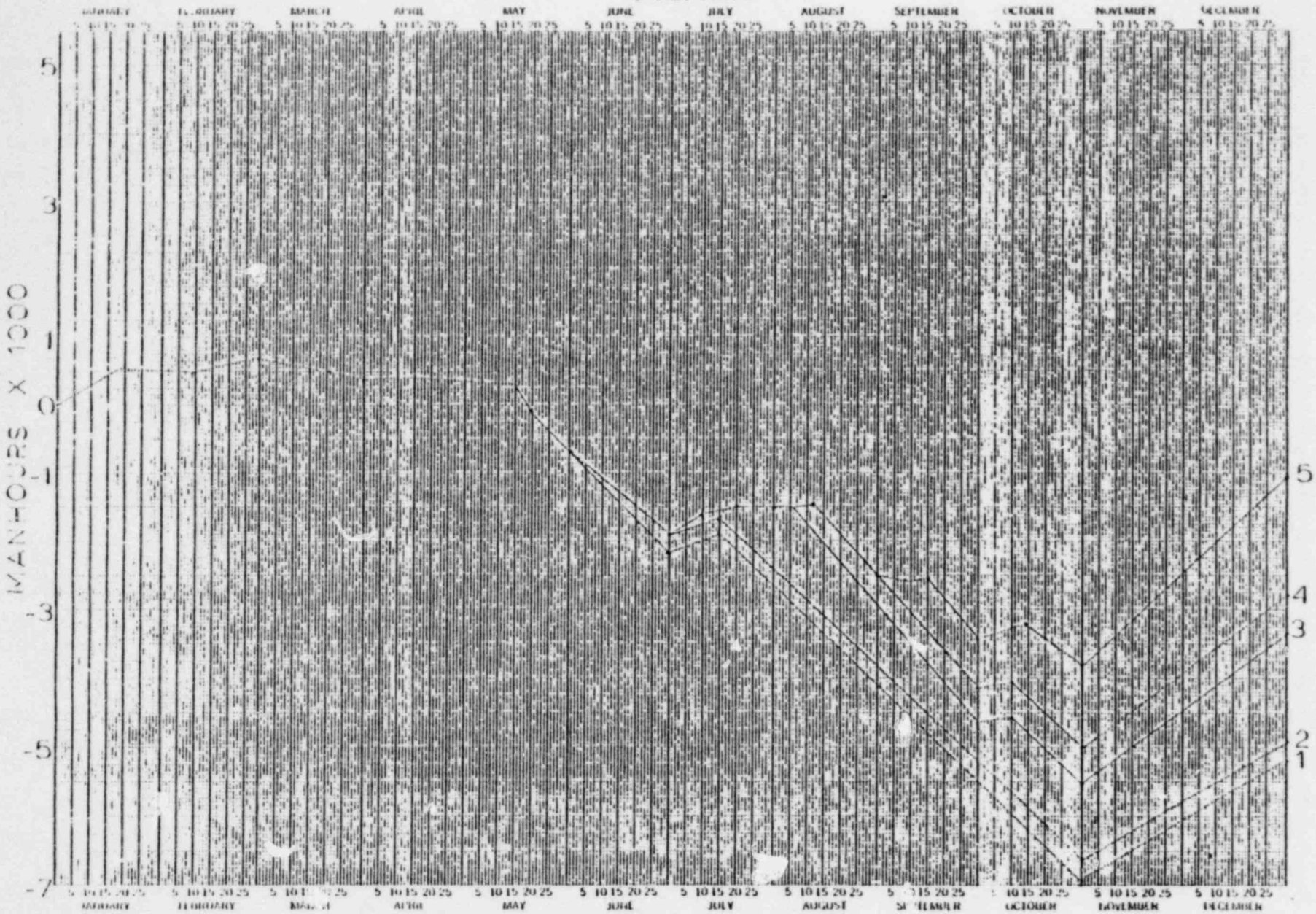


FIG. 18A



1977

Results

1. The correction factors are:

Construction Permits	.61
Construction Permit Acceptance Review	.63
Operating License	.78
Operating License Acceptance Review	.81

2. The correction factor for other ICSB tasks is: .71

3. The total effort for casework as a function of time is given in Figure 17.

4. The best distribution of manpower is presented in Figure 17F.

5. The curves of Figure 18A show the ammount of surplus/deficit in manhours which exist as a result of the various manning schemes given in Figures of 17. The following relationships are displayed:

FIGURE 17	FIGURE 18A CURVE
B	1
C	2
D	3
E	4
F	5

Conclusions

The following conclusions are based on the results of this study:

1. The effect of the reorganization of the EICSB are that the ICSB work effort will be the following percentages of the work which was done by EICSB.

   a. Construction Permits 62%
   b. Operating Licenses 80%
   c. Topical Reports and other tasks 71%

2. The following types of individuals should be hired on the dates indicated:

Engineers	Supervisor
5/20/77	9/1/77
7/1/77	
8/6/77	
8/12/77	
9/1/77	
9/8/77	
10/1/77	

3. Figure 18A curves 1 thru 4 demonstrate that a delay in hiring the recommended numbers of engineers at the dates indicated in Figure 17F will require more men at a later date and will result in greater slippage in schedules than the results given in Figure 18A curve 5.

4. The lack of complete data indicates that this study should be repeated early in 1978.

APPENDIX A

ASSUMPTIONS

1. The data in Tables I, II, and III is 100% accurate at a 100% confidence level. (Data Base)
2. The data in Tables IV, V, and VI are 99% accurate at a 95% confidence level. (Data Base)
3. The number of questions asked in the work effort cited in Table VII is a valid basis for determining the effect of the reorganization on the data of Table V. (Data Base).
4. Where data is not available, data for Construction Permit Work is applicable to Standard Plants PDA and data for operating licenses is applicable to FDA (Methods, Appendix B).
5. The correction factors to be applied to the data of Table V for PDA and FDA is 1 because little if any electrical power design is involved in these reviews. (Methods)
6. There are 7 Federal holidays, each reviewer takes an average of 15 days per year annual leave, does not take sick leave and is allowed 10 days per year for professional development (including short courses). (Methods)
7. It is preferable to underestimate the required number of professional employees than to have underutilization of the staff. (Methods)
8. The annual NRC turnover in personnel is 6%. Therefore, one person will be lost from the ICSB staff in the next two years. (Methods)

APPENDIX B

CALCULATIONS FOR TABLES V

Data Base:

The data source for this Appendix was a special report from the NRC MPS System dated 02/15/77. (Pages 6, 7, and 8 for CP and 3 and 4 for OL) which presented the work effort for 1976. A summary of this data is presented in Table B1 parts 1 thru 12.

The following corrections were made to the data for the reasons stated:

a. Docket 605 was removed from the CP listing because it was a special review.

b. Dockets 535 and 560 were deleted from OL Milestone 12 and the times listed were added to Dockets 535 and 560 CP Milestone 12 because these were CP's not OL's.

c. Docket 535 was deleted from OL Milestone 24 and added to Docket 535 CP Milestone 24 because this was a CP application.

d. Data for Docket 358 time E24 was added to R24 (Milestone 12) because no environmental work was done by EICSB.

e. Docket 219 was deleted from the OL Milestone 24 data because:

(1) This step was started in the preceding year

(2) This was not an OL Project. (It was a full term license re-review of an operating plant)

f. Docket 496 data was removed from the CP listing because it it is a standard plant (SWESSAR).

g. Dockets 518, 519, 520, and 521 were added together because this was a single application (Hartsville).

h. Docket 535 was deleted from the CP listing because it is a standard plant (GASSAR-6). The Milestone 5 data for this docket was deleted because the first round questions were started in the preceding year.

i. Dockets 560 and 561 were deleted from the CP listing because they are standard plants.

j. Dockets 448 and 521 were deleted from CP Milestone 05 because this work was started in the preceding year.

k. Dockets 400, 401, 402, and 403 were added together because this was a single application (Shearon Harris).

l. Dockets 471, 500, and 501 were deleted from CP Milestone 12 because the work was started in the preceding year.

m. The data for Dockets 546 and 547 were added because this was a single application (Marble Hills).

n. Docket 566 data was deleted from CP Milestone 12 because this was post-CP work (CP-follow up).

o. Docket 531 data for CP Milestone 12 was deleted because this work was started in the preceding year.

p. Docket 447 was deleted from CP Milestone 24 because it was a standard plant and the work was started in the preceding year.

q. Dockets 448 and 449 were added because this was one application (Douglas Point).

r. Docket 470 was deleted from CP Milestone 24 because it was a standard plant and the work was started in the preceding year.

s. Docket 471 was deleted from CP Milestone 24 because the work was not completed in the study year.

t. Docket 480 was deleted from CP Milestone 24 because it was a standard plant and the work was not completed in the study year.

u. Docket 488 was deleted from CP Milestone 24 because the work was not completed in the study year.

v. Dockets 496 and 497 were deleted from CP Milestone 24 because the work was started in the preceding year.

w. Dockets 500 and 501 were added because this was a single application (Davis Besse 2/3) Dockets 502 and 503 were added together for the same reason (Koshkonong).

Dockets 508 and 509 were added together for the same reason (Washington Nuclear 3/5).

x. Docket 516 was deleted from CP Milestone 24 because the work was not completed in the study year.

Dockets 522, 523, 528, 529, and 530 were deleted for the same reason.

y. Docket 532 was deleted from CP Milestone 24 because it was a standard plant and the work was started in the preceding year.

z. Docket 593 was deleted from CP Milestone 24 because it was a standard plant and the work was not completed in the study year.

aa. Docket 368 data for Milestone 12 was deleted because it is incorrect (Q1 issued 12/11/75 incomplete)

bb. Dockets 390 and 391 were deleted because this work was not completed in the study year.

cc. Dockets 275, 320,and 323 were deleted because this work started in the preceding year.

dd. Dockets 267 and 296 data were deleted from Milestone 24 because this was OL follow-up.

ee. Docket 366 was deleted from OL Milestones 12 and 24 because some of this work was done in the following year.

ff. Docket 348 and 364 were deleted because this work was not completed in the study year.

gg. Dockets 369 and 370 were added together because this is a single application (McGuire).

Following the listing of docket numbers and times the data was tested for validity. The check was accomplished by determining the mean and standard deviation for each part and assuring that all data was within the range of $(\bar{x} - 3\sigma) \leq x_i \leq (\bar{x} + 3\sigma)$. There were no outliers and, therefore, it was concluded that the mean has an accuracy of better Than 98.7%.(The area under the normal curve from -3σ to $+3\sigma$ is 99.73%.)

Because the MPS system did not provide the time data on acceptance reviews for this year the data for CP acceptance reviews stated in the September 11, 1973, memorandum from R. Ippolito to V. Stello was assumed to be a valid estimate (24 manhours)

TABLE B-1

PART 1 CP MILESTONE 05

DOCKET	TOTAL TIME (MANHOURS)
443	34.5
526	127.0
537	34.0
549	422.0
556	363.0
	$\bar{x}$ = 196.1
	$\bar{x} + \sigma$ = 361.1
	$\bar{x} - \sigma$ = 31.1

PART 2 SP MILESTONE 05

DOCKET	TOTAL TIME
495	61.0
560	202.0
561	208.0
	$\bar{x}$ = 157.0
	$\bar{x} + \sigma$ = 224.9
	$\bar{x} - \sigma$ = 89.1

PART 3 CP MILESTONE 12

DOCKET	TOTAL TIME
400	24.0
477	32.0
518	87.0
537	1169.0
546	691.0
548	163.5
549	2.0

TABLE B-1 (Continued)

PART 3 CP MILESTONE 12 (Continued)

DOCKET	TOTAL TIME
553	484.0
556	161.2
$\bar{x}$ =	300.4
$\bar{x} + \sigma$ =	665.0
$\bar{x} - \sigma$ =	0

PART 4 SP MILESTONE 12

DOCKET	TOTAL TIME
495	83.0
535	227.0
560	232.3
561	155.5
$\bar{x}$ =	174.5
$\bar{x} + \sigma$ =	235.4
$\bar{x} + \sigma$ =	113.6

PART 5 CP MILESTONE 24

DOCKET	TOTAL TIME
376	40.0
448	50.0
477	72.0
500	126.0
502	52.0
508	112.0
514	196.0
518	513.5
546	157.0
548	316.0
549	39.0
553	216.0
556	381.0
$\bar{x}$ =	174.7
$\bar{x} + \sigma$ =	317.3
$\bar{x} - \sigma$ =	32.1

TABLE B-1 (Continued)

PART 6 SP MILESTONE 24

DOCKET	TOTAL TIME
495	467.0
535	132.0
550	109.5
560	227.5
561	274.0
$\bar{x}$ =	242.0
$\bar{x} + \sigma$ =	369.7
$\bar{x} - \sigma$ =	114.3

PART 7 OL MILESTONE 05

DOCKET	TOTAL TIME
322	269.0
358	232.5
366	408.0
$\bar{x}$ =	303.2
$\bar{x} + \sigma$ =	378.8
$\bar{x} - \sigma$ =	227.6

PART 8 OL MILESTONE 12

DOCKET	TOTAL TIME
322	321.0
341	476.5
368	2997.5
$\bar{x}$ =	1265.0
$\bar{x} + \sigma$ =	2491.7
$\bar{x} - \sigma$ =	38.3

TABLE B-1 (Continued)

PART 9 OL MILESTONE 24

DOCKET	TOTAL TIME
320	308.4
368	1889.0
369	1643.0
$\bar{x}$	1281.0
$\bar{x} + \sigma$	1975.1
$\bar{x} - \sigma$	586.9

PART 10 OL Follow up

DOCKET	TOTAL TIME
267	8
296	6
$\bar{x}$	7
$\bar{x} + \sigma$	8
$\bar{x} - \sigma$	6

PART 11 CP Follow up

DOCKET	TOTAL TIME
566	68

PART 12 Special Application

DOCKET	TOTAL TIME
605	131.2

Because complete data was available from the reviewers notebooks (which were started on 9/3/74), 1976 data was used to determine the time requirements for OL acceptance reviews. This data is presented in Table B-2.

Because the MPS report did not provide time data for drawing reviews and site visits, technical specifications review, Supplemental SER, or attendance at hearings, the data presented in Table B-3 was extracted from Mr. Ippolito's memorandum.

This data was then transcribed to Table V. Assuming that Standard plants would be treated like a CP or OL where other data was not available.

TABLE B-2 OL ACCEPTANCE REVIEWS

DOCKETS	TOTAL TIME
373	43.9
390	27.0

$\bar{x} = 35.5$

$\bar{x} + \sigma = 43.9$

$\bar{x} - \sigma = 27.1$

TABLE B-3 (in manhours)

	CP		OL	
	Normal	Minimum	Normal	Minimum
Drawing Review/ Site Visit	NA	NA	160	160
Technical Specifications	NA	NA	160	80
Supplemental SER (Milestone 27)	40	40	80	80
Hearings	0	0	16	16

APPENDIX C

CALCULATION OF TACS DATA

Data Base

The data for current and completed TACS was drawn from the MPS System Table C1 presents a listing of current TACS, by TACS number and code and the time for each number. Table C2 presents similar data for completed TACS.

Methods

The combined statistics for Tables C-1 and C-2 were calculated. The results of these calculations are presented in Table C-3.

The projected TACS work load was estimated by the following procedure:

1. For each applicable TACS number, the active TACS Were listed (TABLE C-1) for 1977.
2. The count (number) of these TACS was subtracted from the N value given in Table C-3. (N is number of this kind of TAC in 1976) Negative values were taken as 0 (null).
3. This difference (Δ) was multiplied by result 2 (0.71) and this product multiplied by $\bar{x}$ (average time) for 1977 and added to the current work load. For 1978 the calculation was 0.71 $\bar{x}$ N.
4. Table C-5 was prepared from the results using the criteria used to estimate the time to complete the active TACS is presented in Table C-4.

TABLE C-1

CURRENT TACS

TAC	TOTAL TIME (MANHOURS)
PART A - R18	
3329	460.5
3916	8.0
3918	36.1
4370	5.0
PART B - R37	
3905	10.7
PART C - R38	
3919	49.1
6372	8.0
PART D - R51	
1075	29.0
1408	2.0
3244	169.0
3297	16.0
PART E - R52	
2370	47.0
PART F - R53	
3256	65.0
PART G - R54	
3436	162.1

TABLE C-1 (Continued)

CURRENT TACS

PART H - R55	
2129	35.9
2140	56.3
2142	15.5
PART I - R56	
3681	8.0
3721	0.6
8300	284.3
8365	36.9
PART J - R71	
1623	23.0
3378	5.0
3971	1.0
4062	14.0
PART K - R73	
3258	142.0
3369	50.7
4209	32.0
4223	2.0
4343	539.6

TABLE C-2

COMPLETED TACS

PART A - R18

TAC	TOTAL TIME (MANHOURS)
1726	70.0
1761	22.0
3239	106.0
3701	8.0
3972	92.0
4248	24.0
PART B - R35	
3658	32.8
PART C - R37	
1079	4.0
1336	26.0
1711	1.1
1736	16.0
1863	3.0
1893	21.0
1914	96.0
1974	8.0
3066	64.5
3073	2.0
3085	237.0

TABLE C-2 (Continued)

COMPLETED TACS

TAC	TOTAL TIME (MANHOURS)
PART D - R38	
1470	70.0
1847	16.0
2137	227.1
2170	29.0
3007	2.0
3070	5.1
3271	51.0
4181	82.0
PART E - R40	
3777	55.4
PART F - R51	
1044	1.7
1146	15.0
1426	36.0
1524	64.0
1717	44.0
1946	4.0
3000	165.2
3344	25.0
3348	92.5
3351	303.0
3358	48.0

TABLE C-2 (Continued)

COMPLETED TACS

PART F - R51 (Continued)

TAC	TOTAL TIME (MANHOURS)
3615	30.0
3642	16.0
3643	6.0
3678	56.0
3695	16.0
3848	16.0
4261	64.0
PART G - R53	
4256	8.0
4293	15.0
PART H - R54	
3019	5.0
PART I - R55	
2130	4.0
2132	6.0
2373	9.0
3298	10.0
3314	24.0
3926	178.4

TABLE C-2 (Continued)

COMPLETED TACS

PART J - R56

TAC	TOTAL TIME (MANHOURS)
1795	5.4
2 33	124.5
2143	18.0
2422	424.5
3001	5.0
3215	26.0
3228	39.0
3295	16.0
3297	8.0
3762	5.0
3926	102.0
6341	1959.6

PART K - R71

1103	5.0
1622	4.0
1942	36.0
2296	52.0
3260	12.0
3312	28.0
3340	22.0
3641	23.0
3763	12.0

TABLE C-2 (Continued)

COMPLETED TACS

PART K - R71 (Continued)

TACS	TOTAL TIME (MANHOURS)
3927	4.0
3928	6.0
3937	64.0
4202	3.0
4268	16.0
4277	4.8
4289	0.7
4343	448.5

PART L - R73

1833	35.3
1835	2.0
2280	1.0
3258	22.0
3324	3.0
3356	50.0
3360	14.0
3378	3.0
3630	16.0
3713	66.0
3714	152.0
3943	27.4
4209	32.0
4210	2.5

TABLE C-2 (Continued)

COMPLETED TACS

PART L - R73 (Continued)

TAC	TOTAL TIME (MANHOURS)
4223	2.0
4238	14.0
4241	38.0
3600	10.0
PART M - JNK	
2137	1.8
2422	14.0
3275	58.0

TABLE C-3

STATISTICS

ITEM	N	$\bar{x}$	σ	$\bar{x} + \sigma$	$\bar{x} - \sigma$
R18	10	83.2	130.4	213.6	0
R35	1	32.8	0	32.8	32.8
R37	12	40.8	65.3	106.1	0
R38	10	53.9	63.5	117.4	0
R40	1	55.4	0	55.4	55.4
R51	22	55.4	70.5	125.9	0
R52	1	47.0	0	47.0	47.0
R53	3	29.3	25.4	54.7	4.0
R54	2	83.6	78.5	162.1	5.0
R55	9	37.7	52.2	89.9	0
R56	16	191.4	470.7	662.1	0
R71	21	37.3	93.4	130.7	0
R73	23	54.6	110.8	165.4	0
JNK	3	24.6	24.1	48.7	0.5

TABLE C-4

Condition Based on Total Time (T)	Result Used
$T < \bar{x} - \sigma$	$\bar{x} - T$
$(\bar{x} - \sigma) < T < \bar{x}$	$\bar{x} + \sigma - T$
$\bar{x} < T < (\bar{x} +\ \ )$	σ
$T > (\bar{x} + \sigma)$	$[\bar{x} - \sigma]$

TABLE C-5

PROJECTED TACS

PART A - R18

TACS		TIME		
Number	Due Date	1976	1977	1978
3329	none	460.5	0	NA
3916	none	8.0	205.6	NA
3918	none	36.1	177.5	NA
4370	none	5.0	208.6	NA
6 x .71 x 83.2		=	354.4	NA
10 x .71 x 83.2		=	NA	590.7
		TOTAL	946.1	590.7

PART B - R35

	TIME	
	1977	1978
1 x .71 x 32.8	23.3	23.3

PART C - R37

3905	none	10.7	95.4	NA
11 x .71 x 40.8		=	318.6	NA
12 x .71 x 40.8		=	NA	347.6
		TOTAL	414.0	347.6

TABLE C-5 (Continued)

PROJECTED TACS

PART D - R38

	TACS		TIME	
Number	Due Date	1976	1977	1978
3919	none	49.1	68.3	NA
6372	none	8.0	109.4	NA
8 x .71 x 53.9		=	306.2	NA
10 x .71 x 53.9		=	NA	382.7
		TOTAL	483.9	382.7
		PART E - R40		
1 x .71 x 55.4			39.3	39.3
		PART F - R51		
300	none	0	55.4	NA
472	none	0	55.4	NA
554	none	0	55.4	NA
969	4/15/77	0	55.4	NA
1075	none	29.0	96.9	NA
1408	none	2.0	125.9	NA
1796	none	0	55.4	NA
3244	5/20/77	169.0	15.1	NA
3297	none	16.0	109.9	NA
3953	1/21/77	0	55.4	NA
12 x .71 x 55.4		=	472.0	NA
22 x .71 x 55.4		=	NA	865.3
			1026.3*	865.3

* less 969, 3244, 3953

TABLE C-5 (Continued)

PROJECTED TACS

PART G - R52

TACS		TIME		
Number	Due Date	1976	1977	1978
2370	none	47.0	47.0	47.0
		PART H - R53		
3256	none	65.0	3.9	NA
2 x .71 x 29.3		=	41.6	NA
3 x .71 x 29.3		=	NA	62.4
		TOTAL	45.5	62.4
		PART I - R54		
3436	none	162.1	5.1	NA
3446	none	0	83.6	NA
2 x .71 x 83.6		=	NA	118.7
			88.7	118.7
		PART J - R55		
2129	none	35.9	54.0	NA
2131	5/20/77	0	37.7	NA
2134	none	0	37.7	NA
2135	9/30/77	0	37.7	NA
2140	none	56.5	52.2	NA
2141	9/1/77	0	37.7	NA
2142	none	15.5	74.4	NA
2 x .71 x 37.7			53.5	NA
9 x .71 x 37.7			NA	240.9
		TOTAL	330.9*	240.9

* less 2129, 2131, 2135, 2141

TABLE C-5 (Continued)

PROJECTED TACS

PART K - R56

TACS		TIME		
Number	Due Date	1976	1977	1978
3279	none	0	191.4	NA
3306	none	0	191.4	NA
3681	none	8.0	462.7	NA
3721	none	0.6	470.1	NA
3926	none	0	191.4	NA
4297	none	0	191.4	NA
8300	7/1/78	284.3	470.7	279.3
8365	none	36.9	625.2	NA
8 x .71 x 191.4		=	1087.2	NA
15 x .71 x 191.4		=	NA	2038.4
		TOTAL	3881.5	2038.4*

* less 8300

PART L - R71

1623	none	23.0	107.7	NA
3378	none	5.0	125.7	NA
3971	none	1.0	129.7	NA
4062	none	14.0	116.7	NA
4385	none	0.0	37.3	NA
16 x .71 x 37.3		=	423.7	NA
21 x .71 x 37.3		=	NA	556.1
		TOTAL	940.3	556.1

TABLE C-5 (Continued)

PROJECTED TACS

PART M - R73

TACS		TIME		
Number	Due Date	1976	1977	1978
3258	none	142.0	110.8	NA
3369	none	50.7	114.7	NA
4209	none	32.0	133.4	NA
4223	none	2.0	163.4	NA
4343	none	539.6	56.2	NA
4378	none	0	54.6	NA
4390	none	0	54.6	NA
16 x .71 x 54.6		=	620.3	NA
23 x .71 x 54.6		=	NA	891.6
		TOTAL	1308.0	891.6
		PART N - JNK		
3 x .71 x 24.6		=	52.4	52.4

L. Woods

= 3

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

JUL 3 1 1979

MEMORANDUM FOR: Roger J. Mattson, Director, Division of Systems Safety

FROM: Darrell G. Eisenhut, Acting Director, Division of Operating Reactors

SUBJECT: SUGGESTED MATERIAL FOR INCLUSION IN YOUR RESPONSE TO C. MICHELSON'S 06/04/79 MEMO TO D. OKRENT CONCERNING PIPE BREAK ISOLATION

This information is provided to assist you in your response to Dr. Michelson's memo.

The potential for incorrect operator action resulting in break isolation on a BWR was a concern which the NRC staff also identified and investigated a year ago. Results of our review were presented in my June 14, 1978 letter to R. L. Gridley of GE (enclosed).

We concluded that BWR/4 plants with the "LPCI Modification" had the greatest potential for experiencing a break with subsequent isolation of the break. This is because both recirculation loop discharge valves would be automatically closed after a LOCA, as required for proper operation of the LPCI system on those plants. A single failure is thus possible (closure of the broken recirculation loop's suction valve) which would isolate the break. However, GE presented analyses and/or acceptable arguments for these plants showing that all 10 CFR 50.46 requirements would be met for any "isolated" break. This is because the ADS would complete the depressurization. The delay in depressurizing would be more than compensated for by the postulated availability of all ECCS (LPCI and CS) once depressurization was completed. (The single failure for this scenario is closure of the suction valve; therefore, no other single failure was required to be assumed in the analysis.)

We concluded that BWR/3 plants could probably be analyzed for an "isolated" break and the consequences shown to be acceptable, just as discussed above for the BWR/4 "LPCI modified" plants. However, it was possible to easily modify the plants so that no single failure could cause an "isolated" break, thereby obviating the need for such analyses. The loop selection logic system previously closed both the suction and discharge valves on the unbroken loop;

Contact: R. Woods, RS/DOR
28053

selection of the wrong loop (the single failure) could cause an "isolated" break. However, since closure of neither suction valve is required for proper operation of the LPCI systems on those plants (only discharge valve closure on the unbroken loop is required), the NRC staff required that the closure of the suction valves by the loop-selection logic system be removed. Thus, break isolation due to a single failure is no longer a credible event for these plants.

We concluded that BWR/1, BWR/2, and non-LPCI-modified BWR/4 plants do not have an automatic closure feature on either valve of the broken loop. Therefore, at least two failures would be required to cause break isolation. The exact definition of the multiple failures required varies among the designs, but break isolation did not have to be considered for those plants, and therefore, no analyses of such events were performed.

All operating BWRs are included in the above discussions.

In addition to the above summary of background material on the "BWR break isolation" problem, we offer the following comments on specific parts of Dr. Michelson's letter.

With respect to the LER he cites on Dresden 2 (77-16), that event occurred in 1977 and indeed at that time, on that plant, it might have been possible to improperly isolate the break in a previously unanalyzed manner due to a single failure. However, Dresden 2 has now been modified so the event is not credible due to a single failure. (It is one of the BWR/3 plants discussed above whose automatic suction valve closure system on the unbroken loop was disabled as a result of our action in June 1978.)

With respect to the need for new operating procedures which would prohibit break isolation, as discussed below we do not believe unacceptable consequences would result from break isolation on any BWR. On this basis, emergency operating instructions do not need generically to forbid isolation of LOCAs in the recirculation system. However, the emergency operating instructions should discourage indiscriminate isolation, i.e., the operator should not isolate the break until he has determined that adequate high-pressure makeup is operational and/or that reactor depressurization capability is available. Non-limiting LOCAs are not expected to be instantaneous or as large as the DBA, but may be more frequent (e.g., LOCAs resulting from recirculation pump flange or seal leakage or pipe crack). For these LOCAs, the trained operator has time to evaluate the situation and perform isolation that prevents further degradation of plant condition; this is in the best interest of safety. Thus, our review indicates that it is desirable to retain approved emergency operating procedures

that allow the operator to perform isolations determined to be beneficial, and that the limiting breaks are acceptable considering a single operator error.

We understand that T. Novak in the Bulletin & Order group plans to look into plant emergency procedures and that he will be looking for any "counter-productive" instructions they may contain. Until this study is complete it seems unnecessary and unwise to further limit the operator's use of installed plant equipment.

With respect to the break isolation concern for PWRs, operating PWR susceptibility is limited to those Westinghouse plants equipped with loop isolation valves (Yankee Rowe, Connecticut Yankee, Beaver Valley, North Anna, Zion, and Surry). All have Technical Specification requirements that these valves be disabled during operation.

In all of the above discussions, it is important to realize that "single failure" applied to manual actions is taken to mean a single operator action such as erroneously opening or closing a single valve. Instead, single failure could be taken to mean the operator incorrectly believes he should isolate the recirculation pump. He would then proceed to perform however many actions are necessary to accomplish the isolation. Such a new definition of single failure would mean that there are potentially more severe, unanalyzed events that could occur for each of the above discussed classes of plants.

For LPCI modified BWR/4 plants, the analyses that were performed took credit for a pressure permissive that will not allow the recirculation discharge valve to close until reactor pressure is below 300 to 350 psig (varies from plant to plant). This minimizes the possible delay between the time when break isolation becomes possible and the time when the LPCI and CS can initiate. (ADS will open two minutes after the LOCA signal, continuing depressurization and thus allowing LPCI and CS to initiate.) If the operator closes both the recirculation suction and discharge valves manually, the pressure permissive is not present and break isolation could occur somewhat earlier.

For plants with loop-selection logic (BWR/3s and non-LPCI-modified BWR/4s), if the loop-selection-logic system selects the wrong loop, the discharge valve on the broken loop would be closed (one failure). If the operator then closed the suction valve (another failure), break isolation would result. Alternatively, the operator could close both valves and cause break isolation.

For BWR/1 and BWR/2 plants, the operator could close both valves and cause a break isolation.

We feel that if events such as discussed above that involve more than one incorrect valve closure, etc. are to be analyzed, they should be required as part of a generic study that re-evaluates single failure requirements for all types of analyses and should not arbitrarily be required for this particular case. This is particularly true for the question of "isolated" BWR breaks because we believe no safety problem is involved. We believe that if analyses of early operator actions isolating a break were performed for all BWR plant types, such analyses would likely show that any delays in LPCI and CS initiation caused by break isolation would be largely or wholly compensated for by availability of all LPCI and CS systems once depressurization by the ADS was complete. Additionally, break isolation is only of concern for breaks considerably smaller than the DBA (where depressurization is slower than the DBA); these smaller breaks typically have PCTs considerably below 2200°F, providing a margin of acceptability even if the greater equipment availability after depressurization should be found to not totally compensate for the delay in LPCI and CS flow initiation.

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors

Enclosure:
As stated

cc: B. Grimes
R. Tedesco
P. Check
T. Ippolito
G. Lainas
D. Ziemann
J. Burdoin
ORB/2 and ORB/3 PMs
C. Berlinger
F. Coffman
B. Mills
R. Woods
Section B/RSB

R. Woods

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

JUN 14 1978

Mr. R. L. Gridley, Manager
Operating Reactor Licensing
General Electric Company
175 Curtner Avenue
San Jose, California 95125

SUBJECT: POTENTIAL FOR BREAK ISOLATION AND RESULTING GE-RECOMMENDED BWR/3 ECCS MODIFICATIONS

Dear Mr. Gridley:

We have completed our review of GE's recommendation to utilities owning BWR/3 plants to remove the post-LOCA auto-closure feature on the recirculation loop pump suction valves. We have concluded that plant operation with this feature removed is acceptable for the reasons given in the enclosed Safety Evaluation Report. Our review of each proposal submitted by individual licensees to remove the auto-closure feature will consider instrumentation and control features such as required valve position indication in the control room.

Also, in the course of our review of this matter we have studied the potential for break isolation on other BWRs. We have concluded that no problems exist for the reasons given below.

LPCI-Modified 218" BWR/4 Plants (Hatch-2 et. al.)

This class of plants was specifically analyzed by GE. The suction valves on both loops are not signaled to close. That is, closure of the suction valve would have to be the postulated single failure and no other failure would then be assumed. Basically, there are two reasons why the postulated failure (suction valve closure) is not limiting:

1. There is a pressure permissive on the discharge valve which prevents it from starting to close until reactor pressure is below 350 psig. The suction valve is not closed by the LOCA signal, on either loop, on any LPCI-modified plant; therefore, a single failure or operator error must be assumed to postulate closure of the suction valve and introduce the possibility of an isolated break. For most breaks larger than 0.5 ft.^{2}

(particularly for the limiting breaks which are well above 0.5 ft.^{2}), the vessel will be depressurized to below the LPCI and CS pumps' shutoff head before the discharge valve can complete its closure (a necessary condition for the accident postulated; suction valve inadvertent closure can't cause isolation until the discharge valve is closed). For all breaks smaller than 0.5 ft.^{2}, the ADS is open before the discharge valve can start its closure (the slower depressurization, with these smaller breaks, keeps pressure above the 350 psig permissive until the 120 sec. ADS delay is completed). Thus the ADS will complete depressurization allowing LPCI and CS flow even if the suction valve is inadvertently closed.

2. For certain size breaks somewhat above 0.5 ft.^{2} but well below the limiting break size, it is possible to postulate that the discharge valve may close before the core pressure reaches the LPCI and CS pumps' shutoff head (and if the suction valve is then closed at precisely the wrong time, the core could be uncovered and isolated with only HPCI water being injected). However, the ADS will open shortly after the isolation (120 sec. following whenever the water level reached the low-low setpoint). LPCI-CS flow initiation might be delayed a short time while waiting for completion of the ADS 120-second delay, but GE analyses for Hatch-2 have shown that this delay is more than compensated for by the availability of all LPCI and CS pumps once ADS gets the system to a pressure where they can function. (The single failure is the suction valve closure; therefore, all ECCS must be assumed available.) Ultimate reflood is considerably earlier in all such cases, resulting in lower PCT even though initiation of the low pressure systems could be somewhat delayed.

LPCI-Modified 251" BWR/4 Plants

Although plant-specific analyses have not been performed, the postulated event should be less severe than for the 218" plant size discussed above because the pressure permissive on the discharge valve closure is set at 250 psig (compared to 350 psig on the 218" plants). Thus it would be even harder (or impossible) to postulate closure of the discharge valve before the LPCI and CS pumps are injecting. Just as with 218" plants, if the time average pressure while the LPCI and CS pumps are injecting is higher due to break isolation, this is more than compensated for because all of the pumps would be assumed to be available, and ultimate reflood will be earlier.

All Other LPCI-Modified Plants (Vermont Yankee and Cooper)

Vermont Yankee has a 350 psig permissive on the discharge valve closure (i.e., the same as Hatch-2, which is higher than the 251" plants), but it is a smaller core (designed for a 180" vessel) installed in a 204" vessel. The resulting extra water inventory would cause a slower

depressurization, allowing more time before the 350 psig permissive pressure is reached (before which isolation is not possible) and thus making it more likely that ADS will be "timed out" and open before isolation becomes possible. This should make the postulated event in question relatively less severe than for Hatch-2. Cooper has a 235 psig permissive on the discharge valve closure, and a LPCI pump shutoff head of about 300 psig. Thus isolation should not occur above the pressure where low pressure systems can operate.

All Other BWRs (1's, 2's, 3's with above-discussed change and non-LPCI-modified 4's)

These plants do not have an automatic valve closure feature on either valve of the broken loop. Therefore, plants other than "loop selection" logic plants (discussed below) would have to assume two failures to isolate the break, that is, suction valve closure and separate discharge valve closure. "Loop selection logic" plants close only the discharge valve on the unbroken loop; therefore, selection of the wrong loop as the assumed single failure could only close one of the two valves necessary for isolation of the break. A second failure would be necessary to close the suction valve. Therefore, an isolated break as the result of a single failure on all of these plants is not credible.

If you require further information, please contact Dr. R. Woods of my staff.

Sincerely,

Darrell G. Eisenhut, Assistant Director
for Systems & Projects
Division of Operating Reactors

Enclosure: As stated

cc: L. Gifford
R. Woods

ATTACHMENT

SAFETY EVALUATION
BWR/3 ECCS MODIFICATIONS

Background

Motor operated valves are placed on BWR recirculation suction and discharge lines. Following a loss-of-coolant-accident (LOCA), if either of these valves on the unbroken recirculation line closes and if the low pressure coolant injection system (LPCI) supplies ECCS water to that loop, then the LPCI water will flow through the jet pump nozzles into the lower plenum where it will contribute to core reflooding. If neither of the valves closes, the LPCI water could flow backwards through the unbroken loops' recirculation pump, around the downcomer, and out the break, thereby not contributing to core reflooding. To provide redundancy, BWR/3 ECCS designs incorporated automatic closure of both the suction and discharge valves (on the unbroken loop only) upon receipt of a LOCA signal.

Statement of Concern

However, assumed single failure of the loop selection logic system can result in selection of the wrong loop as the broken loop. This would cause the following two events.

1) All LPCI flow from both LPCI systems would be directed to the broken loop and would be lost out the break. This effect has been considered in BWR/3 ECCS-LOCA analyses; as a result, no credit is assumed for LPCI flow.

2) Both the suction valve and the discharge valve on the broken recirculation line would close. If the break location were between those two valves, the break would be isolated from the reactor vessel. Although this could be advantageous under certain conditions, under other conditions it could introduce undesirable effects which have not been adequately considered in previously performed ECCS-LOCA calculations. That is, for a certain range of break sizes, it is possible that core uncovery could occur with vessel pressure above the LPCI pump shutoff head. If break isolation were to occur at that time, LPCI flow could be delayed and/or reduced, resulting in a later core reflooding and a higher PCT.

Discussion

Compensating effects exist that partially or wholly compensate for the above undesirable effects. The High Pressure Coolant Injection (HPCI) and the Automatic Depressurization System (ADS) would complete depressurization to the point where LPCI could function. Although such LPCI operation would be delayed, credit can be assumed for the full complement of ECCS equipment since the required single failure has already been assumed (loop selection logic failure, selection of the wrong loop).

Preliminary calculations indicate that the above described compensating effects would result in PCT's for the worst size isolatable break that are below 2200°F. However, a fully approved model meeting all requirements of Appendix K to 10 CFR 50.46 does not exist which is capable of calculating a postulated break that becomes isolated. Also, the preliminary calculations were not performed for all sizes of BWR/3's. Consequently, it is not possible to categorically state that 10 CFR 50.46 requirements are met for all isolatable breaks for all BWR 3's.

Therefore, General Electric Company recommended, and we require, that the automatic closure feature on the suction valve be disabled. This makes break isolation a non-credible event which does not require analysis: two independent failures are necessary, i.e., closure of the discharge valve in the broken loop (requiring loop selection logic failure), and closure of the suction valve in the same loop (for example, by operator error).

Conclusion

No credit has been assumed for closure of the suction valve in any safety analyses other than ECCS-LOCA analyses.

For ECCS-LOCA analyses, closure of the suction valve provided a backup function for closure of the discharge valve on the unbroken loop. With the recommended modification (suction valve closure disconnected), single failure to close of the discharge valve on the unbroken loop will now cause failure of the LPCI system. However, this LPCI failure has already been taken into account by the ECCS-LOCA analyses on all BWR/3 plants. No credit is assumed for LPCI operation on BWR/3 plants, since single failure of the loop selection logic can cause complete failure of LPCI. Stated another way, the recommended change merely creates another potential path to a failure that is already accounted for in the ECCS-LOCA analyses, that is failure of LPCI; however, the recommended change precludes possibility of an event which has not been accounted for in the analyses, i.e., break isolation.

As stated above, elimination of the automatic closure feature on BWR/3 plants is a desirable change since it eliminates the potential for an event which involves unreviewed safety concerns. The change does not create any new unreviewed safety concerns. We, therefore, find acceptable BWR/3 operation with disabled suction valve automatic closure following a LOCA signal.

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

MAR 1 2 1976

Docket No. 50-320

R. C. DeYoung, Assistant Director for LWRs, DPM

UNRESOLVED ITEMS IN ELECTRICAL, INSTRUMENTATION AND CONTROL SYSTEMS FOR THREE MILE ISLAND UNIT NUMBER 2

Enclosed is a list of electrical, instrumentation and control systems items for Three Mile Island Unit Number 2 which remain unresolved as of today. We note that each of these items has been previously identified to the applicant and that it was stated at the December 18, 1975, meeting with the applicant that the responses to these items were scheduled for January 20, 1976.

We request that the applicant be asked to address these unresolved items promptly. Further, it is imperative that adequate responses to these items be submitted by early April, 1976, to facilitate the EICSB to complete their input to the Safety Evaluation Report for Three Mile Island Unit Number 2, consistent with the overall project schedule.

R. Tedesco

Robert L. Tedesco, Assistant Director
for Plant Systems
Division of Systems Safety

Enclosure:
As Stated

cc: K. Kniel
H. Silver
T. Ippolito
M. Srinivasan
E. Ashe
J. Glynn

ENCLOSURE

UNRESOLVED ITEMS IN INSTRUMENTATION, CONTROL AND ELECTRICAL SYSTEMS FOR THREE MILE ISLAND UNIT NUMBER 2

(To obtain corresponding item numbers, see listing entitled "Pre-Safety Evaluation Report Items of Concern for Three Mile Island Unit Number 2.")

Item 1

B&W Topical Report BAW-10003

Item 2

Response Time Testing

Item 5

Emergency Feedwater Pump Control - (Revised drawing has not been received)

Item 7

Safety Related Display Instrumentation

Item 9

Bypassed Status Indication

Item 10

Hydrogen Re-Combiner Qualification

Item 11

Environmental Qualification of Safety-Related Equipment

Item 12

Electrical Penetrations

Item 13

Fire Stops and Seals

Item 14

Redundant Safety Buses - (Conformance to RG 1.6)

PRE SAFETY EVALUATION REPORT

ITEMS OF CONCERN FOR THREE MILE ISLAND UNIT NO. 2

A. INSTRUMENTATION, CONTROL AND ELECTRICAL SYSTEMS

Item 1

Conformance to the Interface Criteria resulting from the completed Generic Review of BAW-10003 for the as-installed RPS equipment.

Statement of Concern

Acceptability of referencing B&W 10003 for RPS equipment qualification.

Required Action by Applicant

With regard to this concern we request that you respond to each of the items under part 6 of the enclosed Topical Report Evaluation for BAW-10003. Where exceptions are noted detailed justification should be provided.

Item 2

Acceptability of the present Response Time Testing Program.

Required Action by Applicant

The present response time testing program does not include the periodic testing of response times of the temperature and flow sensors. The standard B&W technical specifications do not exclude these sensors from periodic response time testing. All forms of ex-system testing should also be considered for these sensors. If after considering these forms, these exceptions continue to exist, detailed justification must be provided. Also, the basis for claimed initial sensor response times for all sensors must be documented.

Item 3

Conformance to the Generic Resolution for Anticipated Transients Without Scram (ATWS).

Required Action by NRC

The staff has completed its review on a generic basis concerning ATWS and is in the process of formulating a detailed position pertaining to implementation of ATWS requirements for Babcock and Wilcox plants. We will transmit this detailed position as soon as it becomes available and will require that Three Mile Island Unit Number 2 conform to this position.

Item 4 Clarified In Amendment #37

Feedwater Latching System (FLS)

Statement of Concern

The FLS as shown on Burns and Roe Drawing Number 3090, Sheet 74, Revision 3 does not appear to be consistant with the information contained in Section 15.1.15.2.3.2 and Figure 15.1.15-1 of the FSAR.

Required Action by Applicant

Provide a clarifying discussion.

Item 5 Error - Will be Corrected

Concerns relating to the permissive start interlocks for the Emergency Feedwater Pumps.

Statement of Concern

Burns and Roe Drawing Number 3090, Sheet 71, Revision 3 shows permissive start interlocks for both the motor driven and steam driven emergency feedwater pumps. It is not apparent from this drawing and/or other information the purpose of these interlocks.

Required Action by Applicant

Provide a clarification discussion.

Item 6

Documentation relating to the Decay Heat Removal (DHR) Low Pressure to High Pressure Isolation Valve Circuitry.

Statement of Concern

The motor-operated valves used to prevent overpressurization of the DHR System by the Reactor Coolant System are required to conform to the following criteria:

1. Two valves in series to isolate the DHR low pressure system from the high pressure system.
2. For motor-operated valves, the valves have independent and diverse interlocks to prevent valve opening at high pressure. The interlocks are designed to comply with the requirements of IEEE Std 279-1971.
3. The motor-operated valves are closed automatically whenever the reactor coolant system pressure exceeds the pressure rating of the low pressure system. The closure devices are designed to comply with the requirements of IEEE Std 279-1971.
4. Suitable valve position indication should be provided for these valves in the control room.

The present information contained in the FSAR does not reflect full conformance to this position. Of primary concern in this regard is the diversity of the interlocks provided and the valve position indication available.

Required Action by Applicant

Revision of FSAR to reflect full conformance to this position.

Item 7

Documentation relating to the Display Instrumentation necessary for Safe

Shutdown and Post-Accident Monitoring.

Required Action by Applicant

Provide additional documentation demonstrating the reliability of power sources for the post-accident display instrumentation. This information should be included in the FSAR.

Item 8

Concerns relating to the Changeover from the Injection Mode to the Recirculation Mode.

Statement of Concern

The limiting time frame for which the manual operations must be completed in order to accomplish the changeover has not been defined. Also, does the physical installation permit easy correlation of all the necessary instrumentation and controls which the operator will use.;

Required Action by Applicant

This information should be provided and docketed.

Item 9

Main Control Room Alarms for selected bypassed Safety-Related Components.

Statement of Concern

Selected safety-related components are bypassed and no control room alarms are provided to indicate the bypassed condition, reference response to Q 22.11.

Required Action by Applicant

Alarms located in the main control room should be provided for these components.

Item 10

Commitment regarding the Hydrogen Recombiner to be used for this plant.

Statement of Concern

The hydrogen recombiner to be used for this station requires additional seismic qualification testing. Atomic International has agreed to perform such testing. Should these tests reveal additional deficiencies we request that the applicant commit to correcting them.

Required Action

Docketed commitment concerning the above.

Item 11

Environmental Qualification for Safety-Related Instrumentation and Balance-Of-Plant Electrical Equipment

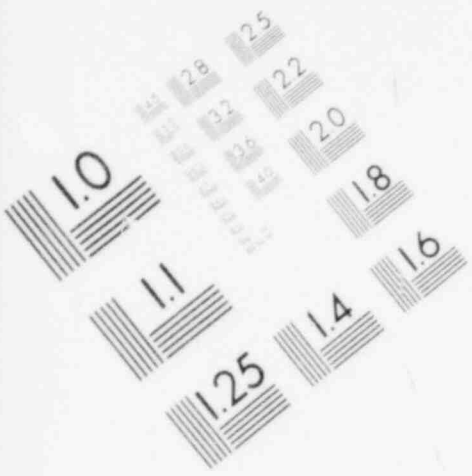
Statement of Concern

Recent re-analyses performed for this plant have indicated that the containment atmosphere temperature exceeds that of the containment design basis temperature for a short period of time following selected steam line breaks within containment. Also, selected equipment within the balance-of-plant scope of supply have no special environmental conditions specified.

Required Action by Applicant

11a Additional documentation which demonstrates the acceptability of any safety-related instrumentation located within containment which will be exposed to the noted higher temperature.

11b Additional documentation demonstrating the acceptability of the noted (above) balance-of-plant equipment for prolonged use at least within the design



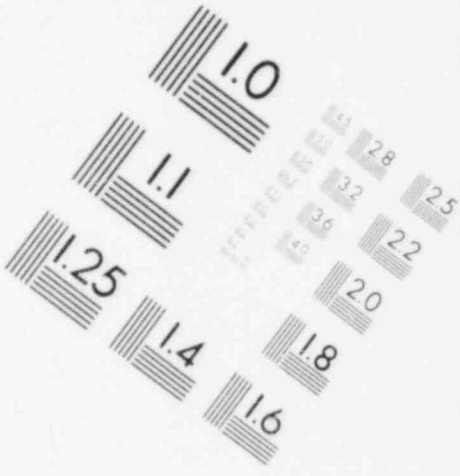
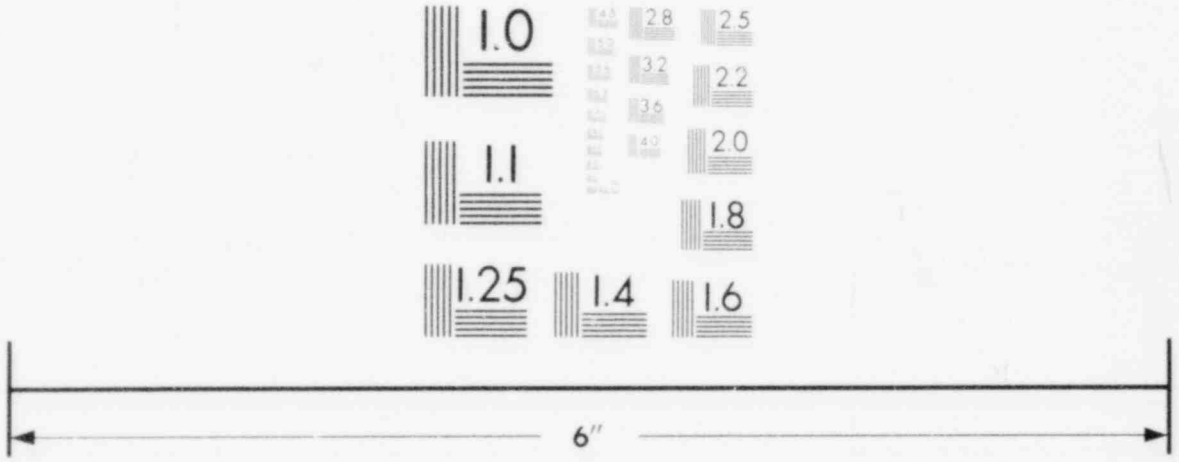
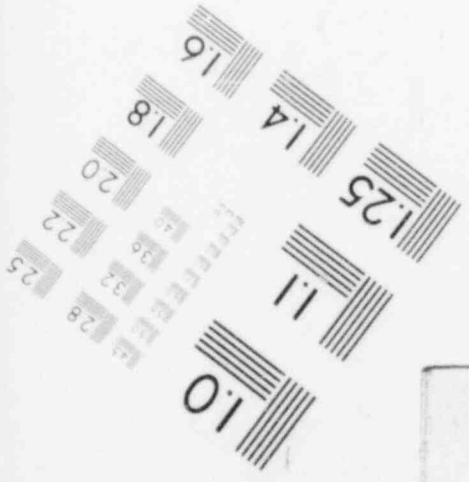
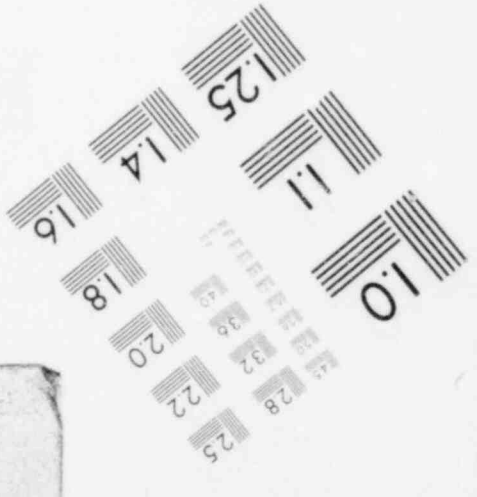


IMAGE EVALUATION
TEST TARGET (MT-3)



MICROCOPY RESOLUTION TEST CHART





normal environmental conditions at the particular locations as specified in Table 3.11-1 of the FSAR.

Item 12

Documentation relating to the Containment Electrical Penetrations.

Statement of Concern

The FSAR does not state that any short circuit tests on medium and low voltage penetration assemblies (both for safety and non-safety cables associated with the electrical penetrations) will be performed to insure that the penetrations assemblies can withstand, without loss of mechanical integrity, the maximum possible fault current versus time conditions.

Required Action by Applicant

These tests should be performed and their results documented in the FSAR. (If they are not to be performed, detailed justification must be provided.)

Item 13

Fire Stops and Seals

Statement of Concern

The interval for periodic insepections to be performed to identify open or deteriorated fire stops and seals has not been specified. Secondly, the administrative procedures and controls that will be followed when it becomes necessary to breach a completed fire stop or seal to add or remove cables has not been provided.

Required Action by Applicant

Documentation concerning these items should be provided. (We note that this general subject including fire detection and fire protection systems is being discussed generically within NRC and any additional requirements developed

will be applied to Three Mile Island Unit 2.)

Item 14

Concerns relating to the control circuitry capability to power selected components from two redundant safety buses and conformance of the design to RG 1.6.

Statement of Concern

Selected Safety-Related components have the capability to be powered from either of two vital buses, reference Q22.24.

Required Action

Discussion necessary concerning FSAR Figures 7.3-4, 7.3-5, 7.3-6, 7.3-7, 7.3-8 and 7.3-9. Also, discussion necessary concerning typical breaker starter combination schematic for which the above (statement of concern) is true. Additional documentation demonstrating conformance of the design to RG 1.6.

Item 15

Details of the Diesel Generator and Diesel Generator Auto Loading Sequencing Safety Feature Actuation System Control Circuitry.

Statement of Concern

Diesel Generator tripping devices and controls as shown on the detailed schematic diagrams: Of primary concern is the detailed design of the two out of three coincident lube oil pressure signal trips.

Also, diesel generator auto sequential loading circuitry: The main concern regarding this item is why on-line testing can't be performed for these

contacts (Should be able to show from detailed schematics).

Required Action

Oral discussion with the detailed schematics. We will required on-line periodic testing of this circuitry.

Item 16

Additional specific information concerning non-conformance to Regulatory Guide 1.41.

Statement of Concern

Identification of specific sources and loads which will not be disconnected during testing in accordance with Regulatory Guide 1.41.

Required Action

Check status of previous oral commitment.

Item 17 To Be Handle On "Site Visit".

Identification of plant areas and justification for less stringent separation distances than those recommended in Regulatory Guide 1.75.

Required Action

Identification of these plant areas. Also, supporting information should be provided which demonstrates that the present separation distances are adaquate.

Item 18

A note concerning the safety-related electrical schematics.

Confirm that the safety-related electrical schematics which were submitted illustrate accurately the "as built" designs. We are aware that some of the

design details will be determined during the course of construction and may not be available now. However to avoid any delay in the completion of our review, revised drawings should be submitted as soon as possible after the detailed design information is available or any design changes are made. We recommend that you review the status of the electrical schematic drawings and submit revised drawings, where appropriate.

Three copies of each electrical schematic drawing should be submitted. Non-proprietary versions of proprietary drawings, if any, must also be submitted. Since we will rely, in part, on the electrical schematics in reaching a conclusion on the design adequacy, the electrical schematics must be submitted under oath or affirmation. In addition, Section 1.7 of the FSAR should be revised to include a list of the proprietary and non-proprietary drawings submitted. The list should include the drawing number, title, revision number and date. The list should be revised when additional or revised schematic drawings are submitted.