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PRESIDENT'S COMMISSION ON THE

ACCIDENT AT THREE MILE ISLAND

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DEPOSITION of BABCOCK & WILCOX by JOHN ALBERT LIND, JR., held at the offices of Babcock & Wilcox, Old Forest Road, Lynchburg, Virginia 24505 on the 3rd day of July 1979, commencing at 9:05 a.m. before Stanley Rudbarg, Certified Shorthand Reporter and Notary Public of the State of New York.

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Associate Chief Counsel

ALSO PRESENT:

RONALD M. EYTCHISON

CLAUDIA A. VELLETRI

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(Document described below was marked Lind Deposition Exhibit 53 for identification, this date.)

J O H N   A L B E R T   L I N D,   J R.,   having been duly sworn by Winthrop A. Rockwell, Esq., was called as a witness and testified as follows:

DIRECT EXAMINATION

BY MR. ROCKWELL:

Q Would you state your current business address?
A My current business address?
Q Yes.
A Babcock & Wilcox, Old Forest Road.
Q And your current employer?
A Babcock & Wilcox.
Q And your current job title?
A I am lead instructor at the training program.
Q Have you brought with you today a resume which we have marked as Lind Deposition Exhibit 53?
A Yes, I have.
Q Is that resume complete and up-to-date?
A Yes, it is.
Q Was it prepared by yourself?
A Yes, it was.
Q When were you first licensed as a senior

reactor operator?

A I took the exam at Crystal River. It was April or May of '77. I received notification I had passed the exam after I returned to Lynchburg.

Q April or May '77?

A That is correct, while I was in Florida assigned to the master service team.

Q I take it from looking at your resume that you were in Florida for about eight or nine months?

A That is right.

Q From the fall of '76 until June of '77?

A That is correct.

Q And you indicate that your job there was shift supervisor augmentor?

A Yes.

Q What is that?

A The personnel that were licensed at Crystal River, the shift supervisors, had very little -- most of them had very little prior nuclear operating experience, and we were required to supply people with previous operating experience to help these shift supervisors run the shifts during the initial fuel load, zero power physics testing and power installation. I was one of the people assigned to do that.

Q How did you happen to acquire your senior

operator's license, senior reactor operator's license, while you were at Crystal River? Was that required that you become a licensed operator at that time?
A No, it was required for my job here, not for the job I had at Crystal River.
Q In what reference was it required for your job here?
A NRC requires that the instructors here who do give examinations to evaluate students and give examinations must have a senior reactor operator's license.
Q So I have this straight, the NRC requires that instructors here who give examinations --
A Yes.
Q (Continued) -- be senior reactor operator licensed?
A That is correct.
Q And is that a formal requirement of the NRC? In other words, is that set out in the regulations published by the NRC?
A I don't know. I know that Mr. Collins requires that we have a license to give the examination.
Q Mr. Collins?
A Paul Collins, head of the license branch.
Q How many of the other instructors in your Training Department are licensed?

A All but two.

Q And does that mean they are all currently licensed?

A Technically speaking, once you leave the plant and no longer operate the controls, you are not licensed as such.

Q I guess my question is how many of the instructors --

A To maintain the license --

Q Let me ask the question.

How many of the instructors in the training program hold current NRC operating licenses?

A I believe right now probably none.

MR. EDGAR: Do you want to define the term "current"? I think you have something to explain on the maintenance of the license, don't you?

THE WITNESS: Right. In order to maintain a license as such on a utility, you have to essentially operate the plant on a periodic basis. You have to stand watches. Since none of us are doing it any more, our licenses are not current. Some of our licenses for some of the people, the expiration date hasn't been reached yet, so in that sense they are still current, but in another sense, since we don't meet the requirements of standing watch on plant, the license has lapsed.

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Q Is it possible to substitute work on the simulator in lieu of standing watch?

A No. You must stand watch on your plant.

Q Showing what has been previously marked as Willse Deposition Exhibit 2, does that appear to be a chart depicting the organization of the Nuclear Service Department, now the Customer Service Department?

A I couldn't tell you anything specific about any of the departments except mine. As far as everybody in it, this chart is not up-to-date, as far as our department.

Q With respect to your department, we are referring to the department entitled "Training Services in Special Projects."

A This is not up-to-date.

Q Would you tell me where it is not up-to-date.

A Walter Perks is not lead instructor. W. E. Perks no longer works for Babcock & Wilcox.

Q Are there any corrections that should be made?

A Michael Penovich no longer works for Babcock & Wilcox. There are two additional instructors that are not on here -- I'm sorry -- there are three additional instructors that are not on here.

Q Who are the additional instructors that are not shown?
A The additional instructors are James Watson, Chuck Gaines, and Richard Sweeney.
Q With the corrections you have now made, would that chart be up-to-date and accurate?
A Yes. One other thing. Sweeney is assigned right now to Crystal River. He is not physically here. He is still working for me but assigned elsewhere. Also, W. G. Ferrell, who works for Rosser, above, but still in the dpartment, is now on assignment to TMI, not physically in the building.
Q As lead instructor -- that is the term you use?
A Yes.
Q -- do you report to Norm Elliott?
A Yes, I do.
Q What are your duties as lead instructor in, would you call it a unit or department?
A Department. My basic duties are scheduling all the training, both at the simulator and any site training that we do, coordinate the schedules. I normally, in addition to scheduling the courses, schedule internally as far as assignment of instructors

to teach different topics. I have responsibility overall for the simulator in terms of maintenance, making sure that the maintenance is performed, correct maintenance, both on the computer and on the hardware in the simulator area. And I also interface between Ralph Rosser and the other instructors, as far as simulator upgrading, software upgrading. I also do some other work, business-type work, relating to writing proposals and cost estimates and things like that.

Q This would be writing proposals for training?

A Yes. I do that on occasion. I don't do it all the time. I also am responsible for assigning the areas of expertise, developing new lesson plans, procedures, casualties on the simulator, yearly evaluations of instructors.

Q Anything more?

A Nothing that I can think of right off the top of my head. It pretty much summarizes most of the things I do.

Q What is Mr. Rosser's job?

A Mr. Rosser is responsible basically for the software on the simulator, developing programs, operations simulator.

Q How long has he been in that job?

A Mr. Rosser was involved with the initial developmental work on the simulator, which is about ten years ago, and he has been doing that ever since. He has had other job assignments in the interim, but he has always been essentially responsible for that type of work.

Q And that would be here within Babcock & Wilcox?

A Yes.

Q Had he come from a simulator manufacturer to this job?

A No.

Q So Mr. Rosser would know really the whole history of how that simulator has been used and how it has been programmed, is that correct?

A That is correct.

(Continued on Page 11.)

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My experience as far as the program goes is limited to an even shorter time span than this.

When I first came here, I did not do very much. I really only became responsible for this a year, year and a half ago. I have been involved with the upgrading, the software upgrading.

Q Let me give you a couple ground rules that may --

A I talked this morning before I came in. I am a little --

Q First of all, when I ask a question, try to wait until I finish asking a question before you answer. It makes it easier for the court reporter.

Secondly, they have to take it down, and if you talk too fast, they are going to tail behind you, and they also have to hear what you say.

A Okay.

Q With those ground rules, I think we will be able to go forward.

MR. EDGAR: Off the record.

(Discussion held off the record.)

Q Can you be more specific about what Mr. Rosser does; you say he is involved with basically the design of the software that is used on the computer?

A That is correct.

Q How does he go at that, what does he do?

A Well, I can probably explain by way of giving an example.

Q That would be great.

A We decided last year, well, it had been in the works for a longer period of time, that we are going to go ahead and convert the integrated control system from a hardware system to a software system; in other words, instead of having actually an integrated control system, we were going to load the ICS onto the computer and to do that we would have to develop the equation which enabled us to essentially make the ICS a software piece of equipment rather than a hardware piece of equipment.

If we upgrade any systems because of field changes that occur out in the industry, say there is a change in the control-rod drive system out in the field, and we decide that is going to be a good change to put in here because it is representative, we would go to Ralph, explain to him what had to be done, and he would essentially convert it into an equation that would be used in the computer.

We remodeled our core last year. We followed or we remodeled after Rancho Seco. Rancho Seco loaded fuel, essentially went on to their second cycle fuel load, and Ralph was responsible for taking that physics test manual data and using that to upgrade our core, so it would now be a second cycle rather than a first fuel load core. Those type of things.

Q Does he devote full time to the software on the computer?

A He devotes all the time we ask of him. He has other things that he does also.

Q What are those other things?

A He does some work for some of the other departments in terms of special projects.

Q Are those related to training?

A No.

Q What kind of special projects?

A The only thing I can think of offhand lately that he has been doing is he has been doing some work with developing alternate means of auxilliary feedwater control.

Ralph was in the Control Analysis Section, so he is still asked by people or who are still in the Control Analysis Section to help them do work on occasion. He does work on the hybrid model, which is another simulator that is not a control room simulator, but a softwear simulator, and he was involved quite extensively in some developmental work for the MK plant in Germany for BBR. Those are the last two important things I can think he worked on in addition to our training simulator.

Q Who makes the decision that a software update or a softwear modification should be made?

A Anybody in the department can make the suggestion, and it is basically up to Ralph, Norm Elliott and myself whether or not that suggestion would be augmented in terms of a change in program.

Q So Ralph is a participant in terms of making those decisions as well as implementing them, or is that really your decision?

A Well, really most of the changes we make now are essentially operational type considerations, refinement, certain things, perhaps adding additional casualties, the response to the plant, some areas that don't look quite right, some people with more operational background will look at it and say -- or Ralph would come in and say, "We want to do this, that and the other thing," because most of the stuff we do now would be putting in additional malfunctions or refining some small area or adding an additional control or something, and most of that would be originated by a person with more operational experience rather than Ralph, but Ralph might have decided he wanted to do something, and we would talk about it and decide if we would do it, if we came up with something that he liked to put in.

Q In this operations area, would those decisions generally be made by either you or Norm Elliott as to the decision to make the software change?

A Most of the time.

Q So that if I understand the relationship correctly, Rosser has a variety of duties; he is available to effect or implement changes in the software package?

A Yes.

Q When you feel that they are necessary you go to him and ask him to do it, and perhaps discuss with him how he will do it?

A That's correct.

Q But generally, the suggestions are initiated not by Ralph Rosser, but by others in the department?

A That is correct.

Q Is that a fair characterization?

A Yes, I would say that is fair.

Q Can you tell me what Norm Elliott's role in the department is?

A Norm essentially is in charge of money.

Q What do you mean by that?

A Norm takes care of most of the business arrangements involving the training. He is responsible for contract administration for training contracts. He is the man who decides how much money we get paid; salary administration.

Q Is he involved in the formulation of the content of the training program?

A Normally, no, not to any great detail. He may be involved to some extent when we discuss a proposal for a specific training course. That would be very general.

When we refine the content and get it down to specific scheduling and lectures and lecture content, normally, Norm is not involved in that, no.

Q Would it be fair to say that you are the primary decision maker with respect to the content of the training programs?

A Yes.

MR. EDGAR: By "content," you mean the technical material presented?

THE WITNESS: Yes. For example, if we are going to do a requalification.

Q What I mean by content is what is taught.

A Yes. The topics that will be given in a course.

Q Do you have a standardized group of courses that you are teaching now?

A Yes, we do.

Q And has that been a pattern as far as you know for the last few years?

A Yes, it has.

Q Can you tell me, let us take as of January 1, 1979 as a reference point.

A Okay.

Q Can you tell me the package of training courses that would be available to a utility. Tell me how many we are talking about first.

A I would be guessing at any number I give you, because we give, perhaps, just roughly, five or six standard courses, but there are a number of other courses available as package courses that can be utilized by a utility, and there is a training catalog which I can get for you if you would like, which essentially contains all the basic courses we do.

Q That would be very helpful.

MR. EDGAR: Off the record.

(Discussion held off the record.)

(Continued on following page.)

MR. ROCKWELL: Would you mark a document entitled "Nuclear Training Services, Babcock & Wilcox," which appears to include an index and a variety of other descriptive materials about the training program.

(Above-described document was marked Lind Deposition Exhibit 54 for identification, this date.)

Q Mr. Lind, showing you what the court reporter has marked as Lind Deposition Exhibit 54, do I correctly identify it as a brochure which is made available by the Training Department, which includes an index of standard courses and a description of various services offered by the Training Program?

A That's correct.

Q With reference to this Deposition Exhibit 54, can you show me the list of standard courses which Babcock & Wilcox makes available to its utilities.

A We will give any of these courses on request.

Q So we are referring to the second page?

A There are some additional videotapes.

Q Are we referring to the second page of the exhibit, entitled "Index"?

A Yes.

Q Are these courses that are listed in the index?

A Yes, they are.

Q And they are standard courses made available by Babcock & Wilcox, is that correct?

A Yes.

Q You indicated that there was a core of four or five courses that are given most frequently.

A Yes.

Q Would you identify which ones those are by their number and title, as indicated on the index.

A The standard courses we give, the ones most often given are "Replacement Operator Training" and "Simulator Requalification Training."

Q Standard Operator Training?

A Replacement Operator Training, T303.

Q And what is the other one?

A Simulator T304. Those are the most common that are given. The New Plant Operator Program, T301, is also given, and we also give a course called an M-3 course, and I would have to thumb through here to tell you exactly what the numbers are. I think it is M-3. Is it all right?

4.3 Q Yes.

A Here, T-103, Nuclear Power Plant Operations for Management which is actually a misnomer. It is a nuclear power plant operation for operators, rather than just management.

Q Of the total amount of training given by Babcock & Wilcox, how much of it is encompassed within these four standard courses?

A I would say probably about 90 percent of the training that we do for customers is one of those four courses.

Q Do you have a standard description or syllabus for each of those courses?

A Yes; a schedule, yes.

Q Can you describe what those schedules entail, in terms of what they say about the programs? Are they the teaching content of the programs, or are they --

A They are the subject matter content of the program, so we would lay out essentially on a five-day basis what topics are going to be taught in the class-room each day and what casualties would be performed on the machine that day.

Mr. Eytchison has the type of thing I am talking about.

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MR. ROCKWELL: Please mark this document as Lind Deposition Exhibit 55.

(Document described below herein marked Lind Deposition Exhibit 55 for identification, this date.)

Q Mr. Lind, showing you what has been marked as Lind Deposition Exhibit 55, can you identify that Exhibit. What is it?

A This is a copy of a schedule, a one-week schedule for a training course given at the simulator to three people from Three Mile Island.

Q Is this typical of a description of a standard course? Do you have descriptions like this for each course?

A We have descriptions like this for each course.

Q Am I correct in understanding that this Deposition Exhibit 55 represents the subject matters covered during one week of training?

A Yes, that is correct.

Q So that if a course were a number of weeks in length, there would be a summary sheet like this for each week of the training program?

A That is correct.

MR. ROCKWELL: Off the record.

(Discussion held off the record.)

Q Are these descriptions which are on file and which are used repetitively with respect to a particular course, and I am now referring to Deposition Exhibit 55?

A We have standard schedules for requal. training which is a one-week course. We would have a standard course for replacement operator which normally runs two weeks. We may make modifications to that standard schedule for any given week or two, if we either decide to change some topics or if we have perhaps given the same course last year, and the same people return, we do not want to give the same course, so for any given week there may be changes internally. We do have copies of this basic schedule that we work with.

Q And these schedules, as you say, represent the basic training schedules which are used as a reference point for teaching the course with some modifications that you may make from time to time, is that correct?

A Yes.

Q Are these outlines of subject matter covered by the course, such as we have marked here as Deposition Exhibit 55, revised from time to time?

A I am sorry, but I do not understand.

Q With reference to Deposition Exhibit 55, that is the outline of the subject matter to be covered in a requalification course in one week?

A Yes.

Q Is that description either of that course or similar descriptions of other courses updated from time to time, or revised?

A If you are asking, has this schedule changed periodically, the ans.er is yes.

Q What process do you use to update or modify those descriptions of course content?

A We will normally, for an example on this one, if this was a schedule of a course that was given to the people at TMI, for example, a year ago, we know they are coming back, and we wouldn't want to give them the exact same course, so there are certain subjects in here, and for example, let's say this, rod withdrawal limits instead of safety analysis, and we would sit down and talk and develop some new topics, and then bounce that against the people at the utility to make sure that those topics were acceptable. We would initiate a rough schedule and work with those people and finalize those schedules in terms of what subjects would be covered.

Q Referring to Deposition Exhibit 55, what I am trying to understand, is that the standard description?

A There is no standard description. It is upgraded periodically. There is no course set in concrete and never changes.

Q Is there any description that you have in the Training Department of what subject matter falls under the heading of requalification training?

A I do not know of any fixed guidelines for exact subject matter that should be covered for requalification. We are looking now at a requalification schedule for a hot license or a cold license course, and we do not have that much flexibility. There are subjects which must be covered. We cannot completely or essentially rewrite that schedule on a whim. This is a course for the utility to provide help in obtaining licensed operators, and as such that provides a framework more than the M-3 course, for example, which is totally internal training within the organization, familiarization with management people and engineers, and that course you can put anything in it, with no constraints on the subject matter.

Q Do I correctly understand that the document marked as Lind Deposition Exhibit 55 is not a standard description of requalification training, but rather is a presentation of the subject matter that happened to be presented in the course taken by the three people whose names appear in the upper right-hand corner?

A Yes, that is correct, it is typical.

Q If I understand you correctly, again, there is no standard written document that you have in your department which describes the areas that are encompassed by requalification training?

A None that I know of.

Q The decision as to what goes into a particular requalification training course is rather the product of your discussion with the utility in question?

A Internal discussion in the department and the utility.

MR. EDGAR: Off the record.

(Discussion held off the record.)

(Continued on following page.)

Q Again making reference to what you have before you as Lind Deposition Exhibit 55, it is understood that that presents the subject matter as developed by discussion within the Training Department and discussion with the utility involved, and is the subject matter of a course that would be presented on the particular date in question, is that right?

A Yes.

Q Would that course be used repetitively during a particular year?

A Yes. We would give the same course to all the people from TMI Unit 1, for example, who came down for requalification.

Q So everybody in a twelve-month period who came for requalification would get a course that is based on that outline?

A Yes.

MR. EDGAR: For a given utility?

THE WITNESS: For a given utility.

Q What factors do you take into account in building the curriculum, if you will, or the schedule of subjects to be covered in a requalification course?

A One thing that we always do, and one thing that does not change in these courses, we always discuss the ICS, Integrated Control System, because it is really the cornerstone of the plant operation.

Q What I am asking you to address is what are the basic underlying factors that you look to in terms of what must be the contents of the course. Is there any standard that you look to?

A What must be the contents of the course?

Q Yes. Let me expand on my question.

Do you simply think, what would be good to cover this year, or do you look at some basic standard or reference source with respect to what you think ought to be covered?

MR. EDGAR: Like a Reg. Guide?

A The modifications that I have made to the schedules that I have been involved with have been basic-type discussion points.

Q Is there any standard that controls the content of the course other than your judgment and the utility's judgment as to what might be necessary to be covered?

A I am not familiar with any standards.

Q To your knowledge, are you and the utilities free, in constructing a curriculum for a particular requalification course, to pick whatever subjects seem appropriate, useful and worthwhile?

A To my knowledge, yes.

Q Are there any subjects which you are required to include?

A No.

MR. EDGAR: This is on the requalification?

THE WITNESS: We are talking about a requalification schedule.

Q In constructing the curriculum for the particular course, and we are talking about requalification, is that right?

A Right.

Q Who do you talk to within the department? You said you have discussed it within the department. Who are the people you talk to within the department?

A The other instructors.

Q Is that essentially a group process that they all sit down together?

A Sometimes more than two, and sometimes only one; normally the discussion is between the lead instructor, at the present time myself, and the instructor who is in charge of that utility.

Q Are each of your instructors given a liaison role with one of the operating utilities?

A Right now not all my instructors have a utility, but yes, there are liaison roles between instructors and utilities, and some instructors have more than one utility at the present time.

Q Tell me which of your instructors have that liaison role and which utilities they have it with.

A Okay. Harry Heilmeier. He is cognizant for Florida Power and Three Mile Island.

Q Which unit?

A Both.

Q Units 1 and 2?

A Yes, and now making a transition and shifting to Consumers Power, shifting away from Midland, and giving that utility to someone else.

Q Did he have Midland before?

A He had Midland; he did not have TMI.

Q Is Midland the same as Florida Power?

A No. He is now in charge of TMI, shifting Consumers Power to someone else, and in the process of making a transition on that utility, and he has only recently assumed the responsibility for TMI.

Q Does he still have responsibility for Florida Power?

A Yes.

Q When did he assume responsibility for TMI?

A I can't give you an exact time frame; that transition has been gradual over the last few months.

Q When did it start, to your best estimate?

A In the spring; the end of the winter.

Q Who had responsibility for TMI before Harry Heilmeier?

A Ted Book.

Q How long did he have that responsibility?

A I would approximate it as about a year.

Q Who had TMI responsibility before Ted Book?

A Harry Heilmeier.

Q How long did he have it during that earlier phase?

A I believe that Harry has had responsibility for TMI since he came to work at B&W.

Q When did he come to work at B&W?

A He came to work at B&W at the beginning of 1977, January-February 1977. He would have begun to assume responsibility fairly early in that time frame. I can't give you an exact date. I was not here at that time.

Q Do you know who had responsibility for TMI before Harry Heilmeier's first round responsibility?

A I don't know.

Q To recapitulate, so I am clear, Harry Heilmeier took responsibility in terms of a liaison role for TMI training at around the beginning of 1977, held it for about a year, roughly early 1978, is that right?

A Yes.

Q And then Ted Book took it over for a year, and then Harry Heilmeier took it back sometime this last winter-early spring, is that right?

A That is close. I can't give you exact dates.

Q With respect to this liaison responsibility which you have described or which you have mentioned, can you tell me what it involves.

A The major liaison responsibility would be that of periodically going to the utility, talking to the people in the Training Department and the operators, and especially right before training courses start, so that if we are giving a requal. course for TMI, now Harry would be the one to discuss with them the schedule and the course contents and also talk to the operators and ask them if they have anything they would like to see and any problems they had down there, and essentially be responsible for the students that they have down there, and if they had anything that needs to get done there, he would take care of it.

Q Has there been one contact person at Metropolitan Edison for the kind of discussions you described the liaison person in your department would be having?

A I am sorry, but I don't understand the question.

Q You have said that your liaison representative within the Training Department here at B&W would go to Three Mile Island, for instance.

A Yes.

Q And talk from time to time.

Is there a person that is the contact person at Three Mile Island?

A He would formally see either Marshall Beers or Dick Zeckman up there first. They are the people in charge of the Training Department up there.

(Continued on Page 33.)

Q Do you know what their titles are?
A I don't know their official title -- training coordinators, I believe.
Q They are in parallel responsibility?
A Essentially, yes.
Q How often would your liaison man go down to, for instance, Three Mile Island, and talk?
A I normally attempt to send an instructor before the requalification program started at Unit 1 and Unit 2, twice. He would probably go up before each replacement operator, which is a startup certification course. If there was three or four in a year, he would probably go up three or four times prior to those. The exact amount of times he would go up would be a direct function of how many courses they had scheduled for a year.
Q With requalification, where you might give the same course 10 or a dozen times, he would not go up before each session; he would go up once before the first program was given?
A That is correct.
Q And you indicate you would talk to Marshall Beers and Dick Zeckman about how to structure the content of the course?
A He would basically go up with a schedule that

we had developed and say, "These are the things we thought might be good to do this year. What do you think?"

If they have any feedback or things they want to see or changes in addition or changes in the schedules we brought up, they would talk about it with Harry, and then Harry would normally talk to some of the operations people. He might talk to quite a few of them. He might talk to a few -- depending on time -- he might talk to the operations supervisor possibly even or the plant supervisor possibly.

Q How do your trainers have contact with the operators themselves before the training programs that are conducted here?

A When these people go up, when the liaisons go up, they normally go up for a two-day period and will make an attempt or will make an attempt to go in and visit the shifts. In other words, they will come in every eight or 10 hours and catch a different amount of people in the course of a day. In 24 hours they can talk with quite a few people at the utility.

Q Are they directed to do that by you?

A Yes.

Q To talk to the operators while they are there?

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A If possible, yes, we like to have them talk to the operators.

Q Is there a record which would reflect when they made trips to the utility for which they are responsible?

A Not a direct record. You could probably back out the trips from expense accounts and travel authorizations, but a direct record of trips made for that, we don't keep.

Q Do Marshall Beers and Dick Zeckman ever come down here during the courses to observe?

A Yes, they do.

Q How often?

A Marshall comes down every year for requalification, and Zeckman comes down occasionally. He will come down a few times.

Q When you say "Marshall"?

A Marshall Beers.

Q When you say Marshall Beers comes down for requalification, you mean he takes the course?

A He comes down and participates for a full week in the course.

Q Does he take the course?

A Yes, he takes the course.

Q Does he come down and observe other courses,

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not as a trainee, but simply as an observer?

A I have no recollection of Marshall ever coming down, except when he was scheduled to participate in one.

Q Has he indicated what his reason for coming down is? Is it simply to take the course or is it to evaluate the course?

A He has never expressed to me why he comes down.

Q Has he ever given you an evaluation of the course after he has come down?

A All students receive an evaluation comment sheet at the end of a course to put comments on, what they thought of the course, changes and improvements or how they felt about it. We give a sheet like that to every student in every course we give.

Q Do you know when the last time is that Marshall Beers was down for a course at B&W?

A I would have to check, but I believe Marshall was down January or February of this year because we did give requalification this year, and I'm just about sure Marshall was here. I am sure Marshall was down at the beginning, as a matter of fact, but I couldn't give you an exact date. We could find that again if it was necessary.

Q And do you know the last time Beckman was

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here?

A Zeckman was down for a startup certification course, but the date escapes me. I would have to check again on the records for that.

Q Would he have been down to observe or take the course?

A The last time that Dick Zeckman was down here, he was actually taking the course. He was preparing to get his license. That was the last time I can recall him being down here. He was actually a participant in a replacement course.

Q Why was he preparing to get his license?

A He wanted a license. I imagine it may be a requirement. They may have decided that they wanted him to have a license, "they" being the utility.

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Q I take it he had not had a license before that?

A I don't believe he had a current license on TMI. I do not know if he had any other operating license.

Q Other than people coming from Three Mile Island to take your courses, is there anyone who comes on a regular basis to observe the courses?

A No.

Q Have you ever had an evaluation from the people at Three Mile Island of your training courses, other than the sheets that you hand out to trainees?

A Nothing that I can recall.

Q With reference to the exhibit which you have before you, Deposition Exhibit 55, that outlines certain subjects which will be covered in the course, correct?

A Yes.

Q Is there in writing any description of specifically how those subjects are addressed in the training, in course outline in other words?

A Some of these would have course outlines. There are a few topics in them that do not have a course outline.

Q How does a course outline get written, and why are there course outlines in some areas and not in others?

A Course outlines are written by the instructors. Sometimes they will write them on their own just to prepare a topic if it is given, and other times I have requested that format outlines be developed, since I have been the lead instructor, for the record so that we can essentially increase the scope of topics that anyone can teach.

Q When you became lead instructor, were there any formal outlines for any of the subject matters covered in requalification courses?

A Yes.

Q What subjects were covered by those formal outlines?

A I can't remember all of them -- Control Rod Drive, Heatup and Coolant Curves, OTSG's, and another three topics here were given by people outside the Training Department -- the Control Analysis lectures and the lecture on Safety Analysis were given by people outside the Training Department.

I know that there are outlines around on those topics. Safety Analysis -- there are lectures on that in the Safety Analysis Department. Control Analysis -- I have lecture plans that Brian has given to me for my records or given to Walt before I was lead instructor, on the topics he has given.

Q Who reviews the material that is to be presented by a particular instructor in a particular course?

A I periodically audit the classes. The outlines, when they are developed, are duplicated and given to all the instructors to look at.

Essentially when a new lesson plan or a new topic is developed, the lecture will be given to the Training Department before it is given to any student. That will be the first case of evaluation of the topic, to make sure it worked well, that it made sense, and we would essentially critique it and bang against the guy to help him develop the thing and get a good understanding.

Q This is with reference to a formal outline?

A Formal outline or any new topics being developed. As we said earlier, we develop new topics periodically.

For a subject, say, that has been given for a period of time, but now we wanted to get a more complete formal outlined developed, we will normally assign someone to develop that outline, and then make copies of it, and it will be distributed to the instructors.

Q If you invite someone from outside the Training Department to give a lecture on a particular subject, is that person given any materials to work from?

A No.

Q Would it be standard operating procedure for you to review the content of that lecture before it is given?

A That is a difficult question to answer because the lectures that we have given by these people, for instance this Safety Analysis lecture given by Scott Lebeau, and the Control Analysis by Brian, have been given many, many times by the same person, so there will be no need to review the material they were doing on a lecture they had given perhaps 25 or 50 times.

Somewhere way back somebody might have originally worked on that, but Brian has been given lectures down here long before I ever worked for B&W, so I would not feel the need to audit Brian's outline.

I would ask Brian to come down and teach that program. He has done it before, and he would just give it.

Q Since you have been in the department, has anyone reviewed, for instance, Brian Delano's Control Analysis lecture?

A People have audited, sat in from departments and listened to Brian's exam. I guess you would call that a review.

Q Is that a systematic feature that you have of auditing?

A I am not sure what you mean by "systematic." I do assign someone to go there.

Q Do you have a procedure for auditing?

A A critique sheet or something that we would fill out, you mean?

Q No, with respect to which courses you audit and how often, do you have a procedure?

A No.

(Continued on following page.)

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Q How do you decide when to do it?

A I just decide when to do it.

Q When you do audit, do you do any critique?

A If I find that there is a problem in the topic or in the way the material is presented, I will normally talk to the instructor afterwards, yes.

Q Is there any written critique?

A Nothing that is retained. There might be notes jotted down on a piece of paper during the course of the two-hour session, but nothing that would be retained formally.

Q Do you have any records of your audits of the courses?

A No.

Q So what you are indicating is that, based on your audit, if you think if there is anything that requires attention, you will speak to the instructor afterwards?

A If I thought there was something that needed to be called to his attention, I would.

Q What is the last time you personally audited a course?

A I sat in on a steam generator lecture fairly recently. It was during the last few months. I don't remember the exact date.

9.2 Q And what would be the last time before that that you audited a course lecture?

A I am trying to think. I know the last course I sat in on, but I'm trying to think of when it was. I monitored rod withdrawal and power distribution limits lecture that was given I believe right at the beginning of this year or the end of last year.

As for my resume, I have only been a lead instructor for about six months now, so I haven't done a lot of auditing.

Q Your best estimate is that you have done perhaps a couple of audits since you have been the lead instructor?

A A few classroom and a number of simulator audits. I also will go in and sit and watch the conduct of the training in the simulator.

Q Who did you take over from as lead instructor? Was that W. E. Perks?

A Walter Perks, that is correct.

Q You say he has now left B&W?

A That is correct.

Q Did Perks give you any indication as to what auditing procedure he had used as lead instructor?

A He said that he periodically sat in on classes and audited them. That is the extent of the guidance.

9.3

Q Would the auditing procedure which you have been describing apply to the whole range of courses which are offered by B&W?

A Would I sit in on a lecture on any course?

Q In other words, is the auditing procedure which you have described the same for all courses? Is your approach the same?

A To sit down and watch the course being given and then talk to the instructor afterwards?

Q Yes.

A Yes. The methodology of the audit would be identical on any course that was given.

Q Did you make outlines available to your customers, that is outlines of material to be covered within specific topics, such as control analysis lecture by Brian Delano in this simulator requalification training that we have marked as Deposition Exhibit 55?

A Yes, we do. Anyone who would like a copy of any lesson plan on any topic that is given can have it.

Q And they get it if they ask for it, is that it?

A Yes.

Q But they are not automatically forwarded?

A No, but anyone who says, "Hey, that was really

9.4 good. Do you have any material I can keep on that," we are always glad to give them anything they like, anything we have.

Q You used the term "lesson Plan." Is there a lesson plan for each topic covered in the course?

A No.

Q In some cases it is just whatever the person lecturing presents?

A Yes.

Q And that may vary from course to course?

A It will vary from lecture to lecture, the exact method in which the material is presented. I feel that continuity of information is essentially identical. It may be given in a slightly different fashion, but the operators are all conveyed the same information on the topics that we don't have outlines on, and there are not too many of those, but the biggest example is ICS.

We don't or we have very few normal lesson plans on the integrated control system, and that except for a general overview type lecture, which we do have a lesson plan for, when we get into detailed discussions of the ICS.

First of all, the ICS varies from utility to utility when you get into great detail, so you could

9.5 not have a standard outline for the ICS that would cover all utilities.

Then again a lot of that type ICS review is a reflection of transients that have been done on the machine the day before and the response questions they have. A lot of the ICS becomes a question and answer-type discussion, rather than a formal "I talk and you listen" type of presentation.

So we normally don't have any detailed outlines on specific small areas of the ICS. We do have a fairly thorough outline and overall big overview, general overview, of the ICS.

(Continued on following page.)

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Q Let us take ICS as an example. Would the same person within B&W be likely to give the ICS lecture each time?

A The lead instructor for each utility normally is the person most familiar with that utility's ICS, and we strive when possible that that person always gives or attempts to give most of the oral ICS lectures to that utility.

Q So these lectures would come from within the Training Department is what you are saying?

A We generally try to have continuity in seeing that giving the ICS lectures, for instance, for Toledo, James Carson is the lead instructor, and he normally gives the Toledo Edison lectures.

Q What other subject matters leading to the requalification, what other subject matters require the most modification because of differences among your operating plants.

A Just take this schedule for an example?

Q Generally.

A The ones which require the most modification is ICS, which is unique. You have the CRDM System.

Q What?

A Control-Rod Drive System -- there are two generations of that system, so it is not one common system to all utilities.

We give lectures on reactor coolant pump parts, and there are a number of pump vendors, so the lecture differs from utility to utility, based on which vendor they use.

But we do have generic outlines for the two different generations of control rod drive, and the outline on the reactor coolant pump has sections in it which would discuss differences or different aspects of it.

Q What other subjects have to be modified substantially with respect to different utilities?

A If we are giving technology courses, which is the type of very extended classroom lecture series that we do as part of a re-license certification, part of that training is a very detailed description of different piping systems and control systems, and that would be unique from utility to utility.

But those are not lectures given on a regular basis as often as the requalification, for example.

Q Well, referring to the four courses which you said comprise 90 percent of the work of the Training Department, what other subject matter areas require modification because of differences among your operating utilities?

Just to recapitulate, you mentioned ICS, Control-Rod Drive, RC pumps, technical courses.

A Okay. For the M-3 course, which is one of the standard courses we discussed -- T103 was the course.

Q Nuclear Power Plant Operation for Management?

A Right. That is a standard course, and we do not -- we teach a power plant. We teach our power plant, so that is the same for any one. We teach a typical power plant because it is one where we can give a general orientation on how a B&W plant operates. We don't get into any great detail in a week.

So that essentially the course content for any given subject would be the same for any utility with almost no change.

For startup certification, the first week of that standard replacement operator training course, which I believe we identified as T301, the course topics like Reactor Theory, Reactivity Balance Calculations, Reactor Startups -- there is no significant difference from utility to utility for that type of information.

One subject that is covered that first week is Control-Rod Drive, and that would be changed.

Q What I am trying to get at here is what areas of difference, rather than the areas of similarity.

MR. EDGAR: He is trying to go through it.

THE WITNESS: I am trying to back into that. I have to review the courses in my head as I go through them.

As we said, ICS, and if we discuss safeguard systems, those vary generally. All safeguard systems -- Emergency Core Coolant Systems are the same for all B&W plants, but actuation systems, the exact number of components controlled by any control block may vary; so that would require, if we get into a detailed description of your safety features actuation system, it requires very specific information on a utility because essentially the utility and vendor design a large part of that system.

Then, as I said, for the longer courses, technology courses, anything which specifically addresses their plant, we talk about their system, their piping systems, and all of that is very, very particular for that utility. I think that about covers the general ones that vary widely.

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Q With respect to the ICS lecture, you indicated that that is generally given by someone within the Training Department?

A It is always given by someone in the Training Department.

Q How about with respect to control-rod drive systems?

A That is always given by someone inside the Training Department.

Q RC pumps?

A That is right now always given by someone in the Training Department.

Q When you say "right now," you mean that it has been changed recently?

A Not recently, but there have been lectures given on reactor coolant pumps to utility people by people other than Training Department personnel. There are some engineers upstairs who occasionally have given lectures on pumps, but that is not common.

Q How about the technical courses? Who teaches those?

A The only technical course that I can talk about with any great surety is the most recent one we gave, which is the Midland course, the one I have been involved with, in any detail since I have been here.

11.2 That course was given to a large degree by us with some aid from engineering.

Q And how about safeguard systems?

A The safeguard systems normally are taught by us, in the Training Department.

Q How do people teaching the courses in these five areas that we have identified, where changes or adjustments in the course material are made with respect to the particular configuration of the plant operated by that utility, how is that information about the differences acquired by the instructor?

A Well, we retain a technical manual and information on all the different pumps, for instance. I can speak for control rod drive because I normally give that lecture.

We have a copy of it available in the Training Department or upstairs in the library of all the plant control-rod drive technical manuals, which is what I use to develop the course outline.

We also are kept up-to-date on any field changes or modifications made to those systems internally.

Q How are you kept up-to-date?

A The lead instructor is on a routing list or mailing list of that type of information, and when I get it, I review it and then it is disseminated to

11.3 the department.

Q Is there any process by which you review the courses in these areas to be sure that the instructors have an adequate understanding of the differences in the systems from utility to utility?

A I'm not quite sure I understand what you mean.

(Previous question was read back.)

Q Do you want me to clarify that?

A Yes.

Q You have indicated that in five areas that we have identified there are adjustments that have to be made to the subject matter of the presentation to account for individual differences in the particular utility system, is that correct?

A That is correct.

Q And you have indicated that the instructors acquire the knowledge of what those differences are from essentially two sources -- technical manuals and information that comes to you from within B&W, which you then distribute within the department, is that correct?

A Yes, that is correct.

Q Is there any procedure which you use to see that your instructors, having that information available to them, adequately understand it, so they can present it in their courses which they teach?

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A No, I think we had better clarify something. I have got a feeling there is a basic piece of information missing.

Q Go ahead.

A When I talk about a system, like either ICS or the control-rod drive system or coolant pumps, I am generally talking about one or two people in the department that we designate as the system experts. Those are the people, and their responsibility is to keep up-to-date, to keep up-to-date information. They are the ones responsible for being aware of the fact that changes are made and exactly in great detail information on that system.

Q With respect to each utility?

A Well, no. Like I am systems expert on CRDM. I keep current on all control rod drive for all utilities. Harry Heilmeier is the systems expert on reactor coolant pumps. That is not to say that he is the only person that can talk on pumps, but he is essentially assigned that task, to keep up-to-date and make sure that the information is disseminated, useful information is disseminated to instructors.

I think that may be want you are looking for, and we were kind of moving in a different direction from that.

11.5 Q So what I understand you to be saying then is that you have two areas of concentration or specialization within your department: One is with respect to an overview of a particular utility, with respect to that utility's requirements?

A And a liaison role for personal interface between their Training Department and our Training Department.

Q And then there is another area.

A Which is a systems expert-type designation.

MR. EDGAR: A guy who is a lead guy for a system and follows it?

THE WITNESS: Yes.

(Continued on following page.)

Q So any particular person in your Training Department might be a liaison person with one utility and he also might be a systems person in the sense that he follows a particular system with respect to all the utilities?

A Yes, that is correct.

Q Are there any other areas of specialization or focus that you assign to your trainers?

A There is one other area that I have designated essentially and set some people to work on there.

Q What is that?

A Research and development of new topics. There are two people currently who work in that area, developing new lesson plans, elaborating on old lesson plans, and that type of work, doing research in the new topics which may be of interest or use in the Training Department.

Q What sources do they call on, these people that you have assigned to research and development, for their design of new lesson plans or their revision of old lesson plans?

A The entire building.

Q Who do they do that with?

A There is a library that is available to them with essentially an infinite amount of material, almost, and they are also quite free to use all the engineering people upstairs to help them develop lesson plans. That is normally what is done. We utilize people in different departments upstairs to help us develop topics.

Q Do they have available to them contact with the facilities in developing new lesson plans and revising old ones?

A Yes. The most recent example of lesson plan development that I have done, that we have done since I have been in charge, were not utility-dependent.

Q Can you tell me what you mean by that.

A We developed a lesson plan on rod withdrawal and power distribution limits. This is a generic topic which applies to all utilities. So the interface there is basically between my staff and the engineers upstairs who develop the limitations, these curves.

They will change slightly from utility to utility, the actual curves themselves, but all of that information is available in this building because that is information we develop here and give to the utilities. We don't need to get information from them, okay? We give it to them.

Q So that to that extent it would be essentially theoretical?

A That is not correct, really. These are limitations which apply to everyday operations of the plants. What we are talking about is the basis for the way these curves were developed -- a better understanding of why you have to follow these curves by explaining how they were developed, the considerations used in developing these limitations. I don't know if you can really call that theoretical.

Q Do you have in the Training Department a complete set of operating and emergency instructions for each of your operating utilities?

A A complete set, no. We do not have all operating procedures and all emergency procedures.

Q What do you have?

A It varies from utility to utility. For some of them we have more complete. At a minimum, for each utility we have the operating and emergency procedures which we use for our courses.

Obviously in the scope of a course this long, you would not utilize anywhere the 120 or 150 or 200 procedures that they might have available to them.

Since our course content is limited to a certain amount of evolutions. we cannot do a lot of things that you do at a plant because we don't even model these areas. We don't hold procedures. We would hold general procedures for moving a plant around, combatting emergencies, standard emergencies.

Q What system do you have for acquiring emergency and operating procedures and keeping them updated?

A We are on the distribution lists for the utilities. I have to jot this down -- let me think. We are on the distribution list for Met Edison, Oconee units of Duke Power, and Rancho Seco-SMUD. Midland doesn't have any procedures yet. Arkansas -- we have not been doing a lot of training with Arkansas. We have just recently begun to get reinvolved with training at Arkansas, and we will get back on their distribution list, or get on their distribution list. Florida -- we are not on their distribution list. We are not on the distribution list for Toledo as of now.

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Q When you use the term, "distribution list," what is it, and what do you get?

A We are sent a copy of all the changes that they make to their procedures. We start with a base set of procedures, and those procedures change, are updated and revised at the site, and we are mailed a copy of all changes that occur, and we incorporate those into our set of procedures, and then we sign the distribution memo or sheet and send it back to them acknowledging that we have received it and incorporated it.

Q To what extent are those procedures incorporated into the course content of a particular presentation?

A We attempt to use as many of the plant's procedures as we can. Obviously, there are certain differences which make it very hard to adapt a procedure wholesale onto our course. There are differences from utility to utility. If a procedure -- if the entire procedure will work, we will use that procedure. If there are a small number of sets which may be modified or deleted in there to make it work on our own machine, we will do that, but if the procedure is just very far afield from ours, then we use ours.

Q You use your procedure?

A Yes, instead of theirs. If their procedure won't

13.2 work on our machine, we use ours.

Q Do you have a set of procedures which are applicable to the simulator?

A Yes.

Q When you are working with those procedures that are applicable to your simulator, are copies given to the trainees?

A Copies are given to the trainees for their own retention if they ask for a copy, but normally they don't ask for one.

Q They are only given one if they are asked for?

A Let me backtrack just a second. For instance, let me take an example. For new operators, replacement training, startup certification, we would give them a certain amount of package information the first day. They will get integrated control system, some basic system descriptions for our plant, and a set of our procedures.

Q "Our plant" meaning your simulator?

A Yes. For replacement operators, they wouldn't get a set of our procedures. They don't need them. We always maintain a set in the simulator, so that when we are operating the plant, we basically have a book in there which is a procedure book for that utility,

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and it will contain their procedures and a mixture of our procedures and modified procedures, or a conglomeration, as it were, of procedures that together are a workable set of procedures for the material we are going to cover in the course.

MR. EDGAR: Off the record.

(Discussion held off the record, following which a brief recess was taken.)

MR. ROCKWELL: Please mark this document as Lind Deposition Exhibit 56.

(Metropolitan Edison Company Unit 1 simulator requal training, bearing on the first page the date, December 15, 1978, herein marked Lind Deposition Exhibit 56 for identification, this date.)

Q How long has the Training Department been on the distribution list for operating and emergency procedures from Metropolitan Edison?

A I think they were just put on that list recently. We may have been on it at some earlier time also, and got off it.

Q When were you recently put on the list?

A It has been in the last month, when we talked to Marshall Beers, and he said they were going to start sending all the material and put us on the list.

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Q Were you on the distribution list as of the 28th of March, 1979?

A No.

MR. GALLEN: This is for operating instructions or emergency?

MR. ROCKWELL: Both operating and emergency procedures.

Q Did you have any Metropolitan Edison operating instructions or emergency procedures in the Training Department as of the 28th of March, 1979?

A Yes; you know, a full set for our use, for our needs.

Q What do you mean by that?

A Distribution is an automatic way of upgrading the procedures. We have always upgraded our copies of the procedures from the utilities before we did it by liaison people.

Q My question is, what do you mean by a full set for our use?

A As I said earlier, we don't hold all their procedures. We hold the procedures of theirs that can be used or are used for the operations performed in the simulator. We would not have a full set of all their procedures.

Q How many of their procedures would you have;

13.5 What proportion of them would you have? Would it be 10 percent or 90 percent?
A I would be hazarding a guess, but I'd say, of operating and emergency procedures, in excess of 50, but not 100. It is hard to say.
Q How did you know they were current?
A We don't know that they are exactly current. When we get ready to do a course like this, we, for instance, over here, Harry had gone up to see those people, and he would make sure that he brought back with him the most updated copies of the procedures that we would use.
Q Would he make a specific request for --
A Yes.
Q -- for procedures?
A Normally, he would go up and say, let me have all of these procedures, and he would have a list, perhaps, of procedures we were going to use, and he would say, let me have a up-to-date copy of all these.
Q Who would he make that request of?
A To Marshall Beers or Dick Zeckman, the guys I mentioned.
Q Would those procedures be mailed back or handed to him?
A Sometimes they are mailed and sometimes they are carried.

Q So it is your recollection that before, for instance, the requal training that was being done for Metropolitan Edison personnel in the winter and spring of 1978-79, that all procedures relating to the subject matters covered by that course have been updated and were current in your files?

A Harry, I am sure, went up sometime before that started and would have requested the procedures.

Q The question is, were all of the procedures related to the subject matters covered by the requal course under way in the wrinter and spring of 1979, current and up-to-date in your files?

A I can't say for sure. I would think they were.

Q Showing you what has been marked as Lind Deposition Exhibit 56, do I correctly identify that as a document the first page of which is dated December 15, 1978, subject "Unit 1 Simulator Requal Training," and giving a list of names and dates for which they apparently were scheduled for requalification training?

A Yes, that is correct.

Q And this document related to Metropolitan Edison?

A Yes.

Q Did your liaison with Metropolitan Edison, Harry Heilmeier, go to Metropolitan Edison before this round of requalification was begun?
A Per my resume, I was not the lead instructor at this time; I don't have the slightest idea whether Harry went up or not -- I am sorry -- no, that is December 1978. I am sorry; I don't remember exactly if Harry went up. He may have gone up prior to the time I took over or right after or during the time that Walt and I were making the transition. I don't remember if he exactly went. He probably did. Harry normally goes up.
Q What specific requests relating to the subject matter of this round of requal training were made by Metropolitan Edison?
A Let me check the schedule.
There is a problem with the schedule. There is a lecture that was given that for some reason is not on this schedule, and I do not know why. There was a lecture given on Control Analysis by Brian Delano on push-pull operation that was requested by Metropolitan Edison and was given.
Q Are there any other subjects which were requested by Metropolitan Edison which were given?

A Not that I know of. I remember specifically that we do have arrangements to have Brian Delano to give that push-pull course.

Q How did you make the decision as to what subjects to include in that training program which is contained in Deposition Exhibit 56?

A I don't remember the exact process, but this would have been bounced against the schedule in the previous year to see what had been done the previous year.

Q Who made the decision as to what went into that program?

A This one here?

Q Yes, Deposition Exhibit 56.

A Well, Deposition Exhibit 56? I don't recall if it was myself or Walt Perks that actually worked up the schedule. This was right after transition time when I was first taking over, and I don't remember if I worked on the schedule specifically, or if it was put together before.

Q But either you or Walt Perks would make the final decision?

A Yes, we would be the ones who would make the final decision.

There are two new lectures on here, lectures that would not have been given the previous year.

Q Of the training instructors in your Training Department, how many have held a senior reactor operator's license on a B&W plant?

A All but one who is currently teaching, who holds a license, holds it on a B&W plant.

I said earlier that two of my instructors, associate instructors, do not have a license. Of the licensed instructors who actually now teach, all but one hold a license on a B&W plant, and another person in the B&W Training Department who does not have a license, who used to be an instructor and does not hold a senior license on a B&W plant.

Q Please give me the names of each person in your department who does not hold or has not held in the past a senior reactors license on a B&W plant?

A Jim Watson; he does hold a senior license at Calvert Cliffs.

Q Who does not hold or has not held in the past a license on a B&W plant?

A Jim Watson, Chuck Gaines, Richard Sweeney and Gene Alden.

Q All of the other personnel in the Training Department either presently hold or have in the past held an operating license for a B&W plant?

A Yes. I didn't mention Mr. Elliott. He does not hold a B&W license?

Q And has never held one?

A No.

MR. GALLEN: I want to clarify that Mr. Elliott does not teach a course.

THE WITNESS: No.

Q Could you define for me what you understand the purpose of the training conducted at B&W for your operating utilities is?

A To put it briefly, we like to teach them enough about the plant so that they can operate it properly, and we like to make our people good operators, knowledgeable, and able to cope with a variety of transient and upset conditions; that is the main thrust of our training.

(Continued on Page 71.)

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Q With respect to the requalification program referred to in Deposition Exhibit 56, were any course materials given to the students in the course?

A The students would have received a copy of the integrated control system analog digitalogic prints, and probably the only thing they would have been handed directly, although they may have requested it be information or material on some of these other courses, but the only thing that would have been sitting on their desk Monday morning when they came in would have been the ICS prints that would have been given them in their hands.

Q What are these prints? Are these computer printouts?

A They are detailed prints showing the control system, the logic involved in the control system, and the general flow path of the control signals, how they developed, and how they flow through the system.

Q Are there any other written materials given to the students in that course bearing on any other course covered, and I am referring now to the requalification training which is referred to in Lind Deposition Exhibit 56?

A To the best of my knowledge, no.

Q Were the students given any assignments to

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review or read materials in the form of homework?

A No.

Q Were the students evaluated in terms of their performance in the course?

A By anyone or by B&W?

Q By B&W.

A No.

Q Were they evaluated by anyone?

A Metropolitan Edison normally on Friday morni.g crosses over the group leaders to evaluate the other shift.

Q What do you mean by that?

A On the last day of the drills, they would come down with two groups, and there would be a person in charge of each group.

Q Who would that be? Would it be somebody from Met Ed?

A Yes, either a foreman or shift supervisor in charge of the three people that he worked with during the week, and on Friday morning normally those two supervisors will observe the other session's training to evaluate it, but B&W did not do any of the evaluation.

Q So what you are saying is that the Met Ed group leaders in the two training courses simply switched positions?

15.3 A Not switched, but they would come in, and say there would be three people that normally worked together all week, and the supervisor from the other shift would come in and observe those three, and then do the same thing in the afternoon, where the other group leader would shift over and observe the other group.

Q And there would be an evaluation?

A I believe that they make some comments on a sheet. I have seen evaluation-type sheets of their own that they use.

Q Have they ever given those evaluation sheets to you?

A No.

Q Are any tests or quizzes given to the trainees or students during the course?

A During the requalification course?

Q That is correct.

A No.

Q Is there any procedure which you have to evaluate the effectiveness of the training?

A We can judge to some extent the effect of the training by observing responses to certain casualties on the machine. In other words, if we talk about a specific topic in the morning on a lecture, and then

15.4 we give a related casualty the next day, and then we can judge the effectiveness of the presentation, and whether or not or how they respond to the casualty.

Q Was there any information given to Metropolitan Edison as to whether the trainees appeared to absorb or failed to absorb the information in the course?

A As far as the requal, no.

Q How about with respect to the initial operator licensing?

A For the longer courses, we give quizzes weekly, and also -- now, I have to speak from things that have been observed because I have never participated in a long training course as such all the way through. The last one was given way back, and we are getting reach to give a new one, but there hasn't been any of these six or eight or 10-week courses.

For Met Ed we came down for cold license, and the supervisory people who came down with them and stayed with them throughout, who essentially were auditing our classes and working with their people in addition to our training to make sure that the training was being done effectively.

As far as hard copy evidence of the effectiveness of the training, we did give weekly quizzes and final examinations.

Q Were the results of the weekly quizzes and the final examinations passed through to Metropolitan Edison?

A Yes.

Q Who makes up the quizzes?

A Training staff.

Q Do you approve them?

A I have never done this. I will when we get ready to do the next course if I am still lead instructor, or I will look at all the quizzes.

Q It is the duty of the lead instructor to review and approve the quizzes?

A I would consider it my responsibility to review the quizzes.

Q Have you given any long courses to Metropolitan Edison people?

A Myself, personally?

Q No, the department.

A Yes, the department has given cold license training to Met Ed people.

Q Do you know when the last one was?

A I believe the last time it was given was during the time I was in Crystal River at the end of 1976, beginning of 1977, in that time frame. I don't have a date.

Q You indicated that quizzes have been given in a so-called long course, and that was the course that you indicated was given at the end of 1976 and the beginning of 1977?

A There were a series of courses given from 1977 back for Unit 2 cold license courses, and during that time frame there were quizzes given on a weekly basis, I believe. Again, I am talking now just touching on this briefly; I was not involved in the training.

Q Have quizzes been given for cold license courses conducted for utilities other than Metropolitan Edison?

A One other example, we have given part or the beginning part of a cold licensing course now to Midland. We have given the technology, the classroom sessions, and there were weekly quizzes and a final exam given on that, and that was done this year.

Q Other than the course you have now referred to, which was given to Midland, and the course that you had referred to as having been given to Metropolitan Edison at the end of 1976 and the beginning of 1977, have you given quizzes or tests in any other courses?

A A final exam is always given for the cold license courses. As far as the quizzes, I just don't know.

MR. GALLEN: To clarify, when you say "you," you mean B&W or Mr. Lind personally?

MR. ROCKWELL: B&W.

Off the record.

(Discussion held off the record, following which a brief recess was taken.)

THE WITNESS: Can I point out one thing before we go any further along, just in terms of these dicussions we have had about evaluations?

Q Yes.

A We evaluate on a constant basis during the week in terms of discussing the response to casualties, better ways that something might have been handled as we go on the simulator, and the supervisors are present at that time because there is a supervisor present with the personnel so there is evaluation as an ongoing thing.

There is nothing in writing formally, but we are constantly in an evaluation situation during the week, and that is all I have to say.

MR. ROCKWELL: Please mark this document as Lind Deposition Exhibit 57.

(Attendance sheet herein marked Lind Deposition Exhibit 57 for identification, this date.)

Q Mr. Lind, showing you what has been marked as Lind Deposition Exhibit 57, can you identify that exhibit as to the type of form it is?
A I can tell you what it is not. It is not a B&W form. It is a form generated internally to the Training Program.
Q Are you familiar with this? Have you ever seen it before?
A Yes, I have.
Q In what context have you seen it?
A Mr. Eytchison showed it to me when he was down here previously and asked me what it was.
Q Before that, had you ever seen it?
A I recall seeing Mr. Heilmeier signing these forms and asking him what they were, but I never really have done very much in terms of looking at it. Mr. Heilmeier is the person who signed it, and he is an instructor, and it appears to be basically an attendance sheet or a lecture signed by an instructor.
Q Do you regularly take attendance of who is present for the courses you give?
A We do not take -- we do not pass out an attendance sheet for each lecture, no.
Q Do you at the end of the course have any

record of who was present and when?

A Yes, we keep track during the week; if anything develops where a person does not attend a lecture, we would know.

Q Do you have any records at the end of the courses as to who attends and when?

A Yes, right here. (Indicating Deposition Exhibit 55.) This schedule shows who was present for the lecture, and who was present here and here during that week and what was given, and we know who attended what and when.

Q You are referring now to Lind Deposition Exhibit 55?

A Yes, that is correct.

Q And you are indicating ath a record of attendance was taken?

A Yes, it is. It is a record of attendance that we have.

Q Isn't this form made up before the course starts?

A Yes. I would like to point something out there, though. If their participation is not complete there, that is noted on the attendance sheet, on this additional sheet which is filled out after the course is finished. (Indicating.) This is another -- this indicates that he has completed a one-week course program consisting of 16 hours of classroom and 20 hours of simulator. If for some reason Mr. Boyer had to leave that course in the middle, this sheet would not reflect 16 hours of classroom and 20 hours of simulator. It would adjust if he missed a portion of this course and bounced against this, and those two things essentially comprise the attendance report.

MR. ROCKWELL: Would the reporter please mark this document as Lind Deposition Exhibit 58.

(Document described below herein marked Lind Deposition Exhibit 58 for identification, this date.)

Q Showing you what has been marked as Lind Deposition Exhibit 58, I would ask you to identify it.

A This is a training summary sheet which is put together by the department for purposes of documentation after a course has been completed.

Q And you are referring to what we have now marked as Deposition Exhibit 58 a moment ago, are you not?

A Yes, that is correct.

Q Am I correct in understanding that this is a form onto which the name of the trainee is placed and the number of exercises performed are also plugged into the form?

A In this form we plug in the student's name, number of classroom hours, and number of simulator hours, evolutions performed, and the number of times evolution is performed.

(Continued on Page 81.)

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Q Doesn't the form include as part of the form the number of hours? There is no blank on the form where you fill in the number of hours, is there?

A Yes, there is. That is an empty space for that, and going back to Exhibit 55, perhaps for some reason those gentlemen indicated they wanted to leave or had to make a plane connection and could not be here for class, and for Rancho Seco, we never gave classes on Friday afternoon, and those spaces are blank and adjusted for the schedule.

I might point out one other thing about this Exhibit 55. This is made up before the course, and any changes made in the course, we will amend this sheet during that, and the one that goes in the jacket should reflect what was done.

Q On Exhibit 58, is there a blank where the number of hours is filled in indicating --

A Yes.

Q There is a blank?

A That is correct.

Q Are you saying there is a line and a blank space?

A Yes.

Q Do you see a line under the number of hours?

A No. Would you like me to get an original form

17.2

which shows where that space is blank? Do you want me to do that now?

MR. GALLEN: Off the record.

(Discussion held off the record.)

Q Do you use any training manuals in any of the courses that you have taught to Met Ed personnel?

A Again referring back to the cold license training programs?

Q Any of the courses that you have given.

A Yes. On some of the courses, training manuals are supplied.

Q Can you tell me what those training manuals are?

A They would be what they call PWR technology course manual, and it would be a three or four volume set of manuals which would describe the B&W scope of supply systems for that utility and might contain some additional information on some other topics which aren't essentially distant, but other topics like soluble poison concentration control or safety analysis.

Q In which courses have you used the PWR technology course manual for the training of Metropolitan Edison personnel?

A In the cold license training program.

Q When was that given?

17.3

A Those were the courses that were given from 1977 back.

Q Do you use any course manuals in hot licensing?

MR. GALLEN: When you say "course manuals," what are you referring to?

MR. ROCKWELL: Training manuals.

A For the hot license replacement operators programs, we normally supply them with a set of procedures, a set of technical specifications and a basic B&W system description manual which is a general description manual.

Q So for the hot licensing, you supply them a set of procedures, and what was the next thing?

A Technical specifications.

Q And the third item?

A Is a B&W systems description manual. It is a very general descriptive manual on B&W systems.

Q With respect to the set of procedures, whose procedures do you supply them with?

A They get a set of ours for their own retention, and the conglomerate set, as we have discussed earlier. There is normally a classroom set available, and then a set that goes on the simulator for use during the simulator exercises.

17.4 Q They are given their own set?

A Normally they get a full set of procedures.

Q When you give them B&W procedures, are those procedues annotated to show where they differ from the procedures that these operators will be using in their home plants?

A The set that we just give them in class for retention is normally not annotated. The set on the simulator that they will use on a daily basis are those annotated sets we just discussed earlier which was a mixture of ours and theirs with annotations on them.

Q Now, correct me if I am wrong, but I thought you told me that you did not give them a set of procedures for them to keep?

A You asked if I give them a set of their procedures. I do not give them a full set of theirs.

Q Do you give them a set of your procedures?

A We give them a set such as this (indicating) of our procedures.

Q You give them a full set of your procedures in the hot licensing program, is that correct?

A Yes.

Q And then you give them a set of technical specifications. Whose technical specifications do you

17.5

give them?

A Their technical specifications.

Q For the plant that they are being trained to operate?

A Yes.

Q Then you give them a B&W systems description which is a manual?

A Yes, it is a manual.

Q And you also say that available to them during the course is a set of procedures, a mixed set of B&W procedures and the procedures that apply to these operators' home plant?

A That is correct.

Q How can they tell which is a B&W procedure and which is a procedure for their own plant?

A They are marked.

Q Where the procedures differ, do you show them both procedures?

A If their procedures apply, we show them their procedure, and if their procedure does not apply, then normally we insert our procedure.

(Continued on following page.)

Q How do they know which procedures are applicable to their own plants and which applicable to B&W? Are they ever told the differences in terms of the procedures they are being trained on and the procedures they will have to use when they go home?

A Well, they are --

Q Are they ever told what those differences are, specifically?

A If we are using a procedure that is ours, we will explain why we are using our procedure instead of theirs.

Q The question is, are they ever told specifically what the differences are between the procedure they are training in and the procedure they will use when they get to their home plants?

A They are told, yes.

Q By whom?

A By the instructor.

Q The instructor goes through the B&W procedure and then shows them the home procedure and compares and tells them?

A If we have a copy of both. Sometimes the procedures we use -- we don't have a copy of their procedures.

Q So it depends on availability of materials?
A Yes, exactly, and what procedure will work.
Q Do you have any idea of how often you have available the procedure from the home plant to compare with the procedure you are using in the training?
A It is hard to put a number on that. I would say that a pretty good amount of the time -- this again depends. As I have said earlier, on some places we have a very extensive set of their procedures, and on some we just carry a more compact version. If we are talking about a utility where we have almost all or all of their procedures, then it is much easier to compare the two. On some of the utilities, we only have the fairly compact set of procedures, in which case we don't have their procedure, and we obviously can't compare it.
Q There is no procedure in the department to see to it that the procedures which you use in the training which differ from the procedures at the home plants have an analog from the home plant available for comparison?
MR. GALLEN: Can you clarify what you mean by "analog."
MR. ROCKWELL: The same procedure for the

same operation, but as applied to the home plant.

A Universally, no.

MR. ROCKWELL: Please mark this as Lind Deposition Exhibit 59.

(Document described below herein marked Lind Deposition Exhibit 59 for identification, this date.)

Q Mr. Lind, showing you what we have marked as Lind Deposition Exhibit 59, do I adequately or correctly identify it as a document which is something on the order of 100 pages with tabs, which is entitled, "Operations Manual for Nuclear Power Simulator," and the front page of this document does not bear a date? Is this a correct description of the document?

A That is correct.

Q Is that a set of the B&W procedures that we have been referring to previously in this deposition?

A Yes, it is.

Q Is it complete? Can you take a moment to review it and see if it is complete.

I would advise you that what we have now marked as Deposition Exhibit 59 was provided to us by B&W.

MR. GALLEN: Off the record.

(Discussion hold off the record.)

Q Mr. Lind, have you now had a chance to review Depo*ition Exhibit 59?

A Yes.

Q Does it appear to be complete to you?

A The procedures appear to be, yes.

Q And those are the procedures that you have been referring toas the B&W procedures for use on the simulator?

A Yes, that is correct.

Q Am I correct in understanding that where a procedure for a particular operating utility does not fit with the software programming of your computer, then you use a B&W procedure?

A Software programming is not correct. It is more a matter of hardware, really.

Q Given that correction?

A Yes, I agree.

Q When was this set of procedures which we have marked as Deposition Exhibit 59 written down?

A Some of the procedures in there were just recently updated; some of them are much older. The latest procedures in there were written in the last few months; some of them are much older than that.

Q When was the last time that procedures in the manual were updated before March 28, 1979?

A They were in the process of being updated before that time. It is not an ongoing process. It is something we do periodically.

Q Who was in charge of that update that was ongoing at the time of March 1979?

A At that time I was essentially working on upgrading the procedures.

Q What procedures did you use for reviewing and upgrading them?

A If you mean did I bounce that against another procedure, no. We used other utilities' procedures as guidelines for working on our procedures.

Q In other words, in revising or updating these procedures, you simply began looking at comparable procedures from other utilities?

A Yes.

Q And on the basis of your review of those, then made some changes in these procedures?

A Yes, or if there were some other obvious problems in other procedures, like bad wording or something that wasn't well put down on the paper, it might have been corrected. In other words, editorial changes and just bad English, bad structure-type things.

Q Did the Engineering Department here at the Nuclear Power Generating Division play a role in the updating of these procedures?

A This recent update?

Q Yes.

A Not on the ones that are in there, no.

Q When you say "not on the ones that are in there," what do you mean?

A There is another procedure being worked on which got some help from the Engineering Department, but not incorporated in the one being used now.

Q That has occurred since March 1979?

A Yes.

Q In terms of the review that was being done on these procedures in March 1979, was any other division or department or section within the Nuclear Power Generating Division involved in terms of reviewing and making recommendations as to what changes ought to be made?

A No.

19.1

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Q Who made the decision that the procedures ought to be reviewed and updated?

A Walt Perks initiated it, while he was the lead instructor made the latest thrust towards revising the procedures and developed and picked out some procedures that he thought needed to be worked on and distributed them to the instructors, and I was essentially carrying on some work prior to my taking over the lead instructor position.

Q Aside from Mr. Elliott, does your Training Department receive instructions or guidance from other departments or other people in management here at the Nuclear Power Generating Division?

A In management?

Q Yes.

A Everybody in our department goes to Norm, Mr. Elliott; there is no bypassing him.

Q Aside from the instruction which he gives to the department as a group, does the department have contact with other departments or other management people here at the company in terms of what its approach to training is, or what its philosophy of training is?

A No.

Q Do you have any informal contacts?

19.2

A You mean, with other managers, management level people, as far as policy?

Q Yes.

A Going directly as relating to training, no. For example, we have, you know, I guess, annual meetings with Mr. MacMillan, but that is a large meeting talking about company policy.

As far as specifically directed to the Training Department other than Mr. Elliott, on the managerial level, no, and no direct contact, and not really any informal contacts as such.

Q To your knowledge, has there ever been any discussion between what was formerly called Nuclear Services and now called Consumer Services and your department in terms of structure and focus of the Training Department?

A To my knowledge, no. That I can remember, no.

Q Does a trainee ever fail any course here?

A Yes.

Q How is that determined?

A We have to talk about what is -- let us take an example, which is much easier.

One of the courses that we give is this replacement operator training course which is a two-week and possibly sometimes a three-week core course that is

19.3 given to people on a plant that is already running, so that one of the steps that they have to do is go through a startup certification examination here, and it is one of the things that they must pass in order to be able to take their examination and get their license, and they are given a startup certification examination as a normal part of the replacement operator training course.

Q Aside from that for the moment, who formulates the startup certification examination?

A The guidelines for the startup certification examination were developed basically by the Nuclear Regulatory Commission which provided us with guidelines into which we had to fit our examination.

Q When did they provide those guidelines?

A Those guidelines have been around awhile. I can't give you a date, and I also know that, and again I can't give you a date, but sometime occurring when I was new in the department, and I am not exactly sure of the time frame, but in the last few years, perhaps, three years, there was an audit of our examinations that were used to help us, essentially, polish up and check that we were giving the examinations according to the guidelines that they imposed and following the spirit of the guidelines and the factual regimentation

19.4 of the guidelines.

Q Did you say when that audit was held?

A We have had an audit since then, but there was some -- specifically some audit done which resulted in slight modifications.

Q When was that?

A It was done within the last few years, but I can't recall an exact date.

Q It was done within the last few years, you say?

A Yes. There have been audits which were satisfactory and didn't require changes. There were a few since then. We were audited by Mr. Buzy from the Regulatory Commission.

Q When?

A Last fall, I believe.

Q Can you spell his name.

A B-u-z-y, but there might be a "c" in there someplace; Joe Buzy.

Q What department of the NRC is he with?

A Inspection and Enforcement.

Q You are saying that he audited the startup certification examination?

A He audited two that were given to operators from Toledo Edison.

19.5

Q Are these written or oral or both?

A Oral and performance; oral examination and naturally, the startup that must be done in the observation of the examiner.

Q Did Mr. Buzy ever give you a written evalu-evaluation of what he had observed while he was here in the fall of 1978?

A When Mr. Buzy was here, Mr. Perks was still here. He talked to Perks. I am not sure of the content or whether any comments were given, and there have not been any audits for our examinations since I have been lead instructor.

(Continued on following page.)

Q You have not seen any written results of his visits in the fall of 1978?

A Nothing that I can recall.

MR. ROCKWELL: Off the record.

(Discussion held off the record.)

Q Do you know whether Mr. Buzy conducted any other audits other than the one conducted in the fall of 1978?

A I mentioned earlier that there was a previous audit conducted.

Q Where some changes were made?

A I believe Mr. Buzy also ordered that examination, and there was an audit of a Metropolitan Edison exam for a cold license course given by two members of the Regulatory Commission, and that took place in the spring of 1976, and so I was just in here then, and I remember the examinations were given, and some people from the Regulatory Commission did come down and audit that cold license course examination that was given.

Q Did Metropolitan Edison administer that examination?

A We administered the examination to the Metropolitan Edison operators, and were audited by the Regulatory Commission.

Q When you use the term "startup operator program," is that the same as "cold licensing"?
A No, I have been using startup certification examination when I am talking about that as being the only exam given, and we are talking in the context of hot license application, hot license training, and the examination for cold license training is for more extensive operation.
Q Do any of the examinations which you give here in the Training Program bear a correlation or bear a relationship to the subject matter of the NRC licensing programs?
A Yes. We give a written examination for cold license certification, and it is an NRC examination. It is put together by drawing on a reserve we have of old Nuclear Regulatory Commission exams, and we take questions on the old exam, put them together and give a new exam, and so we give that -- it is essentially an NRC exam.
Q Are the operators who received that written examination during the cold licensing process then examined by the NRC?
A Yes.
Q They are given a similar exam by the NRC?
A Similar in length and general content, yes.

The examination is made up right prior to the examination by whoever is going to give the exam, made up by the NRC.

Q What is the purpose in drawing your examinations utilizing the old NRC examinations?
A So that th y will follow the general material that is asked on the Regulatory Commission examinations.

Q Your examinations are in part then a preparation for the NRC examination?
A In part, yes.

Q Is the course material geared to some extent toward the kind of questions that in your experience the NRC asks on its licensing exams?
A There is a correlation between the material, but we do not design our courses o : design our course content based on the examination, but they ask important questions, and we try to cover important topics, so there is a correlation between the exams and what we give.

Q Before the examinations which you give based on the NRC examinations, are the students exposed to old NRC exams?
A I can't answer that question. I would think, yes. I am not sure. It is possible that in making

up some of these quizzes we talked about earlier, some of these questions might be gleaned from an exam, some questions pertaining to certain topics taught in the class might be given as a review, and those examinations, the old examinations, are available to the utilities, so they might have done quite a bit of preparatory work with old NRC examinations as part of their training at the utility also.

Q You have revi . quite a few of these old NRC examinations, have you?

A I haven't looked at any in quite a while, but I have in preparation of my own examination.

Q From your experience, are the kinds of subjects and the questions asked in those NRC examinations fairly consistent?

A Yes.

Q Do you know who in the NRC makes up their examinations?

A The examiners make up the exams. That is my understanding, that the examiners make up the examinations.

Q They are administered at the site?

A Yes.

Q And that is both oral and written exams?

A Oral and written, correct.

Q Let me ask you if you have covered certain specific questions in training to your knowledge before March of '79. Do you know whether the question of the relationship of the pressurizer water level to water inventory in core was ever addressed in your training, either in that classroom or simulator training?

A I don't know.

Q To your knowledge was it?

A Based on what I taught, no.

Q Had it ever been brought to your attention before March of 1979 that a transient in September of '77 had occurred at Davis-Besse, which involved the departure of the pressurizer water level indication from the normal relationship that it bears to water inventory in the core?

A I was aware of an accident at Davis-Besse, that there were problems with the pressurizer level. I did not have a tremendous in-depth knowledge like I do with the TMI accident, but I was aware of a problem that it did have.

Q When did you learn about that, do you recall?

A It was fairly soon after the accident occurred. We got the word down there on things like that fairly rapidly, serious things.

21.2

Q Did I understand when you heard about it that there had been some concern about operators prematurely terminating HPI because they were watching the pressurizer water level?

A Yes.

Q Did you have occasion to discuss that in your training programs, either in the classroom or in the simulator?

A Again I can't speak for the whole department in that time frame. I am speaking for myself.

I remember on some occasions, and I can't even really specify, but I remember discussing the accident with operators, what I knew of the accident at that time.

Q What would that discussion have involved?

A Talking about the fact that they had gotten into a saturation condition, that the electromatic had failed to pick it up for what I thought was a fairly long period of time, 20 minutes, the fact that the pressurizer filled up because they lost feedwater.

Q What prompted you to discuss that question? Was that at Mr. Elliott's request or was that at your own initiative?

A I believe it just came up in conversation. We may have been talking about leaks or something like

21.3

that, and it came up.

Q From whom did you learn about the Davis-Besse accident or transient?

A I can't recall. I believe the first information on that was word of mouth that came down from the Service Department or from the site possibly, but it was essentially something that was told to me. I came in one day and it was something like that, that maybe I walked in and we sat around and they said, "Hey, did you hear about what happened at Davis-Besse?"

"What happened?"

"Well, the electromatic failed to open, and there was saturation."

My first information on an accident of that type would probably have been that type of thing, verbal transmission.

Q Are you offering a specific recollection or are you assuming as to how you probably learned?

A My assumption would be something like that.

Q Do you know whether you received more than word of mouth information about it at some time following the initial word?

A There was ultimately a one word writeup in the LER, Licensee Event Report, which was about that long (indicating), very, very short. It was a short

2‡.4

description, which did not get into a lot of detail.

I am familiar with the accident report that has ultimately come out of that with the very detailed description of the accident, with graphs and stuff. I just reviewed that again recently, and it is hard for me to remember how much of that I had seen earlier. I just really don't recall. I did see some written information on it, but it was quite awhile ago.

Q Did you ever see a memorandum written by Bert Dunn in the Engineering Group referring to his concern about premature termination of HPI by operators in the Davis-Besse transient in September '77?

A We are talking in this time frame after the accident happened?

Q In the fall of --

A I don't recall seeing anything like that.

Q Actually, yes, the Dunn memorandum I believe was written in the winter of '78.

A I don't recall ever seeing that memorandum.

Q Do I correctly understand then that you had some word of mouth information about the transient, you probably did see the LER, and did you see anything else?

A Pertaining to the accident?

Q The Davis-Besse transient, yes.

A I don't believe so, no.

2‡.5

Q Do you ever recall discussing that transient with Norm Elliott?

A That is so long ago it is very hard to remember. I talked about that accident subsequently with Norm Elliott, but my contact with Norm was very limited at that time.

I talk with Norm a lot more than I used to when I was just an instructor. I don't remember. I might have talked to Norm or Walt. We might have discussed it with one of the instructors. I just don't remember any of the details.

(Continued on following page.)

Q You say you talked to Elliott about that transient subsequently. You mean since TMI 2?

A We discussed it in light of the TMI accident.

Q What was the substance of your discussion since the TMI accident, your discussion of Davis-Besse?

A Just a comparison basically of the two accidents, how much of it was the same and how much of it was different, nothing in any great detail. It was just general discussions about things. I know I remember Mr. Elliott in one of his testimonies some place, I think it was anticipated he might be asked about that. I didn't know why. We sat and talked about it for a few minutes. He asked me if I had a copy of the thick, full, complete report available, and I had it and gave it to him.

We looked at some of the traces, the pressurizer level responses, general discussion of the accident.

Q Did he have any specific questions to you in his more recent conversations about what information had come into his department back in the fall of '77?

A Did Mr. Elliott have any questions of me?

Q Yes.

A Not that I can recall.

Q Was he saying to you. "Hey, John, do you remember what information we got back in '77"?

A No, not that I recall. We were generally discussing the accident as the accident. He was trying to reconstruct what had happened, basically talk to the charts, so that he felt sure he had, I think, a good handle on it in case he was asked.

Q You say you recall having discussed your understanding of the Davis-Besse transient with some of your trainees?

A Yes. I said that I felt sure that it had come up in conversations with some of the operators that I mentioned, that they had a fairly serious transient at Davis-Besse.

Q Do you remember which utilities these trainees were from?

A No, I can't. That is just too far back and too many students and too many utilities.

Q Was there ever a request from Elliott or from anyone else that you cover the events of the Davis-Besse transient, the September transient of '77, regularly in your classes?

A I don't recall any instruction being given to me or any instruction that I saw.

Q And I take it you did not cover it regularly?

A It did not become a formal topic, such as the TMI transient is now. It was never worked up to that detail.

Q Before TMI, had you ever incorporated particular operating experiences, that is, transients, into the training program by way of an example to illustrate certain points?

A Yes.

Q Can you give me an example?

A Well, like I said, this Davis-Besse thing, I remember discussing to some extent.

Q I mean on a regular basis.

A On a regular basis?

Q Yes.

A Nothing that would have been done with total uniformity to everybody. Other transients I can recall discussing with operators. I know that I was teaching, but I can't recall anything right off the top of my head in terms of classroom discussion, which would have been done on a rigid structured basis.

Q Is there any component of your training which specifically focuses on running the students or the trainees through transients that have occurred in the field to show them actual examples of the kind of problems that come up and how people have responded to them, either correctly or incorrectly?

A We do as a matter of course run certain problems on the machine, on the simulator, as part of our training. Some of them have been incorporated such that they are now a standard evolution, a standard drill that is done for everybody. Others were done where a utility requested it because they had a problem and wanted to look at it again.

There are some accidents that are fairly standard evolutions that we did that were not thought of by us but actually happened some place, and we used them for training.

Q Do you ever use as a training method showing the operators or your trainees how somebody reacted incorrectly, so that they see what the possibilities are for going wrong?

A Can I take an example?

Q Sure.

A Let us take something fairly simple, so it doesn't look too complicated, something that happened, as a matter of fact. It happened to one of our instructors when he worked at a utility.

He had the main feed valve go shut on him at power.

There are a number of ways to handle that correctly. There are a number of ways to handle it incorrectly, almost infinite, depending on what the operator does for his actions for a casualty like that.

Normally what we will do is put it in. If the operator responds correctly, fine. Then they obviously can handle a casualty. If the operator responds incorrectly, then they learn by their own mistakes. We do not make a habit of telling people wrong things to do.

If they make a mistake that they can learn from, we will go back and review what they did incorrectly, but we don't try to emphasize or get into emphasizing incorrect actions as a starting point for a casualty.

Q Have you ever simulated or incorporated in your training a PORV stuck open?

A Yes.

Q And asked your operators to handle that kind of a situation?

A Yes. That is a casualty that the machine is capable of performing, and that is done.

Q Was it capable of performing that kind of casualty before March 28, 1979?

A Yes. You put it in a different way than you do now because we are using it more often. It is now a push button, but we had the capability of putting it in through the teletype before that.

Q To your knowledge, up until March 28, 1979, had the training program ever discussed or addressed the issue of going solid and whether it was advisable or not advisable?

A Operationally, yes, there are a number of ways you can get the plant moving toward a solid condition for certain accidents, and it is discussed and treated as an operational consideration, filling the pressurizer up.

Q Is it discussed and treated in any of your written materials?

A I am not really sure I understand what you mean by "written materials" in that case.

Q Manuals, anything in writing that you use in your training.

A There are documents which specify that there are normal operating conditions, pressurizer levels must be maintained in certain bands, technical specifications, for one, which is a written document. We attempt to cover it quite thoroughly with all the students on different levels, depending on perhaps who is there. That is one document that is used all the time that is always around, very specifically addressing the concept of maximal allowable pressurizer levels. In light of that, yes.

(Continued on Page 112.)

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23.1

Q Do you know offhand what the particular technical specification is that addresses that?

A There is a specification on pressurizer level which calls for a maximum and minimum allowable level with a bubble while the reactor is critical. That level will vary slightly from unit to unit because of the ways the pressurizer level instruments are located, and things like that.

But they do -- the specifications call for utilities specifying a maximum and minimum level under certain modes of operation, which essentially is critical modes of operation.

Q I take it that the students in the training courses are taught that those maximum and minimum pressurizer levels are to be observed at all times, based on the tech spec?

A Based on what the tech spec says, they must be approved, which is not all the time.

Q What does the tech spec say when they are to be observed?

A If we get into the older technical specifications, non-standard, they generally have words that say before you can go critical, the pressurizer must be in this level range, and you must have a bubble in the pressurizer. It is a condition which must be required.

23.2

Q To your knowledge?

A The newer sepcifications will again specify a maximum-minimum level and will describe the time that that condition must be met in terms of modes, which is different conditions the plant can be in, in terms of power level and its margin to criticality.

So, for instance, on the newer specs they would call for modes -- I think it is Modes 1 and 2 I believe -- yes, Modes 1 and 2 would call for pressurizer level to operate and continue to operate in that mode. Pressurizer level must fall in a certain band. Essentially it is a critical consideration.

Q To your knowledge are there any exceptions to the maximum and minimum pressurizer level instructions in the tech spec, while the reactor is critical?

A The only possible place it could be would be under special testing, and I don't believe there is any exception made for pressurizer level on special testing exceptions. These concern slightly different areas.

Q In your training would you have taught the operators anything other than the consideration that pressurizer level should be maintained within the limits contained in the tech spec?

A With the conditions that the plant would have

23.3 been in when we were teaching that, that would be consistent with what we taught, yes. I need to elaborate on that maybe.

Q I'm not sure what you mean.

A If we get into a position where we are worried about the pressurizer level --

Q "Worried" in what sense?

A Well, it is too high or too low. But I don't know -- this may be technical, but I can take an example.

Q Sure.

A A large steam rupture of the system, a steam break, a big one, will very rapidly depressurize the system and actuate safeguards.

The system automatically isolates that leak in a fairly short period of time normally, and as soon as the leak is isolated, there is no longer any need for high pressure injection.

High pressure injection was actuated, and not because there is a steam leak, but because the pressure goes up affecting the safeguard system, and the safeguard system thinks that because the pressure is building up there is probably a leak and starts pushing water into the system.

So we take the example where we have a large

23.4 break. The pressure drops very quickly. The safeguard system is actuated. The HPI has come on full, and there really is no need at all for any HPI because there isn't any loss in the primary. The steam leak is over. The generator has boiled off.

At this point, if the operators didn't terminate HPI, they are going to take the pressurizer solid. There is inventory, and they are putting 1,000 gallons a minute into the system.

In a case like that we tell them to terminate HPI before they take the pressurizer solid.

In this case, of course, if we are telling them that, "Hey, your pressurizer is going full," they will meet the proper condition for terminating HPI. The plant is stablizied. There is no inventory loss. The pressure is increasing and the pressurizer level is increasing.

At that point there is no problem in terminating HPI to keep the plant from going solid.

There really aren't too many times where you can't stop that increase, where you don't have a choice. It is a very limited series of circumstances where you are saying that the pressurizer level is where you don't want to take the plant solid. It is not true, I would say, based on the fact that it is a multiple

23.5 casualty situation, filling up the pressurizer is a last resort for cooling under multiple casualties. It is completely possible that in a very, very long period of time over a course of years you may never get to the point where telling the people to take the plant solid is not the correct statement to make. I can only think of one very limited set of circumstances where not taking it solid or taking it solid would be the correct thing to do.

Q Let me ask you this. In any of your training before March of '79 have you ever suggested to the operators that there was an appropriate time to take the plant solid?

A Probably not, because the only set of circumstances I can think of are compound casualties, which just probably we never got into that specifically. Actually there is almost always an alternative.

Q Is there any particular reason that you were not addressing compound casualties in training?

A We were addressing that. We do address compound casualties. We have addressed compound casualties. We never addressed that compound casualty. There can be an infinite array of compound casualties.

Q It would be a fair statement, I take it then, to say that to the extent that the training

23.6 program addressed the question of the appropriate water level in the pressurizer, that is addressed in the context of the tech spec which gives certain requirements with respect to maximal allowable level and minimal allowable level, is that correct?

A That is our concern for the level is based on the specifications, yes, which limits what you can do in levels. That would be one basis. Right off the top of the operator's head you don't want to go above that. The spec says you can't. It is not a good thing to go above.

There are other considerations which would be discussed. That is not the only concern as far as exceeding that level. It is not just that technical specifications. There are operational considerations.

Q In the sense of damage to the system?

A Possible damage.

Q Pressure spikes?

A Possibility of coming up a code relief and not having re-seat is probably the biggest danger, with filling the plant completely up, then pushing the pressure up to lift the relief up. If that doesn't re-seat, then you have got yourself a problem. You have got a big leak.

Q You have got a small break LOCA?

A If it fails open, you are talking closer to a large break LOCA than a small break LOCA. That is a big hole.

Q The operators in your training would have understood that risk and that concern?

A Yes.

(A brief recess was taken.)

Q Mr. Lind, taking you back for a moment again to the September 1977 Davis-Besse transient, were you ever told that that was a subject of considerable concern with this organization because of the risk of premature termination of HPI and the possibility of core uncovery?

A Nobody ever addressed that concern to me directly. There were some intimations that people in the building were concerned about the fact that Davis-Besse had turned the HPI off.

Q When you say there were intimations in the building, can you be more specific?

A Sometime after the accident, and again I can't give you an exact time frame, Joe Kelly, who works in, I think, Plant Integration, came down and said, "Have you been putting out information down here which would

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

have caused those guys to turn HPI off?"

I kind of went around and talked to different operators about what kind of information we had been putting out. Again, I was an operator or an instructor at that time, not lead instructor, but we talked with Joe about it and basically said, "They turned the HPI off, but they weren't following guidelines we put up because the pressurizer was going down at the time, and we wanted the pressurizer level going up before you turn it off." That was about the extent of my conversation with Joe.

I could tell that they were concerned about the fact that the HPI had gone off, but as far as that flowing out and the core uncovering, nobody ever directly talked about that consideration with me. There was a fairly specific question about "What were you putting out down here to the operators," and did I ever tell them anything that might have caused them to turn the HPI off, and the answer to that was, we asked different instructors, and my answer and everybody else's answer that I heard was "No, we never told them that under the conditions where that is happening, pressurizer going up, pressurizer level going up and pressure is going down, that is not a specifically good reason to turn the high-pressure injection off."

Pressure was not under control. That is really about the accident. You can intimate from that some concern.

Q Again, was there anything in the written materials that you were using in the course at that time which gave that guidance to instructors on termination of HPI, guidance which you told Kelly that you had been giving?

A Our procedures were very specific about termination of HPI on a leak. It called for both pressurizer level and system pressure to be increasing or under control when you terminated HPI.

Q Would that be in the procedures that we have marked as Lind Deposition Exhibit 59?

A Yes, it would.

Q Could you show me where that procedure is.

A Okay, on Page 4 of Procedure 1202.6, Page 4, Step 5.2.5, "If the RC System pressure and pressurizer level stop decreasing or begin to increase upon initiation of high-pressure injection, maintain level as close as possible to normal operating range by varying the number of running makeup pumps."

(There was discussion off the record.)

Q So in answer to my question as to what procedure involved the instructions that you told Kelly you were giving, it is this 5.2.5 of Operating Procedure 1282.6?

A That is correct.

Q Which you just read into the record?

A There is a guideline in that procedure for termination or cutting back on full HPI.

Q And the guideline would be that when the reactor coolant system and the pressurizer level begin to increase, then you would cut back?

A Yes, stop decreasing or begin to increase. That is the conditions which have to be met before you begin to back off from the HPI.

MR. EDGAR: Off the record.

(There was discussion off the record.)

Q In reference to the page from which you have recently quoted, I take it this 5.2.5 is what you were teaching your students in the training course, is that correct?

A We might not teach the specific step because we might never use this procedure, but these guidelines and the fact that you are not supposed to throttle back on high-pressure injection unless pressurizer level and system pressure are under control, stop decreasing or are increasing, is what we do unless pressure and level are under control. If the system is coming down, the pressure is decreasing. Normally the pressurizer level is coming down, and you do not terminate HPI, if HPI is on because of a leak, you do not terminate unless pressurizer level and pressure are under control, stabilized or increasing.

Q Where is "pressure" referenced?

A That is a typo. It is system pressure, because I can refer you back to the first page. We are talking about leaks. We talk about the capability of the high-pressure injection system to maintain system pressure and level.

Q So you are saying that 5.2.5 really is talking about RC system pressure, even though the word "pressure" never appears?

A Yes, that is a typo.

Q Or maybe an editing problem?

A Or an editing problem, yes. We talked earlier about making some editorial corrections in procedures. The concept of pressure and level are addressed in the procedure in the introductory remarks.

MR. EDGAR: What is the caption?

THE WITNESS: That step in the section of the procedure which talks about leaks or ruptures within the capability of the high-pressure injection system to maintain system pressure and pressurizer.

MR. EDGAR: That is the purpose?

THE WITNESS: Yes. This Section 5 discusses, like where you have the ability of controlling these two with high-pressure injection.

MR. GALLEN: What is the title of the operating procedure?

THE WITNESS: Loss of reactor coolant -- reactor coolant system pressure.

Q Do you ever get a specific request from utilities to run certain simulations?

A Yes.

Q How often does that occur?

A I really can't give you a number. We are asked on occasion by the utilities to either duplicate -- to duplicate accidents they have had or to run perhaps for the students, perhaps for an example, let's say -- this may not have happened, but, say, TMI heard about an accident that occurred at Arkansas which they thought might be interesting for their students to see. They might ask us to run the accident.

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25.1

Q Have you actually ever had a specific request from Met Ed?

A I had a very specific request from Met Edison for a transient last year.

Q What was the request?

A Jim Selinger, who is operations supervisor at TMI, called me up and asked me to please run a transient to verify that the response of the plant was essentially correct for what had happened.

They experienced a very, very rapid loss of pressurizer level on a transient, and they wanted to make sure that he hadn't missed anything, that something else happened. He gave me all the initial parameters and asked me to run the transient, and essentially check and see how the simulator responded, in terms of pressure decrease and loss of pressurizer level, compared to the traces at the Island, and we did that.

Q Did he ask you to do this in the context of the training program or just as a check?

A This was a request as a check to get back to him after the casualty had occurred.

Q Have you ever had a request from Met Ed to run a particular simulation in the context of the training program for the purposes of educating the

25.2 operators?

A I can't remember anything specifically. I think that we were requested to run a failure on integrated control system because it had occurred up there, a failure of a specific instrument on the integrated control system.

I know that it did happen to them, but I don't remember if we just ran it and coincidentally it happened there of they had had it and asked us to run it. I don't remember the exact time frame there.

But I remember it was specifically a TMI accident, and we did run it down here for training purposes with students.

Q Is it fair to say that the vast majority of the simulations that are run are simulations selected by you here at B&W during the training program?

A For some utilities, yes, and for some utilities, no.

Q With respect to Met Ed?

A Met Ed, I would say that more of the casualties we run on the simulator are initiated by the staff then by requests from them. They would request a casualty on occasion, but most of the casualties we run we initiate.

Q You referred to two or three. Can you think

25.3 of any more that have been requested specifically by Met Ed to be used during training?

A It is very hard to remember requests specifically for casualties. There could have been more. I can think of these two or three off the top of my head. That is not to say there may not have been others. I can't recall any more.

Q When they make a request, do they send a letter saying, "Please include these in the training program"?

A Nearly all these would be verbal communications.

Q How many students at a time do you put into the simulator?

A Our optimum number that we like to have in there at any given time is three. However, on occasion there will be more students that are in there. Normally for operators we never run more than four, but on occasion we did run as many as six for certain types of courses where it is not that critical for all the people to get their hands on the controls. Three is the optimum number, and on occasion four.

Q At any time one of the three trainees is actually manipulating the controls?

A At least one, unless we are just doing a demo, where they would stand back and watch.

25.4 But on any type of evolution that would require a minimum of one, we have those three, with an absolute minimum of one of them to do what we are requesting. We don't normally do things that that one guy can do all by himself. We try to make it a little more difficult for him.

Q Before March '78 had you run any multiple casualties on the simulator during training?

A Excuse me. March '79?

Q Yes, March '79.

A Yes.

Q Can you tell me the types of multiple casualties you had run?

A You want some specific examples?

Q Yes.

A Okay. We run as a fairly standard accident a leak which gives faulty pressurizer level indication, which is a multiple casualty, a leak and a level instrument. In a sense that is a multiple casualty. We will run casualties which will perhaps get the operator to one place, and then put another casualty in behind it, in which essentially he's just finishing up one casualty and another one takes place.

We quite often defeat runbacks on the integrated control system, so we will initiate a casualty to

25.5 cause the runback signal to be generated and block that control signal.

We will cause an upset condition, so he has to take part of the ICS in hand and then cause another casualty to occur, where he then has to manipulate the controls, rather than let the system do it automatically.

We fail or partially degrade or block partial actuations of safety systems on leaks, which is a multiple casualty because you have a leak, and then one of your high pressure injection pumps doesn't start or something like that.

It is hard sitting here thinking, but there are quite a few multiple casualties. This is something we have been doing for quite a long time ever since I have been here.

Q If Met Edison people had been trained on PORV being stuck open, would you be able to go back to the records and see from the records that this had occurred?

A We can go through the records and find documentation on PORV valve leaks, electromatic relief valve leaks.

Q Where do you go?

A I would go to the summary sheets to find the

25.6 record of these casualties.

Q That is the primary source?

A That is the source.

Q That is the source?

A Yes.

Q How many of the simulations involve actual B&W plant transients from actual experience, what proportion?

A . Things that we do in the simulator that have actually happened at B&W plants?

Q Yes. Do you have an estimate?

A Can I use this as a refresher? (Indicating.)

Q Yes.

A In excess of 50 percent. Now, some of these we have things like steam leaks and reactor coolant system leaks -- those occur, but we don't put in one gallon a minute leaks or very small steam breaks. We would put bigger casualties in.

But as far as reactor trips, turbine trips, load rejection, feed pump trips, control-rod drive malfunctions, instrumentation failures -- almost all of that stuff has occurred in B&W plants to some degree or other.

Q How often do you update the simulations that you use to incorporate recent transients and recent operating experience at B&W plants?

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A This upgrading we just went through is a larger type upgrade. If we find an accident that has taken place which is an interesting casualty that may involve nothing more than just being able to just put it in right away, the machine has a good capability, as far as developing new casualties. Sometimes it requires a small change or the addition of a push botton on the instructor's console. We don't have to go through a lot of upgrading for most casualties, a lot of changing.

(Continued on following page.)

Q Do you have any record of when you plug in new simulations or when you create the capability to do new simulations?

A I don't keep track of that.

Q Does anyone?

A The software packages are upgraded as casualties are put in, but I'm not sure what, essentially. We are working in Mr. Rosser's domain, as far as that goes. I don't know how much of previous capability is retained as a back check.

We have a current, up-to-date statement of the capability of the machine. On what it used to be able to do, I'm not sure how accurately we keep track of that and where we add things on.

Q You don't know, for instance, whether there is a log of changes made in the software package, what those changes are and why they were made?

A There would be. I am sure there is a listing to some extent of what changes have been made, but why and when, I doubt if they are listed.

Q Who would have the summary of what changes were made?

A That type of thing would be Mr. Rosser's concern--the detailed software documentation, keeping track of software changes. I am concerned about the fact of getting them in there and making sure all the instructors are capable of utilizing the things we put in, but I don't keep a record of when things are done.

Q Is there an hourly rate which is charged for use of the simulator?

A The charges for the machine for customers are based on machine time per hour, yes, and hourly rate.

Q What are the hourly rates?

A Can we go off the record?

(There was discussion off the record.)

MR. EDGAR: Is the information really important? We could talk to some people about whatever sensitivity may exist.

Q Can you give us a range for the hourly charges for use of the B&W simulator?

A $500 an hour would be a good ballpark figure for an average rate.

Q Your B&W simulator is basically similar to the SMUD design, is that correct?

A In terms of the control room design, control room layout and core model.

Q During the training of the Met Edison personnel, how do you account to your students for the differences between your simulator and the control room and plant that they use when they are at home?

A I am not sure what you mean by "account."

Q How do you explain that?

A We explain that we are required to basically model some plant, and we model Rancho Seco. If we are modeling Rancho Seco, we can't duplicate TMI.

Q D you explain the details of the differences between your simulator --

A If we have a new set of students, we spend normall between an hour and three hours in an orientation in the c ntrol room before we start doing any exercises. The first day or two the exercises are normally the type of things where they don't have to move fast, which is to do startup or shutdown, so they have a day or two to get acclimated to our control room layout before they have to start moving quickly and find things quickly.

Q Do you explain the differences between your simulator and the TMI control room and plant to the students?

A Yes.

Q The students from Met Ed?

A Yes.

Q Who would make that explanation?

A The instructors.

Q Which instructors?

A The instructor who had them for the orientation the first day.

Q Would that be the instructor who is the liaison with Met Ed?

A Normally, if the scheduling permits, the first instructor that the students come in contact with in the first week is the liaison person.

Q It could also not be?

A It could be another instructor if the schedule doesn't allow.

Q How would the other instructor know what the differences are between TMI and your simulator?

A He would be familiar with the big changes, the big differences, because we all work with all the students. All of us have some familiarity with all of the plants, not in tremendous detail.

Mostly what we do is explain our system. For the people who know more about the plant, they intuitively know the difference because they know their own control room.

Q If one of your instructors had never been at Met Ed, is there any way that instructor would know the difference between the control room layout in Met Ed and the control room layout in the simulator?

A In great detail, no.

Q How many of your instructors have been to TMI 2 before March 28th?

A I am listening to the instructions and trying to think. I believe there are three instructors who have taken a look.

Q Three instructors?

A Yes.

Q And that is in reference to the TMI 2 control room?

A TMI 2 control room.

Q Who would they be?

A Harry Heilmeier, who is liaison; Ted Book, who is a senior licensed on TMI; and I have been in the control room, John Lind.

Q How many have been in the TMI 1 control room?

A I would say probably the same three.

Q Are the students evaluated, other than running feedback, oral feedback, during the simulator sessions?

A No.

Q Is there any measure of their performance in the simulator transmitted back to the utility?

A Not by the Training Center, no.

Q Am I correct that evaluation forms are given to the trainees for evaluation of the training?

A Comment sheets, yes, where they may make remarks on training.

Q Are these supplied by B&W or the utility?

A B&W.

(Continued on Page '35.)

Q Are those returned to B&W? Are they filled out here?

A Yes, they are filled out normally right before they leave on Friday and given to us.

Q Whose responsibility is it to review these forms?

A They are reviewed by all the instructors, myself and Mr. Elliott.

Q Everbody reviews them?

A Yes.

Q Routinely?

A Yes.

Q What procedure is there for evaluating these comments and for adjusting the content or the approach of the training program if it is deemed necessary?

A As I said, at the end of the week we collect all the sheets. They are passed around and routed to each instructor. We look at the comments. If we find worthwhile comments and we feel they are pointing up a problem area or somebody has had a very bad classroom situation, say, with a specific instructor, something like that, we may either upgrade that instructor, in the case where we have had an outside engineer come in and everbody uniformly for six weeks says, "This

27. 2

guy is terrible," then we just won't use that guy any more. That is a simple case.

Some requests, although very good, just can't be augmented. They may ask for us to duplicate their control room or something like that.

If they ask for a course or, for example, they feel -- and one of the suggestions that came out quite awhile ago involved structuring this two-hour and two-hour break, which I believe was initiated by some students who said they thought it would be better if we would restructure the 2 and 2, rather than four hours on the machine. So if we find a worthwhile comment which is feasible, we will implement it. It may be that we won't get a useful comment out of 36 comment sheets.

Q Can you recall the last time that you made a change in the program based on trainee feedback?

A One thing leaps to my mind. We had some not complaints, but some comments on a pressurizer pressure response on transients. This is prior to TMI.

They thought that perhaps our pressure wasn't moving quickly as it should. Myself and Walt Perks, when he was still here, and a couple of other instructors, we talked about it and ran some transients and took a look and expressed our opinions as to

27.3

whether or not we thought the response was a little sluggish.

We had Mr. Ralph Rosser do some work in that area to try to get a little more rapid response on the pressure for a given type change, and things like that.

Q When was that, do you know?

A It was when Walt was here, Mr. Perks, so it was I would think last fall, early fall or last summer. That would be one of the last things I can think of concretely, as far as a good suggestion or good input. This was one of the things where that comment had been coming up so much, that even though we really thought that the pressure response wasn't so bad, so many people were saying that it looked a little sluggish, we tried to do something and take a good look at it.

Q I take it that a trainee doesn't pass or fail simulation aspects of the course; just is exposed to it?

A No. That is true for requalification. As we said, the replacement operator training involves an exam that is pass-fail. The 10 weeks or 8 weeks cold license program involves evaluation that is pass or fail. The startup certification is essentially an oral/ machine examination, oral, and then a physical startup of a simulator. Longer examinations are given

27.4 for the cold license courses. There is a written exam, an oral and an operational aspect, where they go in and actually do startup for this certification, and then do casualties.

Q Do you know what the pass rate of students being tested on the simulator is?

A I don't have any good numbers on cold licenses. As I say, I have had very little experience with that. It is hard to even it out. As far as the replacement operator training, there are times when six will come out and there are times when two or three or four of them will fail. It would average out. Again this is rough. I would say on the average that since I have been really watching that type of thing we probably lose one out of every six or fail one out of every six. I would think that is not a bad number. That is just very rough, just trying to recall back the last few exams I looked at.

Q You were in the simulator on the 28th of March with Floyd?

A The afternoon of the 28th, yes.

Q What was he trying to do?

A What Mr. Floyd and I and Bernie Smith were trying to do was essentially duplicate the initial first few minutes of the transient. Jim and Bernie -- I'm not

27.5

sure exactly what they were trying to do. They were telling me to do different things and watching pressure responses, trying to see if we could generate -- and for the most part we ran quite a number of simulations trying to generate the first few minutes' response, as they understood it, at the plant from conversations they had had with the site. We didn't run any long casualties. We were doing a lot of fast ones, trying to get the first about five or six minutes of the transient.

Q Based on your knowledge of the TMI 2 sequence now, were the simulations that you were trying to run at the time based on accurate information?

A The information was accurate, but it was incomplete.

(Continued on following page.)

Q How was it incomplete?

A We had not specific information on how the high-pressure injection system was manipulated during the first five minutes of the transient, and we didn't know exactly when the emergency feedwater was brought back on, and some of the other things we did were accurate, as it turned out. I don't know if it was guesswork on Floyd's part, or good information that had been transmitted.

In addition, we did some things which had no accuracy at all because we were putting very, very large tube leaks in, and I believe at this point in the game there wasn't a big one, so there were some things that were just guesses, very inaccurate, and some things which were very accurate.

Then there was a lot of incomplete or big holes in the first few minutes of the transient, a combination of quite a few.

MR. EDGAR: Is that holes in information?

THE WITNESS: Holes in information because we said we didn't know.

Q Had you had complete information, could you have duplicated on the 28th the sequence that in fact occurred at TMI 2?

A For a period of time.

Q When would the simulation have fallen apart?

A It would have broke down at about Time 8.

Q Eight minutes?

A Yes.

Q Why would it have broken down at that point?

A Because at that time the voiding in the core begins to play a significant part in the plant's response.

Q Your simulator does not account for voiding in the core or did not at that time?

A No. I can be more specific. In terms of significant parts of the plant's response, they had a very rapid cooldown from Time 8, and the pressurizer level only dropped very shortly on the scale. Without the voiding, the pressurizer level would have dropped about 200 inches. So the action would have started to significantly differ after 8 minutes with the loss.

Q Does the simulator now have the capacity to handle voiding in the core?

A Yes.

Q Tell me what specific changes have been in the simulator since March 28th.

A Again, the actual software work is Mr. Rosser's, okay, but generally speaking from my general knowledge we have included the capability of simulating voiding in the core.

The other thing that has been done is that in the previous model, once the plant reached a solid condition, that was the end of the calculations.

Now we can back down out of a solid condition and re-draw a bubble in the pressurizer and regain the level in the pressurizer. The details of that you would really have to talk to Rosser about. But generally speaking, that is the big things we did.

Q These two things?

A These two things.

Q Anything else?

A The flow degradation that they experienced as a function of pressure and temperature have always been in the simulator. We just kind of beefed them up a little bit.

Q Anything else that you know of?

A As far as new things?

Q Well, changes.

A Okay. Well, the relief valve leak used to be entered on a typewriter. It is now a push button, but not a new casualty. We have put some other casualties in subsequently, but they are not specifically addressed to this area. They are just casualties. We are always putting in casualties.

Q Any other specific changes in the software?

A That is the big ones I can think of now. Again, Mr. Rosser may have done some more subtle things in order to accomplish that, but these are the major things that I can see when I run the simulator.

Q When Mr. Eytchison and Mr. Stern were down on June 21st, a number of simulations were run on the simulator?

A That is correct.

Q One of the simulations that was run was a 1000 to 3000 GPM break in the cold leg, do you recall that?

A Yes.

Q Do you recall what the results of that were?

A I remember the response that occurred. There was a reflood of the pressurizer.

Q It went dry and then refilled?

A The level went off-scale and came back on-scale.

Q Was that an expected result for you?
A Expected, no.
Q Why not?
A Why didn't I expect it? I just didn't expect on a leak that big that the pressure would reflow. It was surprising to me to see that response.
Q Have you looked at that at all since that time?
A I have had Mr. Rosser, Mr. Norm Elliott and I talk to Mr. Eytchison and Mr. Stern about that, to look in and to see really whether that is something that would really happen if it was programmed correct, but a couple of orders of magnitude too big or too rapid, or if it was just something in the simulator that wasn't correct.
Mr. Elliott now talked to Bert Dunn about this, I believe, or Bob Jones, and I really haven't had a chance to talk to Norm and Bert about it. Ralph has looked into it and has been sick and is on vacation now and hasn't done very much about fixing it, but has been informed that we don't think that that is probably quite right. If it is correct, I think it is too fast. He has been looking into it. That is where it stands right now.
Q Have you re-run to see if it --

A If the reflood is repeatable? Yes, it is repeatable. We ran it about three times that evening. I have been informed by Mr. Dunn there are conditions under which the pressurizer will reflood, given a leak that size, but I don't feel that simulation is exactly right, so that we informed Mr. Elliott there are circumstances under which the pressurizer would reflood.

Q Mr. Lind, have you given any statements in connection with your understanding of the circumstances surrounding the accident at Three Mile Island, and by "statement," I mean either a statement which you wrote out yourself or which you gave in an interview form to someone else?

A Beyond?

Q Beyond what we are doing here today.

A Well, I have not written. I have shown this accident to I don't know how many people.

Q I am not talking about that.

A The only other thing I can construe is statements made while I was giving a demonstration. Beyond that, nothing I can remember.

Q You have not been interviewed by the Nuclear Regulatory Commission?

A Interviewed? No, I didn't. I have talked to people from the Commission, as far as the demo.

Q But in terms of their interview, question-and-answer interview, which was taken down in some fashion?

A No.

MR. ROCKWELL: At this time, Mr. Lind, we will recess your deposition, leaving you subject to further recall or recall for additional testimony if it should be required. We don't know that it will be, and we don't have any present plans to recall you.

MR. EDGAR: Before we take off, I want to confer with John.

(Witness conferred with counsel.)

(The deposition adjourned at 1:25 p.m.)

____
John Albert Lind, Jr.

Subscribed and sworn to
before me this_____day
of_______________1979.

____
Notary Public

I N D E X

WITNESS	DIRECT	CROSS
John Albert Lind, Jr.	3	

E X H I B I T S

STATE OF NEW YORK)
) ss.:
UNTY OF NEW YORK)

We, STANLEY RUDBARG and IRWIN H. BENJAMIN, Certified Shorthand Reporters and Notaries Public, and ROBERT ZERKIN, Notary Public, of the State of New York, do hereby certify that the foregoing deposition of BABCOCK & WILCOX by JOHN ALBERT LIND, JR. was taken before us on the 3rd day of July 1979.

The said witness was duly sworn before the commencement of his testimony. The said testimony was taken stenographically by ourselves and then transcribed.

The within transcript is a true record of the said deposition.

We are not related by blood or marriage to any of the said parties nor interested directly or indirectly in the matter in controversy; nor are we in the employ of any of the counsel.

IN WITNESS WHEREOF, we have hereunto set our hands this 21 day of July 1979.

Stanley Rudbarg
STANLEY RUDBARG, CSR.

IRWIN H. BENJAMIN, CSR.

ROBERT ZERKIN

RESUME

of

J.A. Lind, Jr.

Name: John A. Lind, Jr.

Address: 414 West Cadbury Drive
Lynchburg, Virginia

Education: Bachelor of Arts (Mathematics) Boston College - 1969
U. S. Navy Nuclear Power Program
U. S. Navy Electronics Technician A&B Schools

Qualification and Training:

5 years Naval nuclear power plant operation, testing and maintenance
3 years commercial power plant operation, testing and training
Senior Reactor Operator's License - Crystal River #3

Job History: Random House Publishing - 1969 through 1970:
Associate editor mathematics department.

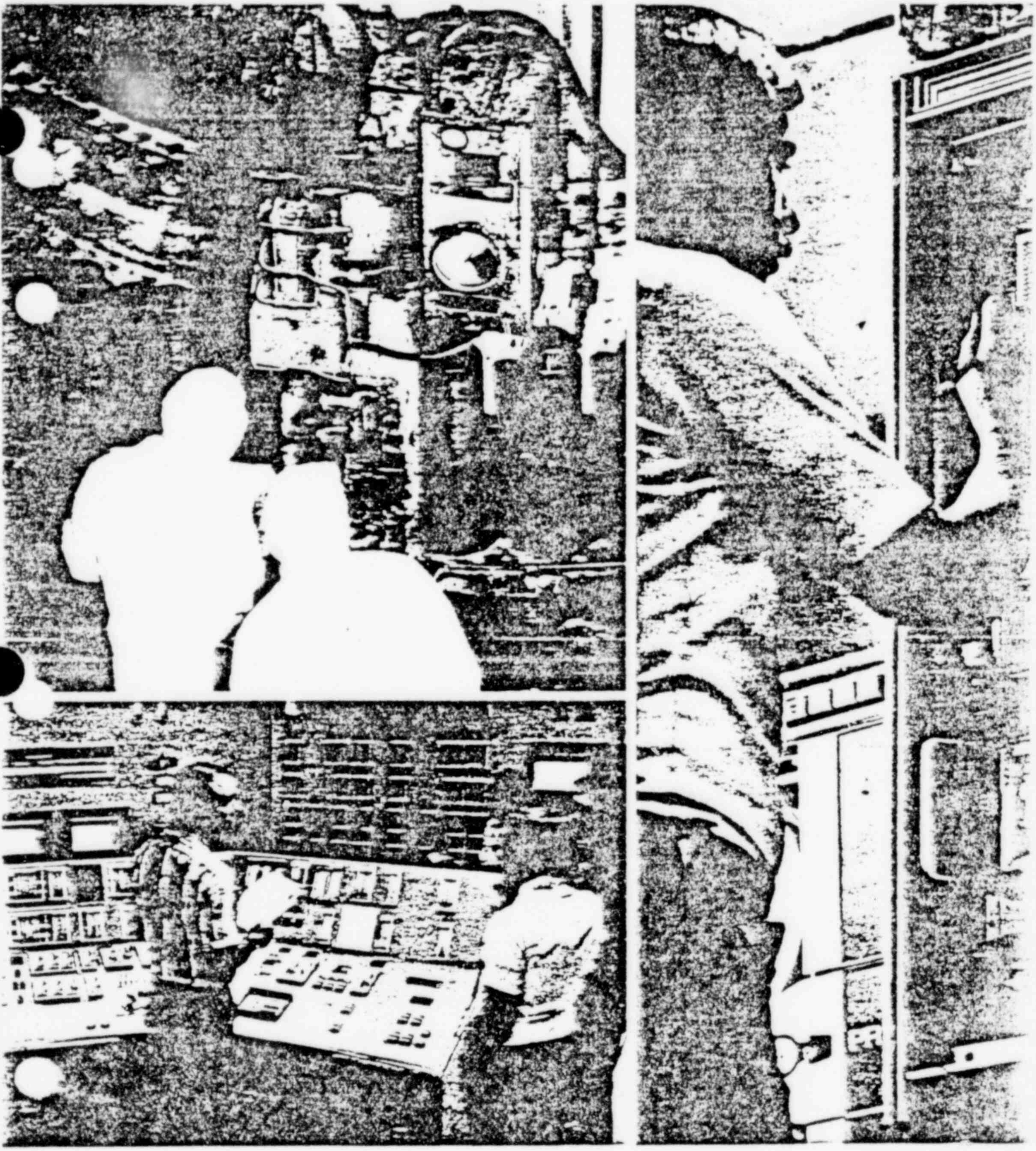
U. S. Navy - 1970 through 1976:
Reactor Operator, Engineering Watch Supervisor U.S.S. Jack(SSN 605) - 4 years.
Special Training - 2 years.

Babcock & Wilcox - 1976 to Present:

March-Sept. 1976 - Associate Instructor, Training Center.
Sept. 1976-June 1977 - Shift Supervisor Augmentor (Master Services Team), Crystal River, Florida
June 1977- Nov. 1978 - Instructor, Training Center
Nov. 1978-Present- Lead Instructor, Training Center - Responsible for scheduling and conduct of operator training courses, maintenance and upgrading of simulator.

LIND DEPO

EXHIBIT 3
FOR IDENTIFICATION
L.H. BENJAMIN



Nuclear Training Ser

EXHIBIT

FOR IDENTIFICATION

Babcock &

Babcock & Wilcox
Nuclear Products and Services

Index

Catalog No.

For more information contact your nearest B&W Sales Office or Manager, Special Products and Service Marketing, Nuclear Power Generation Division, Lynchburg, VA [illegible] or phone (804) [illegible]

Babcock & Wilcox
Nuclear Products and Services

Training Services
T101

Management Seminar/M1

The Management Seminar is designed for personnel who are involved in the management of a utility involved with its first nuclear power plant. The course provides an overview of the nuclear steam system design, startup and operational requirements including a review of major licensing, safety and economic considerations.

A B&W instructor will conduct lecture sessions at the customer's site utilizing material and information specific to the customer's installed system. Utility managers taking this course will be provided with basic information to assist in management decision making during the nuclear plant startup program.

Duration: Two days

Class Size: 30

Location: Customer's site

Typical schedule

Day 1

Introduction

Nuclear plant description

Nuclear power reactor design and operation

Day 2

Nuclear fuel cycle and economics

Liability and public acceptance

Safety and licensing requirements

Operator training and licensing

Startup test program

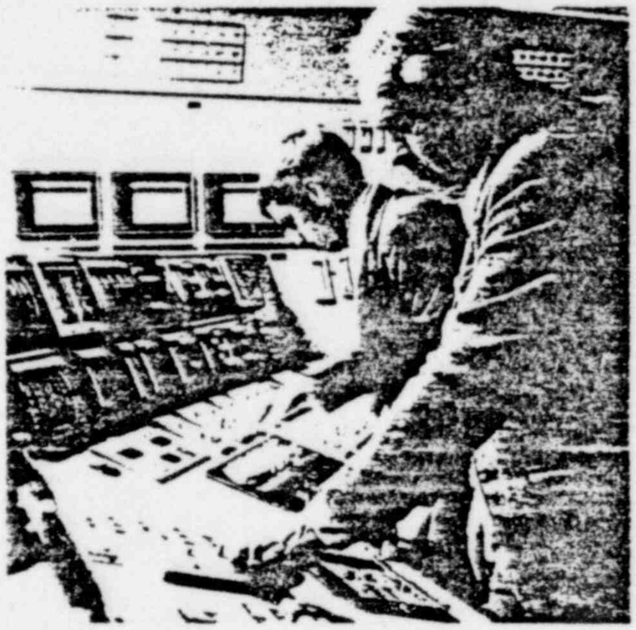
Quality assurance

For more information contact your nearest B&W Sales Office or Manager, Special Products and Service Marketing, Nuclear Power Generation Division, Lynchburg, VA [illegible] or phone [illegible]

FORM T101
JUNE 1977

Basic Nuclear Orientation for Managers /M2

This course provides an understanding and appreciation for the scientific aspects of nuclear power for non-nuclear experienced personnel. Emphasis will be on such subjects as basic nuclear physics, heat transfer, health physics and the safety aspects of nuclear power plants. This training applies to any water moderated reactor and provides a basic understanding of nuclear power for managers who are making the transition from fossil to nuclear power.



Duration: Five Days
Class Size: 30
Location: Customer's site

Day 1 Typical schedule

Introduction
Atomic physics
- Atomic nature of matter
- Nuclear characteristics of elements, fundamental particles and rays
- Radioactivity, decay, and nuclear reactions

Day 2

Reactor physics
- Terminology
- Fission process and chain reactions
- Multiplication factors and criticality

Day 3

Reactor physics (continued)
- Flux and power
- Reactivity coefficients
- Reactor operations, control and safety

Heat transfer and fluid flow
- Thermal properties of bodies, gases and vapors
- Transmission of heat by conduction and convection
- Liquids in motion
- Application to reactor design

Day 4

Introduction to nuclear instrumentation
- Radiation detectors, principles of operation, and typical systems

Shielding
- Functional requirements, materials and types of shielding

Health physics and radiation safety
- Health physics practices
- Biological effects of radiation
- Exposure levels and applicable federal regulations

Day 5

Safety analysis
- Potential plant accidents and hazards
- General public protection and acceptance
- Administrative safeguards

Overview of nuclear power plant operations

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Babcock & Wilcox
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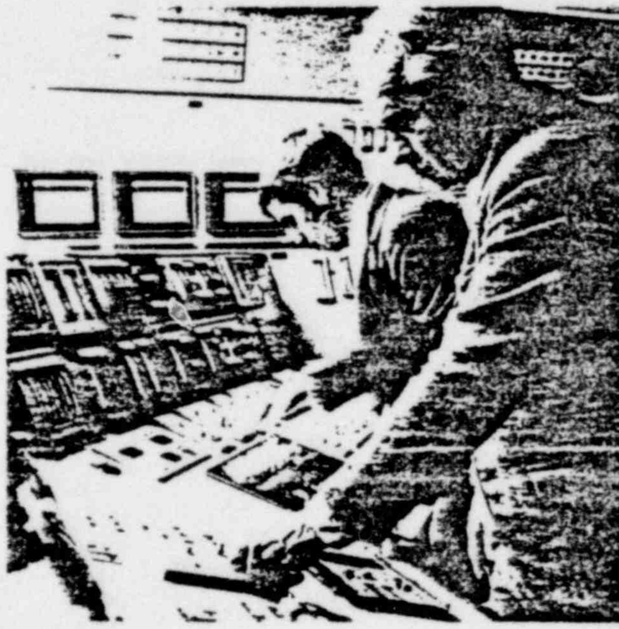
Training Services
T103

Nuclear Power Plant Operations for Management/M3

The purpose of this course is to provide an understanding, and appreciation of the operation of a nuclear power plant to personnel who are involved in utility management.

The course includes both classroom lectures and practical exercises on B&W's PWR simulator. Lecture sessions cover a discussion of control systems, control analysis and safety analysis. Operational exercises and demonstrations on the PWR simulator will be approximately 20 hours. Each student will act as a control operator at the simulator to control the plant in automatic and manual modes for major evolutions.



Duration: Five Days

Class Size: 6

Location: Lynchburg Training Center

Typical schedule

Day 1

Introduction

Basic nuclear plant description

Primary and secondary system components

Control room orientation

Reactor startup and physics demonstration

Day 2

Power plant heatup and startup discussion

Control-rod drive operation
  Safety analysis

Operational exercises
  Reactor criticality demonstration and control-rod drive system operation

Day 3

Control system discussion

Safety analysis

Operational exercise
  Each student brings the plant from hot shutdown to 25% power

Day 4

Control analysis

Power plant shutdown and cooldown

Operational exercise
  Shutdown and cooldown

Day 5

Control Analysis

NSS and steam system failures

Operational exercise
  Demonstration of NSS and steam system failures

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Babcock & Wilcox
Nuclear Products and Services

Training Services
T201

Engineering Staff Orientation/E1

The objective of this course is to provide a comprehensive description of the design and function of major B&W reactor components, equipment and systems. This program is designed as an orientation for the utility's engineering staff during the preconstruction phase. The material presented will promote a better understanding of the Babcock & Wilcox Nuclear System and will assist engineering personnel in the supervision of construction.

The course consists of a detailed review of mechanical equipment, fluid and electrical systems and safety analysis. A description of B&W project support services is also included as well as an introduction to plant operations.

Duration: Sixteen half-day sessions

Class Size: 30

Location: Customer's site

Typical course content

Reactor Vessel and Internals

Reactor vessel, internals, control-rod drive mechanisms, location of fuel assemblies and control rods.

Fuel Assemblies

Description and design parameters.

Control Rod Drives

Description, design considerations and operation.

OTSG and Primary Piping

Unit description, comparison of OTSG with recirculating boiler, arrangements of piping.

Reactor Coolant Pumps

Description, operation and function.

Auxiliary Systems

Engineered safeguards systems, makeup and purification, decay heat removal, spent fuel cooling, chemical addition and sampling, cooling water waste disposal.

Instrumentation and Controls

Integrated control system, principles of radiation detection, nuclear instrumentation, protection systems, self-powered detectors, non-nuclear instrumentation.

Safety Analysis

Sources of radioactivity, protective boundaries between radioactive sources and environment, hazardous conditions, specific abnormalities and accidents, safeguards features, analytical methods for hazards evaluation.

Project Services

B&W support operation, including Project Management, Quality Assurance and Start-up Assistance.

Fuel Handling

Preparation for refueling, refueling, shipment of spent fuel.

Plant Operation

Reactivity control, start-up, power and transient operation, load upsets, reactor trip.

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FORM T201
JUNE 1977

Babcock & Wilcox
Nuclear Products and Services

Training Services
T202

Plant Operations for Engineers/E2

The Plant Operations course provides an overview of the design and function of major NSS equipment and systems plus intensive study and exercises related to power plant operations. The course is given to utility engineers prior to the initial plant start-up. This provides engineering personnel with appropriate knowledge for interfacing with and supporting plant operations personnel.

The course consists of initial classroom instruction covering the B&W Nuclear Steam System as well as balance of plant equipment. Additional background material concerning basic reactor theory will be provided when a review of student backgrounds indicates the necessity for this instruction. The second half of the program consists of instruction and approximately 20 hours of practical exercises on a PWR simulator. Each student, insofar as possible, will operate the simulator for a significant portion of each major evolution as a control operator.

Duration: Ten Days
Class Size: 6
Location: Lynchburg Training Center

Typical schedule

Day 1
Introduction
Reactor coolant system
Reactor vessel and internals
Control rod drive systems

Day 2
Fuel assemblies
Reactor coolant pumps
Once-through steam generator
Non-nuclear instrumentation
Nuclear instrumentation

Day 3
Secondary & auxiliary system
Letdown & purification system
Reactor protection system
Integrated control system

Day 4
Engineered safety features system
Integrated control system

Day 5
Plant start-up
Quiz
Quiz review

(Continued on reverse side)

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Day 6

Introduction

Fluid system review

Reactor start-up/Plant start-up

Practical session
- Control room orientation
- Reactor start-up

Day 7

Classroom instruction
- Review of basic reactor physics
- Reactivity balance calculation

Practical session
- Power operation from 1% shutdown to 100% power

Day 8

Classroom instruction
- Integrated control system review
- Turbine trip
- Reactor trip

Practical session
- Power operation
- Auto & manual integrated control system operations
- Turbine trip
- Reactor trip

Day 9

Classroom instruction
- Steam leaks
- Control rod drive malfunctions

Practical session
- Steam leaks
- Control rod drive malfunctions

Day 10

Classroom instruction
- Engineered safety features

Practical session
- Power operations with selected malfunctions
- Reactor coolant system leaks

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Training Services
T203

Physics Test Orientation for Engineers/E3

The objective of this course is to provide utility engineers with a better understanding of the testing and operational problems associated with the low power physics and power escalation testing. Utility staff engineers and startup engineers gain experience in conducting preoperational physics tests and interpreting the results obtained. This program can be taken as a continuation of the Plant Operations for Engineers course and form a total package for engineers intimately involved in the start-up program.

The course consists of instruction and practical sessions ranging from preparation for initial criticality through plant operations with malfunctions. During the course of the week, B&W experience gained from previous plant start-ups will be made available to the students.

Duration: Five Days

Class Size: 3 to 6

Location: Lynchburg Training Center

Typical schedule

Day 1

Introduction to the physics test program

Related systems and documents

Practical session
- Plant heatup
- Preparation for initial criticality

Day 2

Zero power physics tests

Initial criticality

Temperature coefficients

Practical session
- Initial criticality
- Zero power physics testing

Day 3

Rod calibrations

Ejected and stuck rod

Practical session
- Zero power physics tests
- Ejected and struck rod worth

Day 4

Power escalation test

Standard test at various plateaus

Practical session
- Power escalation
- Reactivity coefficients at power

Day 5

Technical specifications and initial approach to criticality

Practical session
- Plant operation with malfunctions

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Nuclear Products and Services

Training Services
T301

New Plant Operator Training/01.

New plant operator training provides utility personnel with the training necessary to become reactor plant operators. This program has been certified by the Nuclear Regulatory Commission to meet all prerequisites for the "cold" license operators to support an initial plant start-up.

The complete program consists of five different courses, designed as a package to fulfill current NRC "cold" operator license requirements. Included are courses in basic nuclear theory, observation experience, simulator operation, systems design and on-the-job training. The complete program leading to a "cold" licensing of new operating personnel is available, or B&W can provide only those courses necessary to support the utility's operator staff prior to start-up.

Basic Nuclear Theory/01-A

This course provides the basic information necessary to qualify as a nuclear power plant operator. Successful completion provides the prospective operator with a foundation in engineering and physics. This enables him to later pursue the phases of instruction covering operations, construction, technical specifications and plant operations under normal and abnormal conditions. The course is given in two parts — a lecture series and a laboratory portion.

Lectures series

Prospective nuclear power plant operators are provided a basic foundation in nuclear engineering, reactor and nuclear physics.

Duration: Ten weeks
Class Size: 24
Location: Customer's site or Lynchburg, Virginia

Typical course content

Subjects covered include: atomic and reactor physics; heat transfer and flow; introduction to nuclear instrumentation; and health physics, shielding and radiation safety.

Critical facility laboratory

Trainees will operate the B&W pool reactor in at least three different core configurations and will

perform a minimum of ten start-ups each. Additional laboratory experience will be provided in health physics and instrumentation.

Duration: Two weeks
Class Size: 6
Location: Lynchburg Research Center

Typical course content

Experience will cover: control rod calibration; power distribution measurements; increase multiplication with rods and fuel; detection systems; basic health physics; and reactivity effects.

Plant Operation Observation/01-B

This course familiarizes operators, not possessing nuclear plant experience, with an actual facility similar to the one they will be operating. The course is conducted at an operating PWR plant and com-

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bines operation observations with study of the nuclear systems under the supervision of a full-time B&W coordinator. It is designed to partially satisfy Nuclear Regulatory Commission requirements concerning practical experience and to provide the trainee with insights into actual plant operation and procedures.

B&W's full-time coordinator will schedule classes, conduct instruction, validate and sign-off preplanned program check-lists, generate necessary documentation and generally ensure that the trainees become familiar with operations both inside and outside of the control room.

Duration: Eight weeks
Class Size: 6
Location: Operating PWR power plant

Simulator Operations/01-C

The course consists of practical instruction on the full-scale, B&W PWR simulator, related classroom instruction and individual study time. When combined with Plant Operation Observation/01-B. The student will fulfill the practical experience requirement for "cold" license eligibility in accordance with NRC regulations Title 10, Part 55.

The program consists of two weeks of instruction in the classroom, five weeks of PWR simulator and one week of NRC-type written and operational examinations.

Duration: Eight weeks
Class Size: Groups of 3
Location: Lynchburg Training Center

Typical course content

Trainees are instructed in the control room in groups of three, with every student operating the simulator at each of the three operating positions. Emphasis is on operational orientation with the trainee concentrating on learning the basic plant operations, casualty procedures, performing assigned evolutions and handling imposed casualities. Lectures reviewing plant systems from an operational and functional viewpoint are used to supplement simulator operation. Plant operating procedures are presented as a planned sequence coinciding with actual operations on the simulator.

An examination similar to NRC examinations is given by experienced B&W staff members to determine student eligibility for certification of satisfactory program completion.

Nuclear Steam Systems/01-D

This course provides a detailed description of NSS components and the functions of the operator's plant. A series of lectures covers the system and its operation, and allows personnel to study the detailed characteristics of major components, auxiliary and control systems.

Duration: Four weeks
Class Size: 24
Location: Customer's site or Lynchburg Training Center

Typical course content

Trainees are instructed on the design details of the Nuclear Steam Supply System including the primary coolant system and components; once-through steam generator; auxiliary, secondary, fuel handling, instrumentation, and control systems. Safety analysis and basic water chemistry presentations are provided.

On-The-Job Training/01-E

This course provides on-site supervision of operator training during the pre-operational and hot functional testing, in preparation for "cold" license examination. During this period, a resident B&W training coordinator, knowledgeable in design and operation of the nuclear plant, will work on-site to assist the plant superintendent in establishing and conducting an effective pre-license during the final phase of operating staff training. His responsibilities include: scheduling; maintaining training records; conducting evaluations; assisting in development of lesson plans; writing reports; assisting the review and preparation for NRC "cold" license examinations; assisting in developing the plant training program, conducting oral and walk-through examinations.

Duration: Approximately 10 months
Class Size: 24
Location: Customer's site

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Training Services
T302

NSS Orientation for Experienced Engineers and Operators/02

The purpose of this orientation course is to familiarize nuclear power plant personnel with B&W nuclear system design and reactor operations. It is specifically designed for those engineering or operations personnel who have considerable experience with non-B&W power plants and will provide knowledge specific to the B&W reactor.

The course consists of one week of orientation on nuclear system design followed by a two week session on plant operations. The plant operation session includes classroom instruction on systems operation and practical exercises on B&W's simulator.

Duration: Three weeks
Class Size: 3
Location: Lynchburg Training Center

Typical schedule

Week 1

Primary system design
- Reactor vessel and internals
- Control rod drive system
- Pressurizer
- Fuel assemblies
- Reactor coolant pumps
- Once-through steam generator

Instrumentation and control systems

Secondary and auxiliary systems

Fuel plant tour

Plant start-up discussion

Week 2

Classroom instruction
- Fluid system review
- Reactor/plant start-up
- Basic reactor physics review
- Integrated control system
- Plant cooldown

Practical exercises
- Control room orientation
- Reactor start-up
- Manual/automatic Integrated Control System
- Turbine trip, load rejection, reactor trip
- Power operations with malfunctions
- Turbine/reactor shutdown
- Plant cooldown

Week 3

Classroom instruction
- Steam leaks
- Engineered safety features actuation system
- Instrument failure

Practical exercises
- Feedwater and reactor coolant pump trip
- Reactor coolant system rupture
- Unannounced casualties

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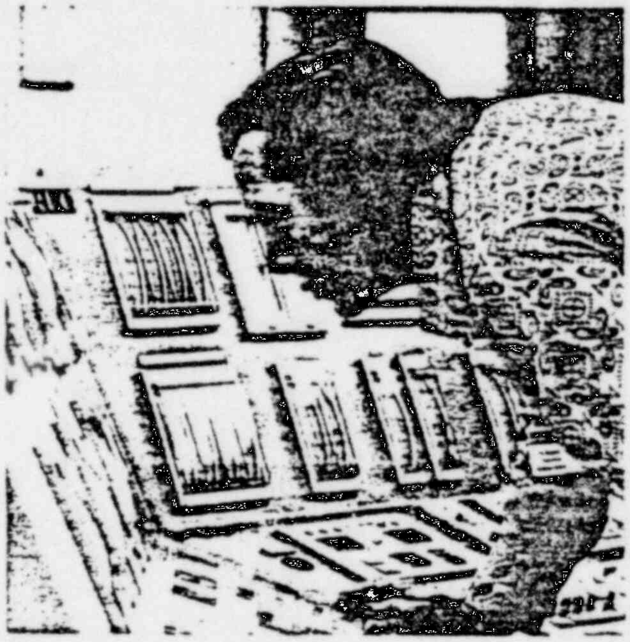
FORM T302
JUNE 1977

Babcock & Wilcox
Nuclear Products and Services

Training Services
T303

Replacement Operator Training/03

During this program students gain a practical understanding of plant operations by operating B&W's PWR Simulator. The course is divided between the classroom (40 hours) and the simulator (40 hours). The course instruction consists of classroom discussion periods followed by practical demonstration exercises on the simulator. The simulator exercises are designed to provide the student with experience in controlling normal and emergency plant evolutions including emphasis on Integrated Control System operation. The course also includes an NRC-type startup examination on the simulator. Certification is made to the sponsoring utility's management for those students who satisfactorily complete the start-up examination.



Duration: Two weeks
Class Size: Groups of 3
Location: Lynchburg Training Center

Typical schedule

Day 1

Introduction
Control panels
Start-up procedures
Reactor criticality

Day 2

Reactivity balance calculations
Plant shutdown (hot shutdown to 25% power)
Plant start-up (hot shutdown to 25% power with turbine-generator in operation)

Day 3

Technical specifications related to start-up.
Power operations and major malfunctions.

Day 4

Review of start-up procedure.
Reactor start-up practice.

Day 5

Start-up examination.

Day 6

Integrated control systems (ICS) operation.
Manual/automatic ICS power operation including turbine and reactor trips.

Day 7

Engineered safety features actuation system review.
Reactor coolant system leaks.
Reactor coolant pump and/or feedwater pump trips.

Day 8

Steam leaks, turbine by-pass valve and feedwater sub-system operation.
Turbine by-pass valve and feedwater sub-system failure.

Day 9

Control rod drive operation.
Control rod drive malfunction and instrument failures.

Day 10

Review and power operation with unannounced casualties.

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Training Services
T304

Simulator Requalification Training/04

The Simulator Requalification Training program provides assistance to the utility's training staff to help meet operator on-the-job requirements for an operator requalification program as defined by NRC Regulation 10CFR55, Appendix A.

The program includes a review of recent abnormal occurrences applicable to B&W plant operations as well as a review of the utility's abnormal and emergency procedures. Instruction consists of 20 hours of practical exercises on the B&W simulator and 20 hours of classroom instruction.

Duration: Five Days

Class Size: Groups of 3

Location: Lynchburg Training Center

Typical schedule

Day 1

Introduction

Review of procedures: reactor start-up and control-rod drive operation.

Control room orientation.

Practical exercise: plant start-up (from all rods in to 20% power).

Day 2

Review of procedures: general power operations.

Integrated control system review.

Control-rod drive malfunctions.

Review of operating curves and limits.

Practical exercise: plant start-up (1% shutdown to 100% power), integrated control system operation in auto and manual, control-rod drive malfunctions.

Day 3

Review of technical specifications.

Evaluation of leak rates.

Practical exercise: reactor coolant system rupture, steam generator tube-leaks.

Day 4

Reveiw of procedures: reactor trip, turbine trip, steam rupture.

Review of selected transients and plant response.

Practical exercise: power operations, manual integrated control system operations, instrumentation failures.

Day 5

Review plant response to selected non-nuclear instrumentation failures.

Review of safety analysis and reactor protective system setpoints.

Practical exercise: power operations, manual integrated control system operations, instrumentation failures.

For more information contact your nearest B&W Sales Office or Manager, Special Products and Services Marketing, Nuclear Power Generation Division, Lynchburg, VA [illegible]

FORM T304
JUNE 197[illegible]

Babcock & Wilcox
Nuclear Products and Services

Training Services
T401

Nuclear Plant Maintenance / T1

These courses provide instruction for plant maintenance supervisors, foremen and technicians in those phases of mechanical maintenance peculiar to nuclear power stations. Instruction provides the on-site maintenance staff with the knowledge and experience needed to improve the utility's maintenance capability.

Nuclear plant maintenance for supervisors/T1-1

This course provides instruction for maintenance supervisors and mechanical foremen in those phases of maintenance dealing with: planning and supervision of plant outages, steam generator tube plugging, pressurizer heater replacement, welding, pump seal replacement, control rod drive mechanical maintenance, radiation control, quality assurance, tools, shielding, and leak detection.

Duration: One week

Class Size: 12

Location: Lynchburg Training Center

Typical schedule

Day 1

Introduction

Planning requirements for routine and emergency maintenance.

Controlling the working environment (non-radiation related).

Controlling the working environment (radiation related).

Quality assurance

Day 2

Description of normal maintenance routines.

Preparation for refueling: head and plenum removal, handling equipment maintenance, head stud and flange maintenance, and incore detector removal.

During fuel handling (canal filled): inservice inspection, control rod drive maintenance, and refueling schedule and equipment.

Day 3

Description of normal maintenance routines (continued)

Following fuel handling (canal drained): pump seal replacement, canal decontamination, head and plenum replacement, incore detector replacement, and pressurizer relief valve maintenance.

Description of non-routine maintenance

Once-Through Steam Generator: chemical cleaning, orifice adjustment, tube plugging, tube inspection, and instrumentation replacement.

Reactor vessel and internals: vent valve replacement

Primary piping: instrumentation replacement and insulation maintenance.

Day 4

Description of non-routine maintenance (continued)

Pressurizer: heater bundle replacement, spray valve replacement, and relief valve maintenance

Primary pumps: wear ring replacement, rotating element replacement, and seal maintenance.

Description of emergency maintenance

Steam generator: feedwater nozzle replacement, seal surface repair, and stuck stud removal

Reactor vessel and internals: incore piping rupture, clad repair, stuck stud removal, and O-ring groove repair

Day 5

Description of emergency maintenance (continued)

Primary piping: weld repair and instrumentation replacement.

Pressurizer: weld and cladding repair

Primary pumps: removal, replacement, and weld repair

Maintenance seminar with B&W site start-up personnel

(Continued on reverse side)

F[illegible] more information contact your nearest B&W Sales Office or Manager [illegible] Products and Serv[illegible] Marketing [illegible] Power Generation Division [illegible] Lynchburg [illegible]

Nuclear plant maintenance training/T1-2

This course familiarizes site maintenance personnel with the plant primary system components on which maintenance will be required, the special tools and fixtures that will be used during maintenance operations and precautions that must be observed. To be most effective, the course should be presented at the plant site just prior to the start of hot functional testing. At this time, all fixtures and tooling should be on hand, with all equipment in place and the construction activities near completion.

The training sessions will consist of classroom lectures utilizing visual aids and appropriate handouts to familiarize workmen with components, fixtures, tooling, and procedures. Following the lecture, the class will proceed to the reactor building or another appropriate area where actual equipment, tooling and procedures are demonstrated.

Duration: Six iays
Class Size: 12
Location: Customer's site

Typical schedule

Day 1

Introduction

Description of overall primary system, precautions to be observed, and overview of common maintenance items.

Tour of reactor building with emphasis on location of components which require maintenance.

Day 2

Introduction to the Once-Through Steam Generators

Description, precautions, orifice plate adjustments, tube maintenance, secondary side cleaning, and feedwater nozzle replacement

Introduction to the pressurizer

Description, precautions, heater bundle replacement, electromagnetic and code safety valve replacement

Introduction to the core flood system

Check valve, flood isolation valve, and flood tank maintenance

Work area tour

D iy 3

Introduction to primary pumps and motors

Description and maintenance procedures

Introduction to control rod drives

Description and maintenance procedures

Work area tour

Day 4

Refueling operation

Description of operation, handling equipment and fixtures

Plant and fuel storage area tour

Day 5

Refueling operation (continued)

Head removal and installation, plenum removal and installation, control rod drive coupling and uncoupling, seal plate installation and removal, and return to operation.

Review

Day 6

Reactor coolant pump seminar

A presentation and supervised discussion on reactor coolant pump maintenance problems conducted by field service engineers

Primary plant pump maintenance seminar

A presentation and supervised discussion on primary plant pump maintenance problems conducted by field service engineers, with emphasis on high-pressure, multi-stage centrifugal pumps

Nuclear plant maintenance consultation/T1-3

The purpose of this course is to familiarize on-site maintenance personnel with maintenance procedures by on-the-job training as the opportunities are presented.

During the construction phase of a nuclear plant, the maintenance personnel can be effectively trained by taking advantage of numerous construction evolutions. This program provides an experienced instructor to supervise the training of on-site maintenance personnel so that they understand the evolution in progress. In addition, this instructor can provide advice and consultation to assist in the supervision of initial staff-performed functions to ensure that they are done properly.

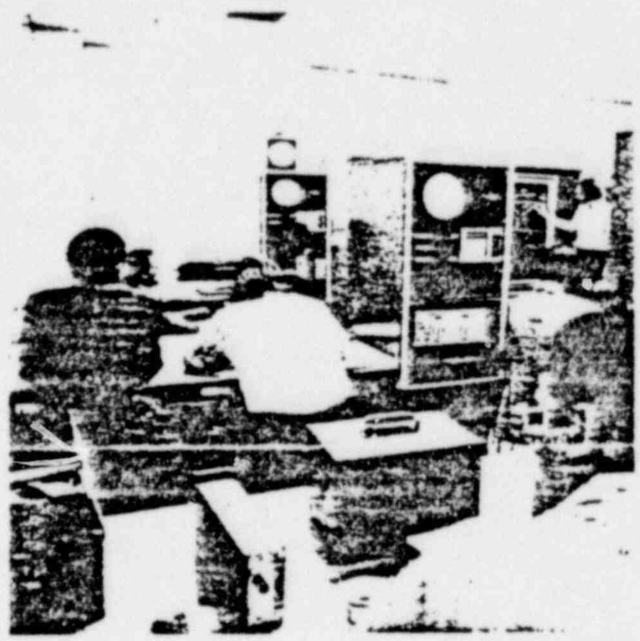
Duration: Length varies
Location: Customer's site

Babcock & Wilcox
Nuclear Products and Services

Training Services
T402

Instrumentation, Control and Computer Systems/T2

These programs provide fundamental knowledge and maintenance capability for utility staff personnel. Instrument supervisors, instrument engineers and other technical personnel achieve in-house maintenance proficiency in the measurement and control equipment used in Bailey Meter Company's integrated control, nuclear instrumentation, reactor protective and plant computer. Throughout the year, B&W's Bailey Meter Company conducts specialized training programs for utility instrumentation, control and computer technicians at their Wickliffe, Ohio facilities. In addition, B&W can provide "in-plant" training programs that allow utility staff technicians to be available for routine duties while receiving the necessary training on your equipment. The courses requires that students in the programs have a basic knowledge of electronics and standard test equipment.



Typical courses available

Analog electronic control circuits and functions
Nuclear Instrumentation
Reactor Protection System
Non-Nulcear Instrumentation
Plant Computer

For more information contact your nearest B&W Sales Office or Manager Special Products and Service Marketing, Nuclear Power Generation Division Lynchburg, VA. 24505 or phone (804) 384-5111

FORM T402
JUNE 1977

Babcock & Wilcox
Nuclear Products and Services

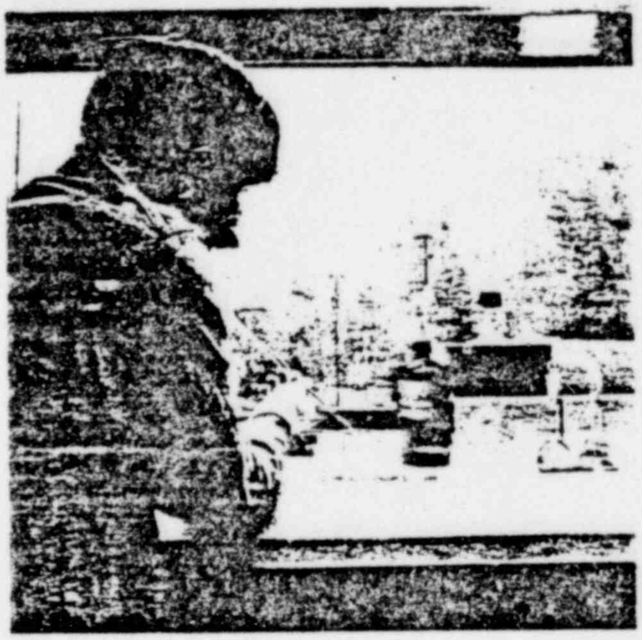
Training Services
T403

Water Chemistry/T3

B&W's water chemistry courses are designed to aid chemistry technicians and supervisors in achieving both "cold" water chemistry and radiochemistry analysis proficiency. Two independent courses are offered in each field of interest.

The first course is for technicians that actually perform these analyses. This laboratory course is normally taught at the customer's plant, using their own equipment and procedures. The technical course is approximately 90% laboratory work. If the customer's facilities are not available, then arrangements can be made to use B&W research center facilities.

The second course is for utility plant supervisory personnel who have a B.S. degree or equivalent experience, but do not have appreciable PWR experience in water chemistry or radiochemistry. Emphasis is on water chemistry theory and interpretation of data as it applies to the operation of a nuclear plant. The supervisory courses are approximately 50% lecture and 50% laboratory practical work. The course will normally be taught at B&W research center facilities.



"Cold" water chemistry for technicians/T3-1
"Cold" water chemistry for supervisors/T3-2

Duration: Ten days each

Class Size: 10

Location: Customer's site or Alliance, Ohio

Typical course content

Overview of reactor coolant and steam cycle chemistry considerations.

Primary and auxiliary system specifications, analytical procedures and sampling techniques: hydrogen, total gas, boric acid, pH, dissolved oxygen, conductivity, chloride, fluoride, Li, NaOH and $Na_2S_2O_3$.

Cooling water systems specifications, analytical procedures and sampling techniques.

Secondary systems specifications, analytical procedures and sampling techniques: NH_3, N_2H_4, Fe, Cu, Pb, SiO_2, H_3BO_3.

Radiochemistry for technicians/T3-3
Radiochemistry for supervisors/T3-4

Duration: Ten days each

Class Size: 10

Location: Customer's site or Lynchburg, Virginia

Typical course content

Gamma Spectroscopy
Gamma calibration
Calibration of alpha and beta
Gross alpha, gross beta
Liquid scintillation and tritium
Iodine separation
Coolant analysis and E-bar
Gas sampling and analysis
Strontium separation
Crud sampling and analysis

For more information contact your nearest B&W Sales Office or Manager, Special Products and Service Marketing, Nuclear Power Generation Division, Lynchburg, VA 24505 or phone (804) 384-5111.

Babcock & Wilcox
Nuclear Products and Services

Training Services
T501

NSS Videotape Seminars.

A series of video tape modules presents a basic overview of a nuclear steam system. Each of the modules is a complete mini-course on a particular subject and lasts approximately 15-20 minutes. The modules are progressively sequenced to instruct various levels of utility staff personnel from secretaries to engineers, and assumes no prior nuclear experience.

Specifications

½ inch, EIAJ-type 1 standard

¾ inch, UMATIC Cassette

Video tape modules

Introduction to NSS
A general review of the way nuclear PWR systems provide electricity.

B&W systems
Basic description of B&W's 205 fuel assembly NSS.

Nuclear physics 1, 2, 3
Atomic theory, nuclear fission and fusion, cross sections, delayed neutrons.

Reactor physics 1, 2
Start up, reactivity changes, xenon, long-term changes.

Reactor vessel and internals
A basic description of the B&W reactor vessel and internals.

Primary system and OTSG
A basic description of the B&W OTSG, reactor coolant pumps, piping, and pressurizer.

Auxiliary systems 1
Makeup and purification, chemical addition and boron recovery, decay heat removal, ESFAS, chemical sampling.

Auxiliary systems 2
Reactor building spray, reactor building cooling, spent fuel, cooling water, waste disposal, waste retention systems.

Nuclear fuel handling
Transportation, scheduling, and plant handling.

Instrumentation and control 1, 2
Basic types and functions of nuclear, non-nuclear, control rod drive, protective and control instrumentation.

Plant operation
Normal heat up, criticality, power and transient operation, reactor shutdown.

For more information contact your nearest B&W Sales Office or Manager, Special Products and Service Marketing, Nuclear Power Generation Division, Lynchburg, VA 24505 or phone (804) 384-5111.

FORM T501
JUNE 1977

Babcock & Wilcox
Nuclear Products and Services

Training Services
T502

NSS Transient Response Videotapes.

The transient response of a nuclear steam system is one of the most difficult training subjects to present to reactor operators. Because of the importance of this knowledge, the Nuclear Regulatory Commission requested that Babcock & Wilcox prepare video tape training material for use in training NRC Operator License Examiners. The video tapes were filmed during the operation of the B&W PWR Simulator. B&W retained the commercial distribution rights and is making these tapes available on a commercial basis.

The series of video tapes describe in full color the total power plant response to various transients. The video tape of each transient has a fully annotated description of each event. Descriptions are provided on plant responses to standard load changes in both automatic and manual control. Abnormal conditions, such as loss of reactor coolant pumps, loss of feed pumps and steam generator tube leaks are also covered. The level of instruction is ideal for power plant operations personnel and is most appropriate for plant staff and support engineering personnel. A fully illustrated text has been prepared to support the video tapes and will enable self-instruction.

Each set of tapes consist of approximately ninety minutes of training material on five individual reels. Twenty copies of the applicable text material is also furnished. The five individual reels are as follows:

- Integrated Control System (part one)

  Increasing Power 30% to 60% in automatic control. Includes an intermediate level presentation of the integrated control system interactions.

- Integrated Control System (part two)

  Four power level changes in manual control including load increase with Steam Generator/Reactor in manual, load decrease with Feedwater Loop Demand in manual, load increase with Reactor Demand in manual and load decrease with Turbine in manual.

- Loss of One Feedwater Pump

- Loss of All Reactor Coolant Pumps

- Steam Generator Tube Leak

For more information contact your nearest B&W Sales Office [illegible] Products [illegible] Marketing, Nuclear Power Generation [illegible]

FORM T502
JANUARY 1978

These video tapes are available in three different formats that are compatible with commercially available color video tape equipment. Formats must be specified with each order and are available as follows:

Specifications

1/2 inch. ElAj - type 1 standard
3/4 inch, UMATIC Cassette
1/2 inch Batamax Cassette

The price for a complete set of video tapes is $6850 plus $2750 for each additional set. Although these tapes were developed specifically for 177 fuel assembly (FA) units, almost all of the information is directly applicable to the 145 FA and 205 FA units. Any differences will be documented in the associated text material.

Please contact your local B&W sales representative or your designated Service Manager should you require additional information.

Babcock & Wilcox
Nuclear Products and Services

Training Services
T503

Integrated Control System Videotapes.

This videotape training program is a series of full color videotapes which describe the Integrated Control System. The approach taken in developing this series was based on the assumption that the student would have a basic knowledge of (1) the mechanical components of the nuclear power plant, (2) heat transfer and fluid flow and (3) the nuclear steam supply system.

A fully illustrated text accompanies the videotape to form a complete training package. The material is presented in a format which will enable self-instruction. Included are questions at the end of each segment which will test the student's understanding of the material presented.

The approximately 100 minutes of video presentation time is divided among five tapes. Each tape, in conjunction with the applicable text material, is a mini-course on a particular section of the Integrated Control System. The five individual tapes are as follows:

- Introduction to the ICS
- Unit Load Demand Subsystem
- Integrated Master Subsystem
- Feedwater Control Subsystem
- Reactor Control Subsystem

The videotapes are available in three different formats which are compatible with commercially available color videotape equipment. Formats must be specified with each order and are available as follows:

Specifications

- ½ inch, E1AJ-Type 1 standard
- ¾ inch, UMATIC Cassette
- ½ inch, Betamax Cassette

The price for a complete set of videotapes is $9,850. Each additional set is $3,750.

For more information contact your nearest B&W Sales Office or Manager, Special Products and Service Marketing, Nuclear Power Generation Division, Lynchburg, VA 245[illegible] or phone [illegible]

Lic Def EXHIBIT 55
FOR IDENTIFICATION
7/3/79 R. ZERKIN

INSTRUCTORS, NUCLEAR TRAINING CENTER

H.J. (Harry) Hollender
J.A. (John) Elst
J.F. (Jim) Carson
A.W. (Al) White

SIMULATOR REQUALIFICATION TRAINING

GROUP 1

Mike Ross
Bob Parnell
Bob Boyer

CLASS ROOM SCHEDULE / CONTROL ROOM SCHEDULE

Day No	Day/Date	Time	Subject	Reference	Time	Operation	Reference Instructor
1	MONDAY 1/9/78	12:00 to 2:00	CRD REVIEW/DIAMOND PANEL REVIEW	J.L.	7:30 to 11:30	REACTOR STARTUP FROM ALL RODS IN TO 100% POWER REACTOR TRIP TURBINE TRIP	H.H.
		2:00 to 4:00	SAFETY ANALYSIS	Scott LaBeau			
2	TUESDAY 1/10/78	9:30 to 11:30	CONTROL ANALYSIS-ADVANTAGES OF PUSH-PULL PLT	Brian Delano	7:30 to 11:30	LOAD REJECTION FEED PUMP TRIP STEAM LEAK OUTSIDE R.B.	H.H.
		2:00 to 4:00	ICS REVIEW	A.W.	12:00 to 2:00	CONTROL ROD DRIVE MALFUNCTIONS STEAM LEAK INSIDE R.B.	H.H.
3	WEDNESDAY 1/11/78	9:30 to 11:30	OTSG REVIEW	H.H.	7:30 to 9:30	REACTOR COOLANT PUMP TRIP OTSG LEVEL FAILURES-FEED PUMP ΔP FAILURES	A.W.
		2:00 to 4:00	ICS REVIEW	J.L.	12:00 to 2:00	Th INSTRUMENT FAILURE RC SYSTEM RUPTURE	A.W.
4	THURSDAY 1/12/78	9:30 to 11:30	CONTROL ANALYSIS-ADVANTAGES OF PUSH-PULL PLT	Brian Delano	7:30 to 9:30	Tc INSTRUMENT FAILURE FEED VALVE FAILURES	H.H.
		2:00 to 4:00	HEATUP/COOLDOWN CURVES	J.L.	12:00 to 2:00	OTSG TUBE RUPTURE PRESSURIZER SPRAY FAILURE	H.H.
5	FRIDAY 1/13/78				8:00 to 12:00	UNANNOUNCED CASUALTIES	A.W.

BABCOCK & WILCOX
NUCLEAR TRAINING CENTER
LYNCHBURG, VIRGINIA

[illegible]

METROPOLITAN EDISON CO.

Enclosure

METROPOLITAN EDISON COMPANY Subsidiary of General Public Utilities Corporation

Subject UNIT I SIMULATOR REQUAL TRAINING

Location TMI Nuclear Station Middletown, Pa.

Date December 15, 1978

To ALL LICENSED PERSONNEL

Unit I RO and SRO licensed personnel and trainees will attend the B&W simulator in Lynchburg, Virginia for training per the following schedule:

Week of 1/2/79

B. G. Smith
T. H. Acker
R. A. Heilman
J. C. Herman
J. R. Floyd

Week of 1/8/79

K. P. Bryan
R. L. Parnell
C. L. Guthrie
T. H. Goodlavage
D. E. Smith
N. D. Brown
Len Garner

LIND DEP EXHIBIT 56
FOR IDENTIFICATION
7/3/79 R. ZERKIN

Week of 1/15/79

W. H. Zewe
D. L. Pilsitz
D. R. Deiter
J. L. Masters
S. W. Brantley
D. B. Mayhue
M. J. Ross

Week of 1/22/79

J. J. Chwastyk
T. L. Crouse
J. C. Banks
V. C. Ruppert
R. S. Hutchinson
J. L. Seelinger

Week of 1/27/79

B. A. Mehler
L. G. Noll
R. E. Boyer
J. E. Keisch
M. P. Kendig
C. E. Husted
M. L. Beers

Week of 2/5/79

G. R. Hitz
D. C. Janes
P. F. Chalecki
D. L. Wooddell
D. A. Smith
D. J. Boltz
R. F. VanStry

Arrangements are being made for hotel accomodations and rental cars. I will forward information to you when these arrangements have been confirmed. You should each see Roxanne Taylor regarding cash advance checks as soon as possible. The advance amount should include enough to cover motel, meals, and fuel for the rental car.

The first week of simulator training will be Tuesday, January 2 through Saturday, January 6, 1979. All other weeks of training will be Monday through Friday. The hours for this series of training weeks is 0730 - 1600.

METROPOLITAN EDISON COMPANY

UNIT II

MARCH 12-16

Brian A. Mehler
Charles D. Adams (Chuck)
Joseph R. Congdon (Joe)
Marvin V. Cooper (Marty)
Donald A. Berry (Don)
Eric W. Orwig

MARCH 19-23

Gregory R. Hitz (Greg)
Adam W. Miller
Dennis I. Olson (Denny)
Lynn O. Wright
W.J. Marshall (Bubba)
George A. Kunder
Mark Cam[illegible]

MARCH 26-30

Bernie G. Smith
Raymond R. Booher
Hal W. Hartman
John J. Blessing
James R. Floyd (Jim)
Kenneth R. Hoyt (Dick)

completed this this week

APRIL 2-6

Kenneth P. Bryan (Ken)
Carl L. Guthrie
Hugh A. McGovern
Earl D. Hemmila
Len P. Germer
Joseph B. Logan (Joe)

APRIL 9-13

William H. Zewe (Bill)
Fred J. Schiemann
Edward R. Frederick (Ed)
Craig C. Faust
James L. Seelinger (Jim)
Jack D. Lawton

APRIL 16-20

Joseph J. Chwastyk (Joe)
William [illegible]. Conaway (Bill)
Ted F. Illjes
John M. Kidwell
Charles F. Mell (Chuck)
Rich S. Hutchison

SIMULATOR TRAINING SUMMARY SHEET

March 26-28, 1979

Mr. Jim Floyd has completed three days of a one week training program consisting of 12 hours of classroom time and 12 hours of simulator operations. The time spent in the simulator consisted of performing the evolutions listed below with the remainder of the time devoted to manual and automatic ICS power operations.

Evolutions Performed	No. Performed
Dropped Rod	1
RC Pump Trip	1
Leaking Pressurizer Relief Valve	1
Reactor Coolant Leak Inside Containment Building	1
OTSG Tube Rupture	1
Steam Leak Outside Containment Building	1
Turbine Trip	2
Feedwater Pump Trip	2
Load Rejection	2
Reactor Startup (Hot shutdown, all rods in, to 10^{-8} amps)	1
Reactor Startup (Hot shutdown, safety rods out, to 10^{-8} amps)	1
Reactor Startup (10^{-8} amps to 5% power)	1
Reactor Startup (5% power to 15% power)	1
Power Escalation (15% power to 100% power)	1
$\geq$10% Reactor Power Change with Reactor Control in Manual	7
Loss of Feedwater	2
Loss of feed with pressurizer relief failed open	2
Loss of feed (main and aux.)	2

[illegible]

Instructor, Nuclear Training Center

[illegible]

Date

3/12-16

SIMULATOR EVALUATION

1. What part of the simulator training was most helpful in improving your knowledge of the plant?

SIMULATOR TRAINING. ICS HAND & AUTO OPS.

2. Would you recommend any changes to the simulator schedule? (i.e. sequence of operations, additions or deletions of any evolutions, more or less time, etc.)

Schedule tour of B&W FUEL ASSEMBLY plant IN BEGINNING OF WEEK FOR THOSE WHO HAVEN'T BEEN THERE.

3. Were the instructors helpful? Were things explained clearly?

YES. EXCELLENT INSTRUCTORS ~~HERE~~.

4. How would you rate the quality of instruction? How could it be improved? TERRIFIC

5. Did the quality of simulation seem real? Do you have any suggestions for improving the quality of simulation?

THE simulator CANNOT simulate the BROAD CONTROL PROBLEMS DURING REAL CASUALTIES at the plant. i.e. Telephones RINGING, CAN'T FIND A O's,

6. Did you get what you expected from the course? Please explain why or why not. YES.

7. Other comments -

THE COFFEE STINKS. I like COFFEE BUT THE STUFF HERE IS REAL CRAP.

SIMULATOR TRAINING SUMMARY SHEET

March 19-23, 1979

Mr. Adam Miller has completed a one week training program consisting of 18 hours of classroom time and 20 hours of simulator operations. The time spent in the simulator consisted of performing the evolutions listed below with the remainder of the time devoted to manual and automatic ICS power operations.

Evolutions Performed	No. Performed
Fail Makeup Pump	1
Defeat Neutron Error Signal from ICS	2
Motor Fault	1
Dropped Rod	2
Stuck Rod	1
Retarded Rod Motion	1
Fail Power Range NI	1
Reactor Trip	1
RC Pump Trip	3
Fail Selected TH Instrument	3
Fail Selected Tc Instrument	2
Fail Pressurizer Spray Valve	2
Reactor Coolant Leak Inside Containment Building	1
OTSG Tube Rupture	1
Fail Turbine Bypass Valve	1
Steam Leak Outside Containment Building	1
Steam Leak Inside Containment Building	1
Fail Header Pressure Signal to ICS	1
Turbine Trip	2
Fail Feedwater Pump ΔP High	1
Fail OTSG Startup Level to ICS	1
Fail OTSG Operate Level to ICS	1
Fail Feedwater Flow Signal to ICS	1
Feedwater Pump Trip	2
Degrade Feedwater Pump Shaft Speed Demand Signal	1
Load Rejection	2
Reactor Startup (Hot shutdown, all rods in, to 10^{-8} amps)	1
Reactor Startup (10^{-8} amps to 5% power)	1
Reactor Startup (5% power to 15% power)	1
Power Escalation (15% power to 100% power)	1
>10% Reactor Power Change with Reactor Control in Manual	7
Feedwater Pump Trip Without Runback	1
Failed Tave to 570 in ICS	1
Fail Intermediate Range NI	1
Leaking Pressurizer Relief Valve	1
Loss of Condenser Vacuum	1
Condensate Pump Trip	1
Fail Steam Generator Pressure to ICS	1
Failed TFW to ICS	2
Failed RC Flow A Loop	1
Failed A Main FW Valve Closed	1
Failed B Main FW Valve Open	1
Failed Power Range Signal to 62.5%	1

Instructor, Nuclear Training Center

March 23, 1979

Date

INSTRUCTORS, NUCLEAR TRAINING CENTER

H.J. (Harry) Hellmeier
T.B. (Ted) Book
V.R. (Vince) Roppel
J.A. (John) Lind
A.V. (Val) White

SIMULATOR REQUALIFICATION TRAINING

GROUP 1

Adam Miller
Lynn Wright
Denny Olson
Mark Coleman

CLASS ROOM SCHEDULE / CONTROL ROOM SCHEDULE

Day No.	Day/Date	Time	Subject	Reference Instructor	Time	Operation	Reference Instructor
1	MONDAY 3/19/79	0730 to 0930	CRD REVIEW/OPERATIONAL PROBLEMS	T.B.	1200 to 1600	REACTOR STARTUP FROM ALL RODS IN TO 100% POWER REACTOR TRIP TURBINE TRIP	H.H.
		0930 to 1130	INTEGRATED CONTROL SYSTEM	H.H.			
2	TUESDAY 3/20/79	0730 to 0930	CONTROL ROD DRIVE	T.B.	1200 to 1400	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	H.H.
		0930 to 1130	POWER DIST. & ROD WITHDRAWAL LIMITS	V.R.	1400 to 1600	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	J.L.
3	WEDNESDAY 3/21/79	0730 to 0930	ICS REVIEW/TRANSIENT RESPONSE	H.H.	1200 to 1400	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	H.H.
		0930 to 1130	HEAT TRANSFER	T.B.	1400 to 1600	PLANT OPERATIONS WITH UNANNOUNCED CASUALTIES	J.L.
4	THURSDAY 3/22/79	0730 to 0930	ICS REVIEW/TRANSIENT RESPONSE	H.H.	1200 to 1400	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	J.?
		0930 to 1130	OTSG REVIEW/TUBE RUPTURE	BRIAN DELANO	1400 to 1600	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	J.L.
5	FRIDAY 3/23/79	0730 to 0930	HEATUP/COOLDOWN CURVES	J.L.	1200 to 1400	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	A.W.
					1400 to 1600	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	T.B.

SIMULATOR TRAINING SUMMARY SHEET

March 12-16, 1979

Mr. Brian Mehler has completed a one week training program consisting of 18 hours of classroom time and 20 hours of simulator operations. The time spent in the simulator consisted of performing the evolutions listed below with the remainder of the time devoted to manual and automatic ICS power operations.

Evolutions Performed	No. Performed
Fail Makeup Pump	1
Fail Pressurizer Level Control Valve	1
Defeat Neutron Error Signal from ICS	1
Dropped Rod	3
Retarded Rod Motion	1
Fail Power Range NI	1
Reactor Trip	4
Reactor Trip Without Turbine Trip	2
RC Pump Trip	2
Fail Selected TH Instrument	3
Fail Selected Tc Instrument	4
Fail Pressurizer Spray Valve	2
Fail Pressurizer Level Signal	2
Reactor Coolant Leak Inside Containment Building	1
OTSG Tube Rupture	1
Fail Turbine Bypass Valve	1
Steam Leak Outside Containment Building	1
Steam Leak Inside Containment Building	1
Fail Header Pressure Signal to ICS	2
Turbine Trip	2
Fail Feedwater Pump ΔP	4
Fail OTSG Startup Level to ICS	2
Fail OTSG Operate Level to ICS	2
Fail Feedwater Flow Signal to ICS	1
Feedwater Pump Trip	1
Reactor Startup (Hot shutdown, all rods in, to 10^{-8} amps)	1
Reactor Startup (10^{-8} amps to 5% power)	1
Reactor Startup (5% power to 15% power)	1
Power Escalation (15% power to 100% power)	1
Plant Reduction (100% power to 15% power)	1
Plant Shutdown (15% power to hot shutdown)	1
$\geq$10% Reactor Power Change With Reactor Control in Manual	7
Plant Temperature Change $\geq 50^\circ F$	2
Failed Feedwater Temp to $0^\circ F$	1
Failed "B" Main FW Valve Closed	1
Feed Pump Trip without Runback	1
Failed Main Feedwater Valve Open	1
Failed OTSG Pressure Low	1
Failed Tave to 570	1
Reactor Trip - Safety Relief Valve Stuck Open	1

[illegible signature]
Instructor, Nuclear Training Center

March 30 1979
Date

INS...CTORS, NUCLEAR ..NING CENTER

H.J. (Harry) Hellmeier
J.A. (John) Lind
A.V. (Al) White
T.L. (Ted) Book
V.R. (Vince) Roppel

SIMULATOR REQUALIFICATION TRAINING

GROUP 2

..ian Mehler
Eric Orwig

CLASS ROOM SCHEDULE / CONTROL ROOM SCHEDULE

Day No.	Day/Date	Time	Subject	Reference Instructor	Time	Operation	Reference Instructo
1	MONDAY 3/12/79	0730 to 0930	CRD REVIEW/OPERATIONAL PROBLEMS	J.L.	1200 to 1600	REACTOR STARTUP FROM ALL RODS IN TO 100% POWER	A.W.
		0930 to 1130	INTEGRATED CONTROL SYSTEM	A.W.		REACTOR TRIP TURBINE TRIP	
2	TUESDAY 3/13/79	0730 to 0930	CONTROL ROD DRIVE	J.L.	0930 to 1130	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	T.B.
		1200 to 1400	HEAT TRANSFER	T.B.	1400 to 1600	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	J.C.
3	WEDNESDAY 3/14/79	0730 to 0930	POWER DIST/ROD WITHDRAWAL LIMITS	V.R.	0930 to 1130	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	H.H.
		1200 to 1400	ICS REVIEW/TRANSIENT RESPONSE	A.W.	1400 to 1600	PLANT OPERATIONS WITH UNANNOUNCED CASUALTIES	J.L.
4	THURSDAY 3/15/79	0930 to 1130	ICS REVIEW/TRANSIENT RESPONSE	A.W.	0730 to 0930	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	T.L.
		1400 to 1600	OTSG REVIEW/TUBE RUPTURE	H.H.	1200 to 1400	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	T.B.
5	FRIDAY 3/16/79	0730 to 0930	REACTOR COOLANT PUMPS	H.H.	0930 to 1130	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	A.W.
		1200 to 1400			1400 to 1600	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	A.W.

BABCOCK & WILCOX
NUCLEAR TRAINING CENTER
LYNCHBURG, VIRGINIA

A1090F27

METROPOLITAN EDISON

SIMU' TOR TRAINING SUMMARY SHEE

March 12-16, 1979

Mr. Chuck Adams has completed a one week training program consisting of 18 hours of classroom time and 20 hours of simulator operations. The time spent in the simulator consisted of performing the evolutions listed below with the remainder of the time devoted to manual and automatic ICS power operations.

Evolutions Performed	No. Performed
Fail Makeup Pump	1
Defeat Neutron Error Signal from ICS	1
Dropped Rod	2
Stuck Rod	2
Retarded Rod Motion	2
Fail Power Range NI	1
Reactor Trip	3
RC Pump Trip	2
Fail Selected TH Instrument	2
Fail Selected Tc Instrument	1
Fail RC Flow Signal to ICS	1
Fail Pressurizer Spray Valve	4
Fail Pressurizer Level Signal	1
Reactor Coolant Leak Inside Containment Building	2
OTSG Tube Rupture	1
Fail Tu bine Bypass Valve	1
Steam Leak Outside Containment Building	2
Steam Leak Inside Containment Building	1
Fail Header Pressure Signal to ICS	3
Turbine Trip	1
Fail OTSG Startup Level to ICS	2
Fail OTSG Operate Level to ICS	2
Loss of Condenser Vacuum	1
Fail Feedwater Flow Signal to ICS	2
Feedwater Pump Trip	3
Reactor Startup (Hot shutdown, all rods in, to 10^{-8} amps)	1
Reactor Startup (10^{-8} amps to 5% power)	1
Reactor Startup (5% power to 15% power)	1
Power Escalation (15% power to 100% power)	2
Plant Reduction (100% power to 15% power)	1
Plant Shutdown (15% power to hot shutdown)	1
≥10% Reactor Power Change with Reactor Control in Manual	9
Plant Temperature Change ≥50°F	3
Dropped Group 5	1
Failed Tave to 570°F	3
Feedwater Pump Trip Without a Runback	1
Failed A Main Feedwater Valve Shut	1
Failed A Main Feedwater Valve Open	1
Failed Feedwater Temp 0°F	1
Failed B Header Pressure to 1200 psi	1
Reactor Trip - Relief valve Secondary Side Stuck Open	1
Failed Feedwater Valve Open (Main Control)	1

Harry Hilmers [signature] — Instructor, Nuclear Training Center

March 20, 1979 — Date

...CTORS, NUCLEAR ...NING CENTER

H.J. (Harry) Hellmeler
J.A. (John) Lind
V.R. (Vince) Roppel
T.B. (Ted) Book

SIMULATOR REQUALIFICATION TRAINING

GROUP 1

Joe Congdon
Chuck Adams
Marty Cooper

CLASS ROOM SCHEDULE / CONTROL ROOM SCHEDULE

Day No.	Day/Date	Time	Subject	Reference Instructor	Time	Operation	Reference Instructor
1	MONDAY 3/12/79	1200 to 1400	CRD REVIEW/OPERATIONAL PROBLEMS	J.L.	0730 to 1130	REACTOR STARTUP FROM ALL RODS IN TO 100% POWER; REACTOR TRIP; TURBINE TRIP	V.R.
		1400 to 1600	INTEGRATED CONTROL SYSTEM	J.C.			
2	TUESDAY 3/13/79	0930 to 1130	CONTROL ROD DRIVE	J.L.	0730 to 0930	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	V.R.
		1400 to 1600	HEAT TRANSFER	T.B.	1200 to 1400	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	V.R.
3	WEDNESDAY 3/14/79	0930 to 1130	ICS REVIEW/TRANSIENT RESPONSE	J.C.	0730 to 0930	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	H.H.
		1400 to 1600	POWER DIST. & ROD WITHDRAWAL LIMITS	V.R.	1200 to 1400	PLANT OPERATIONS WITH UNANNOUNCED CASUALTIES	H.H.
4	THURSDAY 3/15/79	0730 to 0930	OTSG REVIEW/TUBE RUPTURE	H.H.	0930 to 1130	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	[illegible]
		1200 to 1400	ICS REVIEW/TRANSIENT RESPONSE	J.C.	1400 to 1600	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	J.C.
5	FRIDAY 3/16/79	0930 to 1130	HEATUP/COOLDOWN CURVES	J.L.	0730 to 0930	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	T.B.
		1400 to 1600			1200 to 1400	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	T.B.

METROPOLITAN EDISON

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NUCLEAR TRAINING CENTER
LYNCHBURG, VIRGINIA

AT09UT27

[illegible] NUCLEAR [illegible] NING CENTER

H.J. (Harry) Hellmeier
J.A. (John) Lind
V.R. (Vince) Ruppel
T.B. (Ted) Book

SIMULATOR REQUALIFICATION TRAINING

GROUP 1

Joe Congdon
Chuck Adams
Harty Cooper

CLASS ROOM SCHEDULE / CONTROL ROOM SCHEDULE

Day No.	Day/Date	Time	Subject	Reference Instructor	Time	Operation	Reference Instructor
1	MONDAY 3/12/79	1200 to 1400	CRD REVIEW/OPERATIONAL PROBLEMS	J.L.	0730 to 1130	REACTOR STARTUP FROM ALL RODS IN TO 100% POWER	V.R.
		1400 to 1600	INTEGRATED CONTROL SYSTEM	J.C.		REACTOR TRIP TURBINE TRIP	
2	TUESDAY 3/13/79	0930 to 1130	CONTROL ROD DRIVE	J.L.	0730 to 0930	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	V.R.
		1400 to 1600	HEAT TRANSFER	T.B.	1200 to 1400	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	V.R.
3	WEDNESDAY 3/14/79	0930 to 1130	ICS REVIEW/TRANSIENT RESPONSE	J.C.	0730 to 0930	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	H.H.
		1400 to 1600	POWER DIST. & ROD WITHDRAWAL LIMITS	V.R.	1200 to 1400	PLANT OPERATIONS WITH UNANNOUNCED CASUALTIES	H.H.
4	THURSDAY 3/15/79	0730 to 0930	OTSG REVIEW/TUBE RUPTURE	H.H.	0930 to 1130	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	J.I
		1200 to 1400	ICS REVIEW/TRANSIENT RESPONSE	J.C.	1400 to 1600	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	J.C.
5	FRIDAY 3/16/79	0930 to 1130	HEATUP/COOLDOWN CURVES	J.L.	0730 to 0930	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	T.B.
		1400 to 1600			1200 to 1400	POWER OPERATIONS WITH UNANNOUNCED CASUALTIES	T.B.

METROPOLITAN EDISON
Customer

BABCOCK & WILCOX
NUCLEAR TRAINING CENTER
LYNCHBURG, VIRGINIA

AT090T27

Floyd

SIMULATOR EVALUATION

1. What part of the simulator training was most helpful in improving your knowledge of the plant?

2. Would you recommend any changes to the simulator schedule? (i.e. sequence of operations, additions or deletions of any evolutions, more or less time, etc.)

3. Were the instructors helpful? Were things explained clearly?

John Land was great on the CRD's
as was Brian on Control Analysis

4. How would you rate the quality of instruction? How could it be improved?

5. Did the quality of simulation seem real? Do you have any suggestions for improving the quality of simulation?

6. Did you get what you expected from the course? Please explain why or why not.

7. Other comments -

F[illegible]

SIMULATOR EVALUATION

1. What part of the simulator training was most helpful in improving your knowledge of the plant?

Plant evos.

2. Would you recommend any changes to the simulator schedule? (i.e. sequence of operations, additions or deletions of any evolutions, more or less time, etc.)

No - More plant startups

3. Were the instructors helpful? Were things explained clearly?

Yes Yes

4. How would you rate the quality of instruction? How could it be improved?

excelent

5. Did the quality of simulation seem real? Do you have any suggestions for improving the quality of simulation?

?

6. Did you get what you expected from the course? Please explain why or why not.

Yes

7. Other comments -

Lumb Dep EXHIBIT 57
FOR IDENTIFICATION
7/3/79

... PROGRAM ADMINISTRATIVE FORM

2,2,A

1. LESSON/COURSE: SAFETY ANALYSIS

2. LOCATION: Old Forest Road, Lynchburg, Va.

3. PERSONNEL ATTENDING: (PLEASE PRINT)

NAME	EMPLOYEE NO.	NAME	EMPLOYEE NO.
R.E. Boyer	0		0
M.D. Brown	0		0
K.P. Bryan	0		0
P.F. Chalecki	0		0
J.R. Floyd	0		0
R.L. Parnell	0		0
M.J. Ross	0		0
	0		0
	0		0
	0		0
	0		0
	0		0
	0		0

COMPLETION DATE

MO	DAY	YR
1	9	78

CATALOG NUMBER

FILE NO.				LESSON ID
	CAT	TYPE	SUBJECT	

TOTAL COURSE DURATION

HOURS: 2

4. LECTURE HOURS ________ OTHER HOURS ________ EXPLAIN ________

5. ENTER THE PREFIX FOR THE APPROPRIATE MODE:

MODE PREFIX

	PREFIX		PREFIX
CORRESPONDENCE	(COR)	REVIEWER	(REV)
FILM	(FLM)	SCHOOL	(SCH)
LECTURE	(LEC)	SEMINAR	(SEM)
OJT	(OJT)	VIDEOTAPE	(VTP)
ORIENTATOR	(ORN)		

6. METHOD OF EVALUATION (Check at least one)

A. WRITTEN TEST ____________ (Attach copy of answers, and graded test)

B. ORAL QUESTIONING SPOT CHECK ____________

C. QUIZ ____________ (Attach questions and answers)

D. OTHER ____________ (Explain)

7. INSTRUCTOR/TRAINEE'S EVALUATION (This section must be filled in)
(Based on your method of evaluation briefly describe the effectiveness of the training)

____________ INSTRUCTOR/TRAINEE'S SIGNATURE

____________ DATE

8. SIGNATURE OF SUPERVISOR OF TRAINING/TRAINING COORDINATOR ____________ DATE ____________

ATTACH LESSON OUTLINE/ITEMS COVERED

Lind Dep EXHIBIT 58
FOR IDENTIFICATION
7/3/79 R. ZERKIN

SIMULATOR TRAINING SUMMARY SHEET
January 9 - January 13, 1978

Mr. Bob Bover has completed a one-week training program consisting of 16 hours of classroom time and 20 hours of simulator operations.

The time spent in the simulator consisted of performing the evolutions listed below with the remainder of the time devoted to manual and automatic ICS power operations.

Evolutions Performed	No. Performed
Fail Makeup Pump	1
Defeat Neutron Error Signal From ICS	1
Motor Fault	2
Dropped Rod	1
Retarded Rod Motion	1
Reactor Trip	1
RC Pump Trip	3
Fail Selected TH Instrument	2
Fail Selected Tc Instrument	3
Fail Pressurizer Spray Valve	2
Reactor Coolant Leak Inside Containment Building	1
OTSG Tube Rupture	2
Steam Leak Outside Containment Building	2
Steam Leak Inside Containment Building	2
Fail Feedwater Pump ΔP	2
Fail OTSG Startup Level to ICS	1
Fail OTSG Operate Level to ICS	1
Fail Feedwater Flow Signal to ICS	1
Feedwater Pump Trip	3
Degrade Feedwater Pump Shaft Speed Demand Signal	1
Load Rejection	2
Reactor Startup (Hot shutdown, all rods in, to 10^{-8} amps)	1
Reactor Startup (10^{-8} amps to 5% power)	1
Reactor Startup (5% power to 15% power)	1
Power Escalation (15% power to 100% power)	1
Dropped Group 1 Control Rods	3
Slow Vacuum Leak	1
Failed Gen MW to ICS to 500	1
Failed Turbine Header Pressure to 1200#	1
Failed Tave to 570 in ICS	2
Failed Main Feed Valve Full Open	1
Failed Main Feed Valve Closed	2
Reactor Trip Without Turbine Trip	1

[signature]
Instructor, Nuclear Training Center

[illegible]
Date

THE BABCOCK AND WILCOX COMPANY

NUCLEAR TRAINING CENTER

LYNCHBURG, VIRGINIA

OPERATIONS MANUAL

FOR

NUCLEAR POWER PLANT SIMULATOR

Lind Dep EXHIBIT 50
FOR IDENTIFICATION
7/3/79 RZ
L.H. BENJAMIN

7906250.4

This manual has been written specifically for operations of the B&W Nuclear Power Plant Simulator, hereafter identified as Old Forest Road, (OFR-Unit #1).

FOREWORD

This manual contains operating procedures for the Babcock and Wilcox Nuclear Power Plant Simulator. The manual was prepared by the B&W Nuclear Training Center staff for the purpose of training utility operators and engineers during operator training at Lynchburg, Virginia. Procedures contained herein are not to be used for any purpose other than that of training.

INDEX TO OPERATING PROCEDURES

(Short Course)

1102 02

OFR UNIT #1

PLANT HEATUP

OP 1102 02

1. PURPOSE

1.1 To provide the major steps for operations to take the plant from cold shutdown to hot shutdown conditions.

2. REFERENCES

2.1 SMUD Rancho Seco Unit 1 FSAR , Section 15, Technical Specification.

2.2 SMUD Rancho Seco Limits and Precautions

2.3 Chemistry and Radiochemistry Manual Standard 177 Plants

3. LIMITS AND PRECAUTIONS

*(TS) 1. SG secondary side will not be pressurized above 200 psig if the steam generator shell temperature is below 100 F.

(TS) 2. Do not reduce boron concentration in the RC system unless at least one RC pump or one DH pump is circulating reactor coolant. Further, if deboration is required while on decay heat, insure that the calculation uses only the reactor vessel - decay heat system volume.

(TS) 3. One steam generator shall be capable of performing its heat transfer function whenever the RC temperature is above 280°F.

(TS) 4. The reactor shall not be heated above 280F unless the following conditions are met:

a. Capability to supply feedwater to one steam generator at a rate corresponding to a decay heat load of 5% rated reactor power. One of the following are operable:

(1) A condensate pump and main feedwater pump.

(2) An auxiliary feedwater pump.

b. Two or more safety valves are operable/OTSG.

c. The turbine bypass system shall have a capacity equivalent to 5% rated reactor power.

d. A minimum of 250,000 gallons of water shall be available in the condensate storage tank.

(TS) 5. Containment integrity shall be maintained whenever all three of the following conditions exist:

a. RC pressure is 300 psig or greater.

b. RC temperature is 200F or greater.

c. Nuclear fuel is in the core.

(TS) 6. At least one pressurizer code relief valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests.

(TS) 7. The reactor coolant system pressure-temperature relationship and heatup rates shall be maintained within the limits of Figure 1.

8. If safety rods are to be withdrawn, maintain coolant pressure and temperature above the limit of the gas curve on Figure 1.

(TS) 9. Maintain the reactor subcritical by at least 1% $\Delta k/k$ until a steam bubble is formed and a water level between 10 and 316 inches is established in the pressurizer.

(TS) The pressurizer heatup rate shall not exceed 100F/Hr.

(TS) The pressurizer spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 250°F.

10. Do not exceed 350F Δt between SG downcomer and the secondary side feedwater temperature when using the main feedwater nozzles.

11. Do not start fourth RC pump if RC temperature is below 500F.

12. Maintain difference in boron concentrations between pressurizer and RC System $\leq$ 100 ppm.

13. Prior to any rod withdrawal or deboration, source range nuclear instrumentation will read at least 2 cps.

14. The safety groups will be fully withdrawn prior to deboration or heatup unless the boron concentration requirements of Figure 2 are met.

15. Nuclear instrumentation will continuously be monitored during any reactivity addition. During significant subcritical changes, a subcritical source multiplication will be confirmed or the addition will be suspended.

16. All subcritical boron changes will be verified every predicted 30 ppm change.

17. The core flood tank block valves will be closed and the motor control breakers will be racked out and tagged when RC pressure is <700 psig, except for testing.

18. The non-operating high pressure injection pump breakers are open and tagged out if the Decay Heat System is in operation.

19. The maximum ΔT between the OTSG tubes and the average OTSG shell is 60F.

20. The maximum allowable heatup rates of the RC system are:

    a. Ambient to 275F = 50F/Hr.
    b. 275F to 500F = 100F/Hr.
    c. Above 500F = 140F/Hr.

21. The auxiliary feedwater nozzles must be used to refill a dry OTSG or recover the level if it drops ≤8 inches.

22. Maximum OTSG fill rate is 500 gpm.

RC SYSTEM HEATUP

4.1 Initial Conditions:

_________ 1. RC system filled and vented with pressurizer level approximately 100 inches and N_2 blanket at 15-20 psig as read on computer.

_________ 2. Decay heat removal system is operating. (OP 1104 04)

_________ 3. Nuclear service raw water cooling systems operating if required for decay heat removal. (OP 1104 09)

_________ 4. If the OTSG's are in wet layup, draining to a level between 288 and 300" on the operating range should be in progress or completed.

_________ 5. ES channels 1, 2, 3 and 4 are bypassed.

_________ 6. Power is being provided by startup transformers 1 and 2, and at least one source of backup power to the transformers is available.

_________ 7. Auxiliary boiler operating at normal pressure.

__________ 8. Boron concentration in RC system is $\geq$ that required by Figure 2.

__________ 9. Boron concentration in MU tank $\geq$RC system concentration.

__________ 10. Startup, main and emergency feedwater block valves are in hand and closed.

__________ 11. Condensate storage tank and hotwell levels are in normal operating range. $\geq$16' condensate storage tank level.

__________ 12. Turbine plant, component cooling, and plant raw water systems are operating.

__________ 13. Both main feedwater pump turbines are on turning gear.

__________ 14. Main turbine is on turning gear with AC turning gear, seal oil backup and turbine bearing lift pumps operating.

__________ 15. Both core flood tanks are filled to 13.0 feet $\pm$ 0.4 feet with borated water $\geq$ 2270 ppmb and pressurized with 600 psig $\pm$ 15 psig nitrogen.

__________ 16. Required RB cooling units are operating.

__________ 17. Hydrogen header is charged to the makeup tank regulator with ~10-20 psig pressure on makeup tank.

_____(TS) 18. The borated water storage tank is filled with a minimum of 345,000 gallons of borated water with a concentration $\geq$2270 ppmb.

_____(TS) 19. The boric acid mix tank contains a minimum of 5000 gallons of borated water with a concentration $\geq$12,250 ppmb.

4.2 Procedure:

__________ 1. The following prestart/precritical checks, tests, and calibrations have been completed or are in progress:

*a. Nuclear Instrumentation
*b. Control Rod Drive System
*c. Engineered Safeguards
*d. Radiation Monitoring System
*e. Reactor Protective System
*f. Non-Nuclear Instrumentation

_________ 2. Valve checkoff lists for heatup completed or in progress as required.

_________ 3. Establish a steam bubble in the pressurizer. (OP 1103 05)

CAUTION

Do not exceed a pressurizer heatup rate of 100F/Hr.

_________ 4. While establishing pressurizer bubble and increasing RC pressure, continue with preparation for heatup. Pressure may be raised to NPSH if desired in preparation for starting RCP's.

_________ 5. Start two condenser circulating water pumps. (A and B, or C and D).

_________ 6. Line up feed and condensate systems to bypass the FW pumps and recirc to the main condenser. Start a condensate pump. (OP 1106 02).

_________ 7. Establish gland sealing steam to the main and FW pump turbines.

_________ 8. In preparation for establishing vacuum, perform the following:

a. Close vacuum breaker (S37)

b. Position 3-way valves (AR-S34 and S35) in vacuum pump position.

_________ 9. Start electric driven vacuum pumps and establish 15" HG absolute and then switch over to steam jet air ejectors. If the reactor coolant temperature is less than 160F, crack open the turbine bypass valves to establish a vacuum in the OTSG's. Bypass valves may be opened further after a period of time.

NOTE

The steam produced as a result of the reduced pressure will condense on the upper shell of the OTSG and allow for even heating of the OTSG shell and maintain the tube to shell ΔT <60F.

_________ 10. Verify bleed holdup tanks have sufficient capacity for expansion and deboration volumes.

___(TS)___ 11. Establish a positive flow path from the boric acid mix tank to the makeup tank and verify operability.

12. Start feed system cleanup per feedwater cleanup procedure. (OP 1106 02 Section 4). When feedwater is within the limits specified by the Water Chemistry Manual for startup, and if required, fill both steam generators to 288" on the operate range level recorder. (OP 1106 02 Section 5).

CAUTION

Do not exceed a fill rate of 250,000 lb/hr to each steam generator.

13. Close turbine bypass valves to 5% open upon establishing 10" HG absolute in the main condenser and OTSG.

14. Cooling water to DH coolers may be secured to increase RCS temperature with available decay heat and decay heat pump heat for startup.

15. Prior to the reactor coolant temperature exceeding 150F and a minimum of thirty minutes prior to starting a reactor coolant pump, start a makeup pump to establish seal injection flow to the reactor coolant pump seals (OP 1104 02).

16. Establish and regulate letdown flow to compensate for RC pump seal in leakage, pressurizer level control valve bypass flow, and RC system expansion.

17. Adjust pressurizer level setpoint to 100" (25%) and put level controller in auto.

18. a. If a minimum boron concentration, as indicated on Figure 2, is to be maintained until hot shutdown conditions are reached, the withdrawal of the safety rods is not required.

    b. If the boron concentration is not going to be maintained $\geq$ that concentration shown on Figure 2 or the moderator temperature coefficient of reactivity is positive, perform the following:

    *(1) Complete NI and RPS precritical checks/tests/calibrations.

    (2) Reset high flux trip setpoint to 5% full power.

    (3) Place the reactor protective system shutdown bypass switches in "Bypass".

(4) Verify that the reactor will be subcritical by at least 1% Δk/k with safety rods withdrawn during heatup.

(5) Verify that RC pressure-temperature relationship is above RC pump operation curve - Figure 1.

(6) Calculate source multiplication due to safety rod withdrawal.

(7) Withdraw safety rods in sequence to their upper limit. (OP 1105 09)

(8) Verify source multiplication during safety rod withdrawal. (OP 1102 03)

(9) Withdraw Group 8 to 25% if not previously withdrawn.

________ 19. Per OP 1103 06, RC Pumps, run each pump for five minutes. Let RC Pump A-2 run.

________ 20. Stop decay heat pumps and line up decay heat system for ES operation. OP 1104 04.

________ 21. Prior to exceeding 200°F and 300 psig, establish containment integrity.

________ 22. Sample and establish RC chemistry per Water Chemistry Manual.

________ 23. Complete CRD venting.

________ 24. Start pump in "B" loop and one pump in "A" loop. OP 1103 06.

________ 25. During heatup, perform the following:

a. Adjust L/D flow to maintain pressurizer level constant.

b. As required, divert L/D to bleed tanks to maintain makeup tank level in normal range 55-85".

c. Monitor and plot RC system temperature and pressure. Use pressurizer heaters and spray to maintain RC pressure and temperature within limits of Figure 1.

________ 26. When RC temperature is 220F, perform the following:

a. Shut the turbine bypass valves.

b. Commence draining the OTSG's, to minimum level, to the main condensers.

c. When level approaches 30", open startup block valves and place startup feedwater regulating valves in auto.

CAUTION

While heating up the reactor coolant system, do not allow the OTSG average shell temperature and the temperature of the reactor coolant to exceed a ΔT of 60F

_____(TS) 27. Verify the following equipment is operable:

a. One reactor building spray pump and its associated spray header.

b. Two low pressure injection pumps and their associated cooler.

c. Two nuclear service raw water pumps.

d. Two nuclear service cooling water pumps, associated coolers and valves.

e. One borated water storage tank level instrument channels.

f. The borated water storage tank shall contain a minimum of 345,000 gallons of water having a minimum concentration of 2270 ppm boron at a temperature not less than 40F.

g. Three reactor building emergency cooling units and their associated fans.

h. The engineered safety features valves and interlocks associated with each of the above systems shall be operable.

i. Exceptions to the above are listed in the Technical Specifications.

_____(TS) 28. Prior to the reactor coolant system exceeding 300F (OTSG pressure ~50 psig) close the OTSG drain valves to prevent exceeding the design temperatures of the reactor building penetrations.

_____ 29. Place the condensate and feedwater system in service to allow feeding the OTSG's with the condensate pumps. (OP 1106 02 Sec. 5).

_____ 30. Prior to the reactor coolant system exceeding 350F, verify the operability of two high pressure injection pumps to provide redundant and independent flow paths. The associated engineered safety features valves and interlocks shall also be operable.

_____ 31. Verify ES channels 3 and 4 reset by 600 psig RC system pressure.

_____ 32. a. Open CF tank block valves (CF SV81 and SV82) when RC system pressure is 700 psig.

b. The electrically operated discharge valves (SV81 and SV82) shall be locked open and tagged.

c. One pressure and one level instrument channel shall be operable for each tank.

_____ 33. a. Prior to the reactor coolant system exceeding 485F (OTSG pressure ~585 psig), place a feedwater pump in service (OP 1106 02 Section 6).

b. Place FOGG (Feed Only Good Generator) system in Normal.

_____ 34. When the reactor coolant system pressure is at 1650 psig and if the safety rods were withdrawn, perform the following:

NOTE

If the safety rods were not withdrawn, proceed to Step 35.

a. Terminate heatup by opening the turbine bypass valves and operate pressurizer heaters to maintain RC pressure at 1650 psi.

b. Insert safety rods to lower limit (OP 1105 09).

c. Return the reactor protective system shutdown bypass switch to "Normal" and reset the nuclear overpower trip setpoint. (All four channels).

d. Increase reactor coolant pressure to 2155 and place heaters and spray valves in auto.

CAUTION

Do not exceed the pressurized heatup rate of 100F/Hr.

e. Verify engineered safeguards channel 1 and 2 reset by 1750 psig.

f. When reactor coolant pressure is >1800 psig and all RPS channels are reset, withdraw safety rods in sequence to their upper limit. (OP 1105 09)

g. Calculate subcritical multiplication during rod withdrawal.

h. Close turbine bypass valves, place in Auto and resume normal heatup. Verify header pressure setpoint is set at 885 psig (47.5%).

________ 35. When RC system temperature reaches 500°F, start the fourth RC pump (OP 1103 06).

________ 36. Complete main steam line warming prior to reaching 885 psig steam pressure.

________ 37. While continuing heatup to hot shutdown, perform the following in preparation for reactor startup:

a. Complete all precritical checks, tests, and calibrations not completed during heatup.

b. If deboration is required, calculate bleed and feed volumes per soluble poison concentration control procedure OP 1103 04.

________ 38. When RC system temperature reaches 532°F, switch auxiliary steam supply to main steam header (AS-SV161 and AS-SV162), close auxiliary boiler supply to auxiliary steam header valve (AS-SV160) and shut down the auxiliary boiler.

________ 39. Place the makeup and purification system in normal service.

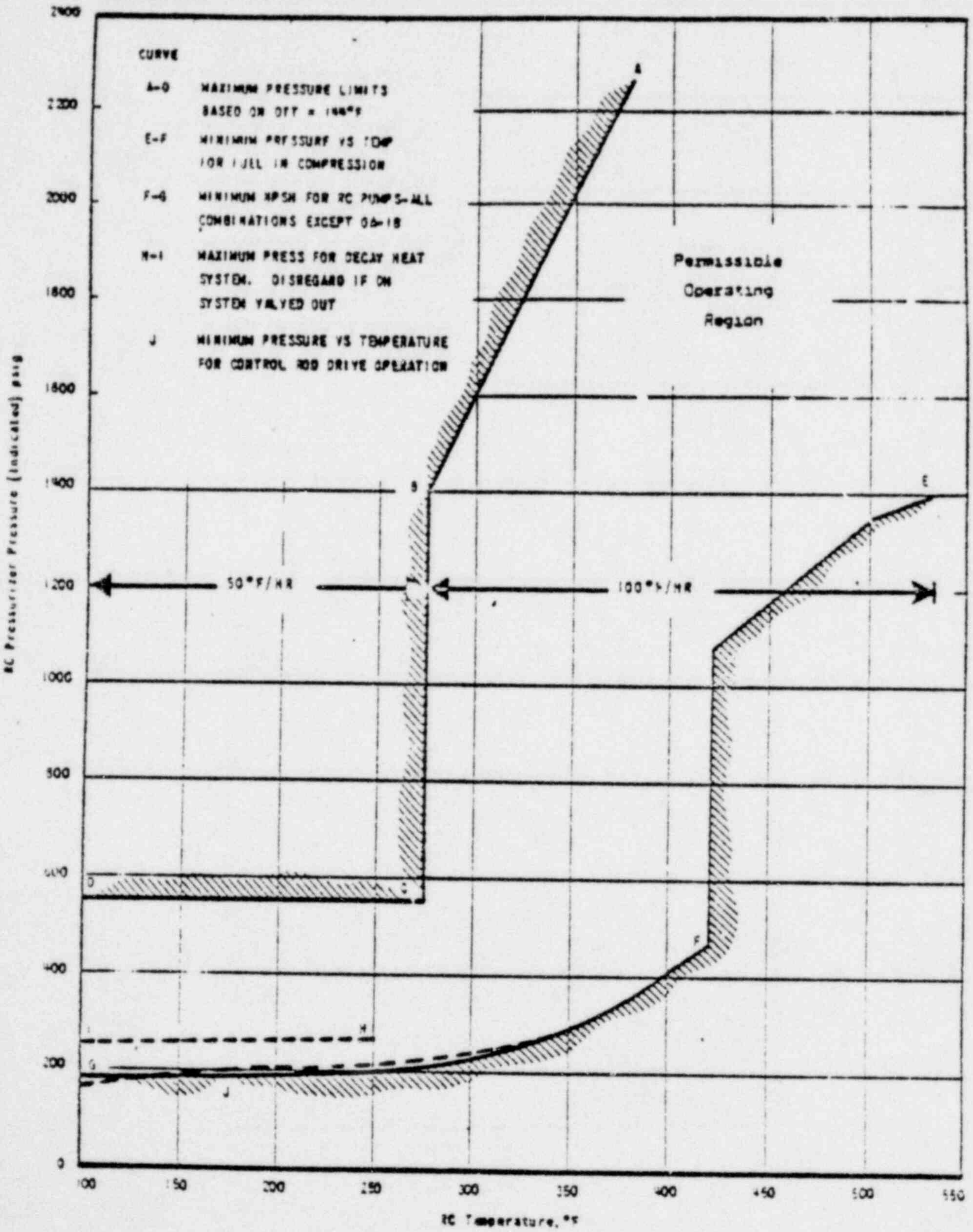
________ 40. Deborate the system to the desired concentration in preparation for startup if required.

CAUTION

Prior to deboration and if not already accomplished, withdraw safety rods in sequence to their upper limit (OP 1105 09).

4.3 Final Conditions

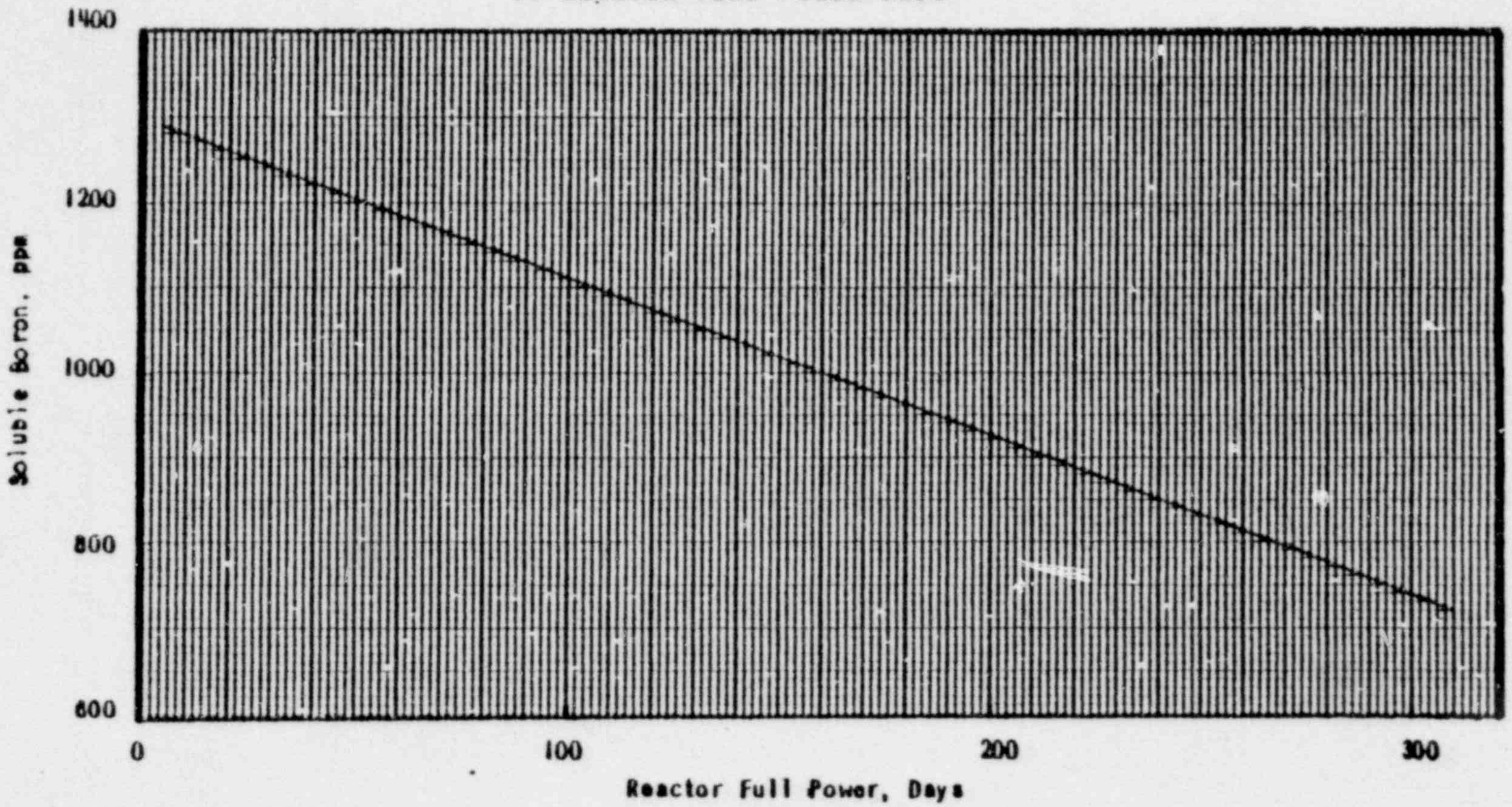
1. The RC system is in a hot shutdown condition (Tave >525F- pressure $\geq$2155 psig).
2. Pressurizer heaters, spray valve, and electro magnetic relief valves are in auto operation.
3. Pressurizer level is being maintained at approximately 100" in auto and normal makeup and purification flows have been established.
4. Safety rods are fully withdrawn. (If required, prior to deboration).
5. Bypass valves are maintaining steam header pressure at 885 psig.
6. OTSG levels are being maintained at minimum level by startup flow control valves feeding through main nozzles.
7. Deboration has been completed.



OP1102 02 RC SYSTEM LIMITATIONS (HEATUP)

FIGURE 1

FIGURE 2 SHUTDOWN BORON REQUIREMENTS (T_{AVE} BELOW 200°F) VS REACTOR FULL POWER DAYS



1102 03

OFR UNIT #1

PLANT STARTUP

OP 1102 03

1. PURPOSE

1.1 To provide the steps for operations listed below to take the plant from hot shutdown to 15% power.

OPERATION	SECTION
Reactor Startup	4
0-15% Power Operation	5

2. REFERENCES

2.1 SMUD Rancho Seco Unit 1 FSAR Technical Specification.

2.2 SMUD Rancho Seco Limits and Precautions.

2.3 OP 1103 15 Reactivity Balance Calculation.

2.4 OP 1103 04 Soluble Poison Concentration Control.

2.5 OP 1105 09 Control Rod Drive System

2.6 OP 1106 01 Main Turbine

3. LIMITS AND PRECAUTIONS

3.1 If any condition occurs which will physically or administratively delay criticality, the reactor will be maintained at least 1% subcritical.

3.2 The reactor coolant system will never be deborated and maintained at a boron concentration which would cause the reactor to be critical on safety groups due to xenon burnout during an increase in power with high xenon concentration.

3.3 Axial power shaping rods may not be used for any purpose except axial power control.

3.4 The reactor will not be critical with the pressurizer level greater than 316".

3.5 Recommended critical rod positions are:

(1) With no appreciable amount of xenon in the core ($\leq .5\% \Delta k/k$) -

(a) safety groups at upper limit.

(b) critical on group 5 at 50% or greater.

(2) With an appreciable amount of xenon in the core ($\leq .5\% \Delta k/k$)-

(a) safety groups at upper limit.

(b) critical on groups 6 or 7 with sufficient reactivity to override total temperature deficits between 0 and the desired power level on control rods.

3.6 The maximum startup rate is one DPM. The prompt change associated with attaining this SUR will be 1.5 DPM or less.

3.7 The minimum count rate prior to rod withdrawal is 2 CPS.

(TS) 3.8 The requirements for the reactor to be critical are specified in Technical Specifications.

(TS) 3.9 Minimum of one decade of overlap between source and intermediate range is required.

3.10 The heatup rate above $530^{\circ}F$ is $100^{\circ}F/hr.$ or less.

4. REACTOR STARTUP

4.1 Initial Conditions:

_________ (1) Plant is at hot shutdown.

_________ (2) Estimated critical rod position calculation completed (OP 1103 15).

_________ (3) All precritical and prestart checks completed.

_________ (4) Source range NI reading greater than 2 cps.

_________ (5) One feed train is operating and SG level is being maintained at minimum level, by startup feedwater valves in auto.

_________ (6) Turbine header pressure is being maintained 885 psig with bypass valves in auto.

_________ (7) Pressurizer level is being maintained at approximately 100" with pressurizer level controller in auto.

4.2 Procedure

_______ (1) If it is desired to verify source range instrumentation response, calculate amount of source multiplication that should occur due to safety group withdrawal. Refer to Enclosure 1.

_______ (2) Withdraw safety groups in sequence to their upper limits per CRD procedure (OP 1105 09) if required. Verify source multiplication.

_______ (3) If deboration is required, perform the following:

(a) calculate feed and bleed volume required.

(b) estimate source multiplication and final count rate.

(c) deborate the RC system per the soluble poison concentration control procedure (OP 1103 04).

_______ (4) Announce "Reactor Startup in Progress".

_______ (5) Withdraw regulating rods while observing NI. Monitor rod position to verify actual rod position vs. ECP is within limits.

NOTE

Criticality will be anticipated during this withdrawal.

(6) If the reactor is critical at the ECP or at a rod position equivalent to ± 0.5%Δk/k, continue with Step (8). If not, proceed with Step (7).

_______ (7) Insert rods to shutdown the reactor by 1%, resolve the problem, and repeat Steps (5) and (6).

_______ (8) If the reactor is critical, raise power to 10^{-8} amps on the intermediate range. Observe intermediate range NI level and rate are on scale and indicating before the source range is at 10^{-5} cps. If not, reduce power to 10^{-4} cps and correct the problem.

(9) At 10^{-8} amps on the intermediate range, record the following:

_______ (a) position of all rods.

_______ (b) boron concentration.

_______ (c) RC system average temperature.

_______ (d) time.

5. 0 TO 15% POWER OPERATION

5.1 Initial Conditions

_______ (1) Reactor is critical at 10^{-8} amps.

_______ (2) One feed train is operating and SG level is being maintained at minimum level, by startup feedwater valves in auto.

_______ (3) Header pressure is being maintained at 885# with the bypass valves in auto.

_______ (4) Pressurizer level is being maintained at approximately 100" with pressurizer level controller in auto.

5.2 Procedure

(1) Verify the ICS hand/auto stations are lined up as follows:

	STATION	POSITION	INITIAL
(a)	Steam Generator Reactor Master	Manual (Set at 0)	_______
(b)	Reactor Demand Tave Setpoint at 582 (~62)	Manual (Pseudo)	_______
(c)	ΔTc	Manual (Set at 50%)	_______
(d)	Loop A FW Demand	Manual (Set at 0)	_______
(e)	Loop B FW Demand	Manual (Set at 0)	_______
(f)	Main FW Pump A	Manual (Set at 0)	_______
(g)	Main FW Pump B	Manual (Set at 0)	_______
(h)	Main FW Valve A	Manual (Set at 0)	_______
(i)	Main FW Valve B	Manual (Set at 0)	_______

	STATION	POSITION	INITIAL
(j)	Startup FW Valve A	Auto	________
(k)	Startup FW Valve B	Auto	________
(l)	Main FW Block Valve A	Auto (Closed)	________
(m)	Main FW Block Valve B	Auto (Closed)	________
(n)	Startup FW Block Valve A	Auto (Open)	________
(o)	Startup FW Block Valve B	Auto (Open)	________
(p)	Emergency FW Block Valve A	Auto (Closed)	________
(q)	Emergency FW Block Valve B	Auto (Closed)	________
(r)	Emergency FW Control Valve A	Auto	________
(s)	Emergency FW Control Valve B	Auto	________
(t)	Turbine Bypass Valve A	Auto	________
(u)	Turbine Bypass Valve B	Auto	________
(v)	Atmospheric Dump Valve A	Auto	________
(w)	Atmospheric Dump Valve B	Auto	________
(x)	Header Pressure Setpoint	Set at 885# (~47.5)	________

(2) Verify the unit master setters are set up as follows:

maximum limit (MW)	- 967 or less
minimum limit (MW)	- 145
rate of change (MW/min)	- 96 or less

(3) Verify main FW flow control valves at zero and place in auto.

(4) Verify the following ICS stations at zero and place in auto.

(a) FW pump station for operating FW pump.

(b) Loop A FW demand.

(c) Loop B FW demand.

(d) Steam Generator-Reactor Demand.

(5) Line up both letdown coolers and both makeup filters for full letdown flow capability.

(6) Increase power to the heatup range and establish a linear heatup rate of 100F/Hr or less. If the reactor has been below 20% power for more than 48 hours, refer to Enclosure 4 for allowable rates of power increase. While heating up, perform the items listed in 5.2 (7) and 5.2(8) below.

CAUTION

While heating up do not exceed a pressurizer level of 316 inches.

(7) Regulate letdown flow to compensate for part of coolant expansion while allowing pressurizer level to increase to the normal operating level within the limits of Enclosure 3.

NOTE

(a) Divert letdown to bleed tanks as required to maintain makeup tank level in the normal operating range.

(b) At 582°F the pressurizer level should be 220 inches and the makeup tank should be between 75 and 85".

(8) While heating up RC system, perform the following:

(a) Monitor with plant computer or plot heatup rate.

(b) Perform turbine testing per OP 1106 01.

(c) Monitor NI to insure proper range NI overlap is proper.

(d) Observe as power is increased, turbine bypass valves and startup feed valves open. Verify feed valves ΔP does not drop below 30 psi.

(e) Monitor reactor power versus RC average temperature and verify it is within limits of Enclosure 2.

(f) Follow pressurizer level increase with pressurizer level setpoint.

_______ (9) When reactor power is 5% verify a zero neutron error exists and place the rod control panel in auto.

_______ (10) Increase reactor power to 10% using reactor demand station in hand.

_______ (11) Startup and load the turbine generator to 5% load per OP 1106 01.

NOTE

While starting and loading the turbine, observe the turbine bypass valves operate properly to maintain header pressure at 885 psig.

_______ (12) Continue heatup by increasing reactor power to 15%

_______ (13) When the reactor is at 15%, verify Tave error less than 2°F. and place reactor demand in auto.

_______ (14) When the turbine warmup period at minimum load is satisfied, with the turbine control in "operator auto", set the load reference setter to 200 MW and increase load by depressing "Go".

_______ (15) When turbine bypass valves are closed, load is ~20% (190 MW) and header pressure is 885#, place the turbine control in "operator ICS".

NOTE

Observe turbine supervisory instruments. Should any limits be approached, depress Hold and correct situation.

_______ (17) When low level limits clear, verify the ΔTc controller is set at 50% and place in Auto.

5.3 Final Condition

_______ (1) The plant is at ~15% power.

_______ (2) The RC system is at 582 Tave and 2155 psig.

_______ (3) Pressurizer level is 220" and the makeup tank level and pressure is within the normal operating range.

_______ (4) All ICS stations are in Auto except the FW pump that is not operating.

ENCLOSURE I

Method to determine neutron multiplication.

$$\text{Multiplication } (m) = \frac{R_1 (100 - R_2)}{R_2 (100 - R_1)}$$

Note: R_1 and R_2 are negative values.

Where R_1 = how far for subcritical prior to reactivity change (%ΔK/K)

Where R_2 = how far subcritical after reactivity change (%ΔK/K)

$CR_2 = M(CR_1)$ where CR_2 = New count rate.
CR_1 = Initial count rate

EXAMPLE 1

Condition: The reactor is 10% subcritical 5.3% reactivity is added by withdrawing safety rods. The initial count rate is 10 counts. The reactor is now 4.7% subcritical.

Calculation: $R_1 = -10\%$ $R_2 = -4.7\%$

$$m = \frac{10\ (100 + 4.7)}{4.7\ (100 + 10)} = \frac{10(104.7)}{4.7(110)} = 2.025$$

$CR_2 = 2.025(10) = 20.25$ cps

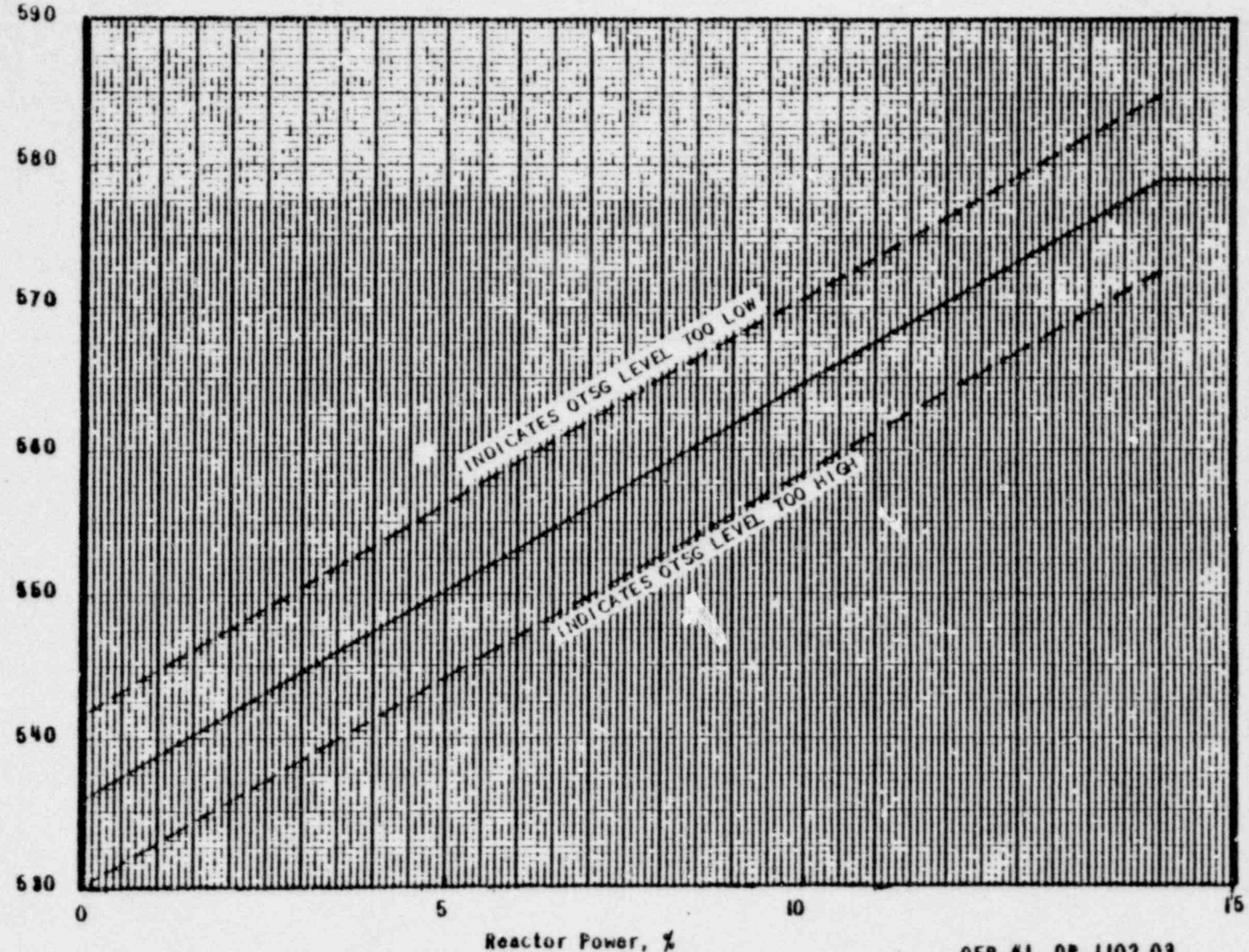
EXAMPLE 2

CONDITION: Safety rods are withdrawn the reactor is 5% subcritical. Deborate 100 ppm CR_1 = 500 cps boron worth hot is 0.1% ΔK/K/ppm.-
$\Delta\rho_b = (100 \text{ ppm})\ .01 = 1\%$
The reactor is now 4% subcritical.
$R_1 = -5\%$ $R_2 = -4\%$

$$m = \frac{5(100+4)}{4(100+5)} = \frac{5(104)}{4(105)} = 1.23$$

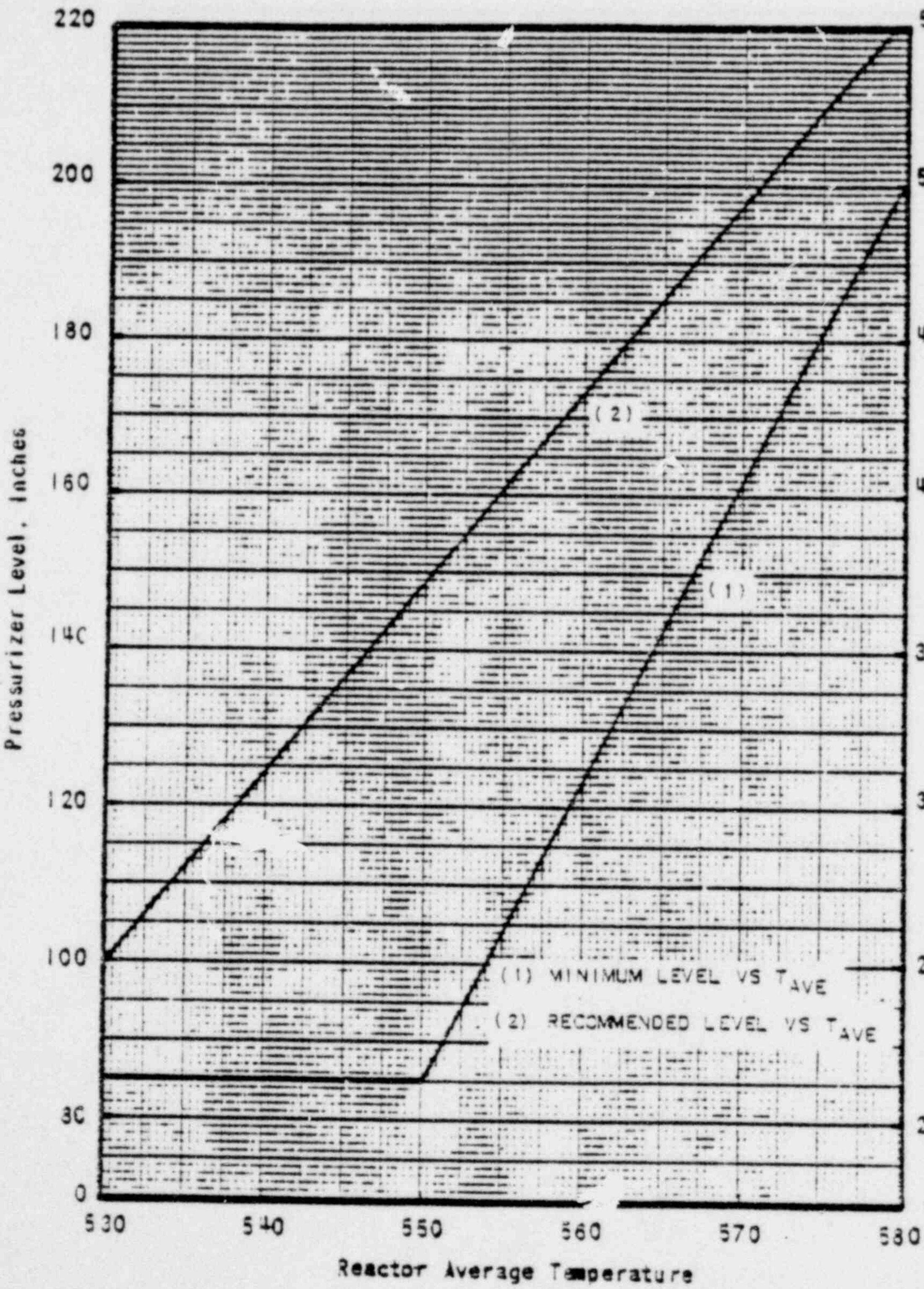
$CR_2 = M(CR_1) = (1.23)\ (500) = 619$ cps

ENCL. 2 REACTOR POWER VS T_{AVE}



OFR #1 OP 1102 03

ENCL. 3. PRESSURIZER LEVEL VS T_{AVE}



ENCLOSURE 4

If the reactor has been below 20% powe for more than 48 hours, the following mechanical maneuvering rates are recommended:

(1) Below 20% power the maximum rate of change should be 10%/hour. This corresponds to a heatup rate of approximately 30°F/hour.

(2) Above 20% power the maximum rate of change should be 30%/hour with five-hour holds at 20% below power level cutoff and power level cutoff.

1102 04

CFR UNIT #1

POWER OPERATIONS

OP 1102 04

1. PURPOSE

To describe operations of the plant at power (>15% load i.e. 127 mwe). Included is routine operations which should be performed.

Operation	Section
Power Increase	4
Steady State Operation	5
Power Decrease	6
Perodic Operations	7

2. DESCRIPTION

System Responses

The expected various parametric system responses for the reactor coolant and steam supply system are given below. Also included are allowable deviations.

2.1 Reactor Coolant System

1. Reactor coolant pressure

   During power operations RC pressure should be maintained at 2155 ± 100 psig by pressurizer heaters and the pressurizer spray valve.

2. Reactor coolant temperature

   The RC system Tave will be maintained by control rods.

   The expected reactor coolant temperature (average, inlet and outlet) along with allowable deviations are specified by Figure 1.

3. Pressurizer level

   Pressurizer level will be maintained at 220 inches by the pressurizer level control valve and the letdown valve.

2.2 Steam and Power Conversion

1. OTSG outlet pressure.

   The OTSG outlet pressure vs. reactor power along with expected and allowable deviations are specified by Figure 2.

2. OTSG outlet steam temperature

   The OTSG outlet steam temperature vs. reactor power along with expected and allowable deviations are specified in Figure 3.

3. OTSG (level)

   The OTSG (level) vs. reactor power for startup range and operate range level instrumentation along with expected and allowable deviations are specified in Figures 4 and 5 respectively.

4. Feedwater flow

   The feedwater flow vs. reactor power along with expected and allowable deviations are specified in Figure 6.

5. Feedwater temperatures

   The feedwater temperature vs. reactor power along with allowable deviations are specified by Figure 7.

6. Turbine

   The turbine load change rates are specified by Figures 9 and 10.

2.3 Control Rods

Control rod sequencing for first (approximately) 385 full power days.

Full length control rod positioning.

1. General description

   The operational control rod position limits as a function of reactor power are specified in Figure 8. These limits serve the following purposes.

   a. The limits insure the capability exists for reactivity control with rods in the maneuverability range.

   b. The limits serve as a simple distinction between transient xenon, which is controlled mostly with rods, and equilibrium xenon, controlled with soluble poison.

      NOTE: In practice it is difficult to distinguish between equilibrium and transient xenon. Descriptive definitions are listed below. Transient xenon would be that reactivity effect associated with immediate after effects of a change in power. (The xenon burnout and initial buildup after a power increase or the buildup and initial decay after a decrease in power). Equilibrium xenon would be that effect following a change in power after the initial xenon transient.

c. The limits insure that design shutdown margin is not decreased by excessive rod control.

d. The withdrawal limit at 100 percent of rated power increases DNB margin for normal conditions.

2. Boration - Deboration

Boration or deboration will occur whenever the following conditions exist.

a. Whenever rod position limits (FIGURE 8) are exceeded during equilibrium xenon changes.

b. Whenever maximum insertion or withdrawal limits are exceeded during xenon transients

3. Axial power shaping rod positioning

APSRS are manually controlled, and will be moved to maintain an imbalance at or near zero at all times.

4. Control rod sequencing near end of life (after approximately 345 full power days).

Near end of life the transient bank (Group 7) is withdrawn to gain increased core lifetime at the expense of power transient capability. The last few days of power operation are at steady state power to permit complete withdrawal of group 7. Figure 8A shows the position of Group 7 as a function of time along with the maximum transient capability available.

3. LIMITS AND PRECAUTIONS

1. Maintain surveillance requirements in accordance with (Technical Specifications).

2. Maintain power imbalance at 0%.
3. If the operator fails to move or moves the APSRS in the wrong direction, the action could result in a reactor trip.
4. If power imbalance is negative APSRS are inserted. However, if APSRS are inserted more than the 10% withdrawn position the power imbalance may increase (become more negative).
5. Prior to performing a heat balance insure quadrant power tilt does not exist.
6. In the event of an unscheduled load reduction, the power level shall not be increased until an investigation has been conducted and any necessary corrective action taken.
7. To minimize power tilts, maintain rod positions within a group at the same level.
8. The turbine may not be operated with a reheat stop or intercept valve closed above 95 mw except for valve testing (only short duration closure).
9. The turbine generator may not be operated below 58.5 HZ. This would result in undue vibration of the last row of turbine blading.
10. The reheat temperature should not be allowed to step change >250/hr.
11. The turbine drains will be open at 170 mw load.
12. If the exhaust hood steam temperature alarm is reached, the operator should attempt to lower this temperature by any of the following means:
    *a. Verify exhaust hood spray system is in operation.
    b. Increase vacuum.
    c. If at low load, increase load above 10-15% of rating.
13. Vibration Limits (double amplitude - mils)
    a. 4.0 mils - satisfactory
    b. 7.0 mils - alarm (investigation is indicated if vibration is continuous and of the unbalanced type).
    c. 14.0 mils - trip or other suitable action (which may be load change, speed change, etc., according to specific conditions.

4. POWER INCREASE

4.1 Initial Conditions

1. The plant is at or above 127 mw load under operator control or dispatcher control.

2. Plant parameters are within the limits of Figures 1 through 8.

4.2 Procedure

1. Determine rate of load change, from Figures 9 and 10, and adjust unit master rate limit.

2. Set load demand to 350 mw.

3. At 170 mw perform the following:

   a. Close main turbine drains.

   b. Transfer all plant loads except Nuclear Service Busses from startup transformers to the station auxiliary transformers.

4. At 350 mw perform the following:

   a. Start two additional condenser circ water pumps.

   b. Start heater drain pumps A and B.

   c. Start second condensate pump.

5. At 425 mw start second FW pump and put in ICS control per OP 1106 02.

6. Set the load demand to the desired power.

7. At 650 mw start the third condensate pump.

8. During power increase verify and or perform the following:

   a. Plant parameters are within the ranges of Figures 1 through 8.

   b. Adjust APSRs to maintain power imbalance at or neat 0%.

   c. Adjust turbine generator hydrogen pressure to the following:

< 350 mw		15 psig
> 350 mw	<650 mw	30 psig
> 650 mw		45 psig

5. STEADY STATE OPERATION

5.1 Initial Condition

The nuclear steam system has been brought to a condition where load equals load demand.

5.2 Procedure

1. Maintain an axial power imbalance at or near zero at all times.
2. Maintain control rod within the recommended rod configurations per power level vs rod positions (Figure 8) by boration or deboration for fuel burnup or equilibrum xenon changes.

*3. Maintain primary and secondary chemistry per water chemistry manual and technical specifications.

4. Maintain routine component test schedule per Section 7.

*5. Maintain shift Health Physics routine.

6. POWER DECREASE

6.1 Initial Conditions

1. The ICS is under operator or dispatcher control.
2. The plant is at steady state conditions.

6.2 Procedure

1. Determine rate of load change from Figures 9 and 10, and adjust unit master rate limit.
2. Set in desired load demand.
3. Move ASPRs as required to maintain power imbalance at or near 0%.
4. At 650 mw, perform the following:
   a. Reduce hydrogen pressure on main generator to 30 psig.
   b. Reduce the number of operating condensate pumps to two (2).
5. At 425 mw, stop one feedwater pump.
6. At 350 mw, perform the following:
   a. Reduce hydrogen pressure on main generator to 15 psig.
   b. Stop one condensate pump.
   c. Stop both heater drain pumps.
7. At 170 mw open the main turbine drains.

7. PERIODIC OPERATIONS

7.1 Daily (each shift)

1. The shift supervisor will make an inspection of the plant as early in the shift as possible.
2. Make a comparison check of all nuclear, non nuclear, protective and E. S. analog signals to see that the channels are functioning properly.
3. Replace all charts as required with starting time and date and end time and date, as well as the instrument name and number. Mark a piece of paper and place in the recorder for the new starting time and date.

7.2 Daily

1. Evaluate RC system for leakage per OP 1103.13.
2. Perform reactivity balance per OP 1103.15.
3. Perform heat balance and after a return to power from an outage. (OP 1103 16)

7.3 Weekly

1. Test main turbine stop valves, reheat stop valves and intercept valves.
*2. Test all main and feed pump turbine lube oil pumps.

(TS) 7.4 Every two weeks.

1. Exercise control rods per Exercising Control Rods at Power OP 1105.11.

(TS) 7.5 Surveillance standards

1. Perform all checks, test, calibrations and verify operability at the frequencies per surveillance standards (technical specifications).

Figure 1. Reactor and Steam Temperature Vs Power

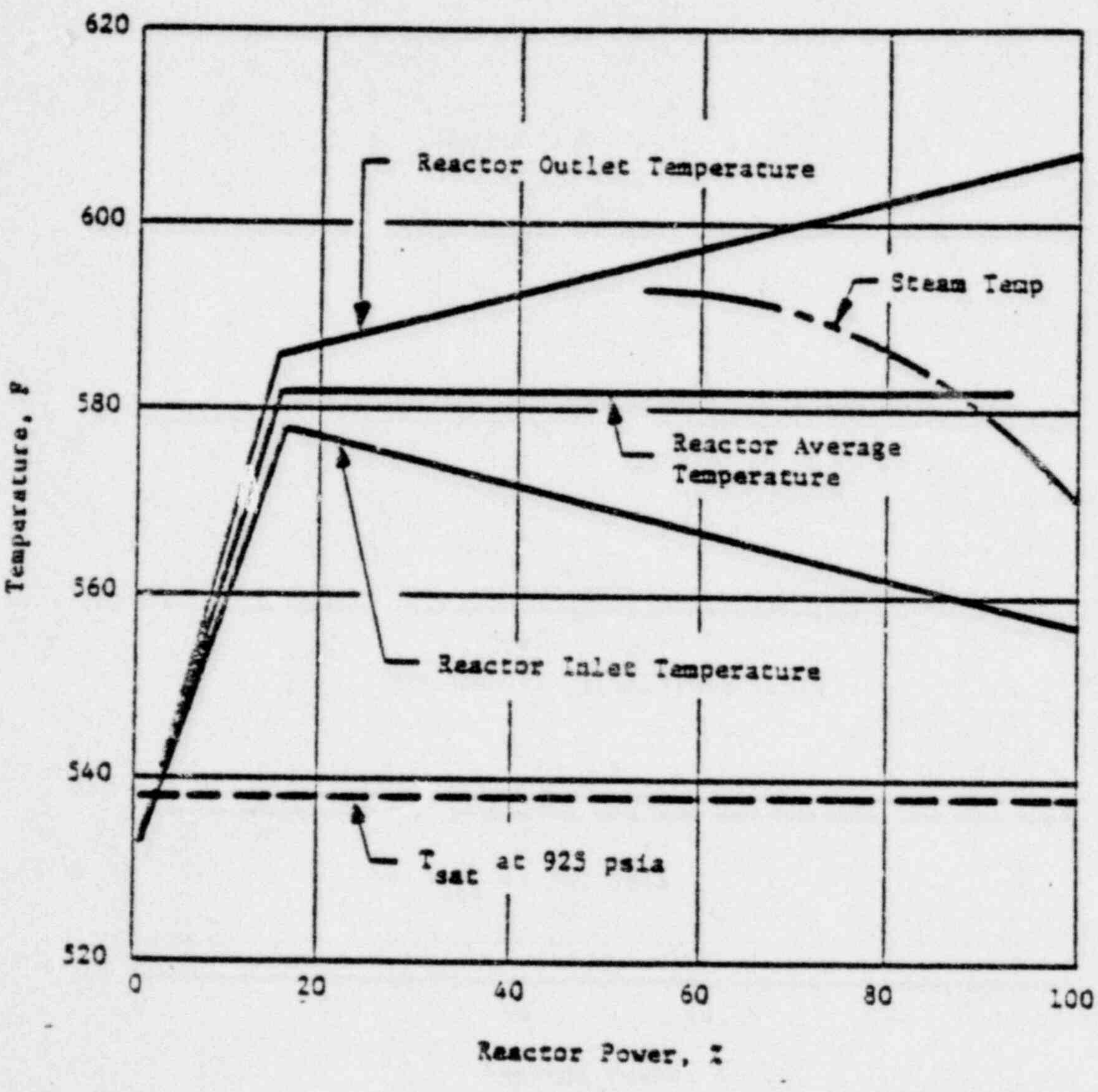
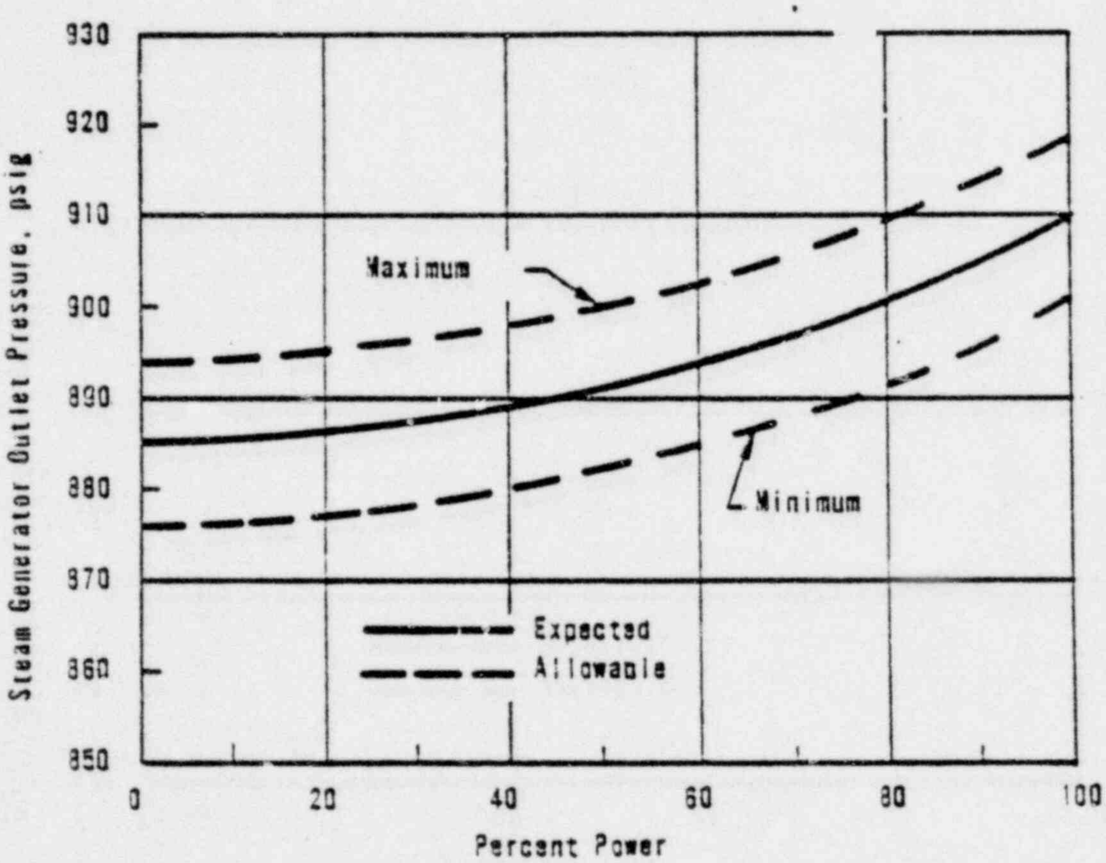
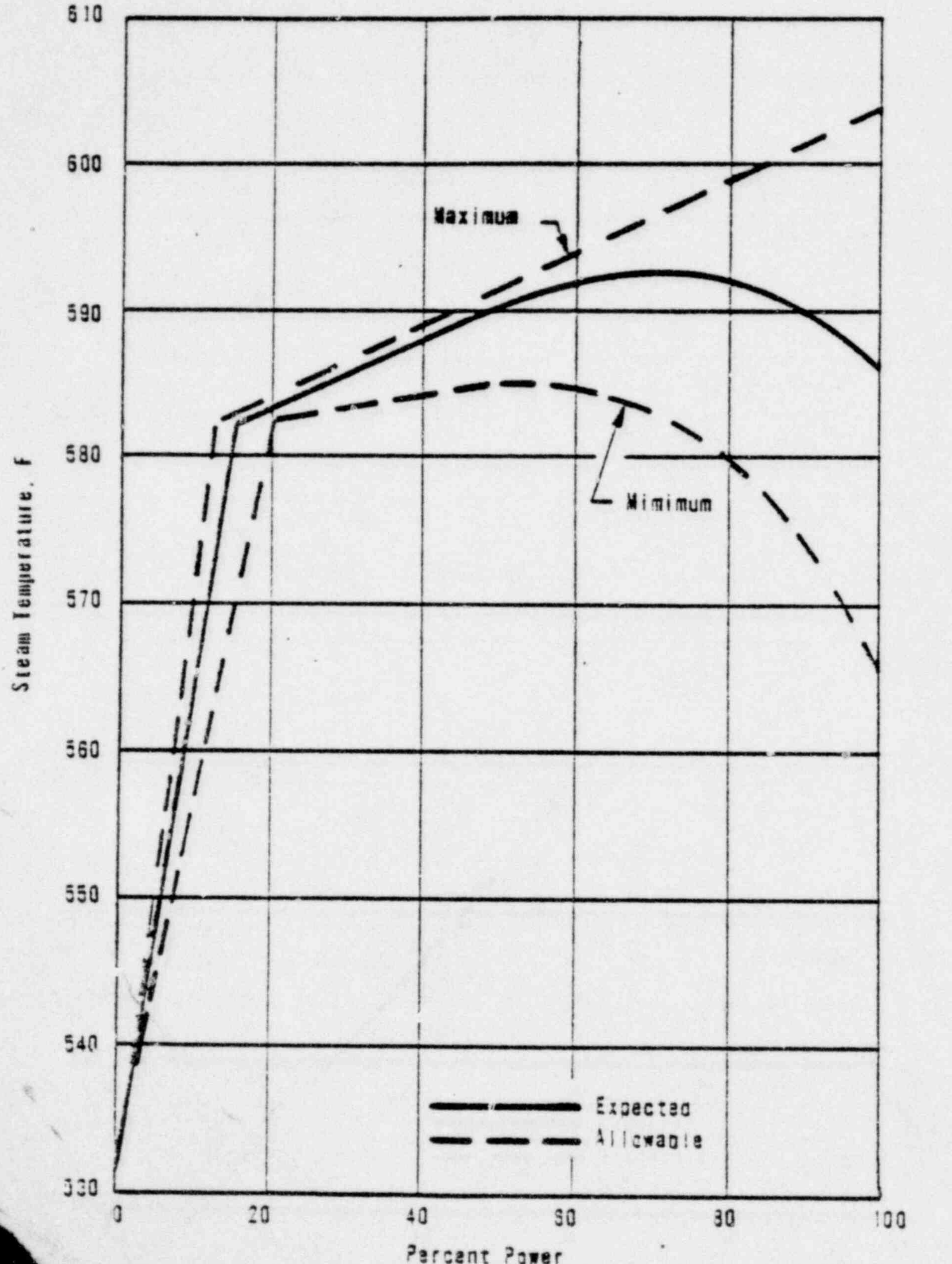


FIGURE 2 STEAM GENERATOR OUTLET PRESSURE VS POWER LEVEL



OP 1102.04

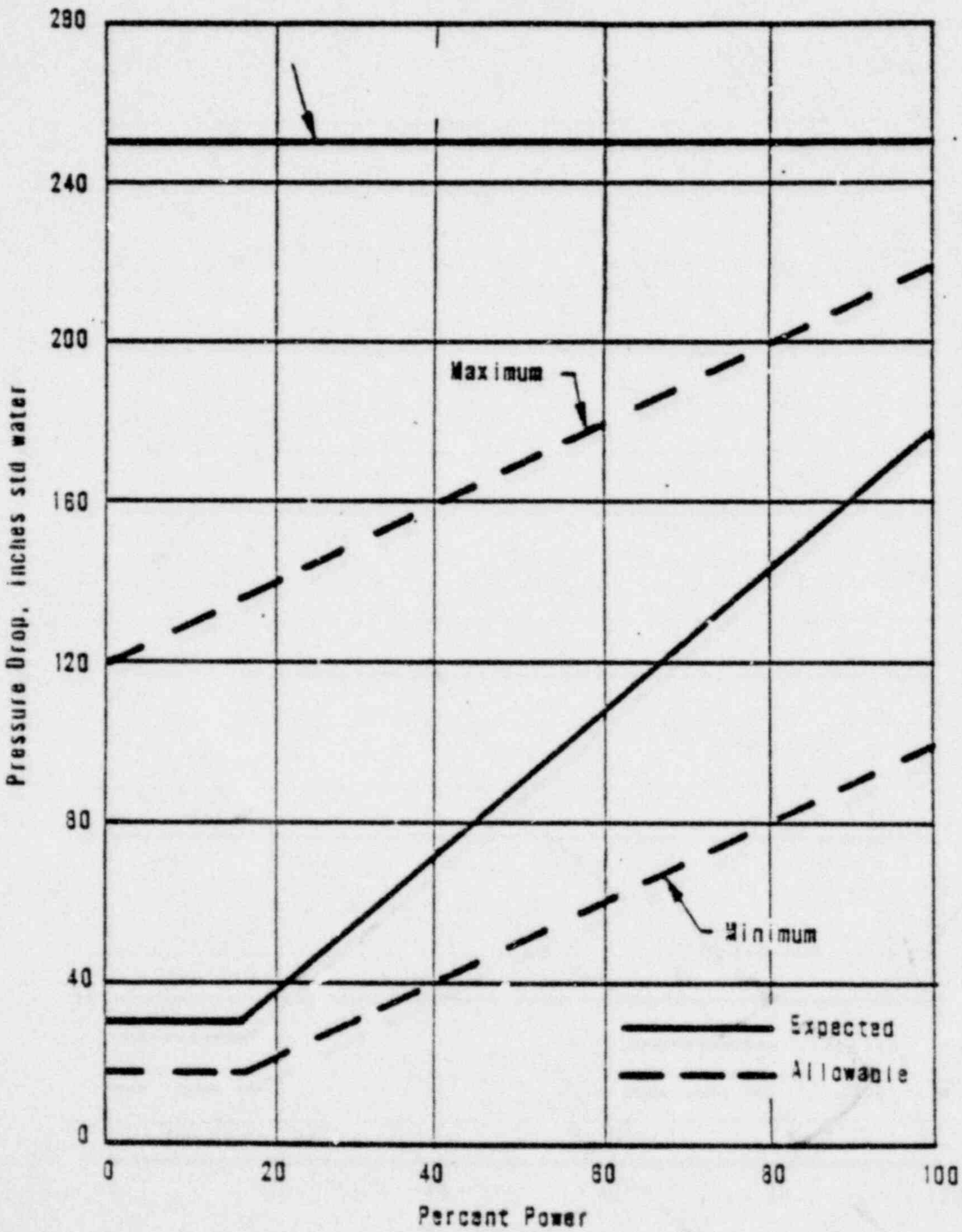
FIGURE 3 STEAM TEMPERATURE VS POWER LEVEL



OFR UNIT #1

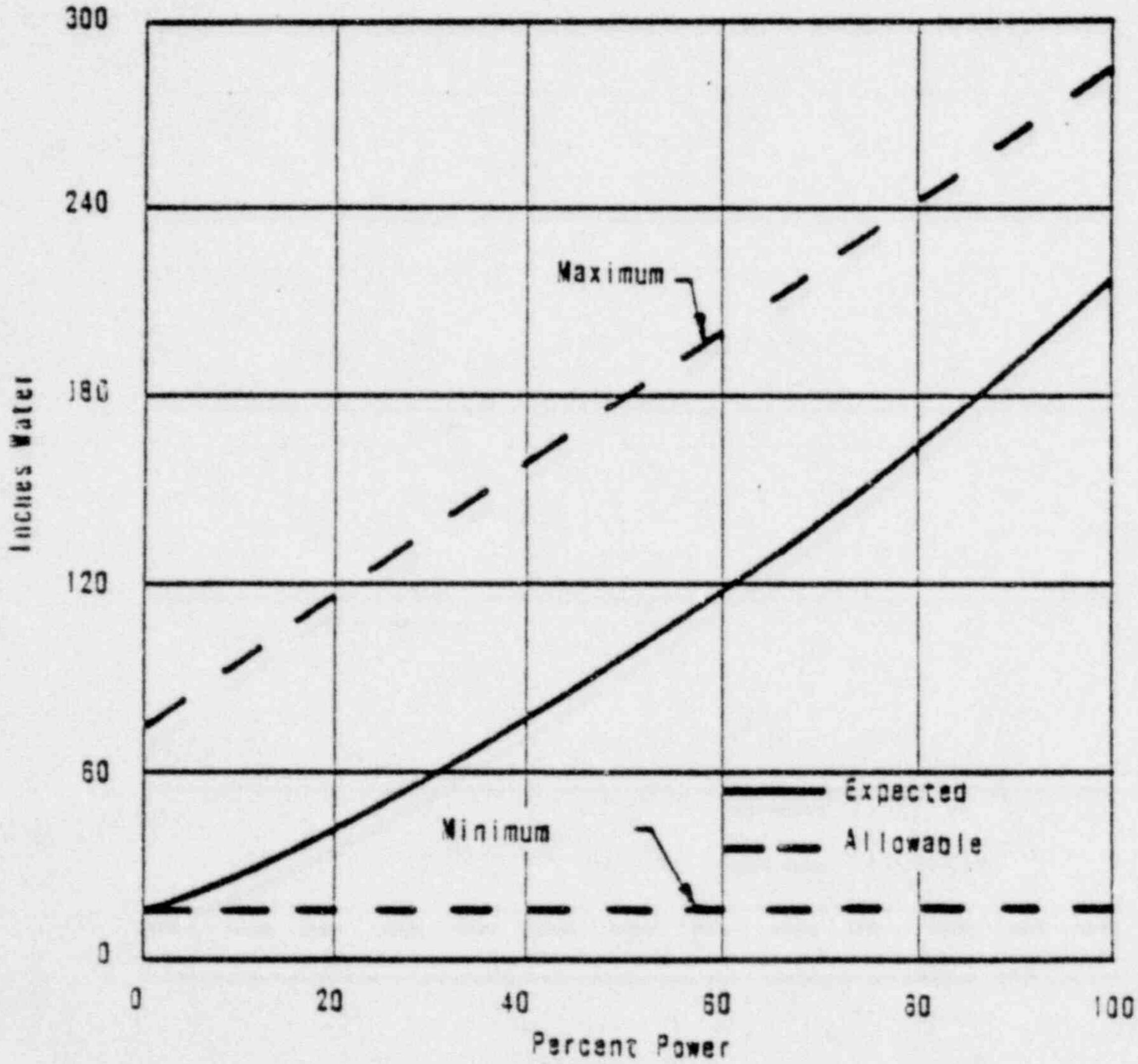
OP 1102.04

FIGURE 4 OTSG STARTUP RANGE "LEVEL" ΔP
VS POWER



OP 1102.04

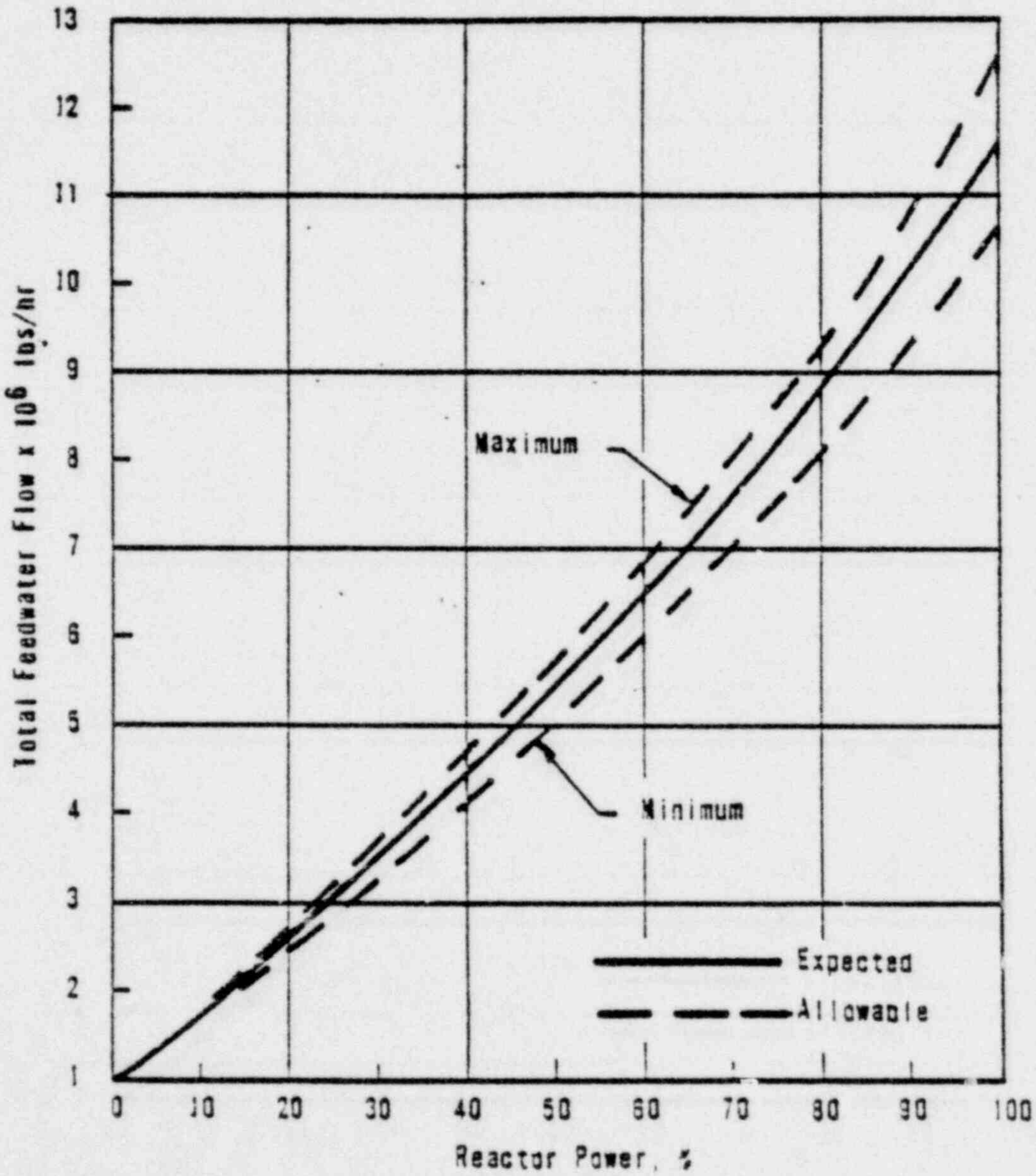
FIGURE 5 OTSG OPERATE RANGE "LEVEL" VS POWER



OP 1102.04

OFR UNIT #1

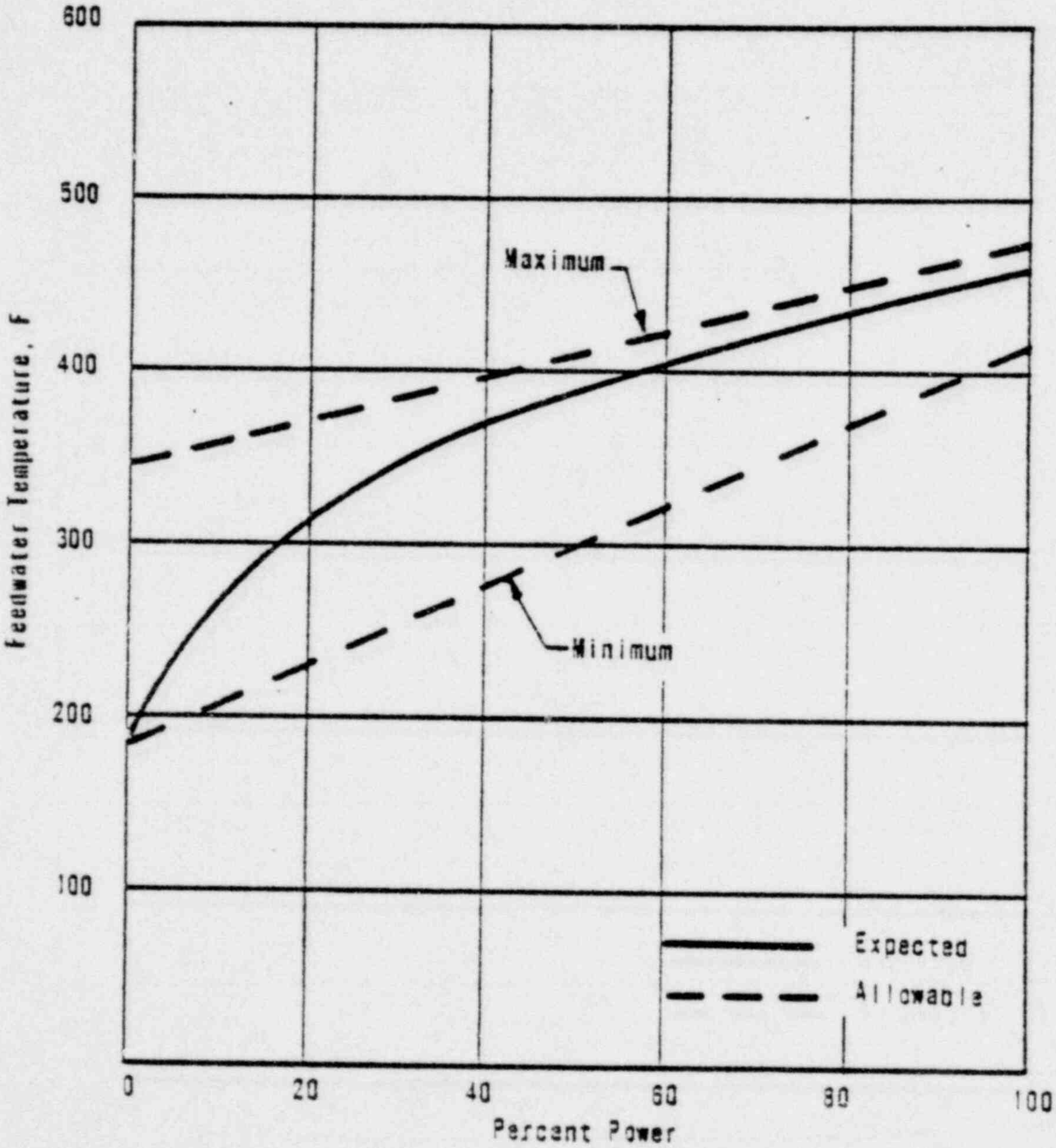
FIGURE 6 TOTAL FEEDWATER FLOW VS POWER



OP 1102.04

OFR UNIT #1

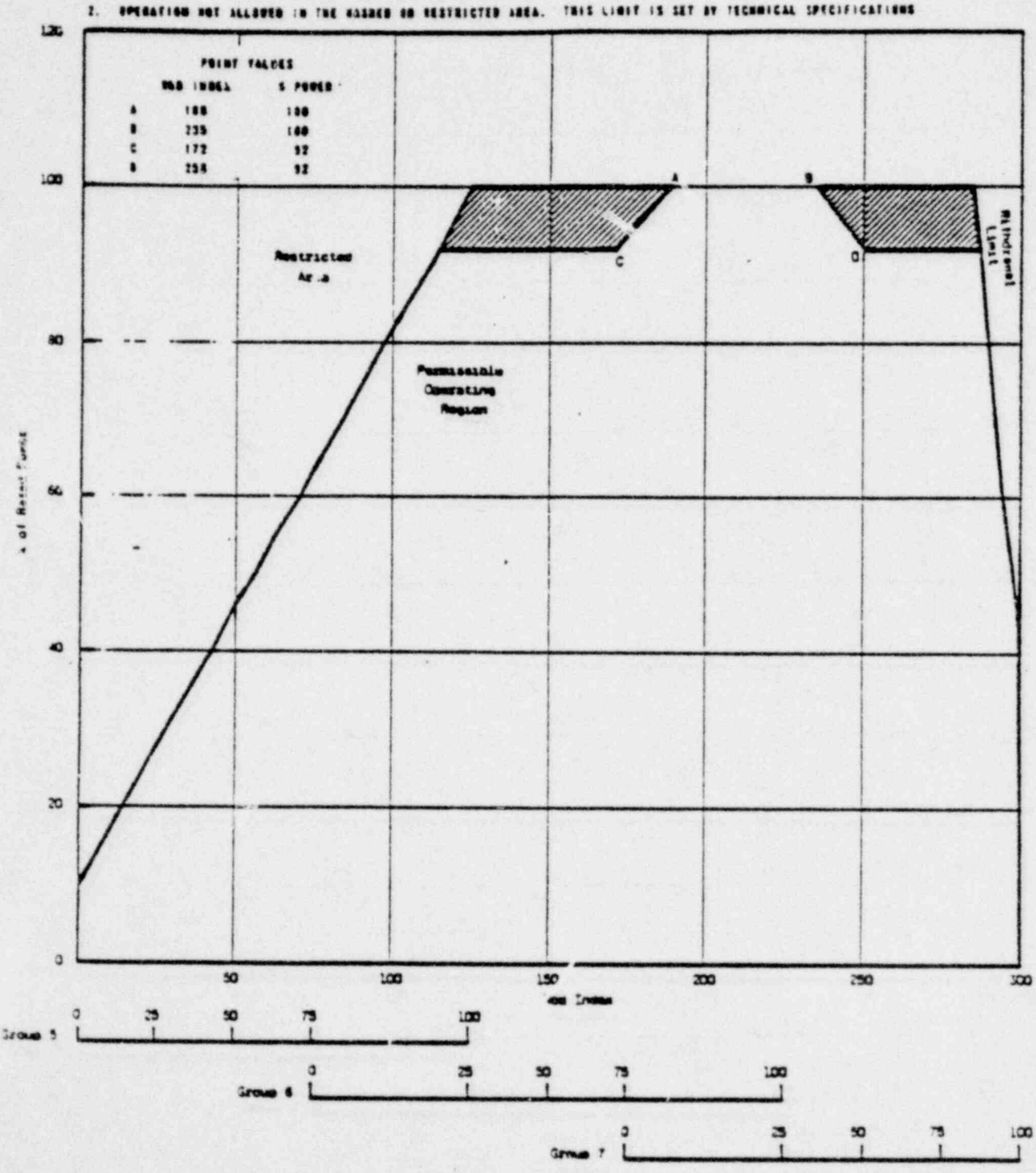
FIGURE 7 FEEDWATER TEMPERATURE VS POWER



OP 1102.04

NOTES:

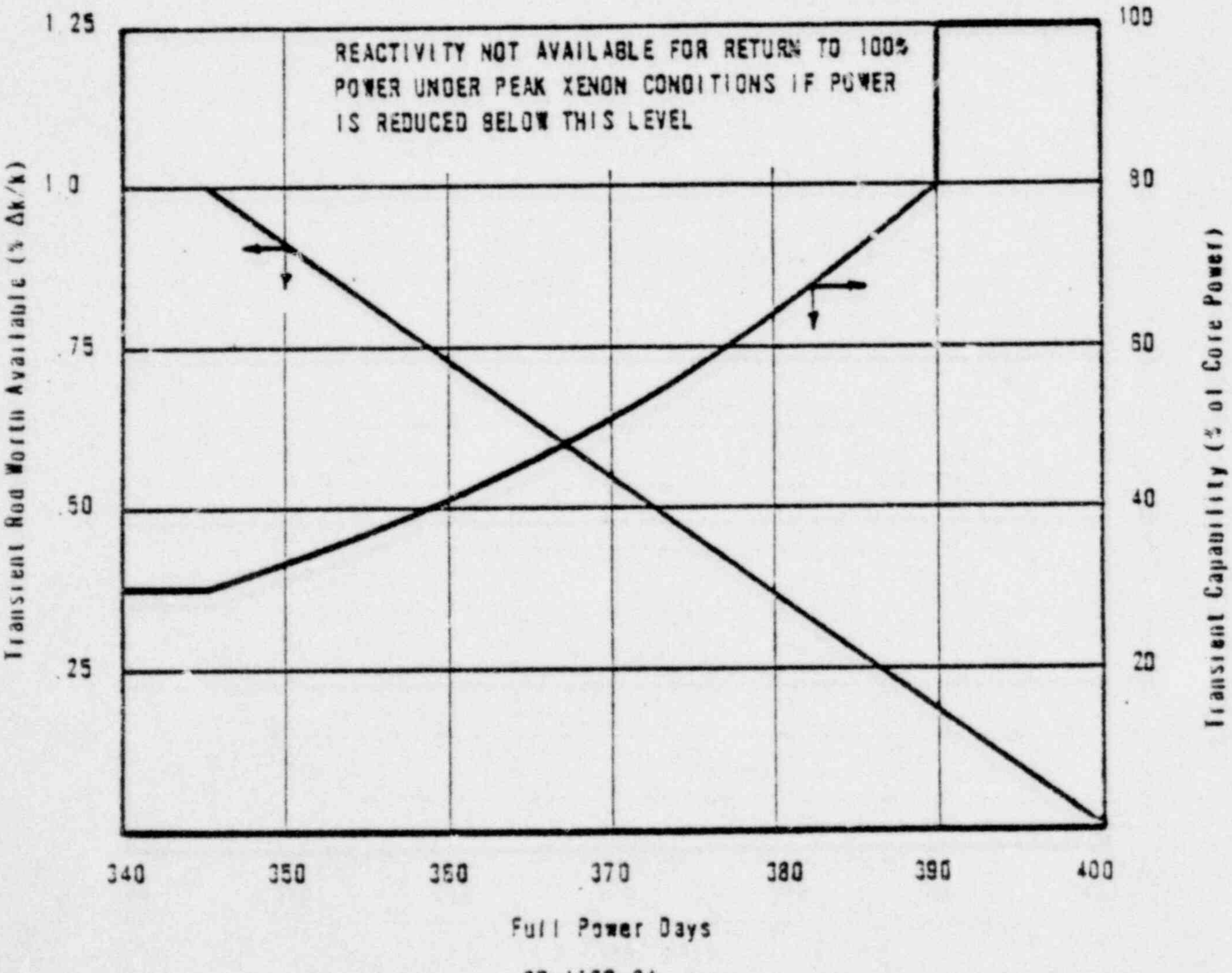
1. ROD INDEX IS THE SUM OF THE REGULATING GROUP POSITION INDICATIONS
2. OPERATION NOT ALLOWED IN THE HASHED OR RESTRICTED AREA. THIS LIMIT IS SET BY TECHNICAL SPECIFICATIONS



OP 1102.04 CONTROL ROD GROUP WITHDRAWAL LIMITS CFR 41

Figure 8

FIGURE 8A AVAILABLE ROD WORTH AND TRANSIENT CAPABILITY NEAR EOL



OP 1102.04

OFR UNIT #1

LOAD-CHANGING RECOMMENDATIONS NUCLEAR STEAM SYSTEM UNITS

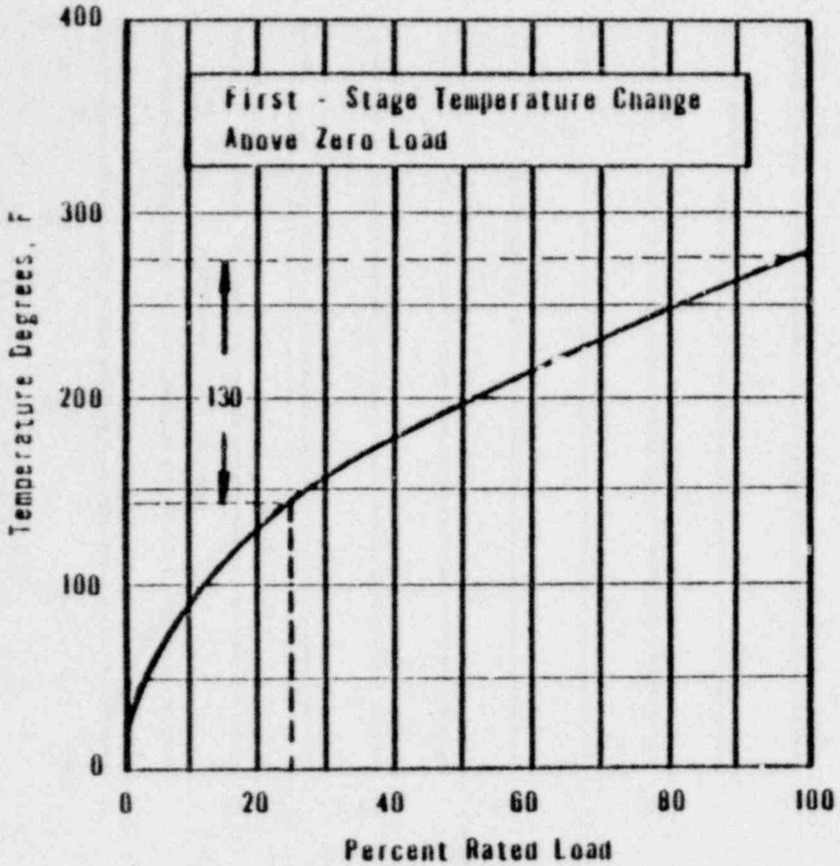


FIGURE 9

OP 1102.04

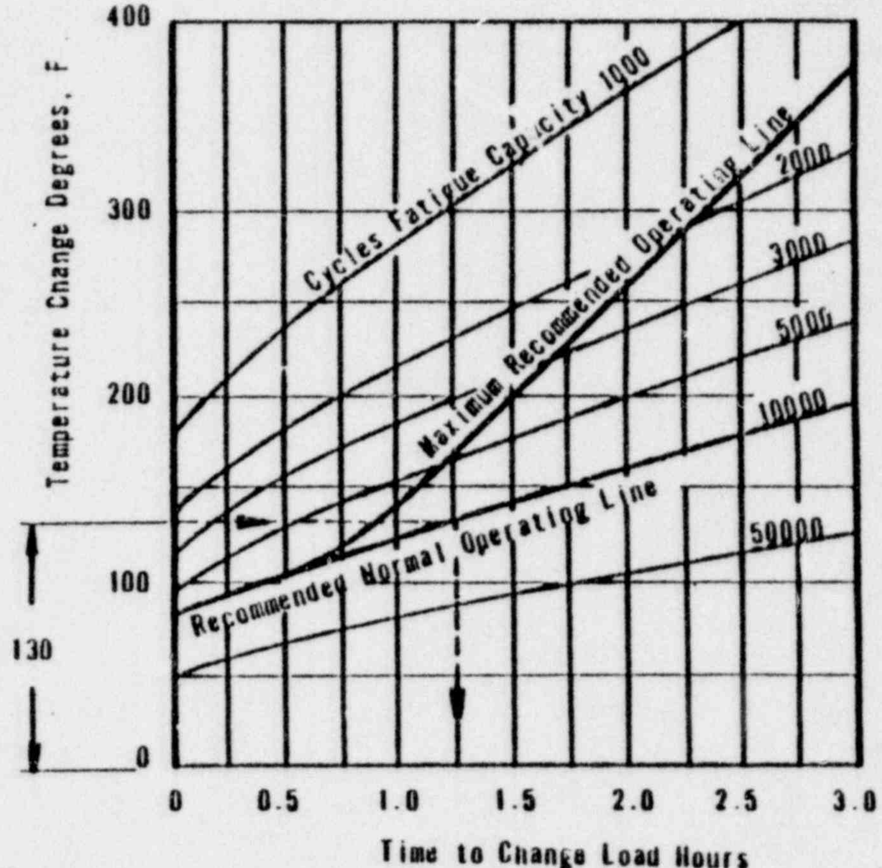


FIGURE 10

OFR UNIT #1

SHUTDOWN TURBINE-REACTOR

OP 1102 10

1. PURPOSE:

This procedure provides the steps necessary to take the plant from 15% power to hot shutdown conditions.

2. DESCRIPTION:

2.1 Tave control below 15% power.

1. With a constant inventory of water in the OTSG's each power level has a given Tave associated with it so to control cooldown rate the operator should adjust the rate of power decrease to adjust the cooldown rate.

2.2 Reactivity control.

1. With all rods in, the method of reactivity control is by adjusting boron concentration in the RC system.

3. LIMITS AND PRECAUTIONS:

1. The cooldown rate limit above 530 F will be less than 140 F/hour.
2. Avoid air being drawn through the turbine glands with the rotor at rest.
3. Maintain as high a vacuum as possible.

NOTE:

The vacuum breaker may not be used to slow the turbine down except in an emergency.

4. As long as steam is going to the main condenser the condenser circulating water pumps must be operating.
5. Maintain pressurizer level within the limits of Figure 1.
6. During boration verify boron concentraiton every estimated 30 ppm change when the reactor is subcritical.
7. Maintain makeup tank above low level alarm.

4. REACTOR-TURBINE SHUTDOWN:

4.1 Initial conditions:

1. Unit is at 15% load.
2. Unit Tave is 582 ~~579~~ F.
3. One main feedwater pump is operating in auto.
4. One main condensate pump is operating.
5. One bleed holdup tank with boron concentration greater than or equal to RC system boron concentration is lined up to feed to the makeup tank.

4.2 Procedure:

1. Set unit master setter at 50 MW.
2. As load decreases below ~~127~~ MW, observe turbine bypass valves open to take excess steam and that header pressure is maintained at setpoint.
3. If shutdown is preparatory to cooldown, stop one RC pump in each loop.
4. Transfer RC pumps to #1 startup transformer.
5. Transfer the rest of the site loads to #2 startup transformer.
6. Place the EHC in "Operator Auto".
7. Place reactor-steam generator demand, loop feedwater demand A & B, and reactor demand in hand and decrease loop feedwater deman A & B, and reactor-steam generator demand to zero.

   NOTE:

   This insures that SG-feedwater is being controlled at minimum level.
8. Close the letdown isolation valve. Monitor makeup tank level, feed as necessary from the coolant storage system to maintain level in the normal operating range.

   NOTE:

   If for any reason the cooldown is interrupted, open the letdown isolation valve and adjust letdown flow rate to compensate for any changes in system temperature and RCP seal in leakage.
9. Gradually reduce reactor power to 10% while monitoring and plotting cooldown rate.
10. Trip the main turbine generator output breakers when load has been reduced to approximately 5% (42 MWE).

11. Trip the generator exciter breaker.

NOTE:

Perform turbine overspeed trip testing if required.

12. Perform the following:
    a. Start the AC bearing oil pump.
    b. Start the bearing lift pump.
    c. Start the seal oil backup pump.
    d. Trip the turbine.
    e. Observe turbine header pressure to insure the turbine bypass valves are operating to control header pressure at 885 psig.
13. Place rod control panel in manual and gradually decrease reactor power to 10^{-8} amps while monitoring and plotting cooldown rate. Log the following when at 10^{-8} amps on the IR.
    a. Rod configuration
    b. Boron concentration
    c. Time
    d. Tave

NOTE:

While decreasing power to 10^{-8} amps, do not exceed a cooldown rate of 140 F/hr and maintain pressurizer level at normal level (220"). If shutdown is to hot shutdown only, allow pressurizer level to decrease to 100".

14. If shutdown is preparatory to cooldown insert regulating rods and leave safety rods withdrawn.
15. If shutdown is to hot shutdown only, all rods may be inserted, the turbine bypass valves placed in manual, the reactor tripped, and the RCS borated to value in Figure 2 by time in Figure 3.
16. As neutron flux decreases note source range is energized at 5×10^{-10} amps $\pm$ .5 decades on the intermediate range.
17. With RC temperature greater than 525 F and the reactor is greater than 1% shutdown, the plant is in a hot shutdown condition.
18. When turbine speed is 0 RPM, start the turning gear motor.

FIGURE 1 PRESSURIZER LEVEL VS RC TEMPERATURE

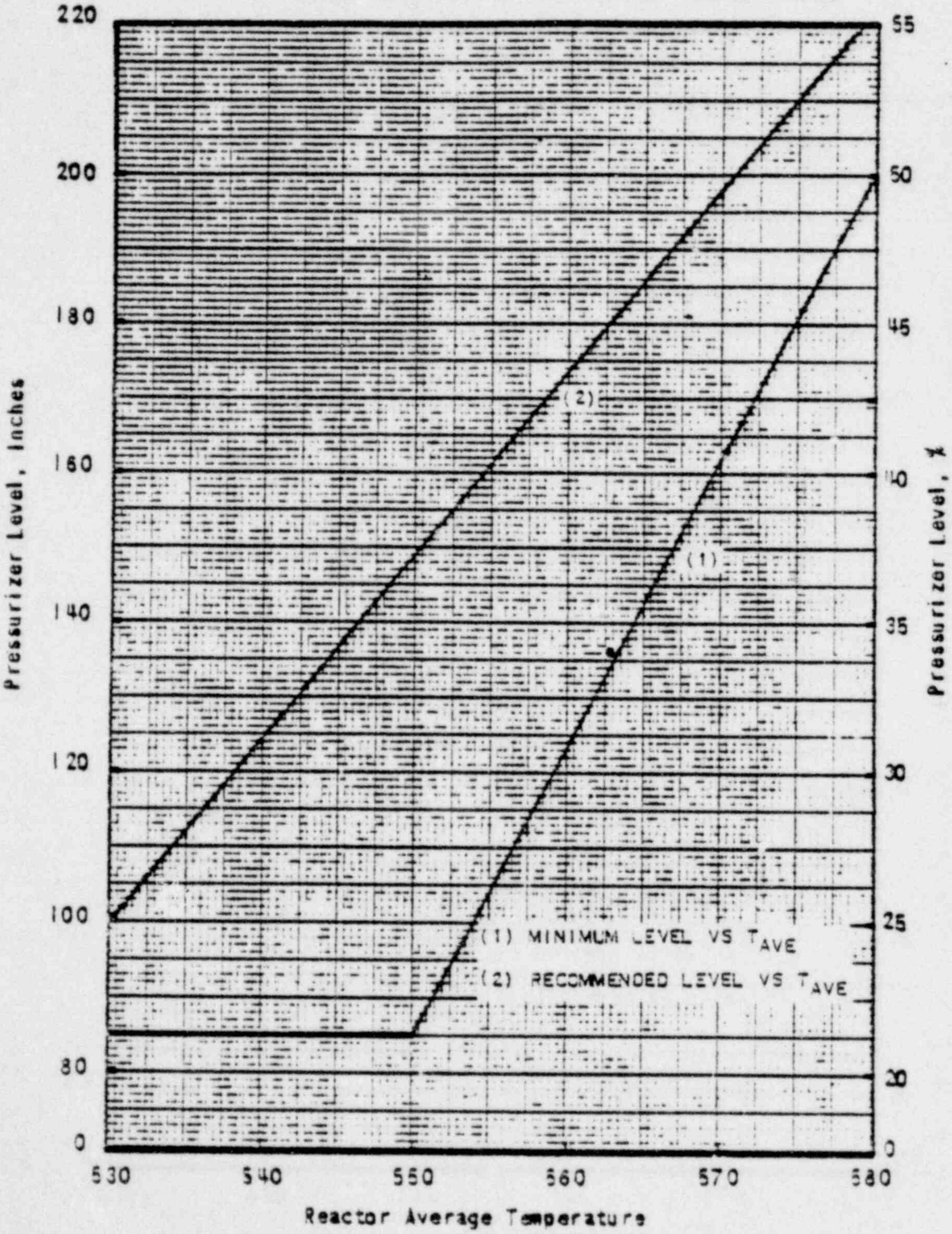


FIGURE 2 SHUTDOWN REQUIREMENTS SAFETY RODS OUT, NO XENON

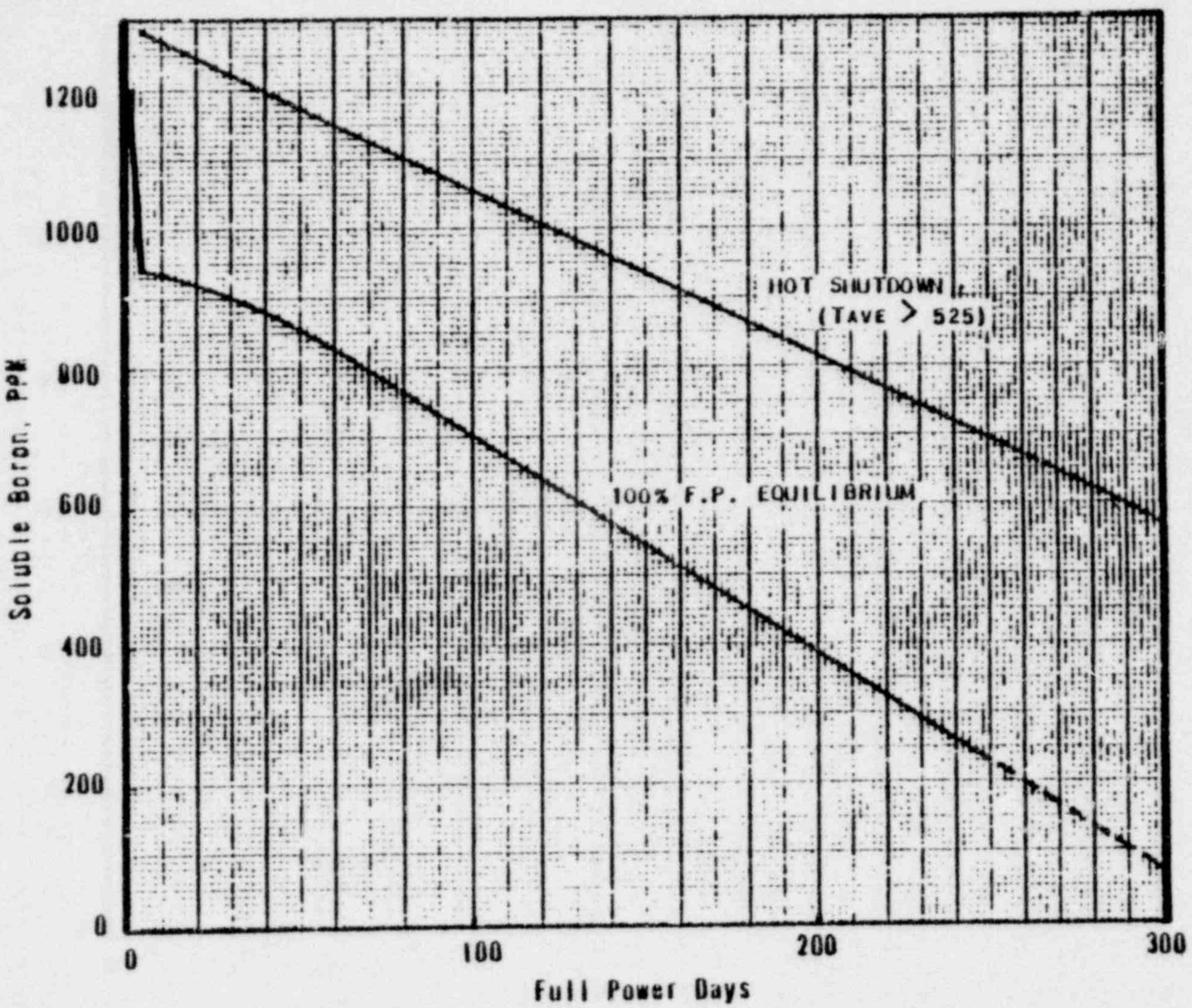
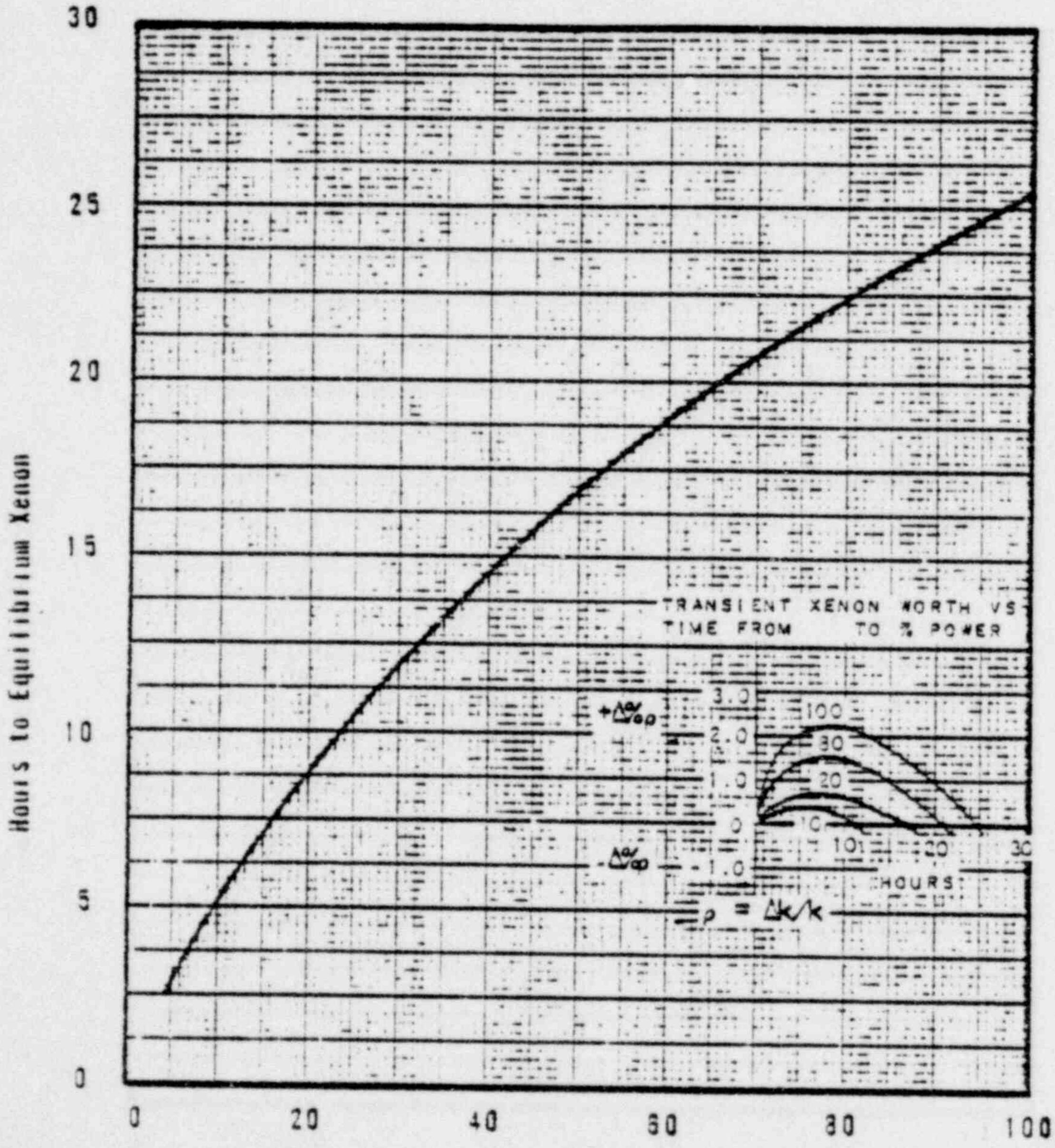


FIGURE 3



Equilibrium Percent Power Before Shutdown

III. LIMITS AND PRECAUTIONS

1. At steady state conditions (equilibrium xenon), if $\rho_{(net)}$ exceeds $\pm$ 0.3 % Δ k/k, analyze for discrepancy.

2. With transient xenon conditions, if $\rho_{(net)}$ exceeds $\pm$ 0.6% Δ k/k, analyze for discrepancy.

IV. PROCEDURE

1. Reactor Shutdown (Reference Condition 532°F, No Rods, No Boron)

   a. Estimated Critical Position (ECP)

      Method:

      Set $\rho_{(net)}$ = 0 and, knowing the value of reactivity associated with each of the components in the reactivity balance equation except control rods, solve for the reactivity worth of inserted control rods necessary to maintain just critical conditions. Complete Worksheet I for determination of Estimated Critical Position.

   b. Estimated Critical Boron Concentration (ECB)

      Method:

      Set $\rho_{(net)}$ = 0, assume the desired critical rod height, and knowing the reactivity value associated with the other components, solve for critical boron worth. Complete Worksheet II for determination of Estimated Critical Boron Concentration.

   c. Shutdown Margin for Hot Shutdown 532°F.

      Method:

      In accordance with Technical Specification and shutdown margin definition, the $\rho_{(net)}$ is determined using the rod worth for Groups 1 through 7 fully inserted and the reactivity addition associated with the worst case stuck rod. Complete Worksheet III for determination of shutdown margin for hot shutdown condition.

   d. Shutdown Margin for Cold Shutdown 70°F.

      Method:

      In accordance with Technical Specification and shutdown margin definition, the $\rho_{(net)}$ is determined using the rod worth for Groups 1 through 7 fully inserted and the reactivity addition associated with the worst case stuck rod. Complete Worksheet IV.

2. Shutdown Margin With Inoperable Rod

   a. Calculate the shutdown margin in accordance with the method of Section 1.c. of this procedure if the reactor is at hot shutdown or if the reactor is at power. Record the calculated value of shutdown margin on Data Sheet II.

b. Determine the maximum reactivity worth of the inoperable control rod using Step 7. Record this value on Data Sheet for operating with an inoperable control rod. If the rod is movable or trippable, enter zero.

c. Sum the value obtained in Step 2.b. and 1.00% $\Delta k/k$.

d. The value obtained in Step 2.a. must be greater than the value obtained in Step 2.c. If not, refer immediately to the "Action Statement" of Technical Specification.

3. Reactivity Balance at Power (for Reactivity Anomaly)

Reference Condition 582°F. In this calculation, the plant is assumed to be at equilibrium conditions. The reactivity effects of all the components are added together. The desired sum is zero (when $K = 1$, $\rho = 0$). However, an acceptable value for $\rho_{(net)}$ is $0 \pm 0.3\%$ $\Delta k/k$. The value of $\rho_{(net)}$ MUST be within $0 \pm 1\%$ $\Delta k/k$ to be in accordance with Technical Specifications. Complete Worksheet V.

4. Comparison of Overall Core Reactivity Balance to Predicted Values

a. Calculate the reactivity balance using the method of Section 3 of this procedure. Record the computed value on Data Sheet I.

b. Forward Data Sheet I with Worksheet V attached, to the Reactor Engineer.

c. Any adjustments made to the curves utilized to calculate reactivity balance shall be described by the Reactor Engineer on Data Sheet for comparison of reactivity balance to predicted value.

1102 11

OFR UNIT #1

PLANT COOLDOWN

OP 1102 11

1. PURPOSE

This procedure provides the necessary steps to take the plant from hot shutdown to a cold shutdown condition. Greater than 280°F, cooldown will be performed using the steam generators and below 280°F, cooldown will be performed using the decay heat system.

2. REFERENCES

2.1 Technical Specifications

2.2 Limits and Precautions

3. LIMITS AND PRECAUTIONS

3.1 The cooldown rate of the RC system shall not exceed 100°F and shall be within the limits of Figure 3.

TS 3.2 The pressurizer spray shall not be used if the ΔT between the RC system and pressurizer is greater than 410°F.

TS 3.3 Maintain RC pressure/temperature relation within the limits of Figure 1.

3.4 The average shell temperature of the OTSG shall not be 100°F above or below RC temperature.

3.5 All subcritical boron changes in the RC system will be verified at least every predicted 30 ppm.

3.6 Following a significant change in the boron concentration of the RC, pressurizer spray should be operated to equalize the concentration in the RC loops and pressurizer.

3.7 No more than 3 RC pumps may be operated below 500°F.

3.8 Control rods will not be operated unless pressure temperature conditions are above curve "O" of Figure 1.

3.9 Safety Groups 1 and 2 will be withdrawn 100% or the reactor coolant system at shutdown boron (see Figure 2) to insure that an adequate shutdown margin can be maintained during the temperature transient.

3.10 Prior to placing the decay heat system in operation, open and tag out the breakers for two of the makeup pumps to prevent overpressurization of the decay heat system.

3.11 When RC pressure is less than 700 psig, the core flood tank block valves will be shut and the motor control breakers opened and tagged.

3.12 When RC temperature is greater than 200°F, a minimum flow of .75 gpm must be maintained through the pressurizer spray line.

TS 3.13 One OTSG shall be operable when RC system average temperature is greater than or equal to 250°F.

TS 3.14 Maximum cooldown rate for the pressurizer is 100°F/Hr.

4.0 PLANT COOLDOWN

4.1 Initial Conditions:

		INITIALS
4.1.1	RC system is at hot shutdown per OP 1102-10.	________
4.1.2	At least one RC pump is running in each loop.	________
4.1.3	Calculation to compensate for contraction and boration have been completed per OP 1103-04.	________
4.1.4	Safety Rod Groups 1 and 2 are 100% withdrawn.	________
4.1.5	OTSG level control at low level limits is in automatic with one feed train in operation.	________
4.1.6	Pressurizer level is being controlled at 100" in auto.	________
4.1.7	RC system degassing is complete.	________
4.1.8	Main steam pressure is being controlled at 885 psig with turbine bypass valves in auto.	________
4.1.9	Auxiliary boiler is in operation.	________
4.1.10	Insure RC system and makeup water are within limits of plant chemistry manual.	________
4.1.11	Flow path to the makeup tank from the borated water source to be used has been established.	________

.0 PLANT COOLDOWN (Continued)

4.2 Procedure:

4.2.1 Establish one RC pump running in each loop per OP 1103-06. ________

4.2.2 Close letdown isolation valve. ________

4.2.3 Establish a flow path to makeup tank from borated water source determined in 4.1.3. ________

4.2.4 Gradually increase the pressurizer level to normal - 220" - while maintaining makeup tank level between 55" and 86". ________

4.2.5 Cycle pressurizer heaters and spray valve as necessary to control RC pressure within limits of Figure 1. ________

4.2.6 Assign the following parameters to pen recorder and monitor:

Tc wide range
RC pressure narrow
Pressurizer temperature
Heatup/cooldown rate
RC pressure wide range ________

4.2.7 Manually open turbine bypass valves to commence cooldown. ________

4.2.8 Place FOGG logics in block when steam pressure is approximately 500 psig. ________

4.2.9 When RC pressure is between 1900 and 1850, perform the following:

(a) Terminate cooldown and stabilize RC temp. ________

(b) If the reactor vessel head is to be removed, insert Group 8 to in-limit ________

(c) Insert Safety Groups 1 and 2 to their in-limit per OP 1105-09. ________

(d) Verify turbine bypass in manual. ________

(e) Trip the reactor. ________

(f) Depressurize RC system to 1650 psig. ________

(g) Reset the high flux trip to 5% full power. ________

(h) Initiate shutdown bypass and reset reactor trip bistables. ________

(i) Reset reactor trip, Latch and withdraw safety groups 1 and 2 to 100% per OP 1105-09. ________

(j) Place ES actuation channels one and two and high pressure injection in bypass by depressing HP Injection Bypass pushbutton. ________

4.2.10 Re-establish Cooldown:

(a) Maintain RC system pressure and temperature within limits of Figure 1. ________

(b) Maintain RC system cooldown $\leq 100^{\circ}F/Hr$. ________

4.2.11 When OTSG Pressure decreases to 500 psig, perform the following:

(a) Line up the feed and condensate system to bypass the main feedwater pumps per OP 1106-02. ________

(b) Depress the turbine-driven emergency feedwater pump stop switch and stop the main feedwater pump per OP 1106-02. ________

NOTE:

A condensate pump is now supplying feedwater to the OTSG's.

4.2.12 Prior to OTSG pressure decreasing below 250 psig, place the auxiliary steam boiler in service and close main steam supply to auxiliary steam header. ________

4.2.13 At 700 psig RC pressure, close core flood tank isolation valves and tag breakers. ________

4.2.14 Between 500 and 550 psig RC system pressure, bypass ES channels 3 and 4, low pressure injection. ________

4.2.15 When RC system pressure reaches 500 psig, reset pilot actuated relief setpoint to 550 psig. ________

4.2.16 When boron concentration of the reactor coolant system is at or above shutdown boron concentration, Figure 2, insert safety rods and trip the reactor. ________

4.2.17 When RC temperature is less than 300°F, take manual control of the startup feedwater control valves and gradually increase OTSG level to between 288" and 299" (96% to 99%) on the operate range instruments being careful not to exceed cooldown limits. ________

4.2.18 When RC temperature is less than 280°F and RC pressure is less than 250 psig, perform the following:

(a) Stop the RC pump in loop B and start the non-operating pump in loop A per OP 1103-06. ________

(b) Open and tag the breakers for the two non-operating makeup pumps. ________

(c) Place the nuclear services raw water and nuclear services cooling water system into operation for decay heat removal per OP 1104-09. ________

(d) Assign DH cooler outlet temperature to pen recorder and use this temperature to determine RC system cooldown rate. ________

4.2.19 When DH flow has been established, close feedwater block valves. ________

4.2.20 Stop operating RC pumps per OP 1103-06. ________

4.2.21 Close pressurizer spray valve SV and open auxiliary spray valve SV as required to control RC pressure and pressurizer cooldown. ________

4.2.22 Control cooldown of reactor coolant system by adjusting bypass flow around the decay heat coolers per OP 1104-04. ________

4.2.23 When main condensers are at atmospheric pressure, secure gland steam and gland exhaust for the main turbine. ________

4.2.24 Secure condensate and feedwater system per OP 1106-02. ________

4.2.25 Secure condenser circ water system per OP 1104-12. ______

4.2.26 At 30 psig RC pressure start adding nitrogen to the pressurizer to maintain 30 psig while quenching the steam bubble. ______

4.2.27 At 150°F RC temp, secure letdown and makeup pump. ______

4.2.28 At 140°F cooldown is considered complete.

4.3 Final Conditions:

4.3.1 The RC system is in cold shutdown conditions.

4.3.2 Pressurizer is blanked with nitrogen at approximately 30 psig and level is approximately 100 inches.

4.3.3 The decay heat system is operating for decay heat removal.

4.3.4 The nuclear services raw water and cooling water systems are operating as required.

4.3.5 The steam generators are in pre-heatup level.

OFR # 1

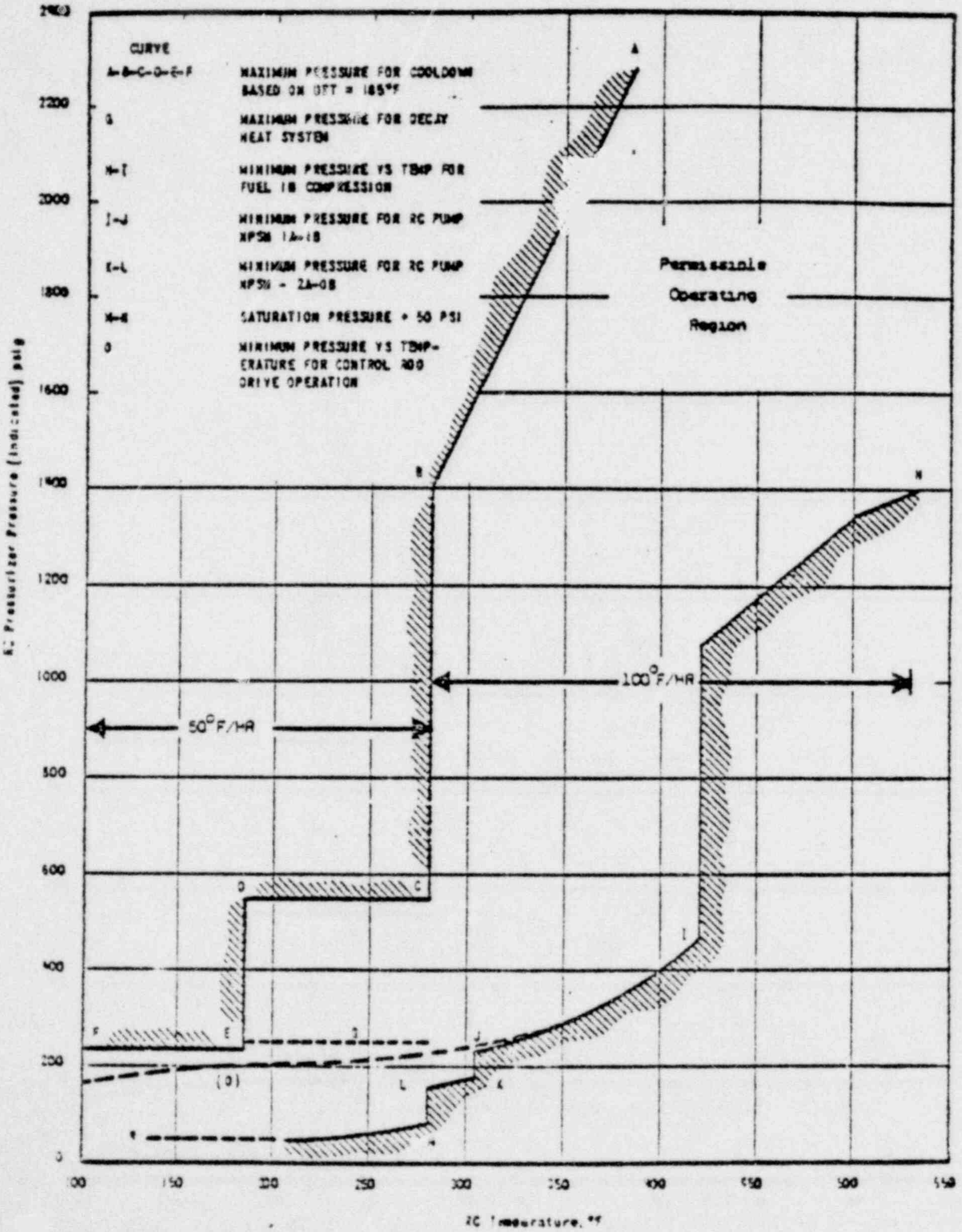
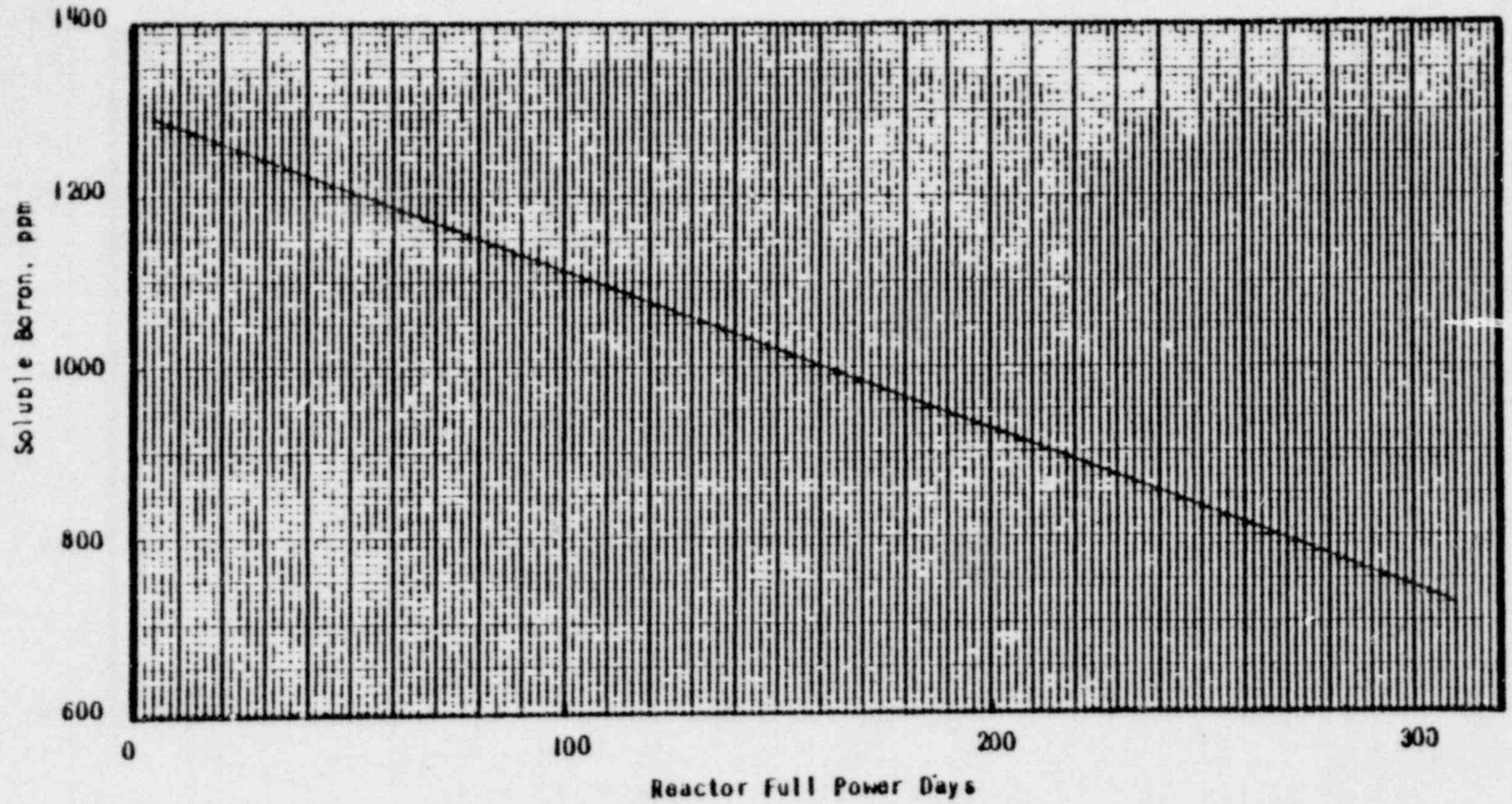


FIGURE 1

FIGURE 2 SHUTDOWN BORON REQUIREMENTS (T_{AVE} BELOW 200°F)
VS REACTOR FULL POWER DAYS



OFR #1 1102 11

1103 04

OFR UNIT #1

SOLUBLE POISON CONCENTRATION CONTROL

OP 1103.04

1. PURPOSE

1.1 This procedure describes the methods for changing and controlling the boron concentration in the reactor coolant system and provides equations used to determine time and volume requirements for boration, deboration, and dilution.

1.2

Operation	Section
Feed and Bleed	4
Boration	5
Deboration	6
Makeup	7

2. DESCRIPTION

2.1 Soluble poison (boric acid dissolved in the coolant) is used as a means of reactivity control. Relatively slow reactivity changes are compensated by increasing or decreasing the amount of boron in the reactor coolant.

2.2 Boron concentration changes in the reactor coolant are required under the following conditions:

1. Deboration from shutdown concentration (refueling, maintenance, or normal shutdown) to that required to approach criticality.
2. Boration following shutdown for refueling, maintenance, cooldown, etc. to assure a minimum of one percent shutdown margin.
3. Boron concentration changes during normal operation to maintain the recommended control rod pattern. This will compensate for fuel depletion and fission product poisons.

4. Feed to the MU system is required to make up for minor leakage and sampling. Metered volumes of concentrated boric acid solution and demineralized water may be required to provide this makeup.

2.3 Boric acid solution or demineralized water or a combination of both is supplied to the makeup system from the chemical addition system and the coolant storage system.

2.4 Manual feed may be initiated:

a. When the three-way valve is in the normal position.

b. If also feeding from demineralized water or bleed holdup tank(s) via feed control valve, when the batch controller has been set for a batch size start/stop switch is in start.

2.5 Manual feed will be terminated:

a. When the MU block valve (SV34) is manually closed.

b. When batch size shutdown occurs or the batch start/stop switch is positioned to stop if also feeding via feed control valve from demineralized water or bleed holdup tanks.

2.6 Manual bleed may be initiated:

a. When the makeup tank level is above the low-low level interlock setpoint.

b. When the feed block valve (SV-34) is closed.

2.7 Manual bleed will be terminated:

a. When the three-way valve is positioned to MU tank by the operator.

b. When the makeup tank level is below the low-low level interlock setpoint.

2.8 Continuous Feed and Bleed may be initiated when all of the following are are satisfied:

a. MU tank level is above the low-low level alarm and interlock setpoint.

b. The batch controller start/stop switch is in the start position.

c. The batch size selected is greater than batch total.

d. Safety rod groups 1 through 4 are fully withdrawn and Group 5 is less than 80% withdrawn
or
Rod group 6 is withdrawn greater than 95%.

2.9 Continuous Feed and Bleed will be terminated:

a. When any one of the items in 2.8 above are not satisified or when the operator positions the 3 way valve (S 30) to the MU Tank or closes the MU Feed Block valve (SV-34).

2.10 The following definitions apply to this procedure:

1. Boration - addition to the RCS of borated water of a higher concentration than that of the RCS from either the BA mix tank or the bleed holdup tanks.

2. Dilution - Reduction of RCS boron concentration by makeup from demineralized water tank or bleed holdup tanks having a lower concentration than the RC system.

3. Deboration - Removal of boric acid by deborating demineralizers, the effluent of which is returned to the RCS. This method is used when the RCS concentration is less than 180 ppmb.

2.11 Enclosures

1. Enclosure 1 provides equations and examplies for determining volume requirements for the various operations.

3. LIMITS AND PRECAUTIONS

3.1 Assure adequate supply of boric acid solution and/or demineralized water is available prior to initiating boration or dilution cycle.

(TS)3.2 If the reactor is critical, assure that minimum volumes and concentrations of boric acid solutions are maintained per technical specifications.

3.3 The CRA safety groups will always be at their upper limits prior to boron dilution.

3.4 Prior to any operation which will change the boric acid concentration of the RC system, the time required to make the change should be determined. If the operation proceeds for a significantly lower time than calculated or normally observed, terminate process until cause is determined and corrected.

3.5 Stop boration or dilution if MU system operation is disrupted. (Loss of letdown flow, loss of makeup pump, ES actuation.)

3.6 Stop boration or dilution immediately if control rod group position indication, neutron count rate, or other reactivity indications are behaving in an erratic or unexpected manner.

3.7 When manually terminating feed to the MU system, secure the pump, being used to feed, prior to shutting the feed block.

(TS)3.8 The RC system, will not be deborated unless a DH, or RC pump is operating.

3.9 When changing RC boron concentration while shutdown insure the reactor is $\geq$1% shutdown.

3.10 The pressurizer boron concentration shall not differ from RC system boron concentration by more than $\pm$ 100 PPM.

3.11 While changing RC system boron concentration, boron samples will be taken every predicted 30 PPM change when the reactor is subcritical.

3.12 Do not allow Group 6 to become <75% or >95% withdrawn, as a result of the addition of the make up solution.

4. DILUTION USING FEED AND BLEED METHOD

4.1 Inital Conditions

1. The MU system is operating.
2. Deboration is required for one of the reasons specified in Section 2.2.
3. Feed and Bleed(contiuous dilution) permit requirements are satisified (see Section 2.8).
4. The RC System is in a hot shutdown condition or at some power level depending on the situation.

4.2 Procedure

*1. Sample source of demineralized water to determine that it meets the water chemistry manual specifications.
2. Complete the data sheet.
3. Using the data calculate the volume of feed required or use curves and verify that an adequate supply of demineralized water is available and adequate storage space is available in the bleed holdup tank(s) for the bleed volume.
4. Based on the Feed and Bleed rates, estimate the time required to make the changes. Use this time in conjunction with Limit and Precaution 3.4.
5. Setup the batch controller as follows:
   a. With the start/stop switch in the stop position, press "clear" and observe that the total registers return to zero and place switch in start position.
   b. Using the feed calculated or from curves set the batch size thumb wheels.
   c. Set the pre shutdown batch size to some value higher than the batch size as this feature at the controller is not required.
   d. Line up demineralized water system valves and set flow control valve (SV-56) control levels to approximately mid position.
6. Establish a flow path from 3 way valve to bleed tank(s)
7. Open makeup valve (SV-34), start demineralized water pump(s), and position 3 way valve (S-30) to bleed tank position.
8. Regulate feed flow control valve (SV-56) and the letdown flow control valve if required to obtain the desired feed rates and maintain MU tank level in the normal operating range.

9. If change in the concentration is to be 100 ppm or greater, but pressurizer heater in manual as required to open the spray valve.
10. Sample RC system every estimated 30 ppm change if reactor is subcritical.
11. When the batch total equals preset batch size the flow control valve (SV-56) and feed block valve (SV-34) will close and the 3-way valve (S-30) will return to the MU tank position.
12. Stop the DW pumps and return system valves to normal arrangement.
13. Sample RC system approximately one hour after feed in terminated to verify new boron concentration.

5. BORATION USING FEED AND BLEED OR BATCH METHOD

5.1 Inital Conditions

1. Boration is required for one of the reasons specified in Section 2.
2. The MU system is operating.
3. Feed and Bleed (continious boration) permit requirements must be satisified or the batch method (alternate Feed and Bleed) must be used.

5.2 Procedure

1. Sample the RC system, MU system and the source(s) of boric acid solu ion to determine the concentrations.
2. Determine which of the following methods is to be used.
   a. Feed and Bleed.
   b. Batch.
   c. Cooldown - Feed at rate sufficient to maintain MU tank level within normal limits. Do not bleed.
   *d. Direct Feed - Direct from boric acid mix tank to MU system via spare line. This is a redundent flow path in the event the normal line can not be used due to blockage etc.
3. Complete data sheet .
4. Calculate the volume of solution being used to obtain the desired concentration by using the appropriate curves. Verify an adequate supply of solution is available and adequate storage space is available for bleed if required.
5. Setup batch controller per 4.2.5 a through c.
6. Establish a flow path to the MU system from the source(s) of feed to be utilized and establish a flow path from the 3 way valve to the bleed tank(s) being used.
7. Open makeup valve (SV-34), start boric acid pump(s) or bleed transfer pump(s) depending on source being used.
8. If bleed transfer pumps are utilized, flow can be regulated with the feed flow control valve (SV-56). If boric acid pumps are used, gross flow control can be obtained by running 1 or 2 pumps. Regulate letdown flow as required to compensate for the desired feed rate.
9. Estimate time required to change the concentration by 30 ppm and sample at that interval if the reactor is subcritical.

10. When the desired volume is added or automatically terminated by the batch controller, stop the pumps being used and return system valves to normal arrangement.
11. Sample RC system and MU tank approximately one hour after feed is terminated to verify the desired concentration is obtained and equilibrium concentration between RC and MU is achieved.

6. DEBORATION USING DEBORATING DEMINERALIZERS

6.1 Initial Condition

1. The RC system boron concentration is 180 ppm or less.
2. MU system is operating
3. Feed and Bleed (continious dilution) permit requirements are satisified. (See Section 2.8.)

6.2 Procedure

1. Complete data sheet.
2. Calculate the volume to be circulated via the deborating demineralizer by using the appropriate curves or calculations
3. Based on the desired letdown flow rate, estimate the time required to make the desired concentration change.
4. Set up the batch controller per 4.2.5 a through c.
5. Verify bleed tank inlet valves are closed.
6. Establish a flow path through the deborating demineralizer to be used by opening SV-32 or 33.
7. Open the makeup valve SV34 and position the 3 way valve to the bleed tank position.
8. Sample RC system.
9. When the flow is terminated either automatically or manually, by closing SV-34 and positioning the 3 way valve to the normal position (MU tank), close the demineralizer inlet valves.
10. Sample the RC and MU systems approximately one hour after deboration has been terminated to verify the desired concentration is achieved, and the MU and RC system are in equilibrium.

7. RC SYSTEM MAKEUP (Batch Method)

7.1 Initial Conditions

1. Makeup to the RC system is required due to minor leakage and sampling or to compensate for coolant contraction during cooldown.
2. The MU system is operating.

7.2 Procedure

1. Sample the source(s) of makeup to be utilized (bleed tank(s), boric acid mix tank) to determine their concentration and if demineralized water is to be utilized determine chemistry is within specification.
2. Complete Data sheet.
3. Calculate the total batch size required and the batch size for each ingredient to be utilized for makeup purposes.
4. Clear the batch total register and set up a batch size for the source with the highest boron concentration.
5. Arrange valves to provide a flow path from the source to the MU system.
6. Start the pump(s) from the source being added to the system. If the bleed transfer pumps are being utilized flow can be regulated with MU flow control valve (SV-56).
7. When the desired batch is added, stop the pump(s) being operated.
8. Perform steps 5 through 7 for the other source(s) of makeup.
9. NOTE: If required, alternately feed the from borated source and the demineralized water source to maintain the normal rod pattern.

Enclosure 1

1. Boration During Cooldown

1.1 RCS volume change - makeup required during cooldown to maintain constant Pressurizer Level.

$$-\Delta V = \left[\frac{\nu_{cc}}{\nu_{ch}} - 1\right] V_r \qquad (1)$$

$-\Delta V$ = contraction of reactor coolant, ft^3, cold.

ν_{ch} = specific volume of reactor coolant at hot operating conditions, ft^3/lbm.

ν_{cc} = specific volume of reactor coolant at cold shutdown conditions, ft^3/lbm.

V_r = volume of reactor coolant system at normal pressurizer level, ft^3.

1.2 Required Feed concentration to obtain final shutdown concentration in RCS.

$$Cf = \frac{\left(\frac{Vr}{vcc} + \frac{Vmu}{vmu}\right) Ccc - \left(\frac{Vr}{vch} + \frac{Vmu}{vmu}\right) Cch}{\Delta V/vcc} \qquad (2)$$

Cf = average concentration of feed solution, ppmB.

Vmu = volume of makeup system, ft^3.

vmu = specific volume of makeup system, ft^3/lbm.

Ccc = boron concentration in RCS at cold shutdown, i.e., final concentration, ppmB.

Cch = boron concentration in RCS at start of boration, i.e., initial concentration, ppmB.

vf = specific volume of feed solution, ft^3/lbm.

$$F = \frac{\Delta V vf}{vcc} \qquad (2a)$$

F = volume of feed required, ft^3.

1.3 Required combination of solutions to obtain volume F at concentration Cf.

A. Concentrated Boric Acid plus Demineralized Water

$F = B+D$ (3)

F = volume of feed solution, ft^3.

B = volume of concentrated boric acid solution, ft^3.

D = volume of demineralized water, ft^3.

$FCf = BC_b + D$ (4)

C_b = Concentration of solution B, ppmB.

If D solves out negative, solution B is not strong enough, i.e., C_b must be > Cf.

B. Combine Two Boric Acid Solutions.

$FCf = BCb + B' C'_B$ (5)

2 FEED AND BLEED

2.1 Boration using Feed and Bleed

$$Mf = M_r \, Ln \left(\frac{Cf - Cri}{Cf - Crf} \right) \qquad (6)$$

$$Mf = F/vf \qquad (7)$$

$$M_r = \frac{Vr}{vr} + \frac{Vmu}{vmu} \qquad (8)$$

M_r = Mass of reactor coolant - makeup systems, lbs.

Vr = Volume of RCS at normal pressurizer level, ft^3.

vr = specific volume of reactor coolant at conditions extant at time of boration, ft^3/lbm.

Vmu = volume of makeup system, ft^3.

vmu = specific volume of makeup system, ft^3/lbm.

F = volume of feed solution, ft^3.

vf = specific volume of feed solution, ft^3/lbm.

Cf = concentration of feed solution, ppmb.

Cri = initial RCS concentration, ppmb.

Crf = desired RCS concentration, ppmb.

2.2 Dilution using Feed and Bleed.

$$Mf = Mr \, Ln \left(\frac{Cri}{Crf} \right) \qquad (9)$$

Mf = Mass of demineralized water required, lbs. (see equation 7)

2.3 Deboration using Feed and Bleed

$$Mf = Mr \, Ln \left(\frac{K-Cri}{K-Crf} \right) \qquad (10)$$

K = Concentration of deborating demineralizer effluent, ppmb.

---

3 BATCH OR MAKEUP

3.1 Boration

$$\frac{FCf}{vf} + \left(\frac{Vr}{vr} + \frac{Vmu}{vmu}\right) Cri = \left(\frac{Vr}{vr} + \frac{Vmu}{vmu} + \frac{F}{vf}\right) Crf \qquad (11)$$

Terms as defined previously.
Vmu must be reduced by amount that Mu tank inventory is below normal.

3.2 Dilution

Remove Cf from Equation (11).

3.3 Deboration

Substitute X for Cf in Equation (11).
X = Concentration of deborating demineralizer effluent, ppmb.

For repetitive batches, repeat calculation for each batch, using Crf as sampled for Cri in the succeeding calculation, however, use of equation (11) will provide close estimate of total batch size required.

4 EXAMPLES

4.1 Reactor Coolant System is to be shutdown and cooled down to 140F from 2155 psig and 532F. Reactor coolant will be borated from 40 ppmb to a refueling concentration of 2275 ppmb.

The following solutions and amounts of each are available, all held at a temperature of 110F.

a. Demineralized water: 10,000 cu.ft.

b. 7 wt. % Boric Acid solution: 1450 cu. ft.

c. Excess in BWST available for other than filling canal: 1,000 cu. ft. @2275 ppmb.

Assume the following system parameters:

$Vr = 10,200\ ft^3$.

$Vmu = 300\ ft^3$.

$vmu = .01620\ ft^3/lbm.$ @120F

$vch = .02085\ ft^3/lbm.$

$vcc = .01630\ ft^3/lbm.$ @140F

<u>CONTRACTION</u>:

From Eq. (1) $\Delta V = \left(\frac{.01630}{.2085} - 1\right) 10,200$

$\Delta V = \underline{2226 ft^3}$ (at conditions of vcc)

<u>Required Feed Concentration</u>:

From Eq. (2)

$$CF = \frac{\left(\frac{10,200}{.01630} + \frac{300}{.01620}\right) 2275 - \left(\frac{10,200}{.02085} + \frac{300}{.01620}\right) 40}{2226/.01630}$$

$Cf = \underline{10,585 ppmb}$

4.1 (cont'd)

FEED VOLUME:

From Eq. (2a) $F = \frac{(2226)\ (.01616)}{.01630}$

$F = \underline{2207\ ft^3.}$

10,585 ppm

Using 7wt% and Demineralized water:

From Eq. (4) $D = \frac{F\ (Cf - Cb)}{1 - Cb}$ 7wt% = 12250 ppmb.

$D = \frac{2207\ (10585 - 12,250)}{1 - 12,250}$

$D = \underline{300\ ft^3}$ Demin. water

From Eq. (3) B = 2207 - 300

B = 1907 ft^3 of 7 wt% Boric acid solution*

*Solution will have to be fed in two batches due to size of tank (1800 cu. ft.), or use following:

Using 7wt% and BWST @2275 ppmb

From Eq. (5) $B' = \frac{F\ (Cf - Cb)}{C'b - Cb}$ Cb = 7 wt% (12250 ppmb) C'b = 2275 ppmb

$B' = \frac{2207\ (10,585 - 12,250)}{2275 - 12,250}$

$B' = \underline{1839\ ft^3}$ of 7 wt% solution?.

$B' = \underline{368}\ ft^3$ of BWST solution

$FC_F = BC_b + B'C_b'$

$C_1V_1 + C_2V_2 = C_FV_F$

1839(12250) + (368)(2275) = (2207)(10,585)

22,527,75 + 837200

30,889,75 = 23,205,000

4.2 Borate RCS at operating conditions from 450 to 550 ppmb using Feed & Bleed, with 7 wt% solution.

From Eq. (8) $Mr = \frac{10,200}{.02085} + \frac{300}{.01620} = 507,727$ lbs.

Pr = 2155 psig
Tav = 532F

From Eq. (6) $Mf = 507,727 \; Ln\left(\frac{11,800}{11,700}\right)$

Mf = 4321 lbs.

From Eq. (7) $F = (4321)\ (.01616) = \underline{\underline{70\ ft^3}}$

4.3 Batch dilute RCS at operating conditions from 450 ppmb to 425 ppmb. Batch size limit is ~150 Ft^3. How many batches required?

Pr = 2155 psig
Tav = 579 F (vr = .02233ft^3/lbm)

From Eq. (11) $$F = \frac{\left(\frac{Vr}{vr} + \frac{Vmu}{vmu}\right)\left(Crf - Cri\right)\ vf}{1 - rf}$$

$$F = \frac{\left(\frac{10,200}{.02233} + \frac{300}{.01620}\right)\left(425 - 450\right)\quad .01616}{1 - 425}$$

F = 453 ft.^{3} demineralized water

Use three batches. Terminate third batch at 100 ft.^{3} and sample RCS before continuing.

FCP-BC 15
D. FCP-BC

5. GRAPHS

5.1 RCS Volume vs. Temperature — Figure 5.1

1. Equation: 1.1 (1)
2. Bases: Vr= 10,200 $ft.^3$ at nominal pressurizer level. Cooldown curve optimized for straightest path between upper and lower temperature and pressure limits. Specific volumes from ASME steam tables.
3. Read makeup volume required to maintain nominal pressurizer level on "ΔV" line at bottom. Calculated contraction between 532°F and 140°F is 2226 cu. ft.

5.2 Boration for Cold or Refueling Shutdown — Figures 5.2 (a-d)

1. Equation: 1.2 (2)
2. Bases: Vmu = 300 cu. ft.
   Shutdown concentrations of 1800 and 2275 ppmb.
   Shutdown temperature of 140°F.
3. On cold or refueling shutdown curves, read required feed concentration versus existing RCS concentration at time of shutdown. Note: Contraction is based on cooldown from 532°F. Any changes in boron concentration between 579°F and 532°F must be calculated prior to using this curve. Required feed concentration is the average concentration of the makeup volume required to achieve 1800 or 2275 ppmb.

5.3 Required Solution Combination to Obtain Average Shutdown Feed Concentration — Figure 5.3

1. Equation: 1.3 (3), (4)
2. Bases: Available concentrated boric acid solution of 12,250 ppmb (7 wt.%).
3. Obtain average required feed concentration from figures 5.2 (a-d). At Cf on figure 5.3, read horizontally to lines for B and D. Read B and D in gallons or cubic feet vertically below where they cross Cf.

   Note: Lines are drawn for three makeup volumes, the middle line is the one usually used.

5.4 Feed and Bleed - Boration — Figure 5.4

1. Equation: 2.1 (6),(7),(8)
2. Bases: Cf available is 12,250 ppmb
   RCS conditions 2155 psig and 579°F.
3. At initial concentration, read upward to diagonal which starts at final concentration. Where the vertical from initial intersects the diagonal from final, read volume required at left.

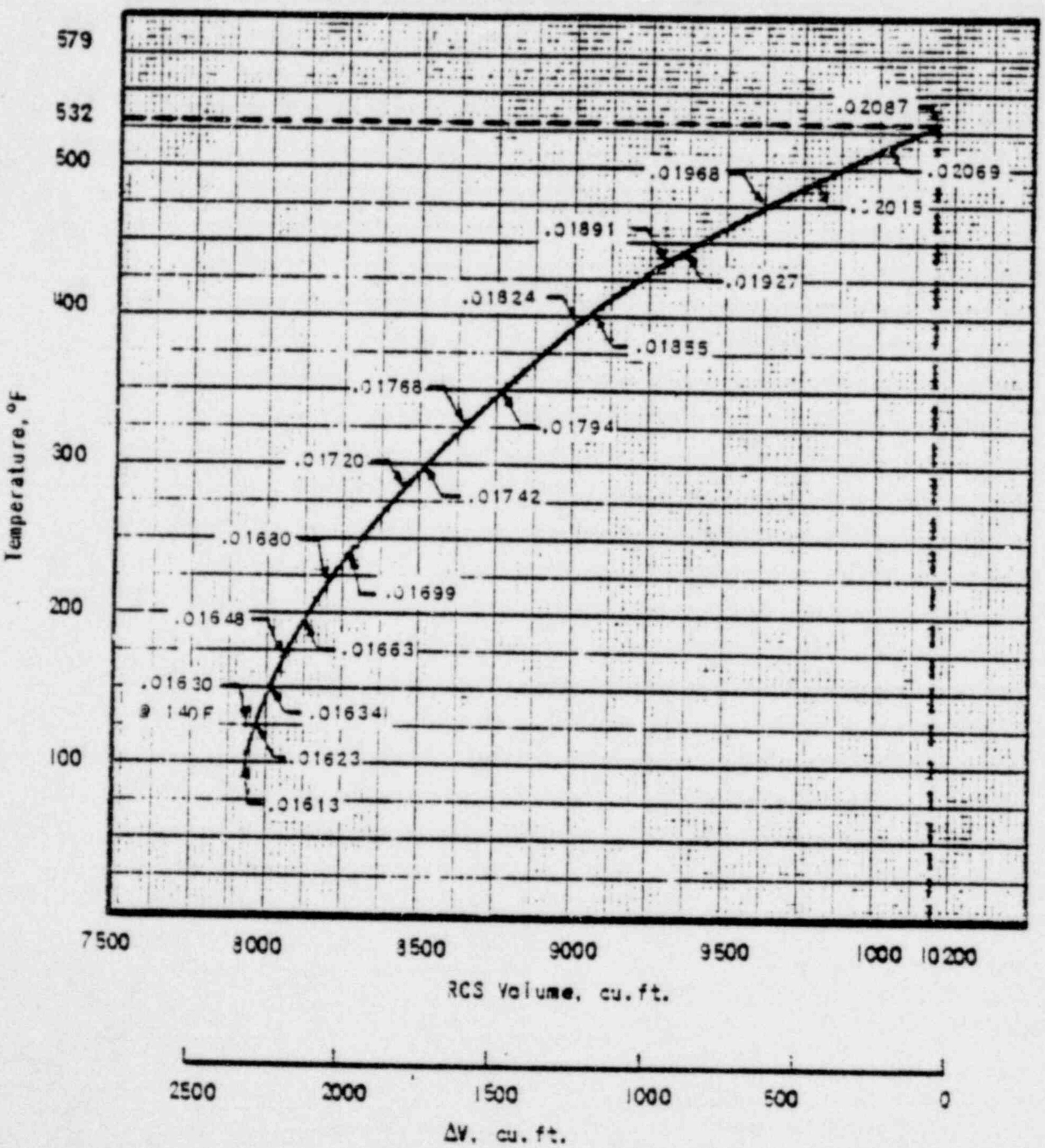
5.5 Feed and Bleed - Dilution — Figure 5.5

1. Equation: 2.2 (9)
2. Bases: RCS at 2155 psig and 579°F.
3. Follow curve beginning at initial concentration up to vertical above final concentration; where they intersect, read volume of demineralized or deborated water required at left.

5.6 Batch Boration or Dilution

Figures 5.4 and 5.5 can be used for each separate batch with negligible error.

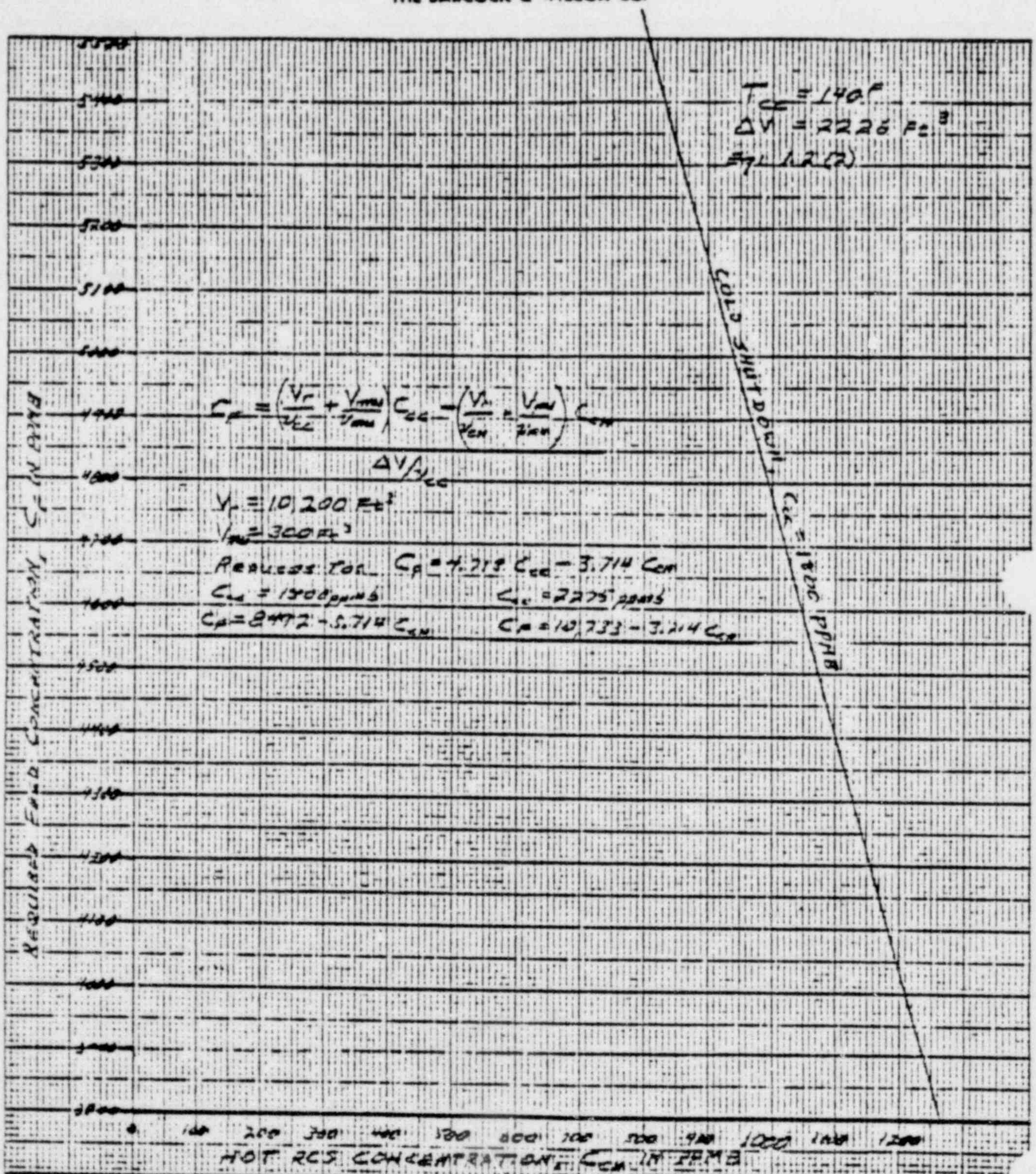
FIGURE 5.1



OFR #1 OP 1103.04

FORM 100

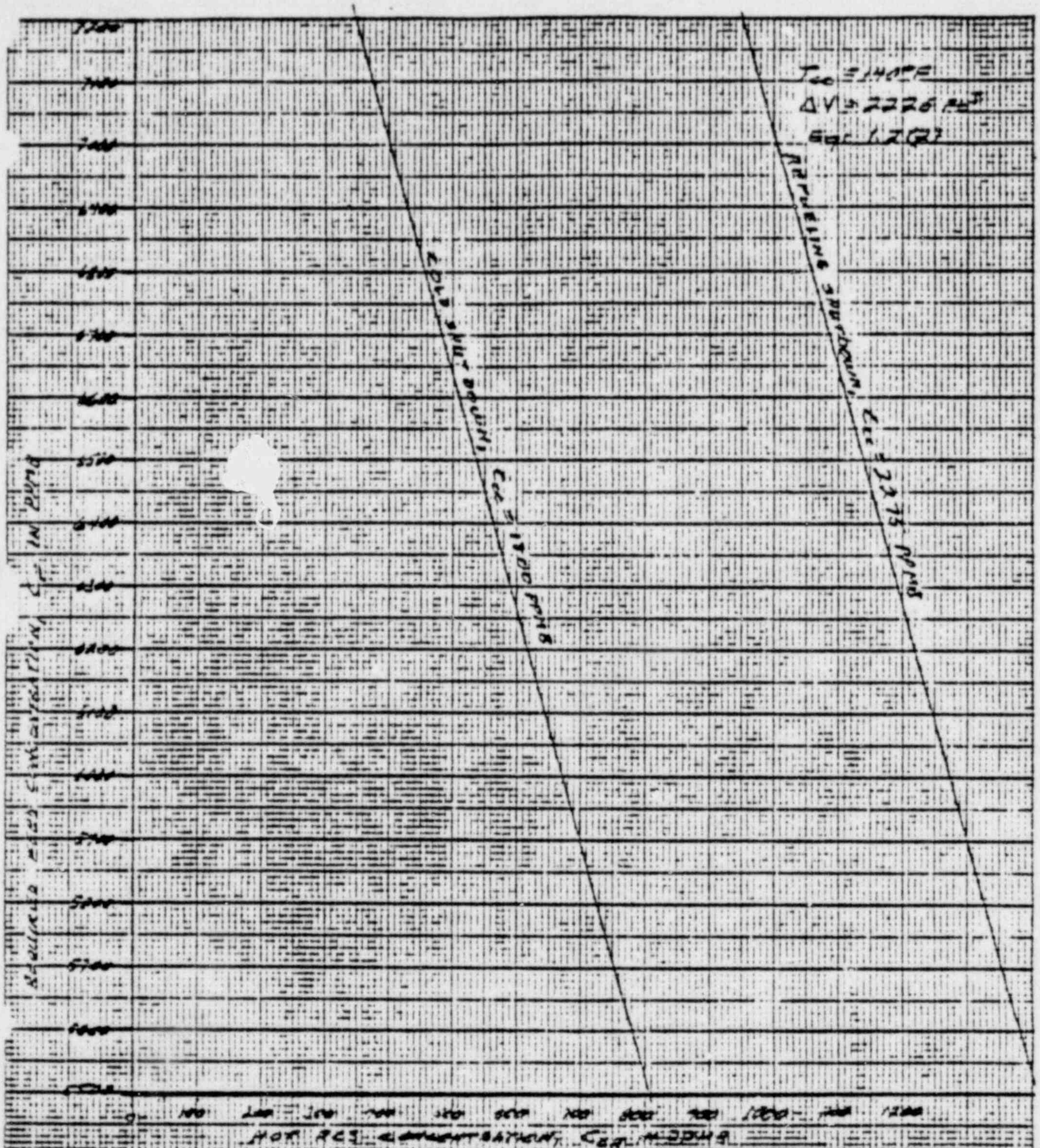
THE BABCOCK & WILCOX CO.



CUSTOMER OFR UNIT #1	JOB NO CP 1103 04
SUBJECT REQUIRED FEED CONCENTRATION VERSUS RCS CONCENTRATION PRIOR TO SHUTDOWN	Figure 5.2(a)
	BY J.A.I.
	DATE 9-14-71

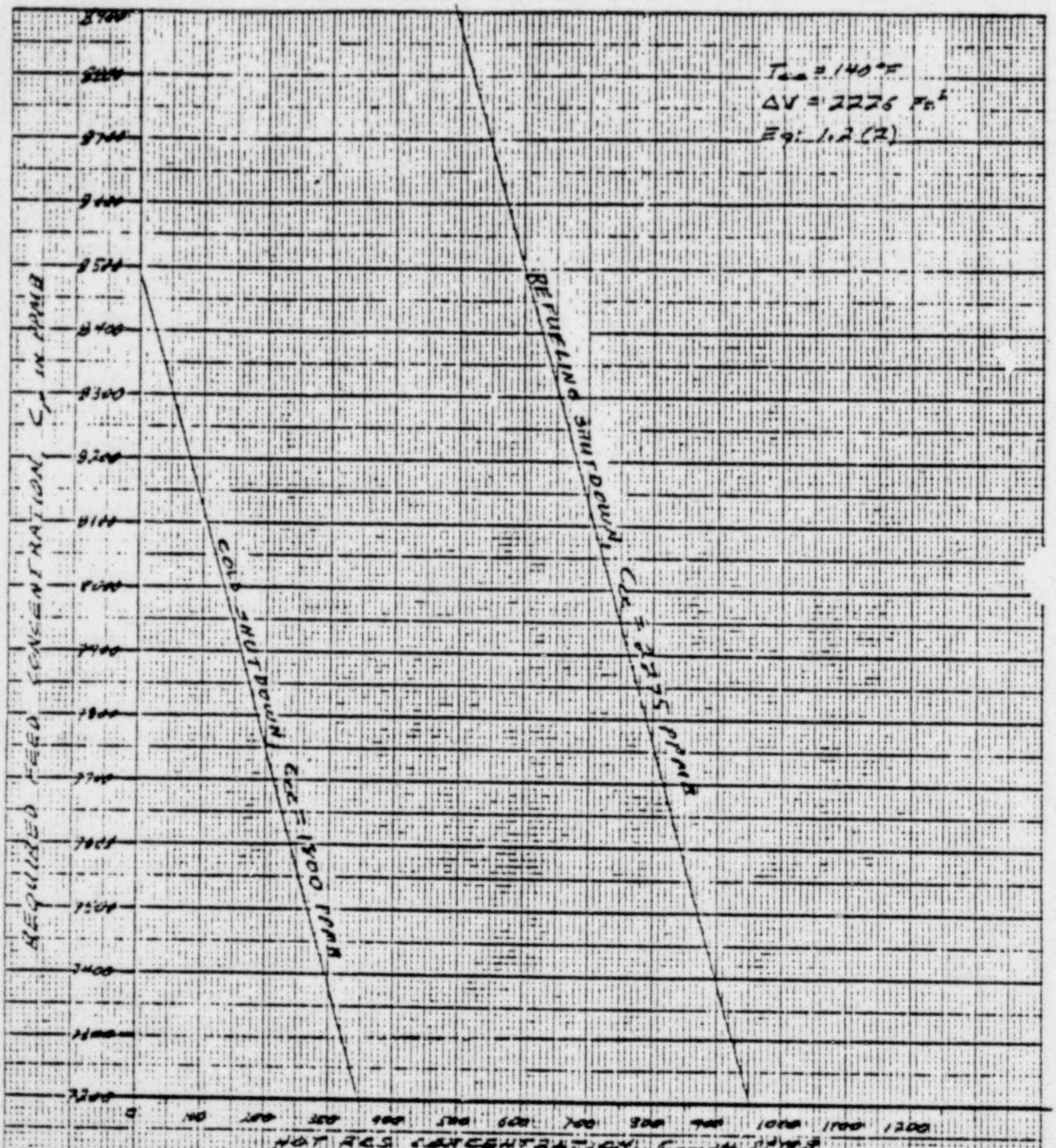
K+E

THE BABCOCK & WILCOX CO.



CUSTOMER OP? UNIT #1	JOB NO OP NO? ON
SUBJECT REQUIRED FEED CONCENTRATION VERSUS	Figure 5.2(A)
RCS CONCENTRATION PRIOR TO SHUTDOWN	BY T.P.I.
	DATE 9-14-71

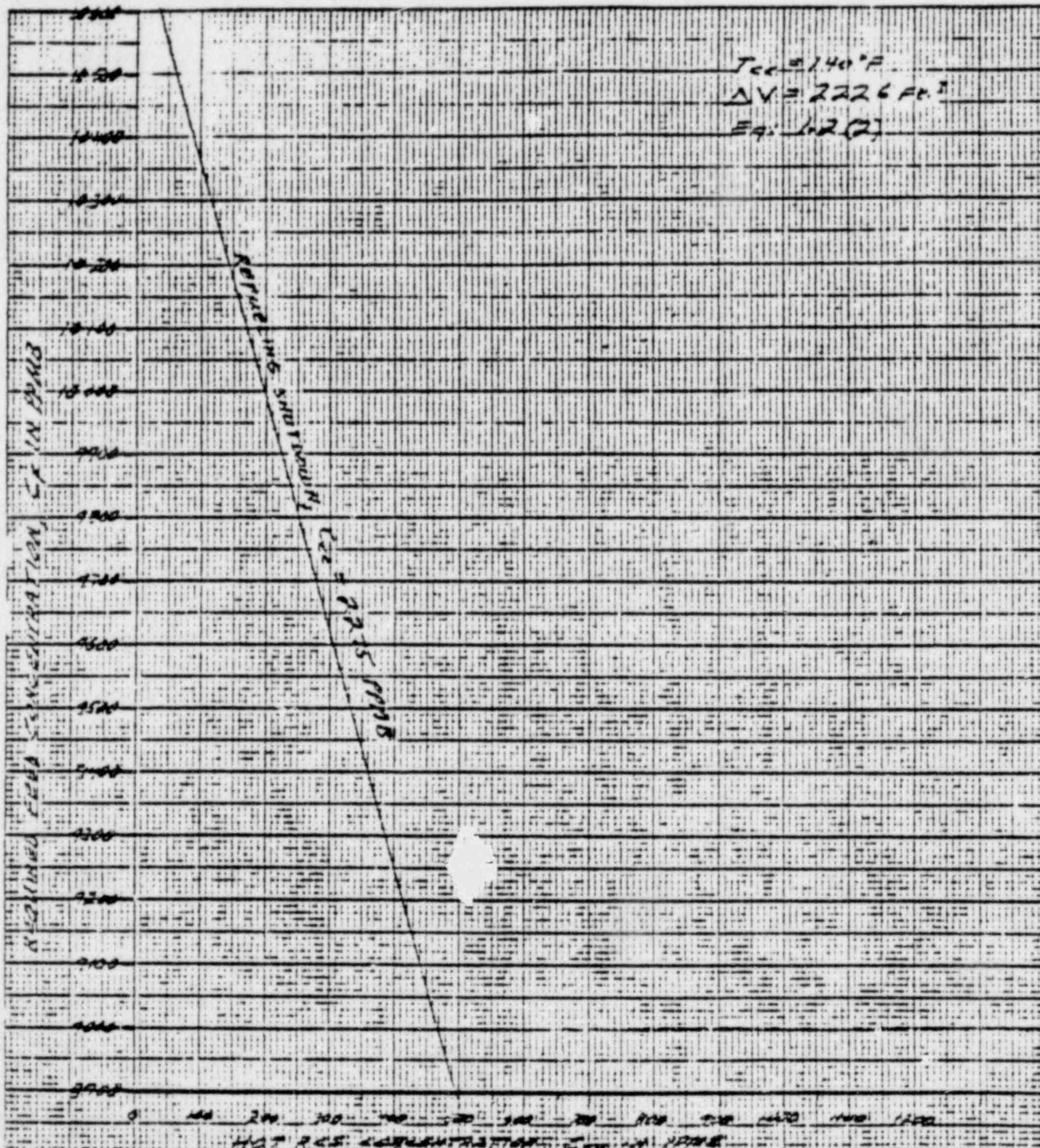
THE BABCOCK & WILCOX CO.



CUSTOMER GPA UNIT #1	JOB NO GA 1103 04
SUBJECT REQUIRED FEED CONCENTRATION VERSUS RCS CONCENTRATION PRIOR TO SHUTDOWN	Figure 5.2(?)
	BY T.P.J.
	DATE 9-14-71

K•E

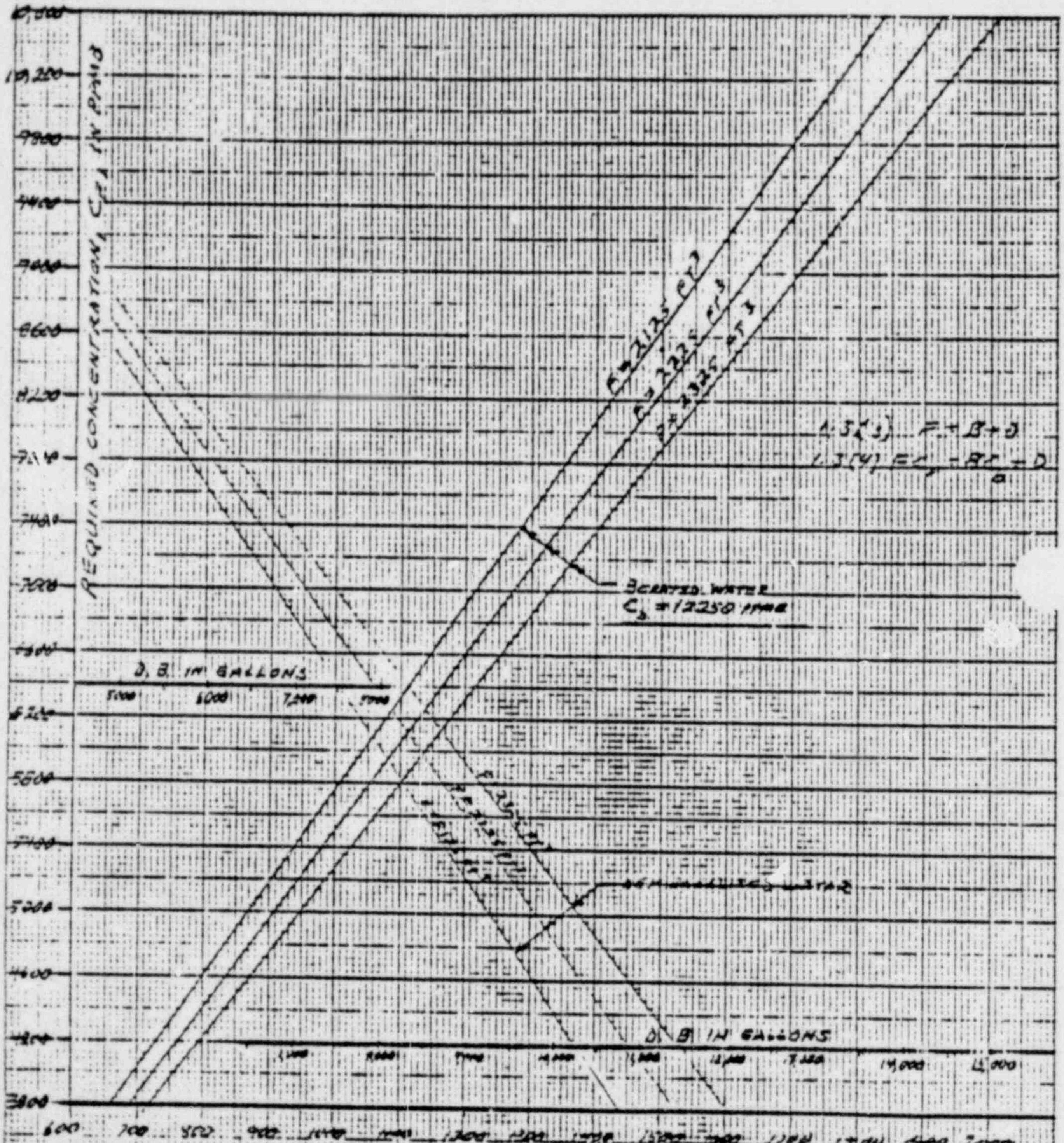
THE BABCOCK & WILCOX CO.



CUSTOMER OFR UNIT #1	JOB NO OP 1103 04
SUBJECT REQUIRED FEED CONCENTRATION	Figure 5.2(1)
VS RCS CONCENTRATION PRIOR TO SHUTDOWN	BY J.P.I.
	DATE 9-14-71

FORM 100

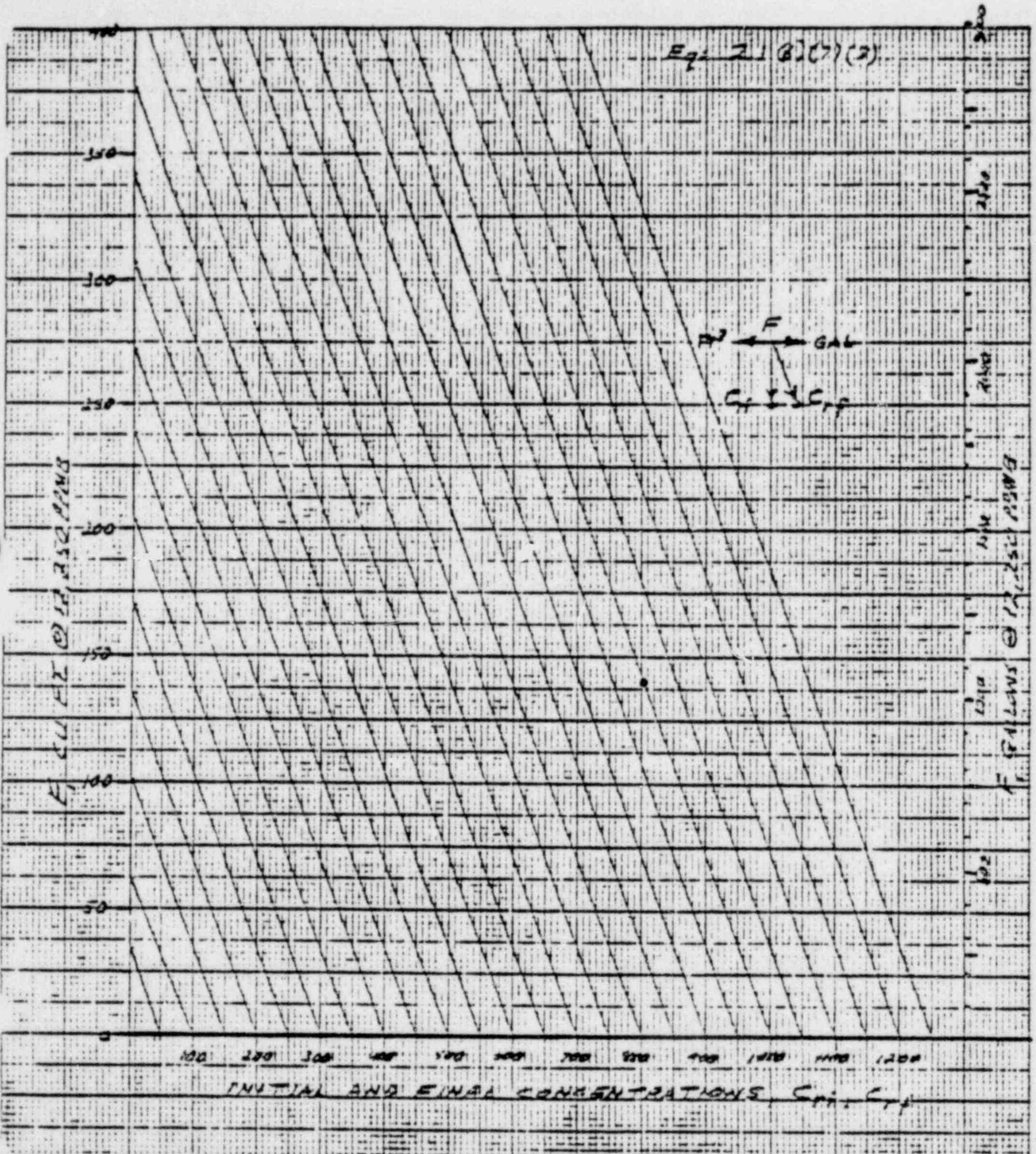
THE BABCOCK & WILCOX CO.



CUSTOMER OFR UNIT #1	JOB NO OP 1103 OH
SUBJECT REQUIRED FEED CONCENTRATION VERSUS	Figure 5.3
EQUIVALENT VOLUME OF BLENDED BORATED	BY J.P.L.
AND DEMINERALIZED WATER C_B = 12,250 ppm	DATE 9-14-71

K+E

THE BABCOCK & WILCOX CO.



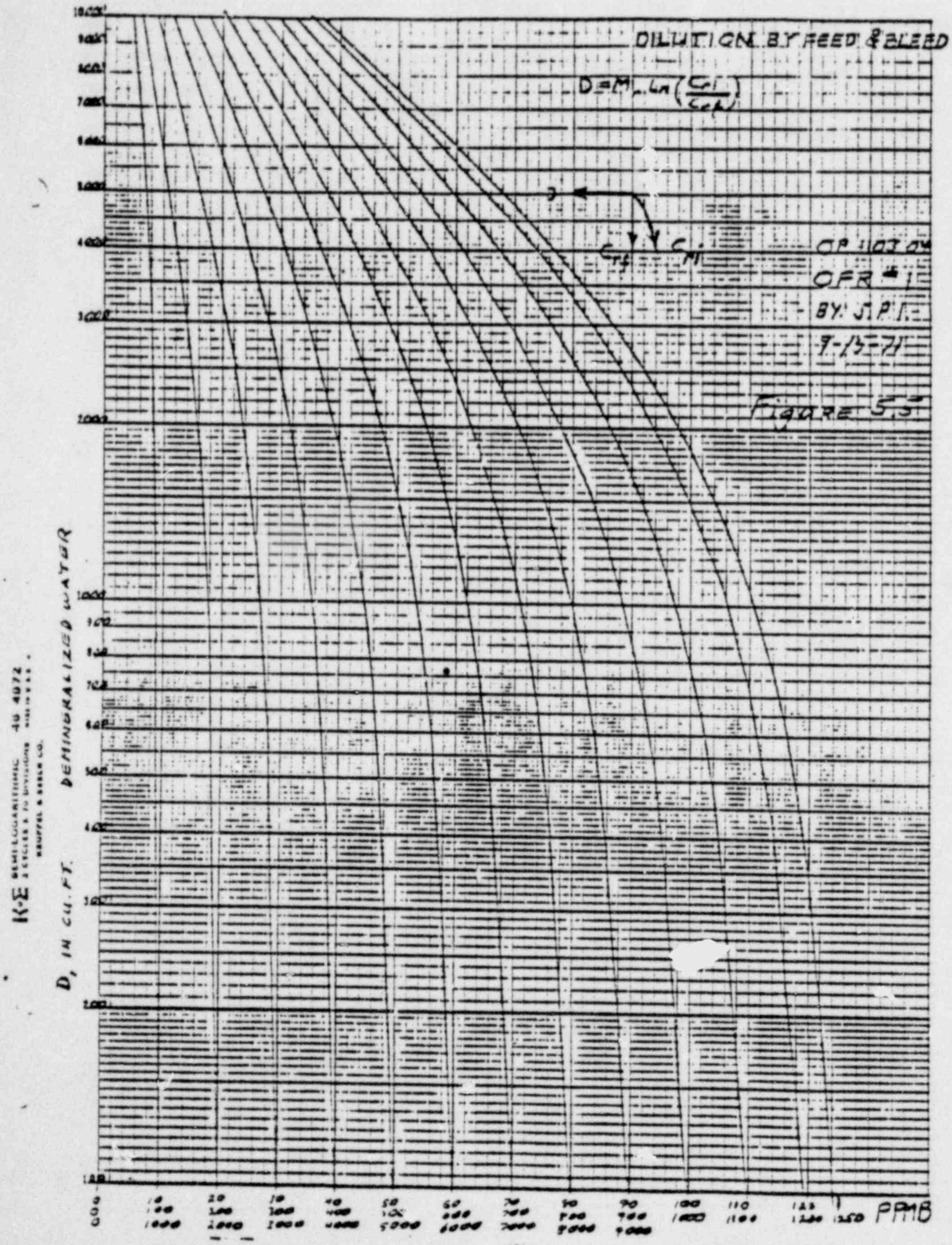
CUSTOMER OFR UNIT #1 JOB NO OP 1103 04

SUBJECT BORATION BY FEED AND BLEED — Figure 5.4

FEED VOLUME REQUIRED VERSUS INITIAL AND FINAL RCS CONCENTRATIONS

BY T.P.I

DATE 9-15-71



DILUTION BY FEED & BLEED
D=M . Ln (Ci/Cf)
OP 1103.04
OFR #1
BY: J.P.T.
9-15-71
Figure 5.5
D, IN CU. FT. DEMINERALIZED WATER
K-E SEMI-LOGARITHMIC 46 4972
PPMB

1103 15

OFR UNIT #1

REACTIVITY BALANCE CALCULATION

OP 1103-15

I. PURPOSE

The purpose of this procedure is to describe the reactivity balance calculation for the following:

Estimated critical position
Estimated critical boron
Shutdown margin at 532°F
Shutdown margin at 70°F
Reactivity balance at power (for anomaly)

II. DESCRIPTION

1. Reference Conditions

There are three (3) temperature reference conditions used in this procedure. Worksheets and curves are based to these three temperatures.

- 70 F, with all rods out, no boron, no xenon, 0% doppler, equilibrium SM

532 F, with all rods out, no boron, no xenon, 0% doppler, equilibrium SM

582 F, with all rods out, no boron, 100% xenon, 100% doppler, equilibrium SM

Note: Rod worth curves are referenced to 532°F and 582°F. For ECP calculations, use the 532°F rod worth curves.

2. The OFR simulator can be operated at three (3) positions in core life:

Beginning of life	BOL	0 EFPD's
Middle of life	MOL	90 EFPD's
End of life	EOL	180 EFPD's

The power doppler and rod worth curves of this procedure will contain information for all three core life conditions. The time in life must be considered when using these curves. Use one of the three listed EFPD's for all similar calculations.

3. Reactivity Balance Equation

$\rho_{(net)} = \rho_{(fuel)} + \rho_{(rods)} + \rho_{(boron)} + \rho_{(temp)} + \rho_{(doppler)} + \rho_{(xe)} + \rho_{(SM)}$

When $K = 1$, $\rho_{(net)} = 0$

WORK SHEET I - ESTIMATED CRITICAL POSITION

Reference Conditions: 532°F, 0% doppler, no xenon, no control rods, Equilibrium samarium, No Boron

1. Fuel Reactivity

   a. Core burnup ________ EFPD

   b. Obtain fuel reactivity from Figure 1 for 532°F, + __________% Δk/k

2. Boron Reactivity

   a. Boron concentration _____ ppm

   b. Obtain boron reactivity from Figure 2 for 532°F - __________% Δk/k

3. Xenon Reactivity

   a. If power has changed more than once in the last 40 hours, consult the reactor engineer for xenon value.

   b. Last power level was _____% FP for _____ hrs.

   c. Time shutdown _____ hrs.

   d. Obtain xenon reactivity from Figure 5 curves - __________% Δk/k

4. Samarium Reactivity After Shutdown

   a. Average power for last 30 days _____% FP

   b. Time after shutdown _____ days

   c. Obtain samarium reactivity from Figure 6 - __________% Δk/k

5. Temperature Coefficient Reactivity

   a. Average RC temperature _____°F

   b. Reference temperature is 532°F

   c. Obtain temperature coefficient from Figure 3 for 532°F

   d. Reactivity = (T(ave) - 532) (Temp. Coeff.)

   e. Reactivity = (_____ - ___) (_____ x 10^{-2}% Δk/k/°F) = ± __________% Δk/k

6. Axial Power Shaping Rod Reactivity

   a. Obtain reactivity of Group 8 at _____% WD from Figure 7 (a,b or c) - __________% Δk/k

7. Estimated Control Rod Reactivity Needed to Maintain Criticality

a. Sum Steps 1-6 and change the sign. - ________% Δk/k

8. Determine ± 0.5% Δk/k

a. Add + .5% Δk/k to the value calculated in Step 7.

- ________% Δk/k + .5% Δk/k = - ________% Δk/k

b. Add - .5% Δk/k to the value calculated in Step 7.

- ________% Δk/k - .5% Δk/k = - ________% Δk/k

9. Estimated Critical Position

Obtain rod positions from Figure 8 (a,b or c) for reactivity calculated in Steps 7 and 8.

Group 1-4 100WD

Group 8 ________% WD
(Same as Step 6)

Boron ________ppm

ECP ________% WD Group ________
(determine from Step 7)
+ 5% ________% WD Group ________
(Determine from Step 8.a.)
-.5% ________% WD Group ________
(Determine from Step 8.b.)

Calculated by ____________________ Date/Time ____________

Checked by ____________________ Date/Time ____________

WORK SHEET II - ESTIMATED CRITICAL BORON CONCENTRATION

Reference Conditions: 532°F, 0% doppler, No Xenon, No Control Rods, Equilibrium Samarium, No Boron

1. Fuel Reactivity

a. Core burnup _____ EFPD

b. Obtain fuel reactivity from Figure 1 for 532°F, 0% xenon, 0% doppler + _________% Δk/k

2. Xenon Reactivity

a. If power has changed more than once in the last 40 hours, consult the reactor engineer for xenon value.

b. Last power was _____% FP for _____ hrs.

c. Time shutdown _____ hrs.

d. Obtain xenon reactivity from Figure 5. - _________% Δk/k

3. Samarium Reactivity After Shutdown

a. Average power level for last 30 days _____% FP

b. Time after shutdown _____ days.

c. Obtain samarium reactivity from Figure 6. - _________% Δk/k

4. Temperature Coefficient Reactivity

a. Average RC temperature _____°F.

b. Reference temperature is 532°F.

c. Obtain temperature coefficient from Figure 3. for 532°F to be _____ x 10^{-2}%

d. Reactivity = (T(ave) - 532) (Temp. Coeff.)

e. Reactivity = (_____ - _____) (_____ x 10^{-2}% Δk/k/°F) ± _________% Δk/k

5. Control Rod Reactivity at Desired Insertion

Groups 1-4 at 100 % WD	Group 5 at _____% WD
Group 8 at _____% WD	Group 6 at _____% WD
	Group 7 at _____% WD

a. Obtain Regulating Group reactivity from Figure 8(a,b,or c) - _________% Δk/k

b. Obtain Group 8 reactivity from Figure 7 (a,b, or c) - _________% Δk/k

6. Boron to Maintain Critical Conditions

a. Sum Steps 1 thru 5 and change sign. - _________ % Δk/k

b. Obtain boron concentration for the reactivity calculated in Step 6.a. from Figure 2 for 532°F.

c. Estimated critical boron concentration is _____ ppmB.

7. Determine ±.5% Δk/k

a. Add +.5% Δk/k from Step 5 - _____% Δk/k = _____% Δk/k

b. Add -.5% Δk/k from Step 5 - _____% Δk/k = _____% Δk/k

8. Estimated Critical Position

Obtain rod positions from Figure 8 (a,b, or c) for reactivity from Steps 6 and 7.

Group 1-4 100% WD

Group 8 ______% WD
(same as Step 5)

Boron _____ ppm

ECP ____% WD Group ____
(same as Step 5)

+.5% Δk/k _____% WD Group _____

-.5% Δk/k _____% WD Group _____

Calculated by ______________________________ Date/Time ______________________________

Checked by ______________________________ Date/Time ______________________________

WORK SHEET III - SHUTDOWN MARGIN FOR HOT SHUTDOWN

Reference Conditions: 532°F, 0% doppler, No xenon, No Control Rods, Equilibrium Samarium, No Boron

1. Fuel Reactivity

   a. Core burnup _____ EFPD

   b. Obtain fuel reactivity from Figure 1 for 532°F, No Boron, No Rods, No Xenon, Equilibrium Samarium. + __________% Δ k/k

2. Boron Reactivity

   a. Boron concentration _____ ppm

   b. Obtain boron reactivity from Figure 2 for 532°F - __________% Δ k/k

3. Xenon Reactivity

   a. If power has changed more than once in the last 40 hours, consult the reactor engineer for value.

   b. Last power was _____% FP for _____ hrs.

   c. Time shutdown _____ hrs.

   d. Obtain xenon reactivity from Figure 5 curves - __________% Δ k/k

4. Samarium Reactivity After Shutdown

   a. Average power level for last 30 days _____% FP

   b. Time after shutdown _____ days

   c. Obtain samarium reactivity from Figure 6. - __________% Δ k/k

5. Temperature Coefficient Reactivity

   a. RC temperature _____°F

   b. Reference temperature is 532°F for hot shutdown

   c. Obtain temperature coefficient from Figure 3.

   d. Reactivity = (T(ave) - 532°) (Temp. Coeff.) ± __________% Δk/k

6. Control Rod Reactivity

   a. Reactivity of Groups 1 thru 7 fully inserted from Curve 9 (a, b or c) 532 = ________% Δk/k

   b. Obtain Group 8 reactivity from Figure 7 (a, b or c) for ____% WD. = ________% Δk/

7. Reactivity of most reactive rod (assume stuck at 100% WD) + ________% Δk/k

BOL	2.60% Δk/k
MOL	2.37% Δk/k
EOL	2.14% Δk/k

8. Shutdown Margin

   a. The shutdown margin is the sum of Steps 1 thru 7. = ________% Δk/k

9. If shutdown margin is <1% Δk/k, notify the Shift Supervisor.

Calculated by: ______________________________ Date/Time ____________

WORK SHEET IV - SHUTDOWN MARGIN FOR COLD SHUTDOWN

Reference Conditions: $70^{\circ}F$, 0% doppler, 0% xenon, no control rods, no boron, equilibrium samarium.

1. Fuel Reactivity

   a. Core burnup ____ EFPD

   b. Obtain fuel reactivity from Figure 1 curve for 70° + ________% Δk/k

2. Boron Reactivity

   a. Boron concentration ____ ppmB

   b. Obtain boron reactivity from Figure 2 for 70° - ________% Δk/k

3. Xenon Reactivity

   a. If power has changed more than once in the last 40 hours, obtain present xenon value from the reactor engineer.

   b. Last power ____% FP for ____ hours.

   c. Time shutdown ____ hours.

   d. Obtain xenon reactivity from Figure 5 curves. - ________% Δk/k

4. Samarium Reactivity After Shutdown

   a. Average power level for last 30 days ____%

   b. Time after Shutdown ____ days.

   c. Obtain Samarium reactivity from Figure 6. - ________% Δk/k

5. Temperature Coefficient Reactivity

   a. Average RC temperature ____°F.

   b. Obtain temperature coefficient from Figure 3 for 70°F

   c. Reactivity =(T(ave) - 70) (Temp. Coeff)

   d. Reactivity - (____ - 70) (____) = ± ________% Δk/k

6. Control Rod Reactivity

   a. Reactivity of Groups 1 thru 7 fully inserted from Curve 9 (a,b or c) - ________% Δk/k

   b. Obtain Group 8 reactivity from Figure 7 (a,b or c) for ____% WD - ________% Δk/k

7. Reactivity of most reactive rod (assume stuck at 100% WD) + __________% Δk/k

BOL	2.60% Δk/k
MOL	2.37% Δk/k
EOL	2.14% Δk/k

8. Shutdown Margin

a. The shutdown margin is the sum of Steps 1 thru 7. - __________% Δk/k

9. If shutdown margin is <1% Δk/k, notify the Shift Supervisor.

Calculated by: ______________________________ Date/Time ______________________________

6. Doppler Reactivity

a. Present power ____% FP

b. Obtain doppler reactivity from Figure 4 for BOL, MOL or EOL.

c. Actual power doppler (____% Δk/k) - 100% power doppler (____% Δk/k) = + ________% Δk/k

7. Net Reactivity

a. Net reactivity is the sum of Steps 1 thru 6 and should = 0. ________% Δk/k

8. Inform the Shift Supervisor if the results are greater than ± 0.3% Δk/k steady state or greater than ± 0.6% Δk/k transient xenon conditions.

alculated by: ______________________ Date/Time ______________________

WORK SHEET V - REACTIVITY BALANCE AT POWER

Reference Conditions: 582^oF, 100% doppler, 100% Xenon, No Control Rods, No Boron

1. Fuel Reactivity

   a. Core burnup _____ EFPD

   b. Obtain fuel reactivity from Figure 1 for 582^oF, 100% xenon, 100% doppler + _________% Δk/k

2. Boron Reactivity

   a. Boron concentration _____ ppmB

   b. Obtain boron reactivity from Figure 2 for 582^oF - _________% Δk/k

3. Xenon Reactivity

   a. If power has changed more than once in the last 40 hours, consult the reactor engineer for value.

   b. Present power _____% FP for _____ hrs.

   c. Obtain xenon reactivity from Figure 5 curves.

   d. Actual Xe (_____% Δk/k) - 100% FP Xe (_____% Δk/k) = ± _________% Δk/k

4. Temperature Coefficient Reactivity

   a. Average RC temperature _____ ^oF.

   b. Obtain temperature coefficient from Figure 3. for 582^oF.

   c. Reactivity = (T(ave) - 582) (Temp. Coeff.)

   d. Reactivity = (_____ - 582) (_____) = ± _________% Δk/k

5. Control Rod Reactivity

   a. Obtain reactivity of inserted regulating rods from Figure 8 (d,e, or f) - _________% Δk/k

      Group 5 at _____% WD Group 6 at _____% WD
      Group 7 at _____% WD

   b. Obtain reactivity of Group 8 rods at _____% WD from Figure 7(d, e, or f) - _________% Δk/k

DATA SHEET I

COMPARISON OF OVERALL CORE REACTIVITY BALANCE TO PREDICTED VALUE

DATE __________ Time __________ Core Age __________EFPD's

Reactivity Balance = __________% Δk/k (0 ± 1% Δk/k required)

Computations are attached.

Computed By __________

The following reactivity values were normalized. __________________________

____

____

____

____

____

____

____

____

____

____

____

____

____

Reactor Engineer ______________________________

DATA SHEET II

SHUTDOWN MARGIN WITH AN INOPERABLE CONTROL ROD

INOPERABLE ROD NO. ____________

EQUIPMENT REQUIRED

NONE

DATE ______

PROCEDURE STEP NO.	DESCRIPTION	00-08	08-16	16-24
6.1	Shutdo : Margin (Attached Work Sheet)			
6.2	Inoperable Rod Worth (Attached Work Sheet, Step 7)			
6.3	Required Shutdown Margin With Inoperative Control Rod. (1% $\Delta k/k$ + Step 7)			

COMMENTS: ______________________________

____

____

PERFORMED BY:

00-08 ____________________

08-16 ____________________

16-24 ____________________

Figure 1a REACTI[illegible] [illegible] ORE LIFE [illegible] BORON, NO XENON,
NO RODS [illegible] POWER, equ Sm

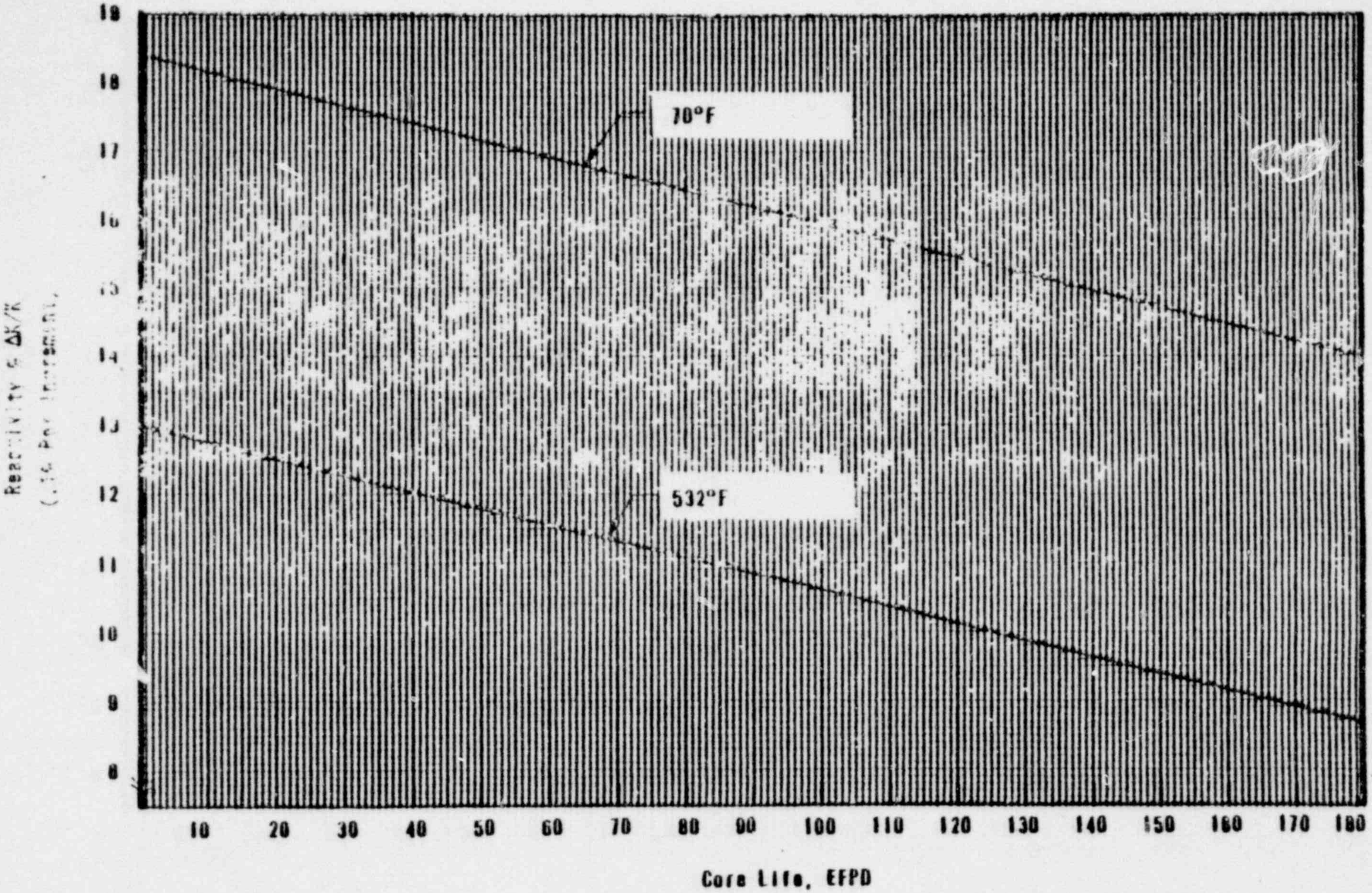
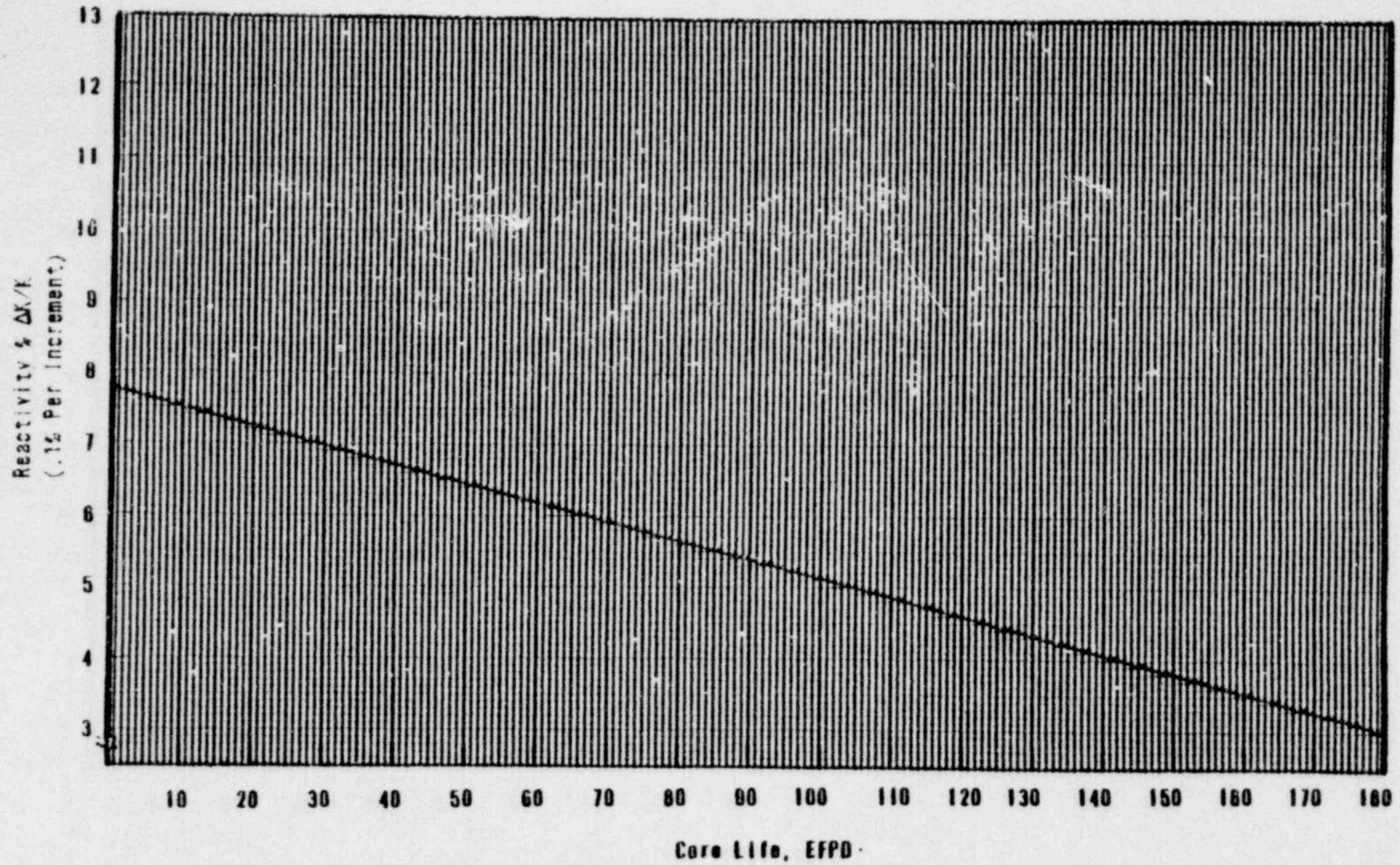


Figure 1b REACTI\ . CORE LIFE NO BORON, NO RODS,
100% POWER, 100% XENON, equ Sm



300

Figure 2A REACTIVITY VS BORON

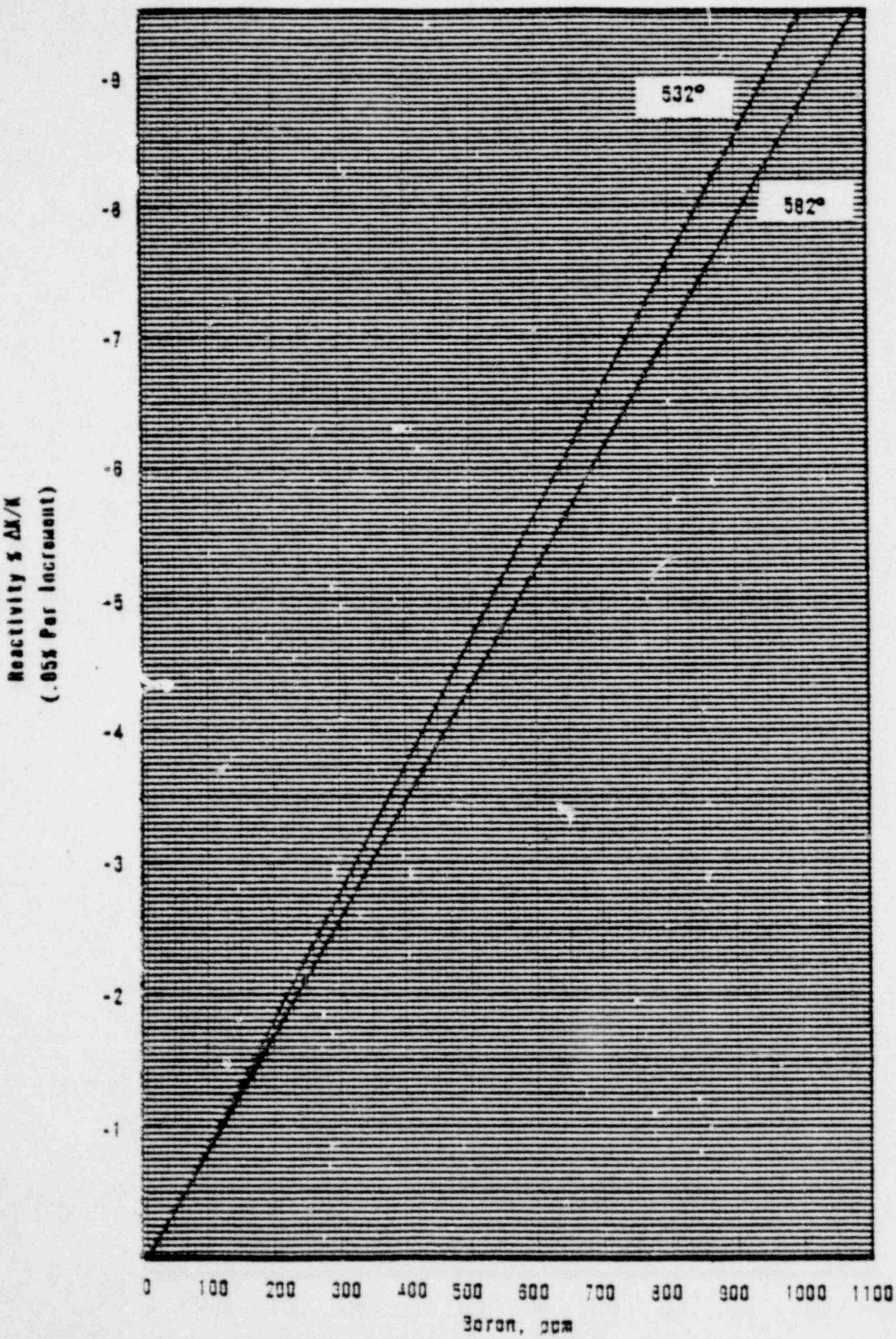


Figure 2b REACTIVITY VS BORON

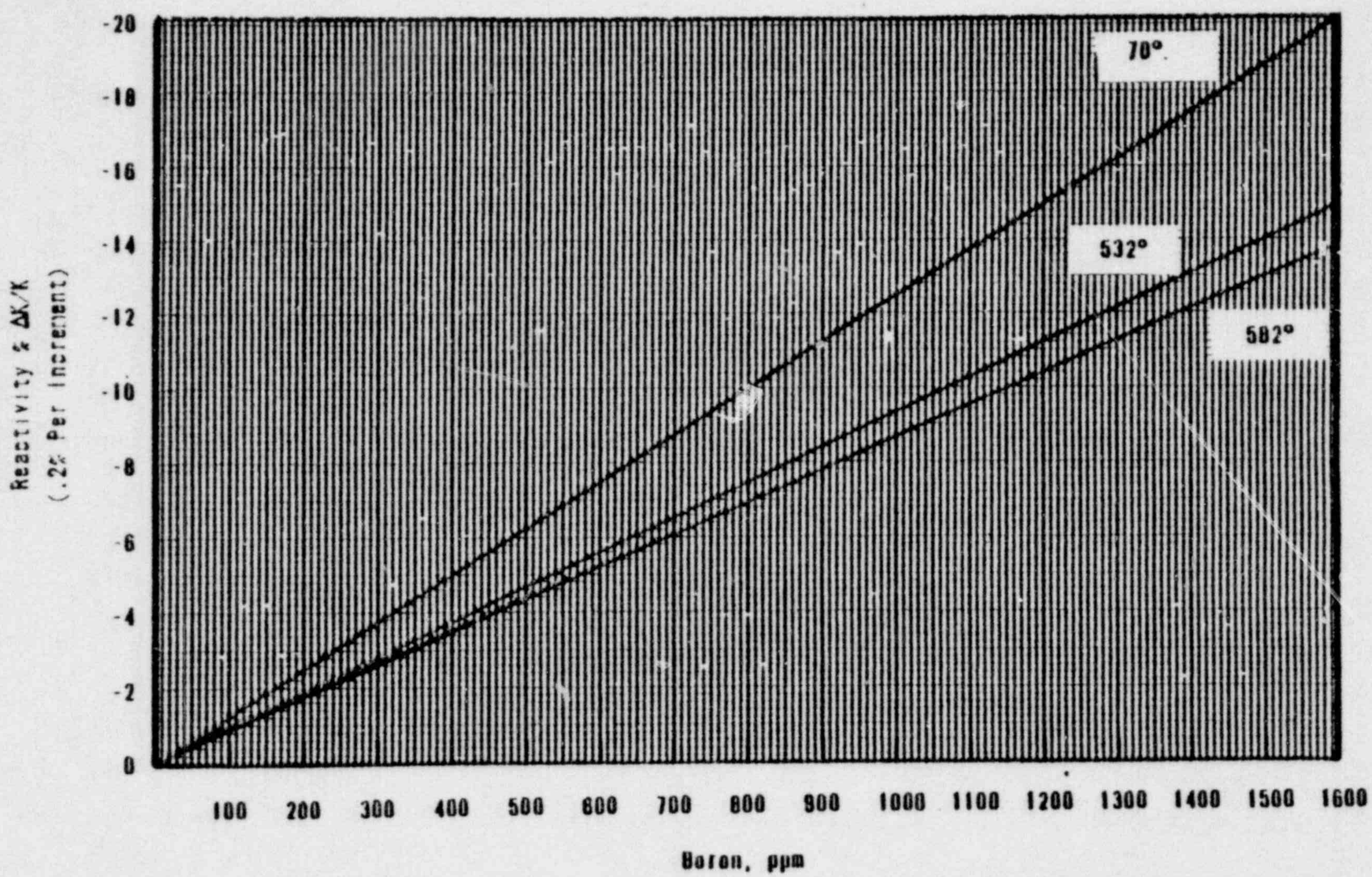


Figure 3 MODERATOR COEFFICIENT VS BORON OFR-1

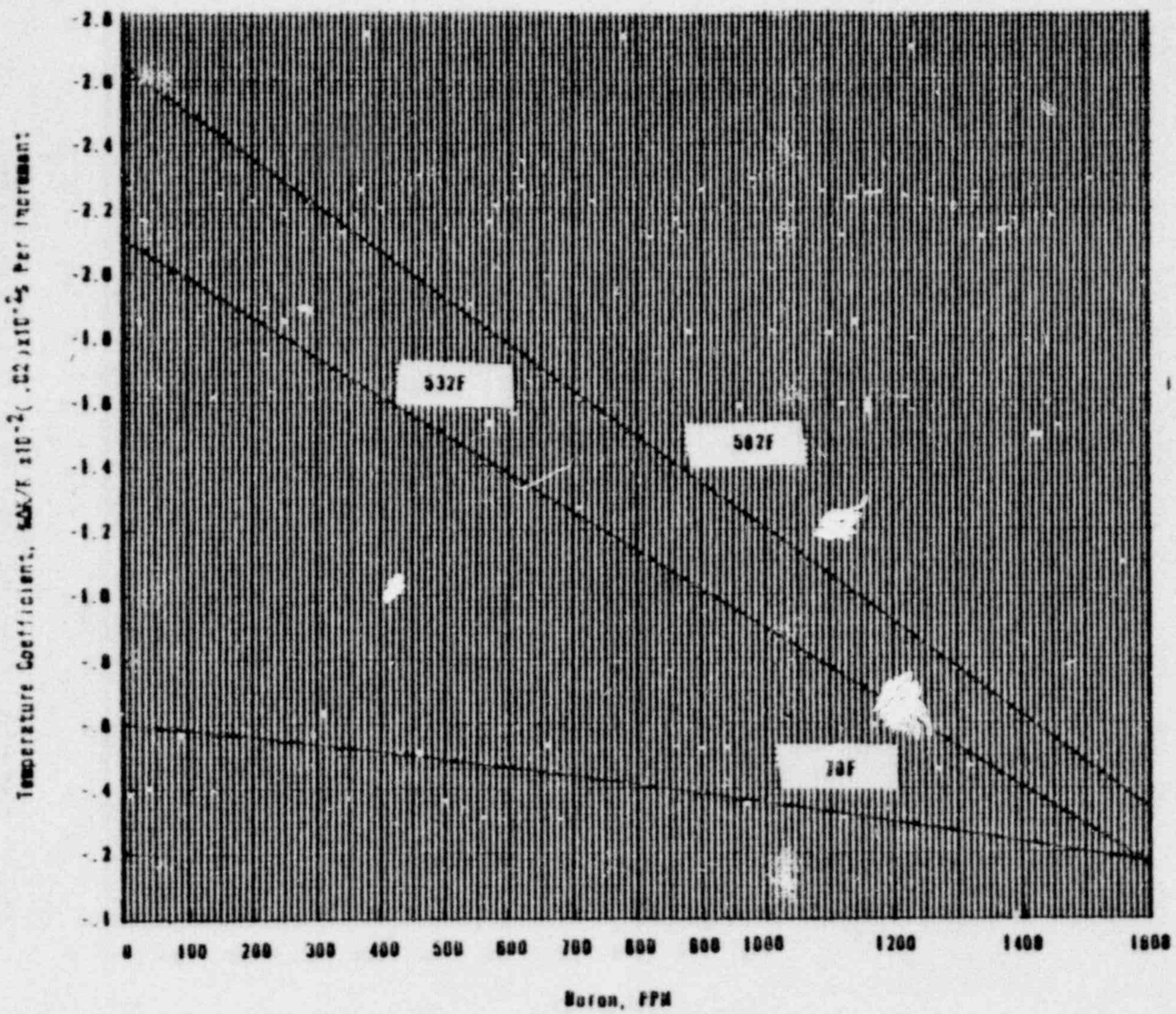


Figure 4 DOPPLER REACTIVITY VS POWER ABOVE 532° DFR-1

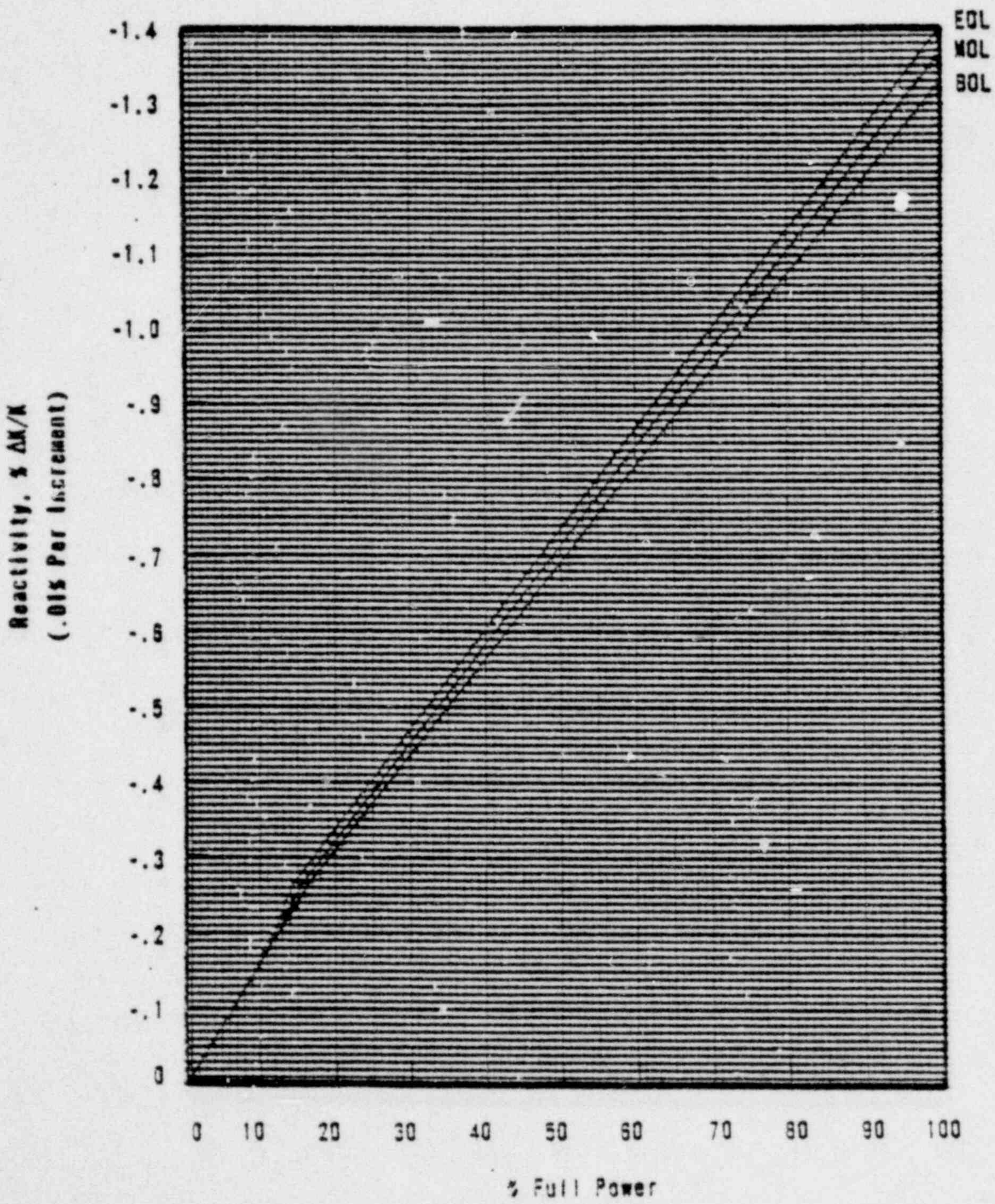


Figure 5a EQUILIBRIUM XENON VS POWER OFR-1

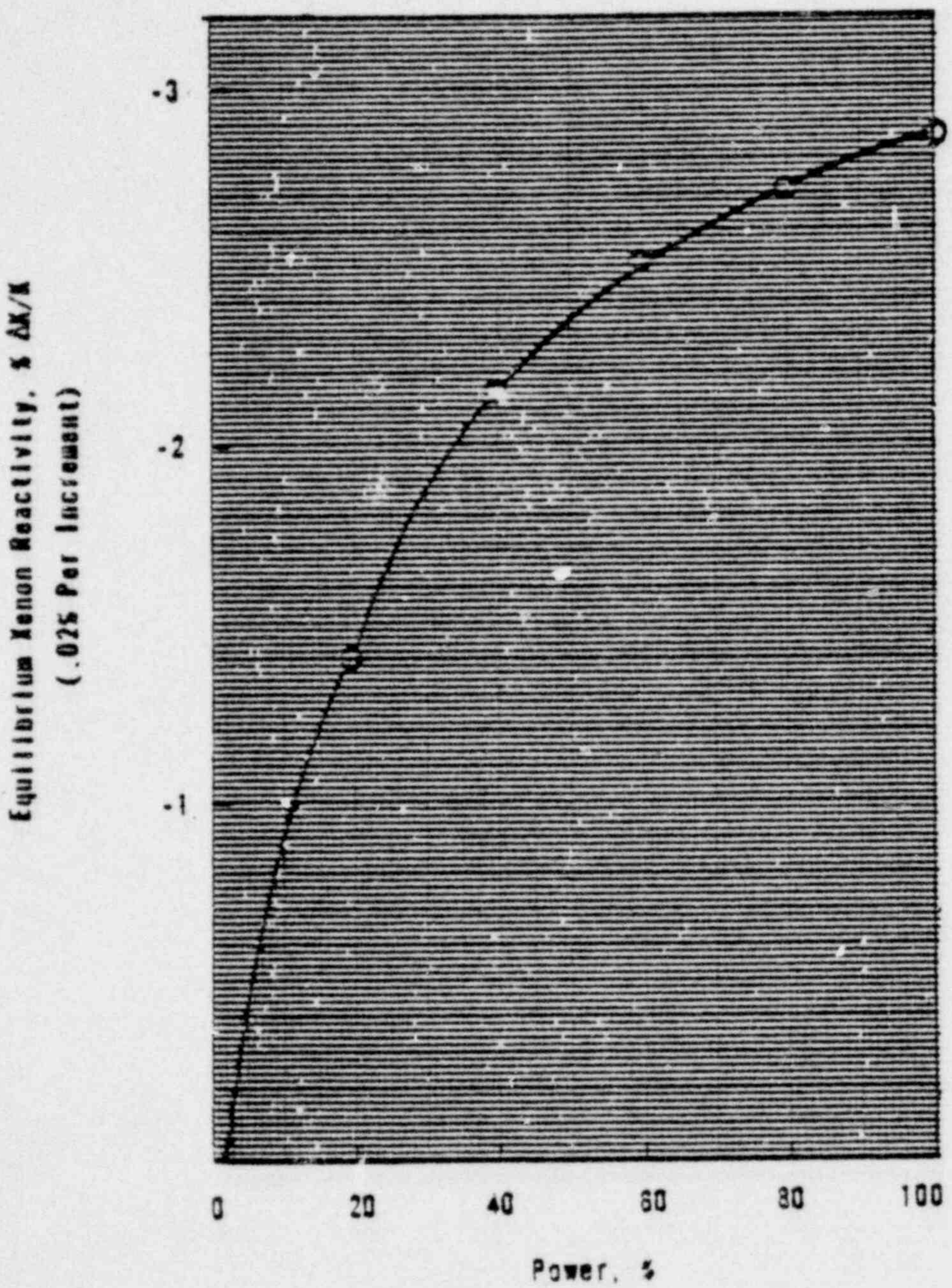


Figure 5b XENON REACTIVITY VS TIME

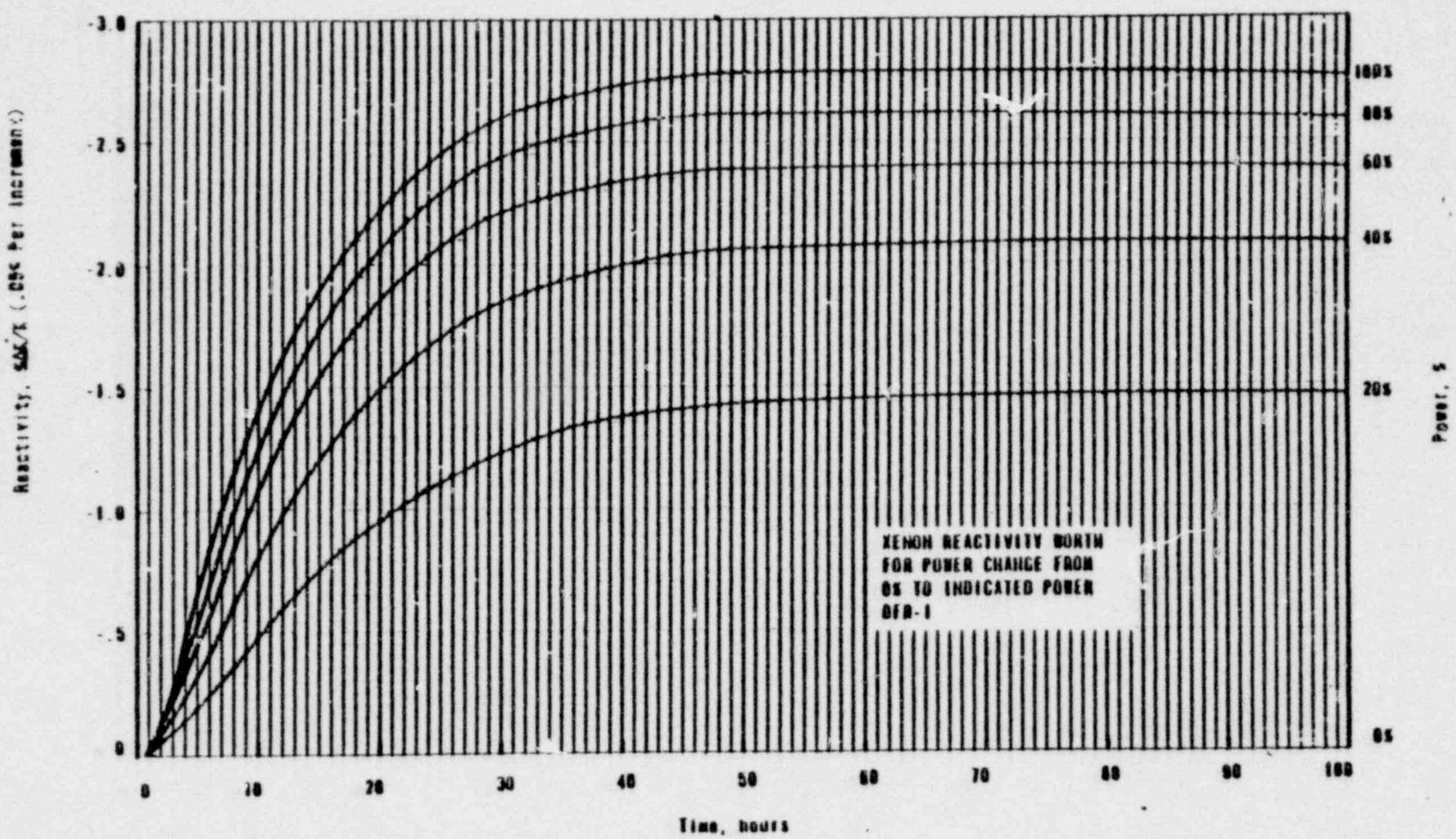


Figure 5c XENON REACTIVITY VS TIME

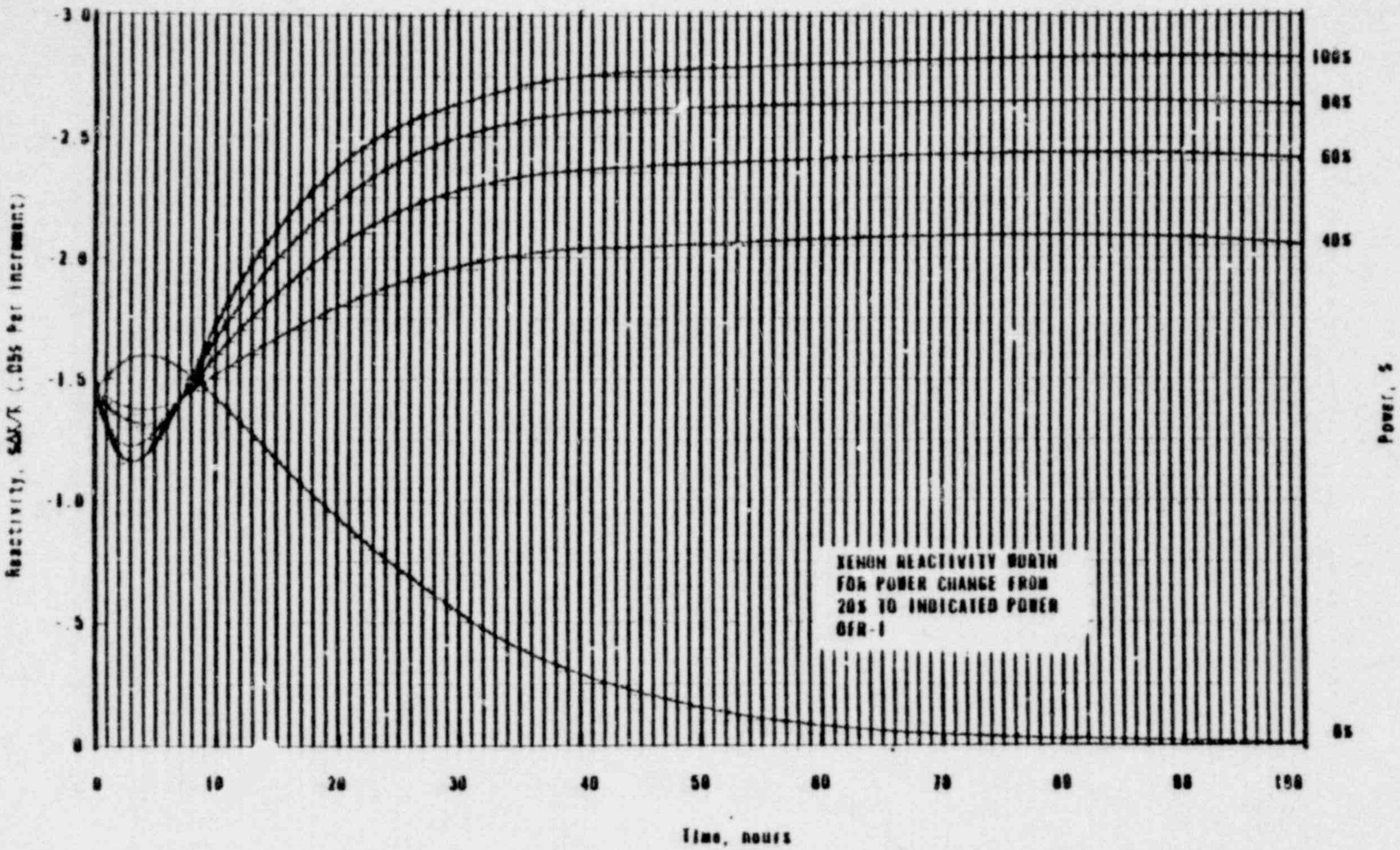


Figure 5a XENON REACTIVITY VS TIME

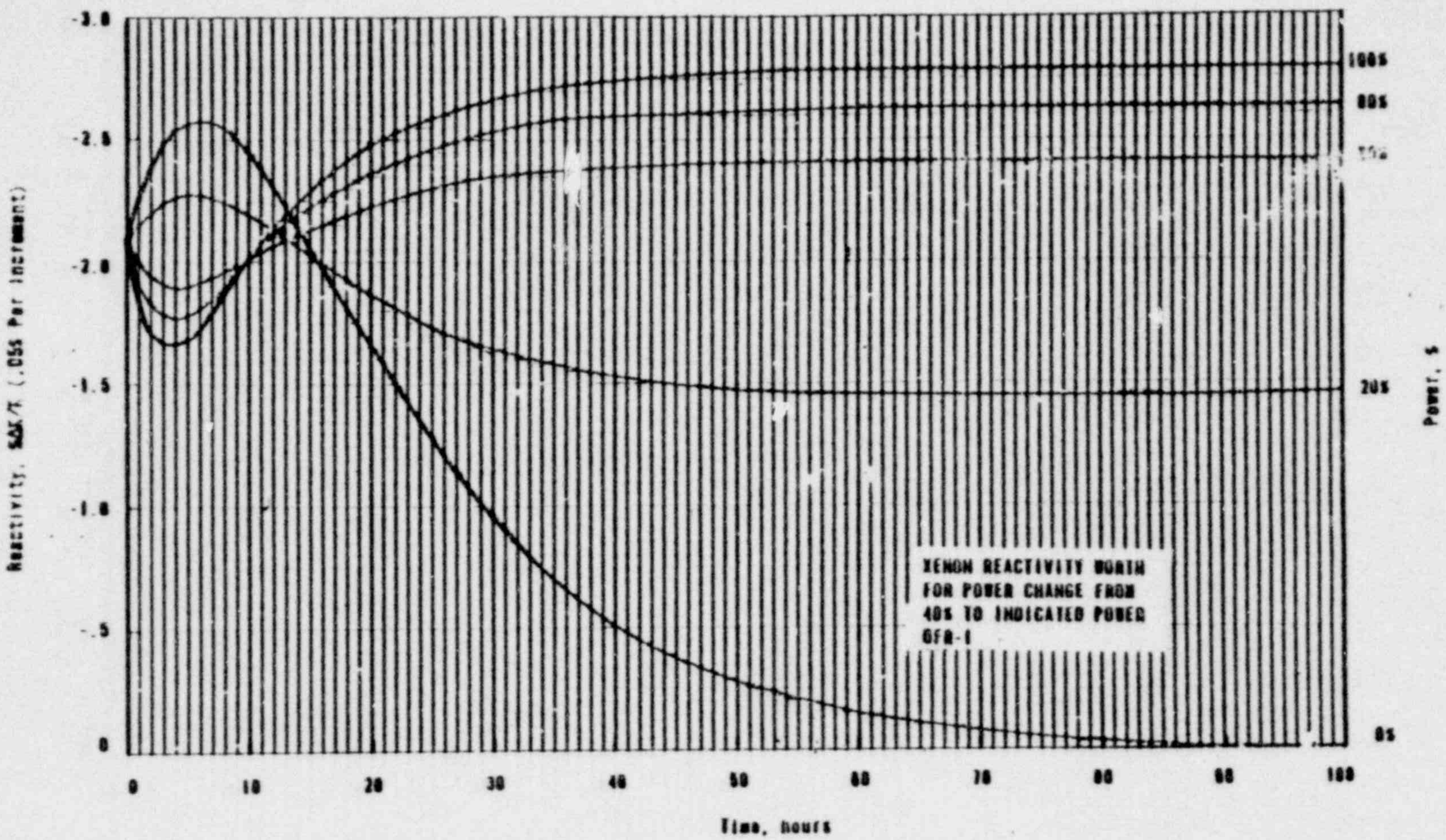


Figure 5a XENON REACTIVITY VS TIME

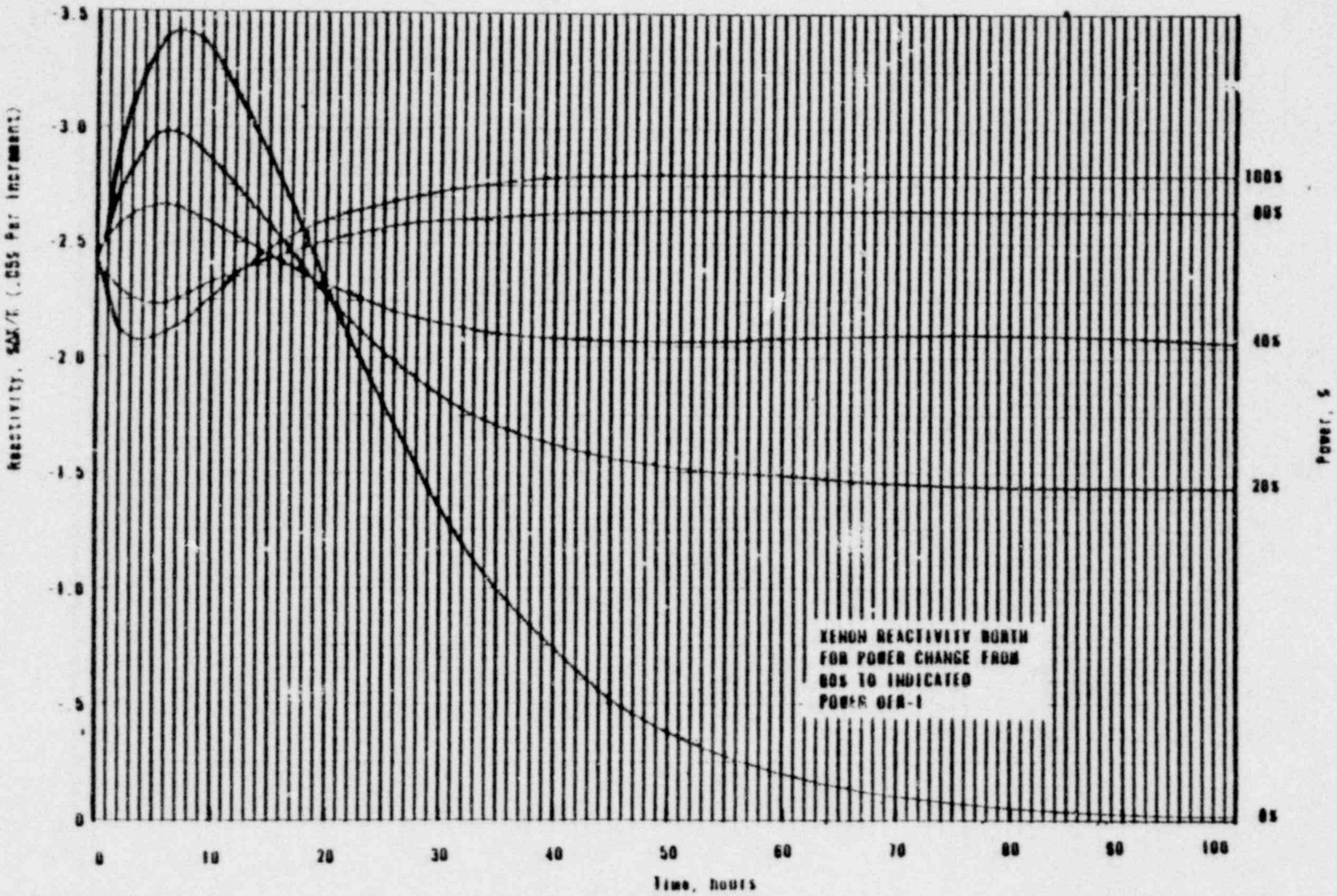


Figure 5F XENON REACTIVITY VS TIME

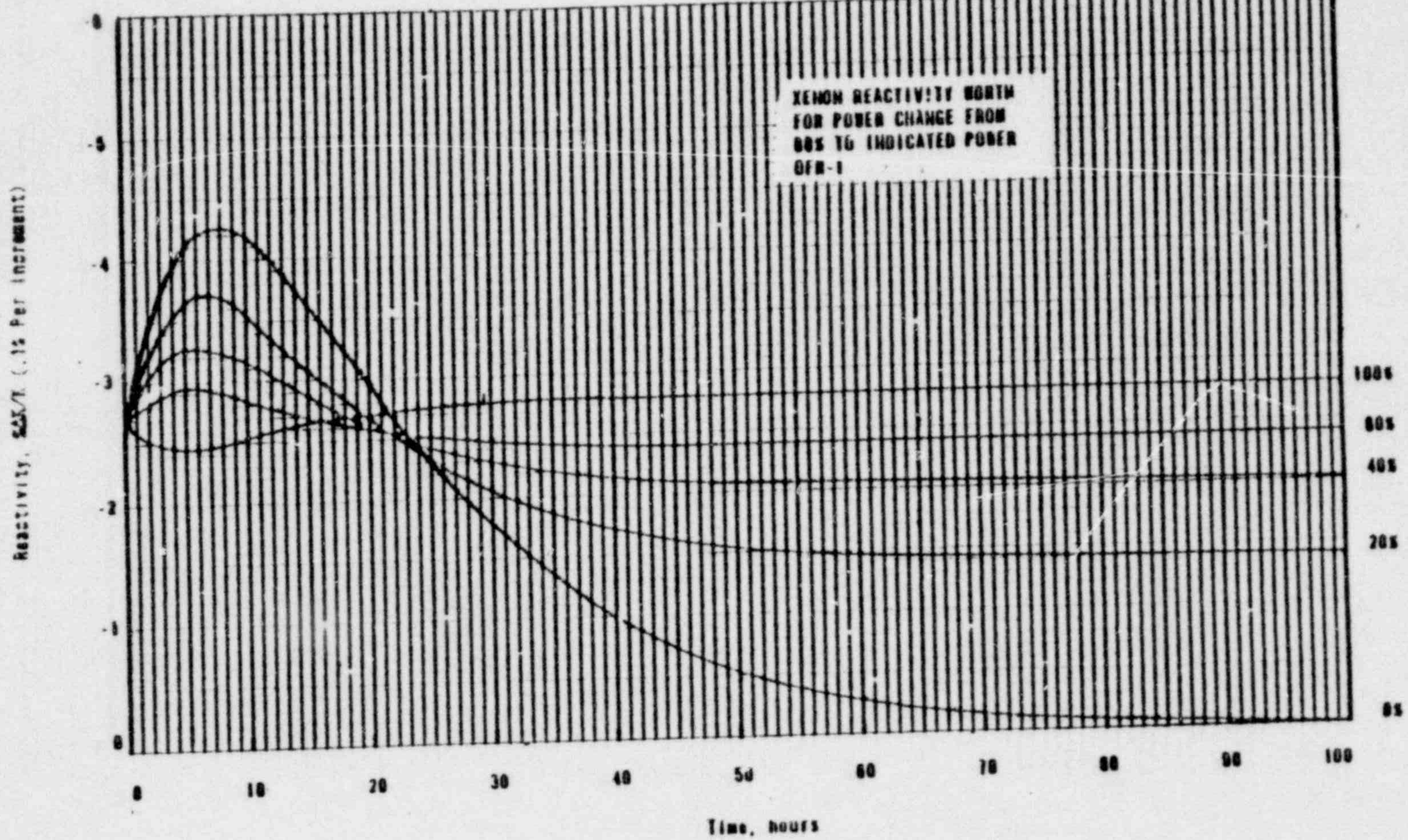


Figure 5g XENON REACTIVITY VS TIME

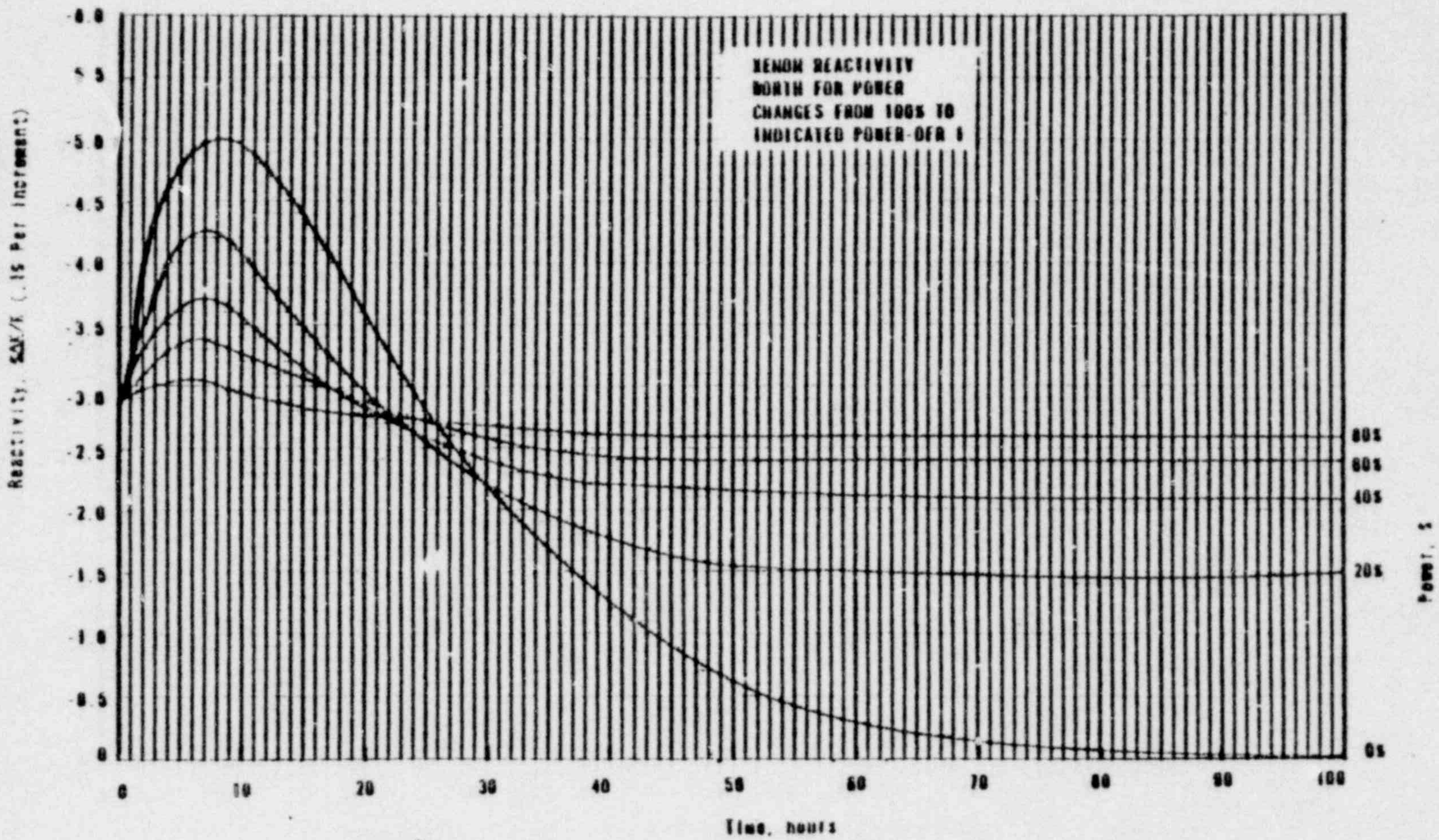


FIGURE 6 SAMARIUM REACTIVITY VS TIME AFTER SHUTDOWN FOR AVERAGE POWER LAST 30 DAYS OPERATION

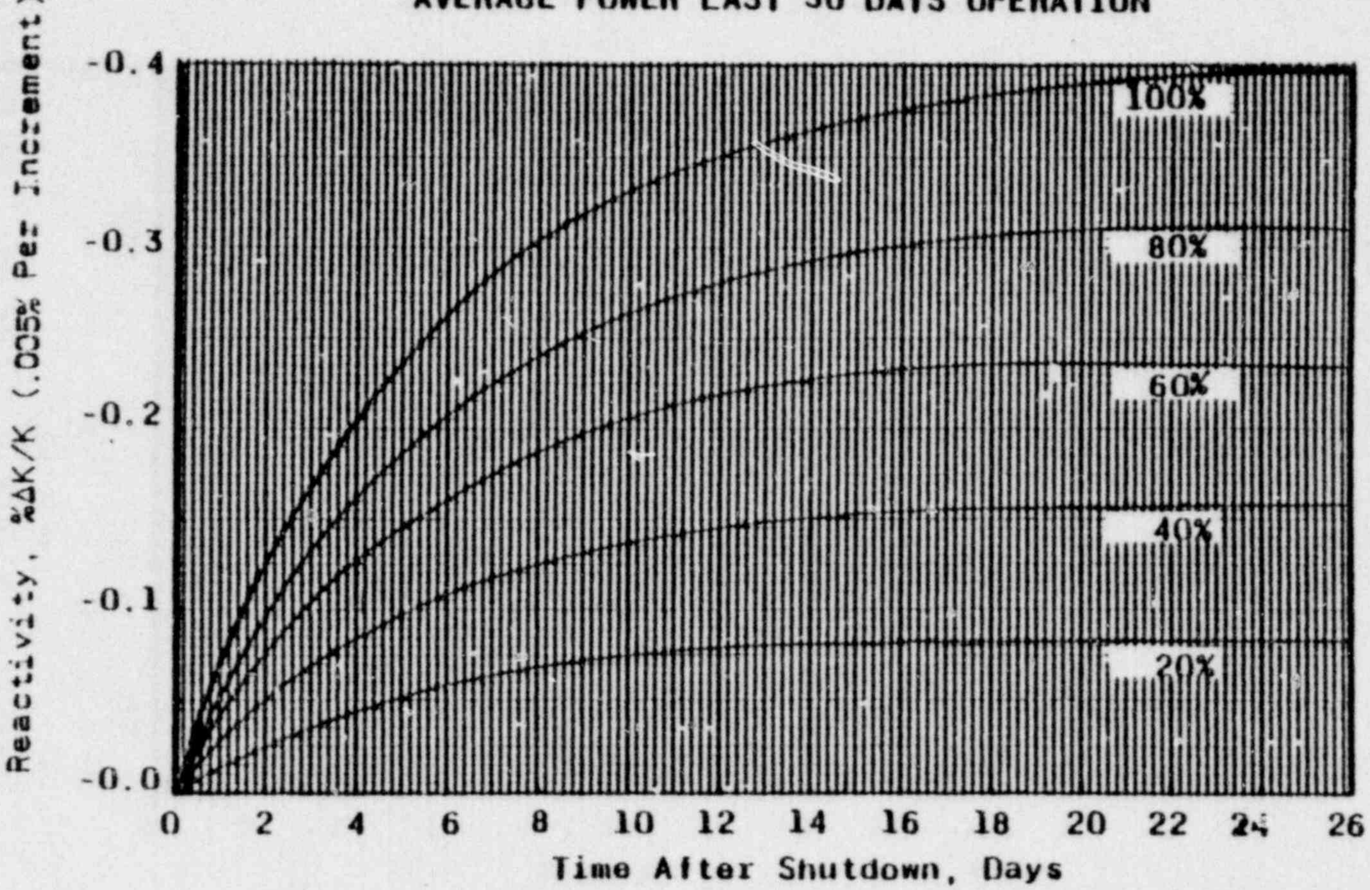


Figure 7a REACTIVITY GROUP 8 VS ROD POSITION BOL, 532°F

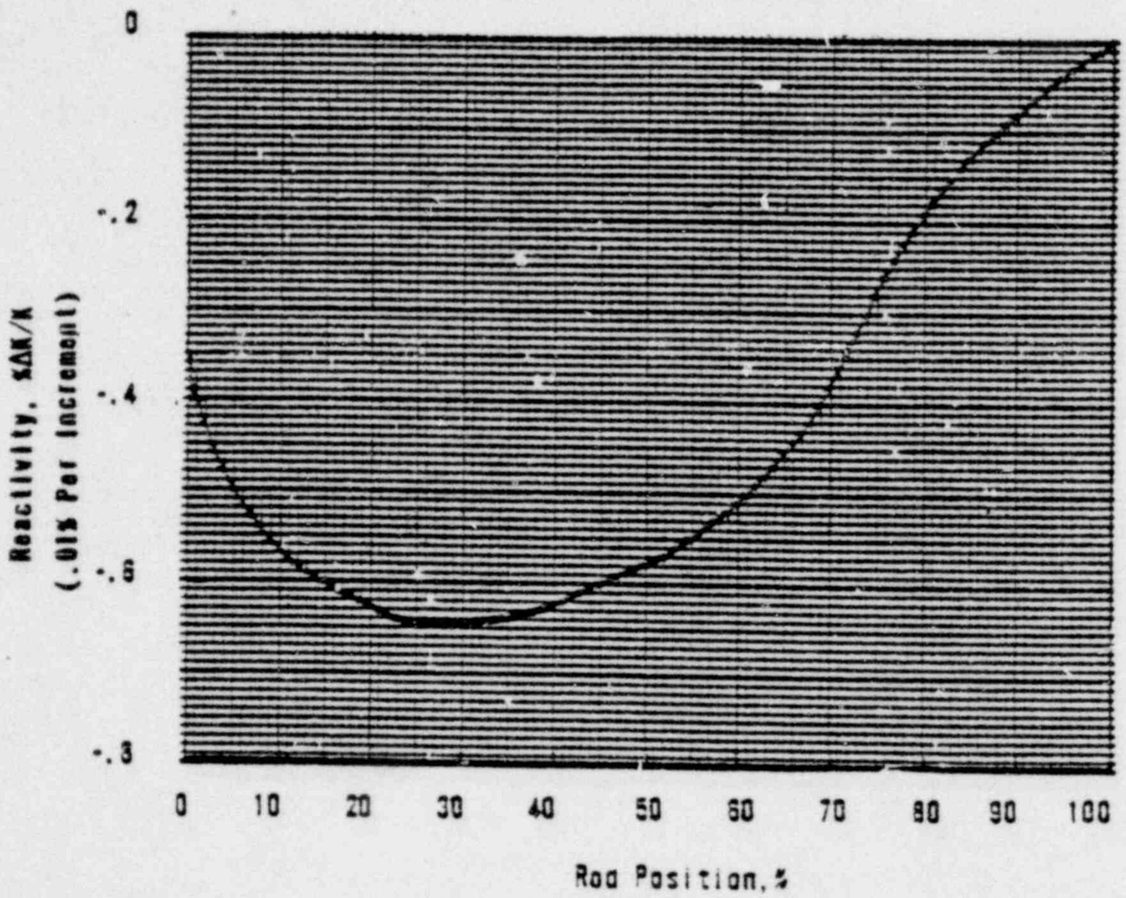


Figure 7b REACTIVITY GROUP 8 VS ROD POSITION MOL 532°F

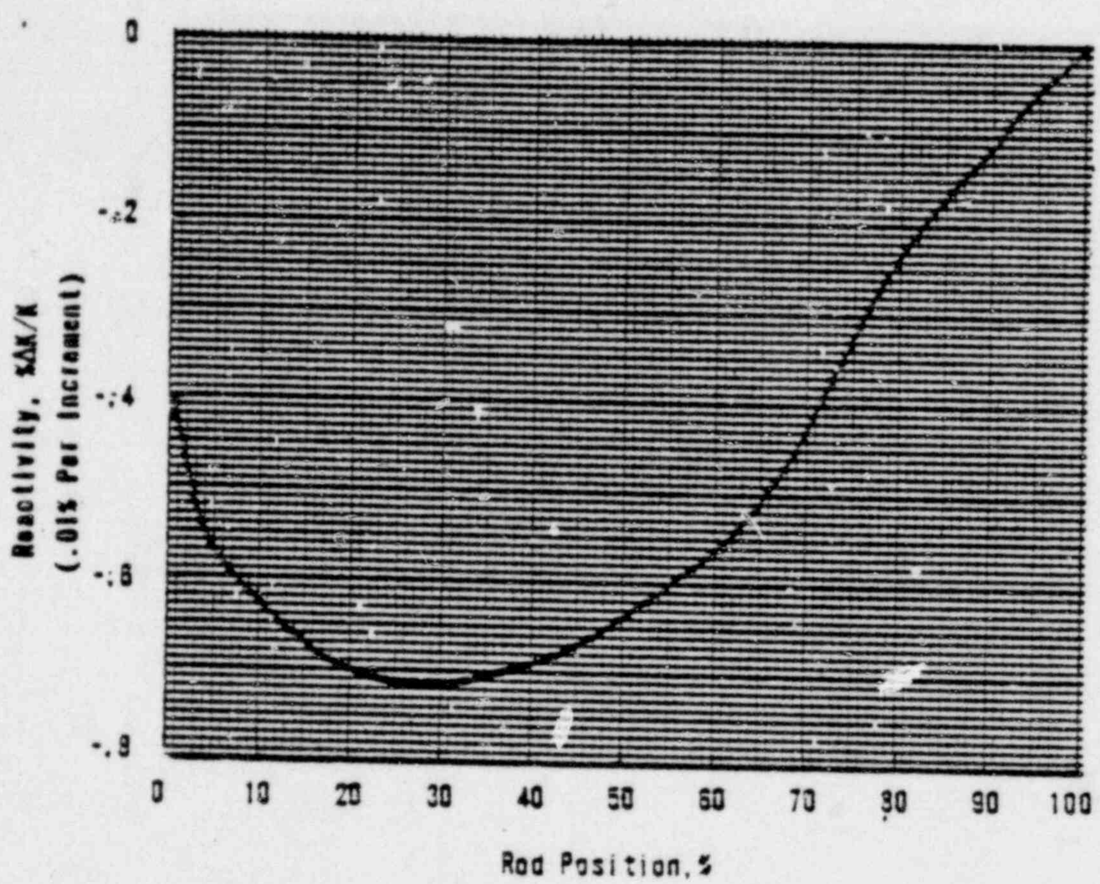


Figure 7c REACTIVITY GROUP 8 VS ROD POSITION EOL, 532°F

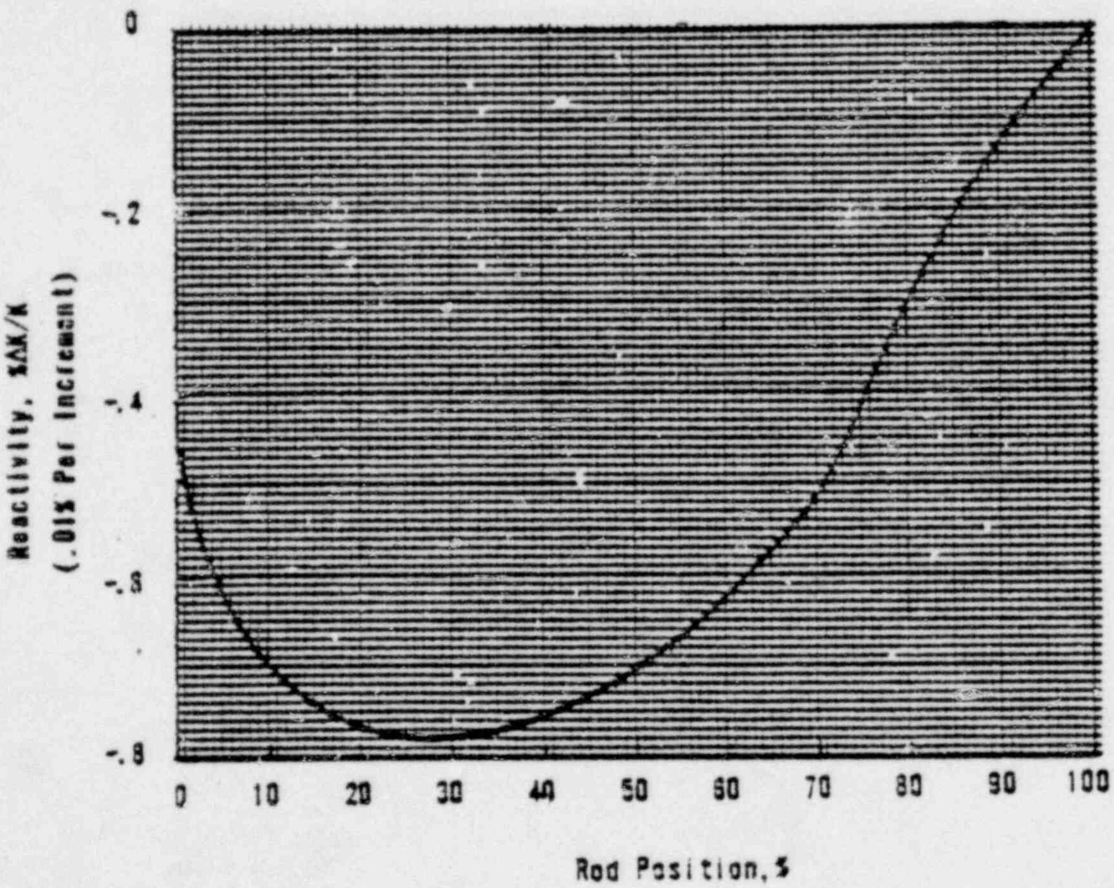


Figure 7d REACTIVITY GROUP 8 VS ROD POSITION BOL, 582°

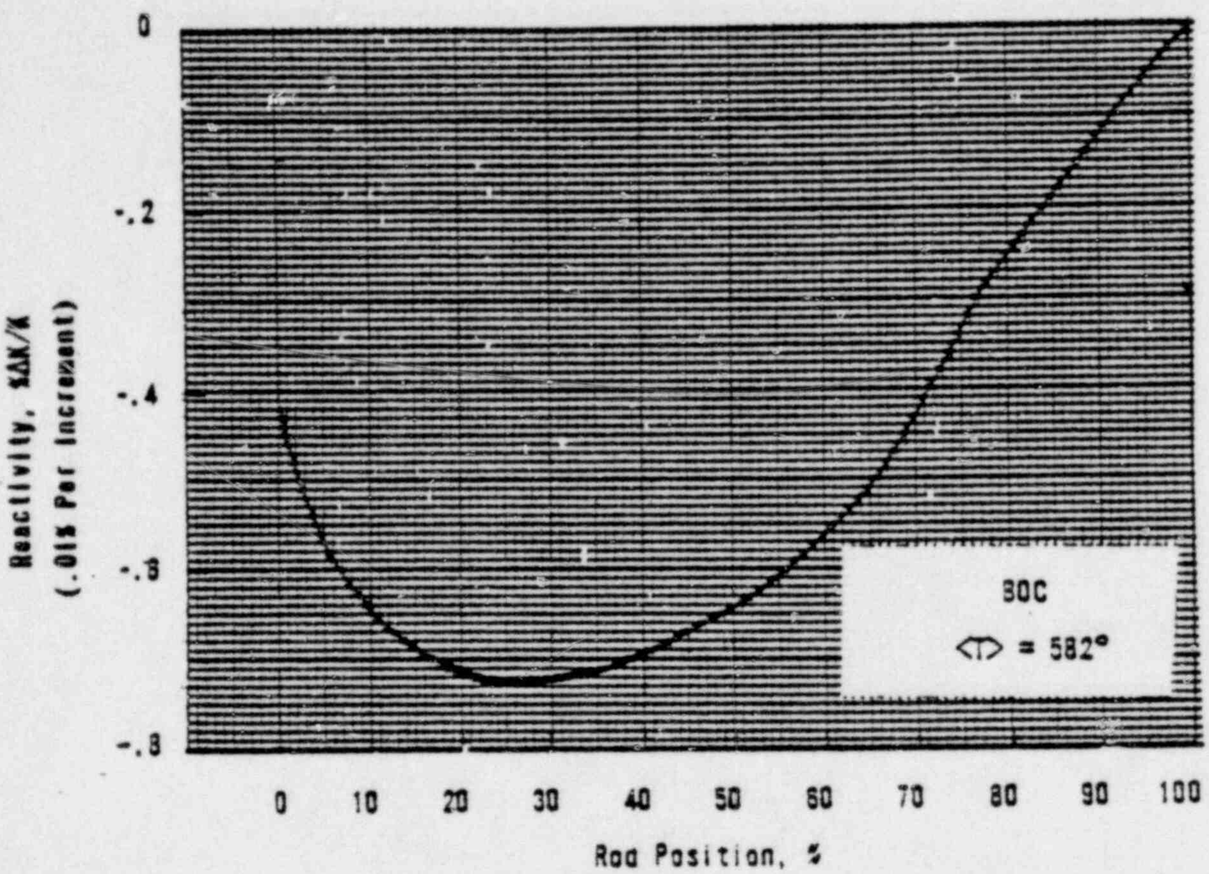


Figure 7a REACTIVITY GROUP 8 VS ROD POSITION MOL, 582°

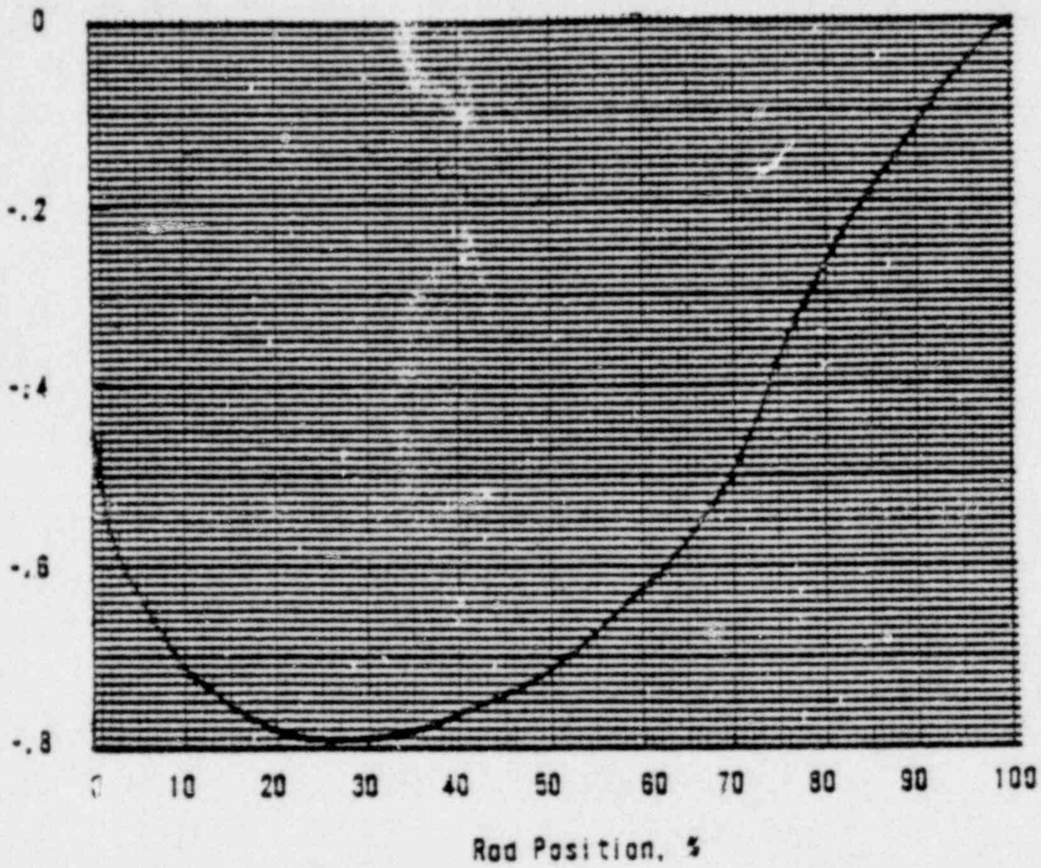


Figure 7f REACTIVITY GROUP 8 VS ROD POSITION EOL, 582°

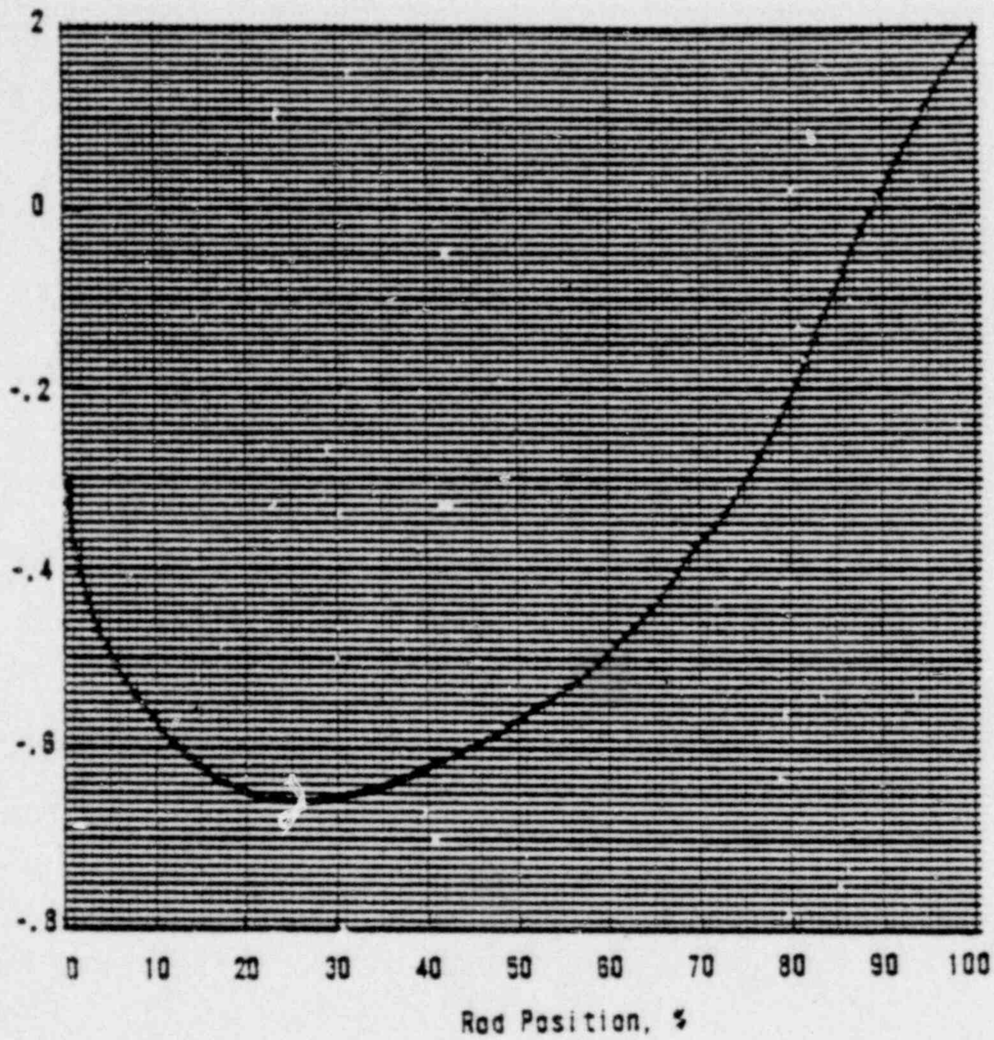


Figure 8a REACTIVITY VS CONTROL ROD POSITION BOL, 532°

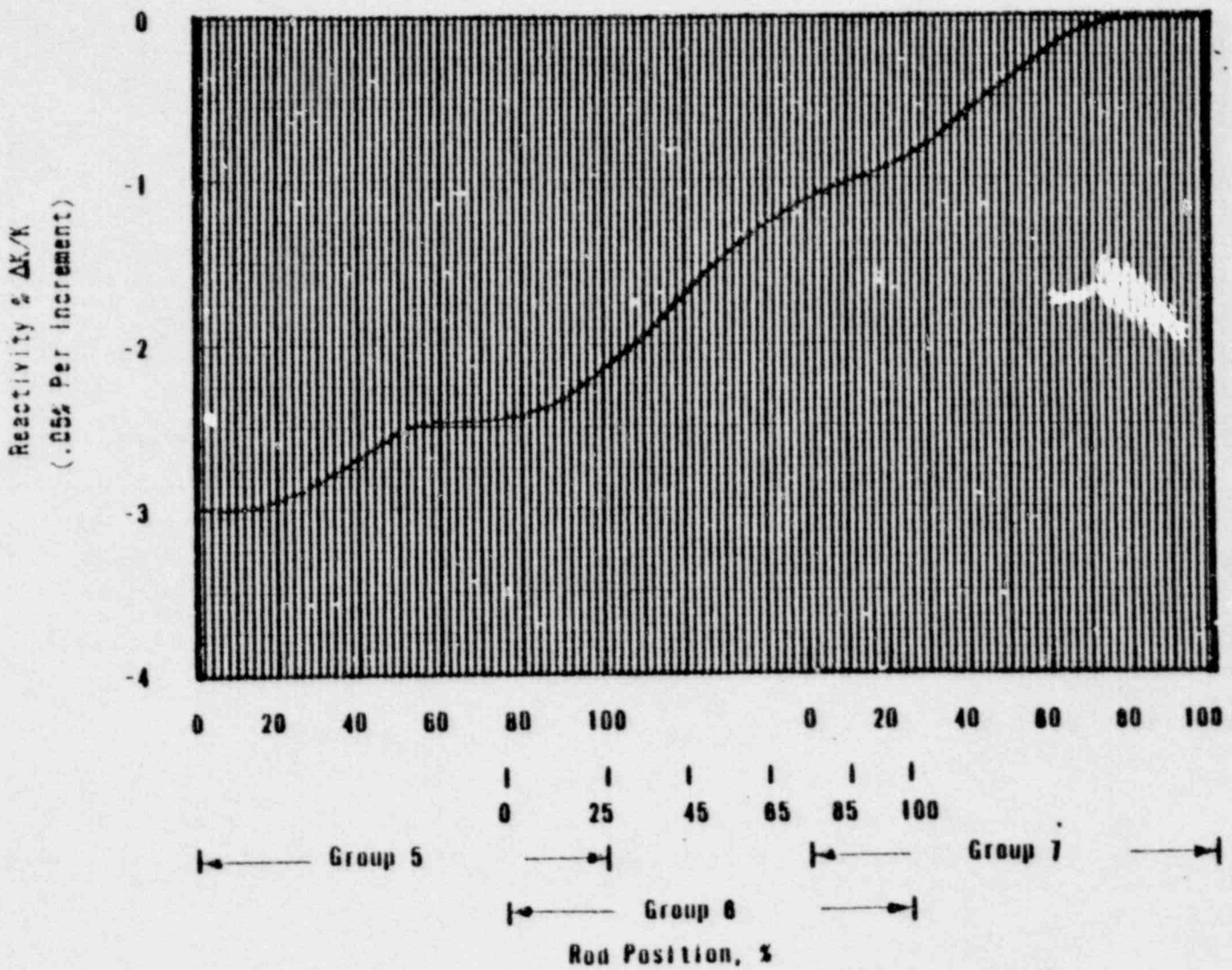


Figure 8b REACTIVITY VS ROD POSITION MOL 532°F

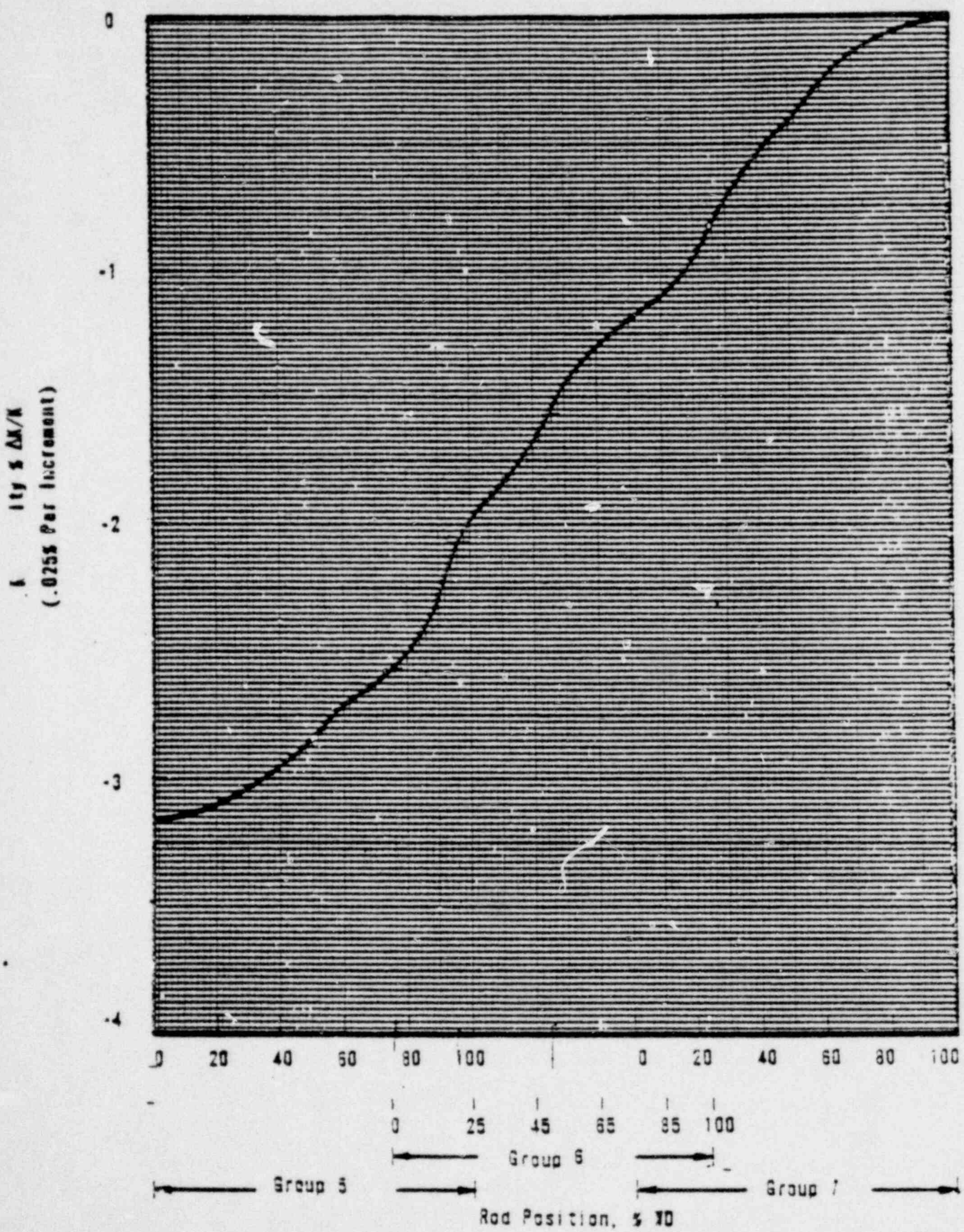


Figure 8c REACTIVITY VS CONTROL ROD POSITIONS EOL, 532°F, GROUP 8 100%

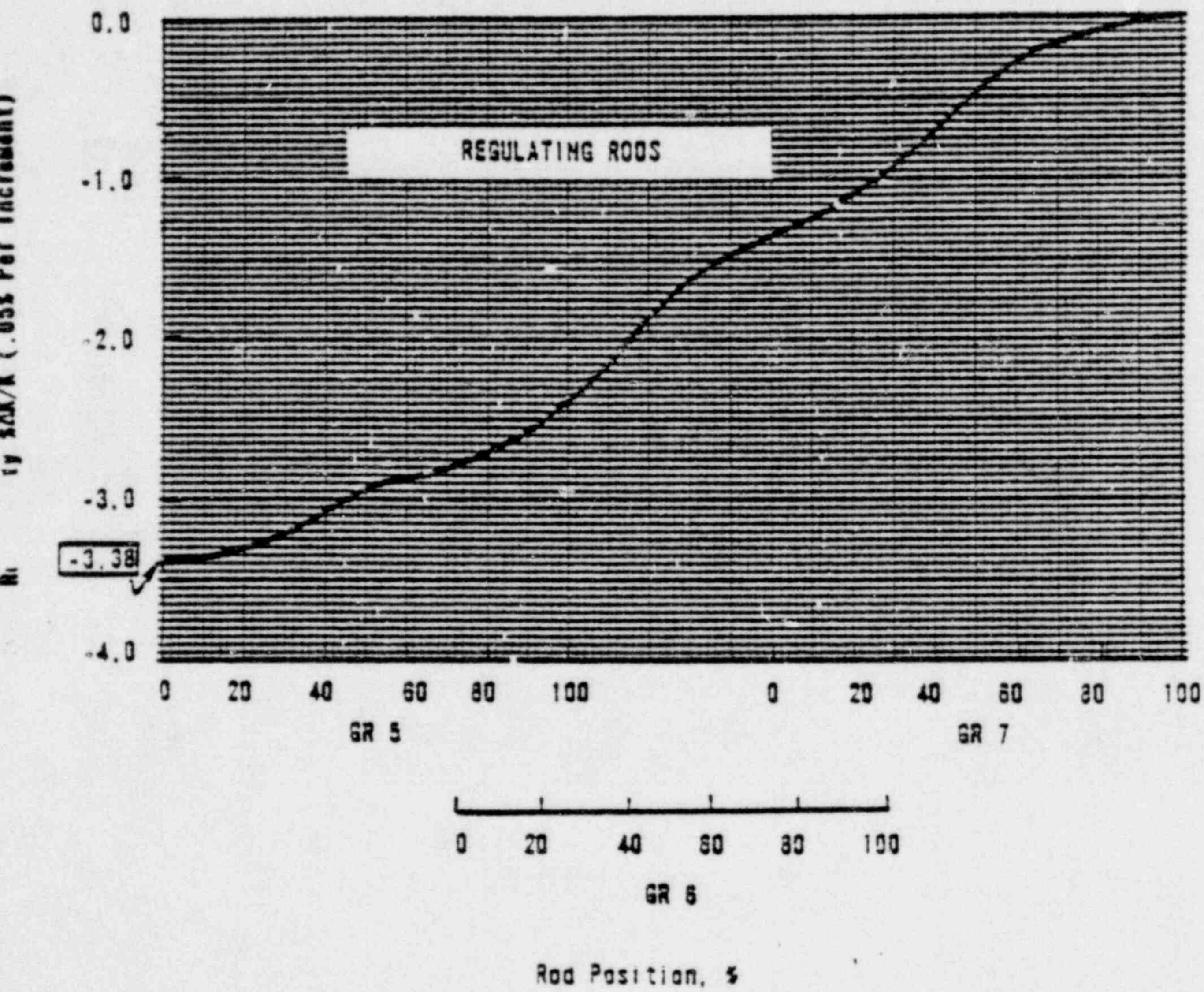


Figure 8d REACTIVITY VS CONTROL ROD POSITION BOL, 582°F, GR8 @ 100%

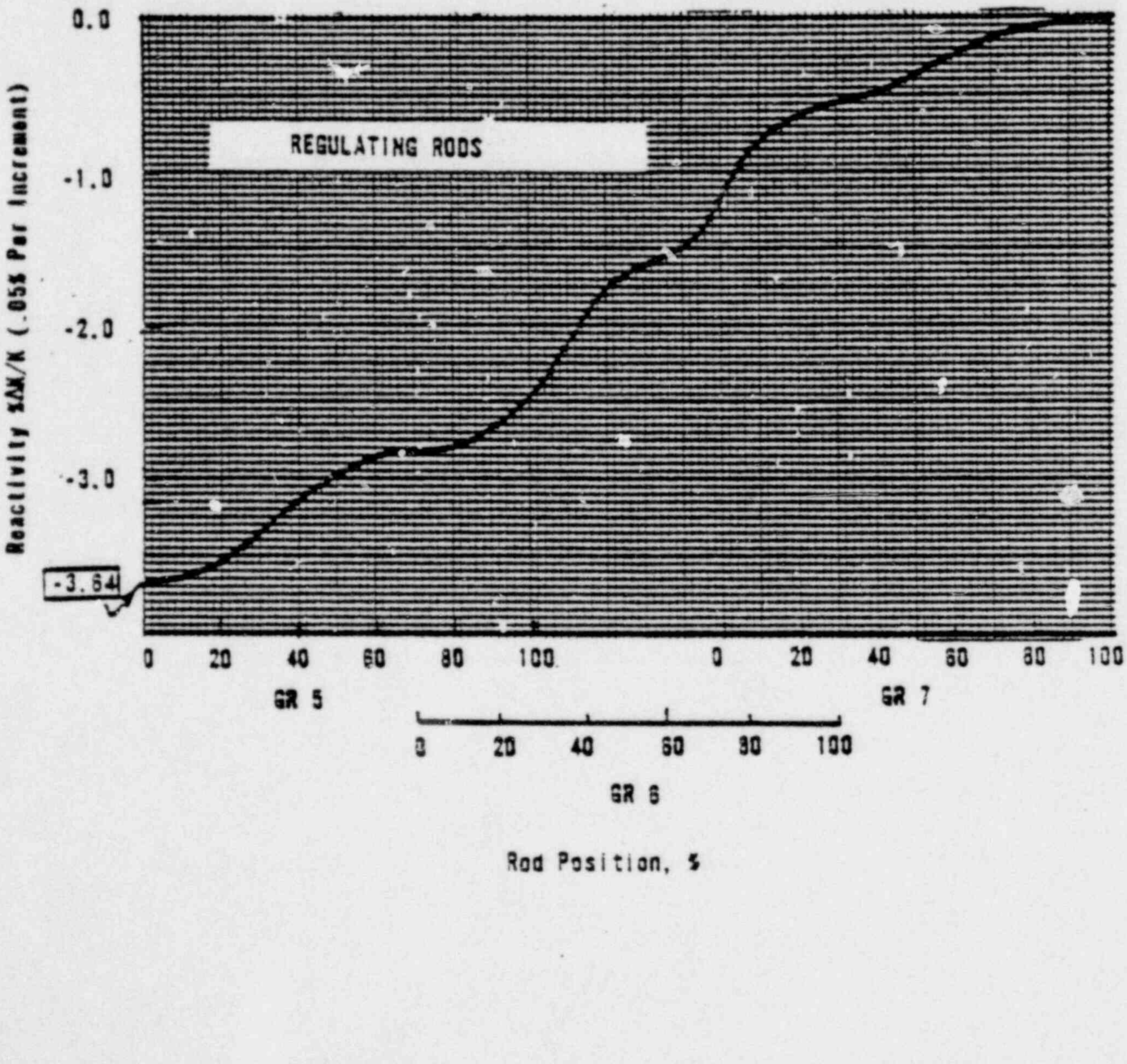


Figure 8e REACTIVITY VS CONTROL ROD POSITION MOL, 582°F, GR8 @ 100%

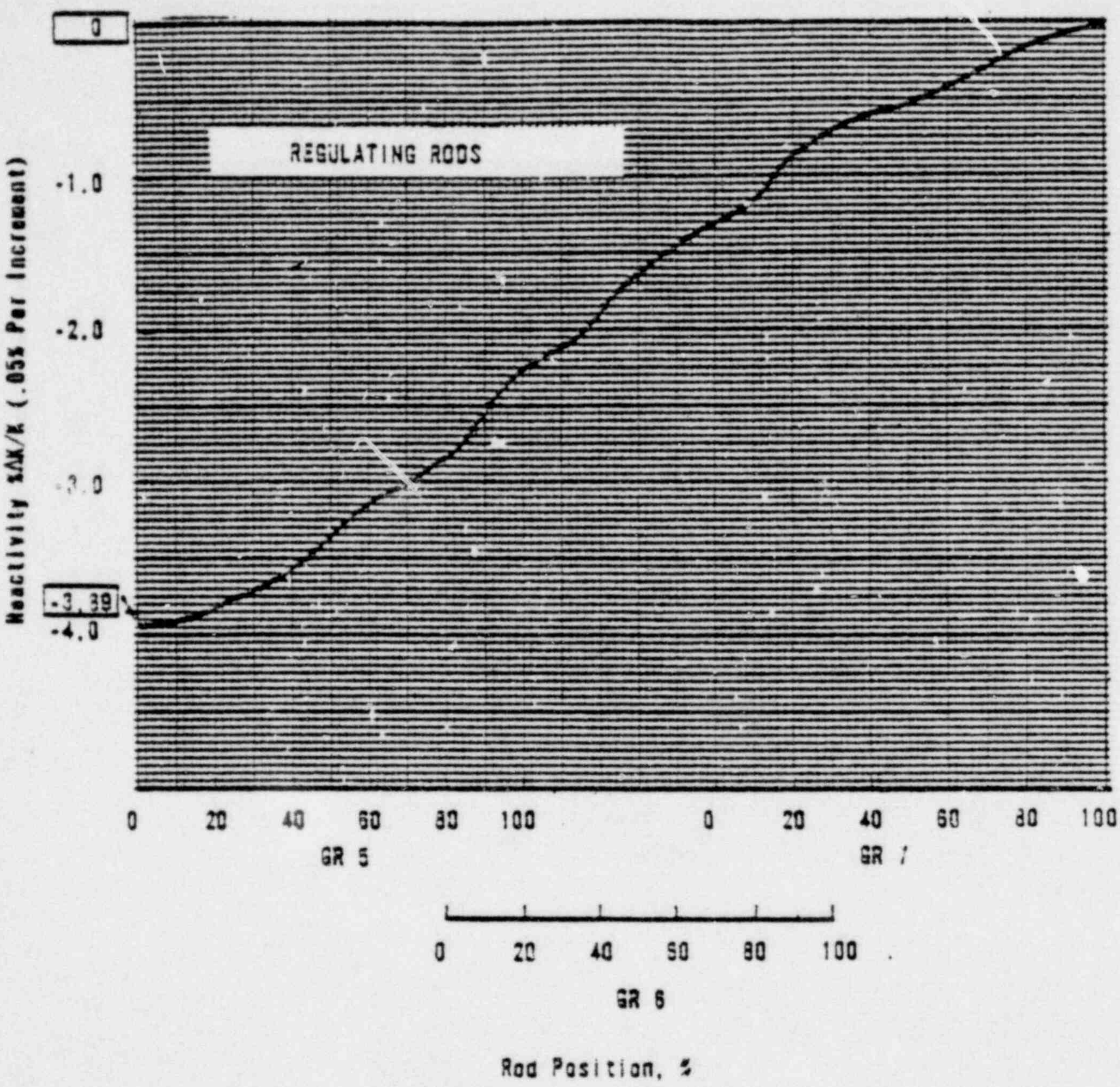


Figure 8f REACTIVITY VS CONTROL ROD POSITION EOL, 582°F, GR8 @ 100%

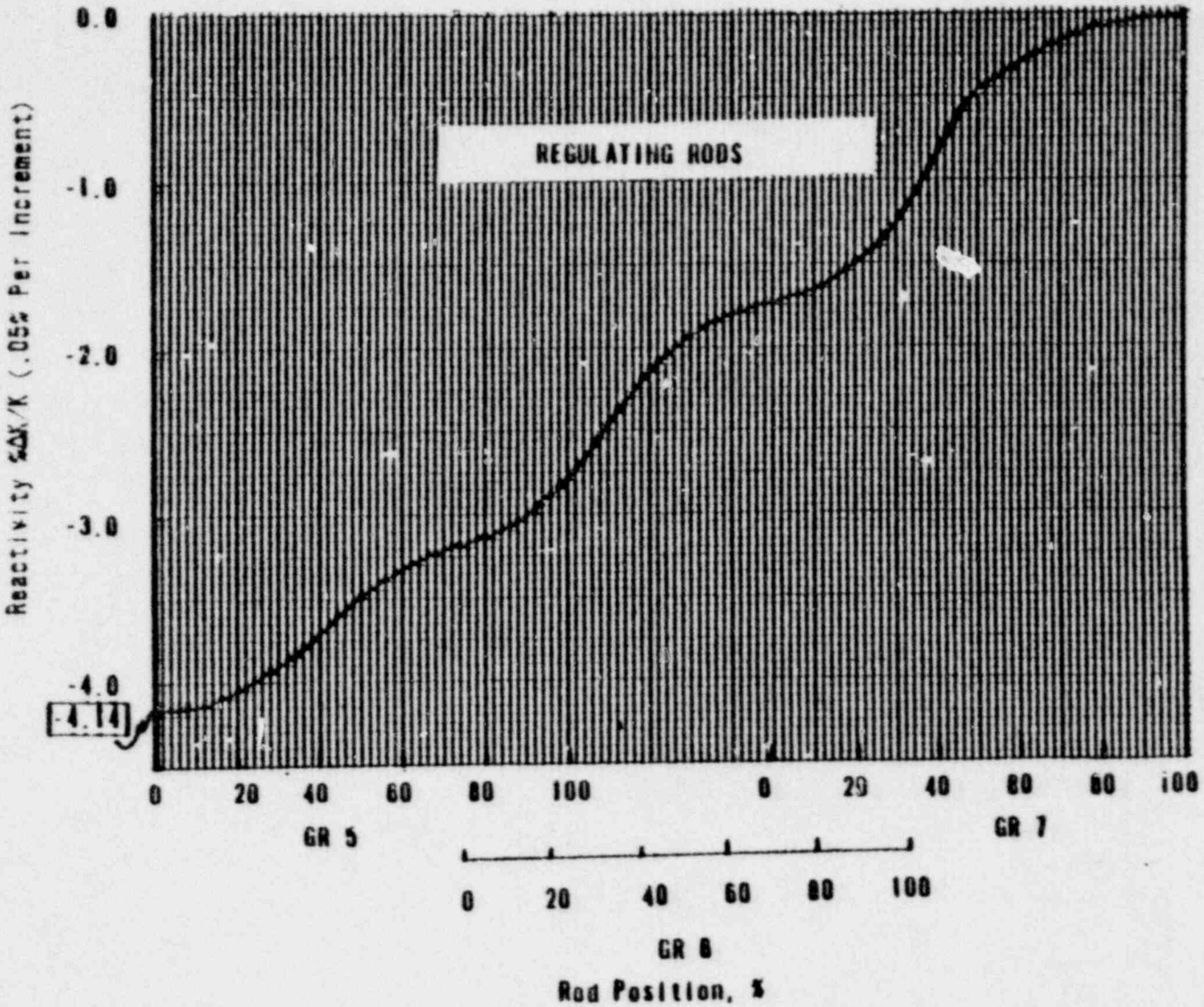


Figure 9a REACTIVITY VS CONTROL ROD POSITION BOL, 532°F, GR8 @ 100%

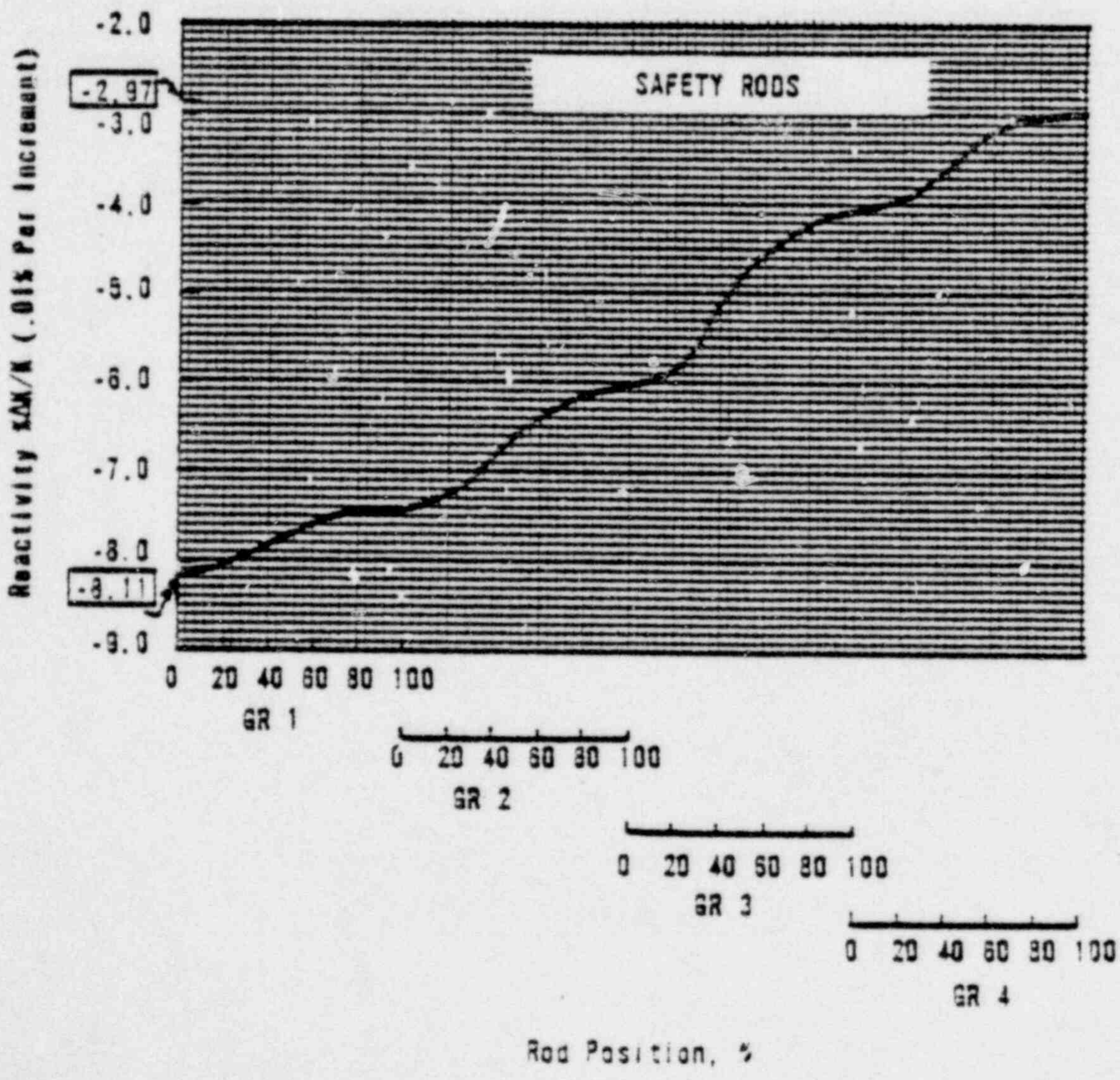


Figure 9b REACTIVITY VS CONTROL ROD POSITION MOL, 532°F, GR8 @ 100%

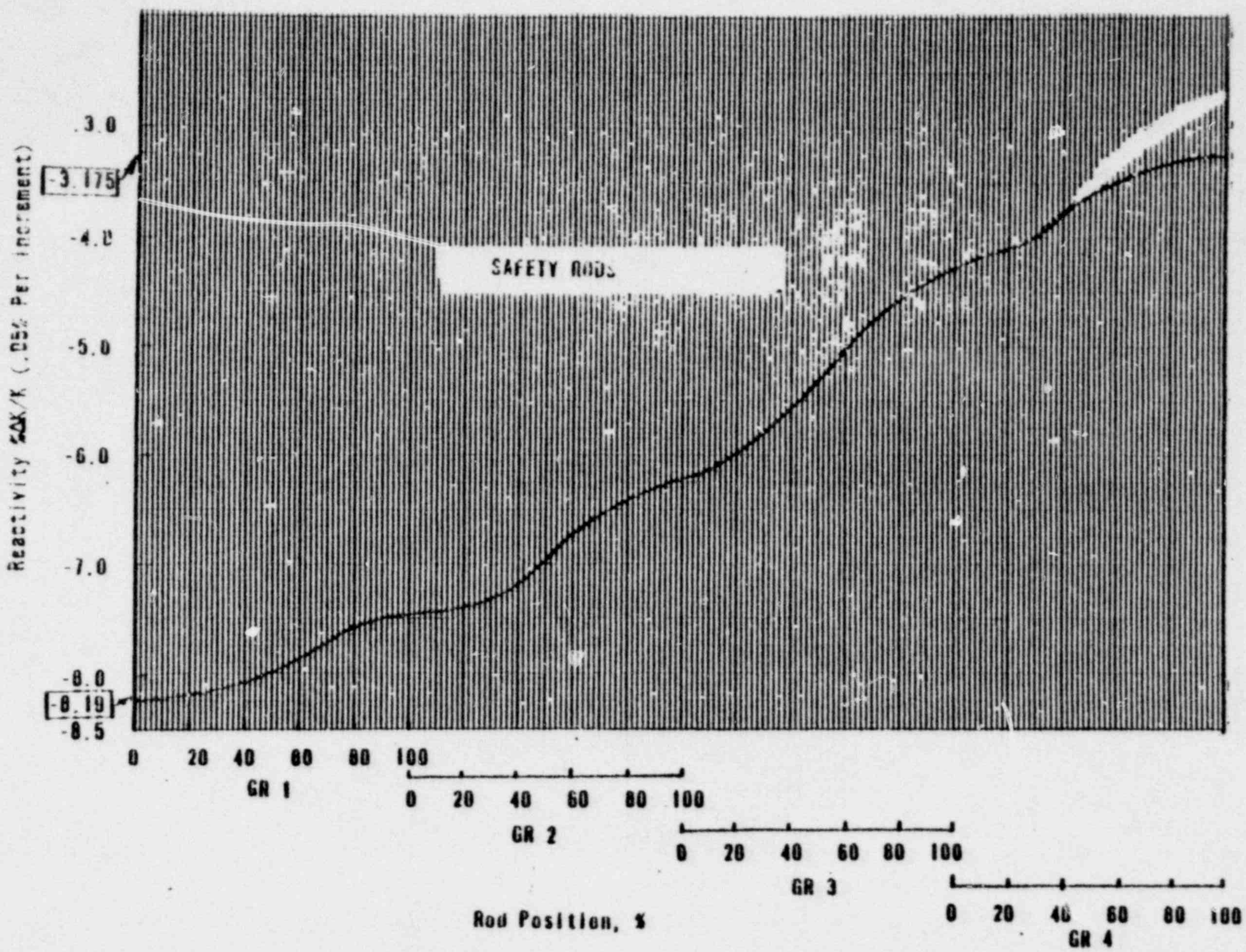


Figure 9c REACTIVITY VS ,ROL ROD POSITION EOL, 532°F, GR8@100%

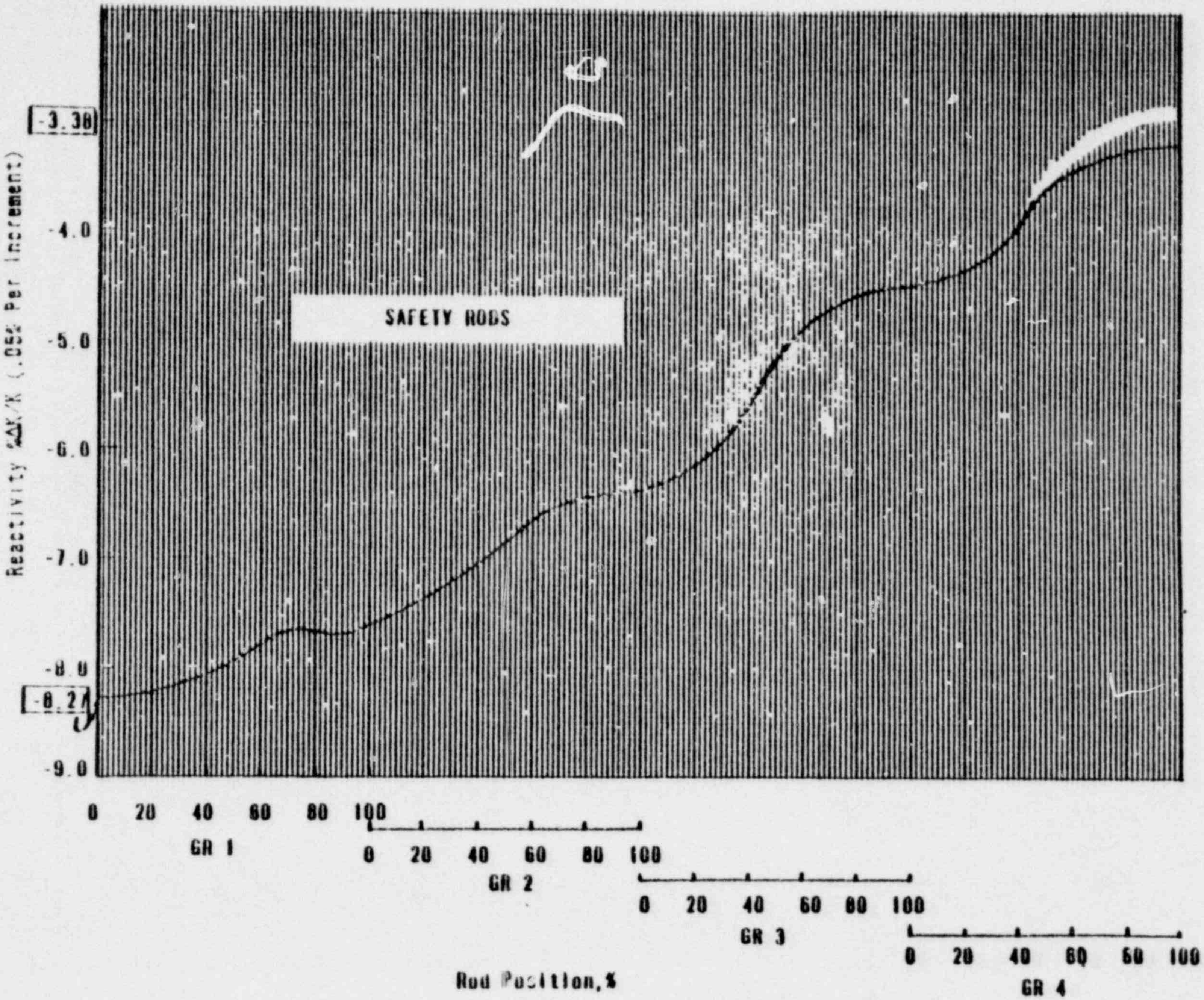


Figure 9d REACTIVITY VS CONTROL ROD POSITION BOL, 582°F, GR8 @ 100%

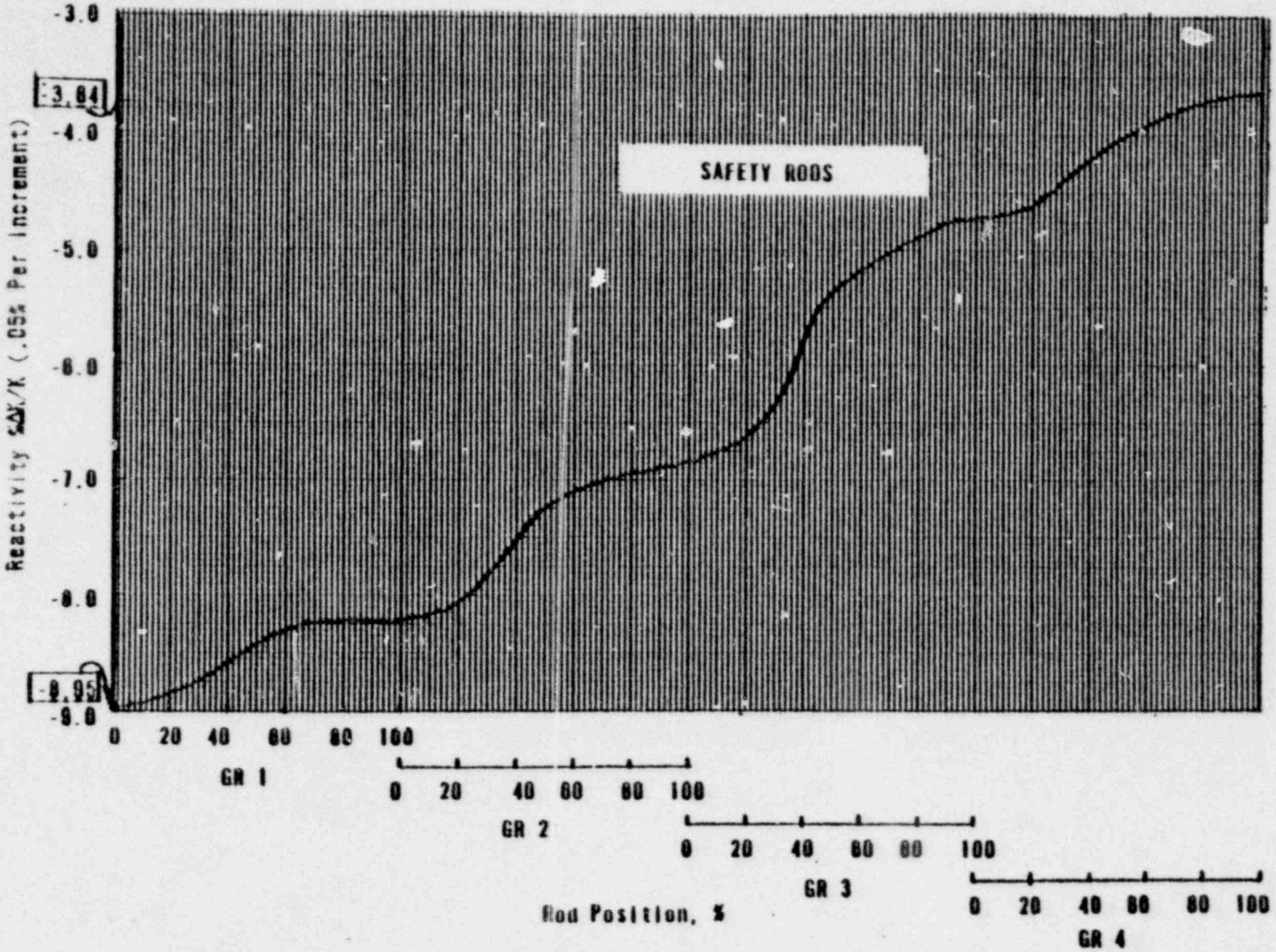


Figure 9a REACTIVITY VS CONTROL ROD POSITION MOL, 582°F, GR8 @ 100%

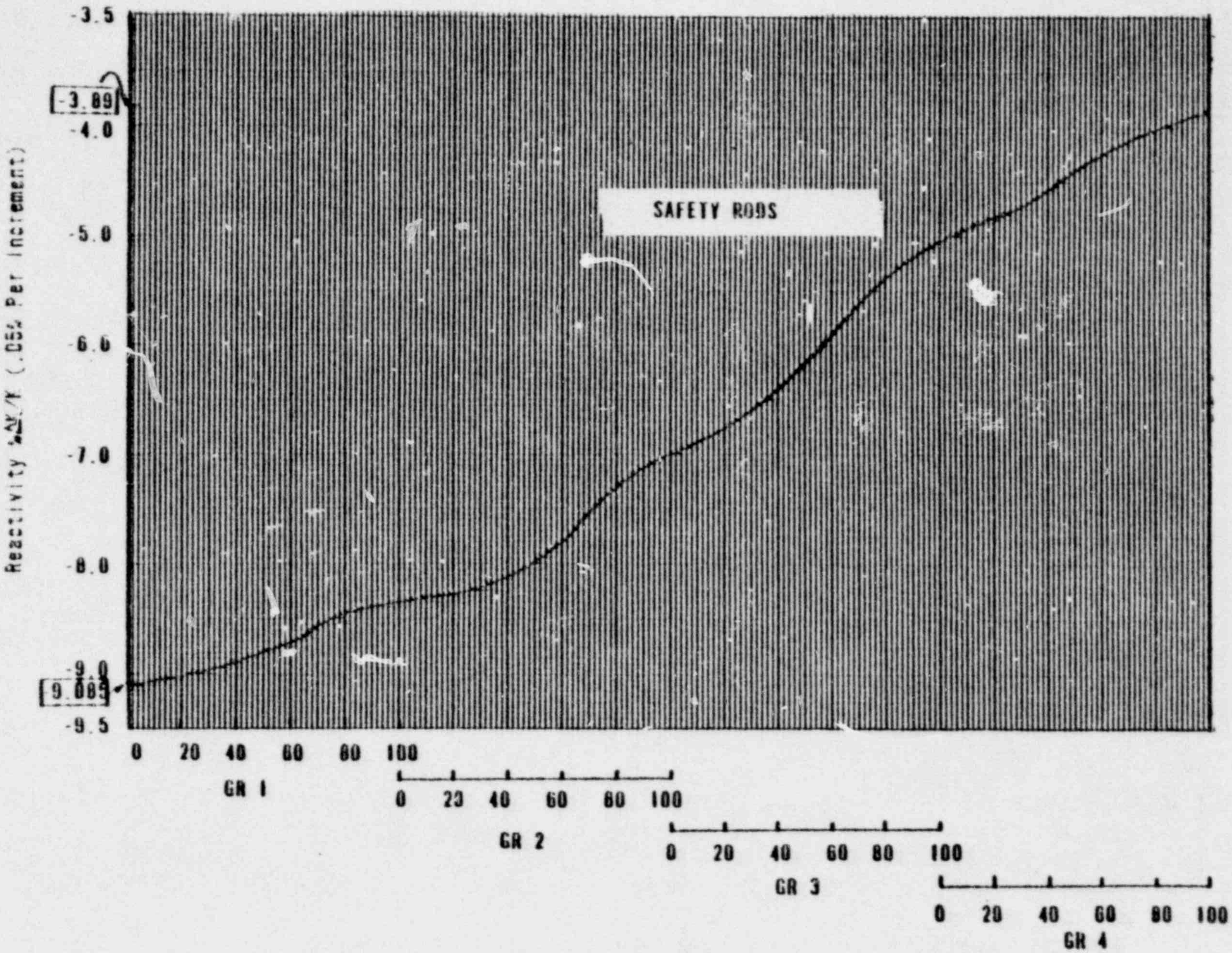
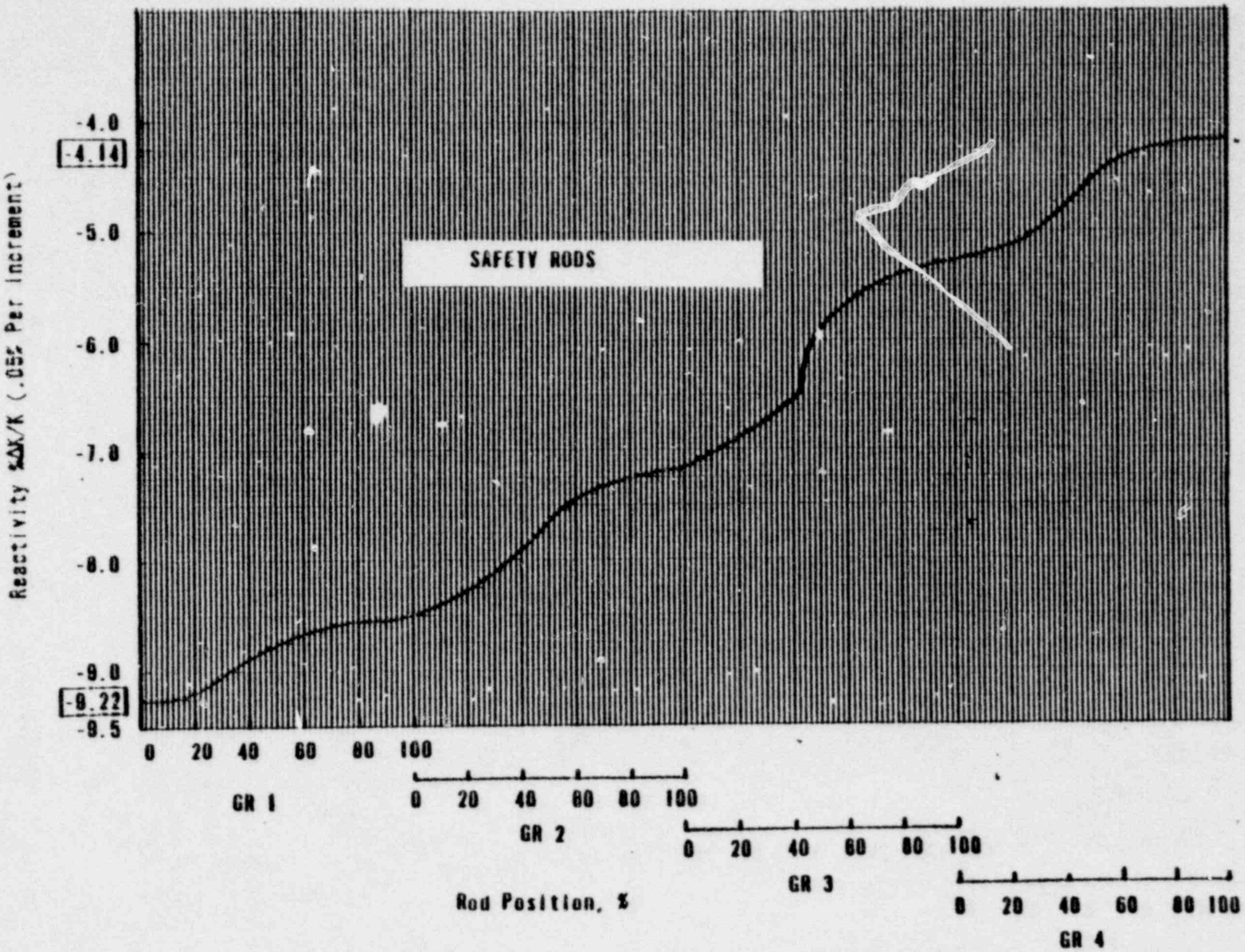


Figure 9f REACTIVITY VS CONTROL ROD POSITION EOL, 582°F, GR8 @ 100%



1105 09

OFR UNIT #1

CONTROL ROD DRIVE SYSTEM

OP 1105 09

1. PURPOSE

1.1 To provide procedures for the normal CRD System Operations listed below:

OPERATION	SECTION
Rod Withdrawal for Plant Startup	4
Transfer to Auxiliary Power Supply	5
Transer from Auxiliary Power Supply	6
Latching and PI Alignment	7
Rod Insertion for Plant Shutdown	8
Axial Power Shaping Group Movement	9

1.2 CRD System Abnormal Operations (OP 1202 10) provides procedures for performing operations related to stuck rod, asymmetric/dropped rod condition.

(TS) 1.3 Exercising Control Rods at Power (OP 1105 11) provides requirement and procedures for exercising control rods during power operations.

2. REFERENCES

2.1 EP 1202 10 - CRD Malfunction Action

2.2 OP 1102 02 - Plant Heatup

2.3 OP 1102 03 - Plant Startup

2.4 OP 1102 11 - Plant Cooldown

2.5 Control Rod Drive Control System, Diamond Power Specialty Corp.

3. LIMITS AND PRECAUTIONS

3.1 CRD Cooling Water must be in operation prior to energizing any CRD's.

3.2 A control rod drive must be de-energized immediately if the stator temperature has reached the upper temperature limit of 180°F.

3.3 CRD's shall not be operated until the Reactor Coolant System final venting is completed and the minimum temperature/pressure conditions for CRD operation are satisfied. (Enclosure 2).

3.4 If an event requiring RCS venting occurs, all CRD's shall be run in (not tripped) to the inlimit.

3.5 If a control rod drive mechanism malfunctions, stop all CRDM operations and check for cause before proceeding.

3.6 Verify that reactor shutdown margin (i.e. RCS boron concentration) requirements are met prior to CRD withdrawal.

3.7 Maximum CRD travel is 420 in/hr. Maximum CRD duty cycle is 30 min/hr.

3.8 Each control rod drive mechanism shall be exercised by a movement of approximately two inches of travel every two weeks. This requirement shall apply to a partial or fully withdrawn control rod at reactor operating conditions.

3.9 If a control rod cannot be exercised, or if neither absolute or relative position indication is operable, the control rod shall be declared inoperative.

3.10 Following each refueling, prior to return to power, the control rod trip insertion time will be checked and shall not exceed 1.66 seconds trip time (3/4 stroke insertion) at hot reactor conditions. If this trip time is not met, the rod shall be declared inoperative.

3.11 The APSR's (Group 8) will be checked to demonstrate that loss of power will not cause rod movement.

NOTE:

Never attempt to withdraw a CRA in JOG speed as the available motor torque is sufficient to destroy the CRA if either the "poison pins" or "spider" is actually stuck.

3.12 If a control rod is misaligned with its group average position by more than an indicated nine inches, the rod shall be declared inoperative until it has been repositioned to the group average position. Refer to Tech Specs to determine the exact action that must be taken if this condition exists.

3.13 If a control rod is declared inoperative, an evaluation shall be initiated immediately to verify the existance of a 1% $\Delta k/k$ or greater shutdown margin.

3.14 With an inoperative control rod the remaining rods must be exercised within 24 hours and weekly thereafter until all rods are operable.

3.15 Reactor operation with more than one inoperative control rod shall not be permitted.

3.16 If the minimum temperature/pressure curve of Enclosure 2 is violated, the CRD must not be tripped except if required for core protection. The drives must be driven to the "in"limit and de-energized.

4. ROD WITHDRAWAL FOR PLANT STARTUP

4.1 Initial Conditions:

(1) CRDs are de-energized (tripped).

(2) System logic cabinets power supplies outputs indicate normal and all control circuits are energized.

(3) Main and secondary power output switches at rear of each power supply cabinet are closed.

(4) Bus duct power disconnect switches are closed.

4.2 Procedure:

(1) Verify the status of the PI panel is as follows:

(a) The 100% lamps for groups 1-7 are EXTINGUISHED.

(b) The CONTROL ON lamps for groups 1-8 are EXTINGUISHED.

(c) The 0% lamps for groups 1-7 are ILLUMINATED.

(d) The POSITION SELECT switch is in the ABSOL position.

(e) Absolute Position Indication (API) for groups 1 thru 7 is 0%.

(f) The indications associated with group 8 agree with the current status of group 8.

(2) Press TRIP RESET switch.

CAUTION

Pressing the TRIP RESET will remove the +125 psig bias from the turbine bypass valves.

(3) Press FAULT RESET switch.

(4) Verify the status of the rod control panel is as follows:

(a)

LIGHT	STATUS
Trip	OFF
Asymmetric Rods	OFF
Motor Fault	OFF
Out Inhibit	OFF
Sequence Inhibit	OFF
Auto Inhibit	ON
Programmer Lamp Fault A&B	OFF
Xenon Overlap Fault	NA
Out Limit (Groups 1-7)	OFF
Control ON	OFF
In Limit (Groups 1-7)	ON
System Power Supply	ON
Motor Power Supply	ON
CRD Travel In and Out	OFF

NOTE:

Group 8 inlimit lights may be on or off depending on plant status.

(b) SWITCH (Back lighted pushbutton)

In Limit Bypass	OFF
Transfer Reset	OFF
Fault Reset	OFF
Trip Reset	ON
Man Trans/Sync/Trans Coef.	OFF
Sequence/Seq. or	SEQUENCE
Group/Auxiliary	(GROUP)
Auto/Manual	(MANUAL)
Clamp/Clamp Release	(CLAMP RELEASE)

(c) SWITCHES (Rotating)

Speed Selector	JOG
Group Position Selector	SAFETY
Xenon Control	(Not Used)
Single Select	OFF
Group Select	OFF

(5) Transfer group 1 to the auxiliary power supply using Section 5. of this procedure.

(6) Latch and align PI for group 1 using Section 7. of this procedure.

(7) Set SPEED SELECTOR switch to RUN. Position INSERT-WITHDRAW switch to WITHDRAW. Note that the CRD TRAVEL OUT lamp illuminates while switch is positioned to WITHDRAW, IN LIMIT lamp and 0% lamps for the selected group are extinguished, and the group PI meter and individual PI meters follow the withdrawal of the selected group.

(8) Release INSERT-WITHDRAW switch when the OUT LIMIT lamp for the selected group illuminates. Note that CRD TRAVEL OUT lamp extinguishes, the selected group's 100% lamps are illuminated, and the group PI meter and individual PI meters for the selected group indicate 100%.

(9) Transfer group 1 from the auxiliary power supply using Section 6. of this procedure.

(10) Transfer group 2 to the auxiliary power supply using Section 5 of this procedure.

(11) Latch and align PI for Group 2 using Section 7. of this procedure.

(12) Repeat Steps 7 and 8.

(13) Transfer group 2 from the auxiliary power supply using Section 6. of this procedure.

(14) Transfer group 3 to the auxiliary power supply using Section 5 of this procedure.

(15) Latch and align PI for group 3 using Section 7. of this procedure.

(16) Repeat Steps 7 and 8.

(17) Transfer group 3 from the auxiliary power supply using Section 6. of this procedure.

(18) Transfer group 4 to the auxiliary power supply using Section 5 of this procedure.

(19) Latch and align PI for group 4 using Section 7. of this procedure.

(20) Repeat Steps 7 and 8. Note that AUTO INHIBIT lamp extinguishes ROD CONTROL SYS alarm clears when group 4 is fully withdrawn.

(21) Transfer group 4 from the auxiliary power supply using Section 6. of this procedure.

(22) Set the GROUP POSITION SELECTOR to REGULATE.

(23) Position GROUP SELECT switch to 5. Note that the group CONTROL ON lamp for group 5 illuminates and individual CONTROL ON lamps for group 5 illuminate.

(24) Latch and align PI for group 5 using Section 7. of this procedure.

(25) Position GROUP SELECT switch to 6. Note that the group CONTROL ON lamp for group 5 extinguishes, the group CONTROL ON lamp for group 6 illuminates, and the individual CONTROL ON lamps for group 6 illuminate.

(26) Latch and align PI for group 6 using Section 7. of this procedure.

(27) Position GROUP SELECT switch to 7. Note that the group CONTROL ON lamp for group 6 extinguishes, the group CONTROL ON lamp for group 7 illuminates, and the individual CONTROL ON lamps for group 7 illuminates.

(28) Latch and align PI for group 7 using Section 7. of this procedure..

(29) Position GROUP SELECT and SINGLE SELECT switch to OFF.

(30) Position SPEED SELECTOR switch to RUN.

(31) Press SEQUENCE/SEQ OR switch to SEQUENCE mode.

NOTE

The Control Rod Drive System is now conditioned for the sequence withdrawal of the regulating groups to bring the reactor up to the correct power level.

(32) Refer to OP 1102 03 for Plant Startup procedure.

5. TRANSFER TO AUXILIARY SUPPLY

5.1 Position GROUP SELECT switch to the group to be transferred.

5.2 Position SINGLE SELECT switch to the mechanisms (1 through ALL) to be transferred.

5.3 Press AUTO-MANUAL switch to MANUAL mode. Note that the MAN lamp illuminates and the AUTO lamp extinguishes.

5.4 Press SEQUENCE-SEQ OR switch to SEQ OR mode. Note that the SEQ OR lamp illuminates.

5.5 Press GROUP-AUXILIARY switch to AUXILIARY MODE. Note that the AUXILIARY lamp illuminates, GROUP lamp extinguishes, and the group CONTROL ON lamp for the selected group illuminates.

5.6 Position SPEED SELECTOR switch to JOG. Note that the SYNCH lamp illuminates.

5.7 Press CLAMP-CLAMP REL switch to CLAMP mode. Note that the CLAMP lamp illuminates.

5.8 Press MAN TRANS switch. Note that the MAN TRANS lamp illuminates momentarily, TRANS CONF (transfer confirm) lamp illuminates, and individual CONTROL ON lamps for the mechanisms transferred illuminate.

5.9 Press CLAMP-CLAMP REL switch to CLAMP REL mode. Note that the CLAMP lamp extinguishes.

5.10 Press GROUP-AUXILIARY switch to GROUP mode. Note that the GROUP lamp extinguishes, AUXILIARY lamp extinguishes, and SYNCH lamp extinguishes.

6. TRANSFER FROM AUXILIARY SUPPLY

6.1 Position GROUP SELECT switch to the group to be transferred.

6.2 Position SINGLE SELECT switch to the number of mechanisms (1 through ALL) to be transferred.

6.3 Press AUTO-MANUAL switch to MANUAL mode. Note that MANUAL lamp illuminates and AUTO lamp extinguishes.

6.4 Press SEQUENCE-SEQ OR switch to SEQ OR mode. Note that SEQ OR lamp illuminates.

6.5 Press GROUP-AUXILIARY switch to AUXIL mode. Note that AUXILIARY lamp illuminates and GROUP lamp extinguishes.

6.6 Position SPEED SELECTOR switch to JOG. Note that SYNCH lamp illuminates.

6.7 Press CLAMP-CLAMP REL switch to CLAMP mode. Note that CLAMP lamp illuminates.

6.8 Press MAN TRANS switch. Note that MAN TRANS lamp illuminates momentarily, TRANS CONF lamp extinguishes, and the individual CONTROL ON lamps for the selected rods extinguishes.

6.9 Press CLAMP-CLAMP REL switch to CLAMP REL mode. Note that CLAMP lamp extinguishes.

6.10 Press GROUP-AUXILIARY switch to GROUP mode. Note that GROUP lamp illuminates, AUXILIARY lamp extinguishes, and SYNCH lamp extinguishes.

6.11 Press TRANS RESET switch. Note that TRANS RESET lamp illuminates while switch is held down.

7. <u>LATCHING AND PI ALIGNMENT</u>

7.1 Press and hold the INLIMIT BYPASS switch. Note that the INLIMIT lamps for groups 1-7 are extinguished.

7.2 Position the INSERT-WITHDRAW switch to the INSERT position for 15 seconds and then return switch to center position.

7.3 Release INLIMIT BYPASS switch.

7.4 Position the INSERT-WITHDRAW switch to WITHDRAW until the INLIMIT for the selected group extinguishes. Then return the switch to the center position.

7.5 Position the INSERT-WITHDRAW switch to INSERT until the inlimit for the selected group illuminates. Then return the switch to the center position.

7.6 Set the position select switch on the PI panel to RELATIVE.

7.7 Adjust the relative PI if necessary, using the RAISE/LOWER switch on the PI panel.

7.8 Set the position select switch to ABSOLUTE.

8. ROD INSERTION FOR SHUTDOWN:

8.1 Initial Conditions:

(1) Groups 5, 6, and 7 have been inserted in sequence per appropriate plant procedures.

(2) Insertion of Safety Groups has been specified by appropriate plant procedures.

8.2 Procedures:

(1) Rotate the group PI Select switch to SAFETY.

(2) Transfer the selected group to the auxiliary power supply using Section 5 of this procedure.

(3) Insert the group selected (4-3-2-1 in that order) until the 0% lights on the PI panel and the "in limit" light on the Diamond panel are on.

(4) Transfer the selected group from the auxiliary power supply using Section 6 of this procedure.

(5) Repeat Steps 2, 3, and 4 for each safety group.

(6) When all rods are inserted, manually trip the reactor to remove power from the CRD Mechanisms.

CAUTION

The steam header pressure controller setpoint will be increased by 125 psi when the reactor is tripped.

9. AXIAL POWER SHAPING RODS (APSR) MOVEMENT:

9.1 To move APSRs (Group 8) when specified by appropriate plant procedures, perform the following:

(1) Rotate the group select switch to 8.

(2) Move Group 8 as necessary by using the insert-withdraw switch.

ENCLOSURE 1

OPERATOR AND PI PANEL
DEFINITION OF TERMS

1. ASYMMETRIC RODS

The individual asymmetric rod alarms at the PI panel indicate a rod is 7 or more inches out of alignment with respect to its group average position. The asymmetric rod indication at the operator panel will be energized if the misalignment is 9 or more inches within any group.

2. AUTO INHIBIT

Indicates control cannot be switched to automatic because the safety groups are not at the out limit or a large neutron error signal is present from the ICS or ICS auto power is not available.

3. CONTROL ON (Group 1 thru 8)

Indicates a particular group has been selected, either manually or due to the automatic sequencer, for control. Unless the control on lamps at the PI panel are also energized, the selected group (or rods within that group) will not respond to insert/withdraw commands.

4. CRD TRAVEL IN/OUT

Indicates an in or out command has been directed to one of the safety or regulating groups. Will not indicate commands associated with Group 8.

5. IN LIMIT (Group 1 thru 8)

Indicates that at least one rod of a particular group is at in limit.

6. MOTOR FAULT

Indicates the power supply programmer motor is running in withdraw direction with an in command, or is running in either direction with no command.

7. OUT INHIBIT

Indicates the CRDs will not respond to out commands. An out inhibit condition may be caused by the following:

> A high start-up rate of 2 DPM in the source range or 3 DPM in the intermediate range;
>
> or
>
> In the auto mode, the safety group is not withdrawn, or an asymm fault with reactor power $\geq$ 60 percent.

8. OUT LIMIT (Group 1 thru 8)

Indicates that at least one rod out of its respective group is at out limit.

9. PROGRAMMER LAMP FAULT A&B

Indicates that one of the 5 programmed power supplies (all regulating supplies plus the auxiliary) has lost one or both of its redundant photo-cell light sources.

10. SEQUENCE INHIBIT

Indicates that withdrawal of the regulating groups is occurring with improper overlap (>25%).

11. SYSTEM AND MOTOR POWER SUPPLIES A&B

Indicates that the 24 vDC, ± 15 vDC and 5 vDC power supplies are available and that power is available on the DC hold buses.

12. TRIP

Indicates the control rod drives are de-energized and, except for group 8, the roller nuts disengated from the lead screw. The AC and DC breakers feeding the CRDs are open and all programmer lamps are off when trip is indicates.

<u>NOTE</u>

The following are integral pushbutton/indicator lamps.
All are momentary contact.

13. AUTO/MAN

Used to select automatic or manual control. To select automatic, the trip, motor fault, sequence inhibit, and auto inhibit lamps must be off. Once automatic is selected, a trip, motor fault, or loss of ICS auto power will revert control to manual. The operator may select manual at any time.

14. CLAMP/CLAMP REL

Used to energize and de-energize the clamping contactors which cross-connect the auxiliary supply with either the DC hold or one of the 4 regulating supplies. Will not go into clamp unless sync is indicated. White indicator lamps indicate the position of the alternate-action switch. Amber and green lamps indicate the clamping contactor as being energized and de-energized, respectively.

15. FAULT RESET

Resets a fault condition if fault has cleared. The faults may be asymmetric rods, motor fault, and motor or system power faults A and B.

16. GROUP/AUX

This switch allows the operation of the selected transfer command circuits. Aux mode must be selected when transfer from or to the auxiliary supply.

17. INLIMIT BYPASS

Disables all group in limits, except group 8, to allow insert motion thereby assuring positive engagement between the roller nuts and the lead screw.

18. MANUAL TRANSFER

Used to transfer individual rods or groups to and from the auxiliary power supply. Will not function unless clamp is indicated.

19. SEQ/SEQ OR

This switch allows the operator to override the regulating group sequence. Override is required in order to select a group for transfer to the aux power supply or to return it to its normal supply.

20. SYNC (LAMP ONLY)

Indicates auxiliary supply is in phase synchronism with either the DC hold supply or one of the regulating supplies.

21. TRANSFER CONFIRM (LAMP ONLY)

Indicates the transfer relay has rotated to a position for transfer of selected CRDs to the auxiliary power supply.

22. TRANSFER RESET

Resets any control on lamp (transfer command relay) that is on and enables the selection of the CRDs selected by the group and single select switch.

23. TRIP RESET

Used to reset the AC and DC trip breakers feeding the CRDs. Will not function unless group 1 thru 7 are at in limit.

NOTE: The following are selector switches not directly connected to any indicator lamp circuits.

24. ABSOLUTE/RELATIVE POSITION SELECTION SWITCH

Used to select API or RPI system for monitoring on the position meters.

25. GROUP POSITION SELECTOR (SAFETY-REGULATE)

Used to select the group average (absolute) position indication for console display. Four separate readouts are available corresponding to the safety groups 1 thru 4 or the regulating groups 5 thru 6.

26. GROUP SELECT SWITCH (POSITIONS 1 THRU 8 AND OFF)

Used to select the group to be transferred. Transfer of more than one group at a time to the auxiliary supply is not permitted. Seq or and aux must also be selected for group 1 thru 7, only aux is required for selection of group 8. Control of groups 5 thru 7 using their regulating (normal) supplies is possible by rotating group select SW to desired position with seq or and group selected. Position of group select switch during manual operation in the sequence mode may be anywhere except at 8. Group select is also used for selection of desired group during relative PI alignment.

27. MANUAL COMMAND SW (INSERT-WITHDRAW)

Provides an in or out command signal to the programmer motors with manual control mode selected. May be used to control group 8 only with control in the automatic mode.

28. SINGLE SELECT SWITCH (POSITIONS 1 thru 12, OFF, AND ALL)

Used to select individual CRDs within a group for transfer, single select is also used to select individual CRD for relative PI adjustment.

29. SPEED SELECTOR (JOG-RUN)

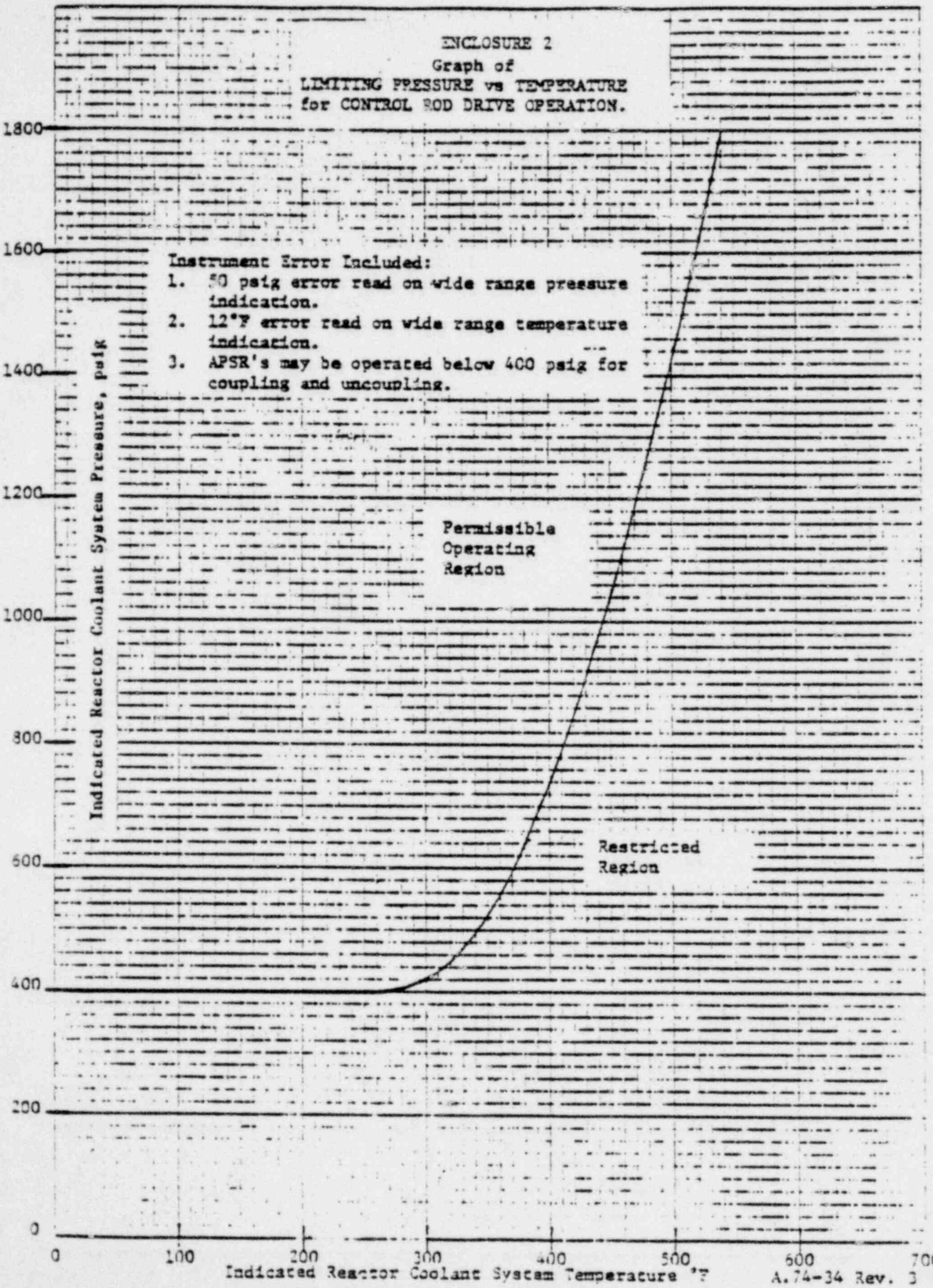
Used to control the rotational speed of the programmer motors, which in turn controls the linear motion of the control rod. Jog and run correspond to 3.0 or 30 inches per minute respectively. If speed select SW is inadvertently left in jog with the automatic control mode selected, the control rods will still move at run speed.

30. XENON CONTROL

Not used.

31. XENON OVERLAP FAULT (LAMP OUT)

Not used.


ENCLOSURE 2
Graph of
LIMITING PRESSURE vs TEMPERATURE
for CONTROL ROD DRIVE OPERATION.
Instrument Error Included:
1. 50 psig error read on wide range pressure indication.
2. 12°F error read on wide range temperature indication.
3. APSR's may be operated below 400 psig for coupling and uncoupling.
Permissible Operating Region
Restricted Region
Indicated Reactor Coolant System Pressure, psig
0
200
400
600
800
1000
1200
1400
1600
1800
0
100
200
300
400
500
600
700
Indicated Reactor Coolant System Temperature °F

A.74-34 Rev. 3

OFR UNIT #1

MAIN TURBINE SYSTEM

OP 1106 01

1. PURPOSE

1.1 To provide procedures to accomplish the following operations:

OPERATION	SECTION
Main turbine startup pre-roll procedure	4
Main turbine operator auto startup	5
Normal operation	6
Main turbine shutdown	7
Loss of electrical tie to system	8

2. REFERENCES

2.1 OP 1102 03, Plant startup.

2.2 Westinghouse Instruction Manuals for steam turbine, EHC control system and generator.

2.3 OP 1102 04, Power Operations.

3. LIMITATIONS AND PRECAUTIONS

3.1 Turbine generator unit must be on turning gear at least one (1) hour prior to rolling with steam.

3.2 No attempt should be made to roll the turbine with steam unless the rotor is being turned by turning gear.

3.3 The Gland Steam System should not be put in service unless the turbine is on turning gear.

3.4 The Electronic Governor must be energized at least one (1) hour prior to rolling the turbine with steam.

3.5 Minimum steam inlet pressure to the turbine is 300 psig.

3.6 Steam removal from the steam generator for increasing the turbine speed about 600 RPM should not be initiated until the reactor is critical at 10% power to prevent cooldown of primary plant.

3.7 The L.P. exhaust hood temperature must not exceed 250°F.

3.8 The L.P. turbine steam inlet temperature should be limited to 400°F when unit load is less than 10%.

3.9 Bearing oil discharge temperatures should not exceed 180°F. Minimum discharge temperature for turning gear operation is 70°F.

3.10 Journal and thrust bearing metal temperature should not exceed 225°F.

3.11 Do not rotate turbine generator if the seal oil system is not in service.

3.12 Maximum permissible back pressure for on-line operation is 5.5 inches of mercury absolute.

4. MAIN TURBINE STARTUP PRE-ROLL PROCEDURE

4.1 Initial Conditions:

_______ (1) Component cooling water in service.

_______ (2) Plant cooling water in service.

_______ (3) The main turbine electro hydraulic oil system in service.

_______ (4) The circulating cooling water system in service.

_______ (5) Auxiliary steam system in service with pressure at ~200 psig.

_______ (6) The plant air system in service.

_______ (7) The main generator seal oil system in service.

_______ (8) Condensate and feedwater system in service.

_______ (9) Air ejector and gland steam system in service.

_______ (10) Extraction steam, reheater and feedwater drain systems in service.

_______ (11) Turbine lube oil system in service with AC turning gear pump, H.P. seal oil backup pump and bearing lift oil pump running.

_______ (12) Turbine is on turning gear.

_______ (13) Generator, excitation and regulator circuits available for operation.

4.2 Procedure:

_______ (1) Verify all supervisory instruments and electronic governor controls have been energized. (Electronic Governor must be energized at least one hour prior to admitting steam to the unit.)

_______ (2) Verify the H_2 system is in operation with pressure greater than 15 psi.

MAIN TURBINE OPERATOR AUTO STARTUP

5.1 Initial Conditions:

______ (1) Section 4., Turbine Pre-Roll Procedure, completed.

______ (2) Plant is at hot shutdown and ready for power operation.

5.2 Procedure:

(1) Check that the following status information is displayed on the turbine panel:

______ (a) "TURBINE MANUAL" mode selector and control stations will be illuminated.

______ (b) "UNIT TRIP" monitor lamp is on.

______ (c) "THROTTLE VALVE CONTROL" status lamp is illuminated.

______ (d) "THROTTLE & GOVERNOR VALVE POSITION" indicators register 0% valve position.

______ (e) "THROTTLE, GOVERNOR, REHEAT STOP, and INTERCEPTOR VALVE" lights show the closed position.

______ (2) Depress the "VALVE POSITION LIMITER" (lower) button until the "VALVE POSITION LIMIT" indicator registers 0% valve limit position.

______ (3) Select "OPER AUTO". Depress the turbine "LATCH" button.

NOTE:

This button must be held for approximately two(2) seconds. When the auto stop and vacuum trip are latched, the "LATCH" lamp will light.

______ (4) Check indicating lamps to see that all interceptor and reheat stop valves are open, and the throttle stop and governor valves are closed.

______ (5) Depress the "VALVE POSITION LIMITER" (raise) button until the valve position limit indicator registers 100% valve limit position. Check governor valves full open.

______ (6) With the governor valves in the open position (indicating lamps OFF), trip the governor, interceptor and reheat stop valves by operating the manual trip button at the Control Room Console. Check the governor valves, interceptor and reheat stop valves shut (indicating lamps ON).

______ (7) Push the "Oper Auto" button. The "Oper Auto" pushbutton will be illuminated, the "Turbine Manual" mode selector and manual control station lamps will go out.

______ (8) Run the Valve Position Limiter to 0% valve limit position. Relatch the unit by depressing the turbine "LATCH" pushbutton, hold for about two seconds, the "UNIT TRIP" lamp will go out and the "LATCH" button will stay illuminated.

______ (9) Run the Valve Position Limiter to 100% valve limit position, check the governor, interceptor, and reheat stop valves re-open fully as indicated by the position lamps OFF and indicators. The throttle valves should be in the closed position.

______ (10) Turn the "Overspeed Protection Controller" switch to test and observe that the governor and interceptor valves close rapidly. Turn key to neutral, remove key and observe that the valves re-open.

______ (11) Using Enclosure 1, determine the acceleration rate to be used for existing H.P. Turbine first stage metal temperature.

______ (12) Set the speed/load reference setter at a terminal speed of 600 RPM. Set the "Acceleration Rate" at the rate determined in Step 5.2(11).

______ (13) With minimum steam inlet pressure of 800 psi, the low vacuum alarm cleared, and reactor critical at least 10% power, depress the "go" pushbutton.

NOTE

If a hold is required during the acceleration of the turbine due to a contingent condition, depress the "Hold" pushbutton. The "Hold" lamp will light and the "go" lamp will go off. The acceleration of the turbine will be stopped.

NOTE

If the turbine fails to respond to the automatic signal and manual operation is desired, depress the "Manual" pushbutton. The "Manual" pushbutton will illuminate, the "Oper Auto" pushbutton will go out. Throttle valve operation will then be actuated by depressing the Throttle Δ/Throttle ∇ pushbuttons to raise and lower the Throttle Stop valves respectively, manually controlling speed

____ (14) Following the "Oper Auto" mode of operation, speed/load "Reference" display will count out and display the rotor speed. After a time interval of several seconds, the rotor speed will increase and the turning gear will automatically disengage. As the turbine rolls off turning gear, listen for rubs and other unusual noises. When the "Reference" display has reached the desired value, the "Go" lamp will go off and the rotor speed will approximately equal the displayed terminal speed.

(15) Keep the turbine rolling at 600 RPM for a sufficient length of time to permit a check of all turbine supervisory instruments to insure that there are no unsatisfactory conditions, per Enclosure 3, and listen for rubs and unusual noises.

____ (16) When turbine speed reaches 900 RPM, set the acceleration rate to 300 RPM.

CAUTION

During the next step, the turbine will pass through two critical speeds as it accelerates. During those periods, increase the acceleration rate to 200 RPM per minute.

____ (17) When the turbine speed reaches 1100 RPM, return the acceleration rate to the setting determined in Step 5.2(11).

____ (18) At 1700 RPM transfer control from the throttle to the governor valves by depressing the throttle to governor transfer button. The "Throttle Valve Control" monitor lamp will go off and the transfer lamp will light. When the transfer is completed, the transfer lamp will go off and the "Governor Control" monitor lamp will light.

____ (19) Set the speed load setter to 1800 RPM and the acceleration rate at 200 RPM and push the "Go" button.

____ (20) At 1800 RPM trip the turbine by depressing the manual trip pushbutton, in the Control Room. Check that the throttle, governor, reheat stop and intercepter valves are fully closed.

Relatch the turbine if test is satisfactory. This test may be made whenever convenient prior to synchronizing. Select "Oper Auto", set the speed load setter to 1800 RPM and push the "Go" button. When the turbine is back to 1800 RPM, transfer control from the throttle to the governor valves by depressing the throttle to governor transfer button.

______ (21) At 1800 RPM check that metal temperature indicators and the turbine supervisory instruments all read within their respective limits as per Enclosure 3, and LIMITS & PRECAUTIONS.

______ (22) Shut off AC turning gear pump, seal oil backup pump, and bearing lift oil pump when turbine speed is 1800 RPM.

______ (23) Close the exciter breaker and raise field current until generator voltage is 22 kv.

______ (24) Put the voltage regulator in auto. Observe voltage holds at 22 kv.

______ (25) Energize synchroscope by turning main generator key switch to operate.

______ (26) Regulate turbine speed to rotate the synchroscope slow in the fast direction. When speed is stable, depress "Operator Auto Sync". Observe main generator breaker closes and generator is loaded to approximately 45 mw and the EHC transfers from speed to load control. If H.P. first stage metal temperature is >300°F, load the turbine to ~100 MW.

______ (27) Turn breaker key switch off and remove key.

______ (28) Close other main generator breaker.

______ (29) Put "Throttle Pressure Limit" in service.

______ (30) Increase the load using the speed/load reference setter in accordance with Enclosure 1. As load is increased, observe turbine bypass valves closing to maintain steam header pressure. When bypass valves indicate closed and load is approx. 20%, push the "Operator ICS" pushbutton to transfer turbine load control to ICS.

6. MAIN TURBINE NORMAL OPERATION

6.1 Initial Conditions:

(1) Main unit on the line and loaded.

6.2 Procedure:

(1) Load should be changed as per Enclosure 2 to reduce thermal stress on the rotor. These curves are to be used for increasing and decreasing loads.

(2) The turbine-generator unit should not be motored for extended periods.

(3) Turbine drains should be opened at <20% of full load.

(4) All supervisory instrument readings are within allowable limits per Enclosure 3.

7. MAIN TURBINE SHUTDOWN

7.1 Initial Conditions:

(1) Main unit on the line and loaded.

7.2 Procedure:

(1) Reduce load gradually as per Enclosure 2 by ICS control or by transfer of turbine control to "Operator Auto" and setting the desired terminal load in the "Setter". Select the desired "Load Rate" and push the "Go" button.

NOTE:

The unloading of the unit can be stopped at anytime by pushing the "Hold" button. Maintain generator hydrogen pressure.

(2) Before load has decreased to 20%, transfer the unit auxiliary power to the startup banks.

(3) When load has decreased to about 20%, open turbine drains.

(4) When generator output is reduced to approx. 50 MWe, separate from the system and open the generator output breakers.

(5) Start the turning gear, the HP seal oil backup pump, and the bearing lift oil pump.

(6) Shut the unit down by pushing the "Turbine Trip" pushbutton.

Check all throttle stop and governor valves, reheat stops and interceptor valves are closed.

NOTE

Maintain condenser vacuum if steam dump is to be used for plant cooldown or if the plant is to be held in a hot shutdown condition.

(7) When vacuum is to be broken, shutdown the air ejectors. Break vacuum with both vacuum breaker valves.

(8) Secure sealing steam system when vacuum decays to zero.

(9) Place the turbine on the turning gear automatically when the turbine speed decreases to zero.

(10) Both the turning gear and the bearing oil system are to be kept in operation for a minimum of 24 hours. Oil temperature from the cooler should be maintained between 95 and 100^{o}F. The unit may be left on turning gear at the end of 24 hours.

8. LOSS OF ELECTRICAL TIE TO SYSTEM

8.1 Initial Conditions:

(1) Unit tied to system.

(2) Unit separates from the system, carrying house loads.

8.2 Procedure:

(1) In the event of a complete or partial loss of electrical load, energy in the entrapped steam will cause the rotor to accelerate. The amount of acceleration is a function of the load level at the time of the load separation.

(2) An Overspeed Protection Controller (O.P.C.) is incorporated in the E.H. Control System. This device performs the following functions:

(a) The load drop anticipator functions:
Senses complete loss of load and rapidly closes all governor valves and interceptor valves to limit the overspeed of the turbine. Senses partial loss of load by comparing turbine input with generator output and

rapidly closes the interceptor valves, to enable the system to stabilize.

(b) The auxiliary governor function of the OPC: Senses excess turbine speed and closes all governor and interceptor valves when the speed is greater than 103% (1854 RPM).

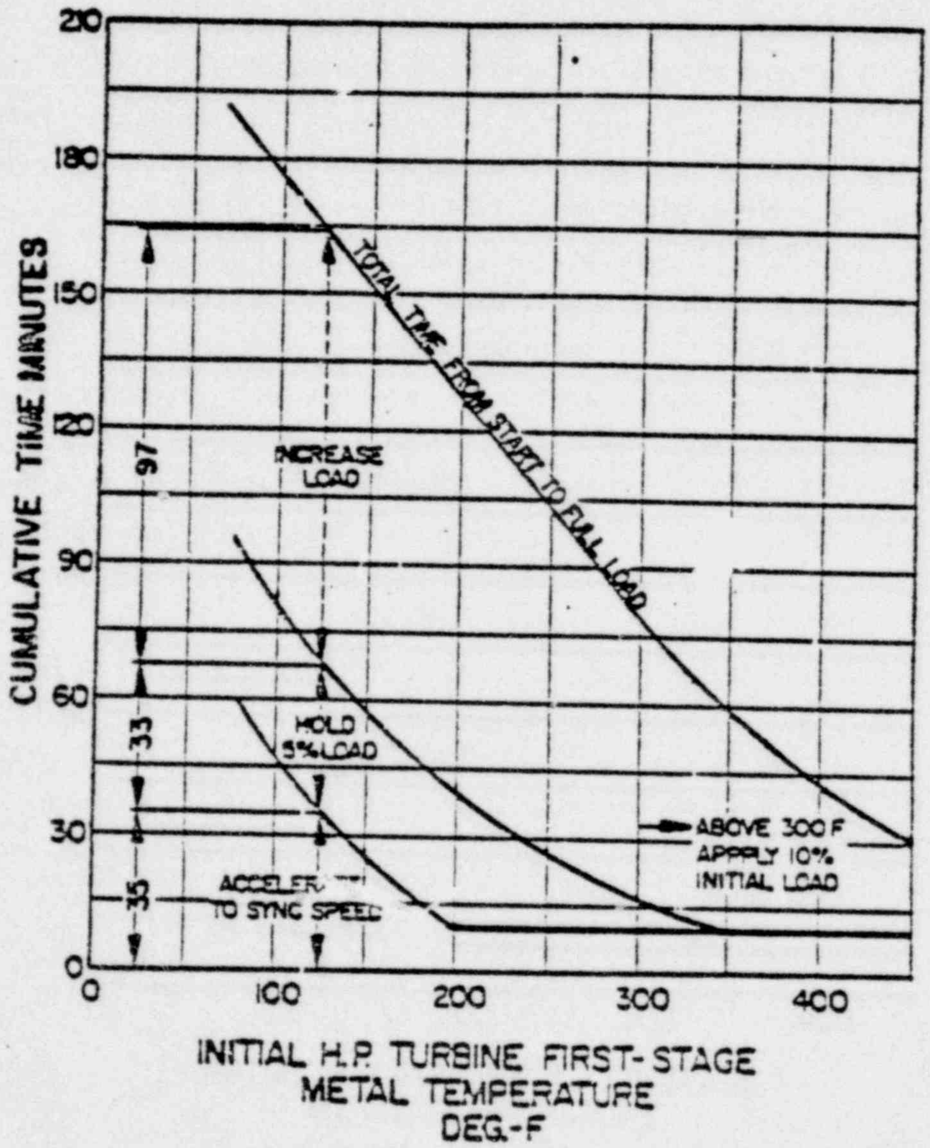
(c) With the opening of the main breaker, the "Setter" and "Reference" displays are automatically reset to 1800 RPM and the "Oper Auto" mode lamp will be on.

(d) After a time interval, the speed of the unit will decrease below the setting of the auxiliary governor which in turn is de-energized permitting the interceptor valves to open slowly. The entrapped steam in the reheat system will cause the auxiliary governor to cut in, again closing the interceptor valves. This mode of control is followed until all of the entrapped steam is dissipated through the interceptor valves.

(e) After the entrapped steam in the reheat system is dissipated, the interceptor valves will start opening and the speed of the unit will decrease towards rated speed. At the rated speed the governor valves will take over the control of the turbine and keep the unit at rated speed.

(f) Check all supervisory instrument readings are within allowable limits. Particular attention should be given to differential expansion readings.

ENCLOSURE 1. Recommended Startup and Loading Times for Nuclear Steam System Units (Westinghouse Drawing CT-22600)



OFR #1

OP 1102 03

ENCLOSURE 2.

LOAD-CHANGING RECOMMENDATIONS NUCLEAR STEAM
SYSTEM UNITS

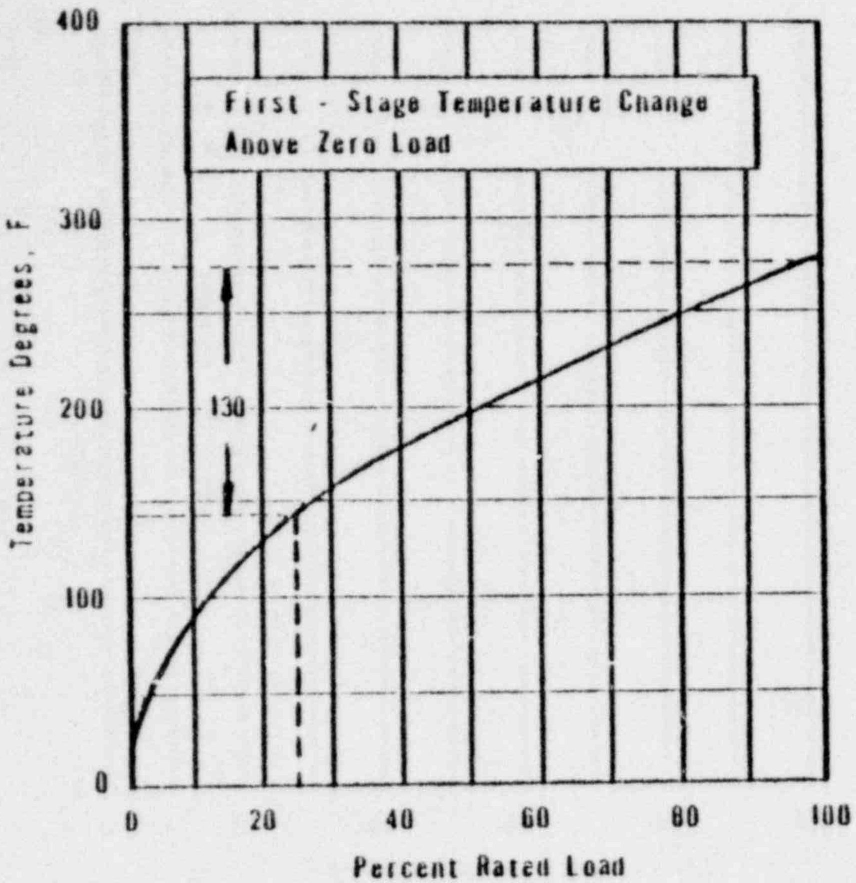


FIGURE 9

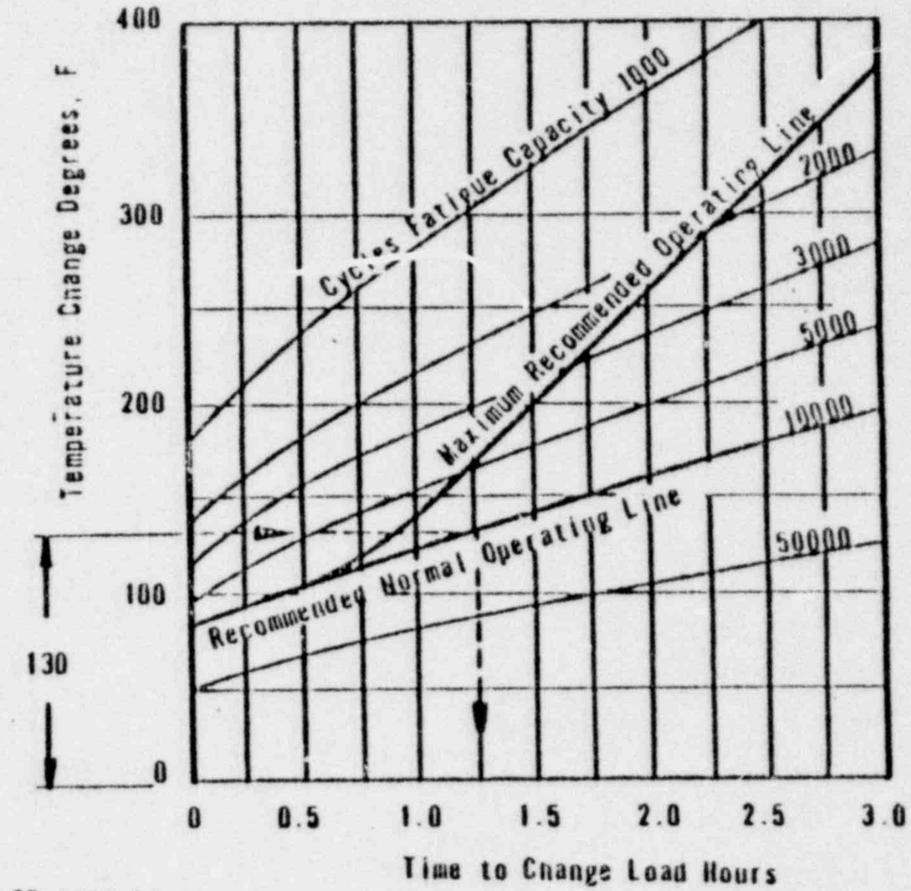


FIGURE 10

OP 1102.04

OFR UNIT #1

ENCLOSURE 3. Turbine Supervisor Instrument Settings

Instrument	Normal	Alarm	Trip
* Eccentricity, mils DA	3	3	--
Differential expansion (gen. end)			
Differential expansion (gov. end)			
Rotor position			
Vibration, mils	3 - 5	7	14
Speed and valve position	--	--	--

OFR #1

OP 1102.03

1106 02

OFR UNIT #1

FEED AND CONDENSATE SYSTEMS

OP 1106 02

1. PURPOSE

1.1 To provide procedures for the following operations:

Operation	Section
Feed and Condensate System Cleanup	4
Feeding OTSG's With Condensate Pumps	5
Feedpump Startup	6
Operation While Increasing Load	7
Operation While Decreasing Load	8

2. DESCRIPTION

The feed and condensate system consists of essentially two parallel cross connected trains of low pressure (condensate) feed discharging two parallel cross connected trains of high pressure feed water to a common header from which the feed control valves again split the flow to the two OTSG's.

In the unlikely event of a loss of both of these feed trains there are two redundant emergency feedwater pumps and associated valving which will supply the OTSG's with adequate makeup water to remove decay heat. The emergency feed water system operation is covered the loss of OTSG feed procedure OP 1202.

3. LIMITS AND PRECAUTIONS

1. Prior to adding feedwater to the OTSG's it must meet chemistry specs. (B&W Water Chemistry Manual.)
2. Avoid at all times passing steam through the feed pump turbine with the rotor at rest.

3. With the governor valve closed the turbine drains should be open.
4. Avoid air being drawn through the glands with the turbine at rest.
5. Do not run feedwater pumps or condensate pumps against a shut off head.
6. Insure at all times the suction pressure for the feedwater pumps is >50 psig.
7. Insure during operation of feed or condensate pumps that the minimum flow recirc valve SV15 is open.

4. FEED AND CONDENSATE SYSTEM CLEANUP

4.1 Initial Conditions

1. Auxiliary boiler is on line.
2. Power available to all pumps and motor operated valves.
3. Condenser hotwell level above low level alarm.

4.2 Procedure

1. Complete valve lineup checklist for feed and condensate system cleanup through LP & HP heater train A.
2. Vent condensate pumps and start. Verify minimum of 1.3×10^6 lbs/hr flow.
3. Cut in full polishing flow.
4. Cut in steam to LP heaters and raise FW temp to between 175°F and 200°F.
5. When iron concentration is within specifications per B&W Water Chemistry Manual, lineup and roll the "A" BFP using aux. steam until iron concentration is again within specs.
6. Repeat steps 1 through 5 for the "B" heater train and the "B" BFP.

*7. Add chemicals as required to meet specifications of the B&W Water Chemistry Manual.

5. FEEDING OTSG(s) WITH CONDENSATE PUMPS

5.1 Initial Conditions

1. Feedwater meets chemistry specs (B&W Water Chemistry Manual).
2. OTSG pressure is <650 psig.
3. Feedwater pumps are shutdown.

5.2 Procedure

1. Complete valve lineup checklist feeding OTSG's with condensate pumps.
2. Start if not already operating at least one condensate pump.
3. Verify feedwater value $\Delta P \geq 30$ psi.
4. Open startup feedwater control valve(s) and feed OTSG(s) as required.

6. FEED PUMP STARTUP

6.1 Initial Conditions

1. Feed system meets chemistry specs (B&W Water Chemistry Manual).
2. OTSG Pressure $\geq$650 psig.
3. Feedwat-r $\geq$ 185 F.

6.2 Procedure

1. Complete applicable portions of valve lineup checklists normal operations.
2. Insure condensate pumps supplying $\geq$50 psig suction pressure.
3. Verify FPT governor control in manual.
4. Reset FPT trip.
5. Using raise pushbutton "Δ" increase FPT speed to 600 RPM verify "turning gear disengage" and stop turning gear motor.
6. Trip the FPT.
7. Verify FPT tripped.
8. Reset trip and raise FPT speed to minimum speed stop $\simeq$3150 RPM.
9. Close FPT drains.
10. Verify ICS feedpump demand station in hand and demand is zero.
11. Transfer FPT speed control to ICS.

7. OPERATION WHILE INCREASING LOAD

7.1 Initial Conditions

1. The plant is at hot shutdown.
2. One BFP is supplying FW on demand to both OTSGs.

7.2 Procedure

1. Complete or verify feed and condensate system lined up per valve line up checklists normal operation.
2. Place all FW block valves in auto.
3. During power escalation when feed water valve ΔP is ~35 psi place operating FWP demand station in auto.
4. At 350 MW perform the following:
   a. Start second condensate pump if not already operating.
   b. Start both heater drain pumps.
5. At 425 MW start up second FW pump as follows:
   a. Verify FPT governor control in manual.
   b. Reset FPT trip.
   c. Using raise pushbutton "Δ" increase FPT speed to 600 RPM. Verify turning gear disengaged and stop turning gear motor.
   d. Trip FPT.
   e. Verify FPT tripped.
   f. Reset FPT trip.
   g. Raise FPT speed to minimum speed stop ~3150 RPM.

   NOTE:

   As FWT comes up in speed the discharge pressure will increase. When the discharge pressure is equal to the other FWP's discharge pressure a surge may be noticed. Increase speed until system stabilizes.

   h. Transfer governor control to ICS.
   i. Raise FWP speed slowly, noting as speed increases the FW valve ΔP increases and the other FWP slows down, match the speed of A and B FW pumps and place in auto.

7. OPERATION WHILE INCREASING LOAD CONTINUED

7.2 Procedure Continued:

6. When unit load is at 650 MW start third condensate pump.

NOTE:

If third condensate pump is inoperable load may be increased but the condensate polishers should be bypassed. The probable result of a loss of heater drain pumps or a condensate pump will be a loss of both FW pumps.

8. OPERATION WHILE DECREASING LOAD

8.1 Initial Conditions

1. The plant is greater than 650 MW.
2. Both FW pumps are operating.

8.2 Procedure

1. At 650 MWe decreasing shut down one condensate pump.
2. At 425 MW perform the following:
   a. Place ICS hand auto station for FW pump to be shutdown in hand.
   b. Slowly decrease speed demand to zero while observing feed water valve ΔP.
   c. Place local control to manual.
   d. Using the lower "7" pushbutton decrease FPT speed until there is a significant difference in feed pump discharge pressures.
   e. Trip the feed pump turbine.
   f. Open the turbine drains.
   g. When speed is zero start turning gear motor and engage clutch.
3. When load is 350 MW shutdown one condensate pump, if two are operating, and both heater drain pumps.

CFR UNIT #1

VALVE LINEUP CHECKLIST

System Condensate System (CS) Operation Feed and Condensate System Cleanup

Valve #	Name	Location	Open	Closed	Initial
CS-S36	Polisher Bypass	TGC		X	
CS-SV6	LP Heater Bypass	Manual		X	
CS-SV7	LP Heater #1 Inlet	Manual	X		
CS-SV8	LP Heater #2 Inlet	Manual	X		
CS-SV5	Recirc LP Heater to Main Conden	Manual		X	
CS-S37	Vacuum Breaker	TGC		X	
CS	MFWP A Drain	TGC	X		
CS	MFWP B Drain	TGC	X		
CS	Main Turbine Drain	TGC	X		

CFR UNIT #1

VALVE LINEUP CHECKLIST

System Feedwater (FW) Operation Feed and Condensate System Cleanup

Valve #	Name	Location	Open	Closed	Initial
FW-16A	Emergency FWP #1 Dischr	TGC		X	
FW-16B	Emergency FWP #2 Dischr	TGC		X	
FW-SV9	Main FWP Bypass	Manual	X		
FW-SV10	Main FWP B Disch	Manual		X	
FW-SV11	Main FWP A Disch	Manual		X	
FW-SV13	HP Heater #1 Inlet	Manual	X		
FW-SV14	HP Heater #2 Inlet	Manual	X		
FW-SV37	Main Feed Block Valve A	RUC		X	
FW-SV38	Main Feed Block Valve B	RUC		X	
FW-S21	Main Feed Valve A	RUC		X	
FW-S23	Main Feed Valve C	RUC		X	
FW-SV19	Startup Block Valve A	RUC		X	
FW-SV18	Startup Block Valve B	RUC		X	
FW-S20	Startup Feed Valve A	RUC		X	
FW-S22	Startup Feed Valve B	RUC		X	
FW-SV157	Emergency Block Valve A	RUC		X	
FW-SV156	Emergency Block Valve B	RUC		X	

CFR UNIT #1

VALVE LINEUP CHECKLIST

System Condensate System (CS) **Operation**

Valve #	Name	Location	Open	Closed	Initial
CS-S36	Polisher Bypass	TGC	X		
CS-SV6	LP Heater Bypass	Manual		X	
CS-SV7	LP Heater #1 Inlet	Manual	X		
CS-SV8	LP Heater #2 Inlet	Manual	X		
CS-SV5	Recirc LP Heater to Main Conden	Manual		X	
CS-S37	Vacuum Breaker	TGC		X	
CS	MFWP A Drain	TGC	X		
CS	MFWP B Drain	TGC	X		
CS	Main Turbine Drain	TGC	X		

CFR UNIT #1

VALVE LINEUP CHECKLIST

System Feedwater (FW) **Operation** Feeding OTSG(S) with Condensate Pumps

Valve #	Name	Location	Open	Closed	Initial
FW-16A	Emergency FWP #1 Dischg	TGC		X	
FW-16B	Emergency FWP #2 Dischg	TGC		X	
FW-S79	Main FWP Bypass	Manual	X		
FW-SV10	Main FWP B Disch	Manual		X	
FW-SV11	Main FWP A Discg	Manual		X	
FW-SV13	HP Heater #1 Inlet	Manual		X	
FW-SV14	HP Heater #2 Inlet	Manual		X	
FW-SV37	Main Feed Block Valve A	RUC		X	
FW-SV38	Main Feed Block Valve B	RUC		X	
FW-S21	Main Feed Valve A	RUC		X	
FW-S23	Main Feed Valve C	RUC		X	
FW-SV19	Startup Block Valve A	RUC		X	
FW-SV18	Startup Block Valve B	RUC		X	
FW-S20	Startup Feed Valve A	RUC		X	
FW-S22	Startup Feed Valve B	RUC	X		
FW-SV157	Emergency Block Valve A	RUC	X		
FW-SV156	Emergency Block Valve B	RUC	X		

CFR UNIT #1

VALVE LINEUP CHECKLIST

System Feedwater (FW) Operation Normal Operation

Valve #	Name	Location	Open	Closed	Initial
FW-16A	Emergency FWP #1 Dischg	TGC		X	
FW-16B	Emergency FWP #2 Dischg	TGC		X	
FW-SV9	Main FWP Bypass	Manual		X	
FW-SV10	Main FWP B Disch	Manual	X		
FW-SV11	Main FWP A Discg	Manual	X		
FW-SV13	HP Heater #1 Inlet	Manual	X		
FW-SV14	HP Heater #2 Inlet	Manual	X		
FW-SV37	Main Feed Block Valve A	RUC		X	
FW-SV38	Main Feed Block Valve B	RUC		X	
FW-S21	Main Feed Valve A	RUC		X	
FW-S23	Main Feed Valve C	RUC		X	
FW-SV19	Startup Block Valve A	RUC		X	
FW-SV18	Startup Block Valve B	RUC		X	
FW-S20	Startup Feed Valve A	RUC	Auto		
FW-S22	Startup Feed Valve B	RUC	Auto		
FW-SV157	Emergency Block Valve A	RUC	X		
FW-SV156	Emergency Block Valve B	RUC	X		
FW-SV12				X	

GFR UNIT #1

VALVE LINEUP CHECKLIST

System Condensate System (CS) Operation Normal Operation

Valve #	Name	Location	Open	Closed	Initial
CS-S36	Polisher Bypass	TGC	As Required		
CS-SV6	LP Heater Bypass	Manual		X	
CS-SV7	LP Heater #1 Inlet	Manual	X		
CS-SV8	LP Heater #2 Inlet	Manual	X		
CS-SV5	Recirc LP Heater to Main Conden	Manual		X	
CS-S37	Vacuum Breaker	TGC		X	
CS	MFWP A Drain	TGC		X	
CS	MFWP B Drain	TGC	X		
CS	Main Turbine Drain	TGC	X		

INDEX TO ABNORMAL AND EMERGENCY PROCEDURES

(Short Course)

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1202 01

OFR UNIT 1

LOAD REJECTION

OP 1202 1

1. PURPOSE:

This procedure describes the necessary actions for operation of the plant during a load rejection if 220kv bus blackout.

2. DESCRIPTION:

2.1 Load rejection describes the condition in which the turbine generator is separated from the 220kv grid.
Load rejection may be caused by a failure in either the generator main output transformer, the 220kv bus or the transmission lines from the 220kv bus.

2.2 As a result of a load rejection, the ICS commences to "track" which results in a 20%/minute runback to 15% reactor power.

2.3 The results of a blackout would be the same as 2.1 and 2.2

3. LIMITS AND PRECAUTIONS:

1. Do not resort to manual control of any parameter except to prevent exceeding allowable limits. When controlling a parameter manually such as reactor coolant pressure, care must be taken not to over control, i.e. only sufficient action should be taken to prevent exceeding limits.
2. Do not allow pressurizer level to exceed 330". If this limit is exceeded, the reactor must be tripped.
3. Operation at less than 5% rated load should be avoided. However, when necessary, Auxiliary Load may be carried indefinitely on the main generator following a load rejection provided:
   a) LP turbine exhaust hood temperature is below 250F.
   b) All supervisory instrument readings are within allowable (alarm) limits.
      NOTE: Particular attention should be given to differential expansion readings. Rapid or continued changes in readings may require

timely action to avoid exceeding allowable limits.

c) If "A or B" cannot be maintained the turbine will be either shutdown or load brought above 10% rated load.

4. PROCEDURE

4.1 Symptoms:

1) Megawatt output to 220kv bus (grid) zero.
2) Reactor power runback.
3) Surge in turbine generator ouput frequency.
4) Rapid increase in pressurizer level and reactor coolant pressure.
5) Black out would be indicated by 1 through 4 and loss of power to the startup transformers. Followed by auto start deisel generators starting and supplying the ES busses.

4.2 Immediate Action:

The integrated control system will ordinarily take the necessary immediate action. This action, however, must be monitored by the operator.

1) Check to ascertain that a reactor runback is in progress. If the runback is not in progress, immediately put the reactor control rods in manual and commence driving rods into reduced reactor power 15%.
2) Verify that the main steam bypass valves open. If these valves are opened manually, they must be either throttled or put in automatic when main steam pressure drops to 885 psig.
3) Manually trip the turbine generator output breaker to the 220kv bus.
4) Monitor the following parameters:

    Pressurizer level
    Reactor coolant pressure
    Reactor coolant temperature
    OTSG level
    Main steam pressure
    Main condenser vacuum
    Reactor power

5. Manually trip the reactor if any reactor trip parameter exceeds its setpoint. The reactor protection system will not ordinarily require actuation either automatically or manually following a load rejection.

4.3 Followup Action:

1) Manually control letdown flow if necessary to prevent the pressurizer from filling on the resultant insurge. Manually trip the reactor if pressurizer level reaches 330".
2) Verify turbine speed being controlled at 1800 RPM if not, take turbine control in operator auto and adjust speed to 1800 RPM.
3) Manually control feed flow if necessary to prevent OTSG level exceeding maximum of operating range.
4) Check to insure all relief valves are seated properly. If a relief valve cannot be reseated, it may require isolation or gagging. Refer to the Tech Spec and Bases for permissible operation with inoperative or leaking relief valves.
5) Investiagate to determine the exact cause of the load rejection; correct if possible and put the turbine generator back on the grid.
6) During a 220kv blackout, perform the above and the following:
   a) Verify the deisel generators are on line supplying the ES busses "1C and 1B."
   b) Open the reservoir Isolation PC-SV1, Raw water to Reservoir Pc-SV2, PCW-PRW supply header X-connect PC-SV4, and place the standby plant CW Raw Water pump in service on the plant cooling water panel.

1202 02

OFR UNIT #1

TURBINE TRIP

OP 1202 02

1.0 SYMPTOMS:

1. Turbine trip alarm

2. Turbine throttle, governor, IV's and RSV's valves shut and generator and exciter breaker are open.

3. Integrated control system in track.

4. Megawatt output zero.

5. Rapid rise in RC pressure and temperature.

6. Safety valves lifting.

2.0 IMMEDIATE ACTION:

1. Manually trip the turbine.

2. Verify the following:

   a. Throttle valves, governor valves, intercept valves and reheat stop valves are closed.

   b. Main generator breakers and exciter breaker are open.

   c. ICS in track and running the plant back to 15% power. If any station is in hand, take appropriate action.

   d. A.C. oil pump and seal oil back up pumps start.

   e. Turbine bypass valves are controlling header pressure at 900#.

   f. Electrical loads have shifted to startup transformers.

   g. RCS pressure increase terminated by spray valve. If not, manually open pressurizer spray valve.

3.0 FOLLOW UP ACTION:

1. Maintain pressurizer level less than 315 inches. If pressurizer level exceeds 315 inches, manually trip the reactor.

2. Ensure main steam safety valves have reseated.

3. Open the main turbine drain valves.

4. With reactor plant stabilized at approximately 15% power, perform the following:

   a. Secure all but one condensate and one main feed pump.

   b. Secure both heater drain pumps.

5. Start the lift pumps.

6. Verify zero speed alarm and turning gear starts and engages.

7. Determine and correct cause of turbine trip.

8. Proceed with either OP 1102-03 (Plant Startup) or OP 1102-10 (Plant Shutdown) as appropriate.

1202 04

OFR UNIT #1

REACTOR TRIP

EP 1202-04

1.0 SYMPTOMS:

1. Reactor trip alarm.
2. Individual and group in limit lights actuate.
3. Rapid decrease in neutron level indicated by nuclear instrumentation.
4. Turbine trip alarm.
5. Rapid decrease in unit load.

2.0 IMMEDIATE ACTION:

1. Manually trip the reactor and verify all rods except Group 8 are on the bottom.
2. Manually trip the turbine and verify the generator and exciter breakers are open.
3. Close the letdown isolation valve.
4. Monitor pressurizer level and maintain greater than 100". Start second makeup pump as required. If pressurizer level drops to less than 80", verify pressurizer heaters are off. Shift suction of A makeup pump to BWST and open HPI valve B to restore pressurizer level to greater than 100".
5. Verify turbine bypass valves are maintaining header pressure at approximately 1010 psig.
6. Verify ICS transfers to track and that feedwater flow control responds properly if in auto. If in hand, take appropriate action.

3.0 FOLLOW UP ACTION:

1. Primary Plant:

   a. Announce reactor trip on page.

   b. Verify RCS pressure returns to normal.

   c. Perform reactivity balance/OP 1103-15 to insure shutdown margin >1% Δk/k. If necessary, borate per OP 1103-04.

d. Lower pressurizer level setpoint to 100" (25%) unless cooldown is anticipated.

e. Monitor makeup tank level. If level drops less than 55", line up transfer pump to appropriate bleed tank and transfer water to makeup tank. If level cannot be returned to normal, shift suction of makeup pump to BWST. Maintain this lineup until makeup tank level recovers.

f. If power is not less than 5% within one (1) minute after trip, begin emergency boration.

NOTE: To emergency borate, lineup transfer pump from BAMT to makeup tank.

Verify source range energizes at approximately $5x10^{-10}$ amps.

g. Take manual control of turbine bypass valves and gradually decrease the header pressure to 885 psig.

h. Investigate and determine cause of trip.

2. Secondary Plant:

a. Reference OP 1202-02, Turbine Trip.

b. Verify OTSG level is maintained at 30" on startup range.

c. Shutdown one main feed pump per OP 1106 02.

d. Shutdown all but one condensate pump.

e. Shutdown heater drain pumps.

f. Verify electrical loads have shifted to startup transformers.

3. In the event of four (4) RCP trip:

a. Verify the following:

(1) Steam driven emergency feedwater pump starts.

(2) Main and startup feedwater block valves close.

(3) Emergency feedwater block valves open and the emergency feedwater valves control at 50% on operating range.

1202 06

OFR UNIT I

LOSS OF REACTOR COOLANT/REACTOR COOLANT SYSTEM PRESSURE

OP 1202 6

1. PURPOSE:

1) To provide emergency procedures to be followed in the event of loss of reactor coolant/reactor coolant system pressure, for leak rates ranging from 30 gpm to and including that associated with a 36" rupture.

2) This procedure is divided into three categories:

SECTION	CATEGORY
4	Leak or rupture within capability of one makeup pump capacity to maintain RC system pressure and pressurizer level.
5	Leak or rupture within capability of high pressure injection system to maintain RC system pressure and pressurizer level.
6	Large rupture (in excess of capability of high pressure injection system).

2. DESCRIPTION:

This procedure describes the action to be taken in the event of a sudden and rapid unexplained decrease in RC system pressure and pressurizer level caused by a leak or rupture in the high pressure envelope of the primary system.

The initial symptoms could be caused by a malfunction of the makeup system or by a steam line rupture, as well as by a loss of coolant from the RC system. The operators should assume the cause of the symptoms described above is a RC leak or rupture, unless another cause can be immediately established.

This procedure applies to the RC system leak rates greater than 30 gpm.

3. LIMITS AND PRECAUTIONS

4. Leak or rupture within capability of one makeup pump capacity to maintain RC System Pressure and pressurizer level.

4.1 Symptoms:

1) Pressurizer level and RC system pressure decrease, without an associated RC average temperature change, initially with one or two makeup pumps running and then hold steady or begin to increase as the pressurizer level control valve opens fully and letdown and seal return flows are isolated (step 4.2.1).

2) Makeup tank level decreases at a rate approximately equal to the system leak rate 1 to 4 inches per minute which is equivalent to a leak rate of 30 - 120 gpm after letdown and seal return flows are isolated (step 4.2.1).

3) High radiation level alarm in reactor building.

4) Possible reactor building sump level high alarm.

4.2 Immediate Action:

1) Close the letdown isolation valve and the RC pump seal return isolation valve.

2) Reduce load as rapidly as possible in preparation for a normal shutdown.

3) Commence normal shutdown and cooldown per Plant Shutdown OP 1102 10.

4) Maintain makeup tank level above the low level alarm point and maintain pressurizer level in the normal operating range.

NOTE: The operator should line up a source of boreated water equal to or greater than RC system, boron concentration.

NOTE: If there is not an adequate supply of borated water available from coolant storage, of if the operator is unable to maintain the makeup tank level above the low level alarm from this source, of if the pressurizer level drops below the low level alarm point, manually initiate high pressure injection then trip the reactor. Carry out the procedure in Section 5 starting at Step 5.2.4.

4.3 Follow Up Action:

1) When the plant has been shutdown the RC system has been cooled down and depressurized, monitor radiation levels to determine when it is safe to send personnel into the Reactor Building to locate and make preparations for repair of the rupture.

5. Leak or rupture within capability of the High Pressure Injection System to maintain RC Pressure and Pressurizer Level.

5.1 Symptoms:

1) Pressurizer level and RC pressure decrease, without an associated RC average temperature change, and continues decreasing as the pressurizer level control valve opens fully and letdown and RC pump seal return flows are isolated (Step 5.2.1).
2) Makeup tank level decreasing rapidly at a rate greater than 3 - 4 inches per minute which is equivalent to a leak rate greater than 90 - 120 gpm. When letdown and RC pump seal flows are isolated (Step 5.2.1).
3) High radiation level alarm in the reactor building.
4) Possible high Reactor Building sump level.

5.2 Immediate Action:

1) Close the letdown isolation valve and the RC pump seal return valve.
2) If the reactor has not already tripped on low pressure, trip the reactor manually.
3) If high pressure injection has not already been actuated automatically, actuate high pressure injection manually.
4) If RC system pressure and pressurizer level continue decreasing after initiation of high pressure injection perform the procedure for large rupture section 6.2.
5) If the RC system and pressurizer level stop decreasing or begin to increase upon initiation of high pressure injection maintain pressurizer level as close as possible to the normal operating range by varying the number of running makeup pumps.

NOTE: Limit HP Injection flow to 500 gpm per flow path to prevent pump runout. (If all three makeup pumps are running the maximum allowable flow through the HP Injection flow path into loop A is 1000 gpm).

5.3 Follow up Action:

1) Reduce the number of running RC pumps to one in each loop.

2) Cooldown and depressurize the RC system, following the normal plant cooldown procedure CP 1102 11, as closely as possible with the exception that Engineered Safeguards Systems, except the core flooding system, must not be bypassed, or isolated. The safeguards systems will thus be available to actuate automatically as RC system pressure decreases or Reactor Building pressure increases. Continue controlling pressurizer level by varying the number of running makeup pumps as necessary.

NOTE: If the borated water storage tank level reaches its lo-lo level alarm point (this would take at least 3 hours) before RC system pressure reaches 200 psig start NSRW Pumps 1A and 1B, open NSCW discharge Pumps A and B valves NS-SV92 and NS-SV93, CW supply and return valves (for DH coolers A and B) NS-SV94 and NS-SV95, NS-SV96 NS-SV97 and start the Nuclear Service Pumps 1A and 1B open RB sump suction valves DH-SV51 and DH-SV60 and DH-SV62, and manually operate LP supply to HP suction A and C valves DH-SV59 and DH-SV61, and start low pressure injection pumps. The low pressure injection pumps will thus take a suction from the reactor building sump and discharge to the suction of the makeup pumps. When Low pressure injection is initiated at 200 psig the low pressure injection valves DH-SV57 and DH-SV58 will open and the low pressure injection pumps will inject directly into the reactor. The makeup pumps must then be stopped and the LP supply to HP suction A and C valves DH-SV60 and DH-SV62 must be closed.

3) As RC system pressure decreases below NPSH required for RC pump operation, stop the RC pumps. (Figure 1 Plant Shutdown Procedure CP 1102 10)

4) Manually initiate LP injection at 200 psig, if automatic LP injection did not occur. Start NSRW pumps 1A and 1B open NSCW pump discharge valves NS-SV92 and NS-SV93, CW supply and return valves for DH coolers A and B. NS-SV94 and NS-SV95, NS-SV96 and NS-97 start NSW pumps 1A and 1B.

5) When the LP pumps are running and the LP injection is functioning properly, stop the makeup pumps.
6) When the borated water storage tank reaches its lo-lo level point, open the RB sump suction valves DH-SV51 and DH-SV52 then shut the borated water storage tank suction valves A and B DH-SV87 and DH-SV88.

6. Large Rupture (in excess of the capability of the high pressure injection system).

1) Symptoms:

1) Rapid and unexplained decrease in RC system pressure and pressurizer level, with associate alarms.
2) High radiation level alarm in the reactor building.
3) Reactor building sump high level alarm.
4) Reactor trip, turbine trip, and safeguards system actuated alarms.
5) Reactor building pressure and temperature increasing.

2) Immediate Action:

1) Verify that reactor trip occurs at or before 1800 psig if not already tripped.
2) Verify that high pressure injection is initiated at 1500 psig (decreasing) RC pressure or 4 psig (increasing) RB pressure as follows: Check on the engineered safeguard panel that all HP injection equipment started (channels 1 and 2 have a solid bar of white lights). If any high pressure injection component fails to assume its engineered safeguards status automatically, the operator should actuate them manually.
3) Verify that injection from core flood tanks occurs at 600 psig decreasing.
4) Verify that LP injection is initiated at 200 psig (decreasing) RB pressure or 4 psig (increasing) RB pressure as follows: Check on the Engineered Safeguards panel that all LP injection equipment started (channels 3 and 4 have a solid bar of white lights). If any LP injection component fails to assume its engineered safeguards status automatically, the operator should actuate them manually.
5) If RB pressure increases to 4 psig, verify that RB isolation is actuated and that emergency RB cooling is initiated as follows: Check on the engineered safeguards panel that all RB isolation valves and RB cooling units started (channels 5 and 6 have a solid bar of white lights) if any RB isolation valves or RB cooling units fail to assume their engineered safeguards status automatically,

the operator should actuate them manually.

3) Follow Up Action:
   1) Stop RC pumps.
   2) Close main and emerg[illegible]y feed water block valves; stop the emergency feedwater [illegible]ps.
   3) Insure that the tur[illegible]ine has tripped; if not trip manually.
   4) After initiation of LP injection, when the LP injection system is functioning properly, stop the makeup pumps.
   5) When the BWST level decreases to the lo-lo level alarm, open DH cooler CW supply and return valves (NS-SV 94, SV95, SV96, and SV97) open the RB sump suction valves DH-SV51 and DH-SV52 then shut the BWST suction valves A, and B DH-SV87 and DH-SV88.

1202 J0

CONTROL ROD DRIVE ABNORMAL OPERATIONS

1. PURPOSE

1.1 To provide the procedures to be taken when a control rod(s) becomes inoperable.

Operation	Section
Asymmetric Rod	4
Stuck Rod	5
Dropped Rod	6
Motor Fault	7
Position Indication Malfunction	8

2. DESCRIPTION

2.1 Definitions:

A CRD is considered inoperable if any of the following conditions exist.

1) A control rod is misaligned with its group average position by more than an indicated nine (9) inches.

2) A control rod cannot be exercised.

3) Neither absolute or relative position indication is operable.

4) Any rod found to be improperly programmed shall be declared inoperable until properly programmed.

5) Control rod trip insertion time does not meet the requirements of technical specifications (reactor control rod system tests).

3. LIMITS AND PRECAUTIONS

3.1 With an asymmetric rod, power shall be limited to 60% of the power allowable for the reactor coolant pump combination.

3.2 With an asymmetric rod in Groups 5, 6, 7, or 8, the other rods in the group shall be trimmed to the same position. Normal operation may then continue provided that the rod that was declared inoperable is maintained within allowable bank average position limits.

However, if a safety rod absolute or relative position indication is inoperable and an energized out-limit light confirms the rod is fully withdrawn, the safety rod shall not be considered asymmetric.

If within one (1) hour of determination of an inoperable rod, it is not determined that a 1% Δk/k hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the hot standby condition until this margin is established. All rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

3.3 Operation with more than one inoperable rod is not permitted.

3.4 If a control rod in the regulating and/or safety rod banks is declared inoperable in the withdrawn position because:

1) It does not meet the trip insertion time as defined in the technical specifications.

2) It cannot be exercised.

3) Neither absolute or relative position indication is operable.

An evaluation shall be initiated immediately to verify the existence of 1% Δk/k hot shutdown margin. Boration may be initiated either to the worth of the inoperable rod or until the regulating and transient rod banks are fully withdrawn, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

If within one (1) hour of determination of an inoperable rod it is not determined that a 1% Δk/k hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the hot standby condition until this margin is established and all rods shall be exercised weekly until the rod problem is solved.

4.0 Asymmetric Rod

4.1 Symptoms:

(1) Asymmetric rod fault lamp (9 inches) illuminated on the rod control panel.

(2) Rod control system alarm.

(3) Meter indication at PI panel and group PI meters.

4.2 Immediate Action:

(1) If in auto control, verify that reactor power is reduced to 60% of the power allowable for the reactor coolant pump combination. If in manual control, take appropriate action.

(2) During the power reduction, evaluate the condition of all control rods to determine proper response of other rods.

(3) Notify shift supervisor.

(4) Commence shutdown margin evaluation.

(5) Locate malfunctioning control rod and confirm its status is as indicated.

4.3 Followup Action:

4.3.1 Asymmetric rod in Groups 5, 6, 7 or 8:

(1) If one rod is asymmetric, transfer the operable rods to the auxiliary supply using OP 1105.09 and move them to the position of the asymmetric rod. Transfer the rods back to the group supply using OP 1105.09. Normal operations may then continue provided that the rod that was declared inoperable is maintained within allowable group average position limits.

(2) If the operable rods are not moved to the position of the asymmetric rod and if within one hour it is determined that the shutdown margin evaluation is ≥ 1%, operation (limited to 60% of the power allowable for the reactor coolant pump combination) may continue, if the shutdown margin is < 1%, the reactor shall be brought to hot standby.

(3) If more than one rod is asymmetric, operation shall not be permitted, continue shutdown in accordance with plant shutdown procedure OP 1102.10.

(4) Determine cause of rod being asymmetric and take corrective action to return the rod to the operable status.

4.3.2 Asymmetric Rod in Groups 1, 2, 3 or 4:

(1) If one rod is asymmetric in the safety groups, operation is limited to 60% of the power allowable for the reactor coolant pump combination.

(2) Determine cause of rod being asymmetric and take corrective action to return the rod to the operable status.

5.0 Stuck Rod

5.1 Symptoms:

(1) A CRD cannot be operated during CRD exercise.

(2) A CRD will not move during normal operation.

5.2 Immediate Action:

(1) Commence shutdown margin evaluation.

(2) Commence a program of exercising all rods.

5.3 Followup Action:

(1) If the 1% Δk/k hot shutdown margin does not exist, boration may be initiated for the worth of the stuck rod or until the regulating and transient rod banks are fully withdrawn, whichever occurs first.

(2) Determine cause of rod being stuck and take corrective action to return the rod to the operable status.

6.0 DROPPED ROD

6.1 Symptoms:

(1) Unexpected in limit indication on the CRD position indication panel.

(2) Unexpected group in limit indication on the rod control panel.

(3) Possible asymmetric rod alarm.

(4) Decreasing reactor power.

(5) Above 60% power ICS runback rod stop alarm and out inhibit light on.

(6) A decrease in rod group position indication.

6.2 Immediate Action:

(1) Verify execution of expected automatic action if reactor power is greater than 60%.

(2) Establish and/or verify less than 60% stable power level.

6.3 Follow-up Action:

(1) Locate malfunctioning CRD.

(2) Correct cause of malfunction.

(3) Select manual rod control.

(4) If the dropped rod is in safety group, transfer all rods in that group to the auxiliary power supply as per OP 1105 09.

(5) Depress inlimit bypass and insert the group until its out limit light goes off.

(6) Transfer the group back to the DC hold supply per OP 1105 09.

(7) Transfer the dropped rod to the auxiliary supply and withdraw it to its outlimit.

(8) If the dropped rod is in the regulating or transient group clear its group outlimit if it is on.

(9) Transfer dropped rod to auxiliary power supply and withdraw it to its group average and transfer it back to its group supply.

7.0 MOTOR FAULT

7.1 Sympton:

(1) Motor fault lamp illuminated on the control panel.

(2) Rod control system alarm.

7.2 Immediate Action:

(1) When the motor fault lamp comes on, CRD control will be switched to manual control at the rod control panel. Using the manual control switch, attempt to insert the malfunctioning group. If control cannot be obtained, manually trip the reactor and follow the reactor trip procedure.

7.3 Follow-up Action:

(1) If reactor is not tripped, rod control is in manual and able to control the malfunctioning group and restore reactor power to its initial level.

(2) Transfer the malfunctioning group to the auxiliary power supply and trouble shoot the malfunctioning motor.

8.0 PI MALFUNCTION

8.1 Symptom:

(1) Both the absolute and relative position indication not operating for the same rod.

8.2 Immediate Action:

(1) Commence shutdown margin evaluation.

(2) Notify shift supervisor.

8.3 Follow-up Action:

(1) Check the absolute individual CRD amplifier for proper operations over its range using an external voltage source and realign amplifier if not in specification tolerance.

(2) Check the position indicator and computer readings while performing Step 1.

(3) Place CRD on auxiliary power supply and run to the lowest possible reference PI light (0, 50 to 75% withdrawn) and check voltage from the voltage divider network in the PI tube to amplifier.

(4) If the voltage measurement Step 3 is out of tolerance, further investigation must be conducted to determine if problem is in the interconnecting wiring or the PI motor tube.

1202 14

OFR UNIT #1

LOSS OF RC FLOW - RC PUMP TRIP

OP 1202 14

1. PURPOSE

This procedure defines the action to be taken upon the indication of a loss of all RC pumps.

Operation	Section
Loss of Flow With The Reactor Critical	4
Loss of Flow During Heatup Or Cooldown With RC Pumps	5

2. DESCRIPTION

2.1 The reactor coolant pump(s) will be tripped by the following signals and/or conditions:

Manual Trip
Loss of Power
Running Undervoltage (time delay)
Instantaneous Overcurrent (short circuit)
Time Delay Overcurrent (overload)
Ground Fault
Phase Balance
Current Differential

*2.2 The "Full Speed" alarm of motor(s) also indicates the no-flow condition and requires an immediate manual trip of the pump.

2.3 Decay heat is removed from the reactor coolant by producing steam in the OTSGs which is bled to:

1. The atmosphere via the main steam safety relief valves.
2. The condenser via the turbine bypass valves.
3. One or more feed pumps or other steam driven auxiliaries with exhaust to the atmosphere or the condensers.

4. The atmosphere via steam dumps.

2.4 Natural circulation will be indicated by a temperature difference between hot and cold legs. For shutdown from full power with a fully irradiated core, the RC outlet temperature will rise to approximately 630F and then begin to drop. The steam generator primary outlet temperature will obtain a constant value of approximately 555F.

3. LIMITS AND PRECAUTIONS

1. During natural circulation of reactor coolant, steam generator level must not be greater than 288 inches on the operating range instrumentation.
2. Do not operate feedwater pumps with steam conditions below the minimum required for feedwater pump trubines.
3. Decay heat removal via atmospheric steam dumps can continue at a constant TAVG of 550F as long as sufficient make-up capability exists to balance the leakage from the seconday plant systems.
4. Do not deborate while on natural circulation.
5. Boration may be done after natural circulation is indicated by temperature difference between hot and cold legs.

4. LOSS OF FLOW WITH THE REACTOR CRITICAL

4.1 Symptoms

1. Reactor Trip
2. RCP's trip annunciated
3. RC Flow decreasing
4. RC Temperature and pressure increasing

4.2 Immediate Action

1. Verify Reactor trip and perform Reactor trip procedure OP 1202 04.
2. Verify the following in the Feedwater System
   a) Main Feedwater block valves shut.
   b) Startup Feedwater block valves shut.
   c) Emergency block valves open.

4.3 Follow Up Action

1. If the plant is >15% power, perform or verify the following:
   a) Electromagnetic Relief Valve reseated.
   b) OTSG levels increasing due to emergency feedwater flow.
   c) When RC pumps have coasted down, verify natural circulation flow is established.
   d) Attempt to determine and correct cause of RCP's tripping.
   e) Restart one RC pump in each loop per RC Pump operation procedure OP 1103 6.
   f) If RCP's cannot be started, maintain RC temperature by manually positioning the turbine bypass valves. For cooldown see cooldown by natural circulation, OP 1202 34.

2. During heatup or cooldown with the reactor critical, perform the following:
   a) Electromagnetic Relief valve reseats.
   b) OTSG levels increasing due to emergency Feedwater flow.
   c) Open the electric driven emergency feed pump discharge pump valve.
   d) Start the electric driven emergency feedpump.
   e) Trip the main feed pumps.

f) Attempt to determine and correct the cause of the RCP's tripping

g) Restart one RC pump in each loop if cooling down or if possible 4 RCP's if heating up per RC pump operation procedure OP 1103 06.

h) If pumps are started secure emergency feed line up.

i) If RCP's can not be started, maintain RC temperature by manually positioning the turbine bypass valves. For cooldown see cooldown by natural circulation, OP 1202 34.

5. LOSS OF FLOW DURING HEATUP OR COOLDOWN WITH RC PUMPS

5.1 Symptomes

1. RC Flow decreasing
2. All 4 RCP's trip annunciated
3. RC temperatures, pressure, and pressurizer level increasing

5.2 Immediate action:

1. If safety rods are withdrawn, verify Reactor Trip.
2. Perform or verify the following in the feedwater system:
   a) Main Feedwater block valves shut
   b) Startup feedwater block valves shut
   c) Emergency block valves open
3. Open the #2 emergency feedwater pump discharge valves and start the pump.
4. If the turbine bypass valves are open shut them.
5. Verify natural circulation flow is established.

5.3 Follow Up Action:

1. Verify OTSG levels increasing to or at natural circulation level. (   ).
2. Attempt to determine and correct cause of RCP's tripping.
3. Restart RCP's per RC pump operation procedure OP 1103 06
4. If RCP's can be started secure emergency feed line up and restablish normal feedwater line up.
5. If RCP's cannot be stared, maintain RC temperature by manually positioning turbine bypass valves. For cooldown, see Cooldown By Natural Circulation, OP 1202 34.

CFR UNIT #1

STEAM SYSTEM RUPTURE

OP 1202 24

1. PURPOSE

This procedure provides necessary operator actions in the event of a steam system piping failure.

2. DESCRIPTION

2.1 Rupture of a steam or feed line has a range of effects based on size and location of the rupture. This procedure treats worse case effects, for smaller breaks the operator action should be dictated by the situation.

2.2 The steam system rupture treated in this procedure is any rupture down steam of the last normal feed line check valves. The rupture may or may not be isolated by the closure of the main turbine stop valves.

3. LIMITS AND PRECAUTIONS

3.1 For a 34 inch double ended rupture at end of life conditions and maximum worth rod stuck out of core will cause return to criticality within the following indicated times, if no operator action to stop flow to the affected OTSG is taken:

ICS Situation	Time
No control: Feed flow max. from One main feed pump	100 seconds
Control at minimum OTSG level	175 seconds

3.2 If the ES systems are actuated by decreasing RC system pressure or high RB pressure operator action will be required to prevent the pressurizer from going solid.

4. STEAM SYSTEM BREAK DOWNSTREAM OF THE LAST FEED LINE CHECK VALVES

4.1 Symptoms

1. Rapid decrease of steam pressure and/or unit electrical load dependent upon ICS mode of operation. The ICS should cause increased feed flow due to megawatt or turbine header pressure errors.
2. Rapid decrease in pressurizer level.
3. Rapid decrease in reactor coolant pressure.
4. Rapid reduction in reactor coolant system cold leg temperature.
5. Reactor trip initiated by high flux (negative moderator coefficient) or low coolant pressure.
6. For a rupture inside the reactor building: Indication of increasing reactor building pressure, temperature, and radioactivity (if an OTSG tube leak existed).
7. For a rupture outside the reactor building: Noise may be heard in the control room or reports made from personnel outside the control room.

4.2 Immediate Action

1. Trip the reactor if not already tripped.
2. Follow normal reactor trip procedure OP 1202 C4.
3. Upon determination of which loop has ruptured, perform the following:
   a) Select the unaffected loop for header pressure control.
   b) Close all feed block valves in the loop ruptured.
   c) Verify minimum level control is established in the unaffected OTSG.
   d) Close the turbine bypass valve in the affected loop.
   e) Close Auxilary Steam supply valve on affected loop.
   f) Throttle the bypass valve in the unaffected loop to establish RC cooldown.
4. If pressurizer level is below the recorder scale, or RC pressure continues to decrease, manually initiate HP injection.
5. If the reactor returns to criticality, perform the following:
   a) Manually initiate HP injection
   b) Trip one RC pump in each loop. If RC pressure-temperature relationship is such that would cause cavitation. Stop all RC pumps.

6. Observe if steam pressure is being restored in one or both OTSG's. If so, perfrom section 4.4. If not, continue with 4.3.

4.3 Follow-up Action

1. Have plant personnel attempt to locate rupture without endangering themselves.
2. One by one, isolate the following systems:
   *a) Reheat steam system.
   b) Auxiliary steam supply from main steam line.
   c) Turbine bypass system.
3. If no effect is noticed upon isolation of the above systems, restore the system to normal operation.
4. If the above actions have not isolated the rupture, the only remaining locations are safety valves on both steam lines. Pressure should be re-established by reseating the steam safety valves.
5. When pressure has been re-established in one or both loops, continue with Section 4.4.

4.4 Follow-up Action

1. Establish feed flow to the unaffected OTSG(s). If necessary, manually throttle the turbine bypass valve(s) to allow feeding from the condensate pumps.
2. Feed water block valves for the OTSG with the rupture should be left shut.
3. The unaffected OTSG is being controlled at minimum level.
4. When pressurizer level is restored, the HP injection may be blocked and normal make-up and boron addition established.
5. Decay heat pumps should be blocked and stopped as long as RC pressure is greater than 200 psig.
6. Upon establishing normal cooldown per OP 1102 11, stop the RC pump in the loop with the rupture.

1202 26

OFR UNIT #1

LOSS OF STEAM GENERATOR FEED

OP 1202 26

1. PURPOSE

1. To define operator action following loss of feed water flow to one or both steam generators.

Operation	Section
Loss of Feed to Both Steam Generators	4
Loss of Feed to One or Both Steam Generators	5
Loss of Feed to One Steam Generator Case I	6
Loss of Feed to One Steam Generator Case 2	7
Feed System Break Upstream of Last Feed Line Check Valves	8

2.1 The combination of feed control valves and variable speed feed pumps are normally used to control Feedwater flow. The Feedwater pump turbines are supplied with steam from the Auxiliary Steam System, which is fed from main steam, low pressure extraction, or the Auxiliary boiler.

2.2 In the event of a loss of both feed water pumps the main and startup feed block valves shut and the emergency feed block valves will open. The #2 (steam driven) emergency feed pump will automatically start. In the event the steam supply to #2 Emergency feed pump is lost, #1 (electric driven) emergency feed pump may be lined up and started to insure a supply of feed to the OTSG's.

2.3 SETPOINTS:

1. OTSG Level following loss of both FW pumps to be at low level setpoint on the start up range.
2. Loss of both FW pumps trips the main turbine and starts the emergency feed pump.

3. FW control valve delta P equals 35 psid.
4. Low Feed pump suction pressure (less than 50 psig) will result in Feed pump(s) tripping.

3. LIMITS AND PRECAUTIONS:

1. Minimum emergency FW temperature equals 90F.
2. Verify a low/zero FW flow indication by comparison with an independant signal, i.e. decreasing level, valve position, pump speed. An erroneous flow signal may otherwise cause overfeeding.

4. LOSS OF FEED TO BOTH STEAM GENERATORS:

4.1 Symptoms:

1. Decreasing OTSG levels
2. Decreasing FW flow
3. Main turbine trip
4. Probable reactor trip if above 5% power.

4.2 Immediate Action:

1. Verify reactor trip or manually trip reactor
2. Verify emergency FW pump starts
3. Verify OTSG levels controlled at low level setpoint on startup range.
4. Verify startup block valves A and B closed
5. Verify main block valves A and B closed
6. Verify emergency block valves A and B opened
7. Verify turbine bypass valves control at 1010 psig

4.3 Follow Up Action:

1. Rectify FW pump trip condition
2. Restart FW pump and initiate normal feed when level is at low level setpoint
3. Shutdown emergency FW pump
4. Prepare for reactor-turbine restart

5. LOSS OF FEED TO ONE OR BOTH STEAM GENERATORS:

5.1 Symptoms:

1. Decreasing OTSG levels
2. Decreasing FW flow
3. Decreasing FW pump turbine speed

5.2 Immediate Action:

1. Attempt to recover FW pump turbine speed by increasing control station speed signal in manual.
2. Attempt to open failed valve in manual or by hand to restore FW flow.

5.3 Follow Up Action:

1. If FW pump turbine speed is recovered in manual and FW flow and OTSG level are restored, return to normal operation.
2. If both OTSG's are affected and FW valve delta P cannot be increased to normal:
   a) Trip the reactor
   b) Trip both feedwater pumps
   c) Verify Emergency Feed control at minimum OTSG level.
3. If one OTSG is affected and FW flow control valve cannot be opened: Verify that load on the unaffected OTSG is reduced by the ICS to balance Delta T_c and that reactor power is reduced to match FW flow. If this automatic action is successful, the system load should settle out at about 15% (capacity of the startup FW valves). If not successful, runback load and trip the reactor and repair valve as required.

6. LOSS OF FEED TO ONE STEAM GENERATOR CASE I

6.1 Symptoms:

1. Decreasing OTSG level.
2. FW flow indication high.
3. Decreasing FW pump turbine speed.

6.2 Immediate Action:

1. Open FW control valve in manual to control OTSG level, and balance the load on the OTSG's to obtain a zero ΔT_c.

6.3 Follow Up Action:

1. Correct flow signal and return to automatic operation.

7. LOSS OF FEED TO ONE STEAM GENERATOR CASE 2

7.1 Symptoms:

1. Decreasing OTSG level.
2. FW flow zero.
3. FW control valve Delta P increases.

7.2 Immediate Action:

1. Attempt to open FW block valve manually.

7.3 Follow Up Action:

1. If FW block valve cannot be opened:

   Verify that load on the unaffected OTSG is reduced by the ICS to balance Delta T_c and that reactor power is reduced to match FW flow. If this automatic action is successful, the system load should settle out at about 15%.

   If this automatic action is not proper runback system load and trip the reactor. Repair the valve as required.

8. FEED SYSTEM BREAK UPSTREAM OF LAST FEED LINE CHECKS VALVES

8.1 Symptoms

1. Loss of electrical load and/or steam pressure, depending upon ICS control mode. Loss rate is slower than in steam line rupture until OTSG (s) have boiled dry.
2. Large upset in OTSG feedwater flow.
3. Increase in RC pressure and temperature. Opening of electromagnetic relief valve.
4. Increase in pressurizer level.
5. Reactor trip initiated by high temperature, high pressure, or pressure-temperature relationship.

8.2 Immediate action:

1. If any reactor protection limits are exceeded, without a reactor trip, trip the reactor.
2. Follow normal reactor trip procedure OP 1202 04.
3. Verify emergency feed flow is established. If not, establish emergency feed flow as follows:
   a. Shut main and start-up feed block valves
   b. Open the emergency block valves
   c. Open #1 emergency feed pumps discharge valve.
   d. Start the #1 emergency feed pumps

   Observe the following:
   a. Approximately equal, non-zero, emergency feed flows have been established to both OTSG loops.
   b. RC cooldown has begun.

   NOTE: If these are happening, perform section 8.4. If not, perform Section 8.3.

8.3 Follow-up action:

1. Line up emergency feed pump discharge valves to feed to one OTSG only.

   NOTE: Feed flow readings may be used as a clue to which OTSG emergency feed line has the leak.

2. If pressure is not restored in the OTSG being fed, line up to feed the other OTSG.
3. If pressure is not restored in the OTSG being fed, continue with 8.4.

8.4 Follow-up Action:

1. Attempt to isolate rupture, if rupture can be isolate the plant may be returned to power. If rupture is in emergency feed line, the plant must be cooled down and the emergency feed line repaired.
2. If rupture cannot be isolated, proceed with RC cooldown using one or both OTSG(s) per cooldown procedure OP 1102 11.

1202 29

OFR UNIT #1

PRESSURIZER ABNORMAL OPERATIONS

OP 1202 29

1. PURPOSE

To provide procedures for the following abnormal conditions:

Title	Section
Leaking or Inoperative Pilot Operated Relief Valve	4
Leaking Code Relief Valve	5
Inoperative Heater or Heater Bank	6
Malfunction In Level Indication Or Control	7
Insufficient Spray Valve Bypass Flow	8
Spray Valve Failure	9

2. DESCRIPTION

2.1 The principal response to a leaking pilot operated relief valve is to close the isolation valve and repair at the next outage.

2.2 The principal response to a leaking code relief valve is to determine the leakage rate and verify that continued operation is permissible.

2.3 The principal response to an inoperative heater is to locate the fault and de-energize the affected heater.

2.4 The principal response to a malfunction in level control depends on its magnitude.

If a discrepancy in level indication exists which does not compromise safety, the malfunctioning component may be recalibrated or replaced and operation continued.

If a discrepancy exists which compromises safety, the plant must be shut down.

(TS) 2.5 The principal response to insufficient bypass spray flow is to manually throttle the main spray valve or to spray intermittently to maintain spray line temperature within limits.

2.6 The principal response to an inoperative spray valve depends on the failure. If the valve has failed open, close the block valve. If the valve fails closed, maneuver with caution.

3. LIMITS AND PRECAUTIONS

3.1 Pressurizer heater capacity of at least 126 KW should be operable during reactor operation.

3.2 The intermediate position limit switch (auto spray setting of 40%) for the spray valve is overridden when in manual control.

3.3 Complete loss of pressurizer level indication requires immediate reactor shutdown.

3.4 The pressurizer spray block valve cannot be jogged. It will open or close completely when a control signal is initiated.

(TS) 3.5 The maximum ΔT between spray fluid and the pressurizer is 430 F.

4. LEAKING OR INOPERATIVE PILOT OPERATED RELIEF VALVE

4.1 Symptoms

*1. Relief valve discharge line temperature alarm (200 F).

2. Excessive makeup flow.

*3. High quench tank temperature.

*4. Increasing quench tank level

4.2 Probable Cause

1. Leakage may be caused by failure of valve to reseat properly or a damaged seat. Failure to operate may be caused by loss of the pilot actuator power supply, binding of the valve or actuator or failure of the pilot actuator.

4.3 Immediate Action

1. Close the relief valve block valve if leakage is greater than that allowed by Technical Specifications. (No immediate action is required if the valve fails to open.)

4.4 Followup Action

1. Repair at the next shutdown.

5. LEAKING CODE RELIEF VALVE

5.1 Symptoms

1. Relief valve discharge line temperature alarm (200 F).
2. Excessive makeup flow.
*3. High quench tank temperature.
*4. Increasing quench tank level.

5.2 Probable Cause

1. Valve leakage may be caused by failure to seat properly or a damaged seat. This procedure addresses itself to those leaks within limits of Technical Specifications. For larger leaks, refer to OP 1202 26, Loss of Reactor Coolant.

5.3 Immediate Action

1. If the leakage is in excess of that allowed by Technical Specifications, the reactor must be shut down and the valve repaired.

5.4 Followup Action

1. Repair prior to resuming operation.

6. INOPERATIVE PRESSURIZER HEATER(S)

6.1 Symptoms

1. A heater bank fails to indicate energized for RC pressure below the setpoint.
*2. Fault indication (if provided in the electrical circuitry) on one ore more heater groups.
*3. Abnormally low heater current.
4. Abnormally slow RC pressure response when heaters are energized.

6.2 Probable Cuase

1. Failed heaters.
2. Defective control system.

6.3 Immediate Action

1. Operate bank manually for defective pressure control system.
2. De-energize heaters showing fault indication for fault condition.

6.4 Followup Action

Reactor operation may continue within the limits of 3.1.

7. PRESSURIZER LEVEL INDICATION OR CONTROL MALFUNCTION

7.1 Symptoms

1. One compensated level indication differs from the other by more than 1.0%, as recorded on RC1-LR.
2. One uncompensated level indication differs from the others by more than 1.0%, as read by plant computer.
3. The uncompensated indication of a transmitter disagrees with the compensated indication by more than:

   7 inches for pressurizer temperature 600 to 670F.

   16 inches for pressurizer temperatures below 600F.
4. The uncompensated indication on the remote indicator disagrees with the compensated indication by more than:

   16 inches for pressurizer temperature 600 to 670F.

   26 inches for pressurizer temperatures below 600F.
5. RC makeup flow does not maintain pressurizer level at set point.

7.2 Discussion

1. Comparison of uncompensated indications will help locate a transmitter problem. Manual compensation of these indicators will help locate a level compensating amplifier or recorder problem.
2. Checking makeup valve position demand may help locate a problem in the amplifier or recorder.
3. Comparing indications using each temperature transmitter will locate a problem in the element or transmitter.
4. Pressurizer level control problems with no accompanying level information discrepancy indicates a problem in the level controller or makeup valve.

7.3 Immediate Action

1. Label the affected level indications as out of calibration. Calibrate the affected component(s) at the earliest available opportunity.
2. If affected indications vary by more than 4.0%, label the offending components as out of service. Take immediate action to schedule repairs as only two channels remain in operation.
3. Control pressurizer level on properly operating chennels. If two level indications agree, within the limits above, switch level control to one of them.

NOTE: When switching transmitters, take manual control of the makeup valve to avoid response to potentially false signals.

4. If there is no agreement, or the remaining chennels drift apart by more than 4.0%, take manual control of the level, shut off heaters and begin plant shutdown. Use the lowest indication for heater protection and the highest to prevent excessive level.

   NOTE: If pressure cannot be maintained, and the operator calculates that the heaters are covered, heaters may be manually operated to maintain RC pressure as required.

5. If all pressurizer level indication is lost, the reactor must be shutdown, if operating, and the NSS must be cooled down. This can be accomplished by doing the following:

   NOTE: This procedure assumes that pressurizer level was in its normal range when indication was lost.

   CAUTION: This assumes zero leakage from the system. Careful attention should be paid to RC system pressure and other parameters to detect any sign of a leak. A sudden increase in RC pressure might indicate that the pressurizer has reached a solid water state. Conversly, a sudden decrease in the pressure might indicate that the pressurizer has emptied.

   a. Shut the plant down per appropriate plant procedures and the following:

   b. Concurrently with cooling down add water to the RC system from the MU tank, in amounts derived from Figure 1202 29-1. By knowing the gal/in of the MU tank -known- amounts of water can be added to the RCS.

      NOTE: Since the MU tank volume is insufficient to supply all of the water required, known amounts of reactor grade water will have to be -batched- into the MU tank from an appopriate source.

      CAUTION: A careful inventory of T(ave) and makeup water, both to the RCS and the MU tank, will he to be maintained.

   c. Shutdown the reactor and continue cooldown. Add water as indicated by Figure 1202 29-1. See example 1 and 2.

7.4 Followup Action

1. Using the analytical checks above, locate and correct the malfunction. Maintaining a constant load (for cases where reactor is not shutdown) will aid in locating trouble.
2. Reactor operation with the level compensating amplifier out of service requires that pressurizer level and headers be controlled manually.

8. INSUFFICIENT PRESSURIZER SPRAY VALVE BYPASS FLOW.

8.1 Symptoms

1. Abnormal difference between RCS and pressurizer boron concentrations.
2. Abnormal difference between pressurizer spray line and loop A T cold temperatures.

8.2 Discussion

1. The bypass spray valve is adjusted for 1 - 1.5 GPM for normal operating conditions to maintain pressurizer boron concentration close to that of the RCS and protect the spray nozzle.

8.3 Immediate Action

(TS) 1. Manually operate the spray valve to maintain line temperature within limits.

8.4 Followup Action

1. At the earliest available opportunity, reset the bypass spray flow

9. PRESSURIZER SPRAY VALVE FAILURE

9.1 Symptoms

1. Spray valve indicates closed when RC pressure is above its opening setpoint.
2. Spray valve indicates open when RC pressure is below the closing pressure.
3. Abnormal change in RC pressure without associated change in pressurizer level.

9.2 Probable Cause

1. Loss of valve motor or control power.
2. Valve motor failure.
3. Valve binding.
4. Control contact failure.

9.3 Immediate Action (Failed Open)

1. Try to close spray valve manually.
2. Close spray block valve if unable to close spray valve manually.
3. Verify heaters return pressure to normal.

Immediate Action (Failed Closed)

1. Attempt manual control.
2. Maneuver with caution observing RC

9.4 Followup Action (Failed Open)

1. Manually control spray flow using the block valve if necessary.

Followup Action (Failed Closed)

1. Stabilize reactor power until repairs can be made.

EXAMPLE 1

Reactor at power, T(ave) 579F, pressurizer level in normal range

1. While cooling down from 579 to 565F. add 1150 gallons of water.
   See Figure 1202 29-1.
2. While cooling down from 565 to 555 F, add 1000 gallons of water.
   See Figure 1202 29-1.
3. Continue cooling down and adding 2000 gallons of water between each of the following temperatures:

   555 - 525 - 500 - 465 - 430 - 395 - 350 - 305 - 250 - 185 - 90

   NOTE: 2000 gallons of water is approximately equal to 84 inches of pressurizer level. This would eliminate the danger of uncovering the heaters or going solid when starting from the normal range of the pressurizer.

EXAMPLE 2

Reactor shutdown, T_c 500 F, Pressurizer level in normal range.

1. While cooling down from 500 to 490F., add 700 gal. of water.
   See Fig. 1202 29-1.

FIGURE 1 RC SYSTEM INVENTORY VS T_{AVE}

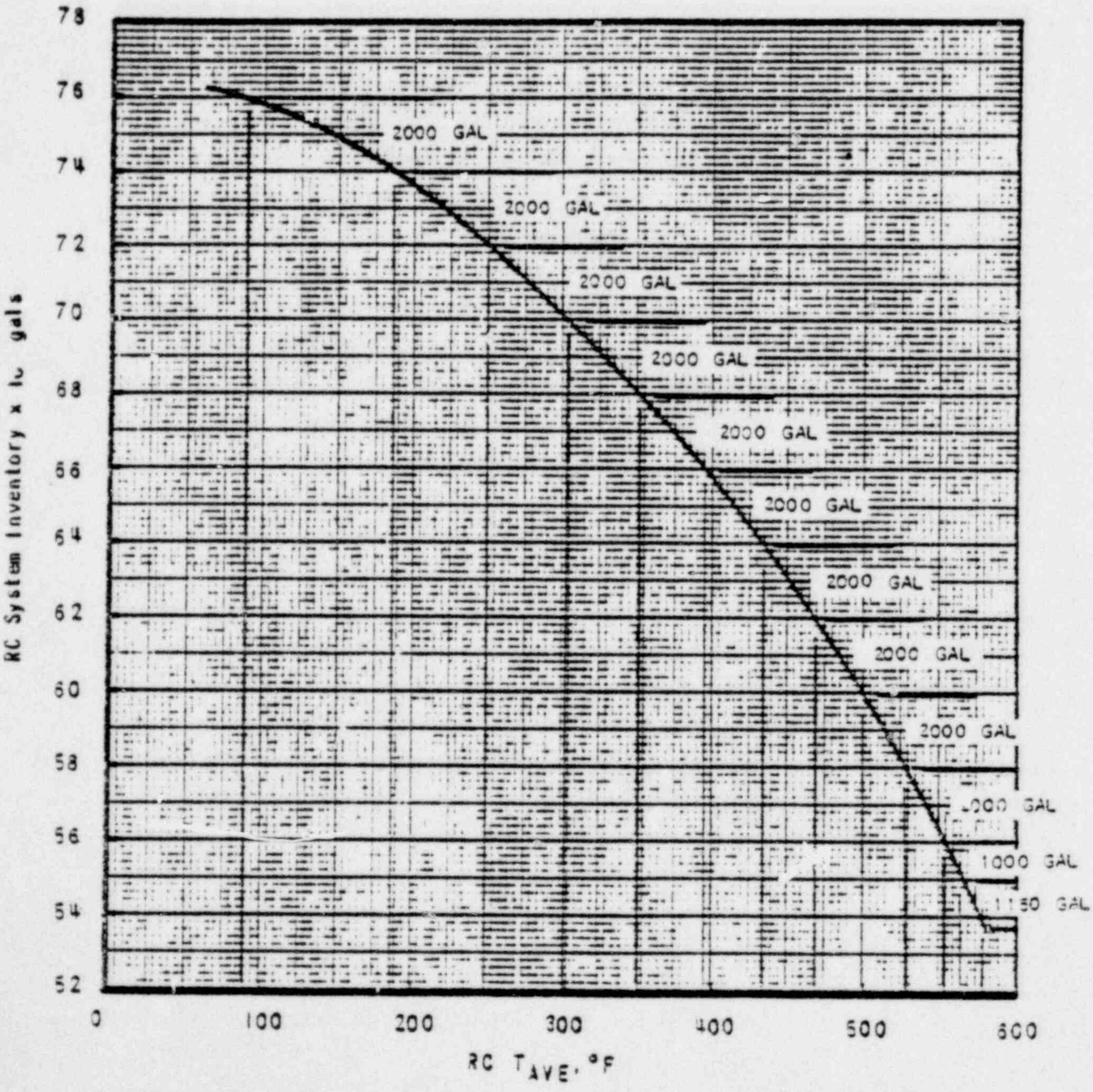

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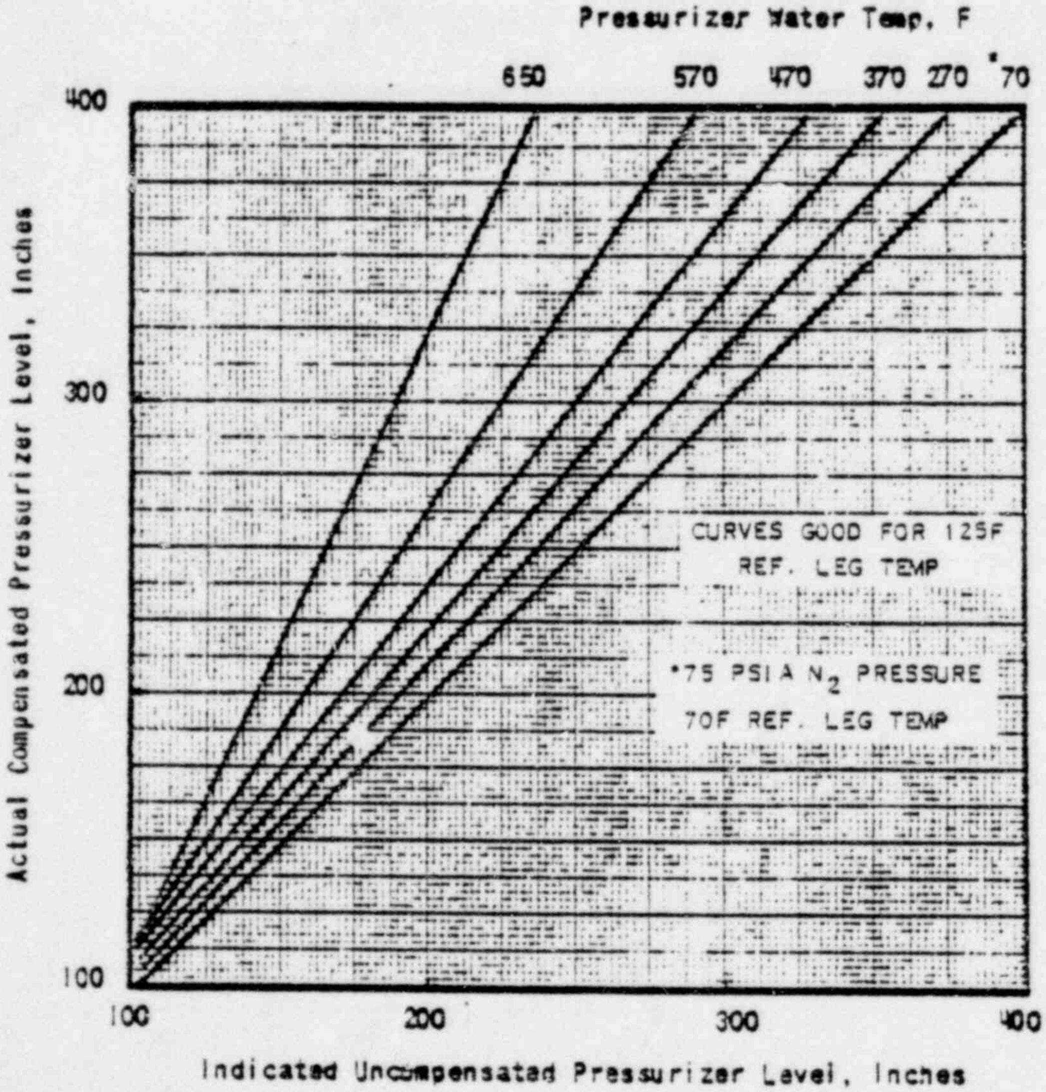
FIGURE 2 ACTUAL VS INDICATED UNCOMPENSATED PRESSURIZER LEVEL



OFR UNIT #1
INTEGRATED CONTROL SYSTEM
ABNORMAL OPERATION
OP 1203 1

1.0 PURPOSE

The purpose of this procedure is to describe operation of the integrated control system during periods when the unit master control station is in track as a result of one or more of the major control stations in manual (or hand) or as a result of other external inputs.

2.0 DESCRIPTION

Tracking is defined as the mode of operation when, due to some abnormal condition, the unit is being limited in its production of megawatts. In order to maintain coordination of all control variables, the unit must track, or follow, that condition. Any one or more of the following conditions will cause the unit master to go into the tracking mode:

(1) Unit operating under cross-limits (when the difference between reactor power and reactor demand exceeds plus or minus 5% or the difference between feedwater demand and feedwater flow exceeds plus 5%).
(2) Transfer of the steam generator/reactor master to manual (turbine following).
(3) Transfer of both feedwater loop demands to manual (turbine following).
(4) Transfer of either the Diamond station or the Bailey Reactor demand station to manual (turbine following).
(5) Transfer of the turbine control station to manual (reactor/steam generator following).
(6) Tripping of both generator breakers.
(7) Reactor trip

2.2 NOTE: In all cases, the unit load demand will follow the generated megawatts in order to maintain the proper relationship between those stations in automatic and those in manual.

3.0 LIMITATIONS AND PRECAUTIONS

(1) Do not exceed specified loading (or unloading) reates.

(2) If a station is in manual, it is the operators responsibility to provide the necessary inputs for unit runback should limits be exceeded. In effect, the operator must perform those functions that would otherwise be performed automatically.

(3) Unless absolutely necessary and except for plant startup, do not put more than one feedwater valve or feedwater pump in manual at the same time. The unit does not necessarily go into track with the feedwater valves or pumps in manual. With the remainder of the system in automatic, feedwater control may become difficult during load changes.

(4) If, while controlling the reactor from either the Bailey reactor demand station or the Diamond station, a FW cross-limit occurs, adjust the reactor power to be compatiable with the total feedwater flow.

(5) If, while controlling with both loop feedwater demands stations in hand, a reactor cross-limit occurs, adjust the total FW flow to be compatable with reactor power.

(6) While in manual control at either the Bailey reactor demand station or Diamond reactor control station, do not exceed 102% power.

(7) Before switching the Diamond Rod Control Station to automatic, ensure a valid neutron signal exists from the selected Power Range N.I. channel to the ICS.

4.0 PROCEDURE

4.1 Neutron and/or Feedwater Cross-Limits

(1) If the reactor, regardless of the unit load demand, changes its power level to a point more than plus or minus 5% different from the neutron power demand, the feedwater control will respond to this change by increasing or decreasing feedwater flow to be compatable with the changed neutron power level. With both feedwater demand stations in manual the automatic control will not function, and the operator must adjust feedwater control as necessary to maintain the proper feedwater flow/neutron power ratio.

(2) If the feedwater flow drops more than 5% below the feedwater demand signal, the reactor control will respond by decreasing neutron power, automatically. If the reactor control is in manual, at either the Diamond Station or the Bailey Reactor Demand Station, the reactor power must be reduced manually. An increase in feedwater flow above 5% of the demand signal will not automatically increase reactor power.

4.2 Transfer of Steam Generator/Reactor Demand To Manual

(1) This places the control in a turbine following mode where steam generation is under operator control and the turbine control (assuming it is in automatic) has responsibility for maintaining turbine header pressure. An increase in steam generation corresponds to an increase in turbine output because the turbine header pressure is maintained while steam flow rate is increased. Likewise, a reduction in steam generation will result in a reduction in turbine output.

(2) The reactor and feedwater controls respond in parallel to manually-initiated load changes at the reactor/steam generator demand station. This control mode is characterized by relatively slow system response, but very accurate and rapid turbine header pressure control.

4.3 Transfer Of Both Feedwater Loop Demand Stations To Manual.

(1) The unit load will respond to changes at either feedwater demand station. The unit delta TC will revert to manual and should be monitored and maintained at less than 5F.

4.4 Transfer Of Either The Diamond Or The Bailey Reactor Demand Control Station to Manual.

(1) Unit load may now be controlled from the Bailey or Diamond Station provided the Diamond or Bailey Reactor Demand Station, respectively, is in automatic. The reactor power must be kept within 10 and 102% to satisfy the steam generator/reactor relationship.

(2) The reactor power must be kept within the specified limits with respect to feedwater flow.

(3) The Diamond Station, if in manual, cannot be transferred to automatic if a neutron error of greater than plus or minus 1.0 % exists or if auto ICS power is lost.

4.5 Transfer Of The Turbine Station To Manual.

(1) This places control in the reactor/steam generator following mode where the reactor/steam generator maintains the correct steam conditions (turbine header pressure) as the turbine output responds to manual control inputs. A rapid response to load changes is characteristic of this control mode.

4.6 Tripping Of Both Generator Output Breakers.

(1) With both generator breakers open, the unit is removed from the grid and no longer supplies power to it. The generator will supply the plant load and the demand upon the unit is a function of the power required by the plant. The unit load demand will follow the actual generated megawatts.

4.8 Reactor Trip

(1) When the reactor trips, the turbine is also tripped, the feedwater is controlled to maintain the steam generators at minimum level, and the turbine bypass valves will control at header pressure set point +125 psig.

(2) The Diamond rod control station will revert to manual upon receipt of a reactor trip signal.

OFR UNIT 1
OPERATION WITH LESS THAN
FOUR REACTOR COOLANT PUMPS
OP 1203 03

1. PURPOSE

This procedure describes operation with less than four RCPs. The operations are as follows:

Operation	Section
Three Pump Operation	4
Two Pump Operation (One in each loop)	5
RC Pump Restart	6

2. DESCRIPTION

2.1 The nuclear steam supply system (NSS) is capable of operation, at reduced power, with less than four (4) RCPs. Operation with less than all reactor coolant pumps calls for equalizing return temperatures to the core inlet and maintaining an average temperature of 579F in the reactor coolant loop with the greatest number of pumps running.

2.2 Loss of operating reactor coolant pumps.

1. The loss of a single RCP (with 4 pumps initially operating) results in the ICS automatically running the unit back to 75 percent of rated load if the load is above 75 percent.
2. The loss of any combination of reactor coolant pumps which results in the following system conditions will cause a reactor trip.
   a. One reactor coolant pump in each loop whenever reactor power is above 55%.
   b. No operating reactor coolant pumps in one loop.
   c. Reactor power 1.10 times greater than flow minus a reduction due to imbalance.

3. LIMITS AND PRECAUTIONS

1. The average reactor coolant temperature in the reactor coolant loop with the greatest number of operating RCPs is to maintained to 579.
2. The maximum reactor coolant inlet temperature difference allowed during steady state conditions is 5F whenever the power level is above the point where neither steam generator is on low level control.

(TS)3. Move the axial power shaping rod assemblies as required to maintain an imbalance within limits for the number of RC pumps operating.

4. THREE RC PUMP OPERATION

4.1 Auto action.

1. Feedwater flow in each steam generator will automatically be controlled by the integrated control system to approximately a 2.4 to 1 flow ratio. The unit will run back to 75% load if the reactor does not trip.

4.2 Operator Action.

1. If any portion of the control system is not in auto the operator should perform the runback function on that control station manually.
2. If loop FW demand stations are in hand the operator should ratio the feed demands feeding to the loop with 2 RC pumps operating approximately twice as much feed flow as the loop with one RC pump operating. Then fine tune his ratio to obtain a zero ΔT_C.
3. If control system is in auto verify runback and proper feed water ratioing.
4. Verify that auto/manual transfer switch has selected the loop with the greatest number of pumps running for Tave control purposes.

5. TWO RC PUMP OPERATION (ONE IN EACH LOOP)

5.1 Auto Action

1. If the reactor is less than 55% full power the system will run back to 40% load and operation will be as with four RC pumps.

5.2 Operator Action

1. If any portion of the control system is not in auto, manually perform runback on that control station.
2. If the control system is in auto, verify runback:

RC PUMP RESTART

6.1 Restart of 1 RC pump operating

1. Reduce the load demand to <22% F.P.

   NOTE: As the unit decreases load, the feedwater loop with one RCP operating will be level limited first. This will cause ΔT_c to be greater than 5F.

2. At 22 percent F.P. perform or verify the following:

   a. The cause of the trip is determined and cleared.

   b. Reactor power is less than 22% . If not, place reactor demand in hand and reduce power.

   c. Restart the RC pump per Reactor Coolant Pump Operation, OP 1103 06.

6.2 Restart of two RC pumps.

1. Reduce load demand to <22% F.P.

2. At 22 per cent F.P. perform or verify the following:

   a. The cause of trip is determined and cleared.

   b. Reactor power is less than 22% if not place reactor demand station in hand and reduce power.

   c. Restart the RC pumps per Reactor Coolant Pump Operation, OP 1103 06.