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268TH GENERAL MEETING

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
268th GENERAL MEETING

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Room 1046
1717 H Street, N.W.
Washington, D.C.
Friday, August 13, 1982

The Committee met, pursuant to notice, at 1:45 p.m., PAUL G. SHEWMON (Chairman of the Committee) presiding.

ACRS MEMBERS PRESENT:

PAUL G. SHEWMON
J. CARSON MARK
MILTON S. PLESSET
CHESTER P. SIESS
DADE W. MOELLER
MYER BENDER
WILLIAM KERR
MAX W. CARBON
HAROLD ETHERINGTON
DAVID A. WARD
JESSE C. EBERSOLE
HAROLD W. LEWIS
DAVID OKRENT

ALSO PRESENT:

J. RAULSTON
R. PIERCE
L. MILLS
D. ORMSBY
B. COTTLE
P. SHEMANSKI
T. KENYON
E. ANDENSAM
T. NOVAK

DESIGNATED FEDERAL EMPLOYEE:

RAYMOND F. FRALEY

P R O C E E D I N G S

MR. EBERSOLE: We're going to talk about Watts Bar Unit 2. I'm not sure how many of the committee here are familiar with the degree similarity between Watts Bar and Sequoyah. To give you an idea about what this is, about 40 years ago we had a hydro project in Tennessee called Cherokee and we needed to make lots of electricity to make aluminum to make bombers in a hurry. So they found a convenient place in the valley changed the elevations above sealevel, and set this plant down in modular form in another part of the country over there.

That entire project, as I recall, was started, and it was from the time of the start of operation until the time they put it into functioning operation, which was one year. In all the intervening years, I think this project represents the sole attempt to attempt to duplicate that process. It is my impression, at least, that TVA has taken a great deal of advantage of the use of identical drawings at Sequoyah and may have in fact reproduced a great many of them. I'm not sure how many.

But it is fair to say that if Sequoyah is a good plant to operate, then Watts Bar is probably better, except for of course the considerations of quality assurance and administrative matters of that sort. If the designs are essentially the same with some improvements, some of which one might argue about, like the Westinghouse 3-D steam generator, which was thought to be an improvement in a recent day.

I want to comment on some of those differences, most of which are mentioned in our proposed letter on the topic. Late in the construction program, in 1981, early '82, there was rather a serious quality assurance breakdown, principally in the construction, but not only that, also in the design area. This has resulted in a monumental flow of deficiency reports.

This problem was picked up and a rather large shift in the organization and management of the QA problem which was invoked. Insofar as I can tell, it's presently working. That doesn't mean the job is done by far. There is still a great inflow of paper and processing of paper on the deficiency reports and how to handle them.

TVA plans to have an independent contractor review the design and the construction of what they call a typical vertical cross-section of this, which is really just a way of saying they are going to pick some fraction of the plant that represents a top to bottom view of what the design is and examine it in intensive detail and see whether or not the quality assurance reorientation has in fact covered the aberrations that occurred in this interval.

Regarding the site, it is interesting to note that from a seismic point of view I think the plant is essentially in the same seismic area as Watts Bar. It's only about 35 or 40 miles away. They have hydrology problems that are quite similar. They use a flooding type concept of design which resembles very strongly that of Sequoyah, which permits the plants to be flooded, the plant to be flooded, after a prescribed interval of time which they think they can forecast, within which time they go in and rig the plant to take the flooded condition for an indefinite interval.

One of the interesting aspects of the plant problem down there was to me the use of cement mortar to line the cooling water system piping and perhaps some other piping down there, in the course of doing this, to recognize a rather new approach to the use of piping, changing out carbon steel to stainless steel for the smaller sizes, but for the larger pipes to go through this process of lining the pipes with mortar.

I believe they have had a little QA problem in this particular aspect itself, and this, as you might expect, one will be interested in the seismic performance of this lining material, particularly if it tends to deteriorate over time and becomes cocked or loosened so that it doesn't perform in the way that it was originally shown to perform when they did bending tests on it.

In short, if it cocks itself to represent a potential cascade failure in the event we have seismic shocks or other mechanical upsets, then by some sudden entrainment of this debris you could have various consequences in the performance of the piping systems.

We are a little bit, not too much, off schedule here already. I think it would be better and more profitable to simply plunge into the presentation by TVA, for which we have here about three and a half-odd hours. And we're going to start that by a status.

However, I want to ask other members of the Committee -- Mr. Bender, Mr. Ward -- if they have anything to add to this presentation.

We had a meeting, incidentally, on the 10th. We had a meeting at the site on April the 30th, including a site visit. And topics will be brought up at least in some of the cases which were discussed at those meetings.

MR. MARK: The Staff and the Applicant will point out the differences between this and Sequoyah?

MR. EBERSOLE: There are differences, outstanding issues, et cetera. The agenda is in the pink-covered sheet. Does everyone have the pink handout?

MR. MARK: Watts Bar has a nice little dam about three miles up above.

MR. EBERSOLE: It has a dam above the plant proper, very few miles. It also has an old steam plant, TVA's original steam plant.

MR. MARK: And it has hydroelectric. Is that important to the backup of the plant?

MR. EBERSOLE: Is it important to the backup of the plant? I can't say that. Maybe TVA can say whether or not local power from Watts Bar is a prime asset for offsite power. I suspect it is.

MR. MARK: It's a very few miles away. There is about a 20-foot dam with a hydroelectric generating unit.

MR. EBERSOLE: I believe it's higher than that. I don't know whether they have deliberately enhanced --

MR. MARK: It has nothing to do except to feed Watts Bar if it wants to.

MR. EBERSOLE: Maybe whoever's going to do the presentation can talk about the value or lack of it above the nuclear plant.

MR. SHEWMON: Whoever is rubbing their microphone, would they stop it.

(Laughter.)

MR. MILLS: Are you ready for us to proceed, sir?

MR. SHEWMON: That's what we're waiting for.

MR. MILLS: I'm Larry Mills, Manager of Nuclear Licensing for TVA.

The Watts Bar nuclear plant, which consists of two identical units, was constructed and will be operated by TVA. These units employ pressurized water reactors furnished by Westinghouse. Each unit will operate at 3411 thermal megawatts, with an electrical output of 1218 megawatts.

Completion of these two units will provide TVA with seven operating nuclear units, including those currently in operation at Sequoyah and Browns Ferry nuclear plants, with a nuclear generating capability of approximately eight million kilowatts of electrical power.

The Watts Bar plant site is located in East Tennessee, and Dave Ormsby is pointing out the location of that.

(Slide.)

This slide also shows the locations of Sequoyah and Browns Ferry. Watts Bar is approximately 31 miles north-northeast of the Sequoyah nuclear plant and 45 miles north-northeast of Chattanooga, Tennessee. The plant site consists of a tract of approximately 1770 acres in Rhea County on the west bank of the Tennessee River. The 1770 acre reservation is owned by the United States and is in the custody of TVa. Also located within the reservation are the Watts Bar hydroelectric plant and the Watts Bar steam plant.

The contract for Watts Bar nuclear steam supply system was awarded to Westinghouse on August 27, 1970, and a construction permit was issued on January 23rd, 1973. TVA submitted a final safety analysis report on July 1, 1976, in support of an application for an operating license. This application was docketed by the NRC on October 4, 1976.

The construction completion schedule of August 1983 for Unit 1 and August 1984 for Unit 2 remains the same as we specified during the last ACRS Subcommittee meeting in April. This schedule includes plant modifications as a result of TMI-related requirements and modifications resulting from the Sequoyah licensing review. These modifications were included at Watts Bar because of the similarities between Sequoyah and Watts Bar.

Later today we will be providing more specific information on TVA organizations and training, but briefly: Staffing of the plant operating personnel was initiated with the appointment of a plant superintendent in July 1976.

Since that time the staff has grown to a present level of approximately 190 engineering and maintenance employees, 65 administrative and support employees, 7 senior reactor operators, 4 reactor operators, and 100 auxilliary unit operators. By fuel loading for Unit 1, 2 more senior reactor operators and 12 more reactor operators will be added.

The onsite operations personnel are currently involved in becoming familiar with plant systems and equipment and operating various systems and equipment during the plant construction test program.

Total plant staff for operation of both units will be approximately 560 people, composed of approximately 130 operators, 315 engineering and maintenance personnel, 120 administrative and support employees.

The remainder of the plant operations personnel are being trained by the Nuclear Training

Branch in the Divison of Nuclear Power at TVA. TVA's training program conforms with the requirements set forth in ANSI N18.1.

When we receive a license to operate Watts Bar, TVA will have accumulated approximately 31 reactor years of operating experience from operation of Sequoyah and Browns Ferry nuclear plants.

Now, David Ormsby, our licensing project engineer, will present a discussion of the comparison between Sequoyah and Watts Bar. And at the conclusion of that, I think we have a gentleman here who can address the location of the Watts Bar hydroplant and steam plant with regard to offsite power of the Watts Bar nuclear plant.

MR. MARK: Mr. Mills.

MR. MILLS: Yes?

MR. MARK: I'm just slightly curious. You have four operators at the moment and seven senior operators.

MR. MILLS: Yes.

MR. MARK: And you are going to add a fair number of operators within a rather -- well, not a terribly short time, but like a year. Where do you get operators? Do you grow them down there in Tennessee? Or do you swipe them from Oklahoma, or what do you do?

(Laughter.)

MR. MILLS: Dr. Mark, I couldn't say that they grow well on the hills of Tennessee. But in reality, we have our own training facility down there, and Mr. Cottle might like to address this, but --

MR. MILLS: When you say you have more coming --

MR. COTTLE: We have them on site. The seven senior reactor operators Mr. Mills referred to are shift engineers, assistant shift engineers, that we have on site that have been licensed on Sequoyah or Browns Ferry. We also have four individuals on site who hold or have held a reactor operator license.

We already have additional people for the remainder of the positions on site. They just do not hold a license on any of our other units.

MR. MARK: Okay. That's exactly the picture I wanted to get a feeling for.

MR. COTTLE: I have a little more information.

MR. MARK: You know the people. They haven't passed their graduation exams, but they are there.

MR. COTTLE: Yes, sir.

MR. MARK: That was all.

MR. MILLS: Thank you, sir.

Dave, did you want to proceed?

MR. ORMSBY: Thank you.

The Watts Bar and Sequoyah nuclear plants are similar in most respects. The cores are similar and each plant utilizes a free-standing steel vessel with an ice condenser and a reinforced concrete shield building. The plants' mechanical systems, containment systems, emergency core cooling systems, instrument and control systems, electrical power systems, radioactive waste systems, and steam and power conversion systems are also very similar in design and materials.

Most differences are either site specific or are the result of the fact that two to three years separate the design phases of Sequoyah and Watts Bar. Although the design philosophy was the same, there were some instances when more current technology was used for Watts Bar.

(Slide.)

This vugraph provides a table showing design differences between the two plants. As I stated previously, there is a two to three-year difference in the design phases for Watts Bar and Sequoyah. During that time, some design changes were made for Watts Bar which increased the efficiency of the system. There are also some instances where equipment was upgraded or provided by a different manufacturer.

A couple of examples are the following. You will notice that there is a difference in the increased primary system flow rate and therefore a difference in the maximum heat flux. That is primarily due to the fact that, although the reactor coolant pumps themselves are the same for the two plants, Sequoyah utilizes a 6,000 horsepower pump motor and Watts Bar utilizes a 7,000 horsepower pump motor.

Because of the increase in the reactor coolant system flow, there is a need for greater PORV relieving capacity for load mismatch and to accommodate a 50 percent load rejection. The PORV's for Watts Bar are provided by a different manufacturer than those for Sequoyah.

You will also note that there is an increase in turbine generator and gross electrical output. These differences are due to increased equipment and system efficiency and also to the difference in steam generators.

As you can see from the slide, we have a model D steam generator for Watts Bar and a model 51 for Sequoyah. The steam generators for the two plants are similar vertical shell and U-tube evaporators, with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering, then leaving through the nozzles located in the hemispherical bottom head of the steam generators. Steam is generated in the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

Now, the primary difference between these, aside from minor differences in surface areas and U-tube design, are the fact that the Model 51 for Sequoyah utilizes a feed ring, whereas the Model D for Watts Bar utilizes a preheater section. The differences in the two steam generators show that we have an increased secondary steam flow rate.

MR. MARK: You mentioned that you had pumps from a different manufacturer. They were bigger, I believe.

MR. ORMSBY: I don't think from a different manufacturer. The pump motors are different in size.

MR. BEAL: The PORV's are different manufacturers.

MR. ORMSBY: Yes.

MR. MARK: Is this a better manufacturer?

MR. ORMSBY: I hesitate to say a better manufacturer. I would say that there is more qualification documentation in the PORV's for the Watts

Bar valves than there are currently for the Sequoyah valves.

MR. MARK: So you would have the expectation that these things are at least as good, maybe better?

MR. ORMSBY: Yes.

Are there any other questions?

MR. ETHERINGTON: The reactors are identical, are they, the core and so on?

MR. ORMSBY: Yes, the cores are essentially identical.

MR. EBERSOLE: Is that the end of your presentation?

MR. ORMSBY: Yes, sir.

MR. MILLS: Mr. Ebersole, would you like for us to hit the items you mentioned before about the Watts Bar hydro?

MR. EBERSOLE: Yes, please, if we have someone here to do that.

MR. GRAVES: My name is Ron Graves, from the TVA.

I would like to address the question of interdependency between the Watts Bar hydro plant and the Watts Bar nuclear plant. The Watts Bar hydro plant feeds power into the 161 KV grid which is the source of two circuits that supply offsite power to the shutdown boards inside the plant. The shutdown boards inside the plant have each a diesel generator as an additional source of power to the shutdown boards.

The 161 KV grid is also interconnected with the entire TVA grid, and I believe that we have two other lines going to the grid in one of the buses in that switchyard, and we have three interconnections with the grid on the other bus.

The final conclusion is that Watts Bar hydro is not essential for the reliable operation of the shutdown system.

MR. MARK: Is there any arrangement whereby the Watts Bar plant would get special attention in case the grid had trouble?

MR. COTTLE: I'll answer that from up here. We do have some postulated failures. We have an Athens-Sequoyah line going out of Watts Bar that is one of our primary feeders. In the event that that line is down, laid on the ground from weather considerations, we do have or are in the process of implementing some procedures where we do get special preference from the hydro plant in terms of splitting the bus, dedicating that basically to the Watts Bar unit plant. And that is to prevent a fall in the hydro switchyard from further degrading loss of offsite power.

MR. ETHERINGTON: Are there periods of low flow when the hydro plant is closed down?

MR. COTTLE: There are some periods throughout the year where we have, mainly during the nighttime hours, essentially zero flow. That is not particular significant to us at the nuclear plant unless, as I said before, we are in a special case where we do have the Athens-Sequoyah line out.

MR. MARK: My point in raising the question here was that I thought that you were particularly favorably located to a possible source of power, or the lines were short.

MR. COTTLE: There are certainly some other considerations, if you have a complete grid blackout, about restoration of power and starting up the hydroplant units first, allowing us to regain a source of power, if you lost that whole TVA grid -- and we do have provision for that, as well.

MR. EBERSOLE: Do the Watts Bar cover generators spend a large part of their time with condensers ready to take water, in the event you need them?

MR. CARBON: Not a large percentage of the time. There are some periods of time when the running is synchronous.

MR. EBERSOLE: Is the old fossile plant in operation?

MR. CARBON: It is in condition for a standby mode of operation. It is typically not run except during --

MR. EBERSOLE: For the general information of the committee, I would ask TVA about the rather ancient practice of deriving auxiliary power from the output of the particular unit, meeting that power. And I came across the relevation that maybe Watts Bar is a type of design that would use that.

They consider the 161 Kv system here is not a particularly better system; not as good as the 500 Kv. Therefore, it is not an optimum system from which to continuously draw power for the operation of the station auxiliary. So they therefore use unit outputs to run the station auxiliary.

In that configuration, they always lose station output when the unit gets in trouble, and it is mandatory that they accomplish an electrical transfer. They do this by, I think, instantaneously transferring to the 161 Kv as an alternate source, inspite of its ordinarily lower stability. And they do other switching to get the other non-1E loads across to a source of power.

I understand that at Belefonte it will be having a steady source of power from other than the other unit itself to run the station auxiliary. The TVA should correct me if I am wrong in that statement. Was that correct?

MR. GRAVES: I believe you are correct, Mr. Ebersole.

MR. EBERSOLE: Any other questions on this topic?

(No response.-

Let's go to the outstanding issues we have. We have Mr. Kenyon down for this.

(Slide.)

MR. KENYON: I am here to present the licensing status of the Watts Bar plant. The SER was issued in June of this year, and the review was unique in that it relied more heavily on the review performed on the Sequoyah plant, because as we have been discussing, the designs were so similar.

The SER contained 17 open items, 41 confirmatory items, and 37 license conditions. Of the 37 license conditions, 21 were TMI related, and can be characterized in a number of different ways. Some of the items were imposed to insure completion of work such as physical plant modifications or to ensure the completion of long-term generic analysis programs that were going on.

Other TMI-related license conditions have been imposed just to insure the submittal of information in accordance with the schedule consistent with NUREG-0737. The non-TMI related open items consisted of items which the staff had reviewed and discussed with TVA and determined that the regulations could be best met only if -- through the imposition of the license conditions.

In addition, some of these non-TMI related license conditions have been imposed to insure that plant modifications would be completed in accordance with the specified schedule.

MR. MARK: In view of the rather large similarity between this plant and Sequoyah, the fact that you are dealing with the same management, are these open items that you mentioned also open at Sequoyah, or have they been closed there but are still not closed here, or are they different items?

MR. KENYON: Sir, it is a mixture of all, actually. Some of these --

(Slide.)

This is a list of the open items remaining since the SER was issued. Some of them are -- were open on Sequoyah and have been either closed or perhaps are still under review. Some of them are very site specific to Watts Bar, and the problem just did not come up on Sequoyah, such as the potential for liquefaction beneath the ERCW pipelies. Some of them are simply due to the normal evolving process of the licensing process.

As you know, it has been about two years since the Sequoyah Unit 1 was licensed. New requirements sometimes come up in addition, or more likely, they are new interpretations of requirements that become policy. In a lot of these cases, sometimes the open issue arose because there was a different reviewer looking at Watts Bar than looked at Sequoyah, and he put a different emphasis in his review in a different area.

Of these open issues remaining, they are primarily just issues where we are waiting for TVA to submit information. In fact, TVA has submitted information on six or seven of these items already, and the staff is just reviewing it, so it is awaiting their review.

MR. MARK: Fine.

MR. KERR: In addition, I guess, one of the plant-specific problems is for the Model D-3 steam generator. As I was going to say, back in December, the staff had issued a draft SER that contained 80 or 90 open items. Probably more appropriately, I should say concerns. They were raised simply because the staff lacked information about the Watts Bar plant.

TVA submitted the information in response to our questions; we reviewed it and closed the item.

MR. MARK: I wasn't meaning to delay you.

MR. KERR: Are there any questions about the open issues?

MR. EBERSOLE: Have there been other designs like this one that used a steel piling retaining wall with an earth fill to act as a causeway, and simultaneously embed the ERCW pipelines in 1E component work? This is a fairly long causeway, I believe. It is a steel piling causeway.

MR. KENYON: Yes.

MR. EBERSOLE: Have you had any designs like this before this particular one that you can recall?

MR. KENYON: Not that I am aware of. I don't believe we have anybody --

MR. EBERSOLE: Anyway, that is one of the differences in this design between Watts Bar and Sequoyah. They do carry these critical water lines and the corresponding electrical services in the earth-filled causeway surrounded by steel pilings, and are retained by deadmen supports to tension bars.

There will be some discussion of this later one, corrosion protection and the long-term viability of this type of construction.

MR. CARBON: I would like to ask one question similar to Dr. Mark's actually. Could you or Mr. Novak or someone make any kind of comment on has there been appreciably -- has it required appreciably less effort on the staff's part to review this plant, in view of all the similarities with the Sequoyah plant?

MR. KENYON: Well, sir, when the review started on Watts Bar, we requested TVA to present a comparison between Sequoyah and Watts Bar. In addition to that, in many instances we used the same reviewer who had worked on Sequoyah to look at the Watts Bar plant. The reviewer, where he could, went back and reviewed the comparison and --

MR. CARBON: I am sure he did that. My question is simply, do you have any feeling for how much less effort it may have taken to review this plant than the first one?

MR. KENYON: Quantitatively, I couldn't give you a good answer. The best response I could give you is I do know in many instances there were -- in many review branches they put less work into the review of Watts Bar simply because they knew what was done on Sequoyah and TVA said that their systems were identical or essentially identical, and that lessened the review work done on it.

MR. CARBON: Fine, thank you.

MR. EBERSOLE: Other questions?

(No response.)

If not, you can proceed into the next topic which is organization and management, and I believe Mr.

Cottle will give that presentation.

MR. NOVAK: I would like to add one comment. I would say overall, the staff review effort was comparable to Sequoyah. Realistically, if you look at how long an application resides in the licensing process there is a building factor. It is an integration. And both Sequoyah and Watts Bar have relatively long periods of time in construction, and in a sense, the licensing process was extended.

I think where there was no reason to go back and do it over again, that was the course that was followed. Just simply because of the difference in time, certain areas were looked at perhaps in more detail on Watts Bar than they were on Sequoyah. I think we continue to learn about plants from reviews. I don't think we will ever say that we won't learn anything about it on the next one.

MR. CARBON: But to a first approximation or a first best guess, it has taken about the same amount of effort, not time but effort, for both plants?

MR. NOVAK: That would be my judgment, yes.

MR. COTTLE: I am Bill Cottle, Plant Superintendent at Watts Bar. I would like to take these next few minutes to go through just a very brief summation of the experience requirements or the experience that key members of my staff have at the plant, and we will get back into the actual staffing numbers that we touched on a little earlier.

(Slide.)

The key members -- and there is a handout, I believe each one of you has. We tried to break it down to the individual, the position he holds and as I said, I am Plant Superintendent. I have two assistant plant superintendents; assistant superintendent for operations and engineering, and an assistant superintendent for maintenance. I am going to have a third individual that is almost on the same level as an assistant superintendent that is a field services supervisors who is responsible for refueling outages and major maintenance outages at the unit.

I would like to go through just briefly the background of those individuals and then focus on just those other key positions that are directly in the line of operation of the unit.

We have got it broken down to where we are only really talking about either general fossile plant experience or experience at a nuclear facility. I have about a total of 13 1/2 years experience in nuclear, between nuclear Navy background, previous experience with the Alabama Power Company and the startup of Farley

Unit 1, the experience at Sequoyah nuclear plant prior to coming to Watts Bar as assistant superintendent for operations and engineering, and intertwined in there two years experience with the I&E staff of the NRC as a project inspector and resident inspector.

Of the 13 years' experience at the plant, I guess if you broke it down between the two categories of maybe pre-operational phase and training, it would be about 7 years, and experience at an operating facility about 6 years.

MR. MARK: Where do you categorize your experience with I&E? It doesn't sound either like operating --

MR. COTTLE: Most of my experience with the I&E group in Atlanta was as resident inspector involving with startup of units. We classify that as operational experience.

(Slide.)

Assisant superintendent in charge of -- direct charge of the operations of the engineering group has some 10 years prior experience in the fossile field, some 20 years' experience in the nuclear field including experimental gas cold reactor project in Browns Ferry and Sequoyah projects. Most of this has been in a pre-operational and training capacity. I would say probably 18 to 19 years in that capacity, and about a year involved with the startup of Sequoyah unit, and Mr. Lewis does hold a senior reactor operator license at the present time.

(Slide.)

Assistant plant superintendent in charge of maintenance, Mr. Ed Ennis has about 13 years of fossile experience, five years of applicable nuclear experience all at Watts Bar and all basically in the pre-operational phase. He has recently completed what we would call the equivalent of a cold license certification course, meaning that training which we would require prior to our certifying him to take an NRC license exam on the Sequoyah facility.

(Slide.)

Field services supervisor, the individual who is primarily responsible for planning and scheduling and conducting our major maintenance and refueling outages has some 11 years of nuclear experience, including five years in the nuclear Navy propulsion program. A breakdown of his experience shows about six years pre-op and training phase, and five years at an operating facility.

(Slide.)

Operations supervisor, that individual

directly over the operations group, has some 15 years of fossile experience with TVA, about 14 years associated with the nuclear program at TVA including a significant amount of time as the senior reactor operator at the Browns Ferry facility. The breakdown of his nuclear experience would be about 10 years pre-operational testing phase, and four to five years in an operational capacity, primarily at Browns Ferry.

(Slide.)

We have two assistant operations supervisors. The first, Mr. Norman, has about seven years' experience in our fossil operations, ten years' experience in nuclear, and all of that ten years primarily in the preoperational and training capacity.

(Slide.)

The second assistant operations supervisor, Mr. Yarbrough, has some fifteen years of fossil experience, about ten years of nuclear experience. Mr. Yarbrough came to us as a shift engineer from Sequoyah and holds an SRO license on Sequoyah and was one of the original shift engineers for the startup of initial operation of the Sequoyah units.

In addition, in the last kind of individual background we'd like to look at we have a special projects supervisor or special projects coordinator that is assigned to assist the assistant plant supervisor of plant engineering, Mr. Eckard. Mr. Eckard has about 19 years of fossil plant experience, 11 years of nuclear plant experience.

(Slide.)

Mr. Eckard was previously the operations supervisor at Sequoyah. He holds a license on the Sequoyah unit and served as operations supervisor during the startup of the Sequoyah units.

(Slide.)

MR. CARBON: Mr. Cottle, I am favorably impressed by the experience that I see here of your people. Just to understand where they came from and so on, I guess they are all long-term TVA people except yourself, is that correct?

MR. COTTLE: Except myself almost all of the years' experience that we've talked about have been in the TVA organization, yes, sir.

MR. CARBON: Thank you.

MR. MARK: A mild question. A number of people that you displayed for us have had a part or maybe even a large part of their experience at Sequoyah or Browns Ferry. Do you have to steal those guys from Sequoyah, or does TVA assign them to Watts Bar? We are very familiar with plants which have been in the spot where they have had to steal or lure their people from somewhere else.

MR. COTTLE: For the most the experience that these individuals got at Sequoyah was while they were in fact assigned as Watts Bar individuals; but the assistant plant superintendent for operations and engineering was licensed on Sequoyah while he was assigned to Watts Bar.

MR. MARK: So it isn't so much a matter of assignment as --

MR. COTTLE: It's more of assignment and trying to accomplish a transfer of experience and get the benefit of the experience on the Sequoyah startup up to Watts Bar.

MR. MARK: That's just what I was trying to get a feeling for.

MR. COTTLE: Yes, sir.

(Slide.)

I'd like to look at just our shift engineer classification, and the shift engineer is synonymous, I guess, with the NUREG terms of shift supervisor: our lead individual on each shift. I won't go over all of the individual statistics. Let me point out a couple of what I feel like are important aspects of the slide.

Six out of nine of our shift engineers have either -- two of them have held a license on the Browns Ferry units; I guess five have held a license on the Sequoyah unit; four have held a license on the Sequoyah unit, and they currently hold that license and are staying current in the Sequoyah requalification program. An additional superintendent does not hold a license on any of our units. However, he has a significant Navy background.

The last individual listed on the slide is Mr. Jenkins. Mr. Jenkins has just come to us. He is working as a shift engineer in a rather limited capacity. We went out and got Mr. Jenkins because of his extensive experience in fossil plants and his extensive experience in the training of assistant unit operators. He is at Watts Bar with the primary purpose of working with our unit operators to get what I call a common sense powerhouse approach towards looking at the equipment.

We are looking at our training program. We are very comfortable with the basics of nuclear theory and the systems knowledge these individuals have. We have identified some areas of weakness in just plain checking out, running pieces of equipment, and diagnosing problems with pieces of equipment. That is what he will be primarily working on.

MR. MARK: Where do the people who we are looking at here and hundreds of others, where do they live? They don't live on the site. Do they live in Knoxville or Chattanooga?

MR. COTTLE: Most of them live in several smaller towns around the site -- Athens, Tennessee; Sweetwater, Dayton, Tennessee -- within a 20-mile radius of the site basically.

MR. MARK: Sweatwater is a fine crossroads on the highway.

(Laughter.)

MR. EBERSOLE: Mr. Cottle, I guess I don't know what you mean by shift engineers. I take it it's really an administrative title and not meant to imply --

MR. COTTLE: The shift engineer is an administrative position type.

MR. EBERSOLE: It doesn't necessarily mean you'll turn the shift over to an individual?

MR. COTTLE: No, sir. We don't in fact have our individual in the license training program.

MR. EBERSOLE: So that's really an administrative title.

MR. COTTLE: We're trying to upgrade our junior operations people.

MR. EBERSOLE: On the other hand, are the others functionally really shift engineers?

MR. COTTLE: The others are functionally shift engineers. The one exception to that would be Mr. Pauley. We have a shift engineer position that we dedicate to operator license training, and he's qualified for that, and he has considerable experience at the Sequoyah plant. His job function primarily is the overall direction and preparation of the license program, that part that we do on site.

MR. EBERSOLE: Mr. Cottle, who is your current counterpart at Sequoyah down the road a little bit?

MR. COTTLE: The superintendent at Sequoyah is Chuck Mason.

MR. EBERSOLE: I see. Thank you.

(Slide.)

MR. COTTLE: The next senior position on our operating shifts is assistant shift engineer. The assistant shift engineer is that individual who would hold a senior reactor operator license and would have accountability for the operation of an assigned unit, either Unit 1 or Unit 2.

We currently have 11. Out of those 11 we have 2 that hold a current reactor operator license on Sequoyah. We will be upgrading them to senior reactor operator. We have one individual who previously held a reactor operator license on a Browns Ferry unit. All of the remaining individuals either have in the very recent past or we are in the process of right now having those individuals participate in the activities down at Sequoyah, and a good number of them will be participating. They have been reviewing the outage on Sequoyah Unit 1.

(Slide.)

So the last classification we will take a look at is our unit operator classification. We currently have 17 individuals, 2 of which hold a current reactor operator license on Sequoyah and are continuing to maintain the validity of that license.

Again, pretty much in the same light, all of our unit operator -- and there will be candidates for reactor operator licensing -- either have participated or will be participating in the activities at our Sequoyah units.

(Slide.)

MR. MARK: Can you make just a comment about your maintenance people? Have they also been drawn from within the organization, knowing how to fix up a motor and that sort of thing?

MR. COTTLE: Most of our maintenance people do have either previous fossil experience, and in some cases significant amounts of it, with TVA in the practical maintenance application and/or a combination of previous nuclear experience either at Browns Ferry or Sequoyah.

We do have some of our key maintenance people who have not spent a significant amount of time in an operating nuclear unit, and those individuals, we are rotating those down to Sequoyah now and have been for the last several months, and they are actually going down and serving in their counterpart positions at Sequoyah either as group supervisor or to cover a special project at Sequoyah.

MR. EBERSOLE: Mr. Cottle, I'd like to ask you to give us your views. You're an ex-Navy man. I think a lot of your experience was on submarines. These plants in one view are sort of compromises between having lots of people and lots of automation. My own view of the Navy is they would have 1,000 men down at Watts Bar with a speaking tube to every valve and a man standing by it.

(Laughter.)

I hear lots of comments from Navy people. We've heard it from high quarters that we're too highly automated and we need more and smarter people to run these plants.

What's your own perspective view on the amount of manning versus the amount of automation?

MR. COTTLE: I guess I might agree with the Navy that we're too highly automated. If we're willing to build plants with a factor of ten less output and drop down to a hundred megawatt plan, we would have very little people and very little automation. You don't accomplish a lot of economy of scale doing that.

On the primary plant I guess I would say I feel very comfortable with the degree of automation and limited amount of manning we have to place on the primary plant. I do feel at times, primarily in the supporting and secondary systems, that we lean a little too much towards our automation phase, and at times we have extra people to compensate for that. It may be because of design problems, it may be because of lack of operations, of individuals' confidence in the design.

MR. EBERSOLE: Some long time ago we were looking at the single failure criterion in the context of its value for preventing spurious trips, not just for a safety function but for preserving continuity of operation.

In reviewing the peripheral safety question that it's better to have a safely running plant than to throw it into shutdown, we didn't get very far with the operating people to invoke coincidence requirements for tripping functions. As a result, many of the plants have places where you can find touch points. I'm sure yours is no exception. You can go up and hit it with a broom, and you can trip it out.

Do you ever look over the plant for such operational weaknesses in this context and maybe provoke a few arguments here and there sometimes that you'd rather not have those weak points?

MR. COTTLE: Yes, sir, we do. As you may be aware, we have made some significant changes on secondary plant originated trips at the Sequoyah unit. We're following those up at the Watts Bar unit with the same type of changes.

We have identified, for example, a single control fuse on the full flow condensate polishing unit at Sequoyah where blowing a single fuse gives you a trip. I proved that twice. We had two unit trips. We don't have the same condition today, and we are continually looking for that, both from the standpoint of challenging the safety systems and from the standpoint of good engineering practice and economics.

MR. EBERSOLE: What kind of staff do you have that are running this sort of thing down?

MR. COTTLE: It would be a combination of the engineering positions of the station, primarily those assigned both to the engineering results group and that engineering support that's assigned to the maintenance group.

MR. EBERSOLE: Do you pump any out of Knoxville?

MR. COTTLE: Yes.

MR. EBERSOLE: Do you have them looking at that?

MR. COTTLE: In Knoxville we don't have a formal program, and maybe someone else can maybe give me some help on this. We don't have a formal program. There have been several instances when I was primarily involved at Sequoyah that there was a motivating factor and those types of items came out as a result of engineering results.

MR. EBERSOLE: Have you done anything with the turbine generator, the point at which the safety and nonsafety systems merge, a trip results in a multi-channel trip by necessity, so what you really have is contact multiplication at the head; and a hydrometer trip or something like that in a single configuration can easily spuriously cut you off.

MR. COTTLE: Most of the improvements we've made along these lines have not been directly associated with the turbine. They've been more associated at this point in time with like inputs into the turbine trips, seal water injection function on main feedwater pumps, loss of both main feedwater pumps which gives the turbine trip. We're still one stage below basically looking at that.

MR. EBERSOLE: I see.

MR. SHEWMON: Does that bring your discussion to an end?

MR. COTTLE: Yes, sir, it does.

MR. MOELLER: Could I ask a question? You have a person who is chief of industrial and radiological hygiene branch that handles radiation protection. Can you tell me his or her qualifications?

MR. COTTLE: No, sir, I can't right at the moment. I can certainly get it for you. We were in the process of accomplishing a reorganization in the area of radiological hygiene. Previous to this time most of the responsibility and direction of our in plant health physics dosimetry/dose assessment program had been handled out of Mussel Shoals, out of our department of radiology. We transfer most of our functions into Chattanooga now.

MR. MOELLER: Could you tell me the qualifications of the top radiation protection program at the plant site?

MR. COTTLE: I can give you a basic outline of our plant health physicist's background. I don't have the details with me right now.

MR. SHEWMON: Would you get it before the day is over?

MR. COTTLE: Yes, sir.

MR. SHEWMON: Thank you.

Any other questions?

MR. EBERSOLE: Thank you, Mr. Cottle.

Our next topic before our break is TVA nuclear safety review staff function by Mr. Harrison Culver.

MR. CULVER: I am Harrison Culver, the director of the nuclear safety review staff.

Following TMI, TVA established a task force to identify actions that could be taken to further strengthen and improve their overall nuclear programs. As an outgrowth of that study, a report was prepared, and a number of actions were identified that related to design changes in the operating organization, changes in the design organization. And the one action that I want to talk about is the action to create the TVA nuclear safety review staff.

What I want to do today is to give you some idea about what our group is all about, the things that we have been doing, and the direction we intend to go in in the future.

(Slide.)

I believe you all got a handout, so please don't fail to look at your handouts.

The basic functions of our staff is shown on this first vu-graph. In this vu-graph you can see that the staff that I direct does have an involvement with all aspects of the TVA nuclear program. We do get involved to some extent at looking at design, construction, operations, training, emergency preparedness programs and the health physics program. We get involved with some of the operational events at the site, we get involved with making investigations, and we also get involved with a program where we look at employee concerns.

Now, as I get into my descriptions, I will be able to show how we have managed to touch upon a number of these items.

(Slide.)

As a point of clarification, I would like to make sure that it's understood that our group does not replace the functions that exist within the line organizations. From time to time people like to think that we get involved with all the details at TVA, and obviously when you see the number of people I've got, it's not possible to get involved with all the details.

So the key thing, our function is really to monitor the overall activities at TVA, not to replace the things that are assigned to the line organizations. In some respects we are similar to the functions of the I&E people or the NRR people. They look at certain aspects of the design, they look at certain aspects of the operation, and most of this is done on a sampling program or on an audit basis.

Now, in order to do these things, when I joined the staff we had built up our staff. I believe when I joined the group in January of 1980 we had about seven technical people. During the past two, two and a half years we have increased our staff to the point where now we have roughly 21 technical people.

Now, in building up the staff the two objectives I had were to try to get as much independence in our group as I could so that we could in fact go out and look at the operations of TVA and make impartial viewpoints on these things. At the same time, we attempted to create a staff that was pretty much senior level, people with a wide variety of backgrounds so that we could in fact get involved with these issues.

As you can see from the vu-graph, I think we were fairly successful. More than half of our staff has come from the outside of TVA. We have had people come from DOE and NRC. Primarily these people were involved with either the safety review or audit activities. We have had people come from other utilities where the intent was to bring in different perspectives from how other people do things. At the same time, we attempted to maintain some people from the TVA system, since we felt like it was important that we have people who understood the system within TVA, and we tried to get people out of the major parts of the system.

As you can see in that upper righthand corner, we have had some people from design, some people from the construction organization, some from the nuke power division, and one individual from the public safety.

Now I have given you some indication of our experience by indicating the distribution of our people. It comes out to be, we have an overall average of about 15 years of nuclear experience in our staff.

In that bottom set of data I indicate where that experience primarily lies. Mainly because of the emphasis that we have placed upon the audit and safety review function, we do have people who are more oriented towards the regulatory audit background as well as from the operations aspect. We have enough people to look at design and construction, and in those areas we have a smaller amount of background.

(Slide.)

I would like to amplify a little bit now about what we do. We make basically two types of reviews. We make either what we call management reviews, which are rather broad types of reviews, and we also make rather specialized types of reviews which are on perhaps one specific area, and these are much more detailed.

In the past two and a half years this will give you some indication of the types of things we've gotten involved with. It also indicates the types of subjects. In the first group I have indicated our management reviews. I will say some more about these in my next vu-graphs.

In our management reviews we have basically gone in and looked at the entire part of the organization to see how that organization is set up to function. In the first grouping I've indicated that we have in fact in the past two and a half years looked at the office of power in one of these reviews. We made two reviews in the office of engineering design construction where we looked at both the design process and the construction process. As a part of that we also had a review of the purchasing part of the organization so that we could in fact look at both the controls that were established by TVA in design, as well as their interface with purchasing.

We had two specialized types of management reviews, one that dealt in security and the other that dealt with the quality assurance program. When we reviewed the office of power, because of the integral tieup between power and the health physics function, we also as a part of that review made an examination of the office of health and safety.

Now, in the next group where I talk about specialized types of reviews, these were mostly reviews that we did in the early part of 1980 before our staff had attained the level of size that we have today. Most of these were rather limited types of reviews in very specialized areas.

From that you can see that we made more reviews of operations than we did of design and construction. The primary reason we did that is that at that point in time most of our experience was in that area.

There was also a time period where we wanted to look at what was going on at Browns Ferry and Sequoyah.

MR. EBERSOLE: Before you take that down, there are some interesting characteristics about the design construction bullet lists there. I look at it and of the sets you have I see one that I guess I can associate, rather materialistic looks at physical features of the plant.

As you well know -- I know your past history, and I see evidence there of why that's there and others are not.

MR. CULVER: Part of that is because of our philosophy of what we're really trying to do.

MR. EBERSOLE: Would you, as a case in point -- I remember some issues in the earlier plants where we were having difficulty ascertaining that the purge valves for the containment probably wouldn't work and that had to be investigated. The main steam isolation valves also probably would not work without certain stroke effects.

Do you get involved in assessment of the detailed mechanical aspects of such matters as that?

MR. CULVER: We have only to a limited degree up to this point.

MR. EBERSOLE: Do you anticipate extending your scope up into these matters?

MR. CULVER: I'll get to that a little bit later.

MR. EBERSOLE: Okay.

MR. CULVER: When I get through with telling you about what we do now, I'll get into that a little bit more.

MR. CARBON: Mr. Culver, before you leave that, unless you are going to discuss it later, a question about your review of the division of purchasing. Was that heavily directed at such things as being sure that the procedures that purchasing followed would ensure that you were getting the equipment to the right specifications, that nothing was falling between the cracks, or what was the real aim there?

MR. CULVER: The basic concern we had or what we were really looking for is that you come out of design with a certain design intent. What we were really looking for were controls that were imposed on the whole process to make sure that you actually ended up going to the vendors with the right information, and then when things came back that you got what you had ordered.

That whole thing, well, our whole review of the design process, the construction process is really directed towards examining the programs to see if in fact a program exists and that it is in fact followed so that you will end up getting what you hoped to get.

Now, to get into it a little bit, as I show you the vu-graphs I'll show you that.

MR. CARBON: My second question, have you developed this review system strictly within TVA, or has it come about in considerable part working with other organizations or INPO or someone, or is it pretty much totally independent?

MR. CULVER: I guess a fair way to say it is when I initially got started in this I, of course, had my own ideas from my own background where I had been involved with the review business for some years back in AEC and DOE. At the same time, when I arrived on my new job I felt fortunate to have some people who had come from NRC who had already begun to do some of these things. So I relied very heavily on both some of the information that they had obtained while working within NRC, and I had one individual in particular who had been working on setting up a system of what they called their PAT, P-A.T. Basically, that was a program where they were going to go in and look at performance evaluations.

I can show you -- in fact, my next vu-graph, I think, shows you what we came up with.

(Slide.)

This particular slide here demonstrates the process that we go through. It really had its beginning in the NRC. Of course, I am a firm believer if someone has developed that looks pretty good, you might as well use it.

This really came out of some of the work that was being done in I guess it was I&E in Washington. One of the gentleman that was on my staff, Morris Englewood, worked on this when he was up here in Washington. We basically in principle use this type of a management evaluation tree when we go in and look at, for example, the office of power or the office of engineering design and construction.

We basically look at, first of all, do they have a program in place to accomplish the functions that we are talking about, and there are a specific number of things we look at. We look both at that program all the way from the top level, the corporate level all the way down to the site. That is the left branch of the tree.

Now, when we look at programs we're really looking at the adequacy of the program as measured against regulatory requirements, commitments you may have in the safety analysis which may eventually get you back to the ASME code standards and what not.

We look to see if the program is fair, if the program is adequate. We look to see, beyond that, was it being implemented. These other two merely indicate that as you are doing this, you really like to make sure that the people who have to conduct the program are in fact aware of the program. As a part of that, you have to get involved with looking at qualification and training of people.

We have basically followed that system in all of our major reviews. In doing that we basically look at each of these functional areas.

(Slide.)

That is a vu-graph I put together when we talked about OECD. The functional areas are basically the same when we looked at operations. When we go out and look at these reviews, we will in fact look at each of these functional areas. We have a review report that addresses each of these areas. By the nature of that these reviews are very time consuming, and we spend a lot of time on those reviews.

Before we go out on the review, for example, we may spend up to -- we spend a lot of time just on the initial review in just pointing out the review and figuring out exactly what we want to look at in each of these areas.

Before we go out and review we in fact have in our group all the procedures, administrative controls, and our people will actually go through those so that we're very familiar with them before we go out to make the review.

We do that for two reasons. One is that you really have a background to make a meaningful review, and you need that to minimize the impact on the people you are reviewing.

So we do in fact do that, and then we will issue a report on it.

MR. EBERSOLE: Can I ask a question? In the time phasing of your activity you had an active role in trying to unsnarl this terrible matter of the QA program. Could you comment on your activities in that period of time when you straightened out the administrative disabilities present at that time?

We will have that as a separate topic, but I thought while you were up here you could make a comment on it.

MR. CULVER: We did get involved in looking at QA. I guess that came about about a year and a half ago. It came about after, I believe it was after one of the SALP reports came into TVA, and the general manager asked us to specifically look at TVA.

The truth of the matter is, and I pointed it out to him, in a sense we had been looking at QA as a part of looking at the overall organization, because when we go in and look at these functional areas, we not only look to see what the line organization is doing, we also look to see what the QA organization is doing.

So we had a head start on that, but we then issued a report which specifically dealt with QA. It dealt with things beyond these functional areas here. We drew very heavily upon our reviews that we had already made, but in our QA report we looked at such things as how the QA organization was in fact organized: the areas we had addressed, the degree of fragmentation, the fact that the individual QA units were in different places in TVA.

MR. SHEWMON: You've about taken up your twenty minutes. Could you get on with it?

(Slide.)

MR. CULVER: Let me just show one more slide. In the other session people were interested in well, what are we going to be doing in the future. What I've shown here, I'll just quickly indicate that as we get more and more out of the types of reviews we have been making, we will be getting more involved with these types of activities which are more directed toward more fundamental issues.

This happened to be a slide I pulled together really from my budget, and this is the basic breakdown on our activities.

MR. ETHERINGTON: All of your people are located at TVA headquarters?

MR. CULVER: All of my people are in Knoxville, yes.

In my handout I indicated how we handle our reports and our dealings with the general manager and the board, which is, I think, an important part of our program.

MR. EBERSOLE: Questions for Mr. Culver?

(No response.)

MR. EBERSOLE: Ten minutes we have scheduled. We will have a break, Mr. Chairman. A pretty short recess.

(Recess.)

MR. SHEWMON: Jeff, I am told that the TVA people have the information on the qualifications of the health physicists, so we can take that first.

MR. EBERSOLE: If you don't mind.

MR. SHEWMON: Can we please quiet things down.

Go ahead.

MR. COTTLE: The plant superintendent, Mr. Ralph Beck, is my health physics supervisor at the station. If I can just briefly cover Mr. Beck's background. He has four years experience in the nuclear navy program as an engineering laboratory technician in the navy chemistry health physics aspect; two years experience in the navy shipyard as a shift supervisor, refueling and overhaul, outages; seven years exerience in the naval shipyard as a lead technician on callibration and dosimetry aspects of the health physics program; and then six years experience with William and Mary College as a health physics technician and a lead health physicists for their space radiation effects laboratory that they were conducting under the direction of the NASA program.

He has been with TVA for three years, primarily at the Watts Bar nuclear plant. And he is currently participating in the activities down at

Sequoyah and will participate during the portion of the refueling outages.

MR. MOELLER: Thank you. That's good.

MR. SHEWMON: No question on level of certification?

MR. MOELLER: I'm sure he's working on it.

MR. EBERSOLE: You might notice by the length of time, at least I regard this as a major topic, the refueling problem. Certainly I am apprehensive to see this enormous flow of deficiency reports and recognize it's a continuing problem and will be throughout the remainder of the system.

To me it's the most significant topic we have on the board.

MR. BEASLEY: My name is Greg Beasley. I am manager of quality assurance for TVA's Office of Design and Construction.

Frequently throughout the presentation, I may refer to OEDC. That's the acronym for TVA's Office of Engineering Design and Construction.

I would first like to talk about the identified problems in quality assurance. Prior to 1980 we did not identify any significant or unusual quality assurance problems in the design and construction program at Watts Bar. We did have the usual number nonconformances, the usual noncompliances that were identified there.

During the 1980 time frame, our QA problems came to light. 1981, 1982, there were several major problems related to quality that arose. One of the first that arose concerned our heating and ventilating systems.

In January 1980, a routine quality assurance audit found that some aspects of the safety-related heating, ventilating and air conditioning systems were being installed without having a QA program over the installation. There was corrective action, which extended over a period of approximately six months.

Work was then resumed with a later identification for more corrective action, and after a period of about a year there was a confirmation of action letter from Region II, and subsequent to that we had a stop work order on this order which lasted for several months. And after about 18 months after the identification, we got the program back on track and work was totally resumed on that and had the corrective action implementation which has continued up until this time.

On the identification of problems, one of the second problems was a series of welding and weld inspection problems that arose. This problem was not unique to the Watts Bar site. There was a task force that addressed these generically for all TVA plants under construction.

There were some problems that were identified relative to the transfer process where we transfer systems and equipment from the construction organization to the operating organization. And in one of these processes while a system was being transferred, we were using some of the safety-related equipment to provide flushing water, and in the course of this one of the safety-related pumps was destroyed.

We did not have adequate control over that situation in the transfer process and the use of this. So there was corrective action in that area.

We also found we had a fairly large number of problems with our quality records. We did not have them all together. We did not have the traceability to trace some of the inspections through from front to back. We still have corrective action going on in that particular area.

Now, these are four major things which we identified which concerned us and which have gotten very high visibility relative to our quality assurance program. There is another factor which came into this, although not directly. In the 1981-1982 time frame, we had extremely low morale at our construction project. We had some major management changes there and we had a significant manpower turnover. While this is not directly related to quality, it does affect it. It does affect traceability of records. It does address application and use procedures.

Now, all these things put together indicates that we just generally have a problem of management control with our quality. It is a serious problem. We have treated it very seriously. It has gotten high visibility, which we think is proper.

But we feel at this time we have identified our own problems and we have our corrective action moving, although it has been slow and difficult to get it moving.

I would just now like to address the steps we have taken to correct this problem. As the usual case, each identified deficiency or inadequacy has had its individual corrective action, and appropriate previous work has been reviewed to be sure that the previous work was adequate.

We have had numerous changes in our quality assurance procedures. Some of the procedures were expanded and broken down into a series of procedures.

Some of the procedures were made more definitive with the new requirements put in them. Some procedures have additional acceptance criteria.

Now, all of this work in the quality area -- the procedure changes, the looking at the deficiencies, and the looking back at the previous work -- has resulted in a lot of review, a lot of manpower, and much more documentation.

In some cases we have actually zeroed our records. That is, we just started over: total reinspection, total new inspection records on pipe location, pipe hangers. We actually went through this process and went back to the plant and relocated dimensionally all of the pipes, so that we had it right to start with.

Now, the rework and the reinspection has resulted in a very large number of nonconformances. Many of these we did determine to be significant on an individual basis. We feel like we have a good program of identifying nonconformances and we have a good low threshold so that the problems do get identified.

On the current status of the quality program, we in the OEDC management feel like we have our quality assurance program back on track. We have some corrective action which we are still in the process of defining. We have even more corrective action where the implementation is not complete. In some cases we are reviewing tens of thousands of records, and this takes a long time to go through and take and see that we have each record and it was properly signed off, and this is being done through an accountability program.

On the current status, one item was disturbing to us. Just recently, just within the past few weeks, we discovered a breakdown in some of our seismic analysis work. The corrective action for that is in the process of being determined and being implemented. This deficiency was defined on nonconformance reports by our line organization and it was duly reported to Region II.

In addition to the work on each individual deficiency, in looking back at our program about a year ago we became concerned that we may not be getting back to our root causes. So we instituted a study to determine what our root causes were for our quality program. This was for all of OEDC, not just for the Watts Bar plant.

We went through and identified some very basic root causes and set about a program to correct those. Just for information, some of the root causes concerned attitude and approach towards our quality program. We have had corrective action in this area. We have given the quality program much more attention and visibility by our top management, and we feel we have seen a turn-around in this attitude and approach to quality.

We found that in some cases a lack of definition of authority and responsibility had affected our program, and not holding people accountable for their responsibilities was affecting it. Timeliness was a big factor in the response to this corrective action. We think this is a major thing. We think this is a very difficult problem to deal with, but we felt that more timely corrective action was important.

As I mentioned, in our procedures we found that 50 percent of all our deficiencies had their root cause in procedures, either the people not following the procedure or the procedure was not adequately interpreted, or we didn't have a procedure that correctly covered this.

So we have had a massive program going on on procedures. All of these root causes are being addressed across the office program. And while some of them we will not complete the action until the end of the year, we feel we have seen some changes in that. And of course, some of these are going to result in further change which will extend out into 1983 or even further.

One of the major concerns when you have a quality assurance problem such as we have had at Watts Bar is, have you found all the things that went wrong, all the things that were not done correctly, all of the deficiencies? This is of paramount concern in OEDC. I think this is one of the concerns of our top management, one of the concerns of Region II.

We in OEDC feel like we have basically accomplished going out and ferreting out our problems. There will be more NCR's as we go through the program, but we don't feel that we have those in the program like occurred in the 1982-83 time frame.

Now, as Mr. Culver mentioned, one of the things that they had done is made two management reviews of OEDC, one on the Bellefont project and one on the Watts Bar project. The one on Watts Bar was recently completed. They have a number of findings against OEDC, some of them against the design program, some of them against the construction program, and some of them against the quality program.

We are addressing each one of these findings and forwarding that back to the nuclear safety review staff and we will resolve these internally. We are in the process of getting the NSRS report and there are responses we're making back to the findings they've had forwarded to Region II and to NRR.

Now, as I said the NSRS review of Watts Bar had a number of findings. We are instituting corrective action for those. They are continuing concerns. We in OEDC did not pick up any major new quality issues there and we did not conclude that we had not covered -- failed to find all the holes in our program.

Now, as part of this review of the Watts Bar program, NSRS did make a recommendation for a further independent review, a further independent review being for the purpose of assuring TVA management that Watts Bar has been designed and constructed in accordance with the requirements.

Now, in some meetings with Region II, specifically a meeting at the site back on February the 18th, a meeting with NRR where Region II was present, and in a routine inspection in Knoxville in late June, in all three of these meetings Region II recommended further independent verification that the plant had been constructed in accordance with design.

Now, we in OEDC are in the process of arranging for this independent review. It will be done by an independent organization, that is an organization outside TVA, an organization that does not have a major dependence upon TVA for their resources. The objective of this independent review will be to enable TVA, TVA top management, to reach a conclusion that Watts Bar has been designed and constructed in accordance with the license application and the license commitments.

The review by this independent organization will be available to NRC -- their findings, their work -- without any sanitization by TVA. The review will be, as Mr. Ebersole mentioned, a vertical slice. It will be on some safety-related feature and it will be broad enough to cover most of the disciplines that go into the design of the system -- the mechanical, the fluid discipline, the electrical and control discipline, the structural, seismic. And it will have some interfaces with the other programs, like our NSSS.

This independent reviewer will basically look at the development of the design details, will look at the calculations and analyses that are necessary to support these design details, and in the development of the information that is given to the Division of Construction to construct the plant. They will also look at the construction up to the point of the transfer of the systems and equipment to the operations organization.

The independent review will recognize the vintage of some of the requirements. It was mentioned earlier that our construction permit dates back to early 1973. We've been at it for about 9-1/2 years. So some of the requirements are vintage requirements and the review will recognize that, that we will not be reviewing against the current state of the art in regulation.

Now, from this review we anticipate that our TVA corporate management will be able to look at that, to look at the NSRS reviews that have been made of our program, plus the internal reviews that we have had, to reach satisfactory conclusions relative to Watts Bar.

I would like to briefly compare the quality program and the quality situation at Watts Bar with that of our other TVA plants. Except for the cases that I mentioned earlier and some related cases, we find no reason to think that Watts Bar is significantly different from our other nuclear plants. Specifically, the heating and vent problem was unique to Watts Bar.

The morale factor -- well, morale was difficult to judge and be objective about, and we can't quantify it, but we feel that the morale problem was basically unique to Watts Bar and much of our management control was there.

Some of the root causes, as I mentioned, were program-wide, but overall we didn't find Watts Bar being that different. Also, we did a detailed comparison of our NSRS report on Belefont to our comparison of the report on Watts Bar. Of course, there were some things that had happened in the time period between those. And we didn't find anything in the NSRS report that made Watts Bar significantly different from our Belefont plant.

We have had a very large number of nonconformance reports. As I mentioned, we feel we have a low threshold for this. And in rectifying some of these problems that we have had, we have issued a large number of NCR's. With respect to one bulletin, there was something in the range of 50 or 60. I believe there were over 70 NCR's written relative to the heating, ventilating and air conditioning problem getting it back on track.

MR. KERR: What do you mean by having a low threshold for them?

MR. BEASLEY: A situation comes up and you can write an NCR for this and make it significant, and it gets reviewed by management and gets reportable. We find that we report things, that we have a low threshold for making items significant and making them reportable. This is part of our TVA policy.

MR. KERR: I heard the use of the term "low threshold." I don't know what it means. How do you judge a low threshold?

MR. BEASLEY: Well, I think by just taking and comparing what we report from what we see reported on other plants, other TVA plants and other organizations'.

MR. SIESS: That doesn't prove anything. You might just be doing a poorer job than anybody else.

MR. BEASLEY: Well, the point is that we don't feel that comparing numbers of NCR's is a valid basis for comparison at the plant.

MR. EBERSOLE: I take it you mean you report items that might be considered marginal?

MR. BEASLEY: That's correct. If it's marginal we compare it.

MR. SIESS: Marginal by whom, and in comparison to what? Could you give us an example of an NCR that you think probably wouldn't be reported by someone else?

MR. BEASLEY: I don't have any specific examples in mind. I can probably think of one. If you look at the bases for reportability, the use of the word "significant," it is used several times. In one person's judgment it may not be significant; in another person's judgment it is.

We have had problems, and the point I was trying to make was that we don't consider the number of NCR's important. We consider the problems we have with the program what was important, and we did have problems, there was no question there.

MR. KERR: Well, I could convince myself that a large number of NCR's were good. It means that people who are responsible are on their toes. Do you look at NCR's as bad?

MR. BEASLEY: No, sir, I don't. Well, they're good and bad. If you have problems they're bad, but it's good when you confess them and report them.

MR. KERR: I like that language. It sounds like you've got religion.

(Laughter.)

MR. BEASLEY: We have been aproaching our quality program with a religious approach. We feel that it has to be something instilled. Our manager of our office has repeatedly told our office that in the quality program everybody has to believe in it, everybody has to carry it out.

It's just like he refers to industrial safety. You can have -- a few people can foul it up. So we've been trying to get everybody on board.

MR. EBERSOLE: Would you say the number of these reports reflects the intensity of the review process?

MR. BEASLEY: I think that's fair, yes.

MR. SIESS: Well, I don't quite understand. I guess one way I could look at it, a large number of NCR's would indicate you have a very effective QC/QA program.

MR. BEASLEY: Yes.

MR. SIESS: It might also indicate you have a lousy construction program. If there have been a lot of nonconformances, deficiencies in construction, I guess even a poor QA program could pick up a lot of NCR's.

MR. BEASLEY: That's correct.

MR. SIESS: So it's a relative thing, and I don't see how you could draw any conclusion regarding quality of the plant as opposed to the quality of the QA program from the number of NCR's.

Could you give us an example of, one or two examples of the actual deficiencies in design or construction that have been discovered at the Watts Bar plant?

MR. EBERSOLE: Let me offer one. The one that struck me as being particularly bad was the progressive installation of hangers on embedded plants up to, evidently, far beyond their ultimate carrying loads, with no integration of the loads as they proceeded compared against the ultimate strength.

MR. SIESS: These are the steel plates cast into the walls for attaching things to? They have those at Watts Bar?

MR. BEASLEY: Yes.

MR. EBERSOLE: People were hanging things on these indiscriminately. There was no definition of the ultimate hanger load that could be put on them. I wondered when I read it, is there any real knowledge as to how many embedment tie rods there are back in the concrete which really maintain the security of the concrete.

MR. SIESS: If that's not shown on the drawing, that's a more serious deficiency than you cite.

MR. EBERSOLE: My question is, even though they're shown on the drawings, do these exist in fact?

MR. SIESS: On one of the plants they use bolts, and you can see them on the other side of the wall.

MR. EBERSOLE: That's good evidence.

MR. WARD: Was that an example of a low threshold?

MR. EBERSOLE: I thought this wasn't a low threshold. I thought it was a high threshold.

Can you give us another good example?

MR. BEASLEY: I think that's a good example.

Larry?

MR. MILLS: We might have a couple of examples here. John Raulston might have a couple of examples on the NCR's, where we think we have a rather low threshold type item.

MR. BEASLEY: By way of introduction while he's getting the mike, John Raulston is the chief nuclear engineer for the Division of Design, and his organization handles the actual reporting of these items to Region II.

MR. RAULSTON: I guess one of the classic examples of an NCR, that I don't know whether it was reported or not but it was deemed significant, was where we had written an earlier NCR and in the corrective action defined certain milestones that had to be met and we failed to meet one of those milestones by, I believe it was, two days, and we wrote an NCR on that failure to meet the milestone and deemed that one to be significant, which meant it was evaluated for reportability.

I kind of feel that's a low threshold of a writing of an NCR. But that's a fairly common practice in the design organization. I've talked to other architect-engineer firms on the subject and I think by comparison with their programs I think ours is quite good and we have a low threshold.

MR. SHEWMON: Fine. Go on.

MR. BEASLEY: When we had --

MR. SIESS: Let me follow up just a minute on the example that Mr. Ebersole gave me. Can anyone tell me whether they have followed up on that problem and found instances where the integrity of those anchors are actually deficient because of that practice? I admit it was a bad practice. I'm wondering now whether it led to a safety issue.

MR. BEASLEY: Ralph Pierce, the OEDC manager for the Watts Bar project. He's manager over design and construction. He might want to respond.

MR. PIERCE: We have made a follow-up on all of the embedded plates. We are looking at the loadings on the embedded plates, and we have looked at some of the worst loadings that we had and did a sampling, and I think the number was 60 of the worst cases of embedded plates.

And we found only one plate that was marginally overloaded in this whole review, and this was reported to OI&E in Atlanta, and that's all we found in this review. We are continuing with the program of better keeping up with the loading on embedded plates.

MR. SIESS: Thank you.

MR. EBERSOLE: May I ask a question in this connection? When you're dealing with something like that, there are two areas I find difficulty imagining how you proceed. Do you take it for granted that what you saw in the construction drawing was actually realized, that the embedment was not just a solid steel plate but it had other attachments thereto and those were in fact put on?

MR. PIERCE: Yes. We have our documentation program on the construction site, which we have the QA records, the accountability records of these inspections of the embedments: the number of studs that were on this, that they did meet the design requirements, that they are in that location and that these records are complete.

MR. EBERSOLE: You have the records of the embedments?

MR. PIERCE: We have paper records of everything.

MR. EBERSOLE: The other question is, if I have a highly responsible weld, perhaps in the raw service water system, and I don't have any paper on it, how in the world do I accommodate myself to this situation, knowing that the quality of that weld is procedurally independent and there's no way to test whether it's any good or not?

MR. PIERCE: Are you saying a pressure-retaining weld?

MR. EBERSOLE: Yes.

MR. PIERCE: Are you talking about ASME Section 3? We have all these records on the inspection of these?

MR. EBERSOLE: You don't have any problem with other structural members other than piping?

MR. PIERCE: Structural members?

MR. EBERSOLE: Yes. Do you have adequate weld records on all of those?

MR. PIERCE: Yes, we do. Cable tray supports, you name it, hangers; we have complete records on it. Documentation right now on Watts Bar amounts to something like 360,000 separate records.

MR. EBERSOLE: Thank you.

MR. MOELLER: Excuse me. You have of course told about your QA program and the problems and your efforts to correct them. Now, I understand that INPO is reviewing or has reviewed your QA program. Is their report out?

MR. BEASLEY: INPO has a new organization looking at design and construction which is in the process. TVA is participating in and supporting that program. We have had some people working with the steering policy group on that. We had people working with INPO in setting up the criteria by which they evaluate, and we have had two people spending full time in the evaluation team.

The brackets for the INPO reviews cover plants that are at least 20 percent and not yet up to the 80 percent point. Watts Bar we figure is too far along to go through the INPO review. It's mainly looking at and evaluating the program.

We are doing a self-evaluation on our Belefont plant. But the program we have with ourselves, with the nuclear safety review staff and with the independent review, we think will be better than going through the INPO evaluation.

MR. MOELLER: Well, the SER prepared by the Staff says at the top of page 1-6: "The Applicant has proposed to have INPO perform an independent audit of Watts Bar design and construction program before fuel loading. The Staff has not yet determined the acceptability of this proposal."

That doesn't --

MR. BEASLEY: I'm sorry, I can't address that.

MR. RAULSTON: This is John Raulston.

I think that refers to about an April 26th meeting we had with the Staff, where at that time we were suggesting that perhaps the INPO review program would be a suitable type of review to do on the Watts Bar units. We have since, in discussions with INPO, decided that that was not appropriate.

MR. MOELLER: Well, again on page 1-11 in the SER, where it has "Summary of principal review matters," it says that -- well, it says: "TVA's QA program for the operation of the facility is a principal review matter." But I guess it doesn't say there that you are depending on INPO.

So the SER on 1-6, it's not -- I've read it wrong or it's not right or what?

MR. BEASLEY: The review by INPO --

MR. MOELLER: Let me ask the Staff.

MR. KENYON: I'm Tom Kenyon, project manager.

The item on page 1-6 was in reference to the April meeting with TVA. At that time they had indicated they wanted -- they were considering using the INPO program.

MR. MOELLER: But it's changed?

MR. KENYON: Since then it's changed, and we didn't have time to take it out of the SER.

MR. MOELLER: All right.

MR. EBERSOLE: Go ahead.

MR. BEASLEY: Okay. I'd like to go back and summarize. We have had our major quality problems. Their roots began prior to 1980. We began finding them out, getting on top of them in 1980. And we now feel very good about our Watts Bar plant. We feel good about the design. We feel we have an excellent design. We feel good about our hardware.

When we were going through our root causes and looking at the major problems back in 1981, we found that in general we had very good work; we just had trouble assuring that work. We had other problems with management control relative to QA, the transfer of design information in the field --

MR. KERR: Excuse me. What does it mean, you are sure you have good work but you have trouble assuring the work?

MR. BEASLEY: We feel comfortable when we look at our work that we have had good workmanship going in. Looking at the plants, the things that we have had to correct after we identify problems, we go back through, we feel like we have had a very good record there.

MR. EBERSOLE: My impression, Bill, is what he means is this -- correct me if I'm wrong -- that he's found good work where he looked at it, where there was absence of the paper record. Is that correct?

MR. BEASLEY: That's basically correct. Overall we feel good about the work. You can see good workmanship. You can see things go around. And we see evidence of good workmanship in many of the things we look at.

MR. SHEWMON: We get the point. Why don't you go on with your summary.

MR. BEASLEY: Okay, I think I was basically through with the summary.

On our quality problems, we're confident that when we complete the corrective actions and based on completing the corrective actions for the nuclear safety review staff, completing action on the items that Region II has identified, we are confident that our independent review will verify that we do have a good design, a good construction, and our quality program will be such that we will have an adequate plant.

MR. EBERSOLE: Thank you. The next topic we have is --

MR. MOELLER: Could I ask, I think this is an appropriate place to ask it. I am again reading in the SER on pages 1-8 and 1-9. On 1-8 it gives the total reactor coolant flow rate. We are talking about design here. I think the numbers are wrong. It says 14,300,000, or 14,030,000 pounds per hour. I think it means 140 million. But then, it has Sequoyah and Watts Bar and McGuire all about the same. And yet, on page 1-9 where it gives the reactor coolant pump flow rates, there is a significant difference between Sequoyah and Watts Bar.

Why is that? Why can you use different pumps to pump the same pounds per hour but a whole lot less gallons per minute?

MR. BEASLEY: Larry, can someone pick up on that?

MR. MOELLER: Do you have the SER, page 1-8 and 1-9, and then, on the same question, like on page 4-21, when you talk about the same -- or, when the staff talks about it, I presume they took these numbers from your safety analysis report. On 4-21 the flow rates are totally different than those on 1-8.

MR. KERR: It is possible that --

MR. NOVAK: Perhaps the staff needs a QA program.

MR. KERR: My experience would indicate that it is possible that something from a previous SER -- that the word "processor" indicates the computer might have stuck in a flow rate from Grand Gulf SER inadvertently.

(Laughter.)

MR. MOELLER: They could look this up and tell me later.

MR. SHEWMON: Fine.

MR. MOELLER: Do you follow on page 4-21 are not the same on page 1-8, and the gallons per minute on 1-9 do not correspond at all with the flow rates in pounds per hour.

MR. SHEWMON: Are you Mr. Williams? Please come on up.

MR. EBERSOLE: The next topic is the margin of safety above SSE. I think there is a growing realization that the assignment of the SSE is, to any extent, somewhat arbitrary. And if one wants to feel warm about whether plant will really survive a safe shutdown earthquake, we have to have some feeling for the range of margins that must effect a safe shutdown.

This really has led to this topic here, and

Mr. Williams is going to present it for us.

(Slide.)

MR. WILLIAMS: I am John Williams, Division of Engineering Design, Supervisor of the Component Qualification Section. All safety-related electrical and mechanical equipment has been qualified to levels which envelope conditions defined for its as-installed configuration.

TVA's equipment seismic qualification program is in full compliance with NRC and industrial recommended procedures, guides, codes and standards and good engineering practice.

Our construction permit was granted in January of 1973. We have gone back and updated the qualification methods to comply with IEEE 344-75. As a matter of fact, our seismic qualification instructions to vendors which were made a part of our equipment specifications included draft versions of 344-75.

We also were looking ahead to expedite licensing for Watts Bar and included operability requirements for active pumps and valves to comply with Reg Guide 1.4A. This included going back and working with Westinghouse for their scope of supply and their nuclea steam supply system for both the seismic qualification and also, the operability assurance.

Equipment qualification reports provide a conservative demonstration that the equipment is capable of withstanding its prescribed seismic conditions. The current philosophy regarding seismic qualification throughout the industry -- and TVA's program is typical -- does not require that the effort be extended to determine how much better the equipment is than it needs to be, nor does the qualification data lend itself to the extraction of such information.

The seismic qualification program, as we know it, cannot be transformed into an equipment reliability program. Re-evaluation effort would provide indications of margins of conservatism in qualification of specific items of equipment.

(Slide.)

MR. SHEWMON: Sir, are you reading from something I have?

MR. WILLIAMS: I hope so.

MR. SHEWMON: If I understand what you just told me, you said that it is very difficult, from the qualification test, to tell what margins are. Now, with regard to reliability I would grant that, but I would hope that if you have to have a piece of equipment that will cope with an acceleration of 1g that you don't quit at 1g, instead of seeing whether it might take 2g. Can you help me?

MR. WILLIAMS: The effort you have referred to would be more in a fragility test. Equipment is qualified both by test and analysis.

MR. SHEWMON: Yes.

MR. WILLIAMS: If it fits the analysis criteria, of course. Typically, equipment is qualified so that the test response spectra envelopes the required response spectra. So that if you have some margin between these, you can evaluate that. But unless you go into a fragility test program, you do not have what you referred to.

MR. SHEWMON: You don't have reliability.

MR. WILLIAMS: You don't have the requirement to go until the equipment breaks, as you would in a fragility test.

MR. SHEWMON: I know that, but how far do you go?

MR. WILLIAMS: The instructions are in the industry guides, which require that you, in the case of a test, that the test response spectra envelopes the required response spectra.

MR. SHEWMON: So someone will do a different test on every piece of equipment they ship, doing just what they have to for a plant with the lowest design base SSE, and testing a little bit higher for that which has a higher SSE?

MR. WILLIAMS: Well, this is typically true. And most of our effort is evaluating previous test equipment to satisfy the TVA requirements.

MR. SHEWMON: So you don't do the test and tell them to test as low as they possibly can? Someone has already done a test that should bracket things?

MR. WILLIAMS: That is correct.

MR. SHEWMON: So indeed, if you read what the person who did the test provided for data, it may well tell you something of the margins, would it not?

MR. WILLIAMS: It would if it is qualified to a higher response spectra than is required for your appliation.

MR. SHEWMON: What is the SSE for your plant?

MR. WILLIAMS: The ground input motion is .18g.

MR. SHEWMON: That is not too exciting, given what goes on in this country. Do you know how often the equipment you have was tested to higher levels than what you needed in your particular location?

MR. WILLIAMS: Not exactly.

MR. SHEWMON: Usually they did go higher, or usually you had to call them up and ask them to have things requalified to meet your specs?

MR. WILLIAMS: The equipment that we purchased is qualified to meet our specifications.

MR. SHEWMON: But you don't even inquire about whether it is qualified to meet more? Is that what you are saying?

MR. WILLIAMS: That is correct.

MR. SHEWMON: But you could find out if you asked, because they sell the same equipment to plants with larger SSEs.

MR. WILLIAMS: Sometimes you have to pay them extra for this information, and there are, of course, generic programs to compare qualification at other plants. Of course, we have a unique opportunity in looking at equipment that is qualified for seven different plants for progressively increasing seismic environments.

MR. SHEWMON: Thank you.

MR. SIESS: Paul, could I help you a bit?

MR. WILLIAMS: Another feature of the NSSS supplier --

MR. SIESS: Excuse me. Could you put up your third slide?

(Slide.)

The one that says "minimum factors of conservatism of Browns Ferry equipment." Was the 250 volt DC circuit breaker board qualified by tests?

MR. WILLIAMS: Yes.

MR. SIESS: Then can I conclude for breaker trip, it was qualified to an acceleration of 5.45 or some value higher, or is all of that 5.45 in the FRs and FRe?

MR. WILLIAMS: I am not prepared to answer that.

MR. SIESS: FRs says structure response factor, which is test response over actual response. It looks to me like some part of that margin is the ratio of a test acceleration to actual. I don't really know what "to actual" -- I don't really know what the words mean. But this is a specific example of the types Dr. Shewmon is askin about. I thought that would help.

MR. WILLIAMS: That is correct, but he was asking for a general idea.

MR. SHEWMON: I was asking for the specific words you used, and I don't have a copy of what you are reading. But go back and read it and I will pay attention.

MR. SIESS: I had the same question, Paul.

The NSSS supplied equipment -- TVA, as you pointed out, has relatively low seismic. For all the equipment that was qualified by test by Westinghouse, you are qualified to much higher levels than are required for the TVA plant. It is probably somewhere 1.5 and 2.

MR. SHEWMON: That would be my impression, but that is not the impression I get from the presentation. Please go on. It is just the viewgraphs, it not what you are reading.

(Slide.)

MR. WILLIAMS: To address the seismic margin of conservatism, the Sequoyah nuclear plant has just undergone a re-evaluation of equipment qualified against the higher seismic levels of the site-specific spectra, and it demonstrated that the qualification had been accomplished with a factor of conservatism of at least 1.5.

In other words, the equipment and structures were re-evaluated to a higher level seismic input by a factor of 1 1/2. The equipment, structures, piping were shown to be adequate for this new site-specific spectra. So that demonstrates a conservatism of at least 1.5. So I am saying that that isn't a minimum, but it has been shown that it at least has a factor of 1 1/2.

MR. SHEWMON: Did the staff change your SSE by 1 1/2? Is that it?

MR. WILLIAMS: We were required to look at the 84 percentile seismic event.

MR. SIESS: Yes, it got changed. I forget whether it was 1 1/2, but it got changed.

MR. WILLIAMS: That is what it represented.

The Browns Ferry nuclear plant has just undergone a probabilistic risk assessment study which included the consideration of equipment qualification. The study found that most equipment reflects large margins of conservatisms beyond the prescribed seismic conditions. The weakest link is relay chatter in the electrical equipment.

The Watts Bar nuclear plant will undergo a similar study, and the current schedule has a target completion day of May of 1984.

To address the results, --

MR. EBERSOLE: May I address the relay chatter? One might address -- if that is the weakest link, then one might consider fixing it. So we inquired at the subcommittee meeting what was the consequence of relay chatter. And I will pass on the answer, as I understand it, to the full committee here.

We were told by the electrical people that whatever the relay chatter did, it did not reflect a terminally ruined set of electrical apparatus. That they could go back and reset, by manual or other means, and therefore, relay chatter should not be considered in the light of broken machinery beyond recovery. Does anybody want to comment?

MR. SIESS: Does it do something when it chatters? Does it open or close components?

MR. EBERSOLE: I am sure it creates a variety of transients. I take it that TVA has stated that those transients are damage of no permanent significance so that you cannot reset the plant and realign it and continue on your shutdown process. Am I correct? Would one of the electrical people comment on this? I have a little difficulty believing that you can start motors in succession within the period of a few seconds without getting a little trouble here and there.

MR. LEE: My name is Ron Lee, Electrical Engineering Branch. The testimony that was given Tuesday was as you stated. I really can't add anything further than that, except it was stated that we could. When you reset the equipment it would not be damaged and we could recover from that.

MR. EBERSOLE: The chattering relay will send contradictory signals to equipment. It will not be in coordination.

MR. SHEWMON: But then, the wave doesn't

continue, and the next time you try, presumably it will work.

MR. EBERSOLE: A momentary impulse can close and seal in a function which you would rather not have occur. Is that a complex investigation, may I ask TVA? Is a relay chatter investigation -- do you really do an intensive investigation of the ultimate potential relay chatter considering seal-in functions and contradictory functions? It sounds like an extensive problem.

MR. WILLIAMS: Mr. Ebersole, we do that when we note this in the seismic qualification report. We ask the electrical people to evaluate the operability or the acceptability of the relay chatter, and in this case, it was indicated Tuesday that it does not cause failure of latching which occurred at a factor of 5.45, as the failure.

MR. EBERSOLE: Is there a summary report that might be titled the effect of relay chatter, uncoordinated relay chatter in the Watts Bar plant?

MR. SIESS: As a transient, not as permanently disabling equipment.

MR. LEE: Not to my knowledge.

MR. EBERSOLE: Where is this information buried, that you can tolerate relay chatter without serious permanent effect? It sounds like it has a degree of casualness about it that may be inappropriate.

MR. WARD: Could I understand something here? You are not saying that at the design basis, your .18 earthquake, that you are getting relay chatter? You say as you go to higher seismic loadings, that is the first thing that you have? I don't know --

MR. SIESS: Dave, that figure up there says they have a factor of 1.4, taking into account three conservatisms. So I don't know what the range is.

MR. WARD: I think it is obvious that they don't have all seismic safety margins defined; nobody does.

MR. SHEWMON: We have a presentation.

MR. EBERSOLE: We are doing pretty good on schedule, Paul.

MR. SHEWMON: I would like to keep it that way.

MR. EBERSOLE: Before we drop this, the staff would like to say something, and I wish they would, about the generic aspects of the problem.

MR. NOVAK: I would only point out prior to the licensing of the Trojan nuclear plant there was a substantial amount of review done on electrical components specifically with regard to chattering effects on electrical equipment. It was done mostly around the nuclear steam supply system equipment, and the resolution was well defined.

I don't want to try to speculate on that, but I do know that because of the seismic design requirements placed on that facility sort of late in the design a question arose as to whether those components would, in fact, perform acceptably. They would induce transients That was clearly one of the things that would happen.

But the emphasis was could you accomplish a safe shutdown, and the answer was yes.

MR. BUSELL: George Buswell from Westinghouse. I would just like to confirm Mr. Novak's statement. As far as my memory goes, we only ended up changing our relays on protection systems for two or maybe three peices of seismic qualified equipment. Protection systems for Westinghouse for this level of plant do not have a relay chatter problem at all.

MR. EBERSOLE: Thank you, let's carry on.

MR. WILLIAMS: Okay. Continuing on the Browns Ferry equipment, the reactor pressure vessel internals, the factor of conservatism 2.45; diesel generator transformer, diesel generators 2 1/2, transformers 2.8 -- no, I am sorry, I am off a line here. The contr-l rod drive housing, 3.85; all other equipment, larger margins.

(Slide.)

There are several generic programs underway to address seismic margins of conservatism. I am sure you are familiar with the seismic safety margins research program underway under the direction of Lawrence Livermore Labs.

I pulled a quote out which I like, "An important observation is that most mechanical and electrical equipment is inherently rugged and will survive acceleration levels far in excess of building responses associated with the safe shutdown earthquake."

Another generic program is the seismic design margins of pumps, valves and piping fluid system components, sponsored by the NRC, done by Mr. Everett Rodenbaum. It was written by Mr. Butterworth. The Watts Bar plant is in a low seismic area relative to other plants, and the numbers that you see are representative of the margin of conservatism that the Watts Bar equipment would have.

(Slide.)

From the first program, the minimum fragility values of Zion, the factors of conservatism -- and remember the other slide which had the definition -- the 125 volt AC distribution panel, the factor of conservatism is 3 1/2; service water pumps, 3.7; 4000

160-volt switchgear, 4.2; other equipment, larger.

MR. SHEWMON: Sir, were those by test or by calculation?

MR. WILLIAMS: The 125-volt AC distribution panel and the 4000 160-volt switchgear would be by test, together with the other factors. The test is only one part of it. The service water pumps would be by analysis.

MR. SHEWMON: I guess it is with data like that that I did not understand your original comment, nor does the qualification data lend itself to extraction of such information. It talks about how much better equipment is than it needs to be for the SSE. But I did find what you are reading, so go on.

MR. WILLIAMS: Okay. The program, as it currently is, does not require that we establish monitors.

MR. SHEWMON: Requiring wasn't the question. It was whether it is easy or difficult.

MR. WILLIAMS: It is difficult.

MR. KERR: Is TVA convinced that the current requirements are adequate?

MR. WILLIAMS: I would say that they are.

(Slide.)

Looking at the design margins for fluid system components, what we see here is simply the margins or the nominal margins, which is simply a ratio of the yield or the ultimate for the material divided by the code allowable stresses. And the figures are as you see. And this is simply what is in the code for the material. Also, for the AISC manual.

In other words, a factor of 1.67 is simply your allowable stress of 20,000 psi. That contains a factor of 1 2/3 on the yield. This does not address the other margin that is available. Typically, material is 1.2 times minimum specified values and other factors that are inherent. For instance, standard flanges are designed to an allowable stress of 7000 psi, which is quite a bit more conservative than these values.

MR. EBERSOLE: John, I am not a structural man. What is the meaning of .55, if I am looking at a pump or a valve and I have a shaft linearity problem, where any kind of distortion of the shaft alignment will cease the shaft and lock up the pump?

MR. WILLIAMS: In this case, you wouldn't be allowed to go to the yield criteria. That is an operability question.

MR. EBERSOLE: Is that a second set of numbers that is not here?

MR. WILLIAMS: It would be the bottom number here of the 1.1 to 4.8. There you would only go to service level B conditions.

At the time, the code of record for Watts Bar does not address operability per se in the code. The current editions of the code do address operability and they limit the stresses to service level B, which would give you the 1.1 to 4.8 factor.

MR. EBERSOLE: You went back and looked at the components to see that?

MR. WILLIAMS: Yes, yes. We do have the operability program in place for Watts Bar.

MR. EBERSOLE: Also in place for Sequoyah?

MR. WILLIAMS: For the seismic qualified components, yes, we did look at that.

MR. EBERSOLE: Did you find any fixes necessary?

MR. WILLIAMS: Well, you are pressing my memory.

MR. EBERSOLE: All right. As long as you have methodical program to turn to to look it up.

MR. SHEWMON: Is that about it?

MR. WILLIAMS: Yes, I believe it is.

MR. SIESS: Mr. Chairman, may I explore a little bit on this margin business? We need some information in connection with what we have putting into letters. Could you go back to that third slide on the Browns Ferry equipment where, at least for that case, the factor of conservatism was defined as being made up of three components.

Let's take again that circuit breaker board which I am sure was qualified by test. Would the FCe strength-load ratio have any application there, or would that factor of conservatism be only involved in the other two?

MR. WILLIAMS: In the case of tests, it is very difficult to evaluate that factor.

MR. SIESS: That doesn't answer my question. Does the FCe equipment capacity factor equal to strength over load having any bearing on that first number?

MR. WILLIAMS: I don't see how it could.

MR. SIESS: The second factor, which is test response or actual response, I don't understand. If it said test spectra over actual spectra I would understand it. Is there any chance that is what it means, or does it mean something else? Test and actual are usually the same thing, to me. Does it mean test over computed or expected?

And the next item takes a ratio of design to anticipated. Could you explain what that means? I know what the design floor spectra are.

MR. WILLIAMS: I would have to go into the report to read it in more detail.

MR. SIESS: Is this a report the NRC has? Could you give us a reference to the report?

MR. WILLIAMS: No. This is a Browns Ferry report. I don't imagine it has been submitted to the NRC as yet.

MR. SIESS: Because in connection with previous cases where the ACRS has asked for seismic margins on things like piping, of course, people have come back and given us stresses pretty much like you had on the last figure, and have essentially ignored the fact that there are other conservatisms built into the calculation of spectra of the earthquake, et cetera and so forth in the analysis method.

This, I think, is one of the first instances I have seen where there is an attempt to look at the conservatisms due to differences between -- conservatisms built into the analysis, actually giving floor spectra and so forth.

But I think we would be interested in seeing what was done there. If you could provide us with a reference, I think that would be helpful.

MR. WILLIAMS: Well, the reference is the Browns Ferry PRA study.

MR. SIESS: It is in the Browns Ferry PRA study?

MR. WILLIAMS: Right.

MR. SIESS: Has this been published? Does the NRC have a copy of that? He says the answers to my questions are in there, so I would like to get the document that has the answers to my questions.

MR. MILLS: Sir, I have just been told that this report will not be issued until sometime next year. But I wonder if we could get back through the staff to you with perhaps extracted material from that report.

MR. SIESS: Anything that indicates what these things are would be very helpful. And other things we are doing, not just Browns Ferry. I mean, not just Watts Bar.

MR. SHEWMON: Okay, thank you very much.

MR. EBERSOLE: I'll comment on the next topic. TVA experienced what I believe was an extraordinary corrosion problem in certain carbon steel piping, and resorted to a process of mortar lining of the critical water pipes. The first thing that springs to mind is whether this material will degrade over time, and while it may not be physically, it may potentially be loosened so that under subsequent seismic loads it will unload and, in a common mode fashion, plug up the process pipes.

We have Chuck Bowman here, who has a presentation on what they did about this and why they did it. It was really a pressure drop problem. Chuck, it's all yours.

MR. BOWMAN: My name is Chuck Bowman. I'm supervisor of a Section in the Mechanical Engineering, Branch of Engineering Design, responsible for the central raw cooling water system.

During preoperational tests in the emergency cooling water system at Browns Ferry nuclear plant in 1976, certain heat exchangers were found to be receiving inadequate cooling water flow due to a buildup of corrosion products in the interior of the pipe of carbon steel piping servicing this equipment.

Since carbon steel piping was extensively used in both safety-related and non-safety-related piping systems at other TVA nuclear plants, a study was undertaken to determine the pervasiveness of this problem in the TVA service system and to develop the recommended practice to mitigate its effects.

Approximately 50 sections of carbon steel raw water piping were removed from 9 different TVA steam plants. Both normally stagnant and normally flowing piping systems, as well as both vertical and horizontal rows of piping were sampled.

In virtually every case, the primary mechanism was found to be corrosion in the steel piping by aerated water river and redeposition of corrosion products. The problem was found to a significant degree at all the TVA plants that were sampled, and the result is a random pitting in the pipe wall and the formation of a tubercle over each pit. And I have a sample from Browns Ferry if you care to look at it.

(Slide.)

The equivalent average diameter reduction as a result of corrosion products buildup as a function of years of service is shown here in Figure 1. The deposit in each sample was removed and analyzed for various constituents. In virtually every case, it was found to be principally iron oxide.

We have plotted a decrease in diameter of the pipe versus service years for the samples that were taken. Tests were performed at Widows Creek, Kingston, and Gallatin steam plants to evaluate the effects of corrosion product buildup on pressure drop. These sites were selected to cover a range of ages as well as a variety of water sources.

Samples removed from each test line were analyzed to determine the percent volume reduction of the pipe interior due to the corrosion product buildup. The corresponding diameter reduction for each test line was then used with the pressure drop test data to develop appropriate equations for pressure drop.

Several figures were generated in an attempt to find a correlation between diameter reduction and Hazen-Williams C factor. Values of C were assumed and corresponding values of diameter were calculated for each test to give you the required pressure drop. Finally, a dimensionless parameter which we call D, was developed for use in correlating the above calculated value of d with the measured value of diameter reduction.

(Slide.)

Using this relationship for each of the three pressure drop tests conducted as shown in the following figure, good agreement was discovered --

(Slide.)

-- good agreement was discovered for values of d equal to 2 and a Hazen-Williams C factor of 57. We have adopted a slightly more conservative value of C equals 55 in our design equations. The result from this analysis is a modified Hazen-Williams equation.

(Slide.)

Figure 3 shows the comparison with the factor of 5 and d equals 2, and reducing the nominal diameter by two times measured diameter we have produced this modified Hazen-Williams equation, which has proven to give us good correlation for all of the tests that were conducted.

(Slide.)

Here we see the comparison between the predictive model and the actual test data taken on a 3-inch line at Widows Creek. Note also the head loss predicted by the normally used Hazen-Williams equals C equals 100.

Similarly, we have comparisons with the other tests that were conducted. Here is a test that was conducted at Kingston on a 6-inch line, which gave you -- which also gave you good agreement with the predictive model.

(Slide.)

And the test that was conducted on the Gallatin 8-inch line, which also gives you good agreement with the predictive model.

(Slide.)

TVA is now using this equation to evaluate the heat rejection system, fire protection, raw cooling water system, and raw service water system at Watts Bar. Most significantly, however, we have completed our evaluation of the Watts Bar ERCW system, and the remainder of this presentation discusses results of that evaluation.

The analysis of the ERCW system determined that delivery of design flow rate to system users over plant life could not be guaranteed with the original design. Consequently, changes were defined to bring the system within the 40-year design basis.

(Slide.)

These changes include: Replacement of selected segments of carbon steel piping within the buildings with stainless steel;

Requalifying certain system users to a lower ERCW flow rate by refinement of the heat transfer design calculations;

And finally, by applying a cement mortar lining to the existing carbon steel piping in the yard in situ.

Before making a final decision to cement mortar line the ERCW system yard piping, a telephone survey was conducted to determine how well it had performed in service. In Table I which you have in your handout, you see there that a total of 11 other utilities, 2 A-E's and 5 municipal water systems which have cement mortar lined piping in service were canvassed.

Although a few problems were reported, in general the experience reported was very good. All of the problems identified could be attributed to either the pipe being out of round or a failure to properly protect the pipe joints.

Mr. Ebersole, in response to your concern about the seismic event, during the 1971 San Fernando earthquake, within three miles of the quake epicenter a 96 inch above-ground water line owned by the Los Angeles Department of Water suffered both vertical and horizontal displacement due to surface acceleration. The pipe was broken from its supports and accordioned. The 34-year-old cement mortar lining was undamaged except at the place where the pipe was accordioned, where it did spall.

A 24-inch diameter steel tunnel liner, the Foothill Feeder in Seguna, California, Metropolitan Water District of Southern California, was buried 50 feet underground and did not deform from the earthquake. Buildings on the surface were destroyed and equipment in the tunnel itself was shaken down. However, the cement lining that had just been in place only 12 hours was not damaged, nor was the older lining damaged.

The Balboa pipeline, owned by the Metropolitan Water District of Southern California, a 14-foot line to the Jennison Treatment Plant, separated about 3 inches in two locations and then was driven back together, one section inside the other. The lining was only damaged where the pipe separated. And the plant was 75 percent destroyed.

It appears that unless there is deformation of the pipe the cement lining is not damaged. We will address the seismic qualification of the piping later in this presentation.

The procedure for applying the cement mortar lining requires that the piping first be cleaned by scraping off existing tubercles. Thereafter, the mortar is centrifugally applied from a spinning head and immediately trowelled onto the inside surfaces, using a machine which is pulled through from one end. Closure pieces and certain elbows are thereafter hand-mortared to complete the process. The closure weld-affected region is hand-lined from inside the pipe.

MR. SHEWMON: What diameter can you go down to on that?

MR. BOWMAN: We lined pipe 24-inch and larger.

MR. SHEWMON: Okay.

MR. BOWMAN: We only lined pipe in safety-related systems.

MR. SHEWMON: For this (Indicating)?

MR. BOWMAN: For that we replace it with stainless steel.

Humidity is carefully controlled after application to ensure proper curing, and each foot of piping is carefully inspected prior to plant operation.

TVA specified the necessary level of quality assurance on the lining process and a number of nonconformances to the specifications have occurred. These have included: both high and low mortar slump; high mortar temperature; low mortar compressive strength; low relative humidity; surface cracks; mortar applied too thin; end caps that were not replaced; pipe damage; and exterior coating damage.

Where appropriate, repairs have been made in accordance with approved procedures.

Very good experience has been reported with cement mortar lining by other utilities. However, since this is the first application by TVA and since the ERCW system is a safety Class 3 system, provision is being made in the design to facilitate periodic inspection of a portion of the system after it has been placed in service.

Now Frank Hand will come and address seismic qualification.

MR. EBERSOLE: A question. Are the tubercles thought to come from the tube itself or from the boiler?

MR. BOWMAN: There's been quite a bit of literature published in the AAWA journals. The general consensus of opinion is that you have anaerobic digestion of suspended solids, creating an oxygen deficient cell, which creates acids, and you begin to have the pipe wall eroded. The corrosions come from the pipe.

On the surface of the tubercle you will find bacteria. That is probably from the water.

MR. ETHERINGTON: It might have been from steam?

MR. BOWMAN: You'll get different people's opinions, but for my judgment it comes from the pipe. It comes out of the pipe, the metal from the pipe.

MR. KERR: That word is spelled p-i-p-e, Harold.

(Laughter.)

MR. EBERSOLE: Some people just don't understood good English.

MR. BOWMAN: I have a sample of a pipe that's been pickled. There is pitting occurring in the pipe wall. We have done studies on the pitting, which was reported in the Subcommittee meeting.

MR. SHEWMON: Fine.

MR. MOELLER: A quick question. In terms of the problems of the corrosion and so forth, the volume of water you are dealing with is just far too great to do any chemical control of it, is that correct?

MR. EBERSOLE: It is once-through.

MR. MOELLER: That's what I mean. In other words, it would just be impossible?

MR. SHEWMON: There's no city water.

MR. BOWMAN: At Watts Bar it is once-through. We have considered closed systems where the water chemistry can be controlled. However, you would have to de-oxygenate the water to be assured of good water chemistry control.

MR. ETHERINGTON: Let me try once again, if I may, on my question. Do you get the same tubercles in stainless steel pipe?

MR. BOWMAN: No, sir.

MR. ETHERINGTON: So it's a local effect.

MR. KERR: Incidentally, I knew southerners didn't pronounce the word "AP[illegible]"*, but I didn't know they started taking iron out of the word.

(Laughter.)

MR. HAND: I am Frank Hand with the Civil Engineering Branch of TVA. We were assigned responsibility for seismically qualifying the cement mortar lining that's going to be installed in the pipe in the test program which will be described here.

(Slide.)

We had several specimens that we were going to line, because we were going to run several different types of tests. We were testing full-sized pieces of pipe, primarily 30-inch diameter pipe. We had one long 40-foot piece, another long 30-foot piece. We had another long piece over here that we subsequently cut into two-foot sections after it was lined for a specific type of test.

We had one 90-degree elbow with two five-foot extensions on it that was lined and then, to see what the difference was between a large diameter pipe and a smaller one, we lined one 18-inch diameter piece of pipe and cut it into two-foot sections also.

(Slide.)

The lining equipment looks like this little dolly that comes through the pipe. The mortar is slung out from this particular head. It is immediately trowelled in this operation with two spinning arms, which will give us a gun-barrel type of an appearance.

In the very foreground of this slide you can see the slicker mortar. There's a little bit of overlap or a rise where the mortar joints coming from the spinning trowel aren't particularly smooth. The orange peel-rough looking surface back here is untrowelled mortar that's been slung on, and the shiny surface in the far back is an unlined piece of pipe.

The reason we have this particular view, we are looking at an elbow. This was machine-troweled, this was hand-troweled, then this would be machine-troweled over here coming in from the other side.

(Slide.)

Of the types of tests we ran on the two-foot specimens, we ran a simple three-inch bearing test, in which we would supply load along the line at the top and we would check the deformation. We were able to squeeze these pieces of pipe three inches --

(Slide.)

-- out of a 30-inch diameter, which gives us a 10 percent distortion rate. We lost no mortar in any of our testing for this particular apparatus.

(Slide.)

We have here a piece before we lined it -- excuse me, before we tested it. You can see the little ridges left from the troweling operation.

(Slide.)

After testing we have a crack in the very bottom, a slight crack over on the side, some other cracks up on the top. The cracks would look worse in the slide because we've accented those with felt-tip magic marker type pens. This is the extent of the damage that we would see when we would squeeze 30 inches of pipe down to 27.

(Slide.)

The maximum loads that the pipes were taking was about 12 Kips. We get small cracks about 3 Kips. So we carried this pipe well into its plastic deformation stage.

We also ran a torsion test on one two-foot specimen, by welding it to a one inch thick plate on the bottom, fixing up a loading apparatus onto the pipe. What we were trying to approximate here is what happens to one leg of a pipe as it goes into an elbow when the other leg is moving and perhaps putting some bending or torquing into it.

For this particular apparatus, we got quite a bit of cracking in the lining itself, but it was all where the cross piece ties into the pipe specimen. And we consider that to be a problem with our loading arrangement.

We carried this up until we actually failed the weld in our loading frame over this particular point. That was at about 60 Kips. Other than the deformation associated with the loading points, there was no major cracking. The pipe with our loading arrangement; we were able to warp it about a quarter of an inch.

(Slide.)

We took one long, 30-foot piece of pipe and we subjected it to a bending test. Due to the size of the specimen and the load we were putting on, we found it was much easier to hold down the ends than to push up in the middle, which is the opposite of what you would normally do in a small test.

We had a 100-ton -- excuse me -- a 100-Kip jack underneath. We ran the first load up to 80 Kips, at which point we ran into a plastic buckling type failure of the pipe at one of our support points and that caused some spalling of the lining on the inside.

We put bigger pads on our support points, rolled the pipe topside-down. This time we went up to 90 Kips. We had no spalling. We had only general cracking running around the pipe. In essence, the lining was relieving itself of any stress. And in this particular one we took the steel up into its plastic range.

We were able to deform the pipe enough so that vertically it was compressing one inch and it was expanding horizontally almost a half an inch.

(Slide.)

We then went into a series of impact or missile drop tests where, from up to 40 feet high, we dropped up to a 35-foot high. On the two-foot specimen, when it would hit it would cause a dent in the steel. This happens to be from a 14-foot drop.

(Slide.)

And only at 14 feet did we get any spalling of the concrete or the mortar on the inside. As this height progressively got up to 40, that spot got progressively larger, but it was always a spalling failure. We never got a lining to collapse.

(Slide.)

We also took the same bending test specimen and subjected it to drops starting at one and working our way up to ten feet. We could not get the lining to fall in that pipe either, even though it had already been tested and severely broken.

We tested our elbow again. Due to testing constraints, it was easier for us to pull the elbow together.

(Slide.)

We did so. During the test the elbow actually came -- the two ends were pulled five inches closer to each other. We wound up with a one and a half-inch set. We were getting close to rated yield stress on this particular area over here (Indicating).

We had no spalling, no cracking -- no spalling, no major cracking of the lining inside the elbow.

(Slide.)

We took our long, 40-foot diameter pipe, buried it in the ground at one of our nuclear plants. As you can see, we were scraping some dirt off it here with a bulldozer. We had put three feet of dirt on it, used a regular 400 vibratory compactor, subjecting it to testing, to vibration; removed six inches of soil, repeated the process; worked on down.

We could not get any cracks in this pipe specimen until we had the vibrator sitting directly on the bare pipe for over 15 minutes.

(Slide.)

Since our lining site was about 100 miles away from our test site, we thought it would be prudent to test it with accelerometers to see what kind of transportation excitation it had received. It turned out that for the pipes loaded on the lower portion of the flatbed truck they received about six-tenths of a g. For those loaded on the top portion they received slightly over 2 g's of acceleration.

We performed the four-inch spectrum analysis on the time histories we got. We found that the frequencies ranged from zero to 100 hertz, slightly richer than our typical earthquake, which would be in the 5 to 10 hertz range.

On the basis of all of our testing, which was on full-scale specimens, we saw that the lining showed considerable flexibility and ductility. We think we put the lining through paces more severe than what it would encounter during our design earthquakes, and on the basis of the testing the observed performance as well as the historical performance that Chuck alluded to, we believe that the pipe is fully qualified for seismic events that it would see in our plants.

MR. EBERSOLE: Well, thank you.

I guess I only have one question about it. That is, I guess I've seen too many WPA jobs with reinforcing heaving concrete. I believe that some heaving and spalling will occur over the years. But I also gather that there is a surveillance program in place that will detect any undue amounts of that. And I have no further things with this, Mr. Chairman.

Does anyone else have any comment?

MR. MOELLER: We presume they also would have filters in case --

MR. EBERSOLE: I don't know what the filter logic is or whether it's been altered by this. I suspect it has not.

MR. MOELLER: Could someone tell us?

MR. EBERSOLE: Have you altered any of your filtration logic as a result of using this material?

MR. PIERCE: No, we have not altered any filtration logic. We prefilter the water, not at the pumps; but we have not altered any of the filter logic used in the concrete lining.

MR. EBERSOLE: Thank you.

MR. SHEWMON: Find. The next speaker?

MR. EBERSOLE: The next topic is the problem with the Westinghouse D3 steam generator. Ralph Pierce.

MR. PIERCE: I am Ralph Pierce. I'm the design and construction project manager for the Watts Bar nuclear plant.

TVA became aware of the tube wear problem due to flow-inducked vibration in November of 1981. It began working with Westinghouse relative to the Watts Bar unit. The discovery of the tube wear problem was in Ringhals Unit 3 in October 1981 in Sweden. Sweden has since conducted two full-scale models of a portion of the previous section of the Model D3 generator at its hydraulics laboratory.

Only one of these test facilities is presently being used. In March-April 1982, Westinghouse entered into agreements with Sweden to have certain baseline and confirmatory tests performed as a part of the development of design modifications for the D2 and D3 generators. Test specifications, procedures, and quality assurance requirements were prepared by Westinghouse.

Operation of the model has been in Sweden.

Final data processing and evaluation has been performed by Westinghouse here in the U.S.

During the week of June 1, 1982, the model was in near readiness for starting baseline testing on the existing D3 design when they had an overpressurization event and caused substantial damage and delay. The model was placed in equivalent full flow operation on June 22, 1982.

Data collection and processing, including high technology photography began. Based on evaluation in the U.S. and at the site, initial baseline testing began on January 5, 1982. While the evaluation of the design of the modification was being performed, TVA began an economic evaluation of the options of making the modifications prior to fuel loading versus operating at 50 percent power level through the first refueling outage, or scheduling an outage at some point in time to do a modification.

It was determined that it would be to TVA's economic advantage to delay fuel loading and to do the modification prior to fuel loading, and specifically prior to our hot functional testing, which we felt was the best spot in the schedule to do this to approve the system and get it back in operation.

Based on preliminary information from

Westinghouse, it was determined that we had a high probability of obtaining an acceptable design, testing and having the work performed by Westinghouse prior to November the 3rd of 1982. The Watts Bar schedule was adjusted to reflect this.

The progress to date on the Westinghouse test is that the full flow test of the Model D3 first full manifold design is complete. This establishes the baseline for comparison purposes. These D3 design full flow tests are to be rerun with force gauges in approximately four of the tubes and will be done in August of 1982.

Based on model testing and analytical work by Westinghouse, they have a high level of confidence that the optimized test manifold will become the production model. Optimized manifold full flow test of the production model is now scheduled to begin in September. In your handout I said the week of September 30th, but Westinghouse informed me this morning that they're a little ahead of schedule.

MR. SHEWMON: The one in Sweden -- there are several of these slightly different models around. Is yours identical to the one in Sweden?

MR. PIERCE: Ours is identical to the on in Sweden.

MR. SHEWMON: Is the one in Spain, Maguire, slightly different?

MR. PIERCE: It's slightly different, but it's so close there's not much difference.

MR. SHEWMON: When you talk about this manifold, what physically are you talking about?

MR. PIERCE: We're talking about a manifold that would be put into the feedwater inlet nozzle, which will disperse the flow of fluid such that velocity at 100 percent load will be dispersed like a shower head, such that we will reduce the impingement on the rows of tubes in the steam generator and cut out the vibration that is causing the wear of the steam generator tubes.

MR. SHEWMON: This will give you somewhat more pressure drop, but it will avoid the vibration?

MR. PIERCE: Westinghouse can address the pressure drop. but nothing that will --

MR. SHULTHEIS: Joel Shultheis.

That's essentially correct.

MR. PIERCE: We've been working with Westinghouse. It's hard to tie down what will become the production model. They have recently informed us that we will be finalizing the schedule hopefully by the end of this month on Watts Bar. Also, TVA is working with Duke and South Carolina Power on their two reactors.

Watts Bar, Duke, and South Carolina at Sumner will be the first three commercial units to get the modifications. It has not been decided which will be first, second and third. This will be worked out mutually between the three utilities and Westinghouse.

So if there is any significant change in the Westinghouse schedule for the final modification, we will make another economic evaluation and note the place in the schedule where we will make this modification.

MR. SHEWMON: Fine.

What is the support plate material in your steam generators?

MR. PIERCE: I will let Westinghouse address this, the technical people.

MR. SHULTHEIS: This is Joel Shultheis from Westinghouse.

The support plate material in the D3 steam generator is carbon steel.

MR. SHEWMON: So they haven't caught that one yet. It's old-fashioned that way.

What about the condenser? What is your tubing material?

MR. PIERCE: Our tubing material for the main condenser?

MR. SHEWMON: Yes.

MR. PIERCE: Help me on this over here. It's copper bearing Admiralty, isn't it?

MR. COTTLE: This is Bill Cottle.

It's 90-10 copper tubing.

MR. SHEWMON: Where are you with regard to accepting the steam generator owners group criteria about operating procedures for when you start getting out of spec in the water, as you would with leaks in your condenser, or oxygen control or other things like that?

MR. COTTLE: Right now the present specs in the operation are very close to the Westinghouse owners group. There may be some slight difference.

MR. SHEWMON: I'm not talking about the Westinghouse owners group. There has been a steam generator group -- I didn't think that was only Westinghouse, but maybe it was -- that EPRI is managing.

MR. COTTLE: It's at EPRI, but the specifications are being written for Westinghouse steam generators and each of the other major classifications.

We are very, very close to that.

MR. SHEWMON: What will you do with oxygen, for example? Will you have procedures or inspection and control on that?

MR. COTTLE: We have procedures. We have an alert level, I believe, for leakage of 5 Cfm, and we have an action level which basically would involve plants that have to gain concurrence from higher level management to continue operation above 10 Cfm.

MR. SHEWMON: This is consistent with what the Westinghouse owners group is saying?

MR. COTTLE: That's fairly consistent, I believe, with the guidelines that are evolving there, yes.

MR. SHEWMON: Have you had a chance to react to what the NRC Staff put out on the 29th of last month as requirements on that, or do you know where you are relative to their checkpoints?

MR. COTTLE: I haven't seen those checkpoints yet.

MR. SHEWMON: You might look.

MR. PIERCE: We are doing additional modifications in the way of improving water chemistry. We are taking the copper out of the main feedwater heater, the condenser, we are modifying the demineralizer. We will be using some nitrogen bubbling through the condensate storage tank. So we have several modifications that have been factored into our schedule in water chemistry.

MR. SHEWMON: These are modified since Sequoyah or will you make those at Sequoyah 2?

MR. PIERCE: We will make them at Sequoyah 2. Some will be done during the outage as the need arises.

MR. ETHERINGTON: Do you have a deaerating heater?

MR. PIERCE: No. One of our long-range products is to look at the aeration of condenstate. But that is long range. Right now we are --

MR. EBERSOLE: Ralph, do you have any plans that you could identify that would preclude your having to do some unfortunate things with steam generators 15 or 20 years from now like having to take them out and throw them awway?

MR. PIERCE: That is what this is all about, the control of the water chemistry to take care of the steam generator denting. So this is the long-range program we have here. We get the vibration modification in. We control the water chemistry which has the denting which has occurred on those that have had the steam generator problems. So we are well aware that we must control this aspect of it.

MR. EBERSOLE: Let me ask one more question about the matter of the tubing. On the component coolant heat exchanger, what affects the tubing material?

MR. PIERCE: On the component coolant heat exchanger?

MR. EBERSOLE: This is the one that rejects the water to service water.

MR. PIERCE: I would have to get you an answer. I think it is stainless-steel.

MR. EBERSOLE: Is this carbon-steel, the tubes that reject heat to service water?

MR. MERRICK: Ed Merrick. I do not know.

MR. PIERCE: I think it is stainless-steel, Jesse.

MR. EBERSOLE: The reason I bring it up is I recall a flap at Westinghouse where they did not want to put three or four moderate levels and draw service water and predicted dire results for them to do this. And I thought that we had carbon-steel tubing in these component coolant heat exchangers next to all service water. Well, I will not pursue this matter. It is a matter of peripheral interest.

MR. KERR: Is there someone who can give me a two-or-three-word explanation as to why this vibration problem did not show up when Westinghouse initially tested the steam generators?

MR. SHEWMON: I am not sure they ever did initially test them, but maybe Westinghouse should try that one. You mean with the economizers?

MR. CANADA: Fred Canada, Westinghouse. The original scale-model test of the steam generator, I guess the best way to describe it was that when you try to scale-model something of the size of a full steam generator, there are certain phenomena which you observe when you run the scale models and certain scale models that you do not observe. And most likely this particular phenomena in the inlet for the preheater was not preserved in the scale-model tests.

MR. KERR: May I translate that to mean you did not test it correctly because you did not know how?

MR. CANADA: Quite frankly, I think the test was aimed at understanding and investigating other parts of the preheater and the steam generator as a whole and was not tested in the vein of anticipating this problem with the inlet configuration. And hence that part of the flow phenomena was not preserved when the model was scaled down to a laboratory size.

MR. SHEWMON: Would you put that thing back on the stand so you are forced to be a little further away from it?

Any other questions?

MR. KERR: I am afraid to ask another one.

(Laughter.)

MR. SHEWMON: Mechanical engineering just is not the exact science that instrumentation and control is.

(Laughter.)

MR. EBERSOLE: Mr. Chairman, the next topic would require that we vacate the room or clear it because of its nature.

Incidentally, I am a little surprised to see that on there. I take it its origin is Dr. Mark?

MR. MARK: No.

MR. EBERSOLE: I am personally willing to forego that topic unless members of the committee here want to go into it.

MR. BEAL: I think it is on there so that you can ask if there is anything that you should know.

MR. EBERSOLE: Is there interest within the committee of wanting to take up the security probelm or not?

MR. SHEWMON: It is off as far as I am concerned. Anybody want it on?

MR. MARK: You mean drop it?

MR. SHEWMON: Yes.

MR. MARK: Well, we could ask --

MR. EBERSOLE: You could ask general questions.

MR. MARK: I did not put it on. It seems to me the question that we do not need to clear the room for is possibly --

MR. KERR: I believe your reporter would like us to hold your microphone a bit further --

MR. MARK: One I could direct to the Staff: Have they looked at it? Are they content with what they found? Is it as good as --

MR. KERR: Browns Ferry?

MR. MARK: -- other plants? Or we can ask that question in open session.

MR. SHEWMON: Let us try something newer than Browns Ferry.

MR. MARK: Or Clinton or anything else. Are they up to snuff in this regard?

MR. KENYON: Sir, the Staff has just recently completed their review of TVA's physical security plan. And there was one item that they were objecting to. The

Staff has made a determination just recently, verbally passed on to TVA, that we will probably be imposing a licensing condition. Other than that, we are satisfied with it.

MR. MARK: You are satisfied except for two or three aspects, I suppose, a matter of procedures, a matter of guarding personnel, a matter of physical equipment, probably those are all different types of things which one would ask about. They are deficient, do you think, in one or another of those fields; that is, the fences or closed-circuit TVs or other such things, mechanical, are not as good as you believe they should be, or arrangements for guards are less good, or their arrangements for scheduling people and not controlling them in and out?

Can you say in which field you have a worry?

MR. KENYON: Sir, all three plants have just recently been reviewed, physical secuity plan, the safeguards contingnency plan and the plant guard training and qualification plan.

All have been accepted with the one exception that the physical security plan, TVA wanted to designate the containment as a nonvital area during the refueling or major maintenance. And the Staff at least at the moment does not believe that it can be done so.

MR. SHEWMON: Is this a rather different request than the request at Sequoyah?

MR. KENYON: TVA has a slightly unique situation in which they present a generic plan and then attach some site-specific addenda. They are in the process -- correct me if I am wrong -- they are in the process of updating their Sequoyah and Browns Ferry plan as well as the Watts Bar plan. But it has been covered on Watts Bar just particularly as part of the normal licensing review.

MR. SHEWMON: Is it your expectation that they will clear off this condition in all three cases?

MR. KENYON: I guess that is up to TVA to answer.

MR. SHEWMON: I do not know what your answer was, but I would have hoped you would say yes or no. The question, as I understood it, was do they already have that sort of a clearance at Sequoyah? Your answer was yes or no?

MR. KENYON: No, sir, they do not.

MR. SHEWMON: Although your answer did say they may be coming in for it, is that what you meant?

MR. KENYON: I believe they have come in for it.

MR. MILLS: Dr. Shewmon, basically what he is talking about and one thing I would like to clear up is when we are talking about this vital area to nonvital area, it was without fuel in the core. We really made this request initially on Browns Ferry when we had a major torus outage out there, and it was very lengthy, several months. We asked to go in and declare a vital area to a nonvital area, do all this work and so forth so that we did not have to go through the security precautions, inspect the plant, inspect all aspects of it in a very similar exact manner as we did before we loaded fuel initially and then declare it a vital area again.

That is what is in our security plan now. That is what I think the Staff is talking about in putting a condition on our license. And in our security plan that is stated, that it will not be allowed at Sequoyah or Watts Bar. We do not know whether that will cause a future problem or not. We are not expecting any long outages with fuel out of the core, but if we did, we are not through pursuing it yet. We would make that request again probably if we had a similar type situation.

MR. MARK: It would seem to me that there is nothing we want to explore here.

MR. MILLS: It has nothing to do with

equipment, personnel, or anything like that.

MR. MARK: Or general positions. You have protection?

MR. MILLS: No, sir.

MR. MARK: It is a matter of procedures under special circumstances, they either change their procedure or the Staff changes its requirements.

MR. SHEWMON: Let me ask a different question oif the Staff. Where are we with regard to separation of vital fucntions? Now, I am thinking more with regard to sabotage prevention. Is there a reg guide on this now? If there is, did it catch this plant? Or did this one come in before that?

MS. ANDENSAM: Dr. Shewmon, the Staff was not really prepared to discuss the physical security plan with the ACRS. We do not have anyone here who is very knowledgable in the physical security planning and what our guidance is.

MR. SHEWMON: I do not know that this has anything to do with physical security. It has to do with plant layout with an idea toward sabotage prevention and protection of vital function. That is design. That does not have anything to do with how much you pay our policemen or what kind of guns they carry. Now, if you are still not prepared to answer it, I guess that is an answer I will have to accept for today at least.

MR. NOVAK: That is correct, sir.

MR. SHEWMON: Would you get back to me on that so I could be a little better informed on this?

MR. EBERSOLE: Mr. Chairman, I want to ask a question. Who in TVA comes up here or gets in the proper channel the classified documents that describe in some detail the actual vulnerability of the typical plants and maybe yours in particular? You know that such documents exist, and people like plant superintendents and others need very badly to read them to understand where the problem really is.

Who in your organization maintains a familiarity with that? For instance, does the safety staff? Are they aware of the security docuents which are not widely distributed?

MR. SHEWMON: It may happen to have been written by people at TVA at the time.

MR. EBERSOLE: In fact, that is the case.

MR. MARK: You are referring to the NRC Sandia reports of 5 or so years ago?

MR. EBERSOLE: That is correct. The NRC Sandia reports is what I am talking about.

MR. SHEWMON: I would still like to see if they tried to do anything on this one.

MR. MILLS: Mr. Ebersole, are you talking in particular perhaps about the Sandia work and so forth?

MR. EBERSOLE: Yes, indeed, I am.

MR. MILLS: Our design organization is aware of the ingredients of that report.

MR. EBERSOLE: By what means do they keep aware? Is there just someone assigned to this or a group? Newt, is it part of your safety assessment to look at security?

MR. CULVER: We look at the security plans involved at Sandia.

MR. EBERSOLE: Do you look at it in all aspects? Does any member of your --

ME. CULVER: We look at he hardware. We look at the training. We look at the adminitrative controls. We look at all parts of the security program.

MR. EBERSOLE: So you are still aware of the Sandia articles?

ME. CULVER: I do not know if the man who looks at security, I do not know if you have seen the Sandia report. I think he has.

MR. SHEWMON: What I am more concerned about is the degree to which the designer is worried about separation so that there were not critical points. What

I am not getting is any particular indication that that was done, although I would hope that with the design organization like TVA they would stay abreast of this some.

MR. PIERCE: Yes, sir. I am Ralph Pierce. We have addressed too the separation of our safety-related systems the train assignments, train A, train B, separation of vital systems. And we have addressed this in our design.

MR. SHEWMON: And is that something the NRC reviews?

MR. PIERCE: The NRC reviews separation.

MR. SHEWMON: Okay. Well, maybe the Staff will find out about it and get back to us.

MR. EBERSOLE: However, is it not true that that separation was not oriented towards the security aspect? It just came about that separation for other reasons gave you a measure of that.

MR. PIERCE: I think that would be a correct assessment of it.

MR. SHEWMON: Whether it is a fire or a guy with a bag of something, either way, it helps.

MR. MARK: Well, the layout of Watts Bar, the plan drawings of where is this and where is that, when was that probably done? 1972?

MR. PIERCE: That is correct. The plant layouts were initially made around 1972-73. You must realize, too, with this vintage plant we do not have as good a separation as we would like to have, and that is apparent with the review we have had with the Staff. Our later plants, we do have much better separation as fas as these circuits go.

MR. MARK: Well, the studies which I believe Jesse was referring to and we are all picking up on, were in fact done in 1966 or 1967. The Sandia reports, those are still classified and discuss what they consider to be sensitive points.

MR. EBERSOLE: Are you correct about that, Dr. Mark? I thought it would be nearer 1976. You said 1966.

MR. MARK: What did I say?

MR. EBERSOLE: 1966.

MR. MARK: I meant 1976-77. That is what I meant. When this plan was done, it was done 5 years before that.

MR. EBERSOLE: Sure. Right.

MR. PIERCE: The physical layout of our plant was more or less settled in the 1972-73 time frame, and by 1976 we essentially had finished off all the concrete work.

MR. MARK: That was the point I was wanting to bring out. The reports which might have changed one's view of the plant layout --

MR. PIERCE: There have been a lot of things that have changed our view on separation.

MR. SHEWMON: Jesse, did you have concluding remarks?

MR. EBERSOLE: One question I could ask Ralph. Ralph, when you lose all the power, do you hold the locks open or the locks shut or should you reveal that?

MR. PIERCE: I do not think we should discuss that.

MR. EBERSOLE: That is my last comment.

MR. MOELLER: I had a few questions I have been patiently waiting to ask.

MR. SHEWMON: Then by all means, begin.

MR. MOELLER: They will not take that long. But at some time I do want an answer to my earlier question on the differences in the flow rate of he pumps.

But let me ask a couple of questions. Who in TVA particularly with respect to Watts Bar keeps up with LERs?

MR. MILLS: Keeps up with categorizing it, Dr. Moeller?

MR. MOELLER: How do they apply to the Watts

Bar plant? How can you use them to your advantage?

MR. MILLS: I will let Bill Cottle, the plant superintendent, answer that.

MR. COTTLE: We have an experienced review group in our division central office in Chattanooga who receives the inputs on LERs from the various publications, NSAC, INPO, information from NRC, et cetera. They do a preliminary type screening evaluation on LERs as well as other experience inputs, and then we get two kinds of outputs from that process at the plant. One would be a simple summation that this type item was looked at, and from their basic knowledge, if it is very clear-cut, it is clearly not applicable to Watts Bar.

The other type of information we would receive from them would be in the way of a directed assignment either assigning the item to the plant or its apparent applicability and examination. Or if it is for more hardware generic type item, we receive information if it had been assigned to a particular down section electrical unit section.

MR. MOELLER: So there is a specific person who makes these assignments?

MR. COTTLE: Yes, sir.

MR. MOELLER: And for instance, at Sequoyah as well as at D.C. Cook you had problems with the freezing of the doors in your ice condenser. What have you done at Watts Bar to prevent the problem at Sequoyah?

MR. COTTLE: I might point out one other thing: We get direct copies of all Sequoyah reports at the plant. The freezing problem at Sequoyah has been primarily one of a heat tracing failure on the individual air handling defrost brains, a tracing failure and, to some extent, inadequate insulation on those brains.

We are in the process, or will be in the process. We have had the ice condensers down in temperature. We will have a little better inspection process on initially placing that in service and I think treat it with a little more care procedurewise than Sequoyah did in the nuclear operation. We will come to the same type of upgrading as they are in the process of doing at Sequoyah now.

MR. MOELLER: Did you make changes at Watts Bar to try to reduce the ice loss, the rate of loss of ice?

MR. COTTLE: We have received -- we are working with AEP and Duke Power Company on overall icing problems. For an example, we just received within the last month the latest comprehensive report on Sequoyah based on their heat load studies and ice sublimation rates. We factor it primarily into our preventive maintenance program and our daily shift check.

MR. MOELLER: You have not made physical changes?

MR. COTTLE: We have made some physical changes in change-outs primarily on what I call solonoid handling valves on the air handling units themselves. Most of the changes have been in either maintenance practices or operational checks. They have not been physical equipment changes.

MR. MOELLER: I note in the SER that you are going to use gaseous and particulate rad monitors to measure the leakage or one of the ways to measure the leakage from the reactor coolant pressure boundary. And that I think I could understand.

But you also for the SER says on page 9-17 that you are going to by analyzing the air in the containment you can determine the extent of core damage. Could you tell me how you do that? This is post-accident, obviously, but maybe if you cannot, the Staff might tell us how you are going to do it.

MR. COTTLE: I do not have the details with me right now.

MR. MOELLER: Could the Staff comment on that on page 9-17? How do you use air analyses within containment to determine the extent of core damage?

MR. KENYON: I am afraid we do not have anybody from the Staff to answer that.

MR. SHEWMON: Just out of curiosity, is there anybody else from the Staff except you three here today?

(No response.)

MR. KERR: I bet they do not have anybody on the Staff anywhere that could answer that question.

(Laughter.)

MR. KERR: So I do not think it would help.

MR. MOELLER: Okay. Another item. On page 9-12 they quote you as assuring us that there are no cross-connections between the potable or sanitary water systems and potentially radioactive water within your plant. And that is fine. I assume it is true and so forth.

However, I also looked for similar assurance that there is no connections, cross-connections, between service air systems and those reserved for breathing. Can you give me such an assurance? Because we have had problems such as that at some other plants.

MR. COTTLE: For the most part, our systems used for breathing air are based on using Scott Air Pack type equipment where we actually have a refilling station located separately in the plant that we use.

MR. MOELLER: But now many plants or some plants have in the control room centralized source manifold, as I recall.

MR. COTTLE: Our experience with using those manifolds are that they are so large and so unwieldly that we have found them pretty impractical to use.

MR. MOELLER: What does Watts Bar have?

MR. COTTLE: We use Scott Air Packs. We have capability for manifolds, but it is not our intention to use them.

MR. MOELLER: Can you assure me, or has someone checked to be positive that the air supply for the manifold can in no way get cross-connected so that radioactive material could be released?

MR. PIERCE: I think I can answer that. I think I can answer that with a reasonable assurance, but we will check it and make sure. But I am fairly certain that we do not have any cross-connections of those type.

MR. MOELLER: The ice condenser volume at Watts Bar is 100 cubic feet less than for Sequoyah. Why? I mean at least that is what the SER says.

MR. KERR: Watts Bar is further north.

(Laughter.)

MR. MOELLER: Everything is exactly the same except the ice condenser which is 110,400 cubic feet instead of 110,500.

MR. ETHERINGTON: It sounds like thermal expansion.

(Laughter.)

MR. MOELLER: Yet the rate of the ice is the same? It is not a typographical error because it carries through to the totals and so forth.

MR. COTTLE: I am sure we do not have anyone who could address that right now. I have no idea.

MR. MOELLER: One last question because I know the committee wants to get on to other things. The air cleaning systems for the reactor building and the control room emergency system are credited with 95 percent iodine removal.

Now, the emergency gas treatment system and the aux building system, they are credited for 99 percent radioiodine removal. Can you tell me why the difference or what that is based on? That is on page 6-36 of the SER.

Or could I ask the Staff, do you have different requirements for those four systems?

MR. CULVER: If I remember correctly, the emergency gas treatment system has two banks of carbon trays in series. The other systems only have one.

MR. MOELLER: Then is the requirement for the control room less than, say, for the standby gas treatment system, being the 95 versus 99?

MR. CULVER: The two banks were put in to reduce as much as possible the effluent going out, and it was not necessary to meet the requirement in the control building.

MR. MOELLER: Okay. Can the Staff tell me if that is just some tech spec requirement that there is a difference there?

MS. ANDENSAM: Dr. Moeller, Eleanor Adensam from the Staff. Could you repeat your question again?

MR. MOELLER: For the reactor building and the control room emergency air cleaning system, they stipulate a 95 percent radioiodine removal. Then for the emergency gas treatement system and the aux building system they specify 99 percent iodine removal. I just wondered. I wanted to be informed.

MS. ANDENSAM: As I understand it, based on my past experience, which is getting a little old and rusty, the efficiencies on the filter systems are based on what the filter system looks like. Then the dose analyses are done after that to determine whether or not that is sufficient.

I know past practice has been that if that is not, then we go back to the applicant and require them to modify their filter system to improve filter efficiency. The requirement, to the best of my knowledge, is not on the efficiency of the filters so much as the dose consequences.

MR. MOELLER: Well, if someone could obtain answers to some of my questions and send the committee a written note, it would be helpful. It need not be too formal, but I would like to have answers.

I will close with this very interesting sentence in the SER on page 6-6, where they found that the rate of ice loss in the ice condenser at Sequoyah was too high. They resolved it in the following manner: "This high loss rate necessitated a reanalysis of the DBA to permit a reduction in the requirement of the plant's tech specs."

That is how you administratively repair the problem.

MR. SHEWMON: Have you left your last question now?

MR. EBERSOLE: Mr. Chairman, I will turn it back to you. I would expect TVA would like to hear an observation from the committee.

MR. SHEWMON: Okay. Is there someone here who feels we could not or should not write a letter on Watts Bar at this meeting?

(No response.)

MR. SHEWMON: Then I presume we will. Thank you.

I would like at this point to take a short break, 6-minute break, and clear the room. Then we will come back and read the letter.

(Thereupon, at 5:25 p.m., the Subcommittee was adjourned.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

____

in the matter of: ACRS/268th General Meeting

Date of Proceeding: August 13, 1982

Docket Number: ____________________

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Jane N. Beach

Official Reporter (Typed)

Jane N. Beach

Official Reporter (Signature)

L. M. MILLS ACRS SUBCOMMITTEE PRESENTATION

I am Larry Mills, Manager of Nuclear Licensing for TVA.

The Watts Bar Nuclear Plant, which consists of two identical units, was constructed and will be operated by TVA. These units employ pressurized water reactors furnished by Westinghouse. Each unit will operate at 3411 thermal megawatts with an electrical output of 1218 megawatts. Completion of these two units will provide TVA with 7 operating nuclear units, including those currently in operation at Sequoyah and Browns Ferry Nuclear Plants, with a nuclear generating capability of approximately 8 million kilowatts of electrical power.

The Watts Bar plant site is located in East Tennessee (Slide 1). This slide also shows the locations of Sequoyah and Browns Ferry. Watts Bar is approximately 31 miles North-North East of the Sequoyah Nuclear Plant and 45 miles North-North East of Chattanooga, Tennessee. The plant site consists of a tract of approximately 1770 acres in Rhea County on the West bank of the Tennessee River. The 1770 acre reservation is owned by the United States and is in the custody of TVA. Also located within the reservation are the Watts Bar Hydro-Electric Plant and Watts Bar Steam Plant.

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The contract for Watts Bar NSSS was awarded to Westinghouse on August 27, 1970, and a construction permit was issued on January 23, 1973. TVA submitted a Final Safety Analysis Report on July 1, 1976 in support of an application for an operating license. This application was docketed by the NRC on October 4, 1976. The construction completion schedule of August 1983 for unit 1 and August 1984 for unit 2 remains the same as we specified during the last ACRS Subcommittee meeting in April. This schedule includes plant modifications as a result of TMI related requirements and modifications resulting from the Sequoyah licensing review. These modifications were included at Watts Bar because of the similarities between Sequoyah and Watts Bar.

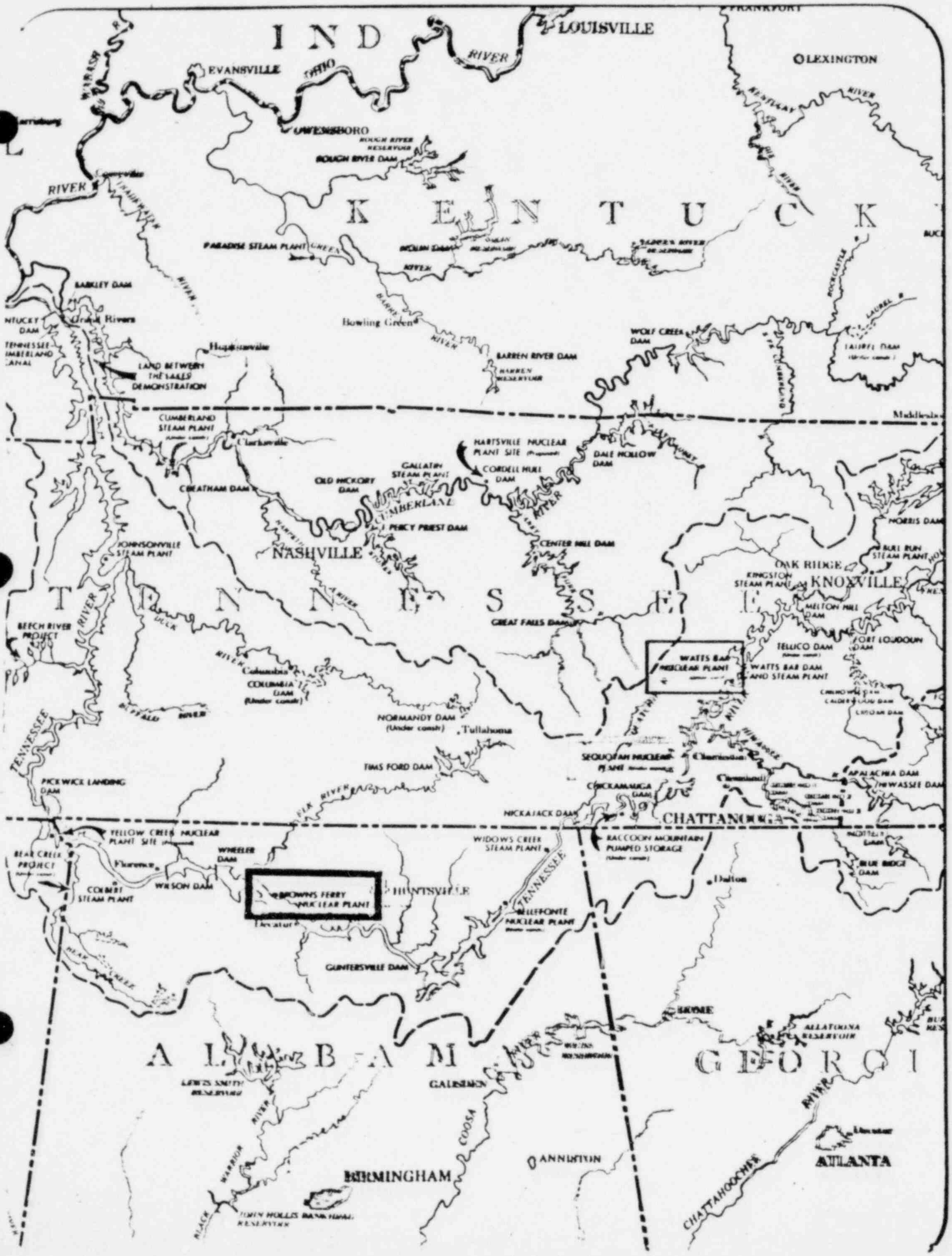
Later today we will be providing more specific information on TVA organizations and training, but briefly: Staffing of the plant operating personnel was initiated with the appointment of a Plant Superintendent in July 1976. Since that time, the staff has grown to a present level of approximately 190 engineering and maintenance employees, 65 administrative and support employees, 7 senior reactor operators, 4 reactor operators and 100 auxiliary unit operators. By fuel loading for unit 1, 2 more senior reactor operators and 12 more reactor operators will be added.

The on-site operations personnel are currently involved in becoming familiar with plant systems and equipment and operating various systems and equipment during the plant's construction test program.

Total plant staff for operation of both units will be approximately 560 people, composed of about 130 operators, 315 engineering and maintenance personnel, 120 administrative and support employees. The remainder of the plant operations personnnel are being trained by the Nuclear Training Branch in the Division of Nuclear Power. TVA's training program conforms to the requirements set forth in ANSI N18.1.

When we receive a license to operate Watts Bar, TVA will have accumulated approximately 31 reactor years of operating experience from operation of Sequoyah and Browns Ferry Nuclear Plants.

A discussion of the Sequoyah-Watts Bar comparison will be provided by David Ormsby.


IND
LOUISVILLE
EVANSVILLE
OHIO
RIVER
LEXINGTON
OWENSBORO
ROUGH RIVER DAM
KENTUCK
PARADISE STEAM PLANT
BARKLEY DAM
Grand Rivers
Bowling Green
WOLF CREEK DAM
BARREN RIVER DAM
LAUREL DAM
LAND BETWEEN THE LAKES DEMONSTRATION
Hopkinsville
CUMBERLAND STEAM PLANT
Clarksville
HARTSVILLE NUCLEAR PLANT SITE
GALLATIN STEAM PLANT
OLD HICKORY DAM
CORDELL HULL DAM
DALE HOLLOW DAM
CHEATHAM DAM
CUMBERLAND
J. PERCY PRIEST DAM
NORRIS DAM
BULL RUN STEAM PLANT
CENTER HILL DAM
NASHVILLE
JOHNSONVILLE STEAM PLANT
OAK RIDGE
KINGSTON STEAM PLANT
KNOXVILLE
TENNESSEE
MELTON HILL DAM
GREAT FALLS DAM
BEECH RIVER PROJECT
TELLICO DAM
FORT LOUDOUN
WATTS BAR NUCLEAR PLANT
WATTS BAR DAM AND STEAM PLANT
Columbia
COLUMBIA DAM (Under constr.)
NORMANDY DAM (Under constr.)
Tullahoma
SEQUOYAH NUCLEAR PLANT
TIMS FORD DAM
APALACHIA DAM
HIWASSEE DAM
PICKWICK LANDING DAM
ELK RIVER
CHICKAMAUGA DAM
HICKAJACK DAM
CHATTANOOGA
YELLOW CREEK NUCLEAR PLANT SITE
WIDOWS CREEK STEAM PLANT
RACCOON MOUNTAIN PUMPED STORAGE
WHEELER DAM
BLUE RIDGE DAM
BEAR CREEK PROJECT
Florence
Dalton
COLBERT STEAM PLANT
WILSON DAM
BROWNS FERRY NUCLEAR PLANT
HUNTSVILLE
BELLEFONTE NUCLEAR PLANT
Decatur
GUNTERSVILLE DAM
ROME
ALLATOONA RESERVOIR
ALABAM
GEORGI
GADSDEN
COOSA
ANNISTON
ATLANTA
BIRMINGHAM
CHATTAHOOCHEE
RIVER

SEQUOYAH-WATTS BAR DESIGN PRESENTATION

DAVID P. ORMSBY

The Watts Bar and Sequoyah Nuclear Plants are similar in most respects. The cores are similar and each plant utilizes a free-standing steel vessel with an ice condenser and a reinforced concrete shield building. The plant's mechanical systems, containment systems, emergency core cooling systems, instrument and control systems, electric power systems, auxiliary systems, radioactive waste systems, and steam and power conversion systems are also very similar in design and materials. Most differences are either site specific or are the result of the fact that two to three years separate the design phases of Sequoyah and Watts Bar. Although the design philosophy was the same, there were some instances when more current technology was used for Watts Bar.

This vugraph provides a table showing design differences between Watts Bar and Sequoyah. As I stated previously, there is a two-to-three year difference in the design phases for Watts Bar and Sequoyah. During that time, some design changes were made for Watts Bar which increased the efficiency of the system. There are also some instances where equipment was upgraded or provided by a different manufacturer. Some examples are as follows:

Reactor Coolant Pumps (RCP)

SQN - 6,000 Horsepower pump motor
WBN - 7,000 Horsepower pump motor

Copy

This difference results in an increased primary system flow rate and therefore, difference in maximum heat flux. Because of the increase in RCS flow, there is a need for greater PORV relieving capacity for load mismatch and to accommodate a 50% load rejection. The PORVs for Watts Bar are provided by a different manufacturer than those for Sequoyah.

Electrical Output

You will also note that there is an increase in turbine generator and gross electrical output. These differences are due to increased equipment and system efficiency including different steam generators.

Steam Generators

SQN - Model 51
WBN - Model D3

The steam generators for the Watts Bar and Sequoyah units are similar vertical shell and U-tube evaporators with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering the leaving through the nozzles located in the hemispherical bottom head of the steam generators. Steam is generated in the shell side and flows upward through the moisture separators to the outlet nozzle at the top of the vessel.

The major difference between the steam generators is that Watts Bar utilizes Model D3 series generators and Sequoyah utilizes Model 51 series generators. Inherent with this model difference are the associated differences in the feedwater entrances (i.e., preheater for Model D3 series utilized in Watts Bar and feedring for Model 51 series utilized in Sequoyah), surface areas, and U-tube design. These differences result in a secondary steam flow rate.

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WATTS BAR - SEQUOYAH DESIGN FEATURES

DESIGN FEATURES	WATTS BAR	SEQUOYAH
Containment design pressure, psig	15.0	12.0
Gross electrical output, MWe	1,171	1,128 (Unit 1) 1,148 (Unit 2)
Total reactor coolant flow-rate, lb/hr	140,300,00	138,100,00
Secondary steam flow-rate, lb/hr	15,140,000	14,250,000
Reactor vessel minimum cladding thickness, in.	1/8	5/32
Maximum heat flux Btu/hr-ft^2	440,300	424,600
Reactor coolant pump flowrate, gpm	101,000	88,500
Steam generator type	W Model D-3	W Model 51
PORV-relieving capacity, lb/hr	210,000	179,000
Turbine generator output, kW	1,218,000	1,185,000
Secondary steam pressure, psia	1,000	832
Secondary steam temperature, ^{o}F	544	522

Status of Watts Bar Review

- SER Issued - June 1982
- SER Contained
  - 17 Open Items
  - 41 Confirmatory Items
  - 37 License Conditions
    - 16 Non-TMI
    - 21 TMI
- SSER Scheduled to be Issued - September 1982

Kenyon
T2

Open Issues Remaining

(1) Potential for liquefaction beneath ERCW pipelines and Class 1E electrical conduit

(2) Buckling loads on Class 2 and 3 supports

(3) Preservice and inservice pump and valve test program

(4) Seismic and environmental qualification of equipment

(5) Preservice and inservice inspection program

(6) Pressure-temperature limits for Unit 2

(7) Model D-3 steam generator preheater tube degradation

(8) BTP-CSB 6-4 and containment isolation dependability (II.E.4.2)

(9) H_2 analysis review

(10) Safety valve sizing analysis (WCAP-7769)

(11) Compliance of proposed design change to the offsite power system to GDC 17 and 18

(12) Fire Protection Program

(13) Quality classification of diesel generator auxiliary system piping and components

(14) Diesel generator auxiliary systems design deficiencies

(15) Boron dilution event

(16) Q-List

Issue Closed Since SER Issuance

(1) Physical Security Plan

9 POSITION: POWER PLANT SUPERINTENDENT

NAME: WILLIAM T. COTTLE

EXPERIENCE SUMMARY:

A. NAVAL NUCLEAR PROPULSION PROGRAM — 4 YEARS

B. NUCLEAR EXPERIENCE

<u>FARLEY NUCLEAR PLANT</u>	5½ YEARS
PREOPERATIONAL TEST ENGINEER	
STARTUP ENGINEER	
ASSISTANT OPERATIONS SUPERVISOR	
OPERATIONS SUPERVISOR	
OPERATING SUPERINTENDENT	
LICENSED AS SRO	
<u>SEQUOYAH NUCLEAR PLANT</u>	1 YEAR
COMPLIANCE SUPERVISOR	
ASSISTANT PLANT SUPERINTENDENT	
<u>WATTS BAR NUCLEAR PLANT</u>	
PLANT SUPERINTENDENT	*1 YEAR
<u>NUCLEAR REGULATORY COMMISSION</u>	2 YEARS
PROJECT INSPECTOR	
SENIOR RESIDENT INSPECTOR	
	13½ YEARS

*6 MONTHS AS WBN PLANT SUPERINTENDENT AND SQN ASSISTANT PLANT SUPERINTENDENT

T2, T3

POSITION: ASSISTANT PLANT SUPERINTENDENT (OPERATIONS & ENGINEERING)

NAME: ROBERT L. LEWIS

EXPERIENCE SUMMARY:

A. FOSSIL EXPERIENCE — 10 YEARS

AUXILIARY OPERATOR
STUDENT OPERATOR
ASSISTANT UNIT OPERATOR
UNIT OPERATOR
SHIFT ENGINEER

B. NUCLEAR EXPERIENCE

<u>EGCR</u> — 6 YEARS

TRAINING & PREOPERATIONAL DUTIES
SHIFT ENGINEER

<u>BROWNS FERRY NUCLEAR PLANT</u> — 2 YEARS

SHIFT ENGINEER

<u>SEQUOYAH NUCLEAR PLANT</u> — 6 YEARS

ASSISTANT OPERATIONS SUPERVISOR

LICENSED AS SRO

<u>WATTS BAR NUCLEAR PLANT</u> — 6 YEARS

OPERATIONS SUPERVISOR
ASSISTANT PLANT SUPERINTENDENT

10 YEARS - FOSSIL
20 YEARS - NUCLEAR

POSITION: ASSISTANT PLANT SUPERINTENDENT (MAINTENANCE)

NAME: EDDIE R. ENNIS

EXPERIENCE SUMMARY:

A. FOSSIL EXPERIENCE — 13½ YEARS

ENGINEER TRAINEE
POWER PLANT MAINTENANCE SPECIALIST
ASSISTANT MECHANICAL MAINTENANCE SUPERVISOR
MECHANICAL MAINTENANCE SUPERVISOR

B. NUCLEAR EXPERIENCE

WATTS BAR NUCLEAR PLANT — 5 YEARS

MECHANICAL MAINTENANCE SUPERVISOR
ASSISTANT PLANT SUPERINTENDENT

COMPLETED COLD LICENSE CERTIFICATION COURSE ON SEQUOYAH NUCLEAR PLANT

13½ YEARS - FOSSIL
5 YEARS - NUCLEAR

Position: Field Services Supervisor

Name: Richard H. Ector

Experience Summary:

A.	Naval Nuclear Propulsion Program	5 years
B.	Nuclear Experience	
	<u>Clinton Nuclear Station</u> Maintenance Engineer	1 year
	<u>Browns Ferry Nuclear Plant</u> Outage Planner	1 year
	<u>Sequoyah Nuclear Plant</u> Assistant Outage Director	3 years
	<u>Watts Bar Nuclear Plant</u> Field Services Supervisor	1 year
		11 years

POSITION: OPERATIONS SUPERVISOR

NAME: GUY T. DENTON

EXPERIENCE SUMMARY:

A. FOSSIL EXPERIENCE — 15 YEARS

MATERIAL TESTER

STUDENT OPERATOR

ASSISTANT UNIT OPERATOR

UNIT OPERATOR

ASSISTANT SHIFT ENGINEER

B. NUCLEAR EXPERIENCE

<u>BROWNS FERRY NUCLEAR PLANT</u> — 8½ YEARS

ASSISTANT SHIFT ENGINEER

SHIFT ENGINEER

LICENSED AS SRO

<u>WATTS BAR NUCLEAR PLANT</u> — 5½ YEARS

ASSISTANT OPERATIONS SUPERVISOR

OPERATIONS SUPERVISOR

15 YEARS - FOSSIL

14 YEARS - NUCLEAR

POSITION: ASSISTANT OPERATIONS SUPERVISOR

NAME: REDFORD NORMAN, JR.

EXPERIENCE SUMMARY:

A. FOSSIL EXPERIENCE — 7 YEARS

STUDENT OPERATOR
ASSISTANT UNIT OPERATOR
UNIT OPERATOR
ASSISTANT SHIFT ENGINEER

B. NUCLEAR EXPERIENCE

SEQUOYAH NUCLEAR PLANT — 4 YEARS

ASSISTANT SHIFT ENGINEER
STUDENT INSTRUCTOR

WATTS BAR NUCLEAR PLANT — 6 YEARS

SHIFT ENGINEER
ASSISTANT OPERATIONS SUPERVISOR

7 YEARS - FOSSIL
10 YEARS - NUCLEAR

POSITION: ASSISTANT OPERATIONS SUPERVISOR

NAME: RICHARD E. YARBROUGH, JR.

EXPERIENCE SUMMARY:

A. FOSSIL EXPERIENCE — 20 YEARS

STUDENT OPERATOR

ASSISTANT UNIT OPERATOR

UNIT OPERATOR

ASSISTANT SHIFT ENGINEER

SHIFT ENGINEER

B. NUCLEAR EXPERIENCE

<u>SEQUOYAH NUCLEAR PLANT</u> — 10 YEARS

SHIFT ENGINEER

LICENSED AS SRO

<u>WATTS BAR NUCLEAR PLANT</u> — 3 MONTHS

ASSISTANT OPERATIONS SUPERVISOR

20 YEARS - FOSSIL

10 YEARS - NUCLEAR

POSITION: SPECIAL PROJECTS SUPERVISOR

NAME: DANIEL J. RECORD, SR.

EXPERIENCE SUMMARY:

A. FOSSIL EXPERIENCE — 19 YEARS

AUXILIARY OPERATOR

STUDENT OPERATOR

ASSISTANT UNIT OPERATOR

UNIT OPERATOR

ASSISTANT SHIFT ENGINEER

B. NUCLEAR EXPERIENCE

SEQUOYAH NUCLEAR PLANT — 10½ YEARS

SHIFT ENGINEER

SHIFT ENGINEER TRAINING

ASSISTANT OPERATIONS SUPERVISOR

OPERATIONS SUPERVISOR

LICENSED AS SRO

WATTS BAR NUCLEAR PLANT — 6 MONTHS

SPECIAL PROJECTS SUPERVISOR

19 YEARS - FOSSIL

11 YEARS - NUCLEAR

SHIFT ENGINEERS

	Fossil	Other Nuclear	WBNP	Comments
Robert E. Bradley	7 yrs.	2 yrs.	5 yrs.	None
C. Richard Cook	6 yrs.	5 yrs.	5 yrs.	SQN/3747-1
Grady R. Davis	3 yrs.	4 yrs.	4 yrs.	BFN/OP 4286
Edward O. Gambill	3 yrs.	2 yrs.	5 yrs.	Nuclear Navy-6 yrs.
David S. King	8 yrs.	2 yrs.	5 yrs.	None
Lacy Pauley	5 yrs.	5 yrs.	5 yrs.	SQN/SOP 3748-1
W. Douglas Stevens	3 yrs.	7 yrs.	5 yrs.	BFN/SOP 2807
Harry J. Voiles	7 yrs.	4 yrs.	4 yrs.	SQN/SOP 3858
Carl E. Wallace	8 yrs.	2 yrs.	5 yrs.	SQN/SOP 3880
Louis Jenkins	28 yrs.	----	3 mos.	None

T3

ASSISTANT SHIFT ENGINEERS

	Fossil	Other Nuclear	WBNP	Comments
Glenn T. Carver	5 yrs.	2 yrs.	5 yrs.	None
M. E. Hastings	1 yr.	4 yrs.	5 yrs.	None
Lewis E. Howard	1 yr.	4 yrs.	5 yrs.	None
James C. Johnson	4 yrs.	1 yr.	4 yrs.	None
John A. Justus	----	1 yr.	4 yrs.	SQN/OP 5405
Richard L. Kimbrough	----	----	4 yrs.	None
J. Patrick McGinnis	----	4 yrs.	3 yrs.	BFN/OP 4883
David L. McConnel	4 yrs.	1 yr.	5 yrs.	None
William V. Rusbridge	4 yrs.	6 yrs.	4 yrs.	None
Wesley T. Stockdale	8 yrs.	----	2 yrs.	None
Ralph E. Schmook	----	1 yr.	4 yrs.	SQN/OP 5405

UNIT OPERATORS

	Fossil	Other Nuclear	WBNP	Comments
Steve M. Baker	----	----	4 yrs.	None
Sheilah D. Baker	----	----	3 yrs.	None
D. W. Barker	----	----	4 yrs.	None
J. M. Childers	----	4 yrs.	2 mos.	SQN/OP 5782
W. R. Collins	----	----	4 yrs.	None
C. M. deBlonk	----	----	3 yrs.	None
R. H. Evans	----	----	3 yrs.	None
R. F. Gallaher	3 yrs.	----	5 yrs.	None
R. W. Ingle	----	----	3 yrs.	None
M. H. Muirhead	----	----	3 yrs.	None
T. L. Newman	----	----	4 yrs.	None
R. S. Scarlett	----	----	3 yrs.	None
A. E. Shultz	----	----	4 yrs.	None
J. W. Smith	----	----	3 yrs.	None
G. D. Stone	----	----	5 yrs.	None
T. E. Tuckier	----	1 yr.	4 yrs.	SQN/OP 5781
T. D. Wallace	----	----	3 yrs.	None

POSITION: DIRECTOR OF NUCLEAR POWER

NAME: HARRY J. GREEN

EXPERIENCE SUMMARY:

A.	NAVAL NUCLEAR PROPULSION PROGRAM	8 YEARS
B.	FOSSIL EXPERIENCE	2 YEARS
	ASSISTANT PLANT SUPERINTENDENT	
C.	NUCLEAR EXPERIENCE	
	<u>EGCR</u>	5 YEARS
	ASSISTANT TO NUCLEAR PLANT SUPERINTENDENT	
	NUCLEAR PLANT SUPERINTENDENT	
	<u>BROWNS FERRY NUCLEAR PLANT</u>	9 YEARS
	ASSISTANT PLANT SUPERINTENDENT	
	POWER PLANT SUPERINTENDENT DIRECTED RECOVERY FROM CABLE FIRE	
	<u>DIVISION OF POWER PRODUCTION</u>	2 YEARS
	CHIEF, NUCLEAR GENERATION BRANCH	
	<u>POWER MANAGER'S OFFICE</u>	1 YEAR
	ASSISTANT MANAGER OF POWER OPERATIONS	
	DIRECTOR OF NUCLEAR POWER	2 YEARS

2 YEARS - FOSSIL

27 YEARS - NUCLEAR

POSITION: ASSISTANT DIRECTOR, NUCLEAR POWER

NAME: JAMES A. COFFEY

EXPERIENCE SUMMARY:

A. FOSSIL EXPERIENCE — 8 YEARS

ENGINEERING AIDE

MECHANICAL ENGINEER

B. NUCLEAR EXPERIENCE

EGCR — 8 YEARS

NUCLEAR DEVELOPMENT ENGINEER

MECHANICAL ENGINEER

BROWNS FERRY NUCLEAR PLANT — 3 YEARS

ASSISTANT PLANT RESULTS SUPERVISOR

DIVISION OF POWER PRODUCTION — 3 YEARS

NUCLEAR ENGINEER

NUCLEAR GENERATION BRANCH — 5 YEARS

ASSISTANT TO THE CHIEF

ASSISTANT CHIEF

DIVISION OF NUCLEAR POWER — 3 YEARS

ASSISTANT DIRECTOR (MAINTENANCE & ENGINEERING)

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8 YEARS - FOSSIL

22 YEARS - NUCLEAR

POSITION: MANAGER, NUCLEAR PRODUCTION

NAME: TOMMY G. CAMPBELL

EXPERIENCE SUMMARY:

A. FOSSIL EXPERIENCE — 6 YEARS

ENGINEERING AIDE

B. NUCLEAR EXPERIENCE

<u>BROWNS FERRY NUCLEAR PLANT</u> — 8 YEARS

MECHANICAL ENGINEER (CONST)

OUTAGE DIRECTOR

NUCLEAR ENGINEER

<u>DIVISION OF NUCLEAR POWER</u> — 3 YEARS

CHIEF, OUTAGE MANAGEMENT BRANCH

MANAGER, NUCLEAR PRODUCTION

---

6 YEARS - FOSSIL

11 YEARS - NUCLEAR

POSITION: ASSISTANT NUCLEAR PRODUCTION MANAGER

NAME: HERBERT L. ABERCROMBIE

EXPERIENCE SUMMARY:

A. FOSSIL EXPERIENCE — 17 YEARS

OPERATOR TRAINING
STUDENT OPERATOR
ASSISTANT UNIT OPERATOR
UNIT OPERATOR
ASSISTANT SHIFT ENGINEER
SHIFT ENGINEER
PERSONNEL OFFICER (GIVING EXAMS)
POWER PLANT OPERATIONS SPECIALIST

B. NUCLEAR EXPERIENCE

SEQUOYAH NUCLEAR PLANT — 7 YEARS
OPERATIONS SUPERVISOR

BROWNS FERRY NUCLEAR PLANT — 4 YEARS
ASSISTANT PLANT SUPERINTENDENT
PLANT SUPERINTENDENT

DIVISION OF NUCLEAR POWER — 1 YEAR
ASSISTANT NUCLEAR PRODUCTION MANAGER

17 YEARS - FOSSIL
12 YEARS - NUCLEAR

POSITION: CHIEF, MECHANICAL BRANCH

NAME: THOMAS F. ZIEGLER

EXPERIENCE SUMMARY:

A. FOSSIL EXPERIENCE — 2 YEARS

MECHANICAL ENGINEER TRAINEE

B. NUCLEAR EXPERIENCE

BROWNS FERRY NUCLEAR PLANT — 1 YEAR

ASSISTANT OUTAGE DIRECTOR

DIVISION OF POWER PRODUCTION — 6 YEARS

MECHANICAL ENGINEER

SUPERVISOR, REACTOR AND AUXILIARY SECTION

DIVISION OF NUCLEAR POWER — 3 YEARS

CHIEF, NUCLEAR MAINTENANCE BRANCH

CHIEF, MECHANICAL BRANCH

---

2 YEARS - FOSSIL

10 YEARS - NUCLEAR

POSITION: CHIEF, REACTOR ENGINEERING BRANCH

NAME: THOMAS D. KNIGHT

EXPERIENCE SUMMARY:

A. NUCLEAR EXPERIENCE

<u>WESTINGHOUSE - ELECTRIC CORPORATION</u>	6 YEARS
NUCLEAR ENGINEER	
<u>TVA - DIVISION OF POWER PRODUCTION</u>	7 YEARS
NUCLEAR ENGINEER	
SUPERVISOR, NUCLEAR SECTION	
SUPERVISOR, REACTOR ENGINEERING STAFF	
<u>DIVISION OF NUCLEAR POWER</u>	3 YEARS
CHIEF, REACTOR ENGINEERING BRANCH	
	16 YEARS - NUCLEAR

POSITION: CHIEF, ELECTRICAL AND INSTRUMENT AND CONTROLS BRANCH

NAME: HUBERT B. BOUNDS

EXPERIENCE SUMMARY:

A. U. S. NAVY — 4 YEARS
ELECTRONICS TECHNICIAN (RADAR)

B. HAYES INTERNATIONAL CORP - SPERRY RAND — 7 YEARS
ENGINEERING TECHNICIAN AND PROGRAMMER

C. TVA - ENGINEERING DESIGN — 8 YEARS
ELECTRICAL ENGINEER

D. NUCLEAR EXPERIENCE

<u>DIVISION OF NUCLEAR POWER</u> — 2½ YEARS

ACTING CHIEF, NUCLEAR MAINTENANCE BRANCH

SUPERVISOR, ELECTRICAL EQUIPMENT GROUP

CHIEF, ELECTRICAL AND INSTRUMENT AND CONTROLS BRANCH

8 YEARS - DESIGN

2½ YEARS - NUCLEAR

Position: Senior Engineer

Name: Richard A. Sessoms

Experience Summary:

A. Nuclear Experience

U. S. Navy	2 years
Nuclear maintenance, repair and research	
Florida Power & Light	4 years
Plant Engineer	
TVA - Division of Power Production	7 years
Instrument Engineer	
Supervisor, Controls Section	
Division of Nuclear Power	3 years
Chief, Controls and Test Branch	
Chief, Technical Services Branch	
Senior Engineer	
	16 years - Nuclear

NUCLEAR SAFETY REVIEW STAFF

DISCUSSION OF PROGRAM

Presented By
H. N. Culver
Director, NSRS

8/13/82

T4

FOLLOWING TMI, TVA ESTABLISHED A TASK FORCE TO IDENTIFY ACTIONS THAT COULD BE TAKEN TO FURTHER STRENGTHEN AND IMPROVE THEIR OVERALL NUCLEAR PROGRAMS.

YOU ARE AWARE OF THE FINDINGS OF THIS TASK FORCE . . . ACTIONS RELATING TO DESIGN, OPERATIONS, ORGANIZATION.

ONE OF THE ACTIONS WAS TO CREATE THE TVA NUCLEAR SAFETY REVIEW STAFF. MY REMARKS ARE DIRECTED AT PROVIDING YOU WITH A BRIEF DESCRIPTION OF THE EXISTING STAFF AND WHAT WE HAVE BEEN DOING AND WHAT WE WILL BE DOING IN THE FUTURE.

THE FUNCTIONS OF THE STAFF ARE SHOWN IN THE FIRST VIEWGRAPH

VIEWGRAPH #1

YOU CAN SEE THE STAFF HAS THE RESPONSIBILITY TO BECOME INVOLVED WITH ALL ASPECTS OF THE TVA NUCLEAR PROGRAM. A LITTLE LATER IN THE PRESENTATION YOU WILL SEE WE DO IN FACT GET INVOLVED WITH MOST OF THESE ACTIVITIES.

AS A POINT OF CLARIFICATION, NSRS DOES NOT REPLACE ANY OF THE LINE FUNCTIONS IN TVA. RATHER, IT IS AN ADDITIONAL OVERVIEW FUNCTION BEYOND REGULATORY REQUIREMENTS --- OR SOME THINK OF US AS ANOTHER GROUP SIMILAR TO I&E, ONLY OPERATING WITHIN TVA.

THE NSRS DURING THE PAST 2-1/2 YEARS HAS BEEN INVOLVED WITH MANY OF THE FUNCTIONS I HAVE SHOWN. IN ORDER TO ACCOMPLISH THESE TASKS THE STAFF WAS ORGANIZED WITH TWO OBJECTIVES--INDEPENDANCE AND BROAD EXPERTISE.

SOME FACTS REGARDING NSRS ARE SHOWN IN THE NEXT VIEWGRAPH.

VIEWGRAPH #2

NOW REGARDING WHAT OUR INVOLVEMENT HAS BEEN.

VIEWGRAPH #3

I WOULD LIKE TO AMPLIFY A LITTLE REGARDING OUR MANAGEMENT REVIEWS. AS INDICATED, WE HAVE MADE SEVERAL OF THESE. THE NEXT 2 VIEWGRAPHS INDICATE WHAT THESE CONSIST OF

VIEWGRAPH #4

VIEWGRAPH #5

I WANT TO NOW INDICATE WHAT WE DO WITH OUR REPORTS AND HOW WE FIT IN WITH THE TVA ORGANIZATION.

ALL OF OUR REPORTS DEALING WITH THE LINE ORGANIZATION ARE PROVIDED TO LINE MANAGEMENT INDICATING OUR FINDINGS AND RECOMMENDATIONS. THESE REPORTS REQUIRE A RESPONSE--USUALLY IN 30 DAYS. THESE ITEMS ARE FOLLOWED UNTIL THERE IS CLOSEOUT.

OUR REPORTS MAY ALSO GO TO THE GM AND/OR THE BOARD EITHER BY COPY OF THE ORIGINAL OR IN SOME CONDENSED FORM.

THE GM AND BOARD ARE WELL AWARE OF OUR ACTIVITIES, FINDINGS, AND THE DEGREE OF RESOLUTION OF PROBLEMS. THE GM AND BOARD ARE SUPPORTIVE AND MORE INVOLVED IN THE NUCLEAR PROGRAM AS A RESULT OF THE NSRS PROGRAM.

WHAT I HAVE TALKED ABOUT UP UNTIL NOW BASICALLY RELATES TO THE NSRS ACTIVITIES ADDRESSING THE REVIEW AND EVALUATION OF THE PERFORMANCE OF THE LINE ORGANIZATION. OTHER FUNCTIONS ASSIGNED TO NSRS HAVE BEEN LIMITED DURING THE PAST TWO YEARS.

CHANGES ALREADY UNDERWAY WITHIN TVA WILL RESULT IN LESS NEED FOR DETAILED REVIEWS OF THE TYPE I HAVE PREVIOUSLY DESCRIBED. FUTURE ACTIVITIES WILL BE DIRECTED TOWARD UNRESOLVED SAFETY ISSUES, PRA, AND SAFETY POLICY DEVELOPMENT.

THIS CONCLUDES MY STATEMENTS REGARDING OUR PROGRAMS AND OUR INVOLVEMENT WITH SAFETY ISSUES WITHIN TVA.

FUNCTIONS OF THE NSRS

- INDEPENDENT REVIEW OF NUCLEAR PLANT DESIGN
- INDEPENDENT MONITORING OF NUCLEAR PLANT CONSTRUCTION
- INDEPENDENT MONITORING OF NUCLEAR PLANT OPERATIONS
- REVIEW OF NUCLEAR PLANT EMPLOYEE TRAINING
- REVIEW OF RADIOLOGICAL EMERGENCY PLANS
- REVIEW AND AUDIT OF RADIATION PROTECTION
- INVESTIGATION AND REVIEW OF OPERATING EVENTS OR INCIDENTS AT TVA PLANTS OR OTHER PLANTS
- RECEIPT AND INVESTIGATION OF EMPLOYEE CONCERNS ABOUT SAFETY ISSUES NOT ADEQUATELY ADDRESSED BY LINE MANAGEMENT

NSRS TECHNICAL STAFF BACKGROUND

STAFF RECRUITED FROM

Outside TVA - 12

- NRC 6
- DOE 2
- Utilities 3
- Other 1

Within TVA - 9

- EN DES 2
- CONST 3
- NUC PR 3
- PSS 1

STAFF NUCLEAR EXPERIENCE

Staff No.	Years
1	0-5
4	6-10
8	11-15
4	16-20
2	21-25
2	26-30
Avg.	14.7

Total Man-Years = 308

Number of Technical Staff = 21

EXPERIENCE BACKGROUND (Man-Years)

- DESIGN 57
- CONSTRUCTION 26
- OPERATION 117
- REGULATORY/AUDIT 108

NSRS REVIEWS COMPLETED 1980-1982

MANAGEMENT REVIEWS

OFFICE OF POWER (BFN)	1
OFFICE OF HEALTH AND SAFETY (BFN)	1
DIVISION OF PURCHASING (BLN)	1
OFFICE OF ENGINEERING DESIGN AND CONSTRUCTION	2
PUBLIC SAFETY SERVICE	1
QUALITY ASSURANCE PROGRAMS	1
TOTAL	7

ROUTINE OR SPECIAL REVIEWS

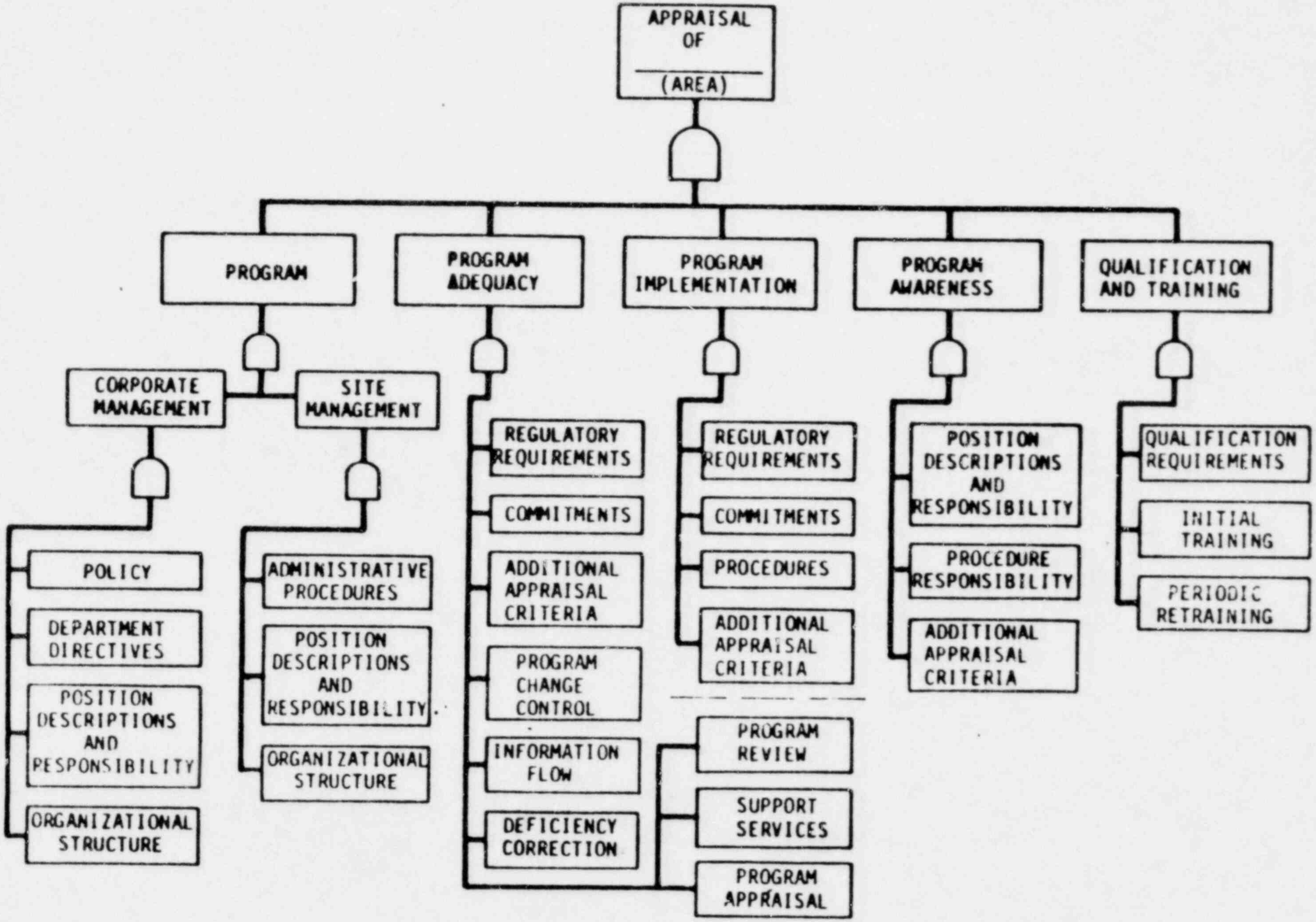
OPERATIONS

• STARTUP PROGRAM	2
• TRAINING	2
• SECURITY	2
• EMERGENCY PLANNING	3
• ROUTINE	16
• EMPLOYEE CONCERNS	2
• OTHER SPECIAL REVIEWS	10
TOTAL	37

DESIGN/CONSTRUCTION

• PIPE HANGERS AND SUPPORTS	3
• QA PROGRAM REVIEWS	2
• INTERFACE CONTROLS	2
• SPECIAL REVIEWS	3
• EMPLOYEE CONCERNS	2
TOTAL	11

MANAGEMENT EVALUATION TREE



FUNCTIONAL AREAS REVIEWED DURING MANAGEMENT REVIEW OF OEDC

- MANAGEMENT CONTROLS
- QUALITY ASSURANCE
- TRAINING AND QUALIFICATIONS OF PERSONNEL
- INTERFACE CONTROLS
- DESIGN PROCESS CONTROLS
- CONFIGURATION CONTROL
- CORRECTIVE ACTION
- RECORDS AND DOCUMENT CONTROL
- PROCUREMENT
- SCHEDULING OF CONSTRUCTION ACTIVITIES
- ASME SECTION III QA PROGRAM
- SPECIAL PROCESS CONTROLS
- EQUIPMENT AND FACILITIES CONTROL

NSRS PROGRAM ELEMENTS

PROGRAM ELEMENT	%
INTERFACE PROGRAM WITH EXTERNAL SAFETY ORGAN	13
GENERIC & UNRESOLVED SAFETY ISSUES	15
REGULATORY REQUIREMENTS	13
PROBABILISTIC RISK ASSESSMENT	13
EMPLOYEE CONCERN PROGRAM	4
INVESTIGATION PROGRAM	32
NUCLEAR SAFETY RELATIONS PROGRAM	4
TRAINING	6

IDENTIFIED PROBLEMS

- PRIOR TO 1980

- 1980 - 1981 TIMEFRAME
  - HVAC
  - WELDING
  - TRANSFER TO OPERATOR
  - MORALE

Beasley
T5

STEPS TAKEN TO CORRECT

- EACH IDENTIFIED DEFICIENCY CORRECTED
- PROCEDURES
- REINSPECTIONS
- REWORK AND NCRs

CURRENT STATUS

- QA PROGRAM BACK ON TRACK
  - SOME CORRECTIVE ACTION STILL BEING FORMULATED
  - SOME CORRECTIVE ACTION STILL BEING IMPLEMENTED
- RECENT NCRs ON ANALYSES
- 1982 QUALITY ACTION PLAN
- FEEL THAT WE HAVE IDENTIFIED ALL THE HOLES IN OUR PROGRAM

INDEPENDENT REVIEW

- NSRS REVIEWS

- NSRS RECOMMENDATION

- REGION II RECOMMENDATION

- ARRANGING FOR INDEPENDENT REVIEW

  - OBJECTIVE TO ENABLE TVA MANAGEMENT TO CONCLUDE DESIGNED AND CONSTRUCTED IN ACCORDANCE WITH LICENSE APPLICATION

  - REVIEW AND FINDINGS AVAILABLE TO NRC

  - SCOPE

  - COMPLETE BY END OF YEAR

COMPARISON WITH OTHER TVA PLANTS

- NOT THAT DIFFERENT

- EXCEPTIONS
  - HVAC
  - MORALE - MANAGEMENT CONTROL

- COMPARISON OF NSRS REPORTS

CONCLUSIONS

- QA PROGRAM IS ON A COURSE OF ACTION THAT WILL PROVIDE AND DOCUMENT SATISFACTORY PLANT

- HAVE A LOT OF WORK TO DO

- OEDC CONFIDENT

  - HAVE AN EXCELLENT PLANT

  - QUALITY PROBLEMS ARE BEING RESOLVED

  - INDEPENDENT REVIEW WILL CONFIRM DESIGN - CONSTRUCTION PROCESS

SEISMIC QUALIFICATION

- ALL SAFETY-RELATED ELECTRICAL AND MECHANICAL EQUIPMENT HAS BEEN SEISMICALLY QUALIFIED TO LEVELS WHICH ENVELOPE CONDITIONS DEFINED FOR ITS "AS-INSTALLED" CONFIGURATION.

- TVA'S EQUIPMENT SEISMIC QUALIFICATION PROGRAM IS IN FULL COMPLIANCE WITH NRC AND INDUSTRY RECOMMENDED PROCEDURES, GUIDES, CODES, AND STANDARDS--AND GOOD ENGINEERING PRACTICE.

- EQUIPMENT QUALIFICATION REPORTS PROVIDE A CONSERVATIVE DEMONSTRATION THAT THE EQUIPMENT IS CAPABLE OF WITHSTANDING ITS PRESCRIBED SEISMIC CONDITIONS.

- THE CURRENT PHILOSOPHY REGARDING SEISMIC QUALIFICATION THROUGHOUT THE INDUSTRY, TVA'S PROGRAM AS TYPICAL, DOES NOT REQUIRE THAT THE EFFORT BE EXTENDED TO DETERMINE HOW MUCH BETTER THE EQUIPMENT IS THAN IT NEEDS TO BE; NOR DOES THE QUALIFICATION DATA LEND ITSELF TO THE EXTRACTION OF SUCH INFORMATION.

- THE SEISMIC QUALIFICATION PROGRAM AS WE KNOW IT CANNOT BE TRANSFORMED INTO AN EQUIPMENT RELIABILITY PROGRAM. REEVALUATION EFFORT WOULD PROVIDE INDICATIONS OF MARGINS OF CONSERVATISM IN QUALIFICATION OF SPECIFIC ITEMS OF EQUIPMENT.

T7 Williams

SEISMIC MARGIN OF CONSERVATISM

TVA PROGRAMS

- SEQUOYAH NUCLEAR PLANT - REEVALUATION OF EQUIPMENT QUALIFICATION AGAINST THE HIGHER SEISMIC LEVELS OF THE SITE SPECIFIC SPECTRA DEMONSTRATED THAT QUALIFICATION HAD BEEN ACCOMPLISHED WITH A FACTOR OF CONSERVATISM OF AT LEAST 1.5.

- BROWNS FERRY NUCLEAR PLANT - PROBABILISTIC RISK ASSESSMENT STUDY INCLUDED THE CONSIDERATION OF EQUIPMENT SEISMIC QUALIFICATION. STUDY FOUND THAT MOST EQUIPMENT REFLECTED LARGE MARGINS OF CONSERVATISM BEYOND THE PRESCRIBED SEISMIC CONDITIONS; THE WEAKEST LINK IS RELAY CHATTER IN ELECTRICAL EQUIPMENT.

- WATTS BAR NUCLEAR PLANT - PROBABILISTIC RISK ASSESSMENT TO BE ACCOMPLISHED--CURRENT SCHEDULE, TARGET COMPLETION DATE MAY 1984.

MINIMUM FACTORS OF CONSERVATISM OF BROWNS FERRY EQUIPMENT

COMPONENT	FACTOR OF CONSERVATISM***
250 VDC CIRCUIT BREAKER BOARD	1.4*/5.45**
REACTOR PRESSURE VESSEL INTERNALS	2.45
DIESEL GENERATOR TRANSFORMERS	2.5
CONTROL ROD DRIVE HOUSING	3.85
. . .	LARGER

*RELAY CHATTER

**BREAKER TRIP

***FACTOR OF CONSERVATISM = $FC_E \cdot FR_S \cdot FR_E$

WHERE:

FC_E = EQUIPMENT CAPACITY FACTOR = STRENGTH/LOAD

FR_S = STRUCTURAL RESPONSE FACTOR = TEST RESPONSE/ACTUAL RESPONSE

FR_E = EQUIPMENT RESPONSE FACTOR = DESIGN FLOOR SPECTRA/ANTICIPATED ACTUAL SPECTRA

SEISMIC MARGIN OF CONSERVATISM (CONTINUED)

GENERIC PROGRAMS

- SEISMIC SAFETY MARGIN RESEARCH PROGRAM, SSMRP, LAWRENCE LIVERMORE LABORATORY (NUREG/CR-2405) - EXTENSIVE RESEARCH PROGRAM INCLUDED CONSIDERATION OF EQUIPMENT FRAGILITIES.

  AN IMPORTANT OBSERVATION . . . IS THAT MOST MECHANICAL AND ELECTRICAL EQUIPMENT IS INHERENTLY RUGGED AND WILL SURVIVE ACCELERATION LEVELS FAR IN EXCESS OF BUILDING RESPONSES ASSOCIATED WITH THE SAFE SHUTDOWN EARTHQUAKE.

- SEISMIC DESIGN MARGINS OF PUMPS, VALVES, AND PIPING (NUREG/CR-2137) - STUDY TO ESTABLISH MARGIN OF CONSERVATISM INHERENT IN CODE DESIGN OF FLUID SYSTEM COMPONENTS--MARGINS LISTED INDICATE LOWER BOUND.

MINIMUM FRAGILITY VALUES OF ZION* EQUIPMENT

COMPONENT	FACTOR OF CONSERVATISM
125 VAC DISTRIBUTION PANEL	3.5
SERVICE WATER PUMPS	3.7
4160 V SWITCHGEAR	4.2
.	
.	LARGER
.	

*REFERENCE PLANT FOR SSMRP

DESIGN MARGINS OF PUMPS, VALVES, AND PIPING
(NUREG/CR-2137)

NOMINAL MARGINS

FAILURE CRITERIA	ASME CODE, FOR PRESSURE BOUNDARY INTEGRITY		AISC MANUAL, FOR SUPPORTS	
	OBE	SSE	BASIC	SEISMIC
BREAK	3.0 TO 10.4	1.43 TO 5.2	2.6 TO 3.1	2.0 TO 2.3
YIELD	1.1 TO 4.8	0.55 TO 2.4* 1.1 TO 4.8**	1.67	1.25

*NORMAL MARGINS FOR PRESSURE BOUNDARY INTEGRITY ONLY--SERVICE LEVEL D

**NORMAL MARGINS FOR OPERABILITY ASSURANCE--SERVICE LEVEL B

NOMINAL MARGIN ON YIELDING = S_Y/S_A

NOMINAL MARGIN ON BREAKING = S_U/S_A

WATTS BAR NUCLEAR PLANT UNITS 1 AND 2

ACRS PRESENTATION

by C. F. Bowman

Cement Mortar Lining

of the

Essential Raw Cooling Water System

Yard Piping

1.0 History

1.1 Problem Definition

While preoperational testing the emergency equipment cooling water (EECW) system at Browns Ferry Nuclear Plant during the summer of 1976, certain heat exchangers were found to be receiving inadequate cooling water flow due to a buildup of corrosion products in the interior of the carbon steel piping servicing the equipment. Since carbon steel piping was extensively used in both safety-related and nonsafety-related piping systems at other TVA nuclear plants, a study was undertaken to determine the pervasiveness of this problem in the TVA system and to develop recommended practices to mitigate its effects.

T9

1.2 Sampling Program

Approximately 50 sections of carbon steel raw water piping were removed from nine different TVA steam plants. Both normally stagnant and normally flowing piping systems, as well as both vertical and horizontal runs of pipe, were sampled.

In virtually every case, the primary mechanism was found to be corrosion of the steel piping by aerated river water and redeposition of the corrosion products. The problem was found to a significant degree at all plants that were sampled. The result is random pitting in the pipe wall and the formation of a tubercle over each pit.

The equivalent average diameter reduction as a result of corrosion products buildup as a function of years of service is shown on Figure 1. The deposit in each sample was removed and analyzed for various constituents. In virtually every case, it was found to be principally iron oxide.

1.3 Pressure Drop Tests

Tests were performed at the Widows Creek, Kingston, and Gallatin steam plants to evaluate the effects of corrosion product buildup on pressure drop. The sites were selected to cover a range of ages as well as a variety of water sources. Samples removed from each test line were analyzed to determine the percent volume reduction of the pipe interior due to the corrosion product buildup. The corresponding diameter reduction for each test line was then used with the pressure drop test data to develop appropriate equations for pressure drop.

Several figures were generated in an attempt to find a correlation between diameter reduction and Hazen-Williams C. Values of C were assumned and corresponding values of diameter were calculated for each test. A dimensionless parameter, d^*, was defined for use in correlating the above calculated value of d with the measured value of diameter reduction.

$$d^* = \frac{(d_{NOM} - d_{CALC})}{\Delta d_{MEAS}}$$

$$= \frac{\text{Calculated Diameter Reduction}}{\text{Measured Diameter Reduction}}$$

Using this relationship for each of the three pressure drop tests conducted as shown in Figure 2, good agreement was discovered for values of $d^* = 2$ and $C = 57$. We have adopted a slightly more conservative value of $C = 55$ as our design value. The result is a modified Hazen-Williams equation

$$h_L = \frac{0.63Q^{1.85}}{(d_{NOM} - 2 \times \Delta d_{MEAS})^{4.8655}}$$

Figure 3 shows the comparison between the predictive model and actual test data taken on a 3-inch line at Widows Creek. Note also the head loss predicted by the normally used Hazen-Williams

C = 100. Similar comparisons are presented for the tests conducted at Kingston and Gallatin and are shown on Figures 4 and 5, respectively.

1.4 Corrective Action

TVA is now using this equation to evaluate the heat rejection system, fire protection system, raw cooling water system, and raw service water system at Watts Bar Nuclear Plant. Most significantly, however, we have completed our evaluation of the Watts Bar ERCW system, and the remainder of this presentation discusses results of that evaluation.

The analysis of the ERCW system determined that delivery of design flow rate to system users over plant life could not be guaranteed with the original design. Consequently, changes were defined to bring the system within the 40-year design basis. Figure 6 shows the changes being made in the ERCW system. They include

1. Replacement of selected segments of carbon steel piping within the buildings with stainless steel.

2. Requalifying certain system users to a lower ERCW flow rate by refinement of the heat transfer design calculations.

3. Applying a cement mortar lining to existing carbon steel piping in the yard in situ.

2.0 Experience with Cement Mortar

Before making the final decision to cement mortar line the ERCW system yard piping, a telephone survey was conducted to determine how well it has performed in service. Table I shows a total of 11 other utilities, 2 A-E's, and 5 municipal water systems which have cement mortar lined piping in service and which were canvassed. Although a few problems were reported, in general the experience reported was very good. All of the problems identified could be attributed to either the pipe being out of round or a failure to properly protect the pipe joints. In one instance lined pipe went through the San Fernando earthquake of 1971 which leveled buildings but only caused damage to the cement mortar lining where the pipe itself was plastically deformed.

3.0 Installation of Cement Mortar

The procedure for applying the cement mortar lining requires that the piping first be cleaned by scraping off existing tubercles. Thereafter, the mortar is centrifugally applied from a spinning head and immediately troweled onto the inside surfaces using a machine which is pulled through from one end. Closure pieces and certain elbows are thereafter hand mortared to complete the process. The closure weld-affected region is hand lined from inside the pipe.

Humidity is carefully controlled after application to ensure proper curing, and each foot of piping is carefully inspected prior to plant operation.

TVA specified the necessary level of quality assurance on the lining process, and a number of nonconformances to the specifications have occurred. These have included:

High and low mortar slump

High mortar temperature

Low mortar compressive strength

Low relative humidity

Surface cracks

Mortar applied too thin

End caps not replaced

Pipe damage

Exterior coating damage

Where appropriate, repairs have been made in accordance with approved procedures.

4.0 Surveillance of Cement Mortar Lining

Very good experience has been reported with cement mortar lining by other utilities. However, since this is the first application by TVA and since the ERCW system is a Safety Class 3 system, provision is being made in the design to facilitate periodic inspection of a portion of the system after it has been placed in service.

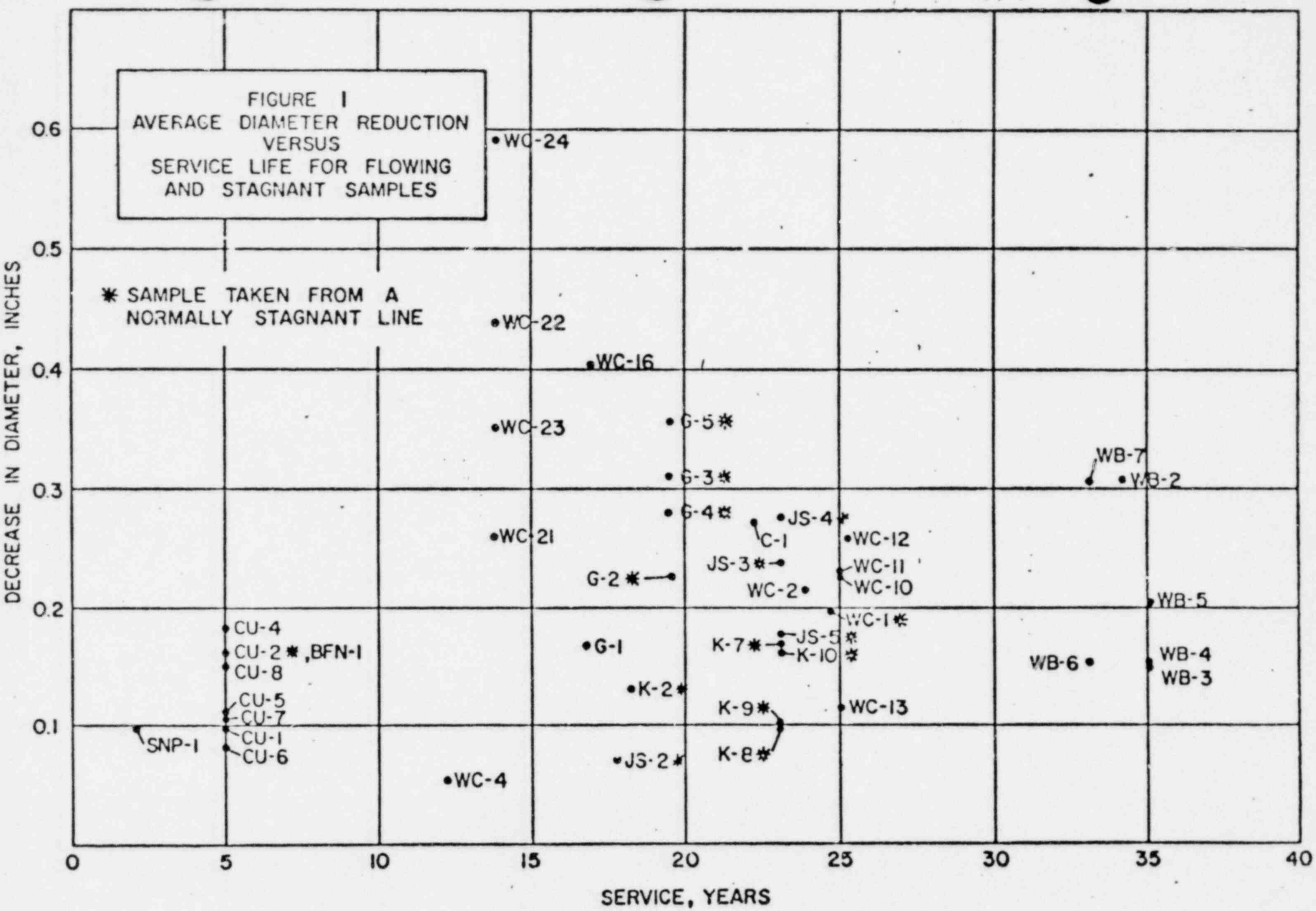


FIGURE I
AVERAGE DIAMETER REDUCTION
VERSUS
SERVICE LIFE FOR FLOWING
AND STAGNANT SAMPLES
* SAMPLE TAKEN FROM A NORMALLY STAGNANT LINE
DECREASE IN DIAMETER, INCHES
SERVICE, YEARS
0.1
0.2
0.3
0.4
0.5
0.6
0
5
10
15
20
25
30
35
40
WC-24
WC-22
WC-16
WC-23
WC-21
WC-4
G-5
G-3
G-4
G-2
G-1
K-2
JS-2
C-1
JS-4
JS-3
WC-2
K-7
JS-5
K-10
K-9
K-8
WC-12
WC-11
WC-10
WC-1
WC-13
CU-4
CU-2
BFN-1
CU-8
CU-5
CU-7
CU-1
CU-6
SNP-1
WB-7
WB-2
WB-5
WB-6
WB-4
WB-3

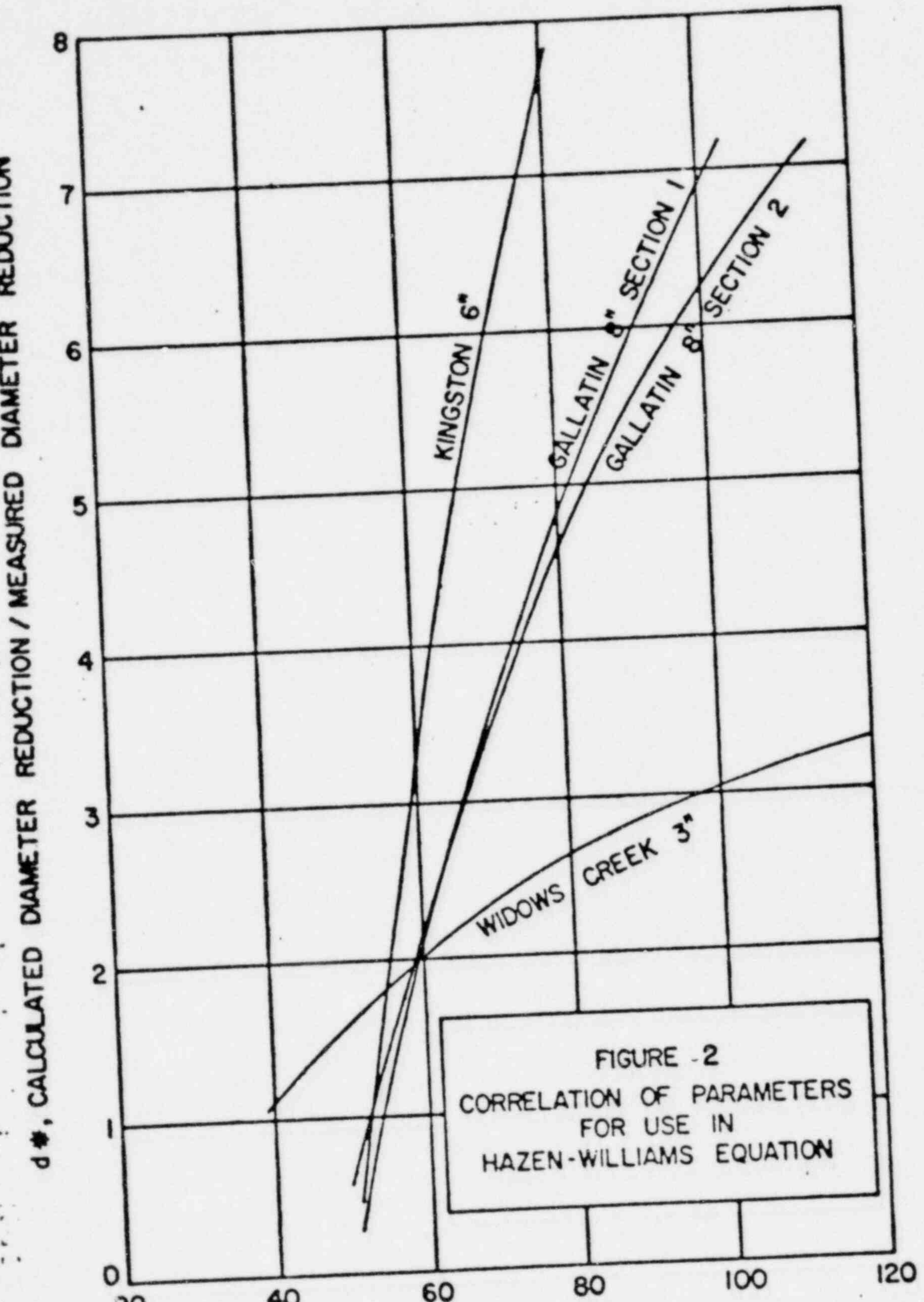


FIGURE - 2
CORRELATION OF PARAMETERS
FOR USE IN
HAZEN-WILLIAMS EQUATION

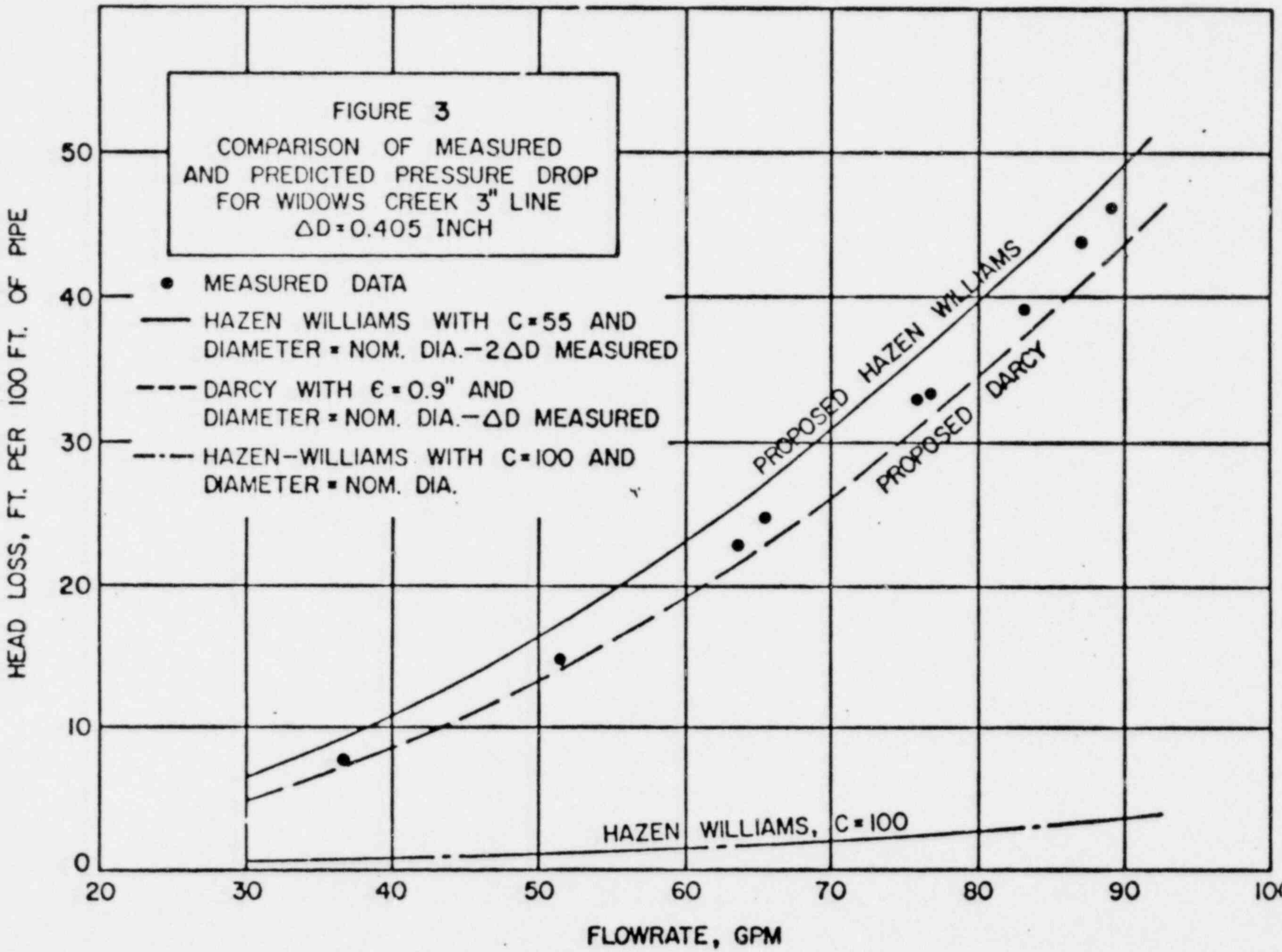


FIGURE 3
COMPARISON OF MEASURED
AND PREDICTED PRESSURE DROP
FOR WIDOWS CREEK 3" LINE
ΔD = 0.405 INCH
• MEASURED DATA
HAZEN WILLIAMS WITH C = 55 AND
DIAMETER = NOM. DIA.−2ΔD MEASURED
DARCY WITH ϵ = 0.9" AND
DIAMETER = NOM. DIA.−ΔD MEASURED
HAZEN−WILLIAMS WITH C = 100 AND
DIAMETER = NOM. DIA.
PROPOSED HAZEN WILLIAMS
PROPOSED DARCY
HAZEN WILLIAMS, C = 100
HEAD LOSS, FT. PER 100 FT. OF PIPE
FLOWRATE, GPM
0
10
20
30
40
50
20
30
40
50
60
70
80
90
100

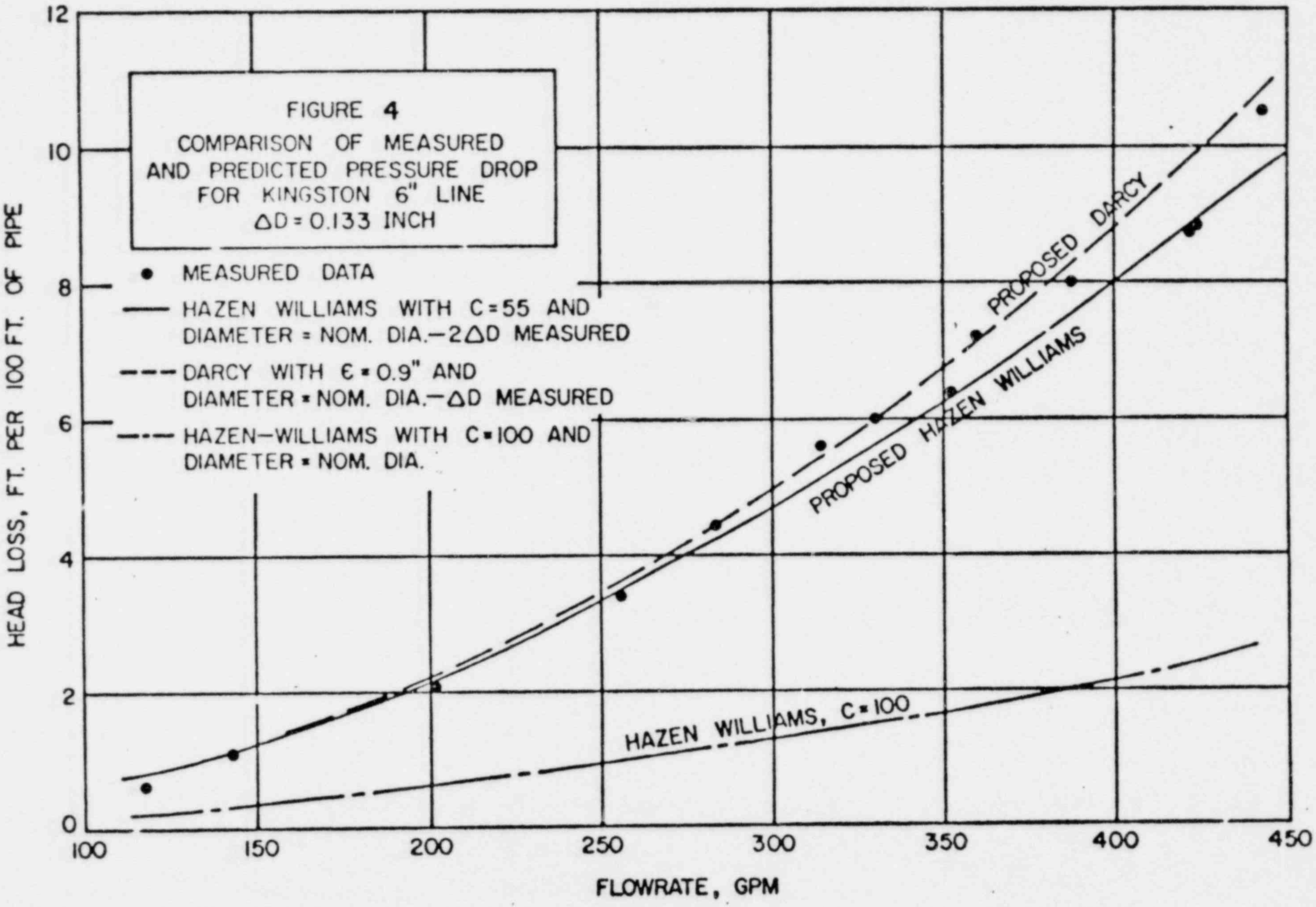


FIGURE 4
COMPARISON OF MEASURED AND PREDICTED PRESSURE DROP FOR KINGSTON 6" LINE
ΔD = 0.133 INCH
● MEASURED DATA
—— HAZEN WILLIAMS WITH C = 55 AND DIAMETER = NOM. DIA.—2ΔD MEASURED
– – – DARCY WITH ε = 0.9" AND DIAMETER = NOM. DIA.—ΔD MEASURED
—·— HAZEN—WILLIAMS WITH C = 100 AND DIAMETER = NOM. DIA.
PROPOSED DARCY
PROPOSED HAZEN WILLIAMS
HAZEN WILLIAMS, C = 100
HEAD LOSS, FT. PER 100 FT. OF PIPE
FLOWRATE, GPM
0
2
4
6
8
10
12
100
150
200
250
300
350
400
450

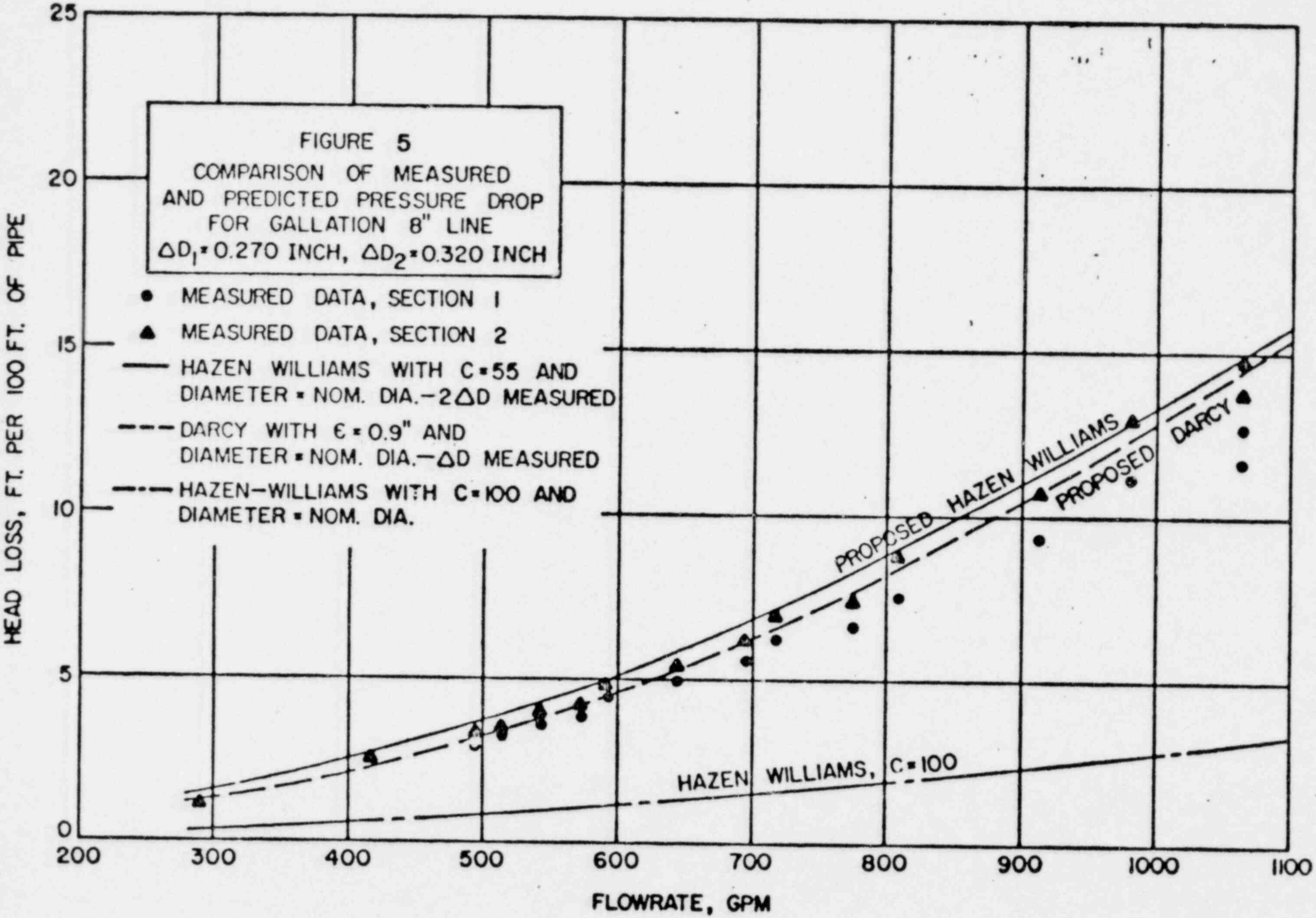


FIGURE 5
COMPARISON OF MEASURED AND PREDICTED PRESSURE DROP FOR GALLATION 8" LINE
ΔD1 = 0.270 INCH, ΔD2 = 0.320 INCH
MEASURED DATA, SECTION 1
MEASURED DATA, SECTION 2
HAZEN WILLIAMS WITH C = 55 AND DIAMETER = NOM. DIA.−2ΔD MEASURED
DARCY WITH ϵ = 0.9" AND DIAMETER = NOM. DIA.−ΔD MEASURED
HAZEN-WILLIAMS WITH C = 100 AND DIAMETER = NOM. DIA.
PROPOSED HAZEN WILLIAMS
PROPOSED DARCY
HAZEN WILLIAMS, C = 100
HEAD LOSS, FT. PER 100 FT. OF PIPE
FLOWRATE, GPM
0
5
10
15
20
25
200
300
400
500
600
700
800
900
1000
1100

CORRECTIVE ACTIONS:

1. STAINLESS STEEL
2. LOWER FLOWRATE
3. CEMENT MORTAR LINING

FIGURE 6

TABLE I

Survey of Operating Experience for Cement Mortar Lining

Utilities

Consolidated Edison[1]
Potomac Electric Power
Long Island Power
Cincinnati Gas and Electric[2]
Southern California Edison[3]
Florida Power and Light[2]
Sacramento Municipal Utility
Carolina Power and Light[1]
Los Angeles Water and Power[4]
Pacific Power and Light
Pacific Gas and Electric

A and E

Bechtel
Ebasco

Municipal Water Systems

City of Pasadena
City of Norfolk
City of Newport News
City of San Francisco
Seattle Water Department

1. Carbon steel pipe prelined with cement mortar using brackish water failed at uncoated welded joints or uncoated flanged joints.

2. Cement mortar lining in 9-foot CCW failed in a transition section and a severly out-of-round section of pipe.

3. Extremely old section of riveted pipe had lining failure after 25 years in some out-of-round sections.

4. During 1971 earthquake, cement mortar lining failed only where piping suffered plastic deformation.

STEAM GENERATOR VIBRATION MODIFICATIONS

D-3 MODEL - WBNP

TVA became aware of the tube wear problem due to flow induced vibration in November 1981 and began working with Westinghouse relative to the WBN units.

The discovery of tube wear problem was at Ringhals Unit 3 in October 1981 in Sweden. Sweden has since constructed two full scale models of a portion of preheater section of Model D-3 generator at its Hydraulics Laboratory.

In March-April 1982, Westinghouse entered into agreements with Sweden to have certain baseline and confirmatory tests performed as a part of development of design modification for D-2, D-3 generators. Test specifications, procedures, and quality assurance requirements were prepared by Westinghouse. Operation of the model has been by Sweden. Final data processing and evaluation has been performed by Westinghouse in the U.S.

During week of June 1, 1982, the model was in near readiness for starting baseline testing of existing D-3 design, when an overpressure event caused substantial damage and delay.

The model was placed in equivalent full-flow operation on June 22, 1982. Data collection and processing, including high speed photography, began.

Based on evaluation in U.S. and at site, initial baseline testing began on June 25, 1982.

U42224.01

T10

While the evaluation of the design of the modification was being performed, TVA began an economic evaluation of the options of making the modifications prior to fuel loading versus operating at 50% power level through the first refueling outage. It was determined that it would be to TVA's economic advantage to delay fuel loading and do the modification prior to fuel loading and specifically prior to hot functional testing. Based on preliminary information from Westinghouse, it was determined that we had a high probability of obtaining an acceptable design, testing, and performing the work by Westinghouse prior to November 3, 1982. The WBN schedule was adjusted to reflect this.

PROGRESS TO DATE

1. Full flow test of model D-3 first full manifold design is complete. This establishes a baseline for comparison purposes.

2. D-3 design full flow test to be rerun with force gauges in approximately four tubes in August 1982.

3. Based on model testing and analytical work W has a high level of confidence that the optimized test manifold design will be the production model.

4. Optimized manifold full flow test of the production model is now scheduled to begin the week of September 30, 1982.

5. W has informed TVA that the testing and installation schedule for Watts Bar will be finalized by August 30, 1982.

If there is any significant change in the Westinghouse schedule for a final modification, TVA will make another economic evaluation relative to the place in the schedule that the modification should be made.

TVA is working with Duke and South Carolina Power on the optimum time that the modifications could be made at the first three domestic plants.

TVA and Westinghouse will keep the NRC staff informed on the developments.

FULL SCALE TESTING
AND QUALIFICATION
OF
CEMENT—MOTAR LINED
CARBON STEEL PIPE

CEB 82-8

T10 - Hand

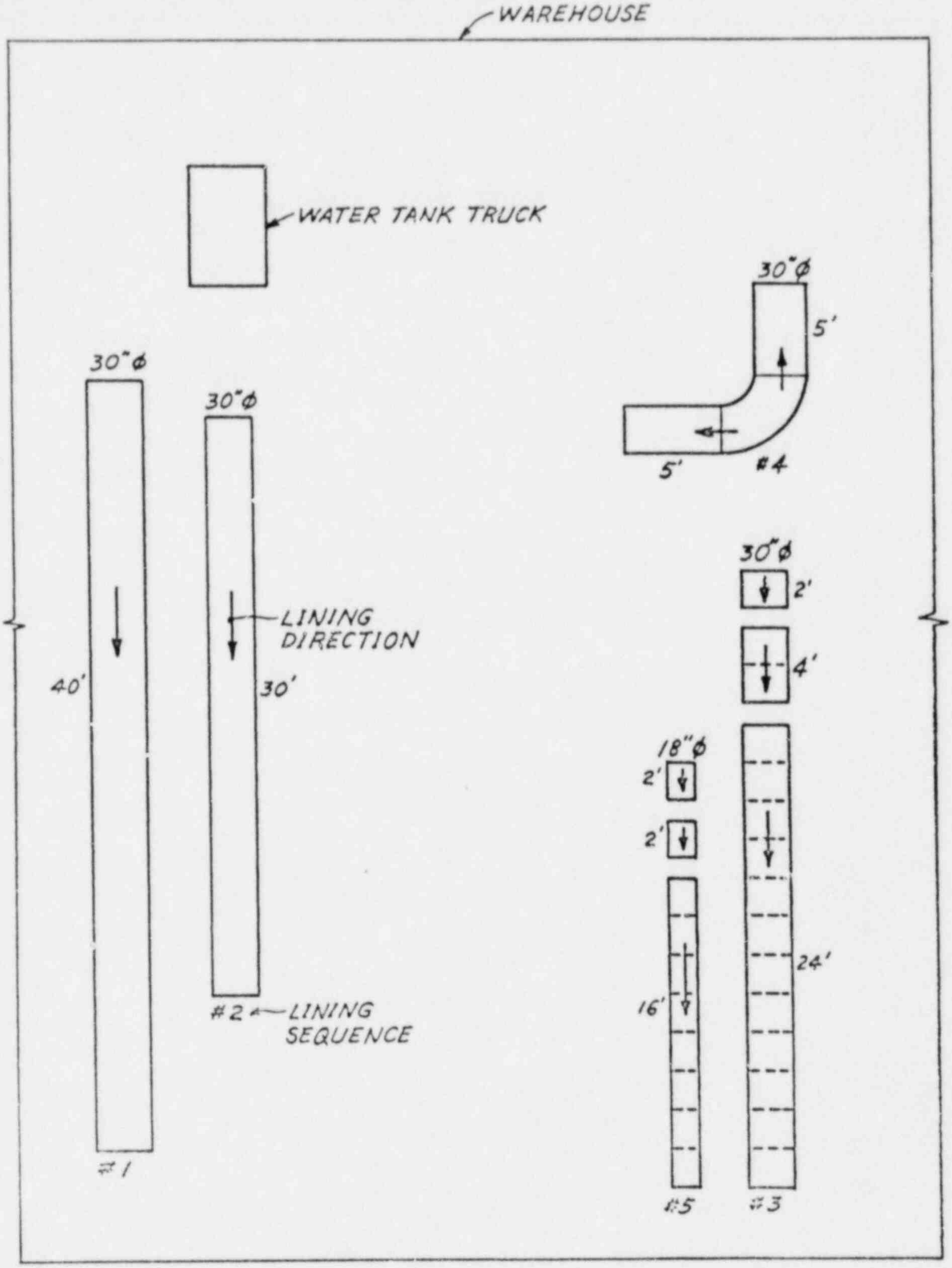
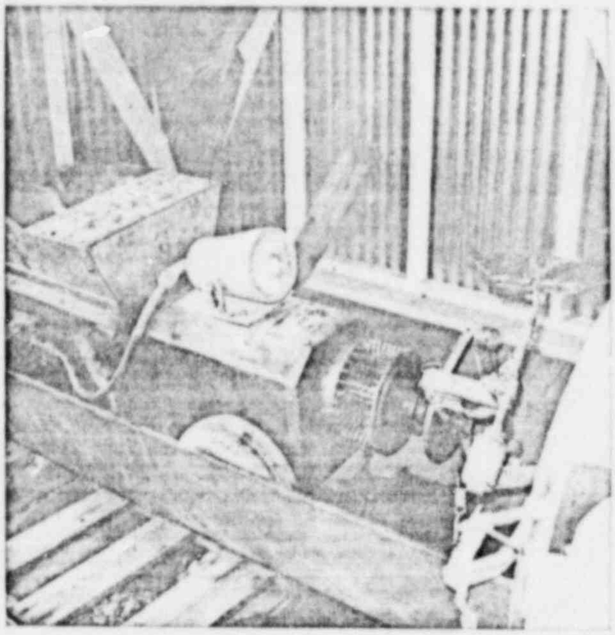
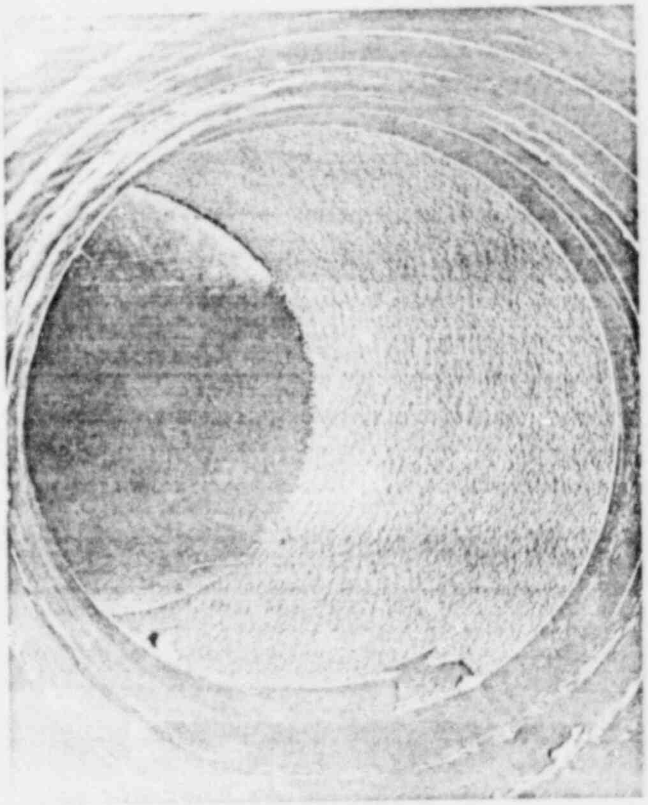


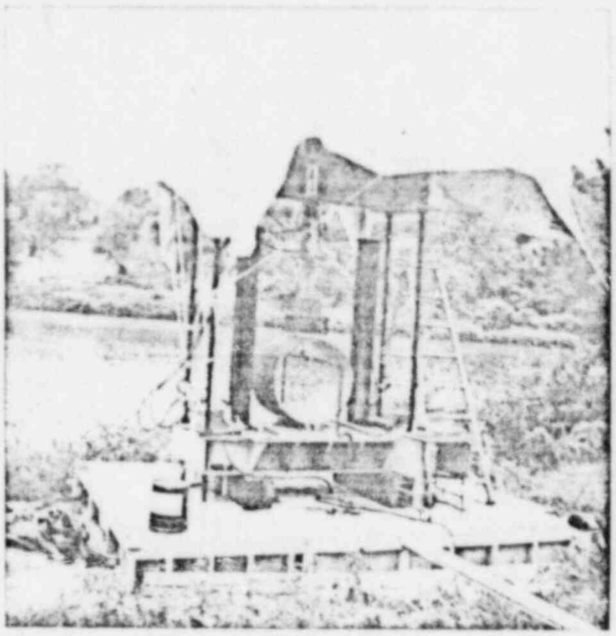
FIG. 1 LINING SEQUENCE AND DIRECTION



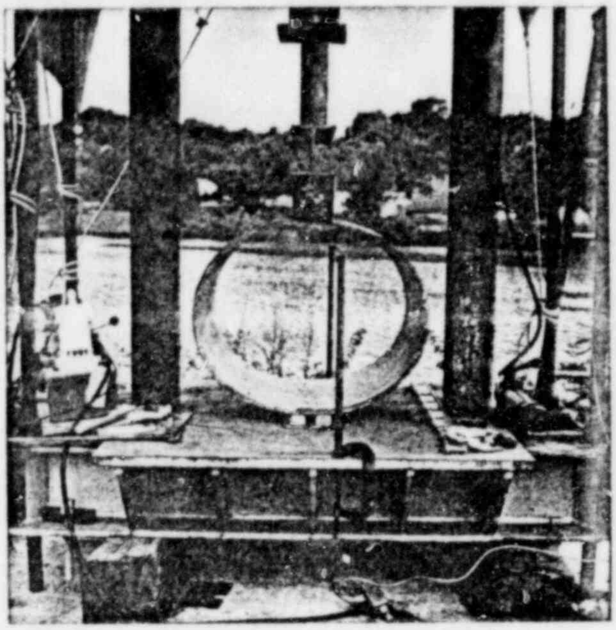
#24 - 30" Pipe lining machine -
front view



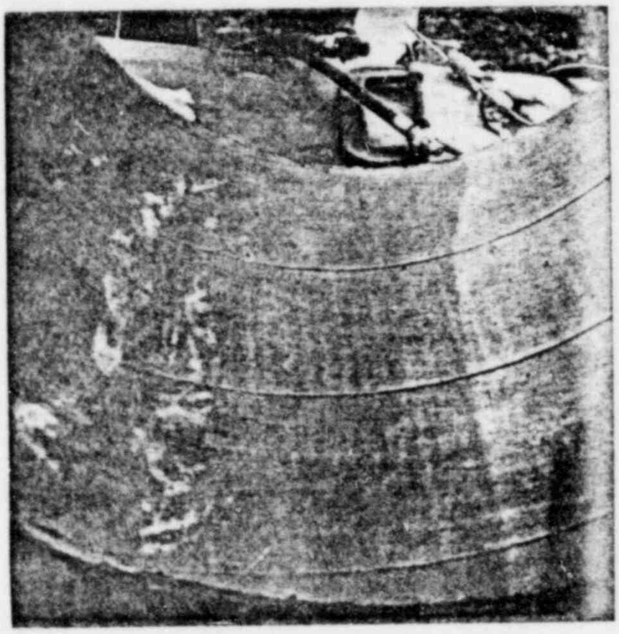
#38 - Elbow lining



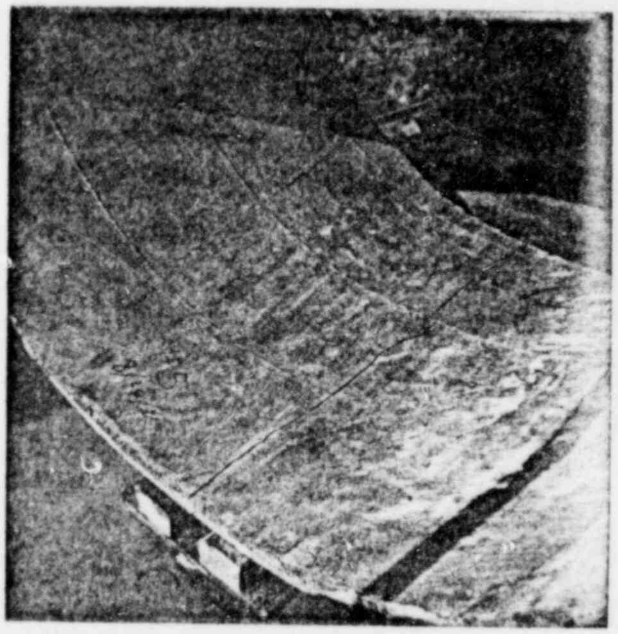
#116 - Three-edge bearing test -
loading machine set up



#158 - Three-edge bearing test -
30" Pipe, after loading,
fully cured (11 days)

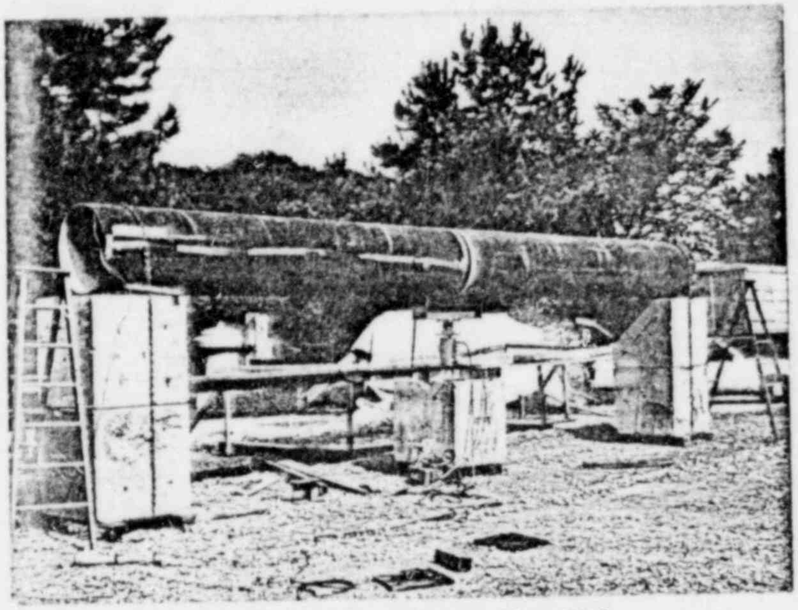


#160 - Three-edge bearing test -
30" Pipe, before loading,
fully cured (16 days)

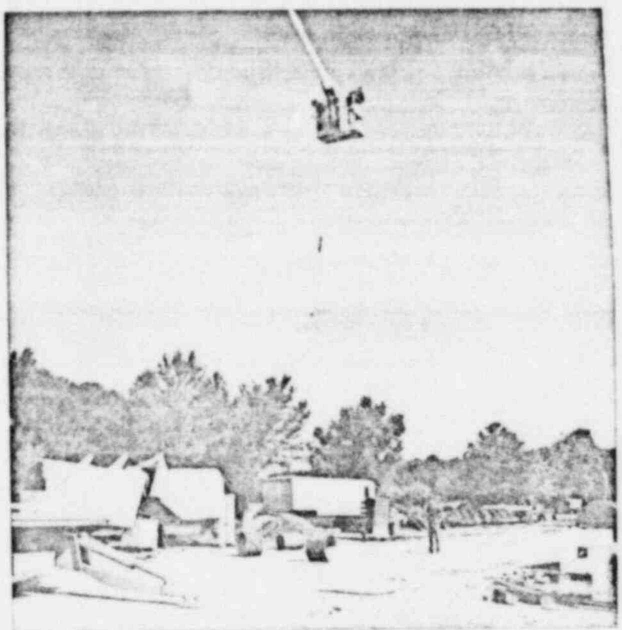


#164 - Three-edge bearing test -
30" Pipe, after loading,
fully cured (16 days)

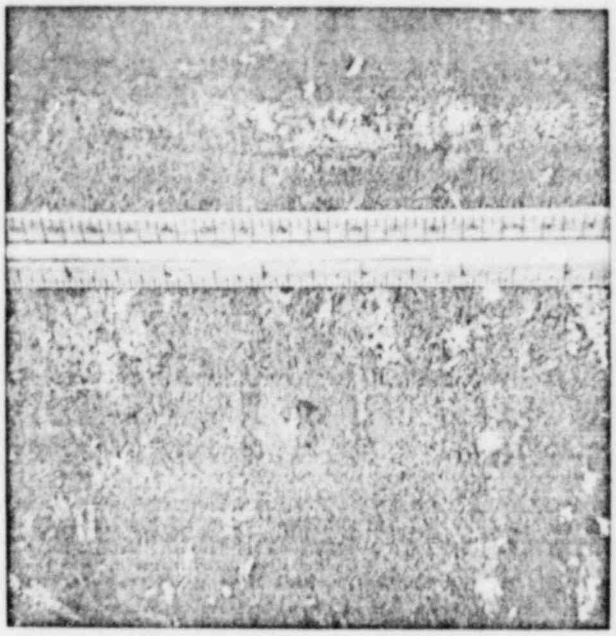
#299 - Torsion test - Initial
setup



#325 - Bending test on the 30' pipe - Initial setup

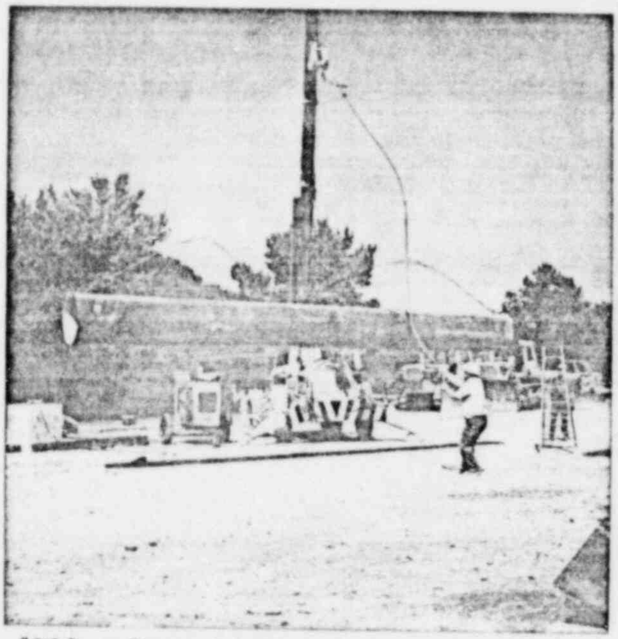


#241 - Impact test - Missile drop
in progress

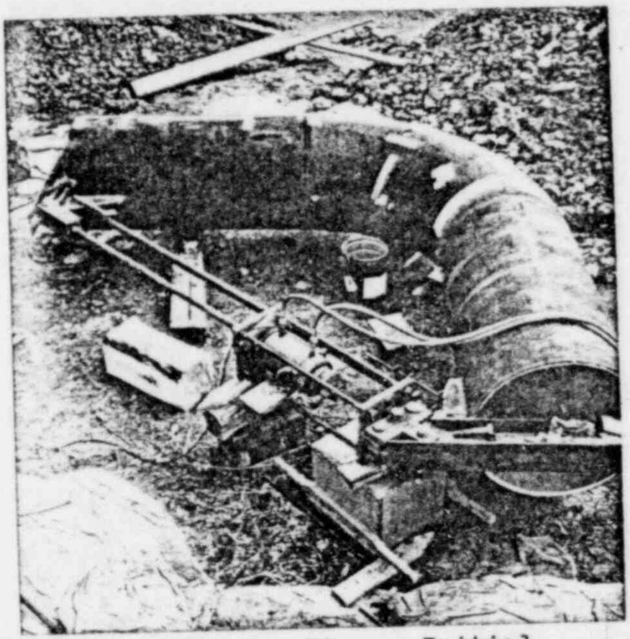


#253 - Impact test - 30" Pipe -
14' Drop

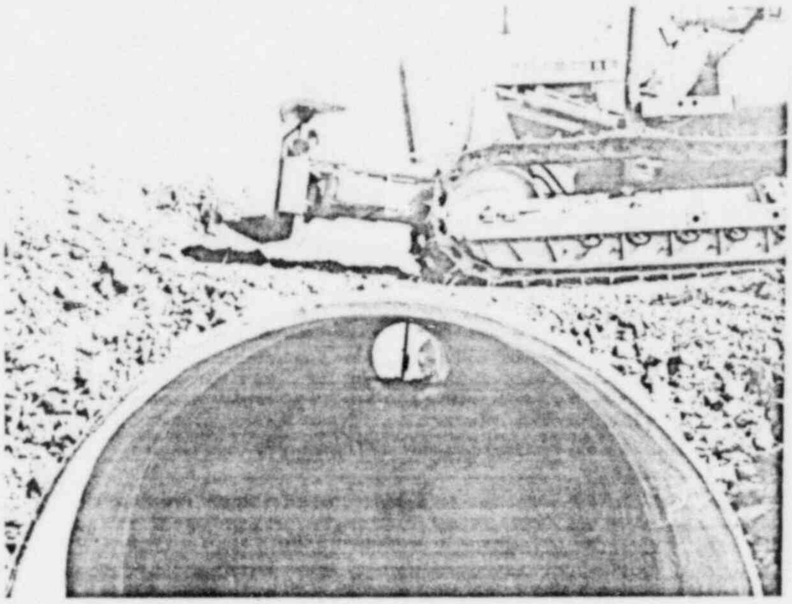
#255 - Impact test - 30" Pipe -
14' Drop



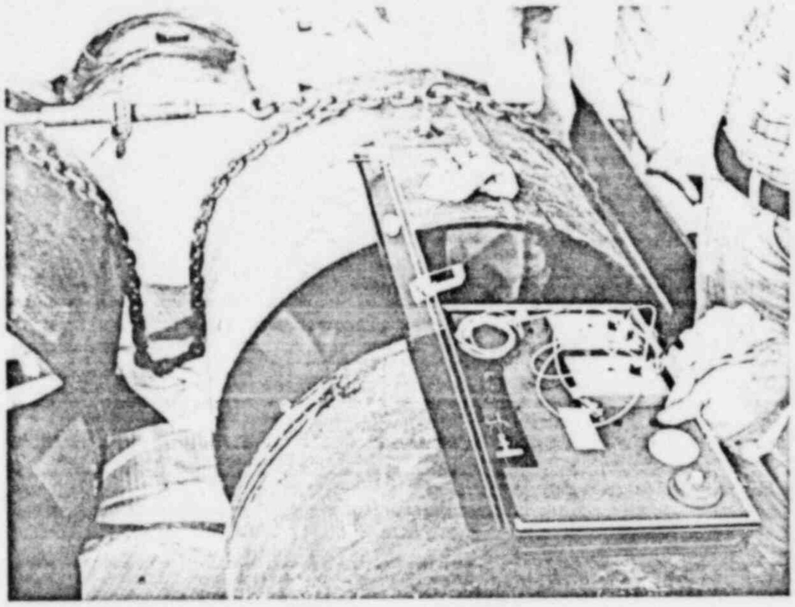
#329 - Drop test - 30' Pipe



#333 - Tests on elbow - Initial setup



#403 - Field test - Loading
6th cut



#117 - Installation of
accelerometer on pipe

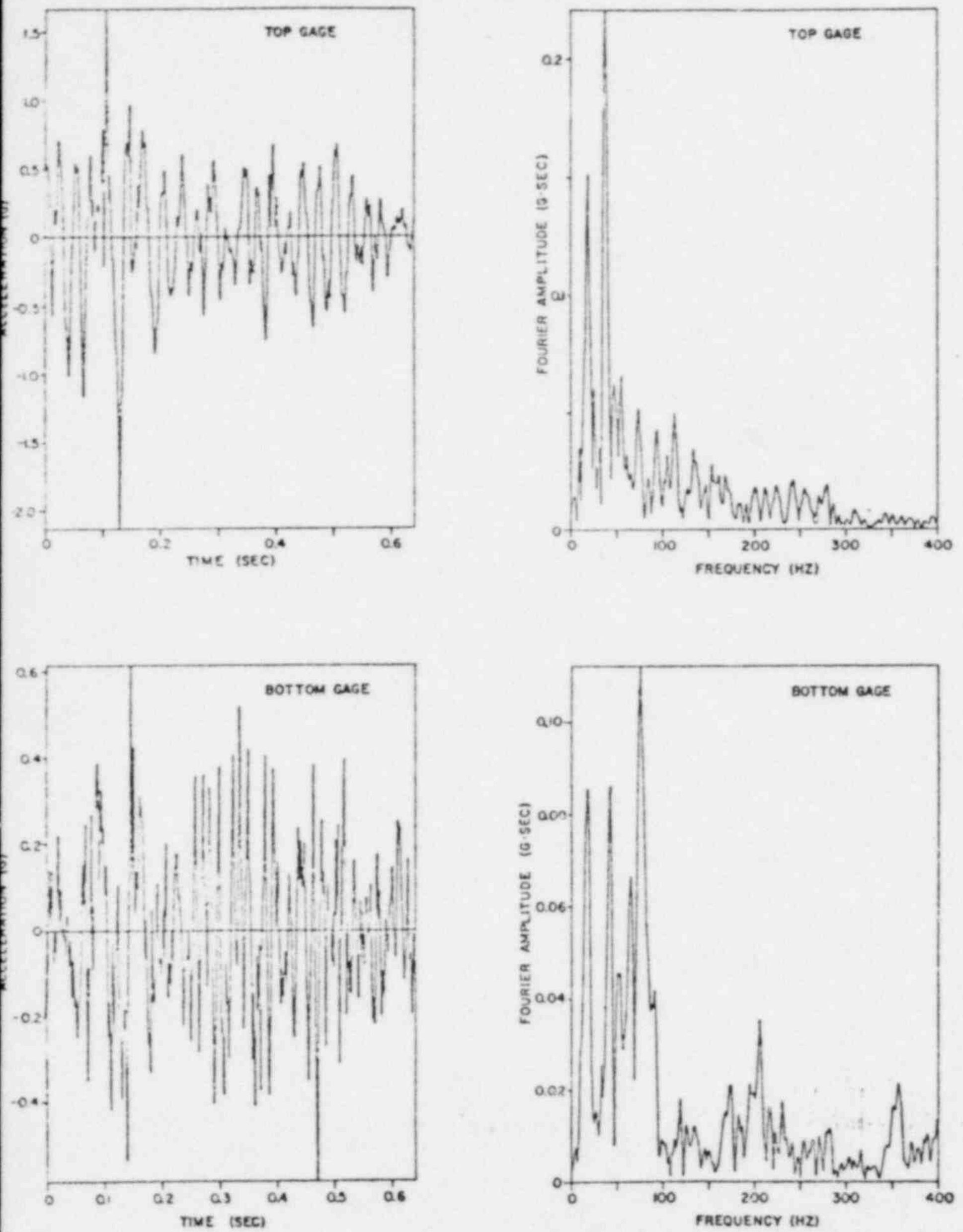
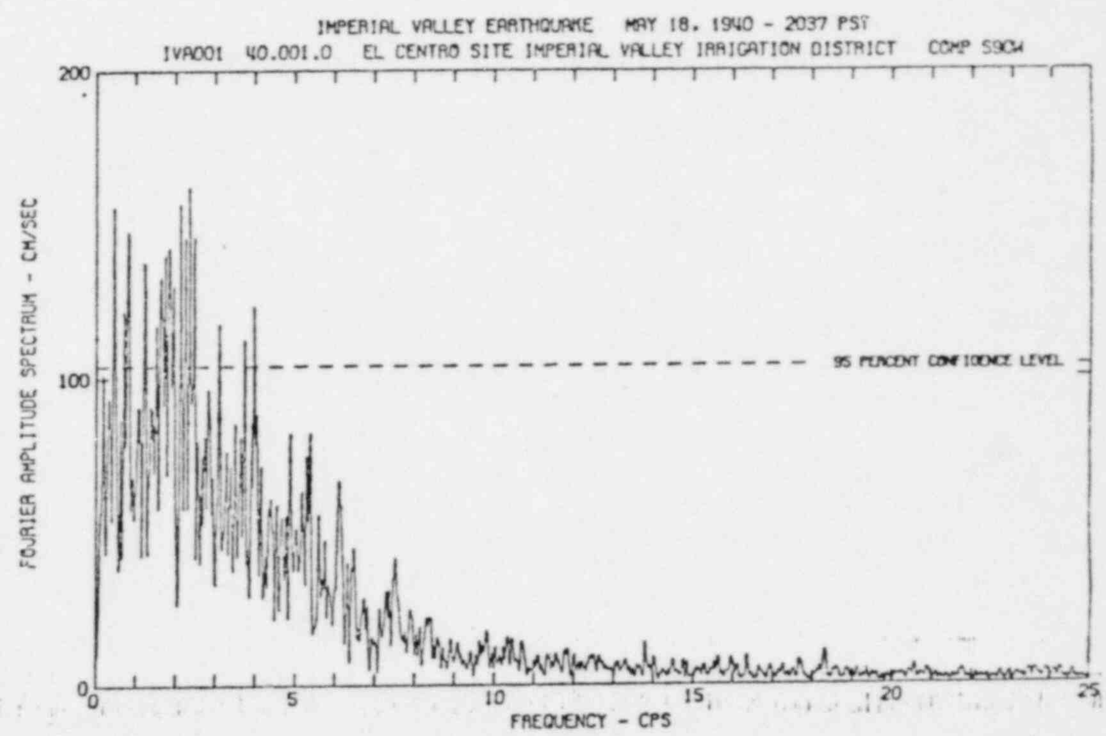


FIG. 2 ACCELERATION TIME HISTORIES AND FOURIER AMPLITUDE SPECTRA



(Earthquake Engineering Research Laboratory, California Institute of Technology, "Analyses of Strong Motion Earthquake Accelerograms, Vol. IV-Fourier Amplitude Spectra," August 1972)

FIG. 10 FOURIER AMPLITUDE SPECTRUM CALCULATED FROM TYPICAL EARTHQUAKE ACCELERATION TIME HISTORY