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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

ACRS Metal Components Subcommittee Meeting

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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ACRS METAL COMPONENTS SUBCOMMITTEE MEETING

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Room 1046

1717 H Street, Northwest

Washington, D.C.

Wednesday, September 4, 1985

The ACRS Metal Components Subcommittee met, pursuant to notice, at 8:35 o'clock, a.m., Paul G. Shewmon, chairman of the subcommittee, presiding.

APPEARANCES:

PAUL G. SHEWMON, Chairman of the Subcommittee,

Member, ACRS

DAVID A. WARD,

Member, ACRS

A. IGNE

ACRS Staff member

R. BRUCE THOMPSON

ACRS Consultant

APPEARANCES (continued):

R. ODETTE

ACRS Consultant

R. DILLON

ACRS Consultant

PRESENTERS:

B.D. LIAU

W. JOHNSTON

W. HAZELTON

D. SMITH

L.E. WILLERTZ

N. RANDALL

C. SERPAN

J. MUSCARA

P R O C E E D I N G S

MR. SHEWMON: This is a meeting of the ACRS Subcommittee on Metal Components. I am Paul Shewmon, Subcommittee Chairman. The other ACRS member in attendance is Dave Ward, on my left. ACRS consultants are Thompson, Odette, and Dillon. Al Igne is the ACRS staff member for this meeting.

The rules of participation have been announced as part of the Notice in the Federal Register on August 16.

We request that you speak up, if you speak, so that everybody can hear you.

We have received no written comments nor requests for oral statements from members of the public.

Today we are here to go over a series of different changes dealing primarily with the pressure boundary corrosion problems, changes in the Reg Guide 199 for changes in mill ductility with irradiation.

Without further ado, I think we will begin, and I will call on Bill Johnston to start things off.

MR. JOHNSTON: Thank you very much, Dr. Shewmon.

I am going to speak from here because I am just going to saya few words of introduction, and then Dr. Liaw and Warren Hazelton will be making the main presentation.

The first topic -- you can't hear me yet? Oh.

The first topic that we'll be talking about and the

one that I am most particularly personally interested in is the first one. It's the girth, shell girth weld cracks that have been appearing in some PWRs. This is an area in which we really would like advice from this group, from this committee, because it's one that we identified several years ago when the first crack was discovered through an actual small leak in the steam generator shell up at Indian Point.

At that time, we pursued the matter and had some boat samples taken and so forth, but weren't sure whether it was a generic problem of whether it was specific to that one particular plant.

Now, subsequent to that time, such things have appeared at the Surry plant. Also, it happened at an overseas plant, not in the same kind of pipe geometry, not in a steam generator shell, but it happened in another piece of pipe. And we were quite interested in learning more about the mechanisms involved and approached Research to start some work in that area.

But we ran into some trouble, as you will hear later, in getting a user's need to our side of the house. And the basic point being a certain amount of difficulty in proving through the PRA process that this is an extremely important safety issue. So we're interested in your insights on this as we present the technical information that we have.

With this introduction, I would like to ask B.D. to

carry on.

MR. LIAW: First I would ask Mr. Hazelton to give you a detailed presentation regarding worldwide experience, history, phenomena for the mechanism of the cracking. And afterwards I would make a brief summary with regard to some internal discussion on where we would like you to concentrate and to give us advice, as Dr. Johnston just mentioned.

Mr. Hazelton.

MR. HAZELTON: Good morning.

(Slide)

There seems to be an extra viewgraph here that I don't think I will need.

I want to talk this morning -- is this working all right? I want to talk this morning about three main events, and I am going to give you an overview on each of them, stressing the details of the occurrence and the conclusions that were drawn by various and sundry people regarding them.

The first I am going to talk about is the case that happened at the Garigliano plant in Italy, where the primary side of a steam generator in a dual-cycle BWR sprung a leak.

Now, this, for those of you who may not understand it, this is dual-cycle BWR like Dresden-1 that has both steam generators and makes steam directly in the reactor vessel. So this was on the primary side of the steam generator.

Then we will discuss first the Indian Point-3

incident in 1982, where we had thruwall leak in one steam generator with extensive cracking in the other three, and this required extensive weld repair, long outage, very extensive, very high man-rem exposure.

Then we will bring you up to date on what happened at Surry 2, where there were no leaks but there was extensive cracking in all three generators but shallow enough so the cracks could be ground out.

MR. SHEWMON: Warren?

MR. HAZELTON: Yes.

MR. SHEWMON: If a crack meets the surface, you can grind it out; if it's an internal one, you can't?

MR. HAZELTON: No.

MR. SHEWMON: There was some talk in here about internal cracks also.

MR. HAZELTON: Oh. Oh.

MR. SHEWMON: Or subsurface.

MR. HAZELTON: Subsurface.

MR. HAZELTON: Well, I think I will get into that in a little bit.

MR. SHEWMON: Fine. Okay. Go ahead.

MR. HAZELTON: We're talking mainly today about cracks that occurred on the surface of the inside of the vessel. There were just normal fabrication cracks subsurface, and we're not going to dwell too much on them.

MR. SHEWMON: Well, but my notes here say that there were at Surry subsurface cracks that were unacceptable indications. And if there are code unacceptable indications, I would like to talk about them even if they were subsurface.

MR. HAZELTON: Fine. That's no problem.

MR. SHEWMON: Good. Okay.

MR. HAZELTON: What I am giving you here now are some viewgraphs that were prepared at the time. The main reports on Garigliano are still labeled proprietary, so I am giving you stuff that is nonproprietary.

(Slide)

When it first happened, EPRI was, of course, very interested, and Carl Stanlkopf prepared a short presentation to pressure vessel advisory group. This is what I am going to show right now.

(Slide)

This was very early in the investigation, so we just wanted to give the story. 150-megawatt PWR designed by GE, two-loop fuel cycle. And power operation began in 1964. And this was, of course, 1978.

But on August 8 they heard a loud whistling noise, and the plant was shut down. They took a look, and they saw a thruwall leaker in the steam generator head well.

Perhaps the best thing for me to do now is to show you what we're talking about.

This is the steam generator. I am pushing this up so you can see over this stuff. Is that what you want?

MR. SHEWMON: That's fine.

(Slide)

MR. HAZELTON: This is down here in what we would call the channel head or water box. It was this weld that leaked. This was a final closure weld because they didn't have a furnace large enough to get the whole thing into it after making that weld. So this was the final closure weld. And then they did a local post-weld heat treatment, and this is the weld that leaked.

The leaker kind of came through in sort of a funnel shape if you look at it in this mode. But then there were extensive other cracks associated with it that we will go into.

I guess we've covered that -- no, sorry.

General Electric obviously went over there and took a look and saw what the problem was. They did a UT of the leaking weld in the area around, but they found seven additional cracks. And it says here that within a few millimeters of going through the wall, that all eight cracks were transferred to the weld, they said at this time.

But additionally, later, they found circumferential cracks that were stated to be over half the circumference of the generator and up to about halfway through the wall.

MR. SHEWMON: Presumably, those two were different, or what's the difference in the definition between -- transverse is transverse to the weld, is that right?

MR. HAZELTON: Right. Right. So there were cracks both transverse to the weld and circumferentially.

MR. SHEWMON: And then a little bit above that, the additional cracks were presumably transverse if they weren't labeled circumferential? Okay. I will wait for that.

(Slide)

MR. HAZELTON: Now, a little later, General Electric made a presentation, and this information is from that presentation.

The steam generators were designed by General Electric but they were manufactured by Stork in Holland, ASTM-A 302B, manganese alloy steel, the same things that we make reactor vessels out of and so forth.

One difference was that instead of the cladding of the water box being stainless, as is our present practice, this was clad with Monel 140. The other things were similar. And the closure weld, local post-closure weld heat treatment is done by induction. And here they talk a little bit about the extensive circumferential transverse cracking.

All the cracks initiated on the inner surface. The Monel cladding was extensively cracked both intercolumnar and intergranular. It looked like stress corrosion cracking,

although there was some argument about it. It could just have been shrinkage cracks.

The base metal cracks were transgranular, randomly branched, and they came to the conclusion it was environmentally assisted corrosion mechanisms -- whatever those words mean -- and that there was fatigue crack growth but it was not the primary cause of the observed cracks.

What was interesting, they calculated that the stress intensity factor for -- and I don't know what these words mean -- for a crack penetrating the clad -- after the crack penetrated the clad, the calculated stress intensity factor was 40-45 KSI root inch. That's important because --

MR. SHEWMON: Leave that on for a minute and let me try to get the picture, will you?

Monel clad --

MR. HAZELTON: Yes.

MR. SHEWMON: -- ferritic base, they feel that the cracks started as corrosion-assisted cracks, they may have been aided in their propagation by a fatigue mechanism?

MR. HAZELTON: Correct.

MR. SHEWMON: Is that it?

MR. HAZELTON: That was what was stated there.

Now, this was very early in the investigation. As I say, we have some later reports after about two years of investigation. The later reports, although they are much more

voluminous, don't really shed much additional information. They still waffle around about the causes.

MR. SHEWMON: How did they repair this? Did they cut the thing out and replace --

MR. HAZELTON: TRhey did not repair it. They just -- I don't know whether they valved the mouth, but they just took the two steam generators out of the system and didn't use them anymore. They didn't repair them.

MR. SHEWMON: Okay.

MR. HAZELTON: This gives another kind of a picture of it. The cracks are a bit confusing because the cracks were circumferential, but in one location where it appeared there might have been some weld repair. Cracks were found that were transverse to the weld and went all the way through.

This is the weld, and then the back cladding after the weld was finally made is in this area, and it appears that a good bit, or most, of the cracking occurred in the back cladding. But of course, that's also right adjacent to the weld. And cracks were found -- it doesn't really say that here -- in the weld, in the heat-affected zone, and some that appeared to even outside the heat-affected zone.

MR. DILLON: Warren, I gather that the issue of the Garigliano steam generator isn't very important but only in terms of how it might be related to the more modern steam generators for PWRs. And I guess my real question is that

there will be no cladding on the shells of ordinary PWR shells?

MR. HAZELTON: That's correct.

MR. DILLON: And I wonder what the relevance of these Monel clad systems might be. There's obviously going to be a substantial perturbing effect.

MR. HAZELTON: I wonder also.

The other thing that I might mention is that General Electric at that time was, of course, very concerned about the relevance of this problem to the normal BWR pressure vessel. The materials were the same with the exception of the cladding. So they took a look at the situation, and they measured some residual stresses at least on the outside surface, and then from that they calculated what the residual stress would be on the inside surface where the cracks started. And their calculations were about 45 KSI tensile stress. Operational stresses, they indicated, were very low.

So they came to the conclusion that the main stress that was of importance was the residual stress due to the weldment. And they also came to the conclusion that the local post-weld heat treatment, which was done locally by induction was right next to the heavy tube sheets, was done -- well, in an inadequate fashion. That is, they assumed that the temperature was not as high as it should have been and therefore you did not relieve the residual stresses from the

welding. As we go on, you will see that that is an important thing.

Again, to point out what General Electric was concerned about, thinking in terms of crack growth and thinking in terms of their reactor vessel, they had some information -- and I don't know exactly where this came from in 1978 -- but that the K1SCC would be approximately 20-30 KSI at 8 ppm oxygenated water but it would be approximately 25-45 KSI root inch in .2 ppm oxygenated water. Then they said they calculated the K1 at Garigliano using the high residual stresses. They calculated this as 40-45 KSI root inch.

So therefore, they feel that they can explain why in that case, at least after the cracks got through the cladding, they continued to progress through the wall. And they also then did some calculations again to show how this related to BWR reactor vessels.

The caluclations indicated that they could justify, they could predict the failure at Garigliano using the same model. They said that they would have no problem with their reactor vessels for the full lifetime of the plant, primarily because they said their reactor vessels, of course, were given good post-weld heat treatment, so they didn't have the extremely high residual stresses caused by the welds.

MR. SHEWMON: Warren, GE has replaced some of the piping in their primary system with unclad ferritic material.

MR. HAZELTON: Yes.

MR. SHEWMON: Can you remind me where that stuff is? Is it in valved-off areas only, or is it in the --

MR. HAZELTON: No. No. In the core spray systems, which is not valved off. And I am not sure, it seems to me that the RHR system, where they have the carbon steel, it may be valved off, but there may be areas that are not. I really can't remember.

MR. SHEWMON: Well, the core spray line is what diameter?

MR. HAZELTON: Generally, ten inch, I believe.

MR. SHEWMON: And if that goes, it just douses the core?

MR. HAZELTON: What do you mean, if it goes?

MR. SHEWMON: If it ruptured, would it be a DPA or is it just an awkward afternoon?

MR. HAZELTON: If it ruptured, steam would come out of it, yes.

MR. SHEWMON: But is it inside the core?

MR. HAZELTON: No, no, no. I'm talking about the core spray piping to the vessel.

MR. SHEWMON: So it's a ten-inch rupture, ten-inch pipe rupture in the primary system outside of the pressure vessel?

MR. HAZELTON: That's what it would be. If you had

it, that's what it would be.

MR. SHEWMON: Now, those get post-weld heat treatment in the field?

MR. HAZELTON: No. No. Those are made of different alloys, lower-strength mater'al. It's basically, what, ASME-106. And that's low carbon, low alloy, essentially no additional alloy. So the yield strength is about 35,000, and you just don't get as high residual stresses or as high hardness, and the code says that material does not need post-weld heat treatment.

Now, this could be a somewhat controversial subject. We could argue. But the point is, the 302B that we're talking about here is higher alloy and therefore builds up higher residual stresses during welding and harder microstructure. So the code does require post-weld heat treatment.

MR. SHEWMON: Fine.

MR. HAZELTON: Okay.

MR. WARD: Warren, let me ask you, the Monel cladding is unqiue and apparently that did crack a lot. Are the explanations for the cracking there? I mean is it necessary to have cracks initiated there? I guess it is, to have the cracking in the vessel wall. But is there -=- what's the explanation for that?

MR. HAZELTON: Obviously, if the cladding maintained

its complete integrity, you'd never get water to the vessel wall. Because it was assumed that the cracking was stress corrosion or corrosion fatigue, therefore it was environmentally induced and you had to get the water to the vessel. And if the cladding hadn't cracked, why, they might have been all right.

Of course, we never take credit for the cladding completely protecting the metal. That has never been the purpose of the cladding or anything else.

But the problem is that if you get a crack in the cladding, then you have a crack in the component and the fracture mechanics stress intensity factor can be pretty high. If you don't have any crack, you don't have any stress intensity factor, essentially.

So with this type of analysis, you can assume that for any significant cracking of the alloy steel, you need to have some kind of a crack present. And the crack in the cladding would do it. We could go into a lot more detail with it later.

MR. DILLON: Well, are cracks prevalent in stainless-steel clad on the reactor vessels?

MR. HAZELTON: We don't really know how prevalent they are. They try to avoid them.

The conclusion -- well, shall we say the position that has been taken is that the cladding on the reactor vessel

is only there just to keep down corrosion product, it is not there to protect the vessel from the environment, and it is assumed that it is cracked or has defects in it, thruwall defects. It is not assumed that it is perfectly intact. That has been their position that both the reactor vendors and the NRC has taken since the beginning. It is assumed to be subject to cracking.

I have -- oh, maybe --

MR. LIAW: Warren, before you start Indian Point, can I make a statement here?

MR. HAZELTON: Yes. Sure.

MR. LIAW: Although we are talking about the steam generator shell cracking, the subject is really broader. As you can see why we spend so much time on Garigliano.

What we are saying is once you have a combination of environment and stress conditions to even the low-alloy steel, yet there is a potential for cracking, either stress corrosion cracking or corrosion fatigue, the difference being the stress in cycles natural, there is strong implication to even BWR vessels where having rust removed for some repair or even steam generator shell or even steam generator PWR secondary side in general. So it's not just limited to the steam generator shell cracking per se.

MR. SHEWMON: Now, the oxygen content of the steam generator secondary side can be relatively high or relatively

low depending on how tight they keep their condensors, is tha it?

MR. LIAW: That's right. That's right. Also, you know, conductivity and all kinds of contaminants.

MR. SHEWMON: Okay.

MR. DILLON: You're talking about Garigliano?

MR. SHEWMON: I was talking about PWRs.

MR. DILLON: Okay. All right.

MR. LIAW: Bob, what I was saying was the subject really is low-alloy stress corrosion cracking, not just limited to the steam generator shell.

MR. HAZELTON: Most of the good pictures that I have from Garigliano are in the proprietary thing. But as a matter of fact, I did have an opportunity to stop in at the site while they were doing the investigation, and they had taken out four-inch plug samples and they kindly let me photograph them. So here are the pictures, two of the pictures, of those plug samples showing the cracks in the cladding and then extending down the side of the plug into the 302B material.

(Slide)

Okay. Now, we can talk about Indian Po nt-3. The leak was discovered. Again, a very similar situation to Garigliano, although I don't think there was a whistling noise. But they discovered a leak in the shell of the steam generator.

MR. SHEWMON: How did they discover it?

MR. HAZELTON: Well, I can't remember the details, except they discovered it by seeing water dripping out of it. They knew they had some kind of a leak, but they thought it was in some seal somewhere else or something. Then when they were shut down, they saw water coming out of it, and then they decided it must be leaking.

This is just a quick overview.

(Slide.)

A boat sample was removed from the inside, and then they did ultrasonic magnetic particle inspection of the weld. And they found many cracks, something like over 100 in each steam generator. They removed a six-inch plug, which was sent to Lucius Pitkin by the utility to do some studies on, and then later we got it or the remains of it and had Brookhaven National Laboratory do some investigations.

The cracks essentially were ground and repair-welded, and post-weld heat treatment was given, obviously at the local --

MR. SHEWMON: You didn't talk about whether these were transverse or circumferential.

MR. HAZELTON: Oh, I will. This is just an overview.

MR. SHEWMON: Okay.

MR. HAZELTON: Okay. They're all circumferential.

Now, the recent inspection, they thought had some cracks, but they finally determiend that they don't have any surface cracks. The thing that they have are some fabrication defects.

Just to orient you, now we're talking about a PWR steam generator, and we're talking about this weld here. It's a transition from the upper cylinder to the lower shell, so there's sort of a stress concentration there, the change in section.

This is also the final closure weld as made. Some of the early Westinghouse steam generators were shipped in two pieces to the field, and this weld was made in the field with the local post-weld heat treatment.

Later versions, including Indian Point-3, were -- the final closure weld was made at Tampa, where they had the facilities to handle the whole thing. And of course, it depends on shipping problems also. But the resulting compmonent was too big to put in the furnace, so a local post-weld heat treatment was done.

MR. SHEWMON: Now, were all the steam generators at Indian Point -- you said this -- you're talking about two now?

MR. HAZELTON: No. They have four. They all cracked.

MR. SHEWMON: The cracking occurred at Indian

Point-3. I am confused on which Indian Point plant it was.

MR. HAZELTON: Indian Point-3. I am sorry. Did I misspeak?

MR. SHEWMON: Okay. You said Indian Point-3 was done at Tampa for heat treatment.

MR. HAZELTON: Yes.

MR. SHEWMON: And now you're saying you should have said Indian Point-3 was welded in the field?

MR. HAZELTON: No.

MR. SHEWMON: Indian Point-2 was welded in the field?

MR. HAZELTON: Yes.

MR. SHEWMON: But the cracking is at Indian Point-3?

MR. HAZELTON: Yes. I might have misspoke. I don't know.

MR. SHEWMON: Okay.

MR. HAZELTON: Some steam generators, this weld was made in the field, others were made at Tampa. In the case of Indian Point-3, the one that had the bad cracking and the leak, that weld was made at Tampa.

MR. SHEWMON: Interesting.

MR. HAZELTON: In the shell.

MR. SHEWMON: Because apparently the Number 2 gave them a lot of trouble, according to Glen Reed, who remembers back to some of those.

MR. HAZELTON: Yes.

(Slide)

To give you a quick orientation of what we're talking about, this is a nasty weld to make, in the first place. They had a lot of repair welds associated with it.

Now, the welder is inside this metal component. It's very difficult to keep things cool and dry in there. The humidity is high. It's difficult for him to work. And you have a final closure weld where you have to take care of the mismatch of the two cylinders. They're not, you know, perfectly round. So they have a problem with that particular weld.

In this case, this is a cross-section through that plug sample where the leak occurred. This is the original weld here. This is the upper shell and the transition, and you can note the massive repair weld on the IV at the location where this crack went through the wall.

MR. SHEWMON: Were there any cracks in that?

MR. HAZELTON: Yes. At this point in the section, here is one crack here. Okay.

MR. SHEWMON: Yes.

MR. HAZELTON: This particular plug sample was riddled with cracks going a lot of different directions.

MR. SHEWMON: And that crack then is not in the weld?

MR. HAZELTON: No. That crack is not in the weld.

MR. WARD: But this repair weld you're talking about is where all the cracks were and they --

MR. HAZELTON: No, no. No.

MR. WARD: Or is in the original --

MR. HAZELTON: The cracks were sort of generally -- there were a lot of cracks inthe weld, but there were cracks in the base metal in the heat-affected zone. The important thing -- one of the important things that we're going to keep harping on, the residual stresses caused by things such things as massive repair welds and --

MR. WARD: When you say "repair welds," you mean done in the fabrication of the vessel?

MR. HAZELTON: Yes. I was going to get into that a little later.

MR. WARD: All right. I will wait. That's all right.

MR. HAZELTON: Part of the problem when making that weld, as I said, there's high humidity in there. They had a little trouble keeping their electrodes dry. And they had a problem with hydrogen cracking associated with the welding. So they had a lot of repair welds associated with this particular final closure weld caused by the difficulty of making the weld in the inside of the component after it's all done.

(Slide)

This shows a slightly different cross-section of that same plug sample, but this shows -- well, I hope that it would show this dark-looking mass of junk here is actually the hole where the leak occurred. So it had a crack -- now, you see the crack is here and it's -- well, partly in the weld, partly in the heat-affected zone and so forth. And then there were other -- porosity and so forth all through this section. So the water just found a path through here and continued to crack or corrode or push, erode its way through.

So this is a cross-section of the leaker.

MR. WARD: Which is the inside one?

MR. HAZELTON: This is the inside. So it came from this way, out there.

MR. SHEWMON: That hole was there from fabrication?

MR. HAZELTON: No, no, no, no. We're not saying that.

MR. SHEWMON: The crack grew as it got outside. You know, it got fatter as it went outside.

MR. HAZELTON: Oh, of course. Yes.

MR. SMITH: It eroded. Yes, steam erosion caused that.

MR. HAZELTON: Yes. Yes.

Now, it's very difficult for us to go into a lot of detail, but a lot of the detail on this is written up in

reports that, well, I am sure you have and can get more detail on. But we can discuss it.

Dave Smith is familiar with this. He was the main reviewer on this subject. He is here, and we can go into a lot of that detail.

(Slide)

But this particular plug had a mess of interconnected and unconnected cracks in it, and it's significantly different from the rest of the place. One can say that we were lucky that they made a massive weld repair in one spot and screwed the stresses up so bad that we could get a thruwall leak there and therefore recognize that we had a problem all the way around.

Anyway, early in the investigation we called Westinghouse and said, "Hey, what's the situation here? What's the generic aspects and so forth?" And they gave us -- I was going to say "song and dance," but that's not nice -- they gave us a presentation to the best of their knowledge. They really didn't know.

They said possible causes were hydrogen cracking. Again they discussed their problems with the fabrication. Pop-in from thermal shock. They really didn't think that was right. Fatigue from design transients or fatigue from thermal transients. They really didnt know what the problem was. Of course, this was before much was done.

But Westinghouse did send a letter to the owners of 15 units, and they recommended that inspection be performed. And they said then we'll see what to do.

At the time, NRC sent out an information bulletin telling people about it and suggesting that they might like to take a look. We never, of course, have gotten any answer. We don't know how many people really looked and how much looking they did or what the results were. We assume that if they found some cracks, they'd tell us. But more about that later.

(Slide)

Westinghouse conclusions were there is insufficient information to rigorously explain the origin of the cracks; fabrication appears to play only a minor role. Of course, as fabricators they feel that way.

(Laughter)

MR. HAZELTON: The best estimate of the cause is corrosion fatigue resulting from abnormal thermal cycles in environments. That leaves it kind of wide open.

(Slide)

The utility, of course, was much more interested in getting the thing back on line, but they did have an analysis done by Lucius Pitkin. And this is sort of a summary of their conclusions as to comments on the cause. One, design -- whatever that means. It probably had some arrangement of weld cracks. Residual welding stresses. Heavy subarc welds.

And this is important, they felt there was inadequate preheat and interpass temperatures. Obviously, when you've got a welder inside this thing, you want to keep it as cool as you can get away with. So it's clear they weren't going overboard on preheat and interpass temperature. More on that later.

Then they had high and heavy major weld repairs. Inadequate post-weld heat treatment. They came to the conclusion that the post-weld heat treatment was probably inadequate. Then thermal cycling.

Now, at this time we have Brookhaven doing analysis for us, and we did get some samples. We did get eventually the plug sample. And there is a report, a NUREG report on the subject that I am sure you have that goes into detail on what Brookhaven found.

What we feel is probably the most important thing that they found as a result of this was the fact that when you take the material out of the plant and then give it a post-weld heat treatment, another one now, after giving it -- I forget how many hours it was -- but at a thousand Fahrenheit, it perhaps dropped a little on average, not much, but after giving it 1125 Fahrenheit, the hardness dropped significantly, and that happened with the heat-affected zone.

The weld metal is particularly significant. So this, of course, tells the metallurgist that, in the case of

the weld metal, it probably didn't even get up to a thousand Fahrenheit in the post-weld heat treatment.

MR. SHEWMON: What was the welds they found in the samples they had -- sorry, the hardnesses? Is that on?

MR. HAZELTON: Hardness is on here.

MR. SHEWMON: As received --

MR. HAZELTON: That's right.

MR. SHEWMON: -- as when it came to the lab?

MR. HAZELTON: Yes.

MR. SMITH: From the boat sample.

MR. HAZELTON: Yes.

MR. WARD: Warren, would you put up the previous one there, the heat-affected zone? Are those little dots?

MR. HAZELTON: These little dots are test points. You've got a big range into --

MR. WARD: And what do the lines mean?

MR. HAZELTON: This is the average, this is the average, and we just put lines in here to --

MR. WARD: Oh.

MR. HAZELTON: -- just for --

MR. WARD: Okay.

MR. HAZELTON: -- to help you analyze the situation.

MR. WARD: Okay.

MR. HAZELTON: What you would normally do is, obviously, if you ran a test to find out how the hardness was

affected by heat treatment, you'd run a series of heat treatments and you'd draw a curve and say there's that's --

MR. WARD: The best you could do is draw an envelope or something.

MR. HAZELTON: That is about what he has. He's got his envelope there.

MR. WARD: Is that the best metallurgists can do, Paul?

MR. SHEWMON: Pardon?

(Laughter)

MR. SMITH: If you take a look on the left side, you'll see these are new hardnesses, which is a very small diamond penetrator hardness. So you have a very localized hardness reading. And so there tends to be a wide scatter. And that's why if you look there on the extreme right on the left side, you can see it's quite a wide range, and that's why you have to go to an average on these things.

The heat-affected zone is a very narrow zone in a weld, and you have tremendous metallurgical changes. And this is how we approached trying to show that there was some response, and you could almost say that at the thousand degrees S from as received, there was no real change. The line is almost level in the heat-affected zone.

MR. DILLON: This profile, this is the heat-affected zone of the weld, and then you've got -- oh, I've got it.

MR. SMITH: It's not a profile.

MR. DILLON: It's weld metal and then base metal. All right.

MR. SMITH: Yes. It's not a profile. It's not a traveler.

MR. HAZELTON: One way to try to find out whether it was given a temper or something to see what hardness is and then give a different indication.

MR. DILLON: I just didn't look ahead to see you had the other two graphs. Sorry.

MR. HAZELTON: Yes. Right. So the base metal, there wasn't much change in the base metal, but some.

MR. WARD: Now, the significance of the thousand degrees, was that the alleged heat treatment temperature?

MR. HAZELTON: Yes. I hoped to get into that, but, yes, the important thing is that the code requires a minimum of, say, a thousand Fahrenheit. This is somewhat controversial, frankly.

MR. WARD: Yes. Well, it looks like that's not a very good number.

MR. HAZELTON: Partly as a result of this and stuff I will talk about later, we've proposed that the minimum post-weld heat treatment for this type of material should be higher. Okay. That is part of the thing we're trying to do is -- but the code does permit 1,000 Fahrenheit, and there's

still a problem when you're doing something like induction heating of a local weld all the way around a big component like this, you don't know whether you really get a thousand degrees everywhere. It's not all that easy.

Just to summarize the Brookhaven conclusions in the analysis of the Indian Point-3, all cracks were transgranular, the hardness indicates the possibility of high residual stress -- that is, you assume that if it's hard, therefore you didn't have a good post-weld heat treatment, therefore you didn't get a good stress relief either.

On the fractography, they found some possible fatigue indications, some possible fatigue striations, but it was a bit confused, and they couldn't be terribly positive. But they felt that it looked, in general, like a low-cycle corrosion fatigue phenomena. But they really weren't able to determine the relative importance of the corrosion and the fatigue. So that's what they came up with.

MR. DILLON: Warren, would you comment briefly on to what extent is this representative of the practice at the time? Is this an unusual situation?

MR. HAZELTON: No.

MR. DILLON: Or is it approximately what you'd --

MR. HAZELTON: In my opinion, no.

MR. DILLON: I see.

MR. HAZELTON: You're talking about the fabrication?

MR. DILLON: Yes, and particularly about the heat treatment, the post-weld heat treatment.

MR. HAZELTON: No. I think it was typical.

MR. DILLON: And do you have any idea what the duration of this post-weld heat treatment presumably at a thousand degrees --

MR. HAZELTON: Somewhere in our records we probably have something on that detail, but I don't know it right at the moment. But that can be dug out possibly. Sometimes all you really know is what they ask to be done.

MR. SHEWMON: Warren, let me come back again. My notes here say that these welds were made in the field. We established that that was not true. They were made in Westinghouse's factory?

MR. HAZELTON: Yes. You're talking about the Indian Point-3 now.

MR. SHEWMON: So unless they screwed up on their control, that's the way they've been doing it since then and that they're --

MR. DILLON: What I was more interested in was the possibility that the welds in the field would even be more screwed up. And that raises a question. You know, we got that -- let me --

MR. SHEWMON: You talked about weld repair, and you said, "Gee, that may have been done because the guy was doing

it in the field."

MR. HAZELTON: No, no. I said --

MR. SHEWMON: But it was done actually back in Westinghouse's plant. The guy was in Tampa --

MR. HAZELTON: Oh, Yes.

MR. SHEWMON: -- inside this container trying to stay dry and cool.

MR. HAZELTON: Yes, sir. He was inside the component.

MR. SHEWMON: Which in Tampa is not easy at any time.

MR. HAZELTON: Inside a metal house in nonair-conditioned Tampa, a boiler shop. Together with preheat and very high humidity.

MR. DILLON: Do you think there's enough left of that transition weld on our Surry steam generator at Hanford to be worth looking at?

MR. HAZELTON: Yes.

MR. DILLON: You know, the thing is cut off right at that transition.

MR. LIAW: Warren, no. You said, "Yes," but the answer is, "No." In the past meeting this past spring, I recall that the steam generator you got at Tampa did not have the girth weld. The girth weld stayed in Surry. That's the reason why Surry --

MR. DILLON: Well, they cut it right off, didn't they?

MR. LIAW: No. It goes below the gird weld.

MR. HAZELTON: Okay. Let's wait until we get to Surry, okay?

MR. DILLON: Right.

MR. SHEWMON: Before you get to Surry, I've got several questions I would just as soon take up on Indian Point.

MR. HAZELTON: Okay.

MR. SHEWMON: You tell me when you're at the end of Indian Point.

MR. HAZELTON: I am at the end of Indian Point.

MR. SHEWMON: Okay. If we talk about the steam separator, the big part of the -- take that screen out of the way and get to a blackboard, will you? A professor can't work without a blackboard.

I guess you've got a figure, actually, that showed this. There are two welds to that cone. What you're talking about, is it the bottom one or the top one?

MR. HAZELTON: The top one.

MR. SHEWMON: And that's where it was -- that's okay, then, he's covered it. And that's where it was also in Surry?

MR. HAZELTON: Yes.

MR. SHEWMON: Do you have any idea why the thermal stresses would be higher up there than at the next one lower?

MR. HAZELTON: The only thing I can say is the water level is supposed to be up in here. There might be a problem with water level sloshing. The feedwater ring is right in here. Sometimes they'll put in cold feedwater just slightly above this weld.

This has been kicked around a lot. But every time we suggest that this might be a problem, they say, "Oh, no, we've calculated it and so forth, and there's no big deal."

MR. WARD: So it's not the stresses that are different, it's the environment, you're saying?

MR. HAZELTON: We don't know. Okay. We suspect both.

MR. SHEWMON: If the mechanical engineers could just calculate stresses, we would be okay.

(Laughter.)

MR. WARD: If this is the art, why aren't there a lot more leaking steam generators around the country?

MR. HAZELTON: Gee, I don't know. Just lucky, I guess.

(Laughter)

MR. HAZELTON: You're talking about generic. Let's talk about -- can I wait until the finish of Surry?

MR. WARD: Sure.

MR. SHEWMON: Okay. Well, let me bring up a different thing. Let me note that I guess you don't necessarily have it -- Al wrote it up -- but it certainly came from notes you had. There is talk about code-allowable cracks here. Could you remind me what a code-allowable crack is in these?

MR. HAZELTON: Well, you were talking about code-unacceptable. Well, there's both.

MR. SHEWMON: Okay. You tell me what's acceptable. It will probably tell me what's unacceptable, too.

MR. HAZELTON: Well, I don't remember. I don't remember details. But in any weld, there's certain small imperfections that are code-acceptable, there are some that are not. Obviously, cracks, linear indications really are not acceptable. They have found a lot of these in both the steam generators, and we've been finding them in some brand-new ones like at Byron, Braidwood, and so forth.

MR. SHEWMON: Let's come back. This is two inches thick?

MR. HAZELTON: Well, this is --

MR. SMITH: Three and a half.

MR. HAZELTON: Three and a half.

MR. SHEWMON: Three and a half. Okay.

MR. HAZELTON: Three and a half to four inches thick, depending on --

MR. SHEWMON: And the way the code must define what's allowable is what fracction of the wall thickness and how long?

MR. HAZELTON: That's primarily section 11. We're talking section 3, any fraction is -- any indication of a crack is not acceptable.

Dave, do you have any good numbers offhand of what would be acceptable?

You know, you're talking slag or you're talking -- you have to talk specifically.

MR. SMITH: Well, you konw, these things -- one of the problems is that these units were built originally to radiographic standards, and afterwards the UT was imposed by section 11. I do not know the criterion. I think it has changed a couple of times. Of course, a surface defect is much more restrictive than a subsurface defect by terms of size, and the sizes are estimated by the reflection of the UT signal and a percent of DAC.

But the particulars past that point, I'd want the code before I talk.

MR. SHEWMON: Well, can the utility come back in and say they don't have to do a UT because they meet the 1971 code and it didn't have UT in it?

MR. SMITH: No. The section 11 has imposed the UT requirement.

MR. SHEWMON: And section 11 applies now to the secondary side?

MR. SMITH: This particular joint, yes. Certain joints, the ones with the structural discontinuity such as this particular transition cone to upper zone, there is a structural discontinuity because it's at an angle, an angle joint, and it is a part of the ISI. So it does have to be periodically inspected, and these particular steam generators have windows where they're supposed to look at them periodically, yes.

MR. SHEWMON: This is windows in time?

MR. SMITH: Windows on how much of the weld they're supposed to look at. They don't have to do 100 percent. It's spread out between the four steam generators and these sort of things.

MR. SHEWMON: So once in ten years they have to look at something unless it meets, is that the --

MR. LIAW: At least.

MR. SMITH: Yes. They are usually broken up so that they might look at a third every other refueling outage so that they constantly are monitoring. It isn't all one shot at ten years. There's a constant monitoring.

MR. HAZELTON: I just want to caution you, they don't have to look at the entire weld in ten years.

MR. SHEWMON: I mean they have to look at something

every few years?

MR. SMITH: Yes. That is true.

MR. LIAW: A section of the circumference.

MR. HAZELTON: Obviously, for section 11, UT is usually preferred, although the code specifically says volumetric. So radiography would be code-acceptable.

MR. SHEWMON: But if it's section 11, then can you tell me how deep a crack can be before it's unacceptable, or can we get back at that?

MR. LIAW: Okay, we can get back to that. If I can just give you a rough idea.

MR. SHEWMON: Yes.

MR. LIAW: As I recall, for vessel, containment vessel, you're talking about an order of one-tenth or 12 percent of the thickness of the wall.

MR. HAZELTON: Right.

MR. LIAW: Okay. Roughly. And I don't recall exactly the number for the Class II steam generator shell, but we can look it up.

MR. SHEWMON: Also, the length must be several times the depth for the --

MR. LIAW: No, there are no criteria on length per se if you look at the code standard.

MR. SHEWMON: Okay.

MR. LIAW: Just depth.

MR. HAZELTON: Actually, the depth that's allowable is a function of the length, but it isn't a strong function when you're talking about fairly long cracks. So you're talking about subsurface cracks could be up to around twice as deep as surface cracks because they're --

MR. SHEWMON: Let me move to a different point. Again, where these near the feedwater nozzles, or were they necessarily away from or at or around the --

MR. HAZELTON: If we can put this thing back on again.

(Slide)

There was a -- do you happen to have that top view showing the feedwater? The feedwater nozzles are somewhere in here. Okay.

MR. SHEWMON: Yes.

MR. HAZELTON: It shows it -- I don't happen to have a viewgraph that shows the location.

MR. SHEWMON: Okay. But are the cracks --

MR. HAZELTON: Then the spargers is a funny-shaped thing in plane view.

MR. SHEWMON: Okay.

MR. HAZELTON: Maybe we can pass that around.

MR. SHEWMON: So there's one feedwater nozzle --

MR. SMITH: And it goes through the ring. These spots here represent, I believe, the crack locations. This is

the original report. And it shows that there was no correlation with the geometry of the rings.

MR. SHEWMON: Okay.

MR. SMITH: Here's another steam generator, and here they darkened in where they had the original weld repairs from the plant, and then, of course, the blended areas, this is the smooth inside surface on Indian-3.

MR. SHEWMON: So that the main point is there is no correlation between the position of the cracks and the position of the feedwater nozzle?

MR. HAZELTON: Except generally.

MR. SMITH: That's right. We could not see any particular orientation.

MR. SHEWMON: Okay. Fine.

MR. HAZELTON: Okay.

(Slide)

Then we have the problem with Surry. Surry, the steam generators were replaced in 1979. However, this particular transition weld was not replaced, and they cut the vessel -- they cut the shell about here, and replaced the bottom and then made another weld here.

Obviously, when they made that weld, they did a post-weld heat treatment. And so the post-weld heat treatment they gave during that repair, that replacement, accomplished some additional stress relief and so forth of the weld we're

talking about.

Now, during this flap on the Indian Point-3 thing, people did take a look, and some people looked better than others. Virginia Power did a pretty decent UT examination, and I can't remember whether they inspected all three steam generators or not. It could be that they did. But at least they inspected one of them.

They found some UT indications. And the conclusion of the examiners was that it was caused by geometry on the inside associated just with the shape of the weld on the inside.

However, some of our Region II inspectors thought they looked pretty cracked, so there was a bit of an argument. It was agreed that they could run for another cycle if they'd commit to go in and do some mag particle, magnetic particle inspection on the inside, the next outage.

So they did. And so in the 1985 outage they went in, and you have to cut some of the hardware inside there really to get in. So they - it's not easy. So they went in and did magnetic particle, and lo and behold, they had cracks.

So they proceeded to investigate the thing, and basically they found that the cracks could all be taken care of by grinding out because in the case of Indian Point some cracks were about as deep as 2-1/2 inches. In the Surry thing, probably the cracks weren't any deeper than half an

inch.

(Slide)

I have again some pretty pictures. Here are three pictures taken of the inside of one of the Surry steam generators, and I am told that the cracks -- you couldn't see the cracks unless they were very tight, unless you polished it off and looked at high-power visually. These are outlined by the mag particles.

MR. SHEWMON: The welds or the guy running the grinding wheel is told to go in half an inch. There's no way he can see that crack when he's in there, is there?

MR. HAZELTON: No. You have to do a mag particle to see if it's gone. That's right. Obviously, where he's supposed to grind is marked off. It's marked. He says, "Here, grind here." So he grinds there. Then they do a mag particle to see if it's gone.

(Slide)

Now, if you'll notice, on one of those you'll see a grinding mark. It may be a little difficult to see where they tried to grind it out. They ground, I forget, an eighth of an inch, and it was still there after the grinding. But that gives you a kind of a picture together with the scale that's there of what they were up against.

MR. DILLON: Warren, in the case of stainless steel, obviously, why, you're pretty careful about the surface

condition of the material if you want to resist the stress corrosion cracking kind of problem. Now, what about carbon steel in terms of grinding artifacts and so on?

MR. HAZELTON: Well, I really can't see how it's going to help it any, but I don't know of any real cautions on it.

MR. SMITH: Warren, there's quite a difference between Indian-3 and Surry-2. Indian-3, you could not see any remnants of the weld at all. It was completely ground so it's a smooth curve from the upper cylinder to the transition column. Okay. There were no remnants of the weld at all. It had been well filled and ground.

At Surry-2, the crack occurred right at the toe of the weld on the inside, the ID, and you could still see the remnants of the weld. In other words, it turned out, and there'd be a little weld crown and then there'd be another little sort of like an undercut, you know, where it entered into the transition cone down at the bottom.

So there's quite a difference between Indian-3, which was thoroughly ground, and Surry-2, which had minimal grinding, if any at all. Indian-2 is the same way as Surry-2; it has the weld crown on the inside visually, so there's not much grinding done on it.

MR. DILLON: I assume that the grinding was done on both the weld material and the heat-affected zone.

MR. SMITH: On Indian-3, definitely, yes.

MR. DILLON: But not on --

MR. SMITH: Surry-2, I would say, was relatively untouched.

MR. DILLON: All the grinding was in the weld material?

MR. SMITH: Well, I don't think there was much grinding done on the ID of Surry-2 at all in the original fabrication.

MR. WARD: Are you saying that grinding is the culprit somehow?

MR. SMITH: No, I am saying that I don't think there is any correlation between that.

MR. SHEWMON: What Dillon brought up was that grinding does have a bad name in stress corrosion cracking in 304 stainless steel in BWRs.

MR. SMITH: Okay. I think Surry-2 shows this thing will happen without it.

MR. SHEWMON: And so if the staff thinks the residual stress plays a significant role here, it's a little bit hard to see why they couldn't feel that cracking would have some -- well, if one thought that residual stress is the culprit, then cracking can leave -- sorry -- grinding can leave residual stresses.

MR. HAZELTON: Presumably, you've given it a

post-weld heat treatment that will accomplish a reasonable amount of stress relief after the grinding. But that isn't a requirement, and in carbon steel nobody really worries about it. But I don't know of any studies that have been done on that at all.

If you notice, on the pictures you've got fits over the general area, and you have the same situation at Indian Point-3 with heavy pitting. There are cracks that appeared to initiate pits and grow from pits.

(Slide)

I am showing you now viewgraphs that were prepared by VEPCO, and I want to go through these. They repaired these welds by grinding in all three steam generators. They only went after the surface flaws only, not the subsurface weld fabrication flaws. They used detailed repair procedures and very careful grinding.

We were discussing with the how they were doing it, and they did an extremely good job. They MT-inspected after each increment of grinding, and the final surface was then MT-clear.

In steam generator A, they were able to remove all the surface flaws by removing half an inch. In steam generators B and C they only had to go 31/1000dths of an inch -- or hundredths of an inch. But in A there were several places where they did grind deeper than they had to just

inadvertently.

Their conclusions, they're talking about two things, really, what were the problems and why are things better for them now. And they said they did have secondary water chemistry problems prior to the replacement. The oxygen was often more than 25 ppb. Chloride was often in the ppm range, occasionally 300 to 400 ppm. They had copper alloy condensors and feedwater heaters, so they had plenty of copper items around.

However, they're saying, since they have replaced the steam generator and the condensor, that the water chemistry has been very good. So they are hoping that they're not going to have a continuation of the problem.

They also discussed what they considered significant construction problems. The preheat on the ID, when the ID welding was being done, 180 to 185, which is on the low side. Even the OD was perhaps on the low side a little bit.

The post-weld heat treatment was accomplished, they said, at a thousand to 1,100. At least that's what the records say. And they were also concerned that that might have been very marginal heat treatment.

So the fact is that they say that mitigate reinitiation of cracking after they've got them ground out, that they have much better water chemistry and lower residual stresses at this weld because of the post-weld heat treatment

of the weld 6; that is, the replacement weld just below it. It was done at 1,150 to 1,180, and they said that that temperature range covered this weld with the cracks in, so it now has a better post-weld heat treatment finally.

(Slide)

Here is sort of a summary of their conclusions: that they had bad water, again; cracks, they feel, initiated from pits; the pitting was caused by the bad water chemistry; and they had cracks progress by stress corrosion or corrosion fatigue. Then the cracks were propagated by corrosion, static stress, operational and possibly residual and fatigue. So you can take your choice or add them all up, I guess.

They feel that preheat was on the low side, and the weld sequence may have left the inside diameter in tension and the post-weld heat treatment left high residual stresses.

I just want to give you a quick summary here of the cracking in Indian Point-3 and Surry. This we tried to put together, Dave Smith tried to put it together from the records we had, which was very difficult. So it's sort of approximate. This is the length in inches.

Correct, Dave?

MR. SMITH: Correct. In 528-inch circumference, the number of inches that were excavated, and of course, you have 711 inches excavated in 528, so what you had was parallel cracks. So in some places there were two or three cracks

overlapping.

MR. HAZELTON: But let's give you a point. On these four steam generators, cracks ranged, the deepest cracks ranged from an inch and a quarter to two and five-eighths, except, of course, that one that punched its way through. These are a percentage of the circumference that had problems.

Surry-2 was similar in the extent of the cracking, perhaps a little less. However, the cracks were much shallower and, fortunately for the utility they could be repaired by grinding out and still have minimal wall that met the code requirements.

MR. THOMPSON: Is there a typographical error on those percents? Those are 77 percent, not .77?

MR. HAZELTON: That is correct.

MR. THOMPSON: Okay.

MR. HAZELTON: Okay. Quick conclusion. I might say that the Materials Engineering Branch has an awful lot of thoughts on this subject. This sort of is a quick summary.

(Slide)

We think that the cracking in Indian Point-3, Surry-2, and Garigliano are all related. The cause is most likely a combination of stress corrosion and fatigue. Contributing factors, we feel, are low post-weld heat treatment temperature that gives relatively high hardness in

the material, which makes it more susceptible to the stress corrosion phenomenon, and also leaves it with high residual stresses.

The environment, apparently, if you have oxygen in the water, it's bad. There's tests that have been run showing that at high stress intensity factors you can crack the stuff just with pure water just with oxygen in. But of course, in the real world you have copper ions, sulfur species and chlorides and a mess of other junk in there that makes it much worse.

We feel that obviously there is stress cycling just from shutdowns and startups and so forth, and heatups and cooldowns. We're not sure but what there might be some impact of the way the cold feedwater or the feedwater comes in close to this transition weld. But the transition weld has a lot of problems anyway.

We feel that the information that is available in the literature and that we've been able to glean from this is not good enough for us to try to quantify any of these synergisms. So it's going to be very difficult for us to predict where we might have a problem in plants, which steam generators out there might have a problem or which ones we wouldn't think would have a problem. And we don't think that we have enough information to make these judgments.

Another information notice is being sent out as a

result of the Surry situation. The most important thing there that's being emphasized is that the UT that was done did not identify these cracks. They had to go in and do a magnetic particle on the inside to find out that these were cracks and not just geometry. So that's, hopefully, the purpose. That's the purpose of the information bulletin, to alert people to that case.

I will let B.D. talk about some of the things that we've done after that. But you have on your pass-outs, I tried to give you some cover sheets and so forth of pertinent reports that you should have or we can help you get that give a lot more detail in this area.

MR. DILLON: Warren, before you leave, let me go back over the actual reconstruction of the Surry steam generators where you attached the steam separator sections to the base part of the steam generator.

MR. HAZELTON: Okay.

MR. DILLON: Before you established that the old weld material was left on the steam separator portion.

MR. HAZELTON: Right.

MR. DILLON: And the question I am really raising is the condition of that remade weld on the existing steam generators atypical of those systems that have not undergone a rebuild?

MR. HAZELTON: I think this particular transition

weld, it's the final closure weld on all the Westinghouse plants, I see no reason that it's any different, and I don't have any details that will tell me that we don't have the same kind of problems in all Westinghouse steam generators. Okay. And I don't think -- I am trying to understand your question a little bit better.

MR. DILLON: Well, the point is that, as I understand the situation, presently steam generators or the final closure weld is made at the Tampa plant -- let me continue.

MR. HAZELTON: There might be some problems with shipping, so they have to --

MR. DILLON: Well, but did they ship these radioactive steam generators -- or rather, the --

MR. HAZELTON: No.

MR. DILLON: -- steam separators back to Tampa?

MR. HAZELTON: No, no, no.

MR. DILLON: Okay. That means they made this transition weld at the Surry plant?

MR. HAZELTON: That is correct.

MR. DILLON: Okay. So it's not typical of what presently is being made.

MR. SHEWMON: Nothing is presently being made. But it's also not typical of what else is in the field.

MR. DILLON: Yeah. Okay.

MR. HAZELTON: That's right.

MR. DILLON: Okay. But the fact is that this has undergone two welding operations in essentially the same region because it was made up once and then again.

MR. HAZELTON: I forget, but there's something like -- oh, I don't know, I will guess -- six or eight inches away. So it might be a foot away.

MR. DILLON: Okay. Now, then I wanted to ask about the thermal cycling problem, which you can conceive of situations which would be perfectly easy to explain the cracking problem based on thermal cycling alone as a result of the feedwater injection. I guess my question is, does anybody have any real knowledge of what the cycling temperatures might be under operating conditions? Is this known at all?

MR. HAZELTON: I don't think we have a lot of detailed knowledge on that. At least I don't. But clearly, sometimes you can have a problem with feedwater heater or they may have fluctuations in the feedwater temperature. And we haven't looked at that in a great deal of detail. This is part of the bigger study. And I think B.D. will talk about that a little bit more.

There is a question whether or not this is a generic problem, how much resources should we spend on it, et cetera, et cetera.

MR. WARD: Warren, a couple of questions. When this

final weld is made on a steam generator at the factory, is the post-weld heat treatment that's done there local induction?

MR. HAZELTON: Local induction.

MR. WARD: Okay. So there may not be any big difference?

MR. SMITH: It is not induction, it is a flame.

MR. HAZELTON: We were told once it was induction, but now apparently that has been corrected.

MR. WARD: Okay. But it's a local heating.

MR. HAZELTON: Yes.

MR. WARD: Okay. What sort of heat treatment is used when that final weld is made in the field? What was used?

MR. HAZELTON: I was told it was local induction, but they could have used resistance blankets in some cases, the Cooper heat type.

MR. WARD: Okay. So there isn't necessarily any big difference in the factory heat treatment and the field heat treatment?

MR. HAZELTON: I think the important thing is, to be honest, they were aiming to get a thousand Fahrenheit.

MR. WARD: Now, when you talked about after the cracks were found at Indian Point, then what I heard you say was all licensees were instructed or requested or something to look for --

MR. HAZELTON: No. We went out an information report --

MR. WARD: Okay.

MR. HAZELTON: And those things don't instruct or require anything. They might --

MR. WARD: Okay. They're informed and --

MR. HAZELTON: They suggest.

MR. WARD: -- some of the chose to look. Apparently, VEPCO looked harder than anybody else did.

MR. HAZELTON: Well, that isn't true. We know that several utilities did look very carefully. We know they looked very carefully at Ginna, at least they told me they did. In fact, they did mag particle at Ginna. But our information is just, you know, if people want to tell us, why, they tell us.

We don't really know how much they did, but we suspect that very few of them went in to do mag particle, because that's a very messy thing to do. They probably just did UT.

MR. SHEWMON: My agenda at this point says we have a ten-minute break. And since I chide Igne for not putting these in very often, I want to encourage him by taking this one.

(Laughter)

MR. SHEWMON: We will come back at quarter after.

(Recess)

MR. SHEWMON: Why don't you go ahead, B.D.?

MR. LIAW: I will try to summarize the staff activities regarding this issue to provide you with some internal discussion among staff as to how we're going to deal with this issue and, hopefully, be able to point out the area where we think some advice from you is necessary in order for us to pursue further.

(Slide)

The first viewgraph I have is regulatory responses. If you recall, in 1982 after Indian Point-3 cracks were discovered, the NRC issued an I&E Information Notice 82-03 on September 15, 1982. As Warren mentioned earlier, we also initiated a quick study at Brookhaven to look into the cracking morphology of the boat sample taken out of Indian Point.

In the meantime, as we learn more about European experiences sometime in '83 -- before '83, as a matter of fact -- we thought it was necessary to propose some research work. I will get back to this viewgraph again.

(Slide)

To have some kind of continuity, I will use Warren's last viewgraph. As you can see, there are several operating mechanisms going on simultaneously and the synergistic effects among all contributing factors are not really well understood

or well quantified. As a result, we thought a research work ought to be initiated to quantify this.

So we generated a research user need request, and that was sent over from Division of Engineering to Division of Systems Safety -- or Safety Technology, rather. And then they requested a prioritization evaluation.

I will discuss some of the main conclusions there later on.

After Surry cracks were found, again we issued another Information Notice 85-65 on July 31, 1985, principally to point out the difficulties in UT examination, because there was some discussion or disagreement among Region II inspectors and the licensee inspectors. As a result, MT was used to verify the UT indications.

About this time, after Surry, the regulatory prioritization was completed in response to Division of Engineering's earlier request for research work.

We also proposed on July 10, 1985, the issuance of an IE bulletin which in great extent would require some kind of response. And then on August 22, 1985, we received a response from Hugh Thompson, who is the director of Division of Licensing. I have a letter, a memo from Hugh Thompson to Jim Knight to distribute to you. I will discuss that briefly later on.

The result of the evaluation, first they conclude

that the issue is a low power issue because in terms of PRA or probability to contribute to coremelt probablility is rather low, we had some problem with the issue of whether or not the regulation has been met, whether it is consistent with a defense-in-depth philosophy underlying the regulation. And some further discussion internally among EST and DE, we got issue we classify as a licensing issue. That means some work can be initiated, although not on a high-priority basis.

Principally, the result shows that it is more of an economic issue than a safety issue. They proposed a writeup for Mr. Denton's signature, although Mr. Denton has not decided whether he agrees with the result of the evaluation. I might as well mention that when they complete the evaluation and when Mr. Denton decides what he wants to do and he will send the -- he will include that in the NUREG-0933 and will request ACRS comment.

What I am suggesting here, you will have an opportunity to look at the numbers, and if you want to comment on it, I encourage you to do so.

MR. SHEWMON: Is it clear that Denton will take a position on it, or is it one of these things that he can just not take on?

MR. LIAW: He will have to take a position on that because it is an Action Item in terms of request for research.

Let me give you some of the basic reasons why it is more of an economic issue than --

MR. WARD: Mr. Liaw, can I ask you a question first?

MR. LIAW: Yes.

MR. WARD: You say who prioritized this?

MR. LIAW: Division of Safety Technology, Warren Minogue's group.

MR. WARD: Okay. This hasn't been identified as a generic issue yet?

MR. LIAW: They did include that as a generic issue.

MR. WARD: Okay. So they're treating it and have prioritized it as one of the generic issues.

MR. LIAW: Right. The number is 111.

MR. WARD: All right.

MR. LIAW: The reason it is classified as more of an economic issue, you can see by looking at this number the degree of cracking at Surry are much less than Indian Point. As a result, the costs associated with fixes, okay, are tremendously different. I will give you some numbers, although, you know, these numbers are rough estimates. You cannot see in any documents.

For Indian Point-3, without additional forced outage that can occur replacing power costs is something like half a million dollars a day. Don't count that. Indian Point-3 repair, grinding and weld repair, post-weld heat treatment,

altogether costing something like $8 million, approximately 800 MR. For Surry-2 --

MR. WARD: Do you mean man-rem?

MR. LIAW: Yes. At Surry-2 they discovered cracks at a much earlier stage of the cracking. They need only to do grinding to satisfy the code requirements. The cost is about $1 million plus about 25-30,000 man-rems.

From that you can see that it is a severe economic issue in terms of the problem. At what stage you might discover and on what basis we thought we are not only regulating safety but we also regulate the occupational exposure.

Let me discuss the response from Hugh Thompson. I will refer you to the letter I passed out to you, and call your attention to the fourth paragraph. The memo is dated August 22. If you remove the word "perhaps" and "the possible reduction," if repairs are required, take the "if" out and replace it with "when," then the first sentence in the fourth paragraph characterizes the problem very well.

It would read, "The major benefits to be gained from the proposed inspections apart from potential economic benefits are reductions in future occupational radiological exposure when repairs are required compared to what would otherwise be required in inspections conducted at this time."

So that characterizes the issue very well. If we

require some kind of an inspection now and if we let them continue to operate and discover cracks later on, the cost difference is tremendous, as indicated by the number I just have passed out to you.

MR. WARD: B.W., the cost for the Indian Point repair was, did you say, $8 million?

MR. LIAW: Yes.

MR. SHEWMON: Now, what's the difference between the inspections they are doing now and the inspections that were proposed?

MR. LIAW: The inspection they are doing now is just on a voluntary basis.

MR. SHEWMON: No, they're committed to an ISI, which means they do some inspection every few years.

MR. LIAW: Yes. Every few years. But as we know, stress corrosion cracking is time-dependent, and every few years means some weld may not get inspected for as long as ten years.

MR. SHEWMON: Fine. What I asked you was what is the difference between what they do now and what was proposed? Are you proposing 100 percent be required at some point, or what?

MR. LIAW: That is again internal discussion. We had in mind if you do an inspection now at the first opportunity, at least 50 percent of the circumference of the

one steam generator weld and the previous inspection report show mass indications, magnetic particle inspection should concentrate on those areas; two, if cracks are found, the steam generator should be declared inoperable -- that means they have to repair it before they continue to operate it.

And then we say, "The result of this inspection should be reported by telephone to NRC Operations Center within 48 hours after inspection has been completed. A recent report describing the area inspected and the results should be submitted within 30 days after the inspection is complete."

More in the regulatory sense -- and again this is just what we proposed in August, some of our colleagues did not quite agree with us. Okay. And that you might consider further and see whether or not you could offer some advice.

MR. WARD: Was there a difference in the lost operating time for between the Indian Point repair and the Surry repair?

MR. LIAW: No.

MR. WARD: There wasn't.

MR. LIAW: Cracks basically has --

MR. WARD: Was there lost operating time in either case?

MR. SMITH: Yes. At Indian Point-3 it was quite extensive. I don't have the dates because they ran into another problem with their generator. They burned out a coil

in their generator and they lost more time additionally. It was never broken up. But the repair of the Indian Point-3 steam generators was over an extensive period of time; I mean, something like on the order of six months or something like that.

Surry-2 was on the order, I would say, weeks. We're talking about two or three weeks, and I don't think it was on the critical path. There were other things going on at the time. Never extended their outage. Indian-3 was certainly extended.

MR. WARD: I see.

MR. LIAW: Indian Point-3 at that time also had a problem with pitting, so they include additional outage. As Dave indicated, we did not have a breakdown of how much was contributed by the shell cracking and how much was contributed by the pitting. And that is the reason I did not want to use the number. I just wanted to give you a rough estimate of the cost.

MR. SHEWMON: Do you believe that these things will leak before they break, if in fact the leak does come through?

MR. LIAW: We believe so. As we are aware of the situation of piping, it is a possibility.

MR. SHEWMON: If you put that into your PRA calculation, then you end up with no contribution to coremelt frequency?

MR. LIAW: That's right.

MR. SHEWMON: Because you have to get a rupture before it starts to be a significant safety challenge, is that right?

MR. LIAW: That's right. And because a sudden break of the steam generator shell would be similar to a main steamline break, which is within the envelope, the design basis. ˆ d they were saying that steam generator shell rupture can be in the envelope of the main steamline break.

MR. SHEWMON: It would be exciting, though.

MR. LIAW: Yes. Frankly, my personal view is --

MR. WARD: I don't know how detailed the prioritization calculation is, but there must be a difference in, let's say, the CE design, let's say, and a four-loop Westinghouse design. There are only two generators.

And then on top of that, if for some reason you lost two generators, some of the newer CE plants do not have the capability to -- I hate to say the word but -- feed-and-bleed. Was the analysis, the risk analysis, sensitive enough to look at that sort of detail, or would you come to any conclusion? Is there a possibility that a conclusion for a CE system 80 plant would be different for a Westinghouse four-loop plant, let's say?

MR. LIAW: It could be, because CE had only two steam generators.

MR. WARD: Plus the newer ones don't have the ability to feed-and-bleed.

MR. LIAW: I am not aware of that.

MR. SHEWMON: At least from the outside, CE's steam generator would look the same as Westinghouse's, wouldn't they?

MR. LIAW: Yes.

MR. SHEWMON: Is there any experience of cracking at the same geometrically similar weld there?

MR. LIAW: We have not received any report of cracking from CE generators. They have not experienced severe water temperature problems. I would expect that if they have some kind of a similar weldment, they might have it. And that's the reason when we proposed the bulletin, we did include the CE plants. In fact, we included all recirculating-type steam generators in it.

MR. SHEWMON: Now, during their ISI, what would they have to find before they'd need to report it to you?

MR. LIAW: Well, any finding of seeing the code-acceptable -- that's again --

MR. SHEWMON: And code-acceptable was 10-12 percent, you thought?

MR. LIAW: Something like that, yes. Somehow an MT signal equivalent -- that is translated to 10 to 12 percent. I will look up the number for you.

MR. SHEWMON: Okay.

MR. LIAW: That number is for vessel, you know.

MR. JOHNSTON: Dr. Shewmon, if I could just comment on it, there are two things that B.D. is talking about. The first one is our desire for a bulletin which would ask the utility to make an inspection, I think it's at their next outage, if I remember correctly. That would ask all of the utilities with steam generators to give us an up-to-date status of what they've got.

And what you're hearing is a reflection of the fact that we don't know exactly what they've got out there because they don't have to tell us. This would essentially be a request that we would get this information about Palisades and other such places.

Also, the problem is that it's very much like in the BWR case: they're not finding it with conventional UT inspection, just as we weren't finding them in the BWRs for years either, and yet all of a sudden somebody, when we began to experience it, then they began to find it.

In this situation, we seem to be still in the pre-mode where nobody's found it and they don't expect to find it, and lo and behold they don't find it. And now we're trying to get people in the mode of maybe you ought to expect it a little bit and maybe question your UT results a little bit more than you have been in the past.

So that's sort of in the mode that we're in, sort of like we were when the BWR stuff was just beginning to be observable.

The second part of our interest in our discussion with you today has to do with our desire to understand more about the mechanism of this, the conditions and so forth under which these things take place. And that's the essence of our research request which has gone thorugh the mill and is coming out as a low priority. So it's two items that we're getting dumped on, if you like.

MR. WARD: To what extent is the latter -- I mean, to understand the mechanism -- to what extent is your concern over whether something similar could occur in a reactor pressure vessel?

MR. JOHNSTON: We'll, it overlaps, and it's an area which we don't have very much information on, we nor the industry.

And it's the kind of thing that we feel Research, really that's a function that Research has in our type of organization, to take the little bit longer look at it in the laboratory sense, put together the information that we have from the field, and do some appropriate work in the laboratory to learn a little bit more about, you know, has the main problem got something to do with the original heat treatment, and if that's the case, how much does the subsequent water

chemistry have to do with it, if anything, plus the combination of the superimposed thermal cycles. So you put all three of those things together, is one of them dominating, in which case we can focus on one type of thing; if not, we've got to worry about all three of them. But we don't know that.

MR. WARD: It sounds to me like there are two concerns. One is Research, and one is the inspections from the licensees. And you're saying that the risk analysis, unless you think it could be somehow tied in to the reactor pressure vessels, you don't think risk analysis is telling you there's is a major safety concern?

MR. JOHNSTON: Well, as long as the leaks --

MR. WARD: If the licensees are willing to risk the money to wait until they get leaks, that's not your business then, but you think you may have a reason for it to be a regulatory concern because you want to avoid the 800 man-rem?

MR. JOHNSTON: Well, that's certainly part of it. I think the other is a prudence factor in that BWR pipe cracks, if you do a PRA on them, don't come out super-high either, but you're well aware of the flap and everything that we had over it.

And therefore, there are things which you sense there's something wrong and something needs to be done even if the case -- even if things like leak-before-break are operational, which, you know, if you're paying attention to

what's going on, you shouldn't really get into a serious problem because you'll find it out beforehand. But that's always a little bit of an iffy thing.

MR. HAZELTON: I would just like to make one comment regarding the relationship to reactor pressure vessels. The main concern, as I mentioned, that GE had was how was the relationship of this problem -- could you correlate it with reactor pressure vessels?

And what they determined was that even with a crack through the cladding, you would have to have a very high stress to get a stress intensity factor enough to drive it, and because reactor vessels, because they're smaller, they can be given a post-weld heat treatment, full post-weld heat treatment in a furnace, when they're completed. And normally, that is on the order of 1,125 or above. So they didn't think that they would have a problem of the high weld residual stresses in a reactor vessel.

There is only one possible fly in the ointment, and that is that the code permits some local repairs without post-weld heat treatment using this half-feed technique, which in the HSST program we pointed out causes high residual stresses in the area. And that's one possible mechanism for getting a local high residual stress situation that could be related to the reactor vessel. I just wanted to point that out.

MR. SHEWMON: Let me come back to what Bob Dillon said, that you don't really need poor chemistry if you get enough thermal fatigue in here. Talk to me a little bit about the feedwater nozzle cracking problems that we had a few years ago on PWRs because of stratification or basically alternating hot and cold water in there. Does that necessarily mean that you'd alternately throw hot and cold water on the inside of the vessel? Is that part of the thermal stress problems?

MR. HAZELTON: We don't really know. It could be related. But they tried to -- we tried to take a look at where the cracks were in relation to where the water was coming out of the feedwater ring. We didn't see any relationship there. That doesn't necessarily mean there isn't a global relationship.

We are also concerned about where the water level was in relation to this. And if you noticed those pictures in Surry, you saw some deposits several inches above this cracked weld that indicated the water level was kind of bouncing around in there, but all above it. But we don't really know the relationship of that.

MR. JOHNSTON: My memory of the discussions we were having at Indian Point when these issues were raised was that the water level was supposed to be above the transition piece in normal operations.

So that whatever the temperature of the incoming

water is, it doesn't -- it would only be modifying the overall water temperature slightly. The direction of the spray and so forth when it comes in is away from the wall. It's not designed to spray in that direction. So we were given several reasons why they didn't think that had much to do with it. The water sprays toward the inside of the vessel rather than toward the wall, plus the water level is supposed to be above it anyway.

Now, you know, whether, as Warren points out, under all operational modes it actually stays up there all the time, we have no exact way of knowing. But the nominal is clearly above that point.

MR. SHEWMON: Are you finished, B.D.?

MR. LIAW: Yes.

MR. SHEWMON: All right.

MR. WARD: May I ask one question? Are there girth welds in the once-thru steam generators that could have a similar problem -- I mean, in the B&W steam generators? Or do you have some reason to believe this is only the --

MR. HAZELTON: Well, there aren't any of these transitional things where there is a difference between two cones, this could happen. But we haven't looked at the B&W situation. We haven't had any particular reason to. We thought about it a little bit but kind of assumed that there probably is not a problem there.

MR. LIAW: Besides, after the completion of the verification, all of it got heat treatment.

MR. WARD: That is what I was going to ask. They are smaller, so they could be heat-treated as a whole.

MR. LIAW: Yes.

MR. SHEWMON: Thank you very much.

We turn next to irradiation assisted stress corrosion. The presentation that we will hear will apparently be aimed primarily at BWRs but there have also been crack events in cracks in treaded fasteners inside PWRs.

I will ask the staff what they know about that or what they think about it, too. I don't know. So I don't really want the presentation just to be about boilers. If the answer is, "We don't know nothing," I guess that's an answer for now.

MR. WILLERTZ: Good morning, gentlemen and ladies.

Paul, thank you very much for inviting me down here to talk about a subject which I became acquainted with about a year and a half ago. I became acquainted with the subject about a year and a half ago when we were looking at obtaining hydrogen water chemistry for a boiling-water reactor, the Susquehanna steam electric station in particular.

In the presentation to us, they were trying to sell us hydrogen water chemistry. It became evident that there was

another phenomena to be taken into account that we were not aware of.

Subsequent to that presentation to us, we made an internal study on the subject of irradiation assisted stress corrosion cracking. Unfortunately, I don't have a lot of information on pressurized water reactors, Paul. I apologize for that.

My job was to look at our own reactor and determine whether or not there was a problem with this particular phenomenon and to try to mitigate it if there was a problem, and to look at some of the safety issues that might come up as a result of this phenomenon.

In my presentation then I am going to review a little bit about intergranular stress corrosion cracking and relate that to irradiation assisted corrosion cracking.

(Slide)

The two phenomena are very closely related. If one were to look at the micro structure of an IGSCC-type crack in material in the stainless steel and look at an IASCC-type crack in the material, I don't think there is any expert in the world that will be able to tell you the difference between the two. Therefore, an understanding of IGSSCC is necessary to understand a little bit of what happens in IASCC.

I will also talk a little bit about the use of stainless steels in reactor, which is basically what we're

talking about when we're talking about in this phenomena, where the components are subject to irradiation assisted stress corrosion cracking, give some examples of parts that have cracked in the past, and look at some of the programs that have been developed over the not-too-distant past.

It's been investigated for probably not much longer than about five years ago, although irradiation damage in materials, of course, is very common and has been well known for as long as there have been nuclear reactors.

The irradiation damage, however, in the stainless steels has never been addressed because there has never been any serious problems with those up until recently.

In talking about the various programs on IESCC, we will also point out the various conditions that one would expect in the different parts, and then give you sort of an overall view of what Susquehanna thinks about IESCC and what we are doing to try to alleviate any future problems that might occur if this phenomena were to appear in our reactor.

Getting into the discussion of intergranular stress corrosion cracking, I know a number of people here are familiar with this triad of components that make up intergranular stress corrosion cracking.

What one needs is a sensitized micro structure, a stress intensity of some sort or strain rate change in the material, and an environment that all match together to

produce a phenomena of crack'ng that parts the grains in the material and causes it to c- ck.

The investigation of this phenomena has occurred over quite an extended period of time, back into the early '70s and maybe even before. The intensity of the investigations has been such that we pretty much know today what the different conditions and what the boundaries are in terms of environment and mechanical and micro structural parameters are concerned which will cause IGSCC to occur in the stainless steel.

As far as the sensitized micro structure is concerned, we know, for example, that a carbon content in the material of around .03 percent is necessary to get you to a high enough carbon level so that carbides will precipitate at the grain boundaries and deplete the chromium that is in the vicinity of the boundaries to the point at which the chromium level gets below approximately 12 percent in that region. At that point the material is said to be sensitizedand it is sensitized to the extent that the environment that's present can preferentially attack the grain boundaries, and under the action of the stress that's present, the grain boundaries will pull apart, be degraded by the environment, and a crack will progress through the material.

The environment is the second paramater that is necessary to produce intergranular stress corrosion cracking,

and that environment has been documented to be basically any environment that contains very small amounts of impurities -- water, high-purity water containing parts per billion of impurities, temperatures in excess of 200 degrees Fahrenheit, and as high as 600 degrees Fahrenheit.

Oxygen levels in the environment generally have to be about higher than 20 parts per billion, and in a boiling-water reactor the normal operating level is 200 to 250 parts per billion of oxygen. So the boiling-water reactor, even though it has "a very high purity," if the environment is a susceptible environment that will attack the sensitized micro structure that we talked about and produce with the stresses cracking of the material.

Stress intensity or stress time or strain time is another part of the triad that's necessary to cause the material to come apart. There's a certain level of mechanical stress that's needed to separate the grains even in the presence of the environment in a sensitized micro structure, and this is generally believed to be somewhere around the yield strength of the material at the temperatures at which you're operating.

So these are the conditions under which one sees intergranular stress corrosion cracking. When some materials started to fail in boiling-water reactors in an environment -- in a condition that will -- that has produced cracking under

conditions that are beyond the normal limits of IGSCC, it was evident that we were dealing with something a little bit different than IGSCC.

And the common thing that seemed to come out of the investigation was that irradiation was present and that the parts that were known to get indications or small cracks associated with them were in high-irradiation environments, neutron irradiation and gamma fields.

So that common thread of evidence indicated that something was happening in the irradiation field that was causing material to become more sensitized than what it would be if it were not in the neutron or gamma radiation field.

We will see later on that even some tests that were carried out on irradiated materials that were irradiated in a reactor, taken out and given constant extension rate tests in a laboratory environment away from the radiation environment, still cracked.

Therefore, the residual damage that had occurred in the reactor was still present in the material. So there was something that had happened to the material, obviously, under the neutron radiation that would cause this material to crack.

It was also obvious from later investigations that stress intensity was another factor that seemed to be a factor that you could almost do away with when you had the neutron irradiation field because materials that were normally in a

very low stress condition and were thoroughly annealed prior to being put into the reactor, had also cracked under seemingly zero stress conditions.

The investigations that followed therefore tried to rationalize what was happening to the material under the conditions of this irradiation. It became apparent during the investigations that certain things happened to the micro structure which caused this to become sensitized.

Even though low-carbon variety stainless steels were found to crack in certain instances in the reactor environment, there were other elements that are present that could possibly be related to the sensitization of the material, not necessarily just carbon and chromium. But there were other residual elements present, such things as phosphorus and sulfur and silicon, that were found by independent investigations to, if they were low enough in quantity, to mitigate IGSCC, and therefore it was thought that maybe these had something to do with IASCC also.

So the radiation environment then has interacted with the micro structure of the material and the elements that are present in this material to cause it to become sensitized in the nonclassical sense of carbide precipitation. There was something else that was happening to the micro structure that was causing it to come apart.

The environment also is extremely different in a

reactor than it what it is outside of a reactor environment. The gamma radiation and the neutron irradiation will cause radiolysis to occur in the water, and radiolysis separates the hydrogen and oxygen molecules and produces nascent hydrogen and oxygen elements and ions which cause the environment to become activated and presumably a little bit more aggressive than what you would normally expect outside of the reactor area, say, in the feedwater system or in the recirculation system where the radiation levels are many orders of magnitude smaller.

So that environment is altered with irradiation, and that's the point to be made here. And it's made much more aggressive than what you would normally consider the high-purity BWR environment outside of the reactor core.

Stress intensity is another factor that seemingly was absent in some of the failures, and yet we all know, or those who are familiar with radiation damage in material know, that a considerable amount of stress can be imposed on micro structure just due to transformations in the micro structure. Boron will transform to helium and lithium and produce these elements which are insoluble in the matrix, will want to precipitate out. If they can't precipitate becuase of the temperature, they'll just sit there and cause internal stresses in the grains which will then be a source of stresses for the IGSCC-type of phenomena.

So externally applied stresses really aren't necessary to produce the IASCC phenomena. The radiation damage itself can produce by various mechanisms defects, interactions in the solids, formation of vacancies and dislocations. All that type of damage that occurs internally in material can cause internal stresses.

And these will then produce, along with the other two conditions, the necessary preconditions to produce an intergranular type of attack.

So we have not violated any of the rules that we've learned about; we've just altered them. The radiation environment has altered these rules and the boundaries have changed now. They are no longer the old boundaries of IGSCC; they've been expanded or contracted with the presence of the radiation damage.

(Slide)

This is a very busy slide, but it gives you a very good outline of what a boiling-water reactor looks like. It is a large vessel with carbon steel walls on the outside that surround and make up the entire vessel cavity. Lining the wall of the vessel itself is a layer of stainless steel which is applied by a welding process.

(Slide)

The reactor core area in a BWR is in this region down here.

All of the internal components in the reactor are either 304, 316, stainless steels, either low carbon or high carbon varieties, depending on what the structures are. There can be some Inconel, nickel-base alloys in there. They are in the minority. There is some Inconel-600. There is some X-750.

However, in the core area, pretty much you're talking about stainless steel parts or zirconium parts today. Zirconium makes up the fuel elements, the fuel cladding, and the fuel bundles are surrounded by a can of zirconium.

The components that receive enough radiation to cause IASCC to become a possibility are contained within the area between the top guide, the lower core support plate, and the shroud area.

Outside of this area -- and I have some numbers later on that will show you -- outside the shroud the neutron radiation level is much too low to see the IASCC phenomena as we know it today. So we are going to be talking about the various components that are in this area, and I will point out in a little bit more detail some of the structures that contain stainless steel.

(Slide)

This is a cross-section of a fuel assembly, but let me show you a picture first. It might give you sort of a bird's-eye view of the top of a core. And these are four fuel

bundles which are separated by a control rod. The fuel bundles, this one segment here is a blowup of a region that surrounds one of these pluses here. Each of the pluses here represents one of these large control rods. So that the four elements of fuel surround the control rod.

The control rod itself then, in a more prospective view, is shown here.

(Slide)

Where we're looking down at the fuel bundle at an angle, you can see these are the four fuel bundles as we saw from the top view. The control rod is partially inserted in this particular bundle. There is a bail handle on it. This is the upper core support. The upper core support has one opening for each four bundles of fuel and one control rod.

Other elements that we will talk about include dry tubes. A dry tube is not shown in here, but the hole through which the dry tube extends can be seen down in the bottom of the picture, and it extends up the side corner of one of the fuel bundles and is fixed at the junction between two plates of the top guide.

There are 12 of these dry tubes in the reactor, and there are a lot of -- there are neutron sources and other tubes that come into the reactor, but the dry tube itself is one of the elements that to date is showing some susceptibility to this cracking.

(Slide)

The fuel bundles thmselves do not contain any stainless steel. As I said, the fuel cladding today is virtually always zirconium alloy, as is the can in which the fuel bundle is embedded.

The component of the control rod that is stainless steel includes the outer shape of the rod as well as the upper portion which supports the control rod itself.

(Slide)

So today we don't have to discuss any stainless steel cladding on fuel that with earlier reactors did contain stainless steel in its fuel cladding. However, that was removed from the reactor for economy reasons. It has to do with the economy of neutrons in the reactor. Zirconium has a very low cross-section and so neutrons will not, shall we say, seize the zirconium as easily as it would seize stainless steel. So the neutrons are free to travel throughout the core and participate in the nuclear reaction as it was designed to do with an economical means of transmitting the neutrons. The stainless steel, having too high a cross-section, made it much more expensive to operate a reactor with stainless steel.

Because of the high fluences in these regions, a secondary issue was the possibility of cracking in the cladding if it were made out of stainless steel. So there is only a couple of reactors today that have fuel as clad with

stainless steel.

(Slide)

The design of the control rod here has stainless steel on the outside of the control rod, and there are also stainless steel tubes inside in which is embedded a boron carbide element which is used for absorbing neutrons in the reactor.

MR. SHEWMON: There has been some experience of cracking in some of those tubes. Is this outer skin of stainless that you're showing us enough to contain the B4C if the tubes do crack, or does the water run through those and around the tube?

MR. WILLERTZ: There are holes in these shades here for cooling purposes. The holes allow water to go in and circulate through the B4C bundles, and if there is a crack in the tube inside containing the B4C, the B4C can escape into the water.

MR. SHEWMON: Thank you.

(Slide)

MR. WILLERTZ: This is one type of control rod that has shown two kinds of cracking. One is the -- actually, three kinds of cracking. The control rods containing the B4C have been shown to crack after extended lifetimes, and the cracking has basically been due to swellingg of the B4C inside the tube. This causes stresses to occur on the tube, and so

after about ten years or so there is a higher probability of cracking in the tubes than earlier in its lifetime.

The fact that a control rod is removed from the reactor after about ten years or so means that if there is some leakage in the B4C, that will generally be taken care of during the changeout of the control rods.

A second kind of cracking that has been shown to occur has been down in these water inlet holes, and these have occurred down in this region or in the region here where there is a spot weld, and presumably produced a little bit of stressing in that area.

The thing about the cracks that were found in these control rods was that it was recognized to be an intergranular attack of some sort and labeled as IGSCC. However, there was no sensitization of the material. It was a low carbon variety of stainless steel that was annealed and therefore should not have cracked in a boiling-water reactor environment.

MR. SHEWMON: Is there a boron spec for this? Has anyone done a helium analysis?

MR. WILLERTZ: Not to my knowledge. I don't believe that there is added requirements for control of the boron. I think the stainless steels that are purchased don't have an exceptionally low requirement for boron. There are varieties which would be common out of a steel mill that would be specified as a low carbon variety stainless steel.

MR. SHEWMON: I asked somebody from England what experience they had with cracking in high exposure stainless steel, and he said they had found it in their advanced gas reactors which operates at an appreciably higher temperature, and they were having trouble with helium cracking, helium generating bubbles, and so I wondered, as I recall, the one they talk about, the helium coming partly from alpha reactions on the nickel and partly from the boron.

MR. WILLERTZ: There is discussion on that one paper in the handout that I referenced, and I think it's the second page there. It's a recent paper by some people at GE who discussed the various interactions that can occur, and the boron reaction with neutrons is one of the reactions that they believe can produce damage in the material.

However, they point out that the helium bubble formation does not normally occur at the temperature of operation of a boiling water reactor. As you pointed out, the

experiments you talk about were done at a higher temperature. And it's generally not found that this damage precipitates out or the helium precipitates out at the low temperatures of 550 degrees Fahrenreit that we operate at.

So the damage remains some place in the matrix or the microstructure of the material.

The other thing to point out about the cracking, the fluence that the control blade was exposed to was on the order of 2 x 10 to the 21 MBTs.

MR DILLON: Excuse me. Is the fluence the only part of it, or is the flux also thought to be a factor?

MR. WILLERTZ: I have not run across anybody who makes any discussion of the flux level as being a contributor. They only talk about fluence, that the total number of neutrons passing through the material is what really counts.

Now I would imagine that flux could be important in extremely high fluxes where you would probably locally produce extremely high temperatures in the microstructure under a high fluence condition where you didn't have enough thermal conductivity to bring the temperature down. Then you would be getting into a different regime of damage at the higher temperatures.

MR. DILLON: Presumably you could anneal some of this out.

MR. WILLERTZ: Under the normal conditions of operating a PWR, flux apparently does not have a significant contribution.

MR. WARD: Well, let's see, Paul. You probably remember this better than I do, but with the stainless steel walls at Savannah River, there is helium formed, I think, and that's at much lower temperatures, but I think -- isn't that related to the flux spectrum?

MR. SHEWMON: The helium -- he wouldn't say that the helium doesn't form, it's just that it's not discernible as bubbles as it is at higher temperatures. One can also talk about actual creep tending to sweep these things up or get into the grain boundaries, too.

So as you -- do you know of any reported problems at Savannah River that have been blamed on helium?

MR. WARD: Yeah, I think -- well, I don't know if we had problems, but we are concerned about being able to do weld repair in the regions where that high fluence --

MR. SHEWMON: Yes, there you get up to high temperatures again and that stuff might bubble.

MR. WARD: Oh. Okay.

MR. ODETTE: 300 degrees C is significantly below where there is believed to be -- there is considerable data from other radiation programs.

MR. WILLERTZ: Well, you're liable to get a lot of

cavitational kind of damage associated with helium. I might mention here, though, while we are talking about this, there is another phenomena that may be very important in this, and that is solute segregation, which at these fluences and temperatures and damage rates may be significant. I think under these conditions chromium segregates away from boundaries, and this may be a contributor to this phenomena as well as hardening.

MR. WILLERTZ: The GE paper that I cited has a brief description on the segregation of various solutes, both large diameter solutes and small diameter solutes, and discusses the segregation of lower diameter solutes to the grain boundaries and the migration of the larger diameter elements away from the grain boundaries.

So there is this segregation phenomena that takes place that is cited in there as being possibly one of the contributors to the IASCC phenomena.

MR. SHEWMON: There was a comment over here?

MR. JOHNSTON: I want to comment on the low temperature possibilities, too, because many years ago we made electrical resistivity measurements and used electron microscopes and at PWR operating temperatures we could see gas bubbles form.

Also one of the correlations I believe that Guthrie came up with in looking at the radiation damage to vessels,

the trend curves, one of his correlations that he discussed with us about a year ago, I believe -- I don't know if it's come out, but he was showing a boron correlation that seemed to fit better than all the other elements. He was playing with copper, nickel and everything else. And actually you look at the boron in the steels and he was apparently finding a better correlation to the transition temperature, which is your loss of ductility to trap rather low levels of boron. So that stuff never gets very hot, but if it does, it acts as a point defect, and interferes with dislocation movement and stuff like that. It doesn't have to be in big bubbles to get a mechanical effect.

So that might be some additional information.

MR. SHEWMON: Let me go back to your first point, which was that you said you did see bubbles at PWR operating temperatures in the electron microscope? Is that what I heard?

MR. JOHNSTON: Seemingly small ones. They're there.

MR. SHEWMON: And do they end up primarily on the grain boundaries?

MR. ODETTE: No, they tend to be fairly well distributed.

MR. JOHNSTON: If you anneal them.

MR. SHEWMON: Yes. Okay.

MR. JOHNSTON: That's 20 year old work.

MR. SHEWMON: I suspect the behavior of the bubbles hasn't changed a great deal.

[Laughter.]

MR. ODETTE: I would like to comment on the second point you make. I think -- I'm not quite sure that description of the correlation is exactly correct. I think that McElroy, with the help of Guthrie, did look for some thermal neutron effects which he hypothesized were associated with boron. They never had any boron measurements in the steel, so that they didn't even know what levels they were dealing with. It was really sort of a minor correction to the residuals.

I don't think it is correct to say that it took on the significance of these major elements, copper, nickel.

[Slide.]

MR. WILLERTZ: Another type of control blade that has shown some susceptibility to cracking has been the plate type control blade where the B4C is embedded in between plates of stainless steel in the form of plate instead of wide, like in the previous example.

In this particular case, we see again that the material should have been a material that was nonsusceptible to IGSCC. It was a low carbon stainless steel. Cracking was experienced close to or in the area of the B4C, and on investigation of the cracking in 29 blades, 10 out of 29 were

found to have some of these small cracks shown here evident.

It was in a high burn-up region where there was approximately 2 x 10 to the 21 nvt neutrons exposed to the material, and it was believed to be related to the fact that the upper portion of the control blade had been inserted into the reactor under conditions that one normally does not operate a PWR in this country. You generally have the control blade either all the way in or all the way out.

In these particular cases, the control blades were only partially inserted. For what reason, I'm not aware. But in any case, it was evident that in the high burn-up region where the blade was partially inserted into the core, that the cracks were evident.

So both types of control blades have been shown to be susceptible as long as the fluence is on the order of 2 x 10 to 21.

That again had occurred after about 10 years worth of service which is about the normal lifetime that one would experience on a control blade, and one would normally remove the control blade and replace it with a fresh supply of B4C and, of course, new sheath material, and so you would start essentially from the beginning again, as far as the IASCC phenomena is concerned.

So one important aspect of these indications that were found was that all of the indications were small, that

they did not result in any loose parts forming, and did not result in any abnormal operation of the control blade. The control blades performed their service and their function throughout their lifetime.

[Slide.]

Another part, which is the top portion of the control blade, has recently been shown to exhibit some cracking. This is the handle portion of the upper control blade, upper portion of the control rod.

In several cases, as far as I know at this point, I think there have been three incidents of control blades like this cracking in the handle region.

Again, there is evidence on the control blade that the material was properly designed and properly put together. It was annealed and was considered not to be sensitized with respect to IGSCC.

On the other hand, there were several cracks in this control blade because of the low stresses and because one part was missing, it became difficult to understand how several -- it was actually four -- cracks could form and come apart without presumably any human interaction.

It is not clear to me that there was not some grabbing of the upper portion of the control blade. Nobody is ever quite sure what happens when you find a broken piece like this, whether somebody inadvertently bumped against it when

they were trying to remove a fuel bundle or something like that. There are these things that happen all the time that generally don't cause any consequences, but occasionally something like this will show up, and then you have to consider, well, did somebody bump into it, did it embrittle and just fall apart by itself? I'm not sure that anybody really has answers to those questions, but we do know that there was a part missing in this particular case, and that it had cracked in pieces, in two other pieces, which were retrieved from the reactor eventually.

The fluence in this particular case was 4 x 10 to the 21 nvts. There was in some cases no evidence of cold work and no evidence of sensitization. The fractures were considered intergranular in nature. There was in some places some cold work, but it is possible that it could have been surface cold work. I'm not aware of the cold work being said to go completely through the microstructure.

MR. ODETTE: What does cold work on a fracture surface mean?

MR. WILLERTZ: The cold work would be on the surface of the material. If there were cold work on the fracture surface, it would generally be from some stressing that occurred during operation or during pulling on the part or whatever happened. If the two faces are bumped together at some time, then you would get cold work on the --

MR. ODETTE: So there's some indication of deformation on the surface?

MR. WILLERTZ: There is some indication of deformation in the microstructure near the surface, not necessarily on the surface of the fracture.

MR. ODETTE: I guess an additional point, one wouldn't expect -- I mean even if this was loaded in some transient way by grabbing or handling, normally stainless steel wouldn't crack this way. So it's been embrittled in some sense.

MR. WILLERTZ: That's right, it has been embrittled. It's been embrittled by the radiation, and because it wasn't sensitized and because there was only a small amount of surface cold work -- it has been shown, for example, recently that in some of the safe ends that have been examined recently from boiling water reactors that were replaced, that cold work was very evident around the welds, and even though the material was considered to be nonsensitized material normally, the solution annealed and should not have cracked by IGSCC.

The fact that you had cold work on the surface initiated a crack, and then the crack became what's considered a crevice, and the crevice environment became more aggressive and allowed the IGSCC to continue through the nonsensitized material.

MR. SHEWMON: You said a couple of times that this stuff is not sensitized, and you know there's a variety of techniques that are used to measure degree of sensitization. When you say that, do you mean somebody did careful metallography on it, or somebody stuck it in an electrochemical bath and didn't see anything by one of the cruder tests that ASTM used to approve?

MR. WILLERTZ: I believe the nonsensitization -- I don't know in this specific case whether or not somebody went in and did measurements. But, as you indicated, there are several different ways of making a measurement of sensitization, and crude or not, they have been accepted. Some of them, like the oxalic acid etch, is one. Electrochemical potential measurements is another. And visually looking for precipitated carbides in the grain boundary is another way of looking at it. And I believe careful metallography was done on the pieces in this case to look at the fracture surface and to look for precipitates.

Whether they did any of the chemical means of proving sensitization or not, I'm not aware.

MR. DILLON: Is there any proposal that the grain boundaries of some of this material be looked at for some of the sulfur, phosphorus kind of things that seem to segregate to grain boundaries?

I'm having trouble understanding intergranular

attack by a process which does not produce sensitization and which presumably does produce some sort of a more or less uniform distribution of precipitated second phase material. It just doesn't easily come to me about how you account for stress corrosion cracking or intergranular type under these circumstances.

MR. WILLERTZ: Well, the evidence is in that precipitation of some other elements may in fact be causing the sensitization. What we discussed before about segregation of these various impurities is one of the ways in which you can get the grain boundary to be more easily attacked by the environment than the grain itself.

MR. DILLON: And you're talking about some sort of an examination by auger or something of that sort to look for bulk surface contamination of grain boundaries?

MR. WILLERTZ: There have been experiments done both in Europe -- I believe KWU and Pacific Northwest Laboratories in this country -- Russ Jones, whom I'm familiar with, has performed some experiments concerning segregation of impurities, and those experiments are continuing. The evidence is not all in yet, but the evidence is strong that, for example, phosphorus and silicon are the two elements that keep coming up in the literature that seem to be the ones that cause the cracking, or at least produce conditions in the material which make the material susceptible to that.

[Slide.]

More recent cracking has been found in what's called dry tubes. Dry tubes are instrumentation tubes that come up through the bottom of the boiling water reactor vessel and pass between the boundaries between fuel bundles and are attached to the top of the top guide, or the bottom of the top guide, excuse me, to perform their function in that particular area, whatever that function is.

There are instrument range monitors and source range monitors are two elements that can be embedded in the bottom of this tube.

Now this is an enlarged section of the top of the tube. The remaining portion of the tube from this point on down is the longest section of the tube. This section is maybe only 10 or 12 inches long. The lower portion may be 10 or 12 feet long.

The lower portion is a pressure boundary portion. It is welded to a hollow tube that contains a spring and a plunger, and this is sort of like a curtain rod, where you have an expanding curtain rod between a window, and it's fixed -- n this case it's fixed at the bottom of the reactor and the spring plunger here is imposed upon the top guide and is held and holds the top of the [illegible] in place and keeps it from moving around.

This tube is in such a small place that it is

totally confined between the intersection between four fuel bundles. There is no room for this tube to move in any direction from its location when fuel bundles are in place. When the fuel bundles are removed, it would be possible for this tube, if it were loose from the top guide, to move in a horizontal direction away from its proper location.

So the function that this plunger performs is simply to hold the top of the tube in place, and where cracking has been found to occur in this particular assembly is above the lower weld of this tube and below the upper weld at this location.

There have been at least four reported incidents of tubes like this cracking.

[Slide.]

The type of cracking that is seen is shown in this particular figure. In this case it was Hatch Unit 1. Both Hatch units have been found to have dry tubes which have experienced some cracking and these were noticed back in late 1984 during their previous refueling outage.

They looked at the tubes and visually were able to pick out the cracking. In another case, when they found cracking, they went down with a television camera tube and were going to take a closer look at it, and in the process of trying to take a closer look at it, the camera tube bumped into the upper portion of this tube and part of it broke

away. The crack had apparently gone 360 degrees around.

Again, the important point to remember in all four of these incidents is that the tube that broke was not a pressure boundary tube, and in no case was it evident that the cracks extended either into the weld or progressed into the lower portion of the pressure boundary tube that is connected to it.

The rationale for why it only cracks in the upper portion is not real clear to me. The materials are the same, the processing of the material in the upper portion of the tube and the lower portion of the tube is identical.

The only thing that is really different on the lower tube which contains the pressure boundary is that it is a thicker weld tube because it is a pressure boundary, and it's also under compression because the pressure is on the outside of the tube in the reactor, and the inside of the tube is under normal atmospheric pressure.

So, consequently you do have this compressive force which we believe is the reason for the absence of any cracking in the lower portion of the tube. It's the only rational reason why it doesn't occur in that region.

MR. DILLON: Is this again about 10 to the 21st as far as fluence is concerned?

MR. WILLERTZ: The neutron irradiation level in these cases was on the order of 2 x 10 to the 21.

[Slide.]

There were some cases that showed lower fluences that produced cracking, but there is a threshold that is not too much below 2 x 10 to the 21 that they believe is a reasonable threshold for any cracking to occur.

A very brief summary then on some earlier cracking that's been seen in various areas of BWRs has been some very old cracking in fuel cladding which, of course, is not being used any more.

Neutron source holders, which are similar to the dry tubes that are in the reactor, control rod absorber tubes, fuel bundles with cap screws and rivets and control rod followers.

Down in this lower portion we see that the cracking has occurred approximately at 5 x 10 to the 20th. The materials that were involved in these earlier component failures involved 304, 304L, 347, 348, and Incaloy 800. The components were solution-annealed in all cases, and there was no evidence of sensitization.

The threshold fluence from these particular observances was somewhere on the order of 5 x 10 to the 20th, and in most of these cases a high stress was required to produce them.

MR. SHEWMON: There has been some experience in PWRs of cracking of threaded fasteners. From that summary I

conclude that you know of none in BWRs, maybe because of differences in design.

MR. WILLERTZ: There are very few threaded fasteners in the core region of a boiling water reactor. Like you say, the threaded fasteners in a PWR have been shown in some cases to crack. I believe they were X-750s, if I remember, X-750 and a nickel base alloy. And I believe it was attributed to either an improper heat treatment or the high stresses that were present on the screws at the time of failure.

MR. SHEWMON: Well, inherently in a screw you tend to tighten it down for heavy stresses, and there's been one case in Point Beach where they had some failures which the ex-plant manager comes back and says, "You metallurgists said if we put annealed stuff in there, it wouldn't happen." So 10 years later he thinks it did and so, yes, that was the excuse for why it happened before. Whether or not they need another excuse now is a separate question.

Go ahead.

[Slide.]

MR. WILLERTZ: More recent data and experiences in boiling water reactors has shown that as late as 1982, control blade handles have been found to experience some cracking, and I believe there was one earlier this year, I'm not sure where that was, but it was reported as just another incident.

Control blade sheaths in '83, plate type control

blades in '83 also, and now the dry tubes. The IRMs and the SRMs in '84 and '85.

So there seems to be enough reason here to look at this phenomena in more detail, and to take an analytical look at the components that are susceptible to this particular phenomena to see whether or not there is cause for concern.

[Slide.]

Some of the data here that is recently published shows that material that had been irradiated in a reactor, taken out and laboratory tests done on them, shows that with fluences of around 1 x 10 to the 18th, you get ductile failures. If you go to 3 x 10 to the 21, this is only two data points widely scattered, but it's obvious that when you get to 3 x 10 to the 21, that IGSCC type of cracking does occur in an irradiated material that would normally not be considered sensitive.

[Slide.]

Some additional CERT tests which have shown examples of tubes that have been irradiated to different fluences a little closer together. We see again that radiation levels on the order of 5 x 10 to the 20th are regions where you begin to see a small amount of IGSCC type of cracking.

Now I want to point out that while the water temperature here is typical of a BWR, the oxygen level here is not. This is 32 to 36 parts per million of oxygen, and

normally you're talking about 200 to 250 parts per billion of oxygen in a normal operating environment, and very low conductivity.

This is also performed with a CERT test which is a very accelerated test and also a very damaging test to the material, because it is deliberately taken above the yield point which is known to cause IGSCC and it's done that way for economy of time in doing the test. And so these accelerated tests tend to show IGSCC type of cracking a little bit earlier under more advanced conditions than what you would expect if it were in a real reactor environment.

On the other hand, when you talk about a real reactor environment, you can talk about the effect of radiolysis under the influence of a gamma neutron radiation which may offset the accelerating effect of the CERT test outside of the reactor area.

So we have those things to consider when we look at the results of all these tests.

[Slide.]

Another data point to consider is the effect of radiation on the ductility of the material, and one sees in these test results here that the ductility of stainless steel, which is normally 40, 50, 60 percent elongation now, under irradiation conditions of about 2 x 10 to the 21st, shows less than 5 percent elongation.

So there is this very drastic reduction in ductility of these materials which may explain the observations that when some of these tubes are hit with camera and other equipment that they go down to look around in the reactor, that occasionally some of these may break if you are too rough with them.

So those are other considerations that one has to look at when doing in-service inspections.

[Slide.]

As far as the components in the reactor that we're looking at that see fluences on the order of 5 x 10 to the 20th or higher, which is getting into the region of susceptibility to IASCC, we are talking about those few reactors left that have fuel cladding, lifetime fluences of almost 5 x 10 to the 22. Control rods, if they were left in, would also be in that region.

However, they are changed out much more often on a normal basis, and so has less fluence than what's shown here by at least a factor of four, and also because the control rods are not in all the time means that the fluence is even lower than a factor of four from this number.

So we are really talking closer to something in the 10 to the 21st regime for the control rods.

Dry tubes also see the very intense radiation in the core area because they're between the fuel bundles, and if

they were left in the full lifetime for 32 full power years, which is equivalent to 40 years of real operating time, yo would also see four times, almost 5 x 10 to the 22nd neutrons.

The core boundary, which is just the boundary between the last fuel element, you see a little bit less fluence. You're getting away from the core center now. The top guide, which is above the fuel element -- yes, Paul?

MR. SHEWMON: These fluxes are greater than 1 MEV; is that right?

MR. WILLERTZ: These are greater than 1 MEV.

The top guide is also in the region of high flux and high fluence, and that would remain there for the life of the reactor, so we're in a region where at least at the lower portion of the guide, one might expect the material to become susceptible to IASCC.

The shroud is just marginally in the region at which it would become susceptible. However, the fact that the shroud is under compressive stresses, just like the lower portion of the dry tube, means that there is very low, if any, probability at all of the shroud itself becoming embrittled to the point where it would fail of its own accord.

The vessel wall, just as a matter of reference, is well below the threshold for IGSCC or IASCC, and so the stainless steel cladding on the wall would be well out of the range of which we'd expect the embrittlement to become

serious.

These are some of the components we are talking about.

MR. ODETTE: What about components like the grid plate that holds the fuel elements and other internal structures? Are they made of stainless steel, or zirconium?

MR. WILLERTZ: The lower support structure is made out of stainless steel and the assembly that the fuel bundle rests on at the bottom portion of the reactor core is all made out of stainless steel.

MR. ODETTE: So these would experience rather high --

MR. WILLERTZ: However, they're farther away from the actual fuel element, and so do not see the same level of fluence that the top guide sees, for example.

MR. ODETTE: Do you know what those levels are?

MR. WILLERTZ: No, I don't have the numbers here with me. But they are considerably below the shroud. And partly that's because also the water is densest at that point, and so the neutrons become attenuated a lot more down in the lower portion of the core than what they do at the upper portion of the core. And so your fluence level where the water goes to steam at the upper portion of the core would be higher in that region.

[Slide.]

So far we have not talked about any of the programs that are going on because of the relative newness of the phenomena of IASCC. A lot of these programs that are listed here are more or less in the formative stages. Some of them have been completed, some of the initial studies have been completed at GE and KWU programs, but they are ongoing, and do not contain all of the necessary data to tell us all of the boundaries that exist for producing IASCC in the stainless steels that we have been talking about.

I don't believe any of these programs, in fact, have ended yet. The GE programs have been designed to characterize the cracking metallurgically. They have been aimed at looking at the high purity alloys because for a number of years high purity alloys have shown less susceptibility to IGSCC.

Water chemistry is one that's actively going on. There's programs for many of the BWRs in this country that are fitting in hydrogen/water chemistry into their operating procedures in order to mitigate IGSCC in their piping systems.

As a side issue of the water chemistry program, then, GE has also been looking at what does this do to the IASCC phenomena, and they have got some preliminary data at this point that shows that at least in the laboratory type tests that IASCC is in fact mitigated.

However, the tests are done out of reactor, away from the conditions of neutron and gamma radiation, which

would make the materials behave somewhat differently under the flux conditions and the temperature and operating conditions, and so the laboratory results don't really give us a good feeling of how the material is going to behave under hydrogen/water reactor conditions in a boiling water reactor environment itself.

Those tests are planned in the future. Right now GE and KWU are devising programs and instrumentation to fit into the reactor, into dry tubes, as a matter of fact, to look at what happens to the chemical potential of the material in the water reactor environment, and to perform crack growth rate tests in the reactor environment.

Did I mention electrochemical potential? That's one of them.

MR. SHEWMON: You don't have the Japanese on your list. Is that because they aren't heard from yet, or they think for some reason it doesn't happen in Japan?

MR. WILLERTZ: The Japanese have been inquiring about the performance of IASCC and the performance of parts under irradiation conditions. The information I have is very sketchy as to what they're doing about it. I do believe that they're looking at the high purity alloys and maybe replacing their dry tubes with the high purity materials, and it may be the reason that we have not seen any reports of cracking of this phenomena in the Japanese boiling water reactors.

But since they have been inquiring and they have been doing some tests in the universities there -- I don't know the details of the tests that they're doing. I'm not familiar with all of the work, and so I --

MR. SHEWMON: Okay.

MR. WILLERTZ: The information available from KWU and GE and the boiling water reactors owners group programs is easy to come by.

The KWU programs, as I said, were also testing different alloy elements in the material and looking at the various 347, 348, X-750, 304, 316 materials in actual CERT tests done in reactors, in both boiling water and PWR reactors, as I understand it.

And the type of tests that they have performed have been tests where the material was manufactured in a tube form and then another material put inside like a B4C which normally swells under reactor conditions, under irradiation conditions, and so produces in effect an in situ CERT test, and these tests can be performed this way in a reactor to show what the real effects of a total BWR reactor environment is, or PWR.

EPRI programs are numerous in this particular area, and many of them are just getting started. Some of them don't even have numbers yet. They are still in the planning stages. They are looking at lab tests on irradiated specimens, purity of microstructure being a couple of the

parameters, in-plant tests on high purity alloys.

MR. SHEWMON: Why don't you let us read that, and tell me what they think they'll find with purity? Is this back to this -- you had sulfur and silicon or something, phosphorus and silicon you implicated in your earlier statement. Is that what they're trying to get out or --

MR. WILLERTZ: Some of the early results of some of the investigations looked at phosphorus and silicon and show that in the accelerated tests that phosphorus levels on the order of .01 or lower show an immunity to IGSCC.

Also the silicon concentration below about .1 percent shows an immunity to IGSCC.

The effects of radiation on the cracking phenomena has not to now been tested sufficiently, either in the laboratory or in reactor tests to give a total picture of immunity in a real reactor environment.

[Slide.]

These tests and the hydrogen/water chemistry tests in reactor are tests that right now have to be finalized and put into the reactor to define totally whether or not the low silicon and phosphorus materials are in fact going to be immune there.

The tests that are being conducted now are all aimed at finalizing what hydrogen/water chemistry will do for you, as well as the impurity levels of the minor elements in the

various stainless steels will do for you.

All the results to date that I'm aware of have shown good results in what they were set up to show. Irr diated materials in a laboratory under gamma radiation .tions, for example, irradiated materials have been shown to be immune to IGSCC without hydrogen -- or with hydrogen and not to be immune without the hydrogen.

The presence or absence of gamma radiation has shown also some effects where the irradiated materials exposed to a gamma radiation in a normal environment would give you IGSCC, and if the gamma radiation were taken away, they would not show the IGSCC.

So there's a lot of loose data right now that's available in the literature and from these programs which shows that, to me, they are on the right track; that there is very high probability that the IASCC will be mitigated by high purity alloys and/or the addition of hydrogen/water chemistry.

The definitive experiments in reactor have not been performed, and I personally will wait my final judgment as to whether it's going to be totally mitigated until I see all of the data.

[Slide.]

This is my last slide.

MR. SHEWMON: Fine. Go ahead.

MR. WILLERTZ: We have also looked at safety issues

which I haven't discussed right now, but I think that's one of your concerns, and so I should bring it up.

As far as the cracking that's been observed to date in the reactor, it has always been after the fact and has never caused, as far as I know, any operational problems in the reactor under any conditions, has not prevented the reactor from coming down from power or caused any safety concern due to loose parts or otherwise in an operating reactor.

We, too, were very concerned in our study last year when we found out about the phenomena of just what parts would come loose, and what would happen to them if they did come loose in a reactor.

The control blade, for example, has never been shown to produce a loose part other than the bail handle at the top in that one reactor that I cited.

The bail handle, on looking at the size and configuration and looking at the size of the holes and locations of areas where this thing could get caught, we have just not been able to find a location in the reactor that this thing could jam itself or embed itself so as to prevent normal operation of a reactor.

The clearances are so small between plates and the fuel bundles that it is impossible for this piece to actually get down below the control rod itself.

The loss of that particular part was attributed to the fact that they didn't notice that a part was loose until they pulled the fuel bundles out, and when they pulled the fuel bundles out, the part that was sitting on top of the control blade apparently fell down and went through the hole in the bottom support plate of the reactor and presumably is laying on the bottom of the reactor some place.

I don't believe anybody has found that piece yet, but they did have two other pieces, if you noticed on that slide, that were loose and they did retrieve those. So apparently the third one is down there some place, too.

So that's the only time that they had a loose part that they didn't know where it was going or what was happening to it.

The cracks in the control blade sheaths themselves are not a problem. They have not produced loose parts, they haven't produced protuberances from the plate so as to cause malfunction of the control blade.

The control blade again is confined in a very narrow space, and there has just been no reason or rationale developed for how the control blade could be jammed due to something protruding from the control blade itself.

The dry tubes are confined to a very narrow area between the fuel bundles, between four fuel bundles at that intersection, and as such, that upper portion that cracked has

no place to go. If it were to come loose in service while the fuel was in place, it just stays there, there is no loose part, because the spring confines it and the fuel bundles confine it in the other dimension.

So it just simply stays there and when the fuel is unloaded, one might notice the part and then will be able to remove the dry tube and replace it at that time.

The last two parts that I can mention that may be susceptible to IASCC, which is the top guide and the shroud -- I talked about the shroud briefly and the fact that it's under compression and also the fact that it has not been found to be cracked under any instances to date.

The top guide has a higher fluence level and has been expressed to us by the manufacturer that it's in a region that could possibly be susceptible to cracking, and that one should be prudent to look at the guide at refueling opportunities, to make sure that it is not cracking.

They have done an analysis for us and they conclude that under the conditions at which this top guide operates -- it's in a totally confined area -- again it separates all the fuel bundles, and when the fuel bundles are in place, it only prevents lateral motion of fuel bundles. It is totally confined. It is designed with very low stresses.

The confining nature and the low stresses indicate that there is no possibility of this thing coming loose and

completely collapsing, which in fact it would have to do in order to cause any problem whatsoever.

Only the lower portion of the top guide -- it's about 12 or 13 inches high -- the lower portion is in the region where the fluence would get high enough to cause cracking.

The upper portion is in a much lower fluence area, it's 12 or 13 inches away from the other area, and has roughly an order of magnitude less fluence than the lower portion of the guide.

Consequently it's expected to remain ductile throughout its lifetime. So even if the lower portion became brittle due to the radiation and became susceptible to IASCC, it would provide no means for propagating a brittle crack all the way to the structure. It would leave the ductile area and stop.

MR. SHEWMON: Well, thank you very much for the presentation. It's been quite detailed and we appreciate it.

Are there any other qusetions?

MR. WARD: In your first conclusion up there, you mention neutron irradiation only, and you have talked about gamma irradiation contributing to this phenomena. Is the difference in that you've seen gamma contribute to the phenomena only in laboratories and in the --

MR. WILLERTZ: It has been shown in laboratory tests

that the gamma radiation can produce an environment which will cause an otherwise nonsusceptible material to become susceptible. The actual level of fluence has yet to be defined, but the laboratory tests that I'm aware of are still considerably lower fluence levels than -- lower flux levels, excuse me, than what would normally be found in a reactor environment. But they have been shown to have an effect on the corrosion rate of the stainless steel.

Consequently it is believed that in reactor there is going to be a similar effect on the grain boundaries of the material.

MR. WARD: Okay. But the problems you have seen in reactor so far have all been attributed to neutrons; is that it?

MR. WILLERTZ: No, it's really a combination of both neutrons and the gammas.

MR. WARD: Well, I was just curious why in your first conclusion you mentioned only neutron.

MR. WILLERTZ: It was an oversight.

MR. SHEWMON: Dr. Johnston, the original memo in June to you on this talked also about threaded fasteners in PWRs. Is the position on that currently that it really hasn't occurred frequently enough or clearly enough to be a problem, and if it did, the conclusion is it appeared to be the same as for boilers; namely that the parts couldn't get you into

trouble?

MR. JOHNSTON: Yes. I think you're referring to the -- we had problems with the bolting in the shrouds at the Oconee plants. We have had it at the Arkansas Nuclear 1, and I believe we have had it at Rancho Seco, three B&W plants. Those were, I think at least in the first case, A286 bolting rather than X-750.

At that time I don't believe IASCC was considered as one of the contributing factors.

MR. SHEWMON: But it was irradiation --

MR. JOHNSTON: No, it was poor heat treatment was the conclusion of the investigation.

MR. SHEWMON: Were all of these failure you're talking about in the core environment?

MR. JOHNSTON: No, the location of the shroud, if you look at the boiling water reactor, the picture that we have here, it's considerably out of the core, and it's somewhat in the same location in a PWR as well. The shroud is out a fair ways and up above to some -- well, it's almost lateral with the top of the fuel assemblies, but it's out a ways.

I'm not aware that -- I think, but I'm not sure about this, but I think it was -- I think B&W told us that this had been considered and that the fluences were just too low to worry about.

Now I can't quote any numbers, but, you know, as part of the general discussion, I believe that was considered as a possibility, but it was -- basically they showed that the particular batch of bolts and so forth had some problems with heat treatment and they had some -- the fact the way it was broached or something about the way the head was formed on it was poorly done, and the upshot of it seemed to be that it was a fabrication problem that was the cause of it, and they subsequently changed it out with a different material, and we have had no problems since then. And we have had similar effects, however, as I said, in two other reactors.

It is not a safety issue in the sense that there's 120 of them or something that go around, and you're going to lose about 90 of them before you really have lost the ability to support the weight. And if it falls, it doesn't stop the flow, anyway, because there's a rest a couple of inches down below that it would settle down on if it fell. But we're not pleased with it.

MR. SHEWMON: If there are no other questions then, we will adjourn until 1:30.

[Whereupon, at 12:25 p.m., the meeting was recessed, to reconvene at 1:30 p.m., this same day.]

AFTERNOON SESSION

[1:30 p.m.]

MR. SHEWMON: This afternoon, then, we shift over to the changes, or at least the revision of Reg Guide 1.99, and I guess Neil Randall begins.

MR. RANDALL: There are a couple of people here who can help me with this. Warren Hazelton and Roy Woods are here, particularly when I get to the implementation aspects of this Guide. We agreed that I would show the Vu-graphs and they would take all the tough questions.

My purpose here is to get Revision 2 out for public comment. I want to be sure the Chairman hears this, so I will wait a second.

[Laughter.]

[Slide.]

I say my purpose here is to get Revision 2 out for public comment.

MR. SHEWMON: Okay.

MR. RANDALL: Now, I realize this is not quite the formal meeting where you gather in to review Reg Guides as such, that we are invited by you to give a technical presentation; but I hope this constitutes what you need to hear in order to say yes, it's okay to go for public comment.

[Slide.]

Here is an outline of what I plan to talk about.

Under safety significance, I want to discuss how we regulate fracture prevention of the reactor vessel, how the Reg Guide fits in with Appendix G in that regulatory process, and just make sure that everyone understands what we do that requires the Reg Guide in many instances.

Under item II, I will explain what's in the Guide and how that differs from Rev. 1. No. III is the biggy to try and explain where I got the calculative procedures that are in the Guide. No. IV, the value/impact part, I may have to skip over in rather brief detail to make the time schedule, but we will see how that goes.

I must be sure to explain to you the relationship to the PTS Rule since that has become an issue now within the NRC.

And finally, under implementation of the Guide, I have to make sure you understand that CRGR Review is not complete. I had a meeting with them on the 24th of July, and when we got to this Item V, the relationship to the PTS Rule, there was considerable confusion about how the two fit together, and so we are due to go back, I hope, on the 25th of September, but that date is not yet firm.

[Slide.]

To spend a minute on the background and the safety significance, I show you a pressure/temperature limit schematic, and as I think everyone knows, there is a high

pressure, low temperature region which I have labeled hazardous vessel integrity.

There is another variable, of course, and that's the cooldown rate, which causes thermal stress, and so I have shown the boundary of that region to be a series of lines depending on the cooldown rate.

The idea of the pressure/temperature limit, of course, is to keep the operation of the vessel during heatup and cooldown well below and to the right of that hazardous region. The reason for that, of course, is that we know at least a couple of transient types that can occur, the low temperature overpressurization, and the PTS events, the rapid cooldown.

The rational of the P/T limit is simply to give the operator as much warning as possible and as much time as possible to manage the transient before he gets into this hazardous region.

Now, with radiation damage, the hazardous region expands to the right and down, and therefore, we have to move the P/T limit to the right and down as well. Now, there is a limit to that, of course, and that's the saturation curve over here. For PWRs they generally keep the boundary some 35 to 50 degrees Fahrenheit lower than the saturation line.

All of this discussion of safety issues thus far has concerned the PWRs. The boiling water reactors ride up and

down the saturation curve except during hydrogen testing.

If you have looked at tech specs for plants, you know that they all have a P/T limit curve with that characteristic shape, and that comes from the fact that the curve is calculated with a simple fracture mechanics approach, the one that is mandated in Appendix G of the boiler code, Section 3.

[Slide.]

The shape of that P/T curve derives from the shape of this KIR curve, it's called in Section 3. I show you here the same figure in Section 11, which also has a lower bound KIC curve in it.

Now, those values of KIR in Section 3, then, amount to allowable values of fracture toughness that are used in this fracture mechanics calculation. Even though it is for crack arrest, we mandate in our regulation that they use that for crack initiation.

MR. SHEWMON: When we get to the transition from the Charpy values that are taken for surveillance to that KIC, will you call my attention to it and explain how we do it?

MR. RANDALL: Yes. I'm coming to that.

You notice the abscissa here is T-RTNDT. I hope you can read it off of there. T is the metal temperature at the tip of the assumed flaw, and RTNDT is the reference temperature of the material at that point, and that value of

RTNDT is the sum of an initial value measured according to the Code, and a delta RTNDT that is the measure of radiation damage that has to be put in.

[Slide.]

So, where does delta RTNDT come from? Well, in a pressurized water reactor that has been running for four or five years, they will have their first surveillance report, and in it there will be a pair of Charpy curves, one for irradiated, one for un-irradiated, and we require in our Appendix G that they measure the difference between the two at the 30 foot pound level, and that is a delta RTNDT. That is one value.

We now say in Revision 2 that if a plant has two surveillance results of this sort, they may use them to get delta RTNDT in calculating their pressure/temperature limits. We describe in the Guide how that calculation should be done. But it takes six, eight, ten years before plants have that, and some of them will never be able to use it because the material they put in the capsules is not the controlling material in the vessel. So for all those cases, they need Reg Guide 199.

[Slide.]

For those, we have taken those individual measurements that I indicated on that graph and worked up a formula to predict delta RTNDT knowing the chemistry and

fluence.

So the heart of the Guide is this expression. Delta RTNDT is a product of the chemistry factor and the fluence factor. The chemistry factor is a function of copper and nickel. Table 1 of the Guide gives those values for a range of copper and nickel contents for welds, Table 2 for base metal.

The fluence, which is in units of 10 to the 19th, one can calculate it or one can take it off of this figure, which is Figure 1 of the Guide.

[Slide.]

You notice fluence function has a value of 1 at 10 to the 19th, and it has this concave downward shape. The slope at 10 to the 19th is .28, which is quite a change from Rev. 1 where we thought the slope was a constant value of .5. Those values in the table are mean values. We then add a margin after we get done calculating the mean.

[Slide.]

To compare Rev. 2 with Rev. 1, I have made a list here. For Rev. 1 we only had a little bit of surveillance data, so about two-thirds of the data base was test reactors. In the present case, we had enough surveillance data and that is all we used. We did not rely on test reactor data, partly because surveillance data is obviously closer to the right fluence rate for vessels, and partly because we feel the two data bases are now somewhat different, but we didn't have a

good expression to meld it all together with one formula.

Rev. 1, the curves were drawn by drawing upper bound curves above the data. For Rev. 2 we used regression analyses to get mean curves and also measure of scatter so that we could be a little more quantitative about the margin that we added.

I told you the chemistry factor includes copper and nickel, and we have separate ones for welds and base metal. I will go into it at length about how we used NRC and EPRI-sponsored work.

I would like to make the point that I have been talking about the basic formula in the Guide with the Working Group on Flaw Evaluation on Section 11 of the Code since January of 1984, and also with the Working Group of Committee E-10 of the ASTM, and Committee E-10 is going to ballot a revision of ASTM E900 this fall, which includes the calculated procedures out of the Guide as the amendment to that standard.

As soon as the ASTM ballot has been completed and the negatives resolved, the Section 11 people plan to also incorporate the guts of the Guide into a figure in Appendix A of Section 11. So we already have some peer review from the people in the industry.

The Guide does go a little bit beyond just the equation for delta RTNDT. You see, that equation is used to calculate the shift at the inside surface of the vessel, and

then we give another attenuation equation to calculate the delta RTNDT at the tip of the assumed crack.

Finally, the Guide, as I mentioned, tells how to use plant surveillance data when that becomes available.

MR. SHEWMON: Tell me what the bullet "Attenuation Equation" means. Basically you are keeping the procedures they used for all effects?

MR. RANDALL: Rev. 1 did not have an attenuation equation. Although we had procedures we used, they weren't published. I will have a figure that shows how that works when I get a little further on.

MR. SHEWMON: Okay, fine.

[Slide.]

To develop what is in the Guide, I used first the work of George Guthrie from Hanford Engineering and Development Laboratory, and then came upon the work of Professor Odette from the University of California, Santa Barbara, and I found they agreed in a number of respects, so I ended up melding the two together.

They both started with almost the same raw data base as far as Charpy curves are concerned, but this Vu-graph outlines a few differences. Professor Odette had a few more data points, mostly of boiling water reactor data. They did both use just power reactor surveillance data, however. Guthrie used 30 foot pound shift values obtained directly from

the surveillance reports, which means they were hand fitted curves done by the authors of those reports, whereas in Professor Odette's case, he used a data base that EPRI had funded to be assembled by other people where those Charpy data had been fitted with a hyperbolic tangent function, and then the 30 foot pound shift taken from that.

Finally, in Guthrie's case he took advantage of some special fluence calculations that had been made by other HEDL people for some plants, perhaps ten out of the total. Professor Odette used the fluence numbers given in the surveillance report.

[Slide.]

To illustrate how difficult the job is, perhaps excuse some of the empiricism that I have applied to that work, I would like to show you how the data are distributed in a nickel-copper space. George Guthrie prepared this Vu-graph for me. He asked the computer to look through and find how many data points they had in this box represented by half nickel and 15 copper, for example, and it said it had four. He then asked it to normalize those measured values to a single fluence, 10 to the 19th, and average them.

So 153 there is the number of degrees shift at 10 to the 19th fluence for that chemistry.

MR. SHEWMON: Normalize means that given some approximate curve, you interpolate to a common value?

MR. RANDALL: Yes. What he did was take the fluence function that is in the Guide and correct the measured value using that fluence function.

MR. SHEWMON: Fine.

MR. RANDALL: Well, the first thing you notice is that, whereas the Guide means to cover the whole range of both variables here, there isn't any data in certain patches of the field. The data are clumped pretty badly in about two areas.

The other thing you would notice if you tried to draw some countour lines or isoshift lines through areas, that there is a fair amount of scatter.

MR. SHEWMON: In some squares I find that four points are melded into one, and in other places I find that they are kept as separate points.

MR. RANDALL: Well, when there are more than one figure in a square, that means they were slightly different nickel contents. This is .75, .70, .65.

MR. SHEWMON: And what does the 153 mean down there, the .5 --

MR. RANDALL: They actually had a nickel content of .55 plus or minus 2-1/2 points.

MR. SHEWMON: Okay. I'm with you. I am with you.

MR. RANDALL: The same thing for base plates. The range of copper and nickel is somewhat smaller in base metal. The data are clumped as far as nickel content is concerned

because either there is no nickel except just the residual amount, or there is a nominal half percent.

[Slide.]

Guthrie and Odette agreed on four areas that I should mention.

They did agree there should be separate correlations of welds and base metal, that the expression should be the product of a chemistry factor and a fluence factor, that only copper and nickel were significant as far as their regression analysis showed, and finally, this statement about the fluence.

So what I did then was take their chemistry factors and prop them up for quite a few combinations of the variables and look at them.

And here is a comparison for weld metal with .2 percent nickel.

[Slide.]

They agree so well within the data base that there's not much problem. So I could use either one.

[Slide.]

For a higher nickel, again they agreed where there was data, from about here up to there (indicating), but the form of the functions that they chose was quite a bit different, especially Odette's, passing through zero. And because that seemed a little more logical and I didn't like

this upsweep here, I chose to use the lesser Odet*e curve with one exception here at the bottom that I will talk about in a moment.

MR. WARD: Let's see. What's the logic of Guthrie's curve there?

MR. RANDALL: Well, the function he happened to choose, it just did that at very low --

MR. WARD: Oh, he was just fitting points above point 1 or something, I guess; is that it?

MR. RANDALL: That's correct. His data base, you remember it's on that scatter plot I had. His data base didn't really guide him down here.

MR. WARD: Okay. But he didn't really have any hypothesis or something that --

MR. RANDALL: No, there is no -- I should say Guthrie did not really use much mechanistic reasoning in getting at his equations, whereas Prof. Odette did certainly take a step in that direction.

[Slide.]

You will recall that this work was really done about three years ago, the basic development of it, two or three years ago.

MR. WARD: It seems like there's a certain logic in going through it.

MR. RANDALL: Well, here it is for base metal.

Being a good regulator, I took the higher branch in each case and simply smoothed the two together.

[Slide.]

But there was a limit to that because sometimes Guthrie's case -- this is base metal with a higher metal -- was even higher than weld metal, and I don't think that was logical, and you remember that for base metal, the copper content only -- the data base only goes to there. So there I used the higher of the two curves, but I made them both the same where the weld was lower than the base metal. So that gave -- when I got done smoothing -- a set of curves like that for various nickel contents.

[Slide.]

MR. SHEWMON: Now is this base metal or weld metal, or is it the same?

MR. RANDALL: No, this is still -- this is weld metal.

MR. THOMPSON: I was thinking that you'd chosen the ones that went through the origin there, and now it doesn't seem to.

MR. RANDALL: Right. It bothered me to have the predicted shift to zero if the copper went to zero. And so what I am showing you this for mainly is to show that -- to try to get a better feel for it, I did look at test reactor data in the low copper range, and it was all fairly high

fluence data, so that to normalize it to 1 x 10 to the 19th, and those are the points that are plotted on here.

MR. SHEWMON: He could follow Odette part way, but not all the way.

[Laughter.]

MR. RANDALL: Well, a good regulator, you know, I just arbitrarily, based on that data, somewhat arbitrarily chose 20 degrees as a cut-off.

I did a similar thing for the base metal.

[Slide.]

And here is a comparison of the two, which I think is what you're waiting for.

[Slide.]

Now you remember I said when base metal curves tended to go above the weld curve, I made it bend over and join, so that's why the two are the same in this region up here.

Once again I remind you it's a region where we don't have any base metal data, so it's a little bit -- it won't get used very much for base metal.

As you can see, in the range of .1 to .2 copper, there is a fair difference between welds and base metal that shows up in those tables in the guide.

This curve is really a plot of those tables, if you like.

MR. SHEWMON: The nickel is important enough to double the effect?

MR. RANDALL: Yes.

MR. SHEWMON: Of what the conclusion is?

MR. RANDALL: Yes. High nickel welds really have an impact.

MR. SHEWMON: I had heard some things from Europe that they had had experience with even higher nickel alloys and didn't seem to see an effect. What's your position on that?

MR. RANDALL: I'm not sure I'm familiar -- that was probably test reactor data, which I didn't study. There is what we call a Rolls Royce weld that was used over here that was about 1.6 nickel, which had tremendous shifts.

MR. ODETTE: 500 degrees Fahrenheit.

MR. SHEWMON: Do the Europeans use a higher nickel subbase metal?

MR. RANDALL: I guess the sub people do.

MR. SHEWMON: We don't talk about those things. I thought I heard -- okay. It may have been something I heard in a conference and that may have had some military stuff in it. Go ahead.

MR. ODETTE: Well, this Rolls Royce weld that Neil is referring to is part of an experimental program that Rolls Royce is sponsoring to deal with this embrittlement issue that

they face with their submarine steels. But to more directly address your question, there is data that suggests once you get above 1-1/2 to 2 percent nickel, that you don't get any incremental effect of nickel in terms of the embrittlement increase.

But up till that point, the European data is very consistent with this amplification of the embrittlement by nickel. So in a range of up to 1-1/2 to 2 percent.

MR. SHEWMON: So in the range of 2 to 4 percent, if the Europeans would use things like that, you'd expect it to be poor and flat?

MR. ODETTE: The limited data that's available seems to suggest the effect of nickel saturates above 1-1/2 to 2 percent.

MR. RANDALL: The guide now has tables up to 1.2 percent nickel which is the biggest number we have found so far.

[Slide.]

This is a comparison of Guthrie and Odette's work as far as the fluence function was concerned. Guthrie's two curves for welds and base metal are practically on top of each other and they run here in the center, and there, and there again is good agreement over the fluence range where there was ample data. The slope of 10 to the 19th was .28 in both cases, which is very gratifying.

So I simply averaged the coefficients in Guthrie's two equations and used those in the guide. So the fluence function in Figure 1 of the guide is this one. I didn't have to do anything very arbitrary to decide that.

[Slide.]

Well, with all of my smoothing and cut-offs and so on, I have to demonstrate to you that I didn't do such violence to their formulas particularly that I couldn't use the values and sigma obtained from their regression analyses in order to get the margin.

So I took the procedures in the guide and calculated a shift number for each line of data in Guthrie's data base, subtracted that from the measured value and plotted that residual versus in this case fluence for weld metal and base metal, and I think at least graphically it demonstrates that, number one, the points are grouped around the zero line in both cases; and number two, that at 2 sigma upper bound it does indeed bound the data. There are only a couple of points out of 100.

MR. ODETTE: These margins for these 2 sigma bounds are based on, from what I read, Guthrie's estimate? You established these 2 sigma bounds based on Guthrie's estimates of the mean residual in these correlations.

Did you go back and calculate with your blended curve what the mean residual error of your application of your

composite model was?

MR. RANDALL: I used his calculation.

MR. ODETTE: I suggest that -- I haven't done what you've done either, but that might be a useful thing to do, to see if the statistical index is similar for your composite as it is for these two cases you used.

MR. RANDALL: Then I would have a numerical comparison.

A similar plot of residuals vs. copper content to show that we have that function reasonably well in hand.

[Slide.]

And I think it does show that.

Notice this droop right in here. That is a consequence of that 20 degree cut-off that I put in for low copper and of the work that I had to do to make the curves come into that ordinant.

[Slide.]

Finally, the residual vs. nickel. This is a good way to see how the data clumped again, either down here or in this group, how the weld metal does indeed have a few points up around 1 percent or slightly above.

MR. SHEWMON: What's the difference between the top and the bottom? One's weld and one's base metal?

MR. RANDALL: Right. And Guthrie got a significantly different sigma value, so that's illustrated on

here, too.

MR. SHEWMON: Now do you feel that that's because -- well, are there fewer data points, or is it just the welds being welds are less homogenous in many regards?

MR. RANDALL: The welds clearly have more scatter. 1 sigma was 28 degrees for Guthrie's fit for welds vs. 17 degrees for Odette's number.

MR. ODETTE: I don't necessarily agree with that, because we had a much -- the scatter was terrible.

MR. RANDALL: You didn't get much difference, I should mention that.

MR. ODETTE: And if you run that up against Guthrie's correlation, they come out similar, too. So I'm not sure there is any inherently larger scatter in the welds than in the base metal. That's all very sensitive to the data base.

MR. RANDALL: Yes, it is. Okay, that's something to look at, certainly.

[Slide.]

In trying to establish how much margin and how to describe it in the guide, I plotted those same residuals as a function of the calculated values, thinking that it might be that we would get a plot that fanned out so that we could say for margin at 20 percent or at something, at some number.

The answer isn't very conclusive. We did conclude

that we would simply add a flat 2 sigma for both welds and base metal, except for very low values, and there we simply said make sigma 50 percent of the calculated value, which is represented by these 45 degree lines here. When you add 2 sigma, you get that.

That is subject to public comment, and we may actually change it a little bit, but I don't think we can move it very much.

[Slide.]

Moving on, then, to the business of how do you calculate attenuations through the wall, I show you this graph with the function that's in the reg guide is this line, and that function is a simple exponential e to the minus .067x where x is the distance in to the wall and e is from the ID.

Well, that is quite a change from what we have done in the past, which is represented by this bottom line in this figure. This is now the attenuation of delta RT/NDT, and to explain how we got from there to there, if you would look at the top figure, this is the attenuation of fluence, and we began -- begin the story with this curve which was an exponential we obtained by fitting calculated values sent to us by vendors from their transport calculations, showing how the "e" greater than 1 MEV fluence attenuated in a calculated way through the wall.

Then we were told by the physical metallurgist that

--

MR. ODETTE: Excuse me, Neil. Can I ask, was there much variation in this parameter for the various vessel configurations?

MR. RANDALL: Yes, there is. I would best label this as a traditional curve. It's one that we've been using for years. It's not a pure exponential. If you look at the curve, it comes off almost level and then it dips down. And that was an empirical function, the one that was used at that time.

Well, when the physical metallurgist said no, the spectrum changes a good deal as you go through the wall, and you should have a weighing function and account for that, they then recommended that we use DPA, the displacements per atom, for which is a table of cross sections in the ASTM standards as your weighting function.

When we do that, we get the top curvehere, and that was based on work done at HEDL, where they look at, I believe, six cases and found that DPA attenuated less rapidly at eight inches in the wall by a factor of a little over two.

So we then changed the exponent here from minus .33 to minus .24 to reflect that change, and but then to get from attenuation of fluence to the attenuation of delta RD/NDT, you have to take that function to some power, and we now use an exponent of .28 as -- really in the guide we have a more

elaborate express, .28 minus .1 log of X -- log of F. But I didn't want to get too fancy in this sort of semi-empirical thing.

So we simply put an exponent of .28 and that gave us the final result, which appears in the reg guide.

MR. RANDALL: The biggest difference in the attenuation formula comes about from our belief that shift varies as the .28 power fluence instead of the .50 power. That's the difference from this curve to this one. And then there is a somewhat smaller increment when you go to dpa.

MR. SHEWMON: Move your slide up and let me tell you what -- see if I can pass a quiz, will you? The top curve, when you say fluence, is that total neutrons or is that fast neutrons?

MR. RANDALL: This curve?

MR. SHEWMON: Yes.

MR. RANDALL: That's the fast neutrons. Oh, I'm sorry. This is the fast neutrons, this is the dpa.

MR. SHEWMON: So they are saying that as you get deeper, because the spectrum changes, you get more dpa per fast neutron, deeper in? Or why is there physically, why is there a shift there?

MR. RANDALL: There is a shift because if you believe that the damage of the energy level just below 1 is significant, then as you go into the wall, there are more in

that bracket and fewer up above.

MR. SHEWMON: Fine. It doesn't quit at 1 MEV.

MR. RANDALL: Right.

MR. SHEWMON: Now, done below, once you have got your dpa curve versus depth, then you have to talk about the relationship, how much change in your transition temperature you get per displaced atom; is that right?

MR. RANDALL: Correct.

MR. SHEWMON: And you are saying that function has changed.

MR. RANDALL: We went from an exponent of .5 in Revision 1 of the Guide to an exponent of .28 is Revision 2. So that made a second change occur, that's right. From there to there and from there to there (indicating).

MR. SHEWMON: And what is raised to the .28 is the fluence.

MR. RANDALL: Is the fluence.

MR. SHEWMON: In the one case, or the dpa in the other, I guess.

MR. RANDALL: Right. We say dpa equivalent because we don't talk much dpa in dealing with the plants. We just say, well, you need a flatter slope greater than 1 MEV.

MR. ODETTE: I might say, Paul, the justification for that is the spectra that surveillance data is generated in have a dpa equivalent which is very similar. That is, it

doesn't differ very much so it doesn't matter whether you talk about dpa or fluence. It's when you go through this 6 or 8 inches of steel where you have scattered a lot of the neutrons that are above 1 MEV down into the .1 MEV where you need to go to this energy weighted dpa unit.

MR. SHEWMON: Let me bring up another point that you can defer till later if you want to, but when I first came on this committee six or eight years ago, there was talk, I think out of some of the Westinghouse points that were relatively advanced at that point, of the saturation of fluence. Is it clear now what causes that, or more of a regulatory nature, where it shows up?

MR. RANDALL: The people that advanced that theory have backed away a good deal.

MR. SHEWMON: You mean as they got more data points, it didn't seem to saturate?

MR. RANDALL: It didn't continue to saturate. Now, we do -- you will recall our fluence function has this concave downward slope, which is a bow in the direction of saturation, if you like, but we don't say it goes all the way to complete saturation. That is not within the fluence ranges that we have any data for.

MR. SHEWMON: But at that time, some steels were saturating and others weren't. You now feel that that reflects scatter in data rather than differences in the

behavior of different lots of steel?

MR. RANDALL: Yes. At least -- well, at one time I had a theory that nickel content affected the exponent, if you like, affected that saturation tendency; but when Guthrie tried that in his regression analyses, he didn't get any better fit. That is, when he tried a nickel term in the exponent. So I gave that up.

MR. ODETTE: Neil, I might say here again that had you waited a little longer, your ideas might have been rediscovered because I think there is now, at least, if one looks at both 'est reactor data and surveillance data and not try to use the test reactor data quantitatively but qualitatively, I think the saturation effect is less apparent, or let's say there isn't a saturation effect, particularly in high nickel steels, but there is more of a saturation effect in the low nickel steel.

So I think you were right a long time ago. The problem with these interpretations of the surveillance data base is that some of these effects can't be discriminated because of the data scatter and the character of the data.

[Slide.]

There is one good piece of experimental evidence that you would not use as a basis for doing the work but I show it to you as a basis for justifying the end result, and this was in the dosimetry improvement program, which performed

a large experiment in, they called it, the Pool Site Facility, the PSF Experiment at Oak Ridge, where they irradiated an 8-inch thick block of steel which was made up for the first five inches of Charpy bars and other test specimens.

When they tested those bars, they got the data represented by the curves here with the dots on them. There are six steels represented, and this, of course, has been normalized to the same shift at the inside surface so they could be compared. So that is the family of experimental attenuation data, if you like, and the arrow points to the Reg Guide curve. On it there is a lot of scatter, but it is a fair fit for four out of six.

That is the only experimental data that I know of at this time. There will be some more in Chuck Serpan's work, at least that one more.

MR. SHEWMON: Is the scatter around two to three inches due to Lord knows what and he hasn't revealed it to us yet, or do we know?

MR. RANDALL: I do not know.

MR. ODETTE: I might also say about this that if you plotted these on an absolute scale, not on a relative scale, against the relative predictions, and in particular if you did it for the in-wall positions and didn't include the two high flux accelerated capsule results, which are like MTR or test reactor results, and in fact you get what looks like much

better agreement.

The significant points are the last three, and when you put those on an absolute scale rather than the scale normalized to the accelerated test reactor capsules, actually these predictions do better than is apparent on these curves.

[Slide.]

MR. RANDALL: Let me go back, then, to the impact of introducing Revision 2 in place of Revision 1 in the calculation of pressure-temperature limits. Here are four trend curves for four different compositions. The top one is typical of a Babcock & Wilcox weld made in the 1960s, and for fluences that they would encounter after a few years, you remember that PWRs accumulate about 10 x 1 to the 19th for effective full power year, boilers maybe a tenth of that. So there isn't very much impact to the plants that have those vessels.

To the plants that have low nickel, like San Onofre, and high fluence, there is a significant benefit because of this nickel factor that we have in there.

The plants that are impacted the worst are those with about two-tenths medium copper and about one nickel, and that is typical of about a half-dozen Combustion Engineering vessels. And there for a fair portion of their life the P/T limits will be boosted upscale, ratcheted, if you like, a significant amount.

The bottom comparison here is for the modern material where they have control of the copper, and the ratchet looks bad but I point out that 200 degrees shift is this line, so while the ratchet is big, the effect on the PWR is minor because --

MR. SHEWMON: Is the Y axis on all of those the same?

MR. RANDALL: Yes. But wait. I have trimmed off the log scale differently on the top ones.

MR. SHEWMON: But the number you have written in is always 100. It looks like it could be 200 in the middle there.

MR. RANDALL: No, that's 100.

MR. SHEWMON: Okay.

[Slide.]

To get the impact of this, mainly for the CRGR Committee, I looked at the chemistry of the critical material for each vessel that we have in operation or that we expect to license, and I said let's assume that we order them next year to change their P-T limits to base them on Rev. 2, whereas they are now basing them on Rev. 1. If they did that, they would use a fluence value for another five years in the case of PWRs, and 10 or 15 years in the case of BWRs.

Doing that for each plant, I then went through and summed up how many plants are ratcheted how much, and this

table is that summary. It turns out that about half of the plants are ratcheted significantly, that is, more than 20 degrees. A few, eight, I found to be ratcheted 50 to 100 degrees. This is using the chemistry values and the fluence values that I had access to. So there is a little slop in these numbers, but the general trend is certainly clear there.

[Slide.]

I then engaged the help of people at Pacific Northwest Laboratories to try to get at an impact in terms of hazard to the public. I show again a temperature-pressure curve here where we have had a fair amount of radiation and the hazardous region is now expanded so that it has approached the Rev. 1 curve.

So the safety impact is expressed as what is the additional risk of vessel failure if we allow the plant to continue with curves based on Rev. 1 instead of requiring them to go to Rev. 2.

Now, there are two ways to look at that safety impact. One has to do with transients. As I indicated in the first Vu-graph, the LTOPs and the PTS events, where if the operator is using Rev. 1, he has less warning and less time to get a transient back under control before he approaches this hazardous region. He has more time if he uses the Rev. 2 curve, but we felt we didn't know how to quantify that in terms of increased vessel tolerability. So we didn't try.

Instead, we took another route. Namely, we said to PNL, assume that a transient will be defined as a heatup and cooldown, number one following the Rev. 1 curve, and number two, following the Rev. 2 curve, and tell us what the difference in probability of vessel failure is for those two excursions.

They had a VESA code that they were working on, improving it for us, which they used for this. When they used the Rev. 2 curve, they could not get any failures in a million iterations. When they used the Rev. 1 curve, they found a probability of failure of between 2 and 3 x 10 to the minus 7 per event, per excursion.

So they then went ahead and parlayed that number on the vessel failure probability into core melt probabilities and finally a calculation of man rem exposure to the public, and that gave them the numerator of the bottom line, which is this dollars per man rem, or I should say the denominator of that number.

To get the costs, they talked to some operator people in their own shop and to some utilities and came up with a number that if the two curves were 100 degrees apart as indicated here, the increased time to get heated up was about two hours. This is for PWRs. They converted that to dollars on the basis of the replacement power cost for $2 worth of power. That is how they got this dollar figure, dollars per

person rem avoided.

We have refined those numbers through the services of something called the Cost Accounting Group in the NRC and got that number down to about $1000 per man rem.

I didn't give you any handouts prior to the meeting involving the PNL work, and I could certainly do that if that is a significant item.

MR. SHEWMON: Let me come back to the slide before that, or the Vu-Graph, and have you tell me what you told me again.

MR. RANDALL: All right.

MR. SHEWMON: Apparently where people are going -- that is, the more recently acquired pressure vessels all seem to fall in this region which is ratcheted 20 to 50 degrees, and that is because the newer ones are low copper and high nickel? If I go back to the slide before that and look at, say, .1 copper, .6 nickel, is that where they have been aiming at for the last 15 years when I assume most of those vessels were made?

MR. RANDALL: Yes, they are in the low copper. The nickel contents are generally about .6, with few exceptions.

MR. SHEWMON: So those things are probably relatively low to begin with, but they are impacted more.

MR. RANDALL: Yes, that's correct.

MR. SHEWMON: Now, if we go to the group of four,

the top line, ratcheted 50 to 100 degrees, where are they? Can you make any similar statements?

MR. RANDALL: Those are the medium copper, high nickel weld vessels.

MR. SHEWMON: Now, I would presume that anything made 15 years ago, that is, the newer plants or newest plants in there, had a relatively low transition temperature to begin with, through heat treatment or something. What about the -- or high toughness, certainly. What about these ones in the ratcheted 50 degrees, the high nickel, medium copper weld that you spoke of?

MR. RANDALL: They do have low initial RTNDTs, that is right, brought on by the nickel content. The high nickel values came about when Combustion Engineering added nickel in the welding process as a pure wire trailing in the weld puddle behind the alloy wire, and for a time, the target was one percent nickel.

MR. SHEWMON: And that was to reduce the ductility temperature?

MR. RANDALL: That was the main purpose, to improve the heat treat response. It came about at the same time people went from A302-B to A533 plate.

MR. SHEWMON: The group down here that is benefitted 25 degrees to 50 degrees?

MR. RANDALL: Those are the low nickel, the .2

nickel.

MR. SHEWMON: And when was that?

MR. RANDALL: Those were very early.

MR. SHEWMON: And were those also relatively high transition temperatures?

MR. RANDALL: Yes. I don't remember specifics. I assume they were, yes.

MR. SHEWMON: If we go at it a different way and I ask you about the half a dozen plants that were part of the PTS study, where they would have relatively high predicted transitions, are most of those benefitted or can't you say one way or the -- can't you generalize that?

MR. RANDALL: With regard to PTS, I have a story on that.

MR. SHEWMON. Yes, but tell me part of it here. Can you or do they all fall into one ratchet category, ratchet or benefit category?

MR. RANDALL: Again, those with high nickel welds are ratcheted the worst in the PTS study as well as in the pressure temperature --

MR. SHEWMON: That doesn't answer my question, but let's go on and I'll wait and ask it again.

MR. RANDALL: All right. Let me show you what I have and see if that is an answer.

[Slide.]

on the PWRs.

[Slide.]

In the BWRs, the systems impact comes in hydro test. In normal heat-up and cooldown when they have a vapor space, they are way over here (indicating), but in hydrotest, where they are water-solid, these are the limits. And again I have shown Rev. 2 with a large ratchet over Rev. 1 just for illustrative purposes.

It turns out that for boiling water reactors, when they get to about 200 degrees F, the amount of pump heat is barely enough to keep raising the temperature. So they begin to raise objections to us any time we approach that 200 degree limit.

To get around that problem, the systems fix of course is to have an auxiliary source of heat for hydrotest. When I say hydrotest, I really refer mainly to the leak tests that are run at the end of each refueling when they put the head back on the vessel.

Well, to fix it by paperwork would require changing the margins that are in the ASME Boiler Code which we have invoked in our Appendix G, and the BWR Owners Group has sent in a letter saying, hey, this may impact us, and we are hoping for more correspondence where they will have some facts and maybe offer some suggestions. So I simply point out that there is some systems impact of that sort.

I want to mention a couple of systems problems that are aggravated by the reg guide. I really can't fix it by changing the reg guide within the constraints that I already have, but they should be aware of a couple of things.

If you remember, in the late 1970s, we had a series of some 30-plus low temperature, overpressurizations as a result of which the NRC mandated that they protect the PT limit curve down in this region with a low setpoint release valve, and generally that pressure was on the order of 450 to 550 pounds. And that valve setting was kept at -- the line to that valve was kept open until they got to about 350 degrees Fahrenheit, and then that was cut off, so that they could go on up in pressure and go to power.

Because Rev. 2 ratchets that curve, it may in fact, as I have illustrated here, impinge on that LTOP setting and require that it be lowered at least for the very lowest start-up temperatures. But there is a constraint on that particularly for Westinghouse-built plants where in order to run the big pumps, they have to have a pressure of around 300 pounds in order for the seals to work properly.

So I expect to get some public comment saying that anything you do like introduce Rev. 2 will squeeze it in this region, and I don't have -- that's a systems question that I can't answer, particularly in the guide, but I want you to know that there will be a specific systems impact of that sort

[Slide.]

Now to get to the PTS story, first off, as I guess everyone knows, in the PTS rule there is a formula for calculating what we call RT PTS --

MR. SHEWMON: Before you leave that, let's stop with the others. I'm a little bit out of my depth with regard to the operating parts here. You have chosen to make Rev. 2 significantly above Rev. 1. My impression was that that curve, at least on the PWRs, set limits on operating, not just heat-up and cooldown, and that that was a certain subset of plants which had had a relatively high reference NDT. And if you talked about that sort of thing in here, it's only been with respect to accident situations, not operating flexibility.

Let me ask one specific question, instead of rambling.

Is there any part of the operating, normal operating limits that are impacted by this?

MR. RANDALL: Not once they get to power, no. It only impacts the process of heat-up and cooldown.

MR. SHEWMON: I guess I'm okay then.

MR. RANDALL: And for boiling water reactors, as I say, they are over on the saturation curve, so there they are only affected --

MR. SHEWMON: Okay, go ahead.

MR. RANDALL: Okay. We get to the relationship of the guide to the PTS rule. I remind you that in the PTS rule that there is a formula for calculating -- we call it RT PTS for comparison with the criteria of 270 degrees for longitudinal welds and 300 degrees for circumferential welds. And that rule came out on the 23rd of July in final form in the Federal Register.

The formula in there for this particular composition is represented by this curve labeled PTS rule, and that really was developed in 1982, and frozen in that form for a couple of reasons:

One, to have changed it at any time would have delayed getting the PTS rule out; but worse than that, many utilities undertook to do some flux reduction programs on the basis of what we were telling them, on the basis of that early curve, and if we had waffled around on what the formula would be, we would have upset all of that flux reduction effort.

So the PTS rule formula really represents an early version of what's now in the reg guide, which when we split off the weld and base metal in separate data bases gave this result. The weld curve popped up, the base metal curve went down.

MR. SHEWMON: Tell me again what the PTS rule says. It gives an equation which is the same for base metal and weld metal, but is a function of composition?

MR. RANDALL: It is a function of copper and nickel and fluence, yes. It was an early version of the reg guide, you might say. The chemistry factor was a little different. The fluence factor was a simple exponent of .27. It's not quite linear on this graph because this is mean plus margin.

MR. WARD: Has the data base changed, Neil, or is it just a more refined interpretation of it?

MR. RANDALL: The PTS rule was based solely on the Guthrie data base. At that time it had about 10 less weld points and 25 less base metal points than it does now.

MR. ODETTE: Do these weld and base metal points have the margin added?

MR. RANDALL: Yes. Yes. I did that so I could also show the Reg Guide 1.99, Rev. 1 curve on here, which is the bound curve.

Well, as you can see, if we change the PTS rule and put in the formula that's now being proposed in the guide, we would indeed ratchet people that had these 1 percent nickel welds.

[Slide.]

Let me show you another one similar kind of figure for .3 copper and .6 nickel. This is typical of high copper B&W weld metal.

There the comparison is much closer, and after they get out a few years in life, and that's really where they are

as far as PTS calculations are concerned, the differences are much smaller.

But if we did introduce a change in the PTS rule and if that looked exactly like the guide looks now, there would be a handful of plants who would be calculated to hit the screening criterion based on my present chemistry numbers about 1995.

So we have to address that issue and the CRGR committee wrestled with it at their first meeting and sent us away to prepare a better answer.

[Slide.]

And the answer that we prepared was this:

Number one, we will -- I guess it's obvious that we will continue to apply the PTS rule as it stands, which has considerable practical effect. Because the first requirement of the rule is that each utility for each plant submit updated values for their copper and nickel and fluence, and we think we would know by now, but it's amazing how much extra work has been done to try to get better numbers in some cases. And also these flux reduction programs are a little hard to keep track of. So we need them and we are ordering them to tell us within six months what the fluence is as of this date, and what the fluence rate is, in the foreseeable future.

So that will go on. In the meantime, we need to get comments on the guide, both for non-PTS purposes and for -- as

described here, to make the supposition that we might some day revise the rule and if we did, how would that affect them, and they are invited to comment on that aspect of it as well.

When we finally get a formula in the reg guide that has cleared all these public comments, and we will have these new chemistries and fluences for all the plants, we will reevaluate where they all stand and we may indeed decide to amend the PTS rule.

MR. SHEWMON: Now the PTS rule will -- can you have contradictory reg guides out there in the field at the same time for calculating it? You do already, I guess.

MR. RANDALL: We do now.

MR. SHEWMON: Because you've got Rev. 1 and the PTS rule.

MR. RANDALL: We are living now with Rev. 1, which is a very nervous thing for Warren Hazelton and me, because in many cases we wish we could go to Rev. 2 and we can't yet. But Rev. 1 is in place. The issuance of the rule doesn't even mention conflict.

MR. SHEWMON: And two years ago when somebody came over here and asked you for what to put in the PTS rule, it wasn't clear to you that base metal and weld metal should be different or --

MR. RANDALL: That's correct.

MR. SHEWMON: But now it seems pretty clear.

MR. RANDALL: Yes.

I think Roy Woods is making a move toward the microphone.

MR. WOODS: Only if asked.

MR. RANDALL: Okay.

MR. SHEWMON: I almost said when Roy Woods came over and asked you, but I wasn't sure he was doing that part that day.

MR. WOODS: I asked.

MR. RANDALL: Yes, indeed.

Well, numbers two and three there may be reversed in time sequence. I'm not promising to do one before the other, and it would be hard to bring them forward in exact parallel, but as I have said, we do need the guide for pressure temperature limit purposes.

MR. SHEWMON: Can somebody at this point tell me which of the -- well, you said half a dozen plants would hit the line in 1995. Are any of those in that set which were looked at seriously by the PTS study group?

MR. WOODS: Okay, let me try that. This is Roy Woods, PTS test manager, NRC.

We looked at Oconee, which was very close to the 270. Neil, I guess that's one that didn't change, since that's not on the list.

MR. RANDALL: Right.

MR. WOODS: Okay. And I can address the other two. Actually the answer is I don't know, but it doesn't matter, because the RT PTS for Calvert Cliffs and for H. P. Robinson are so low now that we have accepted new chemistry values that it doesn't matter. They are not approaching the RT PTS limit even if you were to change to this new correlation. They don't now and they wouldn't on the basis of the new correlation.

The plants that are most badly affected, we believe -- and I would like to emphasize that Neil is correct, we'd like to get the new information from the January submittals coming in for the PTS rule, but based on present information it looks like Ft. Calhoun and Indian Point, I guess Unit 2, would be the most affected. Even though they're affected based on their preliminary information on flux reduction and chemistry and new values and so forth, no plant would exceed the limit before the mid-1990s, even if you were to today replace what's in the PTS rule with this Rev. 2 formulation.

MR. SHEWMON: Now, those would be welds that would be affected, is that a pretty good guess?

MR. RANDALL: Yes, I think all of them will be those where the weld is controlling material.

MR. SHEWMON: And these welds will be axial, not circumferential in a pressure vessel in the core region?

MR. RANDALL: Not necessarily. I don't remember in

every case. I think the answer is yes, but I would have to go back and look.

MR. SHEWMON: I'm only talking about two cases, I guess, which is Indian Point 2, was it, and Ft. Calhoun?

MR. RANDALL: Well, as I said -- I said a handful. I think there might be four or five plants, based on my present --

MR. SHEWMON: I guess my point was if they are vertical welds, they can rearrange fuel some, but then -- go ahead.

MR. WOODS: Well, in fact, a very preliminary version off the top of the head was incorrect, because we didn't take into account the very latest fuel changes at Ft. Calhoun. When you do that, it moves from 1987 to about 1995. So I believe that is an axial weld.

But we really do need this latest information that we will get in January to answer these questions. We are not totally up to speed on exactly what's there chemistrywise and flux-reductionwise at each and every plant.

MR. SHEWMON: Okay.

MR. ODETTE: Beyond the screening criteria, do you anticipate that this much lower attenuation of radiation damage to the vessel will have an effect on the PTS analysis?

MR. WOODS: It won't affect the screening limit at all, because we are talking about inner surface RT PTS values.

MR. SHEWMON: Let me come back. The first one may not affect too many plants, though there are a dozen or two out there that don't have OLs yet. Is that --

MR. RANDALL: Well, we review P-T limits for plants perhaps one every two weeks or so currently, so all of those, once the Reg Guide is in final form, will get reviewed to Rev. 2.

MR. SHEWMON: Then I guess the controlling parameter in that item 1, then, is all operating reactors have to come back in periodically after they bring out a new capsule? Or what is the trip point for bringing them in again?

MR. RANDALL: Their tech specs all have an end date for the currency of their current P-T limits. They generally calculate them for anywhere from two to seven or eight years in advance in the case of PWRs. BWRs go longer than that. So they come in at the end of that period. Very often it is also a time when they have a capsule or two capsules.

Well, whether or not three years will be satisfactory to CRGR, I can't predict, but there has to be some kind of a cut-off.

That's it.

MR. SHEWMON: Now, you never did call my attention to how you got from CV to case of 1. The Code is written in terms of stress intensities, and you regulate shifts on the basis of Charpy numbers; is that right?

MR. ODETTE: But --

MR. RANDALL: May I answer, Roy?

The answer is no, because in thermal shock, the critical crack test is only an inch at most. And so you're not far enough into the wall for the attenuation to have much impact.

Am I being clear?

MR. ODETTE: I guess that's right for initiation. What about arrest?

MR. RANDALL: For arrest, there could be some in the probabilistic analyses. There aren't many cases where arrest and reinitiation turn out to be a governing factor.

MR. ODETTE: Okay. Thank you.

[Slide]

MR. RANDALL: Finally, I have here a Vu-graph on implementation. The first statement there is the one that is now in the Guide for Rev. 1. It simply says when the Guide goes out in final form, we will then review everything that comes in by that.

Item 2 is a tentative thing, very much subject to what the CRGR Committee directs us to do. We are going to propose to them that everyone must come in with some kind of a recalculation of their P-T limits within three years. This keeps people whose P-T limits are now good for 15 years from continuing on and on.

MR. RANDALL: Yes, that's right.

MR. SHEWMON: And where do the twain meet, and how?

MR. RANDALL: It is a basic assumption here that the shift in the KIR curve is truly the same as the shift in Charpy 30 foot pound. There is some experimental data that says it's not bad. There is not a lot of experimental data.

MR. ODETTE: Let me voice a little bit of a concern here, Neil, and maybe my concern can be alleviated. The little bit of data that does compare fairly directly toughness shifts with Charpy shifts is for relatively low sensitivity steels: that is, without big shifts. Do you know if there is information available or becoming available on high copper, high nickel steels?

MR. RANDALL: There is some data becoming available. Would you want to hear Milt Vagins on that issue?

MR. SHEWMON: Yes.

MR. VAGINS: Milt Vagins, Materials Engineering Branch, Office of Research.

Actually, you are going to hear all about it tomorrow. This is our fifth irradiation series, HSST fifth radiation series, which we call KIC verfication curve study, where we are taking two high coppers, .25 and .35, all high upper shelf, unfortunately, irradiating it and taking un-irradiated 8Ts and irradiated 4Ts, drop weight Charpy's,

the whole ball of wax, trying to get a correlation. Charpy, KIC, drop weights, et cetera.

We have data now, a considerable amount of data, by the way, at 41 jewels only. At the 41 jewel shift point, both KIC and the CV shift is the same. 41 jewels, that's 30 foot pounds, and 100 megapascal square meters KIC. They are very, very close and they are consistently close, wherever we can get valid data.

I was trying to fumble around for a slide I'm going to show tomorrow, but basically, every time we do this we get the same data. We did it on the PSF in the dosimetry program. We studied through wall. Those are low sensitivity. We did it it on the fourth irradiation, HSST fourth irradiation, which was a large statistical data base. Unfortunately, present practice, low sensitivity, low copper.

But the early work at NRL done by Frank Loss, Hawthorne, et al, the KJ correlation, was for fairly high copper stuff.

MR. ODETTE: That was upper shelf.

MR. VAGIN: No, this was the early work. I'm talking about late sixties and early seventies. The upper shelf is another -- that's the second and third irradiation.

Tomorrow we will discuss this in more detail, but there is a question of whether it does indeed correlate all the way up and down the KIC, whether KIC shifts the way the

Code says it shifts, and that is an extremely important point.

MR. ODETTE: One of the results of the PSF experiment was that you got good agreement if you made these beta corrections, too, but if you didn't -- and I think there is some question about how that is done.

MR. VAGIN: Well, that is true.

MR. ODETTE: The toughness shifts were appreciably greater than the Charpy shifts if you didn't make that --

MR. VAGIN: I wouldn't use the words appreciably greater.

MR. ODETTE: Thirty percent.

MR. VAGIN: Some were greater, some were less. But they are all invalid.

We will discuss more of this tomorrow, if you want. I think today you may want to continue with this. I have two hours tomorrow.

MR. ODETTE: I won't be here.

MR. VAGIN: Oh, I'm sorry. Okay.

MR. ODETTE: But that's okay. I can talk to you --

MR. VAGIN: All right. Anytime you want. But I have a hundred and some odd slides to go through, anytime you are ready.

[Laughter.]

But you are right. There is doubt. That's why we are spending -- the fifth irradiation is the largest, most

expensive single program the HSST has done. We are irradiating 16 4Ts, hundreds of 2T, et cetera.

MR. SHEWMON: Any other questions on the presentation? Bob?

MR. ODETTE: Maybe I can offer a few comments. Neil did a very effective job with the presentation. I think he did a commendable job in general in formulating this Reg Guide.

It is three years, essentially, since the information that was developed, the data base and the analysis that was done to support this, was put into place, and there has been additional insight and there has been additional data become available.

I have looked closely in a continuing way at the weld situation. I haven't looked so closely at the base. And I have looked most closely at the correlation that we recommended, I guess, in 1982. The situation has been very comforting. I think we found something on the order of 54 additional weld points that have become available.

There are two of them that fell above our recommended upper bound, both for the Jose Cabrera plant, and these are the only two that we found. There is some additional scatter that we observe at low fluences, below 2 or 3 x 10 to the 18th, and I think that may be an inherent characteristic of these materials, that there is more material

variability manifested at low fluences than at higher fluences where some of the variability may wash out.

But if one takes the 47 or 48 points that remain after that, we come up with a mean residual error which is only 11 degrees Centigrade, which is very good. So I think these basic correlations have remained generally robust, and I think that the Reg Guide will reflect that.

I believe the Reg Guide is slightly conservative in some areas, both perhaps in the margin and in the lowest copper levels. That I don't think has a big impact on any of the U.S. programs but it gives some of our European friends indigestion because they do look to the United States for the regulatory rules, and in their new clean steels I think they feel -- and with some justification -- that the reg guide predictions will be conservative. But I don't think that is a major concern of ours.

MR. SHEWMON: When you say conservative, the intimidation is that it is unduly conservative. Is this by 10 degrees or 100 degrees or 50?

MR. ODETTE: Yes, it's a small level of conservatism in the sense that it may be only 20 or 30 degrees, say 10 to 30 degrees Centigrade, but they have imposed upon themselves very stringent margins, like shifts no more than 30 or 40 degrees, and I think there is some justification to the fact that there has been very little surveillance data,

particularly for welds at copper levels below about a tenth of a percent.

Over the last couple of years we have done an analysis which has enabled us to at least qualitatively rationalize a very much larger body of materials test reactor data with the surveillance data, which suggests a different form in the slow copper region. This form is as statistically consistent with the surveillance data base as the other form, and we couldn't choose between then based solely on the surveillance data.

So using the MTR data as an extra weight on that, I think there is perhaps some slight and -- depending on what what imposes on oneself -- significant conservatism at the low copper level. But things have remained very consistent at higher copper levels.

I might add that Dr. Ward's point about the additional availability of data is very much the point with this high nickel experience. That data is just now becoming available. In fact, the first bit of it became available when these correlations were being finalized, which wasn't very long after the PTS Rule came out, and they are, in fact, the things that pushed this high nickel weld sensitivity up. But the data is there to support that change, unfortunately.

MR. SHEWMON: Now, you were talking about the scatter being less with your way of treating the data on the

welds than the other. Does your analysis agree that there is this 50 to 100 degrees F. difference in the reference NDT temperature for welds and base metals? Will your treatment of it also agree with that part?

MR. ODETTE: When we did the analysis, the separation of base and weld gave a slight improvement in the statistical indications of goodness of fit. More significantly, it gave us a slightly more conservative prediction for the welds. That is, if you took the combined data bases, you got a correlation that was sort of a weighted average of the two responses, weighted to the data that had most preponderance in a certain region; and in fact, the copper levels don't overlap all that much between base and welds, so in a low copper region, the base is dominant, and in the higher copper region, the weld is dominant.

I felt at the time it was better to make the separation and to have it hit a slightly more conservative correlation for welds. I'm not sure that that can be absolutely statistically justified, but nevertheless, it does provide a slight additional degree of conservatism.

But I think you misunderstood the point. I was saying that we got roughly comparable residual errors or standard deviations, as Guthrie did for the weld. What we found was our plate errors were larger than his. The reason they were larger than his was the extra data that we had in

our data base, not the differences in our correlations.

MR. SHEWMON: It seems to me most of the impact here is -- well, I'm trying to grasp where most of the impact would be. I am looking, for example, at that first of the correlations that Neil showed that plotted on the PTS Rule, and there is, by the time you get up into the significant range, over 100 degrees Fahrenheit shift or difference in the predicted shift between weld and base metal. What I am trying to get at is how well that is supported.

MR. ODETTE: I don't quite understand that, I guess.

MR. RANDALL: Part of that difference is the fact that that is mean plus margin you are looking at, and the margain we added was 34 degrees for base metal and 56 for welds.

MR. SHEWMON: That margin is independent of fluence, or not?

MR. RANDALL: Yes, independent of fluence.

MR. SHEWMON: Well, the difference between those two curves varies a lot with fluence because it is constant on a log plot here. So it ranges from 50 degrees at 10 to the minus 18 to 125 or 130 degrees at 10 to the 20.

MR. ODETTE: Paul, in that particular one I don't think there is any base data with nickel greater than about .75. There is no data up there that I know about, and I suspect Guthrie doesn't have any either.

MR. SHEWMON: And you are saying there is weld material up there.

MR. ODETTE: There is weld material up there. There is not base material up there. I would be personally surprised if there was that kind of difference. I think if we had base data up there it might be a little higher.

MR. RANDALL: That's a good point. The highest nickel in base metal is about .8.

MR. ODETTE: And there is really not an awful lot of that, is there.

MR. SHEWMON: How many years were we taking one percent nickel as a wonderful kind of weld?

MR. RANDALL: I'm really guessing now. Three or four.

MR. SHEWMON: At a time when people were building a lot of vessels.

MR. RANDALL: Yes.

MR. ODETTE: Actually, one of the problems is that when they go and measure it, sometimes the nickel comes out considerably higher than one percent, as I understand. I mean they get values up to 1.2 or 1.3 percent sometimes.

MR. RANDALL: Yes, there are a few up that high.

MR. SHEWMON: I'm a little confused as to where we are in the program. Do we have you next on the program here?

MR. SERPAN: Yes.

MR. SHEWMON: My program says Wood Hazelton, Implementation.

VOICE: We already covered that.

MR. SHEWMON: Fine. You just put us back on schedule then, Roy.

So we are up to Serpan on introduction and foreign interactions. Is this primarily with regard to the NDT? So we are shifting now. Then let me stop before Neil sits down.

The question is whether or not the Subcommittee would have any concern about recommending that this go out for public comment, and I guess from the questioning, I suspect none.

Is that a fair reflection of the position? I see no nodding heads the wrong way. Okay. Then if you feel like a letter would be desirable the next time we get together, which is probably next week, then it could be done, I think.

MR. RANDALL: We appreciate that.

MR. IGNE: As far as the full ACRS is concerned, Paul, we were planning the Staff for one hour on GDC-4, and we were planning to have a Subcommittee report to the full committee, but if you would like to write a letter, you could do that.

MR. WARD: I think that could be the basis. That could be the basis for the letter, that simply says it is okay to go out for public comment.

MR. SHEWMON: Okay. Thank you.

MR. SERPAN: Charles Serpan, Materials Engineering Branch, Office of Research.

I am leading off a series of talks now from the Materials Engineering Branch on a series of subjects that have been selected by the subcommittee chairman. It will amount to a review of most everything within the Materials Engineering Branch, but not quite everything.

I will also show -- inasmuch as we're going to have a great deal of presentation from other members of the branch, I am not going to go into great detail, but I am going to try to give it a summary overview.

At the end I will also give a summary of the foreign work that we have underway.

[Slide.]

In the Materials Engineering Branch, we believe the key safety issues are these:

An understanding of the properties and behavior of reactor materials degraded by aging.

Limits of acceptability have to be established for operating reactor service of cracked and degraded piping.

We have indicated some dates here when we think we are going to have the major part of that safety issue resolved.

The limits of acceptability have to be established

for operating reactor service of cracked and degraded steam generator tubing and methods for detection and size of flaws in vessels, piping and steam generator tubing have to be validated and, in some cases, improved.

[Slide.]

Each one of those you may recognize works directly into the four areas that we have in the branch, which are reactor vessels, piping, steam generators, nondestructive examination. This is a very -- this is an overview of what we expect to have for the budget for the next coming years, and I have grouped it in the general areas here, rather than by specific contracts.

So in reactor vessels you see we've got the bulk of the resources in the HSST program, but we have irradiation work, surveillance dosimetry, tailing off. Now we've got something called plant life extension, which we hope in '87 we're going to start picking up. If you can see it, you'll see that it shows up in nondestructive examination of welds.

It's a long-term start, we hope, for plant life extension.

Again you will see here the steam generator program is tailing off and, in fact, '87 is the last year we expect for that. That amounts to decommissioning for that program.

MR. SHEWMON: When you get to plant life extension, there will certainly be stuff like we just got done talking

about on metal components. Do you do work on polymeric material, or where does that get covered?

MR. SERPAN: My branch does not touch the polymeric materials. The only one that would come close to that would be Bill Morrison in the Electrical Engineering and the general aging program that Jip Bor is putting together for him. I think those kinds of things will be looked at in that program.

But there is a large effort underway by Bill Morrison's branch now to get that aging program off the ground and to have definition of what needs to be looked at. And I think in the next year or two, what they are going to be doing in large measure is trying to identify what has to be looked at.

So what I am planning for out here is to have some money available so that about the time they start finding out what we really think we have got to look at, I've got some money to try do it. That's the plan on that.

MR. SHEWMON: Okay.

[Slide.]

MR. SERPAN: At the very back of your handout, you will see -- this is weird because I've got -- you will see a listing of the FINS, and you will see the amount of money I've got for them, and I can't show it in public because I've got it for you for '86 and '87, and normally I'm not allowed to do that.

I'm not going to dwell on that. My point is that the discussion today is going to encompass most of these FINS, and those are the ones that we are concerned with. So I don't want to make a great deal about that. But if you want reference to them as we're going through all the presentations, you will have it in the back of mine.

[Slide.]

The four areas in primary system integrity that I want to go through quickly start with reactor vessels. The safety problems are to the aging, degradation of vessels in the primary system. That consists mostly of radiation embrittlement, degradation of fracture toughness, crack growth and prediction and evaluation methods.

The point of all of this work, the regulatory use of this, is to evaluate the accuracy and feasibility of applicants' submittals on FTS, to provide validation of the PTS rule and screening criterion, to evaluate submittals on revision of PT curves and exactly what you've just heard about.

But much of the research that's done to provide that comes out of this branch.

To evaluate submittals on vessel flux reduction through adoption of the low leakage cores, and to provide the basis for NRR decisions on operation of reactors and components in the presence of cracks. Fracture mechanics

work.

[Slide.]

Piping and steam generators. The problems here are the integrity of cracked and degraded piping -- I'm sorry, this includes piping and steam generators. Cracked and degraded piping. And you have heard a great deal about BWR pipe cracks over the last couple of months or year or so. The integrity of cracked and degraded steam generator tubing, and corrosion and cracking by the coolant environment.

I will have to say that our corrosion program is quite small, compared to what we would like, but that's the fact of life.

The regulatory use for all of this is to form the basis for NRC evaluation of the ASME IWB-3640 rules for the evaluation and acceptance of flaws in BWR piping; the experimental basis for formulation and acceptance of the rule on leak-before-break in piping; to form the basis for NRC acceptance of fixes, such as weld clad overlay, and replacement materials, such as 316NG for long-term operation in BWRs; and finally, to form the basis for upgraded steam generator tube plugging criteria and improved in-service inspection plans.

Yes?

MR. SHEWMON: If I wanted to ask questions about the toughness of weld overlay material, will an opportunity for

that come up tomorrow morning or this afternoon?

MR. SERPAN: The toughness of the weld clad overlay, Milton may have some information on it, but we weren't planning on going into that. The schedule did not lay out that kind of -- let me see -- no, Mike has some of that. Yes, that's right. Mike Mayfield. He'll be here tomorrow morning. He will have informatior on that.

MR. SHEWMON: Well, I see the utilities now want to come back and say that that's such wonderful stuff we can probably leave it on there forever.

MR. SERPAN: Well, interestingly enough, some of the research results that we're coming up with seem to suggest that the stuff is quite good, and the cracks don't seem to be extending.

However, it still is uninspected. So --

MR. SHEWMON: Well, they would argue that if you prepare the surface, it is inspectable. That's something else we can discuss, too.

MR. SERPAN: Mike Mayfield should have answers to that.

MR. SHEWMON: Good.

MR. SERPAN: Okay.

[Slide.]

The last of the general overviews is in nondestructive examination. And the safety problem is

accurate detection and sizing of flaws in carbon steel vessels and nozzles, in cast stainless steel elbows and bimetallic joints and, of course, I didn't put in there wrought stainless steel piping as well.

The use is to form the basis for recommendations for upgraded ASME Code or Reg Guide requirements for inspections; to evaluate qualification of personnel, procedures and equipment for UT of vessels and piping. And we have been making great progress in the area of qualifications these days and Joe, I think, will tell you about that shortly.

It is as a basis to review applicant's procedures for plant inspections, and to judge the acceptability of the NDE results and the ISI-NDE results, and to provide independent third party checks of flaw detection and sizing. And we have a couple of very nice instances of that from the Staff UT, which I think Joe will probably talk about.

[Slide.]

The last couple of viewgraphs summarize the foreign interactions that we have in the branch.

First of all, let me mention -- I'm sorry, I got that off the page.

The exchange agreements, the formal signed exchange agreements that are particularly active. This doesn't imply that we don't have others. These are the ones that are active in the areas in which they're mostly active with.

In Germany, it's mostly in the area of materials with Prof. Kussmaul at Stuttgart. But we also have exchanges in the HDR, and we have a resident engineer in Germany.

In the UK, we have a great deal of exchange and cooperation in NDE. In fact, we have more in dosimetry, I would say. We have three or four groups working in dosimetry. But we have good interchange in fracture mechanics and piping and vessel fracture mechanics.

With the Swiss, we have an active exchange in fracture mechanics, primarily for materials and vessels and piping.

It's interesting the way we get a number of calls from the Swiss, I would guess every month we get a couple of calls on specific direct licensing questions. And very frequently we have to refer some of those calls right back down to Bethesda. But there's a very active exchange with the Swiss, between ourselves and licensing.

With Belgium, we have a very active exchange. That is mostly in dosimetry and also annealing, where we have been following annealing.

MR. SHEWMON: What are the main questions the dosimetry effort is working on now, or in that area?

MR. SERPAN: Well, the Belgins are doing the Venus program which is the extrapolation of the flux from the outer edge of the fuel through into the pressure boundary. And the

problem with that before was that the reactor physics guys would take their code and they'd start in the core and then go out to that edge. But they would never carry that all the way through, and the shielding fellows would just have to take whatever junk they got at the core edge and take that all out into the vessel. And that hasn't worked too well.

So the Belgians started at the core and they went all the way through that core edge, through all the thermal shields and the water temps and out into the pressure vessel wall to make an absolute correlation. And that's proved to be very, very interesting work.

They are going to now put in a low leakage core and they are going to validate the flux extrapolation through that low leakage core. We think that will be interesting.

MR. SHEWMON: The low leakage core in the sense of putting one in the computer?

MR. SERPAN: No, no, they actually have a mock-up, they have a very small low power mock-up there. It's called the Venus assembly, and they literally select the pins and put them in there and then they run that thing at very low power and it's loaded with dosimetry. So they measure that at the core edge and they measure it in the shielding, simulating the pressure vessel, but they calculate it as well.

MR. SHEWMON: And the British?

MR. SERPAN: The British? NESDIF. This is cavity

dosimetry, where they are simulating cavities and also the streaming that might go up there toward the nozzles, and they are very interested in that particular phase, so we are cooperating with them. They are doing the primary model work. We are doing some calculations and some measurements. So that is cavity dosimetry.

MR. SHEWMON: Okay.

MR. SERPAN: Okay. The steam generator group project is the famous project with the Surry steam generator out at Battelle. This is the one where we were able to get France, Italy and Japan from overseas to contribute $1 million over five years and, of course, EPRI also contributed $1 million over five years in that program.

The ICCGR cooperative program is the International Cyclic Crack Growth Rate Program. That's been going on since the late 1970s. It's got the United States, virtually all countries in Europe and Japan. They pool all their research in crack growth rate, they coordinate the results and they actually plan work, and it's been enormously successful in minimizing the amount of work in crack growth rate that any one country has to do, but everyone benefits.

MR. SHEWMON: Let me back up. The five-year period on the steam generator was when to when?

MR. SERPAN: '87 is the -- well, let's see. '86 is the last research year, then, isn't it? January '87. Okay.

MR. SHEWMON: Okay. Any end in sight on the ICCGR, or --

MR. SERPAN: Well, that runs -- because we are doing crack growth rate work within the HSST program, and as long as we do that, we simply put that into that larger pool. So as long as people are independently doing that work, it will go. So it's just simply coordinated by those people.

MR. SHEWMON: Maybe Milt will get into this tomorrow, then. Do we have information, or are we getting information out of that on whether cracks grow faster in irradiated pressure vessel steel than annealed?

MR. VAGINS: Bill Vagins.

We have done work on irradiated pressure vessel steels and shown no effect on environmental crack growth. But this is out of reactor.

The only people who are doing work in reactor are the Germans that I know of. I have not been able to get any data out of them. I think they still have an ongoing program. They have a string of specimens which they put into the Karl Reactor,and that's still ongoing. As soon as that becomes available, we'll get it.

But as far as --

MR. SHEWMON: When you say you couldn't find an effect, was this at 30-year predicted life --

MR. VAGINS: Yes, sir. Taking specimens,

irradiating them to -- you do it kind of backwards. You take a specimen and irradiate it 2 times to the 19th neutrons per centimeter squared e greater than 1 MEV,a nd then we put into a BWR-PWR environment, both, and get no effects, no change.

In other words -- it's interesting, though. The fracture surface is different, but there seems to be two conflicting processes going on. The fracture surface is different from the irradiated specimen as compared to an unirradiated specimen, but the rate --

MR. SHEWMON: It is stronger, but the cracks don't run fast--

MR. VAGINS: Exactly. There is a hardening effect, and then some other effect, and the cracks do not go any faster.

But again, this is out of reactor. So we haven't got the presence of gamma heat.

MR. SHEWMON: Well, it gets back to what Willertz was talking about this morning some. At least there they can see an effect from the irradiation, and if you went simply by does it harden it or does it cause segregation, you'd expect that to carry out.

MR. VAGINS: Well, of course, they were limiting themselves to austenitic steels and stainless steels, and they were talking about thresholds of 10 to the 22nd.

MR. SHEWMON: Yes, I know.

MR. VAGINS: We're getting nowhere near that kind of fluence.

There is some doubt as to whether the in core performance with radiolysis going on might make a difference at the crack tip. But as you can imagine, these are extremely expensive things to do.

MR. SHEWMON: Yes. I figured no university would have given you that sort of data, so it's something we'd have to do.

Thank you. Go ahead.

MR. SERPAN: Okay. Gundremmingen is the next foreign program. The purpose is to evaluate in situ irradiated pressure vessel steel in the German reactor. The Germans at MPA Stuttgart now have a contract from us to machine and test the reactor pressure vessel steel. That's proceeding slowly, but nonetheless it is proceeding.

I am now working up a parallel agreement with that, with the British, to do heavy water irradiations of the archive steel and to do radiation damage analysis of all of the steel.

We have identified two pieces of archive steel of the Gundremmingen. We will be doing light water reactor irradiations here. The British will be doing heavy water reactor irradiations. We think that we will both bracket and even match the thermal to fast ratio of the steel from the

reactor itself, but we will also then have test reactor irradiations, we will have the in situ irradiations, and we even hope that we might be able to con somebody in Germany into doing the surveillance capsules. And we will also have high thermal to fast ratios from the D20s. We think we may have enough spectra variations to help understand that.

But this is perhaps as important: We've got the British as well as, I think, many throughout the rest of the world interested very strongly in radiation damage analysis, and Bob Odette is going to be chairing a radiation damage workshop at Monterrey in I guess it's next week, isn't it. And we hope out of that, that we get the same kind of an international group as the ICCGR to work on radiation damage over the next four or five years and see if we can't come up with some good answers on what happens.

We have strong interactions with the CS&I in Paris. The working group No. 3 on primary system integrity is chaired by myself. We worry about fracture mechanics changes, international intercomparisons of problems. We are trying to get some stress analysis international intercomparisons going. And of course, the PISC-III NDE work is a very strong part of that.

There it is. The PISC-III, the idea is to evaluate advanced NDE methods and to study NDE on real flaws in real environments, with real equipment.

MR. THOMPSON: Before you go on, you indicated some exchange agreements in NDE in the UK. With whom are those?

MR. SERPAN: That would be Roy Nichols again and Brychan Watkins, primarily at Risley, with the defect detection trials. But Joe can tell you more on that. But it's primarily there.

MR. SHEWMON: Do you ever talk to the CS&I on the inspecting of cast stainless steel?

MR. SERPAN: Oh. Goodness, yes. There is a round-robin that Joe has organized and is underway throughout Europe and the U.S. on cast stainless steel. He can tell you more about it.

MR. SHEWMON: I'm interested because I sent a question over with a group to talk to the group and they said that's an interesting question, we'll get back to you. And in four months they haven't, so either it got lost or I don't know what.

MR. SERPAN: No, there is a very strong effort on casting the steel. In fact, the first phase of that round-robin is over with, and they are in the process of evaluating and reporting that now and they are also now starting planning for the second phase of that cast stainless steel round-robin.

So there's a lot of work.

Okay. Quickly, the last area of dosimetry. We have

a very large cooperative program on dosimetry. I've told you about Belgium already.

[Slide.]

The Venus, and doing transport calculations in England. It's primarily NESDIP in radiation damage analyses.

With Germany, there is radiation damage and spectrum analysis. We expect to have the fellows at the IKE in Stuttgart actually doing the calculations on the Gundremmingen reactor, and we are getting gamma heating numbers also from them, which is also very important.

We also, in dosimetry, cooperate with the ASTM/EURATOM dosimetry symposium, which was held every two years.

Piping IPIRG. This is for piping tests under extreme conditions under -- and that is under formation now. I don't know whether we will have an international piping integrity research group or not. But in November -- October 31st, November 1st at Battelle-Columbus, we will have the organization meeting of that and everybody, we hope, is going to join, and I hope that by November 3rd, we've got an IPIRG and Mike Mayfield can tell more about that tomorrow.

That is all I have.

MR. SHEWMON: Why don't we take a 10-minute break and then we'll come back.

[Recess.]

MR. SHEWMON: We now turn to NDE.

[Slide.]

MR. MUSCARA: My name is Joe Muscara. I'm with the NRC Research Office in the Materials Research Branch.

I had a number of topics to cover in the agenda. I have shifted them around a little bit. I will be talking mostly about our NDE work, almost all of the work, but not quite. The first topics will include the acoustic emission work, then I will get onto the SAFT-UT project. I will then talk about the NDE FM program, which covers a number of areas of interest to us at this point, and then I will finish up with some results on the steam generator eddy current testing.

[Slide.]

The first project has to do with the acoustic emission monitoring, continuous monitoring on line of operating reactors, and the work is being conducted at Battelle Pacific Northwest Laboratory.

This work began in 1976 and was set up in three distinct phases. There is a laboratory phase, an intermediate scale phase, and a field validation phase of the work. In effect, we wanted to prove the field usability of using acoustic emission for continuous crack monitoring in operating reactors.

In order to do so, we want to establish that, number one, we do have acoustic emission produced by fatigue

stress corrosion cracking, et cetera. We then want to be able to pick out this acoustic emission from the noisy reactor environment, so we need to develop methods for doing this.

We need, of course, to be able to monitor in the reactor environment, so we need transducers and techniques that would stand the high temperatures in the operating environment, and also, once we do detect a growing flaw, one wants to know what is the severity of this flaw because regulatory decisions have to be made. So that another portion of the work has to do with correlating acoustic emission to flaw severity.

[Slide.]

Now, this first part of the laboratory work is essentially completed. We, in effect, have completed intermediate scale testing, which was a test at MPA in Stuttgart, where we were essentially tested our relationships on a large vessel, about a 5 foot in diameter, 15 foot long vessel, 5 inches thick.

We had a number of cracks implanted in this vessel, and these were cycled for about a year and the crack growth was monitored and predicted during the test, and at the same time, several cracks occurred where they were natural cracks, not expected, and these were identified and the severity more or less evaluated.

We are now at the stage of getting ready to monitor

an operating reactor.

So in the laboratory work, we have identified methods for monitoring the reactor environment. The major problem there is to be able to get around the fluid flow noise.

We do get acoustic emission from fatigue and from stress corrosion cracking, and that has been established in the laboratory. A signal identification method has been developed, and this, in effect, distinguishes signals from cracks versus signals from noise sources, and it uses a pattern recognition method.

An interpretation method for the severity of flaws has been developed, correlating the acoustic emission rate with a fracture mechanics parameter such as the crack growth rate, or a delta K.

We have developed instrumentation that is field usable instrumentation for actual reactor monitoring. Actually, we are now ready to monitor an operating reactor. This is Watts Bar Unit 1.

MR. THOMPSON: There was a report on some of this work given at a meeting in February by Phil Hutton, and he had reported some very positive results on some of these points, but unfortunately he couldn't describe exactly what he had done because there were some patent issues at Battelle that kept him from saying exactly how he did this.

MR. MUSCARA: No, there are no patent issues.

MR. THOMPSON: Can you say something about any special advances? Some of these are problems a lot of people haven't been able to solve.

MR. MUSCARA: I have a few data graphs. I was not going to concentrate on data, but let me show those, and if you still have questions, we can get into more detail.

MR. SHEWMON: I was always some bothered by this setup not because they couldn't follow a crack if they knew it was there -- that they have done very impressively on -- but at least several years ago with whether or not they could tell the noise from a crack which they hadn't ever studied and identified before from the general noise in the background.

MR. MUSCARA: In effect the philosophy of the project was to study the crack growth in the absence of noise in the laboratory to first establish feasibility. We have done that. We also separately have studied noise sources in the laboratory. And then we have pattern recognition methods that distinguish the two. Then, of course, we run tests where both crack growth and noise sources are there and we can distinguish.

Now, let me amplify on that a little bit. The pattern recognition methods we developed in the laboratory were then tested on the MPA vessel. Our objective in the laboratory had been to be able to use these pattern

recognition methods to do a correct decision 95 percent of the time. We want to be correct that there is crack there, that we are calling cracks cracks and noise noise. The error we always wanted was in calling the noise a crack, so we wanted to be conservative. We were able to do that and be correct 95 percent of the time with the techniques we developed in the laboratory.

When we tried these techniques on the pressure vessel test at MPA, in effect we were only doing about 70 or 75 percent correct identification. We have then taken the data from MPA and developed a new technique for distinguishing the crack from the noise source, and we are back to 95 or better. We are actually 99 percent correct crack detection versus all the noise sources.

So we had a laboratory model which worked quite well in the lab. It did not work as well in the intermediate scale tests. We developed a new model, and now that has worked on --

MR. SHEWMON: How does somebody convince himself that this would work on 95 percent of the crack when they are probably working with three cracks? They aren't working with a thousand different cracks in different materials, so you can't say that they got at the 90 percent that way.

MR. MUSCARA: That's true. I'm talking about we had a number of cracks grow in the laboratory tests, and we have a

number of signals as a function of time. So we are correctly interpreting 95 percent of all the signals that come from cracks as being cracks.

MR. SHEWMON: From the same two or three cracks in the same material.

MR. MUSCARA: We have had half a dozen cracks on the MPA test. We have had dozens of cracks in the laboratory work, but it has been the same heat of material under many different conditions, under different conditions of crack growth rates, different conditions of R ratio, different conditions of crack growth rates, cyclic rates, and so on. But the material we have concentrated on has been A533B pressure vessel steel at one heat, that is correct.

We have also, while I am on this point, checked out this pattern recognition method on data that was obtained on Watts Bar reactor during their hot functional testing, and in effect, we are correctly distinguishing crack sources versus noise sources.

Now, on this test we had implanted --

MR. WARD: How do you know that? From the Watts Bar data how do you know?

MR. MUSCARA: At the Watts Bar we had several sources. We had one indication from their nozzle, which corresponded to some radiographic indications. We cannot say that is correct. But then we also had a fracture specimen

that was --

MR. WARD: But you were getting a noise indication that it was a crack; is that what you said?

MR. MUSCARA: The preliminary indication was that even that it looked like a crack by recognition method.

We also had a fracture specimen attached to a pipe in the primary coolant system in the Watts Bar, and in effect had crack growth going on in the specimen, so that the crack growth signal was coupled from the specimen through the reactor piping into the sensors. That also was detected and it does fit the pattern recognition method we are working with now.

So we are fairly comfortable that actual AE from large cross-section of materials can be distinguished.

[Slide.]

In this graph in effect we are showing the level of noise versus level of graphs, and this is actually work from the Watts Bar reactor during its hot functional testing compared to signals obtained in the MPA test. So we can see at low temperatures as the reactor is coming up there is a lot of flow noise, and the level of the noise is somewhat higher than the maximum level of cracking sources that we get.

But as the temperature goes on up to operating temperatures, the flow noise goes down and there is quite a bit more signal to the cracks. So that we could not monitor

during startup and shutdown, but we definitely can monitor during reactor operation.

Our technique, essentially, for getting rid of the flow noise is to monitor using transducers. They are peaked at a certain frequency level. We are looking at about 375 to 500 kilohertz. In this range there is very little flow noise and there is still a lot of content on acoustic emission from cracking.

Very briefly, the new pattern recognition method takes advantage of the nature of the signals that are obtained from cracking. We have found that all the cracks, when they travel through a wave guide to our sensor, always exhibit this pattern. There are always three bursts, with reasonably constant spacing between the bursts, and an algorithm has been developed whereby automatically we can match a pattern to this kind of burst.

MR. SHEWMON: This is a fatigue crack growth pattern? Is that a sample that --

MR. MUSCARA: This is, in effect, a piece of data from the MPA test, which was a vessel test undergoing cyclic fatigue, but the same kinds of signals were obtained from the Watts Bar test and all laboratory tests.

MR. WARD: So this is what you get on each cycle? You would get a pattern like this?

MR. MUSCARA: No. For some cycles they are quiet,

and other cycles give out several of these bursts, but always the crack comes in bursts like this. Noise sources don't have the noise of burst-type signals.

MR. SHEWMON: Is there any idea whether this signature comes from the fact that the flow itself on a dislocation level comes in three bursts, or is it some sort of an echo that is characteristic of what happens if you pulse a piece of metal?

MR. MUSCARA: We don't know that. It's empirical data. We do know that the spacing is affected by the wave guide itself, the diameter and length of the wave guide. So it is affected by the wave guide geometry, and I'm not sure what the signal would look lik at the source, but it seems to be correlated with the cracking.

I must say there are usually at least three of these. There may be more. But noise sources do not give this noise, burst-type pattern.

[Slide.]

On the correlation for acoustic emission versus severity, we have done many, many tests under different conditions: room temperature, high temperature, different cyclic frequencies, different R ratios, base metal, weld metal, and all of the data seems to fall on this plot. We can calculate the occurrence versus the crack growth rate or delta K.

So from this, one could get the severity of the flaw, knowing something about the stresses in the reactor.

[Slide.]

Briefly, to move on to the -- I might say most of the work I am talking about is completed, and by 1987 the work on the acoustic emission will be finished, and we are really now in the validation stage. There are a few things that were brought out by the test at MPA that we want to look at in the laboratory.

One of the remaining tasks, of course, the major task is to do on-line monitoring of the Watts Bar reactor. In conjunction with this work, TVA is joining the program to help fund this kind of work on the reactor so that they can get the technology transfer and use this technology for monitoring reactors.

We do need some more work on IGSCC crack monitoring, and that is scheduled and is ongoing right now. We do know that we can get AE from IGSC, but we have not obtained any correlations between the severity of the IGSCC and the acoustic emission. So the major emphasis on the IGSCC work is to try and correlate the signals to the severity.

One of the items that we want to look at is to test in fatigue crack growth at lower crack growth rates than we have looked at so far. The work at MPA indicated that we do get acoustic emission but not as high a level as we get with

the higher crack growth rates, so that we want to test on lower crack growth rates. We have looked at about 10 to the minus 6 so far. We want to go down to as low as 10 to the minus 8, just to see whether there is a saturation effect where no acoustic emission would be developed.

We want to finalize the data interpretation method. That is essentially pattern recognition work by doing some more work in the field to make sure that it does work.

We have already been working in developing an ASTM standard, and I believe that is in the validating stage so we are going to try and finalize that, and of course we are going to work with the Code to try and get this technology accepted by the Code.

MR. SHEWMON: This is a standard for what?

MR. MUSCARA: It is a standard for using acoustic emission for pressure vessel testing.

MR. SHEWMON: For testing. This is on hydrotesting.

MR. MUSCARA: No. That standard has already been developed. We are now developing another standard for on-line monitoring. It is the methodology, to include calibration and methods of use on an operating vessel.

MR. THOMPSON: So, for example, would that include this empirical observation you have observed about the pattern of the signals? Would that be a part of it or would it be something more general than that?

MR. MUSCARA: It would probably be more general, but for the ASME Code, we will have to give them the specific details on how to monitor for reactor operation, and in that case we will give the details on the pattern recognition, on the modeling, on how to evaluate the cracks, et cetera.

MR. THOMPSON: Okay.

MR. MUSCARA: I might say we are also planning on putting out a detailed final report so that the technology would be available to the industry.

Again, I want to tell you there's no proprietary nature of this work at all, so if you need more information from Phil and you're not getting it, let me know, and we'll try to --

MR. THOMPSON: I'm sure that was something short term. I didn't mean to imply that he was being secretive or anything.

MR. MUSCARA: But there isn't any secret.

[Slide.]

The end project essentially is a demonstration of this technology and developing standards and codes for its application, and the technology transfer, primarily through TVA interaction at this point.

MR. SHEWMON: What is the operating schedule for the Watts Bar plant right now?

MR. MUSCARA: They don't have a license yet. We are

ready to monitor as soon as they go up and start the fuel loading.

MR. SHEWMON: Yes, but TVA is having trouble operating any of their reactors, and I don't know whether that -- does that slow down the Watts Bar work or --

MR. MUSCARA: No. I have heard that they are aggressively pursuing licensing of the Watts Bar. I also heard that they would be monitoring in May and then it was June and then it was August. I'm really not sure when.

MR. WARD: You can't move the program, can you. You can't move the program?

MR. MUSCARA: Not really. It takes quite a bit of time and money and instrumenting the reactor. We essentially were lucky to be able to instrument the reactor while it was under construction. And that by itself was a costly project. But trying to go into a reactor and do backfitting would probably be a bit more difficult.

MR. WARD: Joe, earlier Chuck mentioned the cooperative programs with other countries, and I have the idea that the Swiss are doing some work in acoustic emissions. Are they doing something significant there or do you have any --

MR. MUSCARA: Yes, they are.

MR. WARD: -- contact with them?

MR. MUSCARA: They are doing more theoretical work, trying to determine the signal at its source. The philosophy

we had used was let's do the engineering work and try to apply the technology and so we're quite removed from the kinds of things that they are doing.

We have cooperated with the Federal Republic of Germany on this work, where we have actually transferred some of our technology to them and they are doing some additional vessel testing there. In fact, they're taking a full sized vessel and pressurizing it with flaws, and using acoustic emissions to detect these flaws.

Up to this point the Germans have been concentrating on the hydrotesting. Our work is really only work that's been concentrating on actual on-line reactor monitoring. It's a much more difficult problem than hydrotesting, but of course if it does work, it has a much higher payoff.

MR. SHEWMON: On the Watts Bar plant, how many sensors do you have, or how many per running meter of heavy pipe do you have to have to be able to survey it?

MR. MUSCARA: We are monitoring three areas on the Watts Bar. About a third of the cylindrical section of the vessel, and that is taking one array of sensors. Each array in our case is four sensors. We then are looking at a nozzle, and that has another array, and we are looking at a primary piping. And again that's also one other array.

MR. SHEWMON: This was one end of the vessel, one of the nozzles?

MR. MUSCARA: The barrel section, one third of the barrel section of the vessel, and that takes one array. One nozzle, we are looking with one array. Now we may be able to get away with fewer transducers than that. But I think because of the geometry the one array is quite acceptable at this point. And we do have another array on a pipe system. And we will be doing additional work on the plant itself to try and determine what is the least number of transducers we can get away with.

But it looks like for this application that we are going to be looking at things 20, 30 feet away.

MR. SHEWMON: And these will record into a tape, the intensity goes above some threshold, and otherwise it's just static, or what?

MR. MUSCARA: The way we have set up -- we can do two things. For the research work we can actually tape almost every second. We have done that. For the operating work we essentially try to make use of different pieces of information. One is to try the algorithm. So first the signal is sensed, it's located, and then it goes through this black box to see whether it's a crack signal. If it's a crack signal, then we put it onto a recorder that will be saved for future reference, or we can have an alarm, depending on the severity. So it goes through a number of steps. It's detected, recorded if it's a flaw signal, and then goes

through the analysis black box to see what the severity is. And it's followed as a function of time.

MR. DILLON: As far as the location process is concerned, does that work equally well on events that are not cracks but some other form of noise? Can you identify your noise sources?

MR. MUSCARA: Yes. The location does help, and at the noise sources tend to move around in the system. But you can locate a noise source. You can locate most noise sources. A lot of the noise sources are white noise, so there is no signal to that source, so then there is no location of that, although I get into some work on that where I show otherwise.

But if it's a continuous noise source in this work, it would not be located.

MR. WARD: Joe, I know you don't want to spend too much time on this, but I want to get a feel for this. These bursts of noise characteristics -- I mean how often do these bursts come in say a crack in a vessel? I mean are these repeating every --

MR. MUSCARA: In a lab test we almost always get acoustic emissions on the up ramp on the cyclic loading.

MR. WARD: Okay.

MR. MUSCARA: Now many people will report data where they're getting their E in unloading the sample. That usually

turns out to be rubbing. They are able to discriminate from a crack growth and we have done work in the lab to show that the factor of E comes from the peak load more or less.

MR. WARD: Okay. Now a vessel is under -- is under a continuous load, or is there pressure noise that's giving you cycles, or what?

MR. MUSCARA: Oh, in the laboratory, in the work we've done?

MR. WARD: No, either. Let's say what do you expect in a plant? In a reactor vessel, let's say?

MR. MUSCARA: Vibrational factigue, thermal fatigue. What we have done in the lab is a cyclic fatigue where we essentially have cycled through to pressurize the vessel.

MR. WARD: Okay. Well, what do you expect in a pressure vessel in a plant; in a reactor vessel in a plant? What sort of --

MR. MUSCARA: Possibility it would be thermal fatigue or vibrational fatigue. And if it was in piping, we would look at IGSCC.

MR. WARD: Okay. Now IGSCC, what sort of cycling are you getting there?

MR. MUSCARA: In the IGSCC, you really don't -- the preliminary work we've done did test about one cycle per hour in the high oxygen water environment to get IGSCC. The work

I'm doing now is to --

MR. WARD: Okay, so there's a -- I mean a stress cycle.

MR. MUSCARA: There is a stress cycle so far. The new work I'm doing is I want to assure myself I'm not looking at growth and fatigue in conjunction with IGSCC. So the work we are doing now on the pipes, they have been welded, they have high residual stresses similar to what is in a plant. But we are not putting many cycles on it, maybe one cycle a day or very low load. Most of the work has been done on IGSCC for studying the corrosion phenomenon and other phenomena has been done under a cyclic load where the load is almost up to yield. So it cycles between essentially zero and E.

MR. SHEWMON: Watts Bar is a Westinghouse PWR? Or what is it?

MR. MUSCARA: Yes, I believe so.

[Slide.]

Well, the other work we have on acoustic emission is to essentially develop the acoustic emission for leak monitoring of primary systems.

Now the reason we have gone to the acoustic emission is that we expect a much better potential, using this technique, for increased sensitivity of leak detection, better location capability. In effect, many of the techniques that are used now do not have a location capability.

We would like to be able to discriminate the leaks from other sources and would like to be able to discriminate leaks from cracks vs. leaks from seals, for example.

And, of course, we'd like to get a quantitative information on the leak rate. So we have set up laboratory work to try and study these different parameters. This work is going on at Argonne National Laboratory. We have developed a test loop for testing in the lab. We have carried out studies on intergranular stress corrosion cracking remote from the field, and also on fatigue specimens, both thermal fatigue and mechanical fatigue.

We have acquired background data from operating plants in conjunction with the reactor monitoring work, and we already have built an advanced system for leak monitoring. It does essentially signal processing for location of flaws, identification of flaws, and it is a flaw source vs. a non-flaw source, and for quantifying the leak rate.

[Slide.]

The status is that leaks can be detected, located and probably sized -- and I can show you a little data on that -- using commercially available equipment, but we are using some equipment that has been built specifically for this project.

And, of course, all of the analysis algorithms are specific to the project.

We have been able to detect leak rates as small as .005 gpm at about 1 meter away from the source. At 1 gpm we can detect it a meter away. This is for high background noise. The high background noise essentially being an area next to a pump in a plant.

We have been using spectral analysis to distinguish IGSCC from other types of sources, so we can now distinguish IGSCC from valves or seals or fatigue cracks, and we are able to locate the leaks with fairly high accuracy.

All the work that's going on, that has gone on in the past for leak detection, the location essentially is by a single damping evaluation, because the acoustic emission from a leak source is a continuous source. There are no burst signals. They are received at different times by different transducers. So you cannot do any triangulation work.

What we are doing is a cross correlation technique where in effect we are matching the signals from two different transducers and shifting them in time so that knowing the speed of travel within the material, you can locate the other source.

So the cross correlation electronically performs this function where the signal is evaluated from different transducers, shifted so it matches up, and then we look at essentially the shift in time. And we can get very accurate location with this technique.

We also have obtained correlations between the leak rate and the level of acoustic emission.

MR. WARD: Joe, let me ask you, why are you -- what's the incentive for this, for leak detection by this method? I mean, you know, this morning we were talking about more traditional methods of leak detection, such as looking for drips.

MR. MUSCARA: The traditional methods don't seem to have the sensitivity. The reg guide requires sensitivity 1 gpm. I'm not sure that the sump pumps and the other systems have that kind of sensitivity.

Beyond that, we are supposed to be able to discriminate leaks from cracks vs. leaks from unidentified sources. And there is a different requirement. Leak from cracks, you're supposed to shut down as soon as you identify.

On the other hand, depending on the reactor system, most BWRs have a 5 gpm leakage limit, and up to 5 gpm people don't even try to discriminate whether this leak is coming from an unidentified source or an identified source. So that it's assumed to be unidentified, and if we could have a very large crack leaking and not doing anything about it.

And of course the systems are available now to not have any location capability to help locate the leak and determine whether it's a crack or not a crack and take some action.

So there are requirements for leak detection right now. I think the systems in place cannot meet them, and most importantly, what's happening very recently, of course, is that we're accepting leak before break criteria for certain piping systems, and one of the major defense-in-depth items that you have left is the leak detection system.

So we have to make sure that we can detect leaks from cracks and locate them.

A particular problem with IGSCC, for example, is that you could have a very large crack -- and we have seen this, for example, Duane Arnold, where the crack is quite large on the inside, penetrates a high amount through the wall. It can only penetrate through a small section, and you have very low leakage from the crack. And in fact not detected by standard means.

So we could have a situation we have a large crack with a small leak rate, because of the crack geometry and also because they foul up. There's a lot of corrosion product that fouls up the crack. So it becomes important to have sensitive leak detection systems.

[Slide.]

Just a data graph to show the capability for detecting leaks in the plant. We are comparing the signal-to-noise ratio under different levels of noise within the reactor. We have a low, a medium and a high level of

noise. And you can see, for example, that in the -- if you look at the low noise level, one could detect a leak up to about six meters away for a .1 gpm leak to have considerable signal to noise.

In a high noise region, we are more limited. We can detect a 1 gpm leak, for example, at about a meter away.

MR. SHEWMON: How is that transducer attached to the pipe?

MR. MUSCARA: In both projects we have gone to using a wave guide. That is it's a rod, stainless steel rod. The attachment of the stainless steel rod is a couple of ways, either magnetically attached or through a spring-loaded mechanism.

MR. SHEWMON: Now does that have to be a polished surface on both sides or --

MR. MUSCARA: No. Of course, it helps to help a good surface. But we found there is enough pressure that the rod itself comes down to a point. And we have enough pressure to have good contact with the surface. The optimal way would be to drill and tap and screw the rod into the component. But we find that all this work uses the rod either under magnetic force or spring force. At the end of the rod we then have our electric transducer and preamplifier.

MR. THOMPSON: What approximately is the diameter and length of that rod?

MR. MUSCARA: The diameter of the rod we are using mostly is 1/8th inch, and we use rods as long as six, seven, eight feet on the Watts Bar reactor, for example. In most cases we just need enough in the piping to get through the insulation and get away from the high temperature. And in that case, one or two feet.

[Slide.]

This is just showing the correlation between the signal strength, essentially, and the leak rate for IGSCC, thermal fatigue cracks and fatigue cracks here, and we have looked at some flanges and valves on here and it looks like there is a consistent relationship between the flow rate and the acoustic emission level.

Unfortunately, so far we only looked at flow rates up to about 1 gpm, and our following work will look at larger flow rates.

The way we get variations in the flow rate is by using a crack specimen and then using different amounts of loading on the specimen to open up the crack and get more flow through, but this is flow using normal reactor conditions, that is, 1100 psi, 550, or 2200 psi, 550.

MR. SHEWMON: That one curve with no data points labeled IGSCC is to show that you get a louder signal from that than you do from --

MR. MUSCARA: That's true. The IGSCC does produce

more noise than the fatigue cracks, but it also has a different spectral content so we can distinguish this crack from these other sources.

MR. SHEWMON: That is because they are finer and more --

MR. MUSCARA: Because they are finer, more tortuous. The fatigue cracks tend to be smooth. As a matter of fact, the fatigue cracks in mechanical fatigue are not so different from two plates or a drill hole.

MR. SHEWMON: What does TFC stand for?

MR. MUSCARA: Thermal fatigue crack.

MR. THOMPSON: On the thermal fatigue crack data, it would intuitively seem to me that the noise it makes per gallon would have to do with the tightness, the distance between the faces, and what percentage of the fluid had to be disturbed by the faces. I don't see anything indicating that here. Has that been studied?

MR. MUSCARA: To a limited extent. It seems, though, that most of the AE is connected with the flashing of the steam once it exits.

MR. THOMPSON: I see.

MR. MUSCARA: Quite often the path that it takes may affect the flow rate, but for a given flow rate it does not seem to affect the AE.

MR. THOMPSON: So it's not the turbulent flow of the

water through the crack but something that happens when it --

MR. MUSCARA: In general, but the IGSCC does look different so that has got to be playing a role but I think it is probably a smaller role.

MR. THOMPSON: Okay.

MR. MUSCARA: And of course, there are temperature effects and we studied those. Different temperature, same leak rate. It would have a different AE. But again, we do have parameters for that and one can correct for the temperature.

[Slide.]

The future efforts on this project are essentially to complete the software development for the leak detection system. We are righ' now evaluating the field system and we are improving the software. We are doing the automated signal analysis.

We want to carry out experiments from cracks that give leaks more than 1 gpm. That is not by opening them up but essentially getting longer cracks. We want to see the effect of geometry also. So if there is a 1 gpm leak from a small crack, is it the same as a long crack.

We will be carrying out field tests both at Watts Bar and at the reactor close to Argonne.

MR. SHEWMON: What does GARD stand for?

MR. MUSCARA: That is just a small research lab in

Illinois. It used to stand for General American Research Division of GATX, and they are no longer GARD. As a matter of fact, they are no longer GARD. They have been bought out by some other company, Peabody, I believe, but I'm not sure.

MR. SHEWMON: What is their part of the project?

MR. MUSCARA: GARD had worked for us before in developing acoustic emission for weld monitoring, and they were quite successful in that project. What they did for us was to develop equipment for this work, for the field work.

MR. SHEWMON: Now, a Reg Guide would say that if a utility would want to use that, here is a technique we would recommend? Or if the NRC should decide they wanted to put this in as a suggested procedure for monitoring leaks, then there would a Reg Guide there, or what?

MR. MUSCARA: At this point leak monitoring is required. Under the recommendations of the Piping Review Committee, we had recommendations that leak detection be improved and that the capabilities of current systems be evaluated so we can upgrade the codes and the Regulatory Guides and standards.

Depending on that work, if we find that the current systems are not adequate, then this would go in more as a requirement than as a recommendation. At this point a recommendation to meet the requirements, the existing requirements for leak detection. So it really depends on the

results of the study evaluating current systems and on the needs for this leak before break.

[Slide.]

The next project I would like to talk about is development of the synthetic aperture focusing technique for ultrasonic testing. The current work is going on at Battelle Northwest. The objectives essentially are to engineer a fuel system and to validate the system and to come up with techniques for applying it in the field and for standardizing it within the Code.

The work was initiated in this area at the University of Michigan in 1974, and it represented an entirely new way of processing ultrasonic signals to essentially obtain an image with very high resolution. Now, all the laboratory work at Michigan has been completed. The techniques were developed but essentially did not go to the applications stage at the University of Michigan.

The development at Michigan essentially used only the longitudinal wave. The were not concerned so much with angle beams or contact transducers. In effect, they were developing th theory and proving out the theory.

The work we have been doing at Battelle, again, is to try to make the technique field usable. We have developed the angle beam SAFT-UT technique because this becomes more effective for field inspections. The development there really

is just to modify the algorithms to process having had this kind of geometry for inspection.

We have done algorithm optimization. We had found in the past that this technique was optimum not only for accurate characterization of flaws but also it improves sensitivity, which meant we would use it also for improved detection. A drawback, of course, was that it was slow. There were literally millions of calculations to be conducted per second, and it just takes time even in fast computers to get these calculations performed, and then we are talking about thousands of signals per small area that we want to evaluate, so you are taking a lot of time.

Some of the improvements at Battelle have already managed to get an increase in the processing speed of a factor of 100 over what they were doing at Michigan, so we are at the point now that a pipe can be inspected, let's say a 16-inch pipe, and the data from this pipe processed in about 27 minutes, was the last data we had.

But beyond that, we are also developing a special purpose processor which, in effect, would do only the SAFT mathematics and do them very fast, and with that development, we will be able to improve over Michigan by a factor of 500 to 1000, and with that we can effectively do inspections and process the data in real time even for pressure vessel sections.

As a matter of fact, just yesterday we put together, Battelle finished putting together this black box and have done the first test and it works, so it would be fairly exciting to do lots of lab work now which would get all our parameters worked out and try to get it in the field.

MR. THOMPSON: Are those times for what some people would call line SAFT or for a two-dimensional --

MR. MUSCARA: No. I'm glad you mentioned that. We have been concentrating in the work, even at the University of Michigan -- well, we started out with line SAFT, went to two-dimensional processing, and then we went on to three-dimensional. That, of course, is a much more complex problem, computing and the structure of the algorithms, et cetera. We have been working with three-dimensional SAFT.

Now, most of the work in the rest of the world is still at the two-dimensional SAFT, but we do look at three-dimensional SAFT. The closest that people come to is they do a two-dimensional SAFT and then they stack up planes, but that still does not give you the benefit of 3-D because the relationships don't have to hold in three dimensions and you could get some noise sources building up a signal which is, in fact, noise instead of cracking.

We have found that that helps quite a bit, for example, in cast stainless steel to use the 3-D versus the 2-D. But this processor does three-dimensional processing.

The work, of course, at Battelle has been going on since 1982 and has been concentrating mostly on the IGSCC. To do IGSCC, we have found that just the data from pulse echo is not adequate to image the flaws, especially since they are many of them quite vertical and branching and of low signal response. So what we have done is to develop a tandem technique to image the IGSCC, and that has worked out very well.

We have had some field experience with the system at Battelle already, and this has been at Dresden 3 in February of '84 and at Vermont Yankee. What we did here, in effect, were other teams had inspected these plants, and the teams were qualified NDE processors, but the teams could not agree on what was in the piping, so that as a third party, NRR requested us to go out with Staff to try and resolve the issue: were there cracks or were there not cracks, and if there were cracks, could we size them?

It was quite fruitful to go out. We learned quite a bit from it and were able to resolve some of those problems that were going on between the inspectors.

I must say in all the work up to this point, even the field work, we obtained the data in the field and shipped it back to the laboratory and used a large computer to process the data. We have put together a field system utilizing a smaller computer, and we have now been able to process on

site, and with the improvements in our speed, in June we were able to do this pipe test at Dresden in about 27 minutes.

We have just completed putting together the field system. The black box on the processor works out. That black box really at this point uses only two boards to prove the principle, and to get increased speed, we just add more boards. It's a master-slave type setup.

[Slide.]

Just very briefly -- I don't want to show any complex data, but just to show the difference between the tandem SAFT and the pulse echo, this works on IGSCC and I can show you some pictures that are more complex. This is just a notch, a half-penny shaped notch. If you look at the data from the pulse echo, in effect you can see return signals from the crack from the inner surface, you can see a signal from the tip, but it's not connected. There are no signals from the sides of the notch.

With the tandem SAFT -- well, this is an end view, and this is what we call a front view. It's the same thing in this view. With the tandem SAFT, the entire image is there.

MR. SHEWMON: This tandem SAFT; you talked about 2-D and 3-D before, but not tandem.

MR. MUSCARA: Okay. The pulse echo essentially uses a transducer. It does both the sending of the signal and the receiving of the signal. With the tandem SAFT we use two

transducers, one for ejecting signal and one for receiving, and in this case we have a fixed sender and we scan the receiver, so that we can be a' different positions and pick up signals that are reflected off the crack.

MR. DILLON: Are those indications dimensionally satisfactory?

MR. MUSCARA: Oh, yes. All the data is in the captions. But this notch can be very accurately sized.

MR. DILLON: And that is the whole corona or just the red part, or what?

MR. MUSCARA: These are different signal strength levels. One of the things that remains to be done for IGSCC and cracks is to section specimens that we have evaluated to determine at what signal level do we do the sizing. The SAFT signals themselves, once processed, they have enough resolution that the peaks are not very broad, so you can almost do the sizing at any cut-off level with respect to the amplitude.

If we take a sizing at 50 percent amplitude, we get a fairly accurate sizing. But we want to do some more correlation work on section samples to get that as accurate as we can.

[Slide.]

The field system that we have together now essentially uses modular software. There are many aspects of

SAFT, of the entire process. There is a need for scanning, and that's an automated scan, there is need for doing the SAFT processing itself, which is not mathematically complex but it has many steps to it. There also are display parts to this technology. So all the software is in a modular form so that we can make improvements in any section of the entire system and replace it very easily.

The components we have used are commercially available in the field system. Some of them are very good components, high sensitivity, but they are commercially available.

We have implemented both the pulse SAFT and the tandem SAFT techniques on the field system. It is remotely operated, it is flexible and it is portable. We have taken it on site already.

The remaining tasks on this work in effect are to get a cooperation agreement together. There is a private industrial organization that is interested in taking SAFT and using it for field inspections, so they are willing to join us in the program to help us finish the program, help us to do the field validation type work, and to really be concerned about the kinds of things you are faced with in the field. They have experience in that area and they are quite a help to the program, not only financially but from their knowledge of field inspections.

They will be putting quite a bit of money into the project. They have an interest in utilizing technology. Their position has been that they have evaluated techniques that are available, they want to be able to use the most advanced techniques for competition, and they have decided to go with SAFT.

So we are working with them right now in finalizing an agreement and we will work with them in doing tests together, both in the lab and in the field.

Now that the real time processor is in place, we want to do much laboratory testing on different crack types, crack sizes, crack geometries, pressure vessel steels, cast stainless steel, rod stainless steel, et cetera. We have done this kind of work up to now, but only limited specimens. We now want to do a lot of specimens to essentially prove to ourselves that it does work for all these different conditions, and also to optimize inspection for cast stainless steel, for example, for IGSCC and so on.

We have not done so much up to now because it takes so much time to process the data. With the real time processor, we should be able to move ahead very quickly.

In field validation tests, we will conduct some more tests in the field both on piping and on vessels. We have not done any vessel tests yet in the field, and of course, based on the field results, we will upgrade the system if we need to

do so, and we will be working with the Code in trying to get the technology accepted by Code, probably through a Code case.

[Slide.]

There is our schedule, and I won't spend any time on it. Essentially, by 1987 we hope to have this technology in the field implemented.

MR. SHEWMON: You are going to have to skip a few Vu-graphs.

MR. MUSCARA: Okay.

[Slide.]

Briefly, another project, also with Battelle Northwest Laboratory, which is our NDE/FM program, which is the integration of NDE reliability and fracture mechanics. I will not cover the whole program, but there were some items that you were interested in, so I will just cover those specific topics.

Essentially the objective is to quantify the reliability of in-service inspection for piping and vessels and to try and make improvements in this capability and to get the techniques applied in the field.

We are using round robin testing, especially for piping. We will make use of international work on pressure vessel.

And then finally, based on the material properties, on the inspection reliability for a given component and for

its use within the plant, we hope to come up with advanced in-service inspection criteria that is both how reliably it should be inspected, how often it should be inspected, and what the acceptance for a flaw in a given component should be

[Slide.]

One of the tasks we have been looking at in the last year is to look at piping removed from service. This has IGSCC, or supposedly has IGSCC. Very briefly, and we have not done much validation yet, but it looks like there is a lot false calls going on in the field. There is piping that has been repaired because of flaws that have no flaws in them, as far as we have been able to determine so far.

In the cast stainless steel, we have done a round robin inspection a number of years ago and we have reported on this to you. We have recently completed a first phase of our round robin using international teams. Some 18 teams have looked at a set of samples, 15 cast stainless steel samples with thermal fatigue cracks. These samples were used in our previous round robin two years ago, and the data is just being analyzed now.

Very preliminarily, it doesn't look like there is much difference between the results now versus the results from two years ago, although it looks like some of the European teams can do somewhat better. The problem seems to be there is a large, high false call rate. There are people

calling metallurgical signals cracks, so almost every signal they call a crack, so they do detect a crack but they also get a lot of false calls.

Briefly, if I look at the data right now, it looks like the best people can do in detection is about 50 percent POD, but going along with that, there is also high false call rate. So we need to section these specimens.

MR. SHEWMON: Are you still on IGSCC or cast stainless?

MR. MUSCARA: No, this is the cast stainless.

MR. SHEWMON: Okay.

MR. MUSCARA: Our continuing work on this is to put together a broader set of samples that have more cracks of different types, different heats of cast stainless steel because that's a very major parameter, and we will be using a number of techniques, again round robin inspections under the testing program. So that will be an international program.

We are trying to determine the effectiveness and try to put limits on variations of equipment parameters so that the equipment reliability is not affected. And if we are otuside of these specs, one must requalify the equipment. We are trying to develop what these bounds in the equipment parameters should be.

I already touched upon this development of inspection criteria.

We have conducted a round-robin meeting on stainless steel this summer. And the major objectives here were to see whether it was an improvement from our last round-robin of about three years ago.

At that point there had been no qualification at the NDE center, and now there has been. So we want to see if those teams are performing better now under the qualification program.

We are also trying to see whether a team vs. an individual would give us any difference. Qualifying a team or qualifying an individual evaluator.

We are also trying to see whether large cracks respond different than small cracks.

Again, this work is just completed. As a matter of fact, there's two more teams that have not done their work yet that will be coming in within the next month or so. But on a preliminary look at the data, there doesn't seem to be much improvement over the data results from three years ago.

On the pressure vessel activities, we are mostly following the international work on pressure vessels to see how we can benefit from those results and be able to improve on the code.

Now if work needs to be done that was not covered in international work, we would plan on doing that, but I think even more so than that with our interaction with PISC-III, we

will try to have this work that we might have missed in the past.

MR. SHEWMON: When you back up -- if you back up, you talk about the many round-robin -- no, no, not that much, on stainless steel -- and there wasn't too much improvement, did you mean primarily sizing or detecting?

MR. MUSCARA: Both. And my comment right now relates to the detection. The sizing -- we have no specimens here. But there really seemed to be not much improvement. The longer the cracks, the easier to detect, but there is also a cut-off here, I suspect. Much of our cracks in the previous round-robin were about one inch, maybe two inches at the longest. We now have cracks all the way up to several feet. And we do see an improvement with crack length, but I don't think beyond the two or three inches. I think once it's three inches or so -- and that's speculation at this point, I have not looked at the data in detail -- then there's no longer an improvement.

MR. THOMPSON: There have been some indications at various places that going to low frequencies gives some improvement, although certainly not excellent results. Were those kinds of ideas incorporated in this round robin?

MR. MUSCARA: Low frequencies? No, we're using essentially typical field --

MR. THOMPSON: So there were just field techniques?

I'm sorry, I --

MR. MUSCARA: Well, I did not mention that work, but at Argonne National Laboratory we are looking at lower frequencies. In fact, we are now looking at .2 m.gahertz, .2 to .25, and it does seem to be an improvement.

Of course, it's not as sensitive to the crack size.

MR. SHEWMON: There is an industry-sponsored program on that.

MR. MUSCARA: On casting of steel.

MR. SHEWMON: Yes. And detection. And some guy, when whoever came in for their meeting last time -- Vogtle -- yeah, they had a system that was working beautifully, they used lower frequencies in some particular shape, and I went away fat and happy and stupid, apparently. Obviously we weren't asking the right questions, or I wasn't. Are there any sort of standards being evolved or --

MR. MUSCARA: We have used some of our samples for our field people and for NRR to check against the industry's equipment. In other words, they have taken our standards out to the field and said okay, detect the cracks in here, and very often they cannot do it.

The problem with the cast stainless steel is that its results are highly dependent on the microstructures, and the microstructure is highly variable. And the microstructure is much easier to inspect.

So if you run your test on an optimal material, you have a crack that's not so tight, thermal fatigue cracks, and the crack is long and deep, you can probably do better at detecting. And there's some results that indicate that they detect. But it's based usually on a material that's easier -- not that it's easy, but easier, and the cracks are wide open and they're quite long.

In the international work we are planning on incorporating all the different kinds of crack sizes and depths and lengths and different materials. But we have found out if you go down in the frequency for the longer grains, especially, you can get a more reliable detection. But, of course, the wave length is so long that you are not sensitive to small cracks.

Well, maybe I should skip some of these viewgraphs. But we have also been concentrating this last year on inspection weld overlay. And the first phase of the work essentially has been a literature survey, evaluation of the EPRI position in this area, and our own work on looking at wave distortion going through the weld overlay.

We already have developed an interim report we sent to the Staff a week or two ago and I think I have sent the ACRS a copy, and we plan on having a final report in December of '86.

[Slide.]

Some conclusions from this work:

[Slide.]

Again, based on the literature search on the distortion experiments performed at Battelle and on the review from the EPRI work, the conventional shear wave examination is not effective for the weld overlay.

Detection and sizing of flaws more than 50 percent is also not reliable.

Longitudinal probes with angles between 40 and 70 degrees work the best, so far, and using frequencies between 2 and 4 megahertz.

Surface preparation of the weld overlay is very important, and actually preparation is required to get the best inspections possible at this date.

The conditions for the surface are 250 microinches on a local basis, and .06 inches from peak to valley within a 1" x 1" area. So in most cases we need to have some kind of surface preparation of weld overlay.

But even with that, the limitations I mentioned before still apply. Small cracks cannot be detected.

MR. SHEWMON: I don't convert very gracefully out of microinches. But that's a quarter of a thousandth of an inch; is that right? Is that what 250 microinches is? That's --

MR. MUSCARA: 250 millionths.

MR. SHEWMON: Pardon?

MR. MUSCARA: 250 millionths. Millionths.

MR. SHEWMON: So that means -- I don't -- that's a not very --

MR. MUSCARA: 2 and --

MR. SHEWMON: That means it is fairly finely ground. Okay. Next time I see you, I'll ask you to translate that into 120 grid or something.

[Laughter.]

MR. WARD: That's about what you get from not too good turning, isn't it?

MR. SHEWMON: That's not a polished surface?

MR. MUSCARA: No. But what's important is that the waviness, it's almost more important than the other parameter. You really cannot have very much waviness in the surface.

[Slide.]

Well, the recommendations I will skip, because they are based on the conclusions, anyway, and they are in your handout.

We have been doing work with qualifications, and a number of studies indicate we need to improve our inspection capabilities. And we believe we can do that through qualifying essentially the inspectors, the procedures and the equipment as they operate together.

MR. SHEWMON: Who will do that? Do you think the

NDE center will do it or --

MR. MUSCARA: Well, our efforts -- the most recent efforts is to work with the ASME code and there has been a group that's been set up under the ASME to develop qualification criteria based on the documents we already have developed.

What the code has done is to develop three subgroups to look at different aspects of qualification. One of those aspects is the implementation.

Unfortunately, they haven't come up with any answers yet. But one of the options is the NDE center. The way the criteria has developed, though, anyone who wishes to perform qualificat'on can. All the requirements are listed in the criteria. The requirements for samples, for the laboratory, for the instructions, for the instructors, and so on. So it is a very detailed piece of work and anyone that wishes to set up a qualification laboratory can take this criteria and set it up.

MR. SHEWMON: So the initiation fee for setting up such a lab is a good deal higher than it is for requiring somebody, the way they certify technicians now at grades 1, 2 and 3 classes or something, is that --

MR. MUSCARA: Yes, because part of what our requirements include are performance demonstrations. It is they must perform on the actual type flaws and components.

MR. SHEWMON: Go ahead.

MR. MUSCARA: Well, I don't need to cover the instance where they tell us we should do qualifications. There have been a number of studies. The recent activities on the qualification -- well, the elements, the assumptions, the developments criteria.

We do want blind test demonstrations. These are the performance demonstrations. We are addressing personnel, procedures and equipment as what we're calling the NDE system qualification, not just the personnel qualification.

We do want to address unique applications, for example, casting the steel vs. IGSCC vs. pipe cracking. And the criteria for all of these is written up.

We do require annual training courses for a person who already has a level 1, 2 or 3 to update himself on new occurrences of the past year, new developments or new flaw type mechanisms there.

We have strengthened the written exam requirements so that they cover more specifically UT. The pass/fail criteria is statistically based so that we can determine what performance we need, and then based a simple set on the performance required.

We are developing tolerances on the equipment, so that if the equipment is validated, it cannot go outside of the required specifications. If it does, parts need to be

changed or qualifications need to take place.

The documents we are developing apply to all the UT ISI. That is, any components required to be inspected by code should be inspected by a qualified NDE system. Most of the activities at this time, of course, only relate to the IGSCC. And again, all the criteria is provided so that anybody could set up a qualification laboratory.

[Slide.]

A little bit on the status.

We have been working on this for several years. We had a workshop early in June of '83. A second draft was developed and was reviewed with the NRC Staff in '84, in October of '84. The Staff essentially had accepted the idea of having qualification requirements in the details of the document.

Based on that, we had a workshop with the NRC to present our proposed qualification criteria document. The result of this was a recommendation that the ASME code put together a task group to review and evaluate needs for qualification and start using the NRC-developed documents for developing such criteria.

Now three subgroups were developed. They have been working very hard since November, and just this last August they developed documents, three different documents, to cover different aspects of qualification and have started them

through the process of approvals through the code.

Right now the proposed demonstration part of this document only deals with the IGSCC, but it's stated that it will cover all the different possibilities; cast stainless steel, cracking, nozzle inspection, vessel inspection and so on. They have been prioritized so that the IGSCC was on top and ferritic piping, for example, was at the bottom. But they will all be developed.

And I guess I would also like to mention that the industry has worked very hard on this document, and many people have been coming to meetings. We have had meetings once a month to get a document in place in a hurry. So in about four to six months we have documents out for review and evaluation.

That's that. Now I have about 300 slides on the Surry generator work. Some of these you have seen,so if you don't mind, I might skip many of those.

MR. SHEWMON: Yes. You have my permission to skip any of those.

MR. MUSCARA: Okay.

[Slide.]

Thank you.

Very quickly, the kinds of inspections that we have done on this generator, we had at the beginning two baseline examinations. Once the generator was removed from service, it

was placed in the facility at Battelle. We invited two teams to do a complete baseline.

Now these are teams that performed most of the inspections in the U.S. and internationally. So the baseline essentially was detecting using a MIZ 12 system, and the other baseline was conducted Intercontrole, using equipment that's used widely in Europe.

Well, let me mention right now, the other types of testing we have done, up to the baseline we chose a number of specimens of 325 specimens we used in round-robins.

What we wanted to do here was to evaluate a number of teams, and so they have done what we have called a data acquisition and analysis round-robin. So each team came in with their equipment, obtained the data and did the analysis of the data.

The next round-robin was a data analysis only round robin. We took one set of data and five, six, seven teams, evaluated the data, see what kind of results they would come up with.

A following type of round robin was an improved technique round-robin, where we invited people with advanced systems to come in and take a look at either all these 320 systems or a subset of those. And the results were quite interesting. There's a lot of variability in the detection by factors of two or three. Some people are reporting only 200

flaws, others may be reporting 600 flaws.

Interestingly enough, there's more variability in the data acquisition and -- sorry, in the analysis round-robin than there is in the data acquisition analysis.

What we are finding out really is that the signals are there, people are detecting signals, but they are confused on how to evaluate them. Many teams are calling signals artifacts, not flaws, so they don't properly identify them.

So there is more variation on the same set of data by different teams looking at the data than there is in the acquisition and analysis.

Now I'll skip the summary of the inspections. It is in the handout.

[Slide.]

Well, let's look at this briefly.

In the baseline, one team sized defects on the average larger than the other team, about 5 to 7 percent larger. They also called many more flaws, many more small flaws below 20 percent than the other team.

There were a few instances where teams did not have signals in common, and there were some instances where even though they were signals they were not called consistently, where some teams did not have signals.

But in general, most of the signals were there. It's a matter of interpretation.

So one of the things we are going to concentrate on is to try and determine and evaluate what the best procedure are for evaluating the data so that we can optimize and improve on the procedures for evaluating.

I really think I will skip the graph on the selection of the round-robin teams.

In fact, what we wanted to do was to get a range of flaw sizes and types within this generator, so that we could evaluate the techniques in different kinds of flaws. And we looked at the data from the baseline, so we had samples where the teams had calls in common, we had samples where the teams disagreed on the calls, and we had samples where one team called a signal and one team did not find a signal.

We then also used data such as the secondary site inspection to include tubes that had, for example, bulging. We found quite a few number of tubes in a generator that had bulged through service. And also areas where we had loose parts, we took tubes next to the loose parts. And so that made up a round-robin of about 320 specimens.

[Slide.]

Well, I already covered this. This is the different types of round-robins we have performed. Very quickly, this is data acquisition and analysis. These are the different teams and it's all the number of indications that they reported. This is also the number of indications reported by

-- some teams had multiple indications in the three inches above the tube sheet. And in that case, we only called the one flaw there. But if people had multiple calls, we still gave them one. Just to see how the data would compare. But you can see that teams ranged from 202 to about 300 flaws.

MR. WARD: Well, wait a minute, though. How many flaws were there? I mean --

MR. MUSCARA: We don't know. We had 320 samples. It's a blind test. So what we've done is to leave the samples in the generator. Based on the inspections we've done previously, show us a set of samples for the round-robin, perform inspections, now take out these samples and cut them up and verify what's in the samples.

So we really don't know what the baseline is. We don't know who's right and who's wrong. All it shows is that there's a great discrepancy amongst the teams themselves. They all looked at the same tubes.

MR. WARD: Okay. So you'll know --

MR. MUSCARA: Just beginning to line out the tubes from the generator.

[Slide.]

Well, in your handout we have the same data broken down by size category, how many did they detect in the 40 to 70 percent, et cetera, and there's quite a bit of discrepancy there also.

Just to get an idea who was involved in the data analysis round-robin these were the teams. We tried to get teams that do most of the inspections in the states. So we had J.A. Jones, he actually used two different techniques. Rochester Gas and Electric; Combustion Engineering; Babcock; Westinghouse; Zetec; Conam; and UTL.

Now UTL uses German technology, KW technology.

[Slide.]

Let me show you again for the analysis round-robin the kind of range of reported flaws. 600 down to about 220.

One interesting aspect, the 600 and the 200, those are two teams from the same organization. So almost the broadest range obtained within the same company.

MR. SHEWMON: Is a defect an indication or something which is more than 25 percent above some threshold other than detectability?

MR. MUSCARA: We had asked them to report any defect that they detected. Generally you cannot detect flaws over the 20 percent threshold. But we did have a lot of teams, like the French team, they put a lot of flaws below 20 percent. We don't know if they're correct or not. Most of the American teams try to concentrate on flaws other than 20 percent.

The rules we gave them was the ASME code procedures, but do the best that you can, and we told them what kind of

things to report. For example, report denting or not report denting.

[Slide.]

On the advanced technique round-robin, we essentially looked at the NRC-developed equipment at Oak Ridge, and we have covered that in the past. Unfortunately, the equipment we used from Oak Ridge on the generator had been trained on a set of conditions in the laboratory that did not span the kinds of conditions in the generator. So that we have the data, but we need to do some more training and calibration work before we can process the data. We can process the data, but it will give us a less accurate result. So we need to do a little bit more lab work on that.

But we have had teams from J.A.Jones using the state of the art techniques, and that is the Zetec MIZ 12 with a double mix in the frequency to eliminate the conductive deposits, support plates, the dents, and the Zetec MIZ 18 is the newest equipment that's available from Zetec, and it's a digital piece of equipment which has a bit more sensitivity than the MIZ 12.

Mitsubishi Heavy Industries from Japan came in and they did more or less spot our inspections. They looked at our U-bend region and they looked at the tubes and the tube sheet and some distance above the tube sheet, looking for essentially IGA, and IGSCC.

KWU came in with their multi-frequency probe. This work is not conducted yet. We expect before this year is over the French will send in their latest equipment. It's multi-frequency, it's a rotating probe, and it's supposed to do some imaging of the flaws.

These last two pieces of equipment essentially do imaging. What we are trying to do is to characterize the density.

Well, preliminary results from the advanced technique.

The Japanese found some indications in the U-bend that were not reported by anybody else. Stress corrosion cracks. No other team reported any of these. They may be correct, they may not be. The Japanese have put a lot of emphasis on stress corrosion cracks because that's one of the problems they've had in Japan.

They also reported IGA.

[Slide.]

That is, the intergranular attack at the tube sheet area. None of the American teams reported this IG, except for one team, but this one team's supervisor later said we would never report that, we don't believe it's IGA. So I don't know whether to classify that. The representative said he thought it was IGA, the supervisor said no, it's not, if you were out in the field, you would never report that.

The KWU used ultrasonic testing to help them upgrade their eddy current sizing. So once they found some flaws, they went in with an ultrasonic, and then applied a correction factor to their eddy current data.

The only team, I believe, that detected pitting was them, and it seems that there's extensive pitting in the generator tube sheet.

Well, the rest of the work essentially, most of the new work is done, the rest of it will concentrate on validation. And tube sheet sections will be removed for evaluation of both the surface on the carbon steel and the tube sheet and the tube itself, evaluating the corrosion products and evaluating the sludge above the tubes.

[Slide.]

We have removed already some tube support plates just to see what would happen and, in fact, there's so much damage in this generator, that we had to clamp the support plate before it was cut out, and so we removed the section of the support plate with the clamp. Once we took the clamp off, the plate crumbled with the tubes. There was no tube that stayed within its support hole out of sections of tubes of about a dozen tubes. So there's extensive damage in the support plate and in the tubes, and it's amazing how the thing held together. I think probably the corrosion product held it together.

We plan on removing the tubes from the generator that were in the round-robin set of samples. We found totally about 850 indications reported by various and different teams. Any one of these sections of tubes will be removed for evaluation to use many of those for destructive examination or try to do advanced laboratory type NDE, so we're going to have to section them off.

MR. SHEWMON: If you have a conclusion slide, fine. Otherwise, I'd like to get a couple of other questions, and why don't we read this.

MR. MUSCARA: Okay.

[Slide.]

Let me just -- it won't take much time -- show you some of the tubes have been removed. Essentially the tube sheet ends here. We are finding IGA just about at tube sheet and you can see extensive pitting a few inches above the tube sheet.

Now these kinds of problems have not been reported by any inspection teams. We have gone back and looked at the data from the actual in-service inspections and there were signals there, but not evaluated or reported.

[Slide.]

Very briefly again, let me show you a section of tube sheet that has been removed. We removed the section essentially by overdrilling the surrounding holes around the

tubes we wanted to remove.

Let me show you that. Essentially the tube sheet ends here. You can see all the sludge that's in between the tubes. You see a couple bends of copper. We believe that the pitting relates to this copper deposition. You can see also the demarcation of the sludge pile that was produced by the phosphate water chemistry. It's quite dense, tenacious, and here is the AVT type sludge, and there's a lot of it in there.

[Slide.]

Okay, summary of current status, I guess.

Well, I think I don't need to go through this.

Future activities I also mentioned. I was mostly concentrating on the evaluation.

MR. SHEWMON: Are there questions on this?

There's been some talk about whether or not cleaning -- years ago there was talk about hoping to be able to try some cleaning out on this and I didn't hear anything about that today.

There's also been talk in the industry about whether or not cleaning damages and opens up more cracks, and I would be interested in your comments on that.

MR. MUSCARA: Yes. I had a viewgraph, as a matter of fact.

With respect to the cleaning, we did some work at the beginning to clean the water box, and they evaluated a

couple of techniques which have beyond that time been used in industrial applications in the States. We were the first. Nobody wanted to try it, but based on that work, they did try some of that.

We have so much validation and additional work to do that's quite expensive in order to evaluate all these flaws, and we decided not to pursue any chemical cleaning or decontamination work. We felt it was more important to do the characterization work than that kind of work. And many of the people, frankly, in the NRC were not so anxious to do the chemical cleaning, decontamination, and so our international partners decided it would be okay for us to concentrate on the NDE part of the work.

Now you are correct in your assessment that the chemical cleaning brings up flaws that are not noticed during inspection.

As a matter of fact, we also ran into that. One of these groups that I showed you, we did not detect the IGA until we did essentially brush the tube to get rid of the corrosion product deposited on it. And I believe before this tube, we had another tube we had removed from service, some ultrasonic cleaning improved the signals.

Now I don't recall doing any chemical cleaning to clean the tube, but some of the mechanical cleaning techniques have shown flaws after cleaning and not before cleaning.

MR. SHEWMON: Well, my source was Nucleonics Week or something, and the question there was whether or not it had cleaned it up so you could see things that were there before that you couldn't see before, or whether the cleaning procedure had actually generated some flaws of its own.

MR. MUSCARA: No, I don't think the cleaning procedure -- even when we mechanically clean, we find flaws we did not find before the mechanical cleaning. The flaws are very difficult to detect. And what happens with this crud that's on the tube, it complicates the signal. What we found after cleaning, the signal was nice and clean, the signal itself was of a nice typical shape that could be evaluated. With the corrosion products and with the copper deposit, the signals are difficult to evaluate.

So if people are detecting signals, they are so "weird" that they don't know what to do with them, they don't know how to evaluate them.

Of course, there was work at I believe it was Millstone where they had done inspections before chemical cleaning and after chemical cleaning and had many, many more indications after the chemical cleaning.

MR. SHEWMON: That's what I saw, yes.

MR. MUSCARA: And I don't believe that was due to the chemical cleaning process. In our own work, we evaluated the effect of chemical cleaning on corrosion of the samples.

We saw no indication at all that it changed the nature of the samples. There was no corrosion developed, no cracking. We had samples that were sensitized and not sensitized, stress, unstressed, a great number of samples that were immersed in the solution and it did not affect in any way the samples.

The only thing we saw was on some carbon steel that the samples blackened some, a black oxide formed.

MR. DILLON: I have already indicated the importance I attach to this sectioning of the tube sheet and the tubes in it. But the point I wanted to make was to the degree that we analyze the signals that you get from some sort of an NDE process in the more conventional kinds of flaws in the steam generator, I think that's tenfold more important in this very complex region in the tube sheet, and I would guess that Joe is going to have a lot better basis for interpreting some of his NDE after he has done this corresponding metallography and mechanical examination of the section he's taken out of the tube sheet region.

MR. MUSCARA: I did not concentrate on any of the work, either the NDE or the generator, because this was supposedly only an update. But we are planning other work. The most important is the NDE work in the generator. You will characterize tube sheet sections both with respect to metallographic and chemical and evaluate the change in composition of the sludge as we go on and look for any

corrosion products in the crevice and evaluate for any cracks in the crevice.

MR. SHEWMON: Okay. Any other questions? I think we've had enough.

Thank you very much, Joe.

Tomorrow at 8:30.

[Whereupon, at 5:15 p.m., the subcommittee meeting was adjourned, to reconvene at 8:30 a.m., Friday, September 5, 1985.]

CERTIFICATE OF OFFICIAL REPORTER

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of: ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Name of Proceeding: ACRS Metal Components Subcommittee Meeting

Docket No.:

Place: Washington, D. C.

Date: Wednesday, September 4, 1985

were held as herein appears and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission.

(Signature) Marilynn M. Nations

(Typed Name of Reporter) Marilynn Nations

Ann Riley & Associates, Ltd.

(2) (2) (1)

STEAM GENERATOR SHELL CRACKING

1978 GARIGLIANO, ITALY

PRIMARY SIDE OF SG IN DUAL CYCLE BWR
LEAK, AND EXTENSIVE CRACKING

1982 INDIAN POINT 3

LEAK IN ONE SG
EXTENSIVE CRACKING IN OTHER 3
REQUIRED WELD REPAIR

1985 SURRY 2

NO LEAKS
EXTENSTIVE CRACKING IN ALL 3 SGs
CRACKS COULD BE GROUND OUT

2)

FACTS SURROUNDING

GARIGLIANO STEAM GENERATOR

FAILURE

K. E. STAHLKOPF

DECEMBER 4, 1978

GARIGLIANO REACTOR

TYPE

- 150 MWe BWR
- DESIGNED BY G.E.
- TWO LOOP DUAL CYCLE
- OWNED BY ENEL, ITALY

HISTORY

- POWER OPERATION BEGAN IN 1964
- OPERATED IN EARLY YEARS AS LOAD FOLLOWING
- SWITCHED TO BASE LOADING IN 1968
- AUGUST 8, 1978 LOUD WHISTLING NOISE WAS HEARD WHILE PLANT WAS AT POWER
- PLANT SHUT DOWN AND VISUAL INSPECTION SHOWED A THROUGH WALL LEAK IN LOWER STEAM GENERATOR HEAD WELD

KES/RMT
12/78

DESCRIPTION OF FAILURE

- AUGUST 8, 1978 LOUD WHISTLING NOISE HEARD AND PLANT WAS SHUT DOWN

- VISUAL INSPECTION SHOWED A LEAKING THROUGH WALL CRACK IN LOWER STEAM GENERATOR HEAD WELD

- COMPLETE UT OF LEAKING WELD PERFORMED BY GE, SAN JOSE

- IN ADDITION TO LEAKING CRACK, SEVEN ADDITIONAL CRACKS WERE FOUND WHICH WERE WITHIN A FEW MILLIMETER OF GOING THROUGH WALLS. ALL EIGHT CRACKS WERE TRANSVERSE TO THE WELD

- ALL WELDS IN BOTH STEAM GENERATORS WERE RADIOGRAPHED. ONE ADDITIONAL TRANSVERSE CRACK WAS FOUND IN ORIGINAL FAULTY WELD SEAM AS WELL AS A CIRCUMFERENTIAL CRACK WHICH WAS OVER HALF THE CIRCUMFERENCE OF GENERATOR IN LENGTH AND HALF WALL IN DEPTH

- THE CIRCUMFERENTIAL CRACK WAS IN THE BASE METAL OUTSIDE THE WELD AND HAZ

- ALL CRACKS WERE CONFIRMED ON ID OF GENERATOR BY DYE PENETRANT

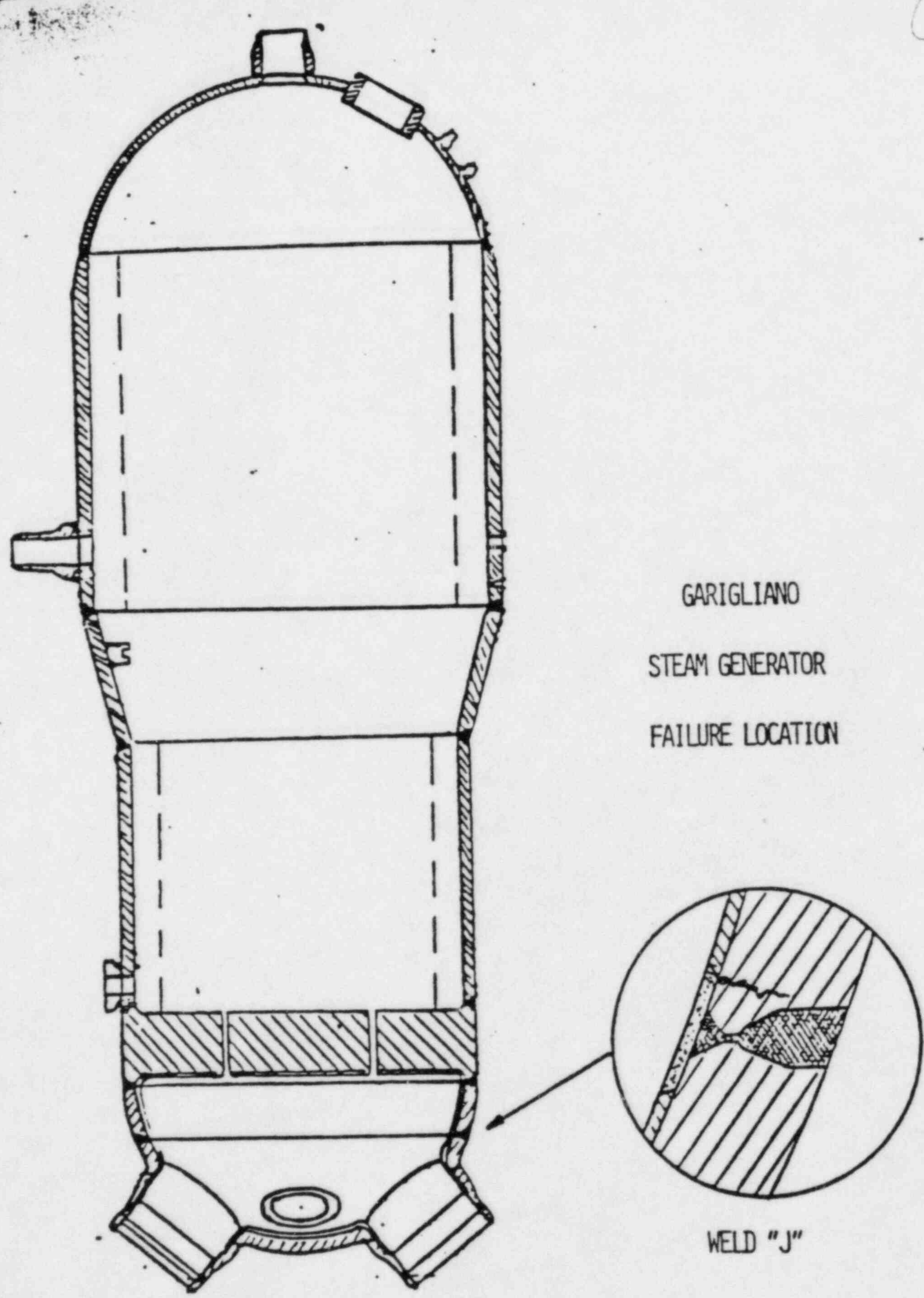
OVERSEAS SSG FAILURE

- BWR-1 WITH TWO STEAM GENERATORS (IN OPERATION SINCE MID-1964)

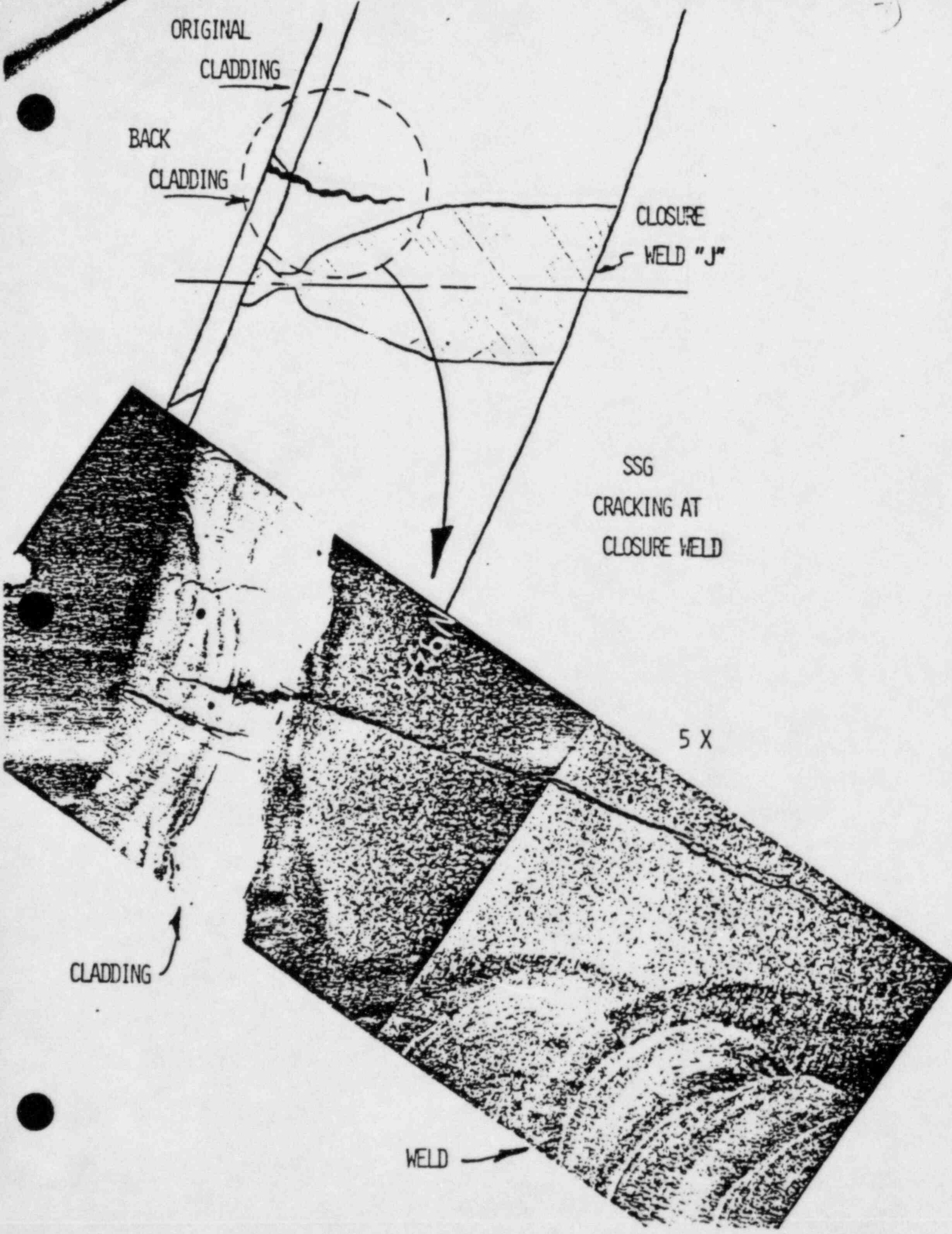
- STEAM GENERATORS -
  - MANUFACTURED BY STORK (HOLLAND)
  - ASTM-A 302B (MN - MO ALLOY STEEL)
  - WATER BOX CLAD WITH MONEL 140
  - WELD FILLER 1/2 CR - 1/2 MO WITH PREHEAT OF 400°F
  - CLOSURE WELD LOCAL PWHT BY INDUCTION HEATING

- EXTENSIVE CIRCUMFERENTIAL AND TRANSVERSE CRACKING
  - CRACK INITIATION ON INNER SURFACE OF VESSEL
  - MONEL CRACKING: INTERCOLUMNAR AND INTERGRANULAR (SCC)
  - BASE METAL CRACKS - TRANSGRANULAR AND RANDOMLY BRANCHED - ENVIRONMENTALLY ASSISTED CORROSION MECHANISM
  - FATIGUE CRACK GROWTH - <u>NOT</u> PRIMARY CAUSE OF OBSERVED CRACKS
  - CALCULATED STRESS INTENSITY FACTOR FOR CRACK RUST PENETRATING CLAD IS 40 - 45 KSI IN

(6)



A Cross-Section of Garigliano Secondary Steam Generator with Seam J Shown in Inset



SSG
CRACKING AT
CLOSURE WELD

5 X

GARIGLIANO - SSG - STRESSES

CLOSURE SEAM WELD FAILURE

RESIDUAL STRESSES DUE TO FABRICATION

MEASURED (O.D. SURFACE)	40 KSI COMPRESSIVE
CALCULATED (O.D. SURFACE)	APPROXIMATELY 35 KSI COMPRESSIVE
CALCULATED (I.D. SURFACE)	45 KSI TENSILE

OPERATIONAL STRESSES

NORMAL PRESSURE STRESSES	APPROXIMATELY 7 KSI
MOST SEVERE OPERATING THERMAL TRANSIENT STRESS* ($\Delta T = 148^{\circ}$ F IN 7 MINUTES)	29 KSI PEAK COMPRESSIVE TRANSIENT
NORMAL THERMAL TRANSIENT STRESS ($\Delta T = 20^{\circ}$ F DUE TO SECONDARY STEAM INTERRUPTION)	4 KSI TRANSIENT
PIPING LOAD	500 PSI

*A REACTOR SCRAM WITH LOW STEAM DRUM WATER LEVEL (ONE TIME OCCURRENCE)

<u>OBSERVATION</u>

LOCAL POST WELD HEAT TREATMENT
OF WELD ADJACENT TO LARGE HEAT
SINK AND CONDUCTED WITH UNKNOWN
CONTROL APPARENTLY RESULTED IN
HIGH POTENTIAL STRESSES.

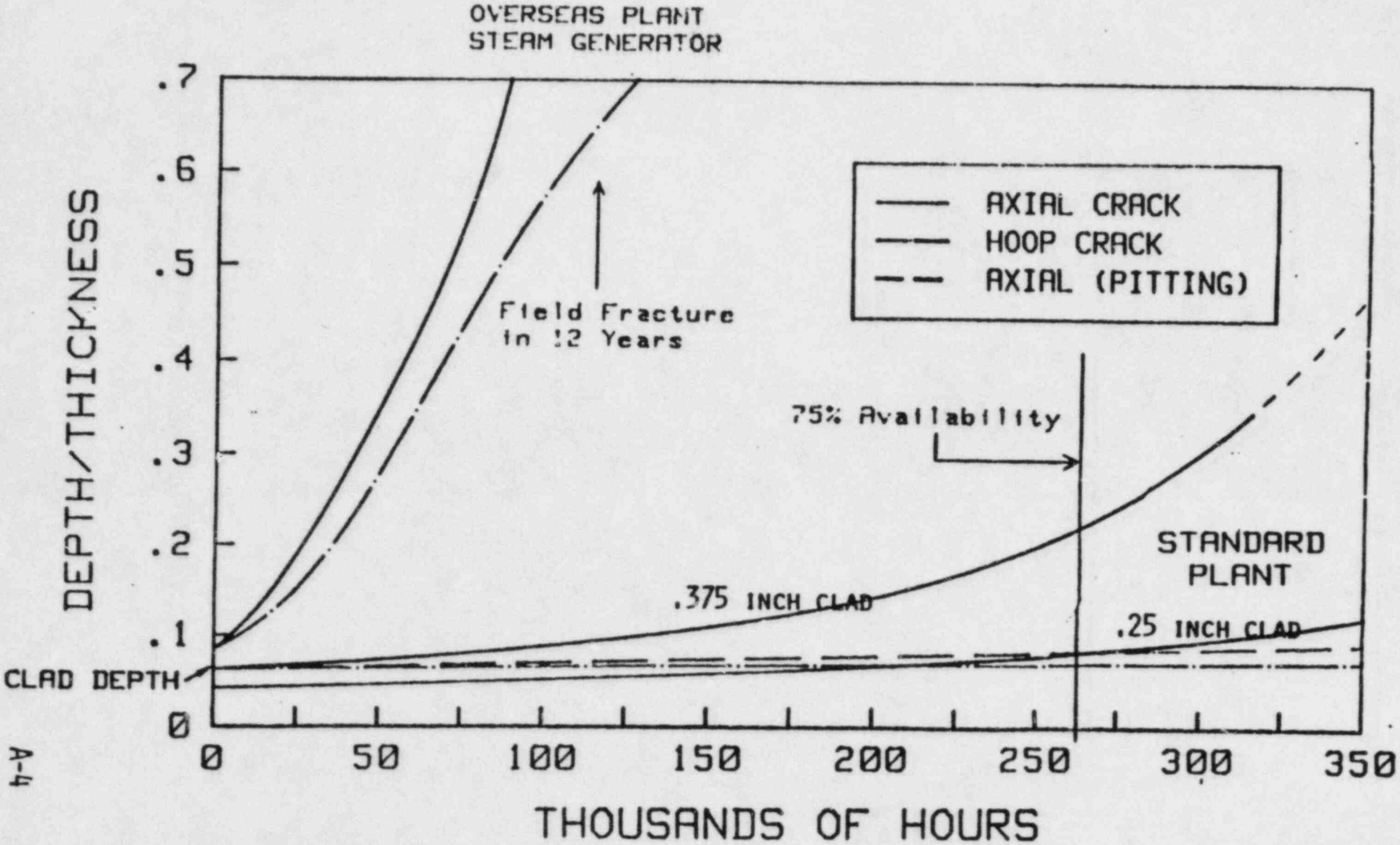
OBSERVATIONS

CRACK GROWTH

1. K_{Iscc} IS APPROXIMATELY 20-30 KSI $\sqrt{IN}$ IN 8PPM OXYGENATED 550° F WATER.

2. K_{Iscc} IS APPROXIMATELY 35 TO 45 KSI $\sqrt{IN}$ IN 0.2PPM O_2 OXYGENATED 550°F WATER.

3. K_I FOR GARIGLIANO FAILURE FOR A THROUGH-CLAD CRACK AT A LOCATION OF HIGH RESIDUAL STRESSES = 40-45 KSI $\sqrt{IN}$.



CRACK EXTENSION PREDICTIONS

OVERALL CONCLUSIONS

- Overseas steam generator material not unique.
- Combination of high weld residual stress and Monel cladding make fabrication process unique and contributed to cracking.
- Thru wall cracking in observed time to crack can be predicted.
- Cracking of normally post weld heat treated component is unlikely - very high stress threshold.
- If cracks should occur, growth is predicted to be very slow.

(13)

INDIAN POINT 3

LEAK DISCOVERED MARCH 1982

BOAT SAMPLE REMOVED

ULTRASONICS AND MAGNETIC PARTICLE INSPECTION

- OVER 100 CRACKS IN EACH SG

6 INCH PLUG REMOVED

- LUCIUS PITKIN, PASNY
- BNL, NRC

CRACKS WERE GROUND OUT, REPAIR WELDED, PWHT

RECENT INSPECTION FOUND NO SURFACE CRACKS

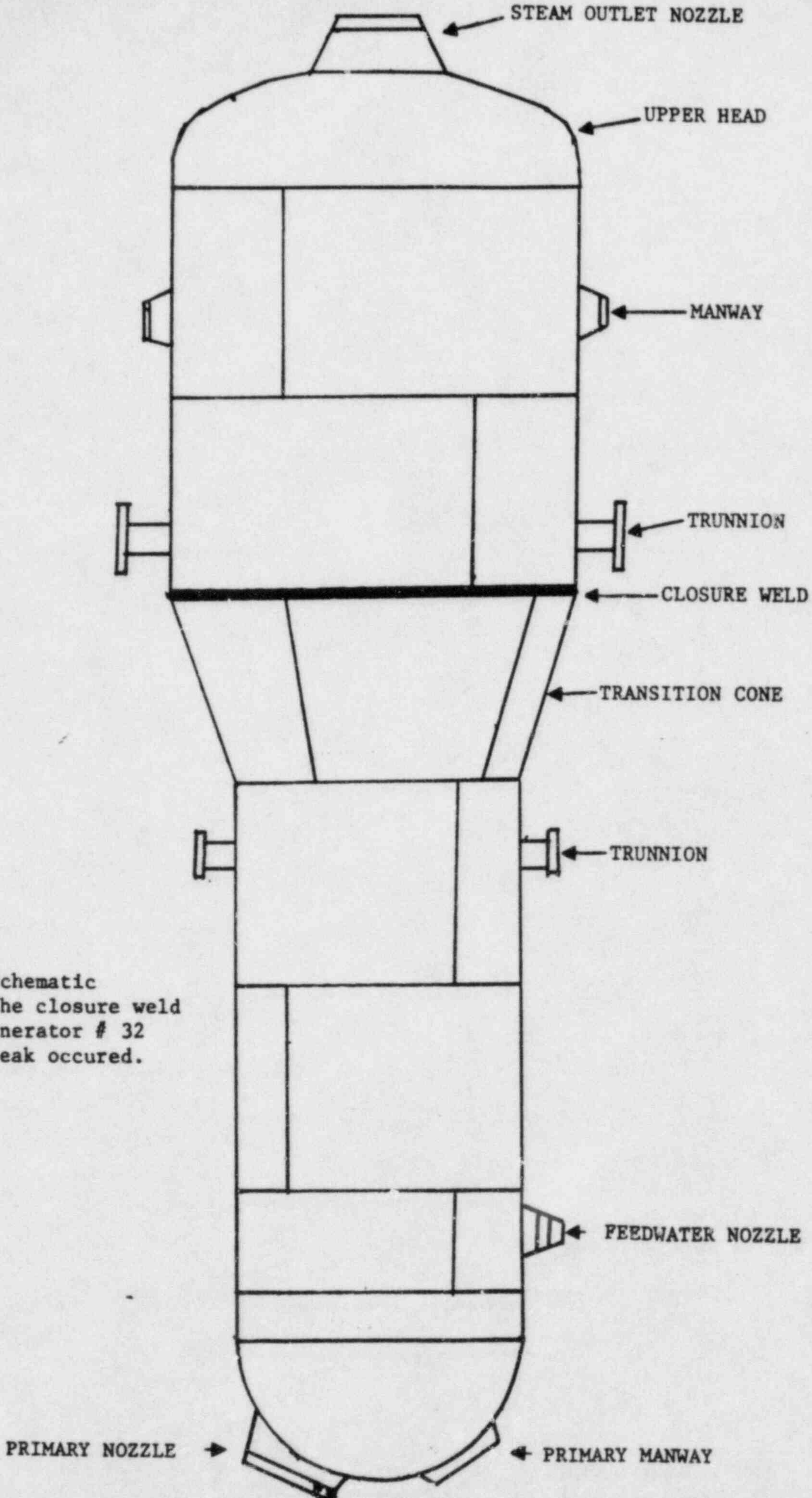


Figure 1. Schematic depicting the closure weld of Steam Generator # 32 where the leak occured.

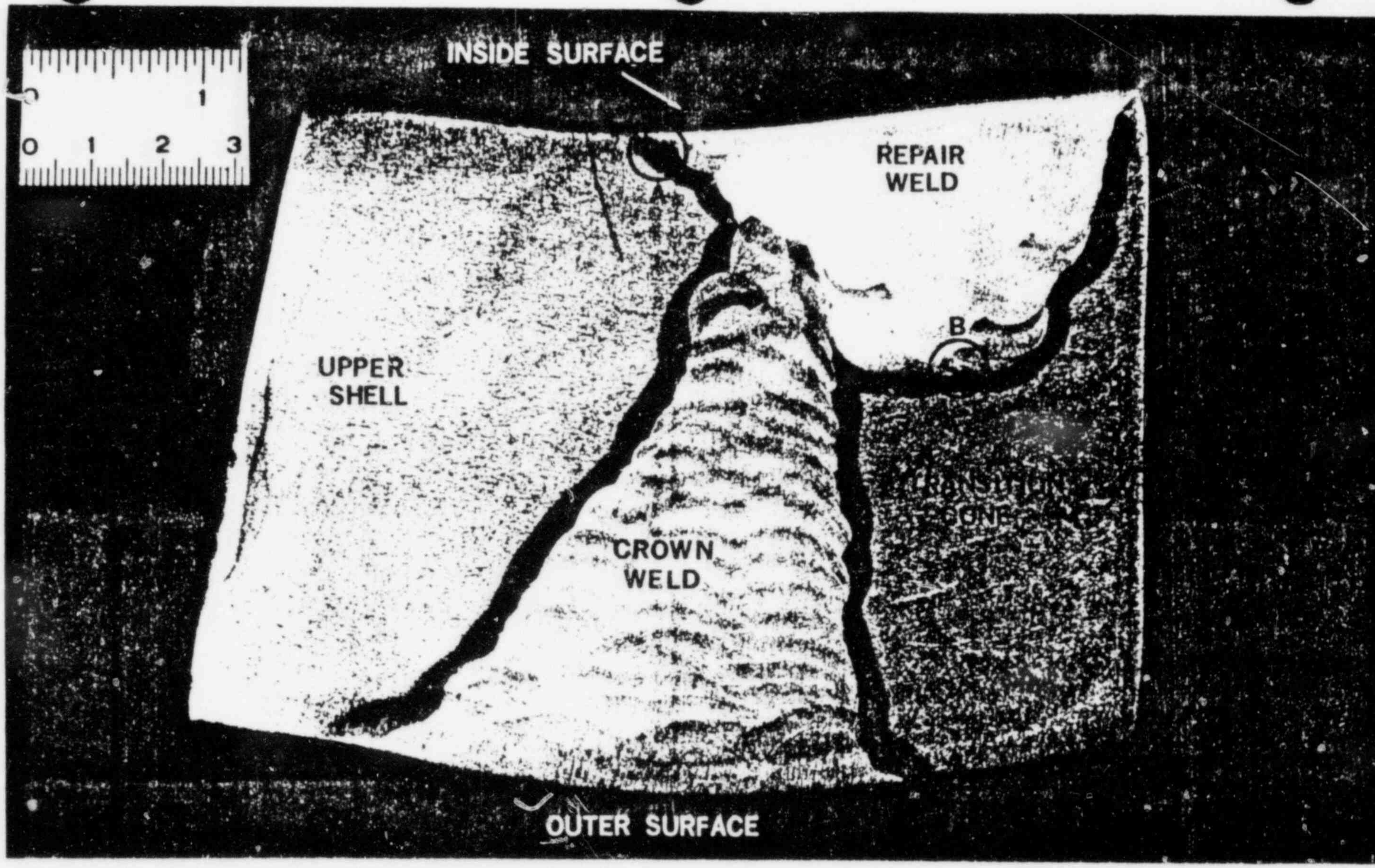


Figure 4. Photograph of a cross-section of the 6" plug after etching. Evident in the photograph is the relative position of the crack to the weld. Additional areas of interest include Area A- a region of lack of fusion and Area B- a region of porosity.

Figure 5. Photograph of the 6" plug cross-section offering a definitive outline of the leak's path.

POSSIBLE CAUSES

- o HYDROGEN CRACKING
- o "POP IN" FROM THERMAL SHOCK
- o FATIGUE-DESIGN TRANSIENTS
- o FATIGUE - THERMAL

W RECOMMENDATIONS

- A LETTER HAS BEEN SENT TO THE OWNERS OF 15 UNITS NEXT SCHEDULED FOR INSPECTION OUTAGES. THIS LETTER RECOMMENDS THAT AN INSPECTION BE PERFORMED ON 100% OF THE UPPER SHELL/CONE WELD ON AT LEAST ONE STEAM GENERATOR.

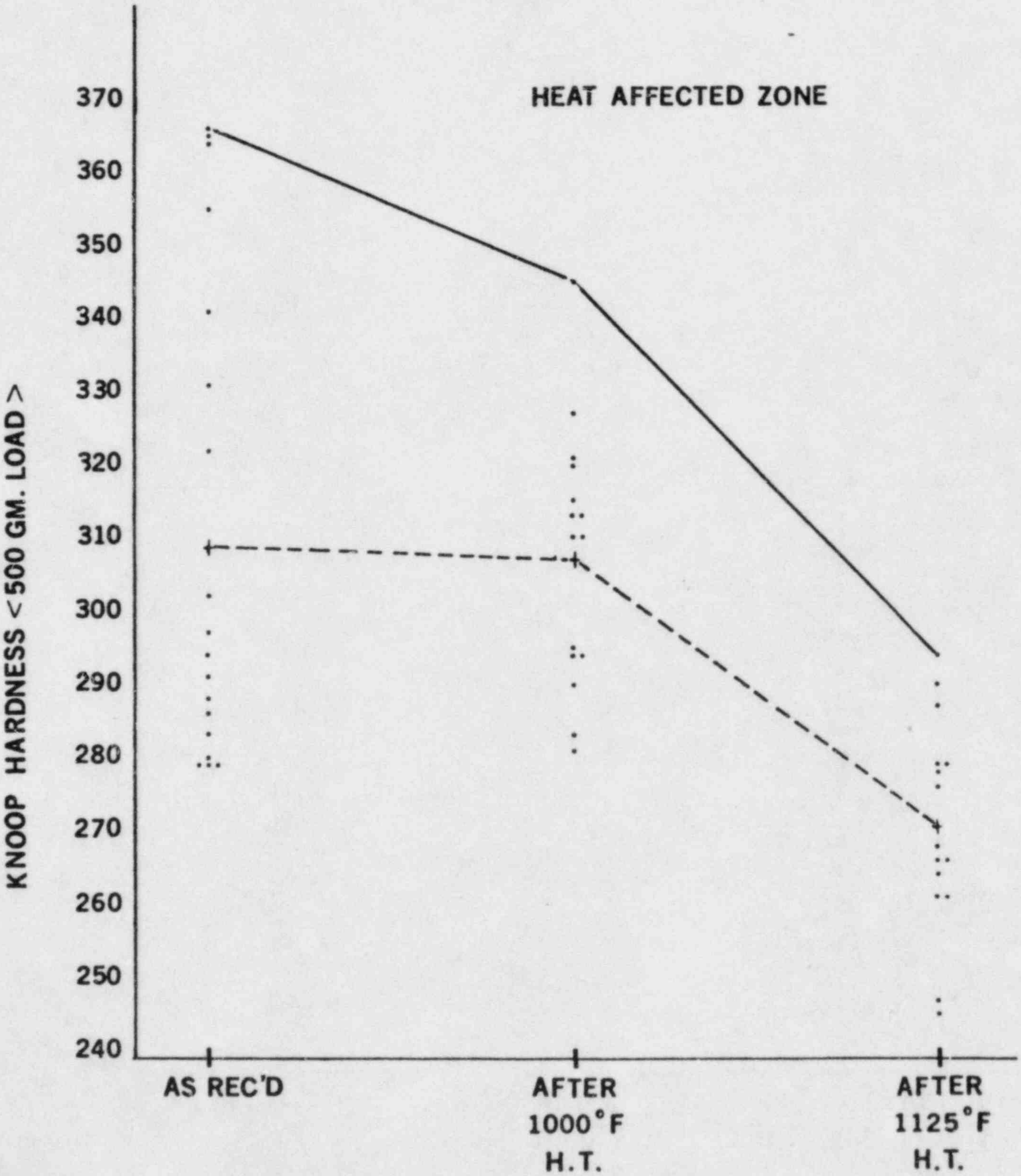
- IF THE RESULTS OF THESE INSPECTIONS DO NOT VERIFY THAT OTHER UNITS ARE NOT AFFECTED, A GENERIC PROGRAM INVOLVING INSTRUMENTATION, ADDITIONAL SAMPLING, LABORATORY TESTS, ETC. WILL BE PROPOSED.

CONCLUSIONS

- THERE IS INSUFFICIENT INFORMATION TO RIGOROUSLY EXPLAIN THE ORIGIN OF THE CRACKS
- FABRICATION APPEARS TO PLAY ONLY A MINOR ROLE
- BEST ESTIMATE OF CAUSE IS CORROSION FATIGUE RESULTING FROM ABNORMAL THERMAL CYCLES AND ENVIRONMENTS

GRAPH #3

A Graphical Comparison of the Hardness Values of the HEAT AFFECTED ZONE After Heat Treatment



— = Maximum Hardness Values

--- = Average Hardness Values

(20)

PASNY COMMENTS ON CAUSE

1. DESIGN

2. ORIGINAL WELD CRACKS

3. RESIDUAL WELDING STRESSES

4. HEAVY SUB-ARC WELDS

5. INADEQUATE PREHEAT AND INTERPASS TEMPERATURE

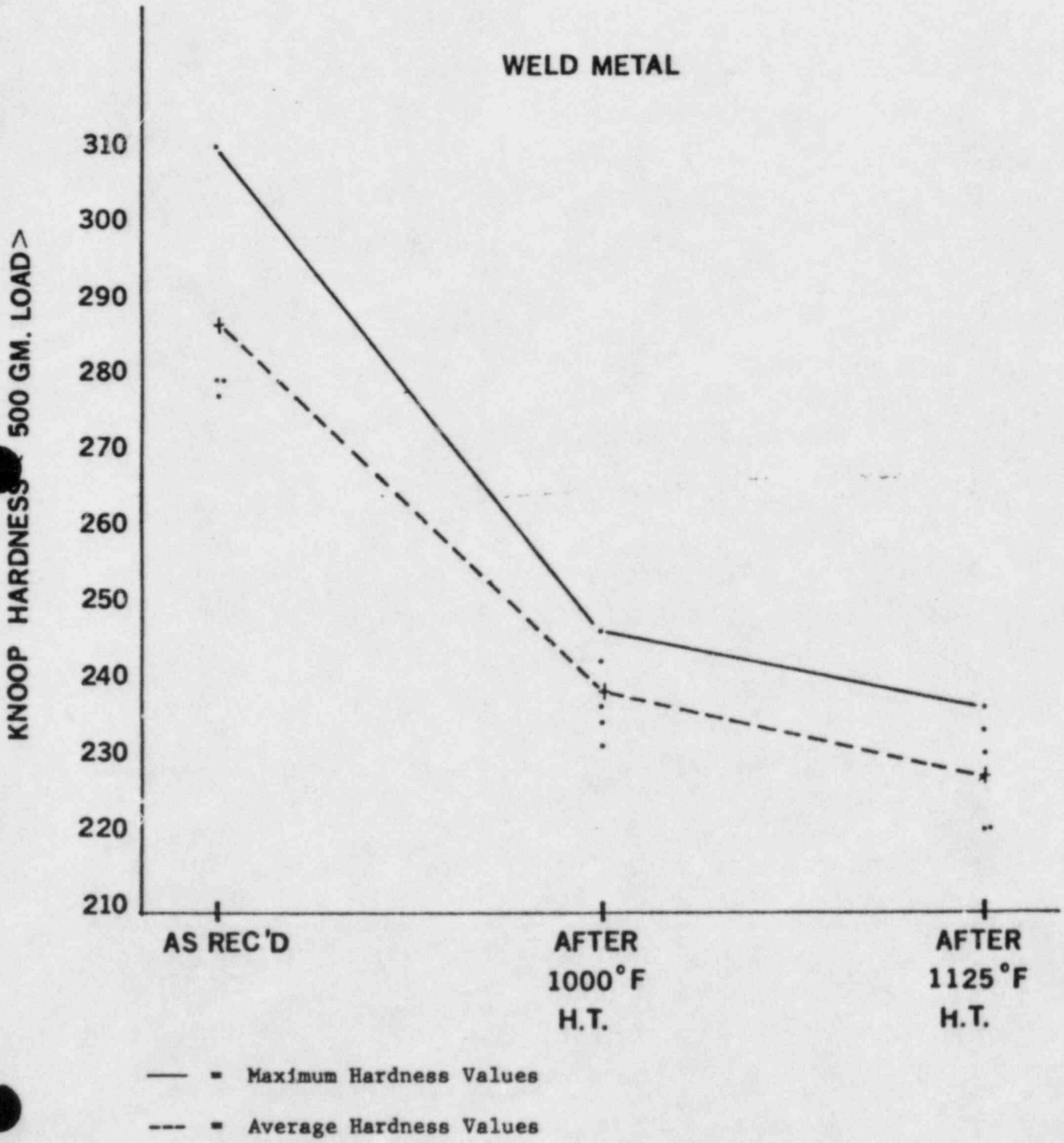
6. WIDE AND HEAVY MAJOR REPAIR WELDS

7. INADEQUATE POST WELD TREATMENT

8. THERMAL CYCLING

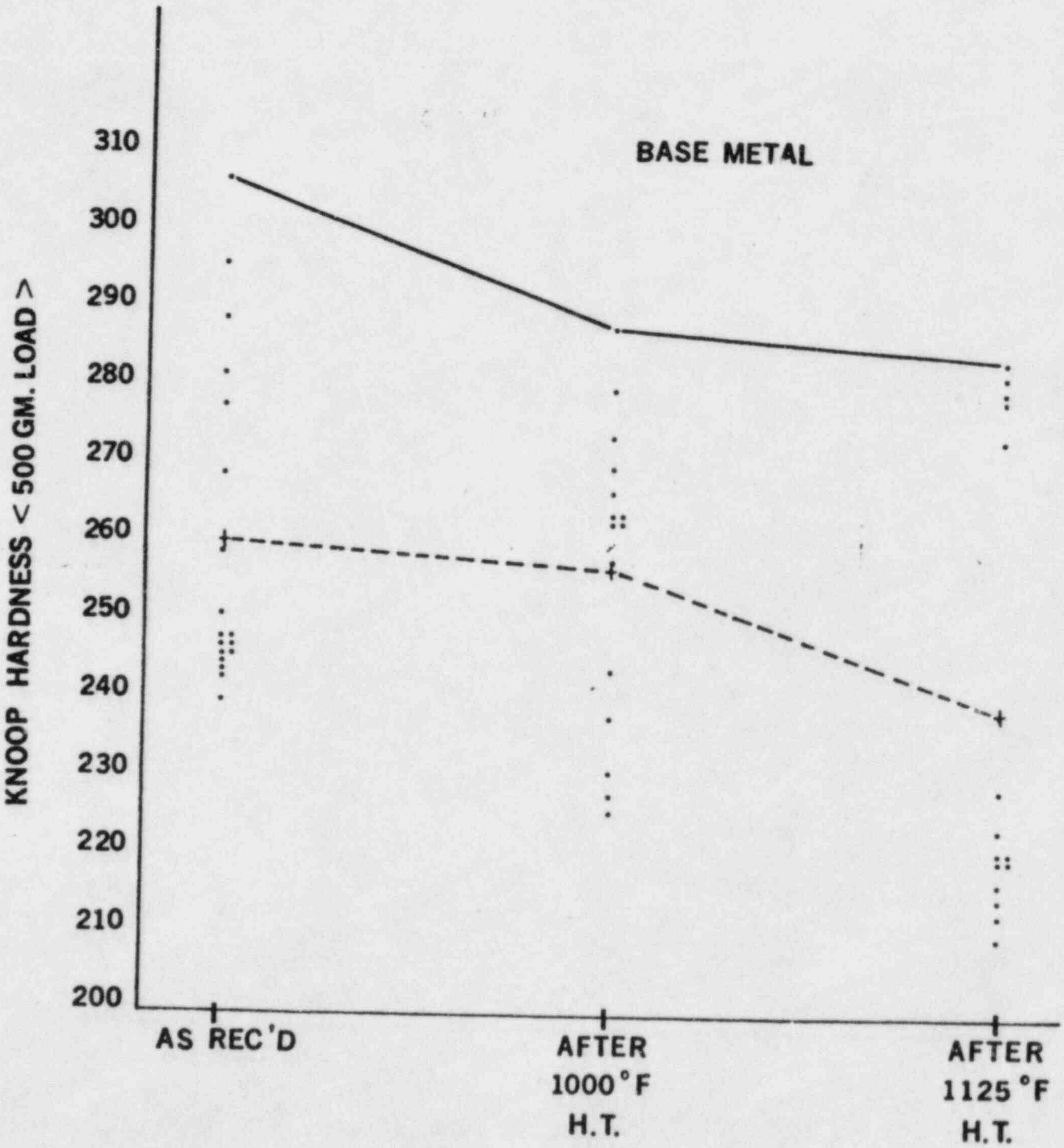
GRAPH #2

A Graphical Comparison of the Hardness Values of the WELD METAL After Heat Treatment



GRAPH #1

A Graphical Comparison of the Hardness Values of the BASE METAL After Heat Treatment



——— = Maximum Hardness Values

--- = Average Hardness Values

BROOKHAVEN CONCLUSIONS

1. ALL CRACKS TRANSGRANULAR
2. HARDNESS INDICATES POSSIBILITY OF HIGH RESIDUAL STRESSES
3. POSSIBLE FATIGUE INDICATIONS - LOW CYCLE CORROSION FATIGUE
4. RELATIVE IMPORTANCE OF CORROSION IS UNDETERMINED

SURRY 2

SGs REPLACED IN 1979

TRANSITION WELD WAS NOT REPLACED

WELD WAS UT INSPECTED 1983
(IE INFORMATION NOTICE)
INDICATIONS WERE CALLED GEOMETRY

WELD WAS MAGNETIC PARTICLE INSPECTED IN 1985
INDICATIONS FOUND TO BE CRACKS

CRACKS COULD BE REPAIRED BY GRINDING

OH-1

UT Indication
Background

- *UT indication identified August 1983*
- *Initial evaluation by Virginia Power was contour*
- *NRC questioned evaluation*
- *NRC evaluated indications as either contour or cracking*
- *Virginia Power agreed to MT inspect in 1985*

1

(25)

OH-2

Indications

- *Pits over general area/visual*
- *Linear crack-like at toe of crown of weld #6/visual 10x MT, UT*
- *Subsurface in region of OD weld root/UT*

OH-3

Repair of Welds by Grinding

- *All three steam generators*
- *Surface flaws only*
- *Use of detailed repair procedures*
- *Grinding in increments of 1/16" or 1/8"*
- *MT inspection after each increment of grinding*
- *Final surface MT clear*

Surface Flaws

- *All surface flaws removed by 0.500" grinding depth in "A" SG*
- *All surface flaws removed by 0.31" grinding depth in "B" and "C" SG*
- *"A" SG contour ground more deeply than necessary at several locations*

04-10

Chemistry Considerations

- *Secondary chemistry problems prior to SG/condenser replacement*
- *Oxygen often greater than 25 PPB*
- *Chloride often in PPM range occasionally 300 to 400 PPM*
- *Copper alloy condensers and feedwater heaters*
- *Since S/G/condenser replacement chemistry has been very good*

Construction Period Welding Considerations

- ID preheat temperature 180 - 185°F
- OD preheat temperature 210 - 220°F
- ID welded completed before OD
- PWHT was accomplished at 1000°F to 1100°F

(35)

OH - 12

Factors Mitigating Re-Initiation of Cracking

- *Much better water chemistry*
- *Lower residual stress due to PWHT of weld 6 with weld 11 at 1150 to 1180°F*
- *Removal of all linear indications*

SUMMARY OF VEPCO CONCLUSIONS

1. WATER CHEMISTRY DURING PRE-REPLACEMENT HAD OVER 25 PPB OXYGEN, CHLORIDES AND COOPER IONS, CAUSING PITTING.

2. CRACKS INITIATED FROM PITS - STRESS CORROSION OR CORROSION FATIGUE.

3. CRACKS WERE PROPAGATED BY CORROSION, STATIC STRESS, (OPERATIONAL AND POSSIBLY RESIDUAL) AND FATIGUE.

VEPCO

ORIGINAL WELDING FINDINGS

1. PREHEAT ON THE LOW SIDE

2. WELD SEQUENCE MAY HAVE OUT ID IN TENSION

3. POST WELD HEAT TREATMENT (1000 to 1100F) MAY HAVE LEFT HIGH RESIDUAL STRESSES

STEAM GENERATOR GIRTH WELD CRACK SUMMARY

INDIAN POINT 3

SG	Total Excavation lengths in 528	Max. depth of excavations	% circumference with MT indications
31	711	2	~95
32	650	2 5/8 (also through wall)	~80
33	579	2 1/8	~85
34	397	1 1/4	~70

SURRY 2

		Max. depth crack contour	% circumference ground
A	406	.500 .850	.77
B	375	.3125	.71
C	484	.3125	.92

MTEB CONCLUSIONS

THE CRACKING IN IP3, SURRY 2, AND GARIGLIANO ARE RELATED

THE CAUSE IS MOST LIKELY A COMBINATION OF STRESS CORROSION AND FATIGUE

CONTRIBUTION FACTORS

LOW PWHT TEMPERATURE

- HIGH HARDNESS-HIGH SUSCEPTIBILITY
- HIGH RESIDUAL STRESSES

ENVIRONMENT

- OXYGEN IN WATER
- COPPER, SULFUR SPECIES, CHLORIDES

STRESS CYCLING

- NORMAL START-UPS & SHUTDOWNS
- POSSIBLE COLD FEEDWATER EFFECT

INFORMATION AVAILABLE IS NOT SUFFICIENT TO QUANTIFY THE SYNERGISMS FOR PREDICTIVE PURPOSES.

NUREG/CR-3281
BNL-NUREG-51670

Investigation of Shell Cracking on the Steam Generators at Indian Point Unit No. 3

Manuscript Completed: April 1983
Date Published: June 1983

Prepared by
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Brookhaven National Laboratory
Department of Nuclear Energy
Upton, NY 11973

**Prepared for
Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN A3400**

NUREG/CR-3614
BNL-NUREG-51736

Constant Extension Rate Testing of SA302 Grade B Material in Neutral and Chloride Solutions

Prepared by C. J. Czajkowski

Brookhaven National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

NUREG/CR-3728
ORNL/TM-9149

OAK RIDGE
NATIONAL
LABORATORY

MARTIN MARIETTA

Effect of Temperature on the Stress-Relaxation Response of a Pressure Vessel Steel

G. M. Goodwin
R. K. Nanstad

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreement DOE 40-543-75

OPERATED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

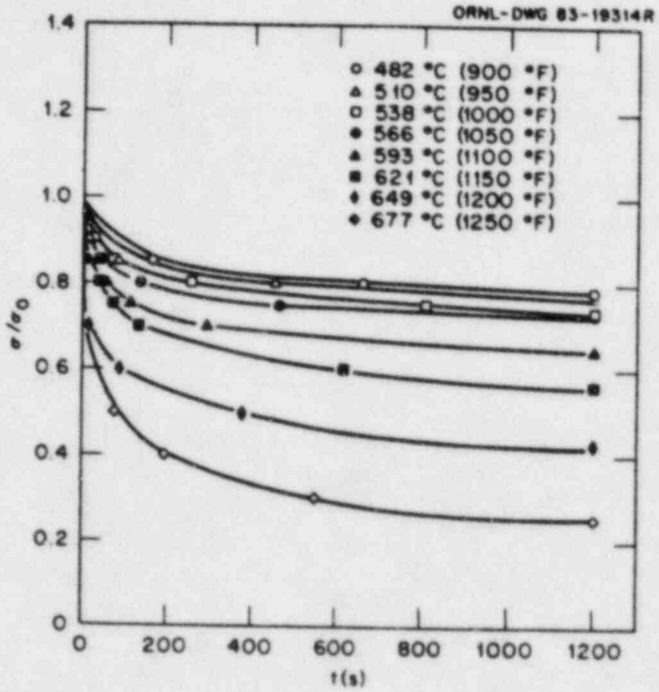


Fig. 4. Stress relaxation versus time for SA-533, grade B, steel.

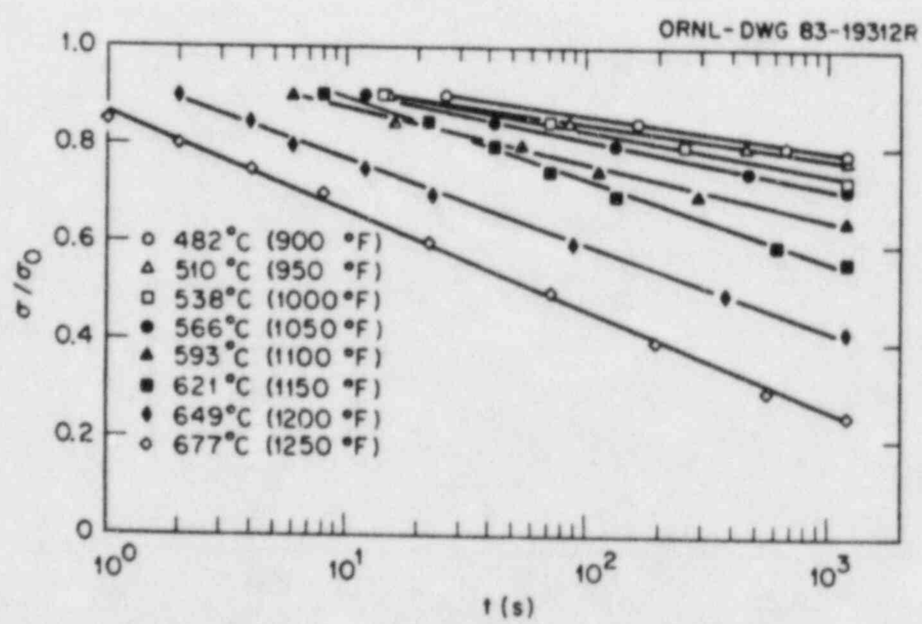


Fig. 5. Stress relaxation versus log time for SA-533, grade B, steel.

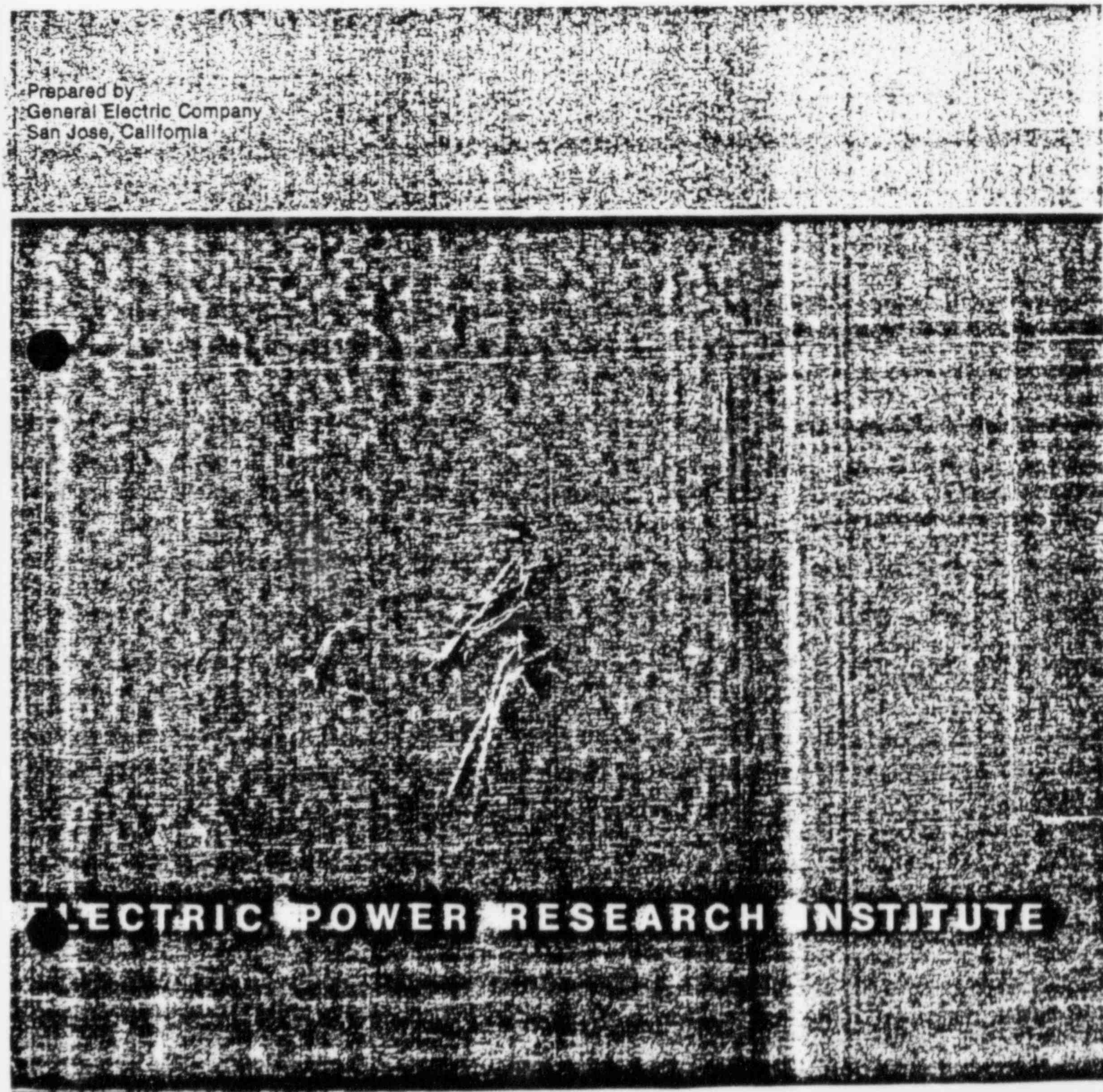
BWR Environmental Cracking Margins for Carbon Steel Piping

EPRI

EPRI NP-2406
Project 1248-1
Final Report
May 1982

Keywords:
Nuclear Reactor Piping
Carbon Steels
Corrosion Fatigue
Stress Corrosion
Reactor Coolant
Piping Design Criteria

Prepared by
General Electric Company
San Jose, California



2. SCC of the sensitized steel is electrochemically controlled. (a) The cracking could be stopped at potential more negative than −0.5 V; (b) the rates of SCC increased with potential; and (c) the rate of cracking was proportional to the current.

3. The cracking is considered to be a result of dissolution of the chromium depleted grain boundary and is controlled by the rupture and formation of a salt layer at the crack tip rather than a passive oxide film.

Acknowledgments

The authors thank Kenneth A. Sutter for his substantial contribution in conducting the measurements, Robert S. Sabatini for the scanning microscopy, and John R. Weeks for his helpful discussions.

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Stress Corrosion Cracking of ASTM A508 Cl 2 Steel in Oxygenated Water at Elevated Temperatures*

H. CHOI, F. H. BECK,* Z. SZKLARSKA-SMIALOWSKA,(1) and D. D. MACDONALD**

Abstract

A508 Cl 2 steel has been found to undergo stress corrosion cracking (SCC) in pure water containing 1 or 8 ppm of oxygen at temperatures ranging from 100 to 288 C, as determined using constant extension rate tests (CERTs). At temperatures of 100 and 150 C, cracks nucleate at corrosion pits. At higher temperatures, however, cracks nucleate beneath hematite crystals which grow via a dissolution-precipitation mechanism upon a base oxide film at sites of high anodic dissolution activity. The susceptibility of SCC increases with increasing oxygen concentration, but passes through a maximum as a function of temperature at 250 C.

Introduction

Intergranular stress corrosion cracking of some austenitic stainless steels in boiling water nuclear reactor (BWR) environments (oxygenated water at high temperature) occurs in heat affected zones adjacent to welds. Accordingly, considerable effort is being expended to identify substitute materials that are not prone to SCC in this environment. One of the candidate materials is plain carbon steel.

Indig[1] found that carbon steel is less susceptible to SCC than sensitized Type 304 stainless steel. At 273 C, stress corrosion cracking of carbon steel in air saturated water occurred only in constant extention rate tests, but not in constant strain test (U-bend test). Transgranular SCC and pitting of carbon steel SA 333 GR 6 was found to occur in oxygenated water at temperatures in the range 50 to 287 C.[2]

The aim of this work is to determine the susceptibility of A508 Cl 2 carbon steel to pitting and SCC in water containing varying amounts of oxygen at temperatures ranging from 25 to 288 C. This material is commonly used for the fabrication of pressure vessels in water cooled nuclear reactor systems.

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*Submitted for publication June, 1981; revised November, 1981.

*Fontana Corrosion Center, Department of Metallurgical Engineering, The Ohio State University, Columbus, Ohio.
(1)On leave from the Polish Academy of Science, Warsaw, Poland.

0010-9312/82/000045/$3.00/0

First, let me clarify why ... *

Although our main presentation is on S/G shell cracking there is a strong implication to reactor vessels where water chemistry may be similar - BWR vessels, PWR secondary sides. - The overall subject is really on the stress corrosion cracking in low alloy steel that we used to believe that it is immune to SCC -

REGULATORY RESPONSES

- I&E INFORMATION NOTICE 82-37 ISSUED ON 09/16/82 SHORTLY AFTER THE CRACKING WAS FOUND AT INDIAN POINT 3
- INITIATED A STUDY @ BROOKHAVEN NATIONAL LAB.
- PROPOSED A RESEARCH WORK - GET PRIORITIZED
- I&E INFORMATION NOTICE 85-65 ISSUED ON 07/31/85 AFTER SURRY 2; PRINCIPALLY TO POINT OUT THE DIFFICULTIES IN UT EXAMINATION
- REGULATORY PRIORITIZATION WAS COMPLETE IN RESPONSE TO DE'S REQUEST FOR RESEARCH
- DIVISION OF ENGINEERING PROPOSED ON 07/10/85 THE ISSUANCE OF AN I&E BULLETIN
- H. L. THOMPSON'S RESPONSE BY MEMO DATED 08/22/85

I will call your attention to 4th paragraph - If one removes the words "perhaps" "possible" and change "if" to "when". Then the the first sentence would be perfect in characterizing the ~~[illegible]~~ principal benefit of doing something about the problem —

#*

RESULTS OF PRIORITIZATION EVALUATION

- INITIAL RESULTS CLASSIFIED IT AS A "LOW PRIORITY" ISSUE
- AFTER WE RAISED THE ISSUE OF WHETHER REGULATION IS MET; WHETHER IT IS CONSISTENT WITH THE "DEFENSE-IN-DEPTH" PHILOSOPHY UNDERLIED THE REGULATION, THE ISSUE WAS RECLASSIFIED AS "LICENSING ISSUE"
- PRINCIPALLY, THE RESULTS SHOW THAT IT IS MORE OF AN ECONOMIC ISSUE THAN A SAFETY ISSUE
- OFFICE DIRECTOR HAS NOT DECIDED WHETHER HE AGREES WITH THE RESULTS OF THE EVALUATION

When and if he approves, it will be sent to ACRS for comments. Any comments you may make will be incorporated into ~~the~~ NUREG-0933 "Prioritization of Generic Safety Issues".

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

AUG 2 2 1985

MEMORANDUM FOR: James P. Knight, Acting Director
Division of Engineering, NRR

FROM: Hugh L. Thompson, Jr., Director
Division of Licensing, NRR

SUBJECT: STEAM GENERATOR SHELL TRANSITION JOINT CRACKING

This is in response to your memorandum dated July 10, 1985, which recommended issuance of an I&E Bulletin to require inspections of steam generator transition cone girth welds at all Westinghouse and Combustion Engineering plants.

We have reviewed your proposal and have discussed it with members of the Materials Engineering Branch, DE, and with I&E. As you are aware, IE Information Notice 85-65, "Crack Growth in Steam Generator Girth Welds" was issued on July 31, 1985. This information notice describes the recent experience with girth weld cracks and advises utilities of the limitations of normal inspection methods in detecting such cracks. Based on our understanding of this issue, and in light of the recent IE Notice, it appears to us that there is no immediate safety concern to justify issuance of a bulletin at this time. As stated in your memorandum, any significant cracking in the girth weld would be expected to cause a leak (thus providing a warning) prior to a catastrophic failure of the shell.

In the absence of an immediate safety concern, the most appropriate action at this time would be to submit the subject proposal to a regulatory analysis in accordance with existing NRR policy.

Perhaps the major benefit to be gained from the proposed inspections, apart from potential economic benefits, are possible reductions in future occupational radiological exposures (ORE) if repairs are required, compared to what would otherwise be incurred during an inspection conducted at this time. However, in the absence of a value-impact study (which would be part of the regulatory analysis), it is far from clear that the potential reductions in ORE would actually be realized or would be of sufficient magnitude to make the proposed inspections a high priority item.

CONTACT: E. Murphy
X 27457

Pending completion of the regulatory analysis, DE may wish to recommend issuance of a supplemental IE Notice describing which plants have the highest potential for having such cracks and to emphasize the potential benefits in the areas of averted cost and occupational exposures which can be gained through timely inspections.

Dennis M. Crutchfield for
Hugh L. Thompson, Jr., Director
Division of Licensing

cc: G. Holahan
R. Wessman
W. Johnston
B. Liaw
D. Smith
W. Hazelton
A. Dromerick, IE
R. Baer, IE
T. Speis
W. Minners

(3)

IASCC

1. REVIEW OF IGSCC AND THE IASCC PHENOMENA
2. USE OF STAINLESS STEELS IN REACTORS
3. SOME EXAMPLES OF CRACKED PARTS
4. PROGRAMS INVESTIGATING IASCC

   MITIGATION

   FUNDAMENTALS

5. SESS VIEW OF IASCC

AMERICAN SOCIETY FOR METALS
Metals Park, Ohio 44073

Metals/Materials Technology Series

IRRADIATION-ASSISTED STRESS CORROSION CRACKING AS A FACTOR IN NUCLEAR POWER PLANT AGING

A. J. Jacobs, G. P. Wozadlo
General Electric Company
San Jose, California

ASM's International Conference on Nuclear Power Plant Aging, Availability Factor and Reliability Analysis
San Diego, California
8-12 July 1985

8507-002

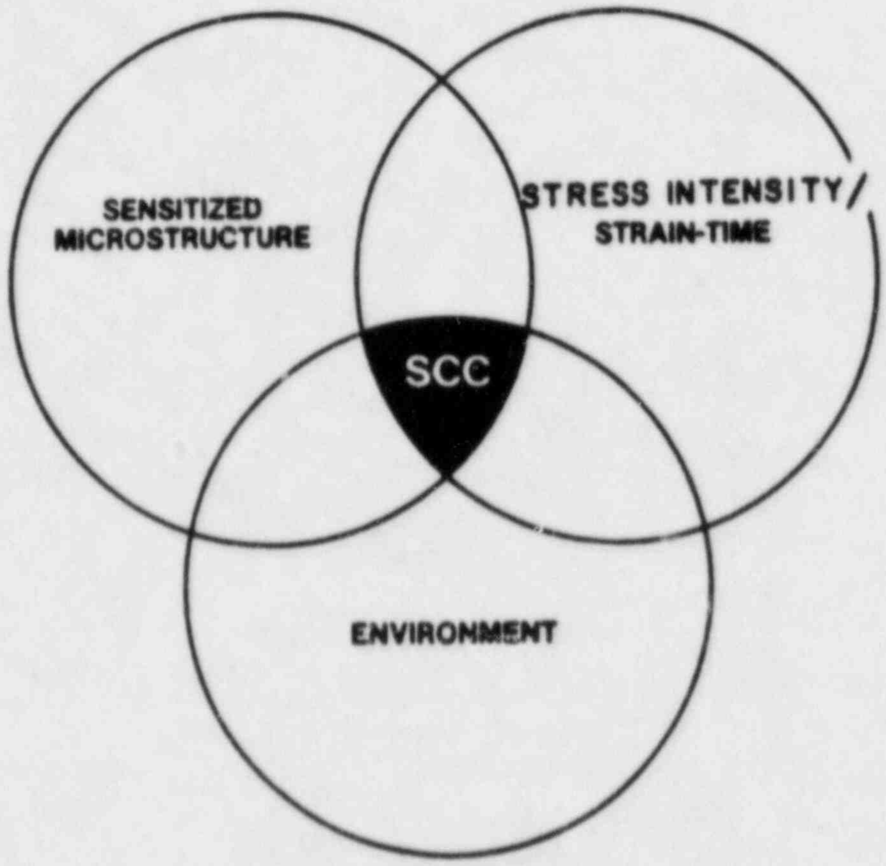
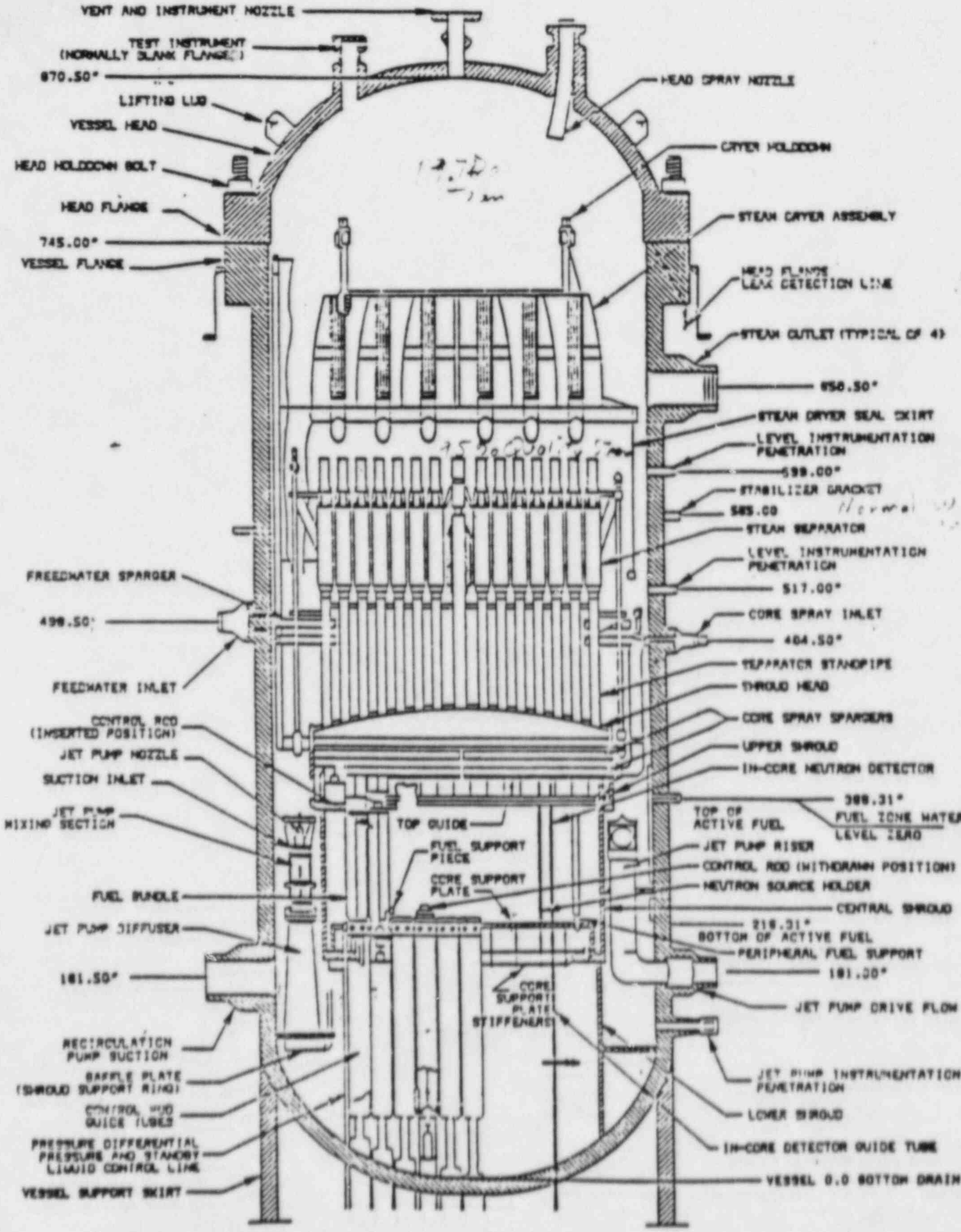


Fig. 1 - Critical overall factors for stress corrosion cracking of 304 SS.

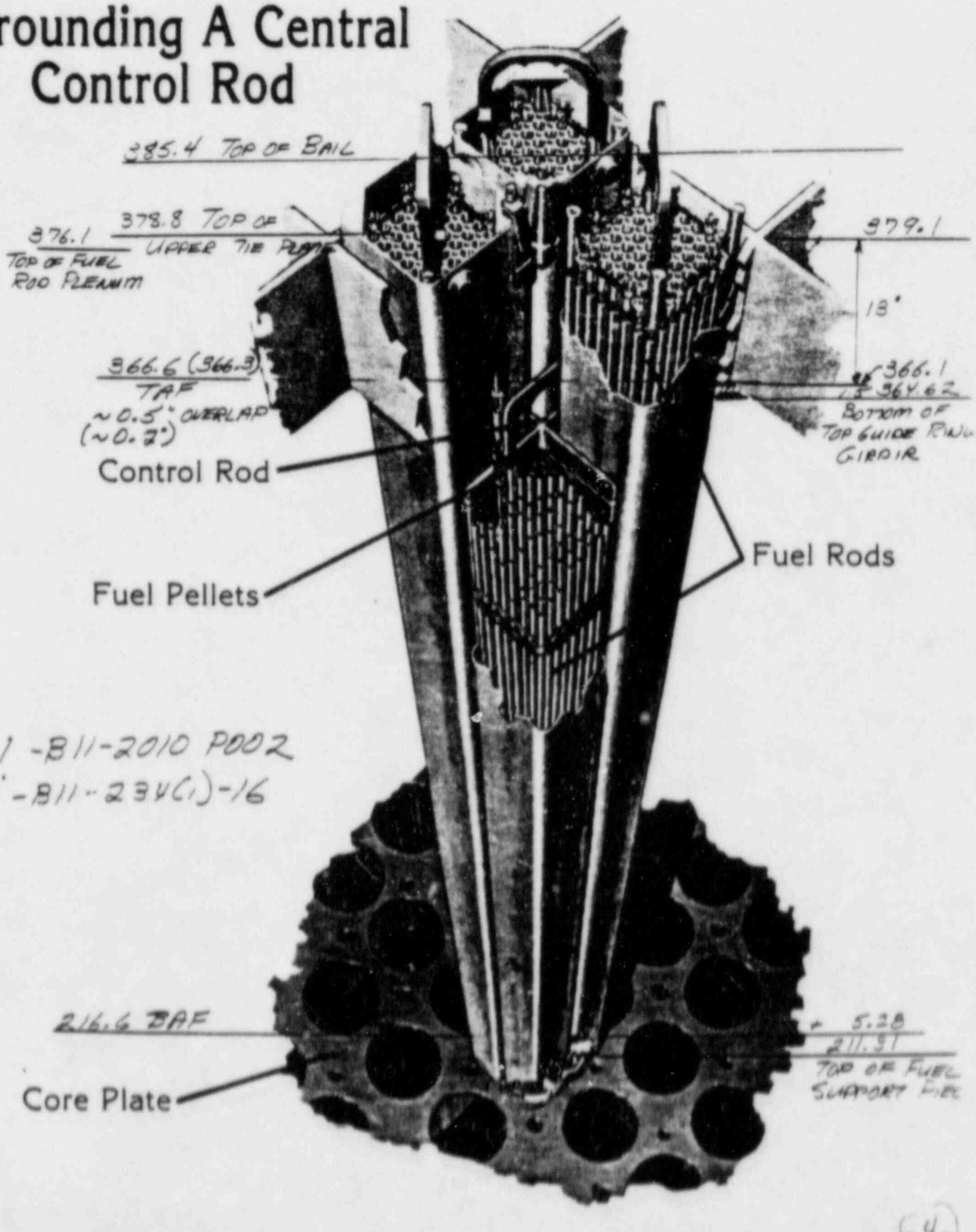
(2)

FIGURE 1



REACTOR VESSEL COMPOSITE DRAWING

Four Fuel Assemblies Surrounding A Central Control Rod



M-1-B11-2010 P002
mi-B11-23V(1)-16

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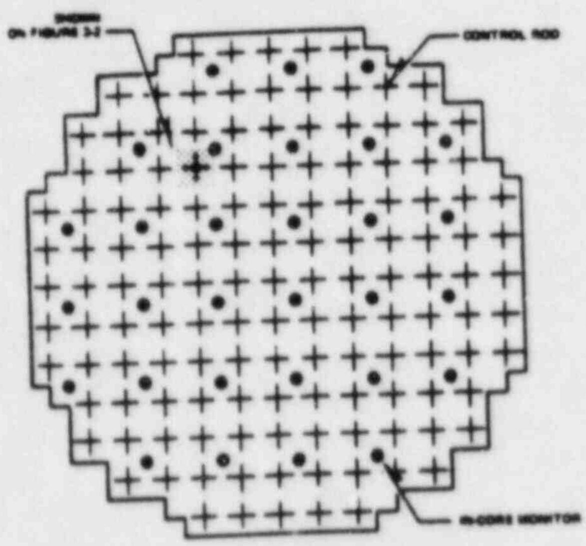
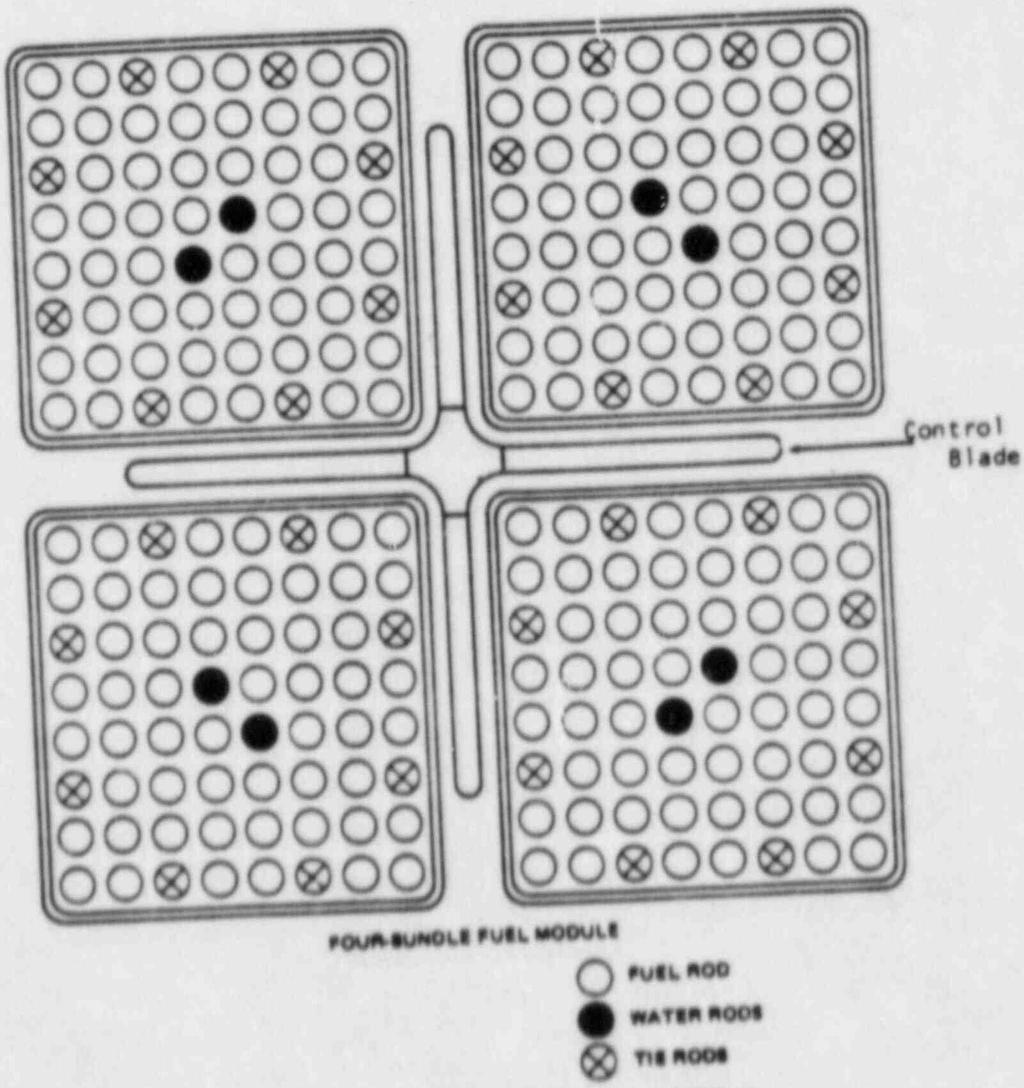
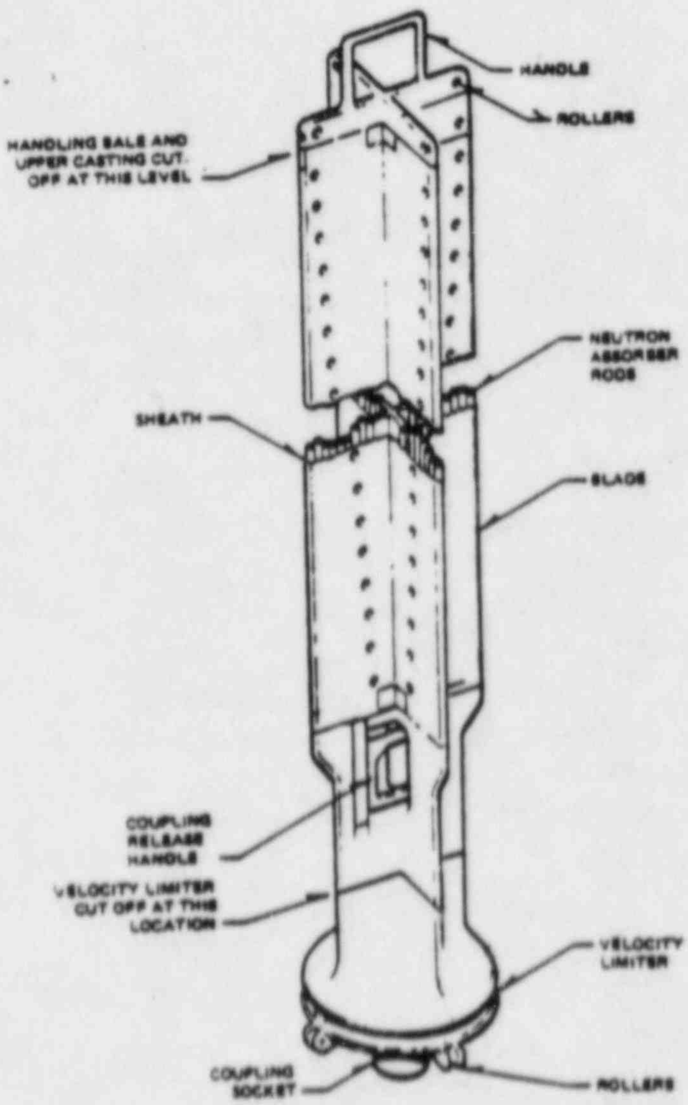




FOUR-BUNDLE FUEL MODULE

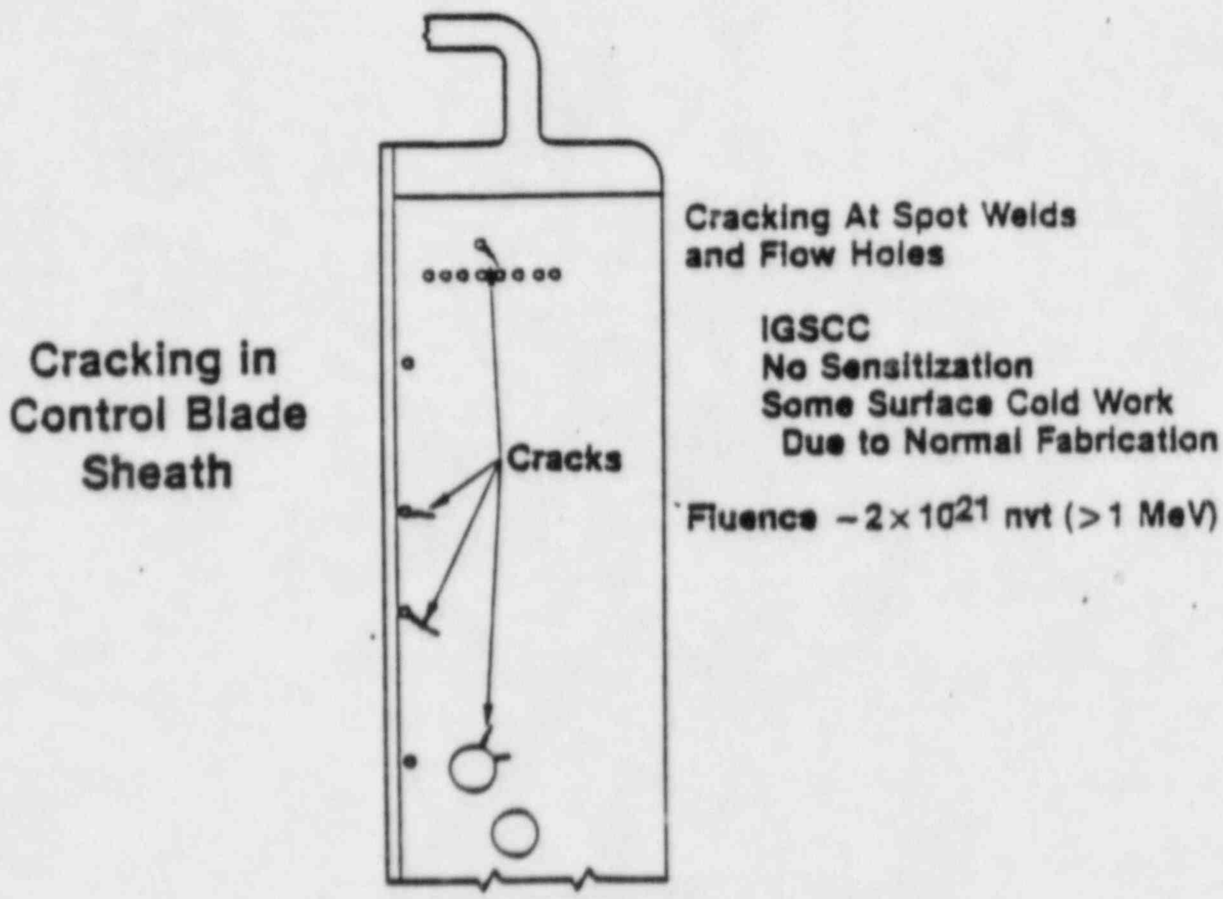
- GENERAL ELECTRIC COMPANY -

NEDE-25281
GENERAL ELECTRIC COMPANY [redacted]
Class III



B_4C Control Blade

FIGURE 4-3



Cracking in Control Blade Sheath
Cracks
Cracking At Spot Welds and Flow Holes
IGSCC
No Sensitization
Some Surface Cold Work Due to Normal Fabrication
Fluence ~2 × 10²¹ nvt (>1 MeV)

FIGURE 4-2

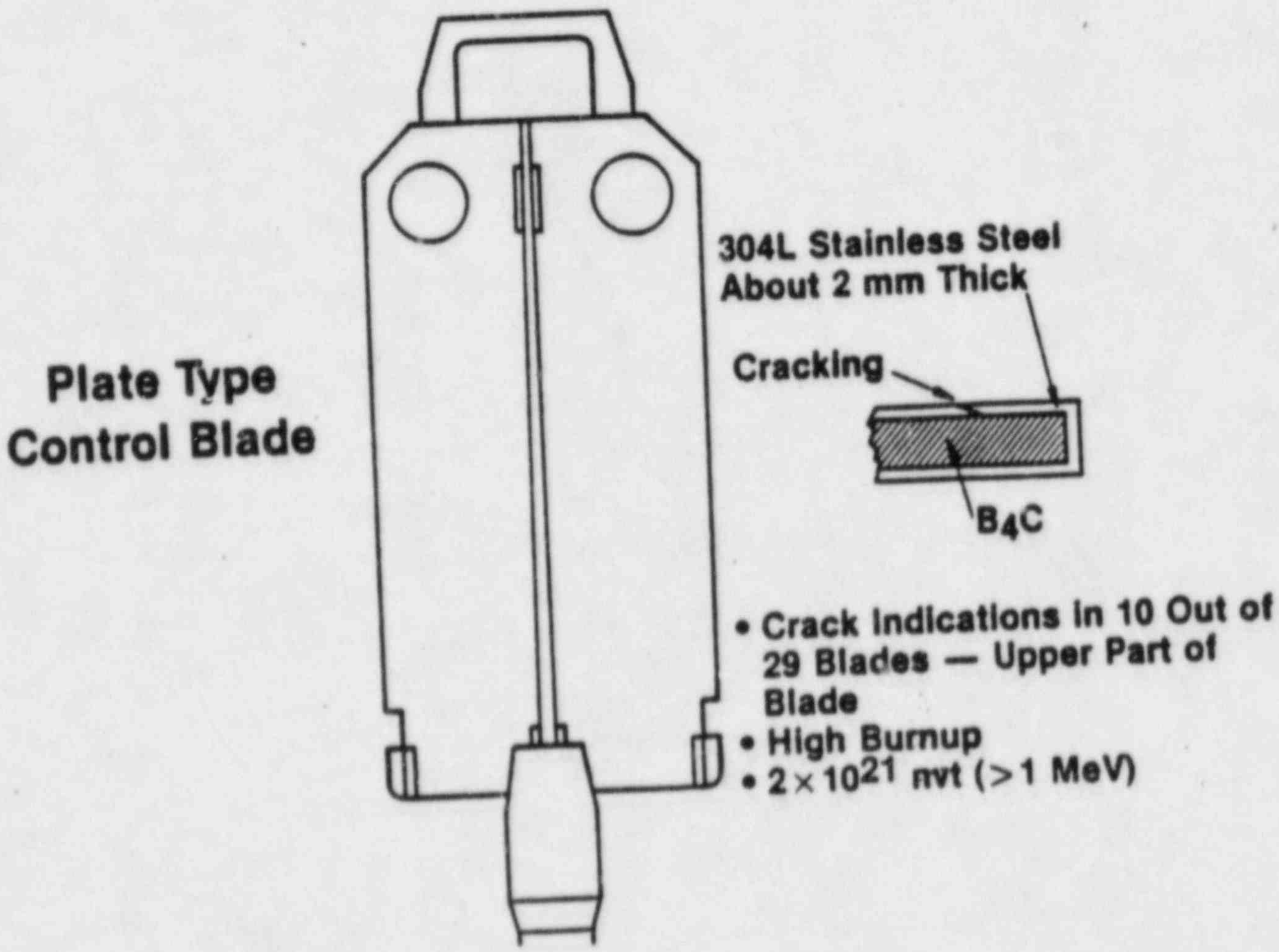


FIGURE 4-7 Configuration of Swedish Plate Control Blade failed by IASCC in 1983

Microstructure Examination of Control Blade Handle

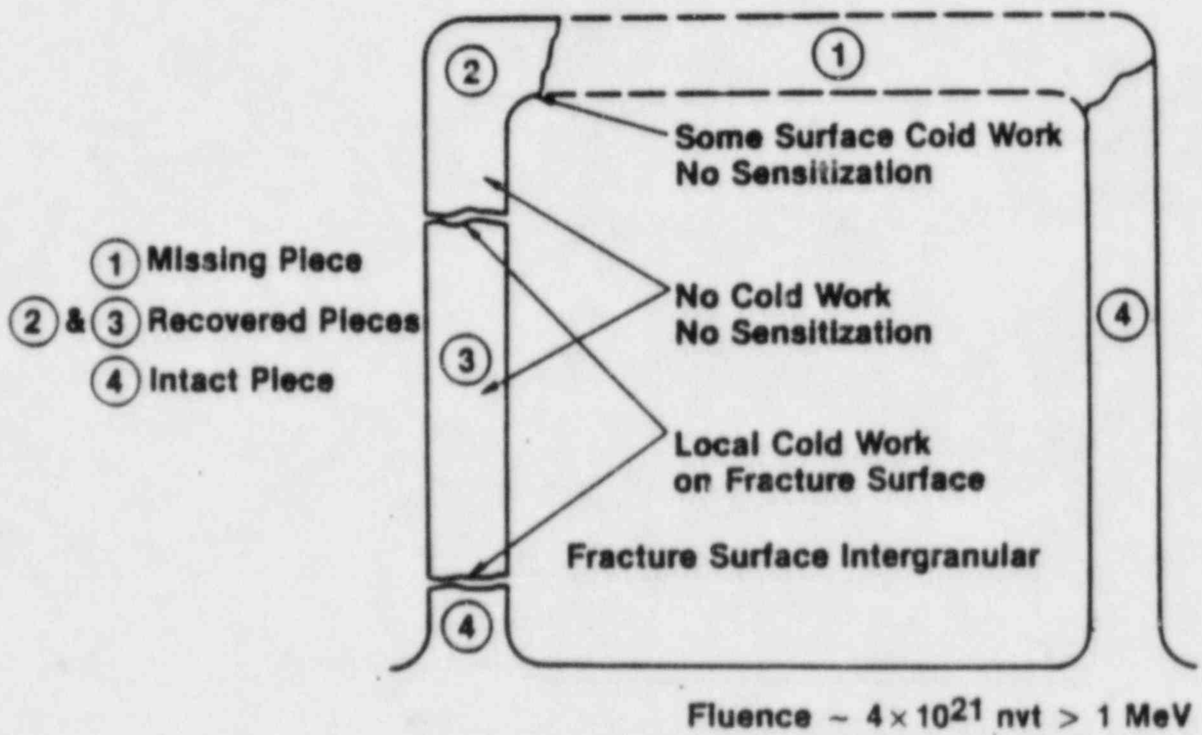
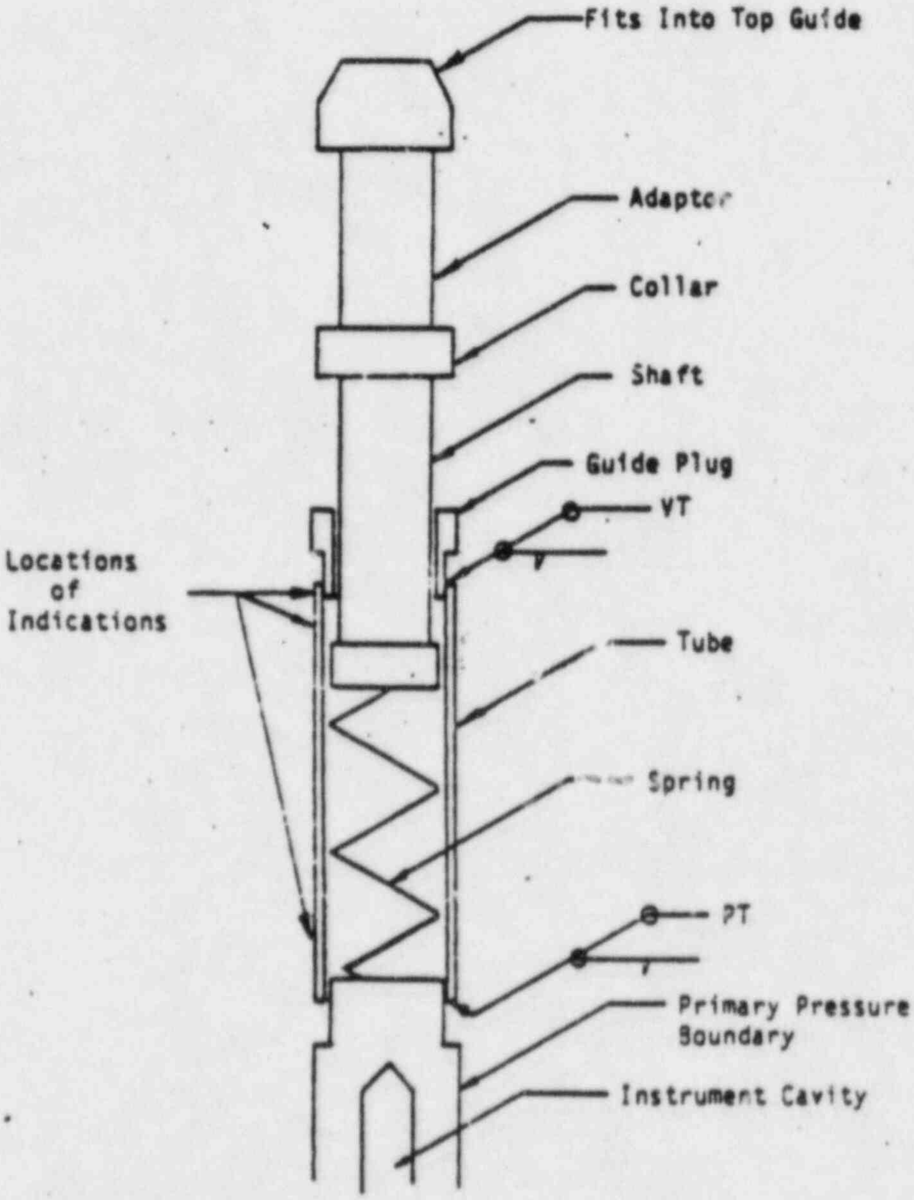


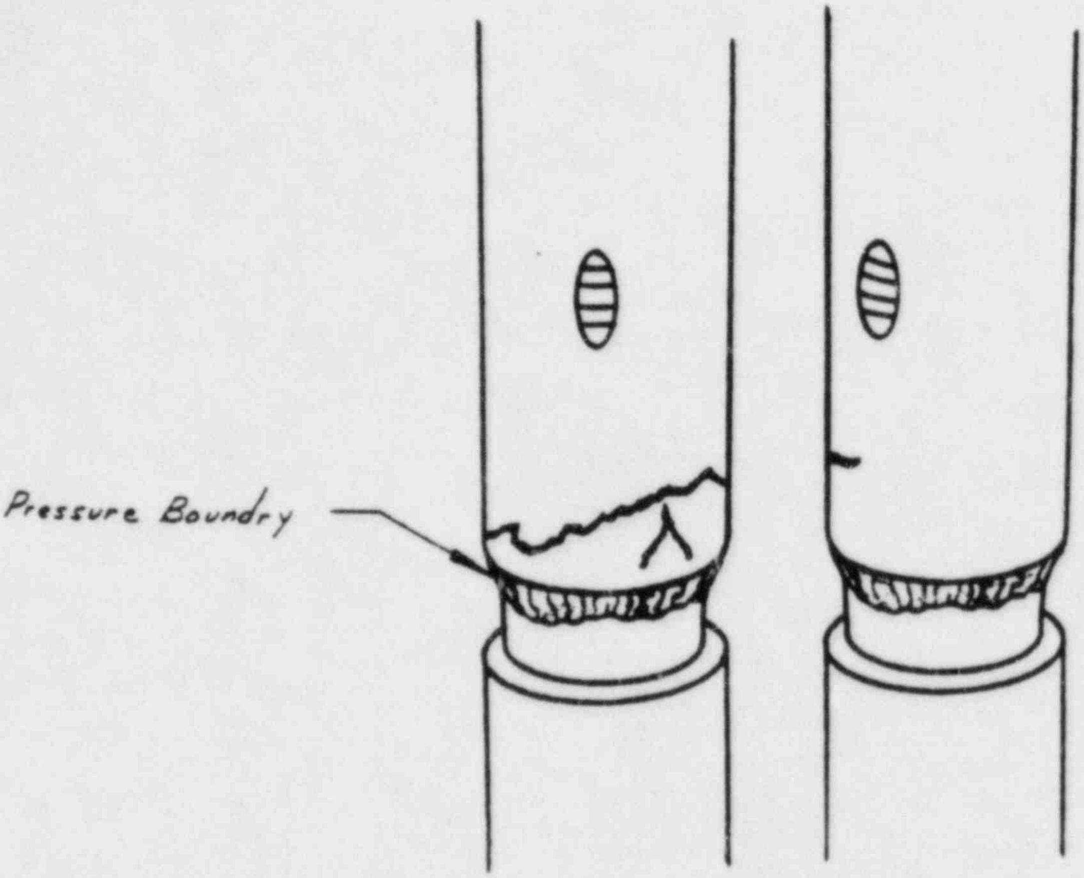
Figure 4-1



SCHEMATIC OF TOP PORTION OF DRY TUBE

FIGURE 1

Hatch Unit No. 1
IRM 12-41
12 Nov. 1984



Position B   Position C

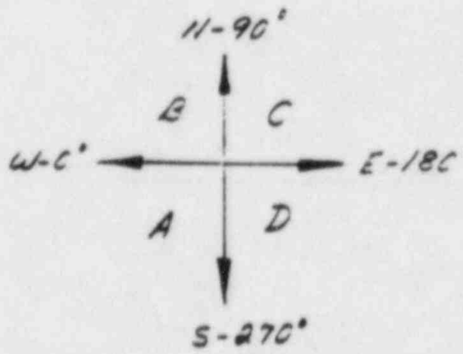


TABLE 4-4

SUMMARY OF FIELD EXPERIENCE
UP TO 1980

COMPONENT	PLANTS AFFECTED	FLUENCE (N/CM^2)	SOURCE OF STRESS
FUEL CLADDING		$5X10^{20}$ - $2X10^{21}$	• FABRICATION • FUEL CLADDING INTERACTION
NEUTRON SOURCE HOLDERS		10^{21} - 10^{22}	• WELDING • BERYLLIUM SWELLING AFTER INITIAL CREVICE ATTACK
CONTROL ROD ABSORBER TUBES		$5X10^{20}$ - $3X10^{21}$	• B_4C SWELLING
FUEL BUNDLE CAP SCREWS		10^{21} - 10^{22} (ESTIMATED)	• FABRICATION AND/OR ASSEMBLY
RIVETS IN CONTROL ROD FOLLOWER		$\sim 5X10^{20}$	• NOT WELL UNDERSTOOD

- 304, 304L, 347, 348SS, INCOLOY 800
- COMPONENTS SOLUTION ANNEALED - NO SENSITIZATION
- THRESHOLD FLUENCE > $5X10^{20}$ N/CM^2
- HIGH STRESS REQUIRED

TABLE 4-3

I. FIELD SERVICE DATA

- STAINLESS STEEL FUEL CLADDING	1960s
- CONTROL BLADE ABSORBER TUBES	LATE 1970s
- STARTUP SOURCES	MID 1970s
- FOLLOWER RIVETS	OLD
- CONTROL BLADE HANDLE	1982
- CONTROL BLADE SHEATH	1983
- PLATE TYPE CONTROL BLADE	1983
- IRM'S/SRM'S	1984

Table 1. Room Temperature Tensile Properties of Irradiated Plate Materials

Fluence	Yield Strength (MPa/Ksi)	Ultimate Tensile Strength (MPa/Ksi)	Elongation (%)
3×10^{21} n/cm^2	820/118	859/125	18
1×10^{18} n/cm^2	195/28	505/73	75

Table 2. CERT Results for Annealed and Irradiated Type-304 Plate Material (288°C Water With 32-36 mg/l O_2)

Test No.	Sample Fluence (n/cm^2, E>1 MeV)	Strain Rate (min^{-1})	Test Time (h)	Max. Stress (MPa/Ksi)	Fracture Mode (SEM)
HC-1	3×10^{21}	4×10^{-6}	191	710/103	IGSCC (100%)
HC-2	1×10^{18}	4×10^{-6}	408	372/54	Ductile
HC-3	3×10^{21}	1×10^{-4}	25	717/104	IGSCC (40%)
HC-4[a]	1×10^{18}	4×10^{-6}	382	382/52	Ductile

[a]High radiation source (1/2 of high-fluence sample) attached to test sample.

Table 3. CERT Results for Samples Fabricated from Irradiated Reactor Tubing (288° Water with 32-36 mg/l O_2)

Test No.	Sample Fluence (n/cm^2, E>1 MeV)	Strain Rate (min^{-1})	Test Time (h)	Max. Stress (MPa/Ksi)	Mode (SEM)
3-13-DL	2.77×10^{21}	1×10^{-5}	1	194/28	IGSCC (100%)
4-13-DH	2.30×10^{21}	1×10^{-5}	28	303/44	IGSCC (11%)
3-13-DG	2.08×10^{21}	5×10^{-6}	41	641/93	Ductile
3-13-DF	2.01×10^{21}	1×10^{-5}	93	572/83	IGSCC (45%)
4-13-DE	2.01×10^{21}	1×10^{-6}	329	503/73	IGSCC (55%)
3-13-CB	1.28×10^{21}	5×10^{-6}	98	406/59	IGSCC (5%)
3-13-BK	9.50×10^{20}	1×10^{-5}	85	537/78	IGSCC (5%)
4-13-BH	5.0×10^{20}	1×10^{-5}	32	462/67	IGSCC (8%)
4-13-AL	6.0×10^{19}	1×10^{-5}	56	331/48	Ductile

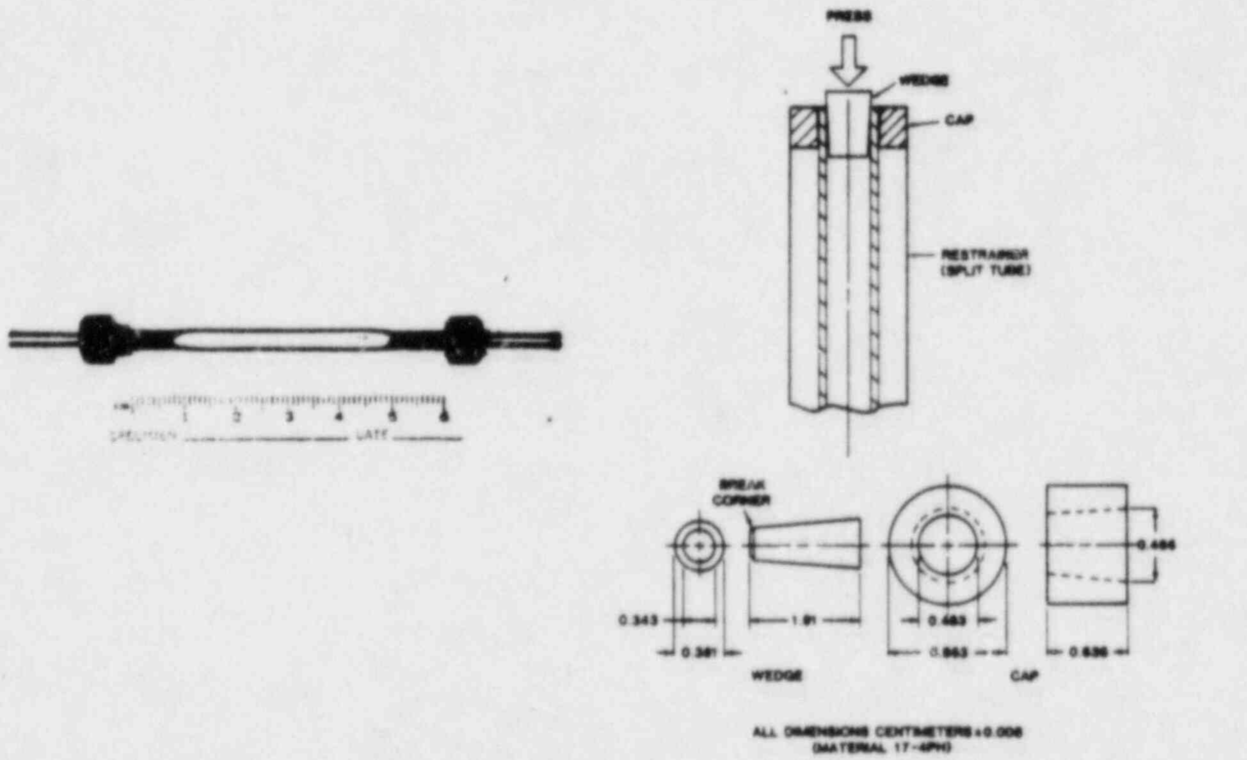


Figure 1. Button Head Uniaxial Tensile Sample Design for SCC Testing Small Diameter (<1.61 cm) Tubing.

-15

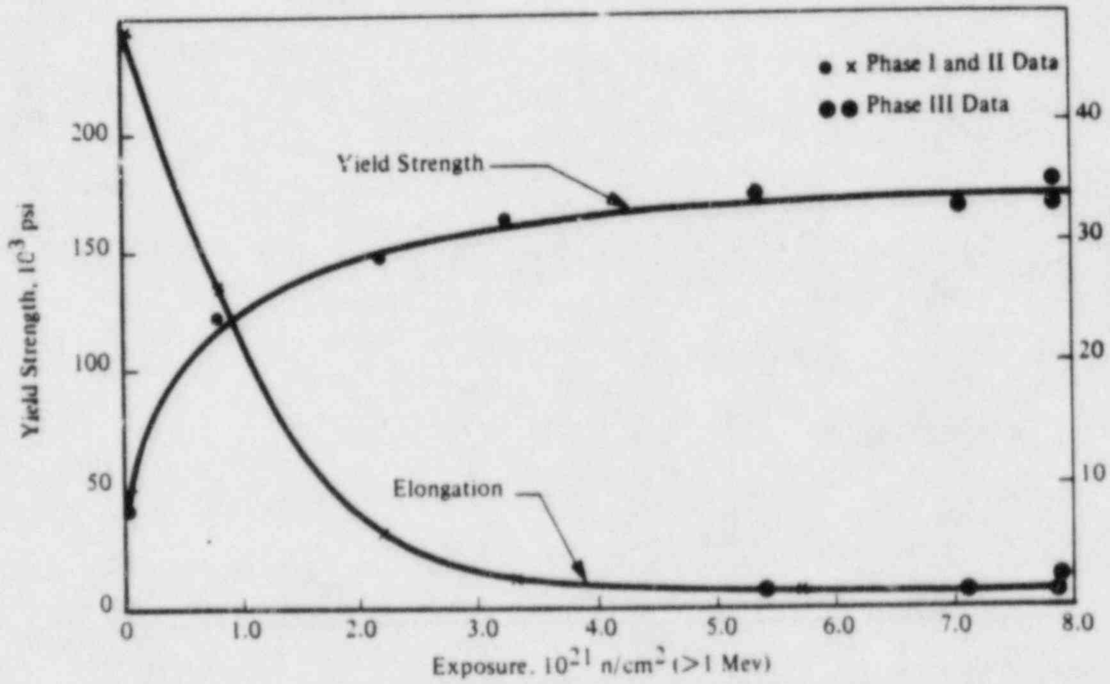


Figure 3-3. Room Temperature Tensile Properties of Irradiated Yankee Fuel Cladding. (3.15)

-16-

STAINLESS STEEL COMPONENTS IN BWR'S AND EXPECTED FLUENCES
****

	FLUX	FLUENCE(32 EFPY)
1. FUEL CLADDING	4.8E13 N/CM^2-S	4.8E22
2. CONTROL RODS	4.8E13	4.8E22
3. DRY TUBES	4.8E13	4.8E22
4. CORE BOUNDRY	1.7E13	1.7E22
5. TOP GUIDE	9.0E12	9.0E21
6. SHROUD	9.0E11	9.0E20
7. VESSEL WALL	6.1E8	6.1E17

GE PROGRAMS

A. CHARACTERIZE CRACKING
B. HIGH PURITY ALLOYS
C. WATER CHEMISTRY
D. FRACTURE EVALUATION

KWU PROGRAMS

A. TESTING DIFFERENT ALLOYS WITH CERT TESTS IN REACTORS, BOTH BWR & PWR

EPRI PROGRAMS

A. RP 2058-8 LAB TESTS ON IRRADIATED SPECIMENS LOOKING AT MATERIAL PURITY AND MICROSTRUCTURE

B. RP 2181-3 IN PLANT TESTS OF HIGH PURITY ALLOYS

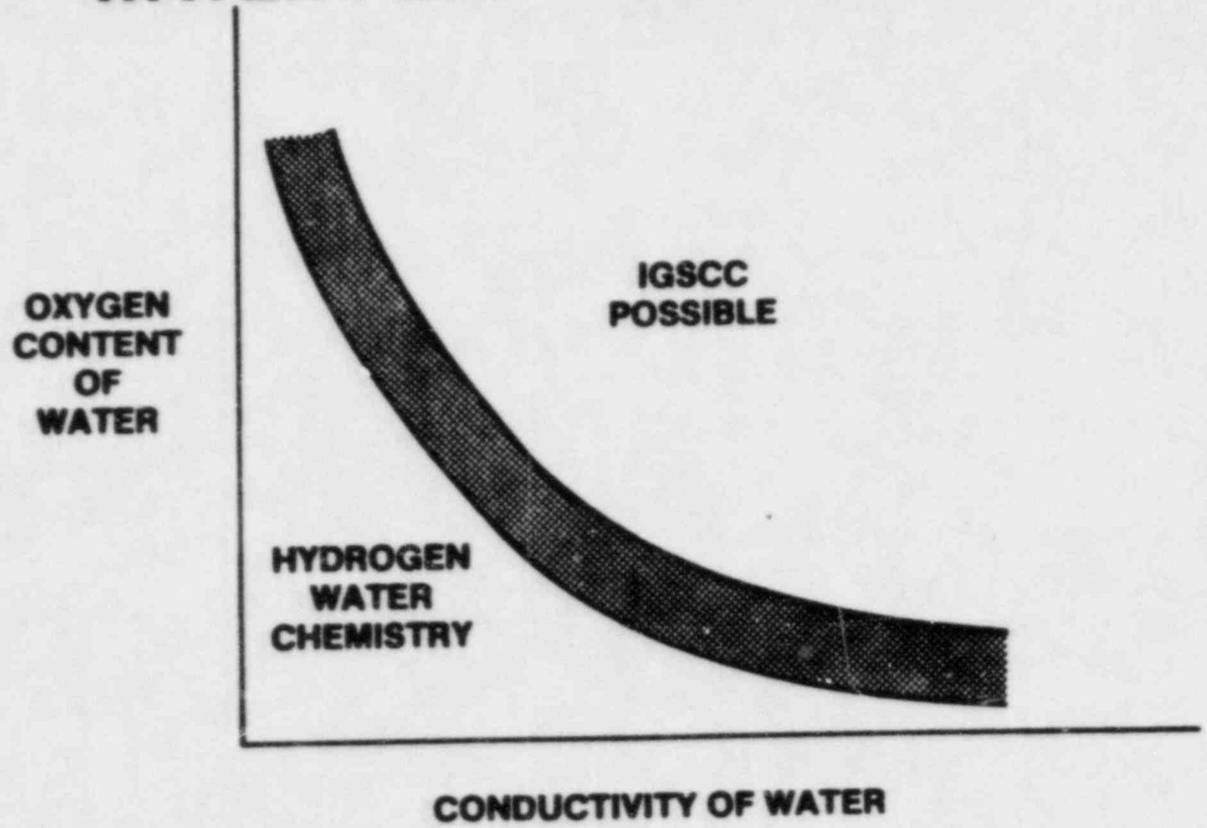
C. RP 1930-XX IN PLANT CHEMISTRY AND ECP MEASUREMENTS TO BETTER DEFINE SERVICE ENVIRONMENT

D. RP 1930-XX DETERMINE EFFECTIVENESS OF H2WC AS A CRACKING REMEDY

E. RP 1628-7 LITERATURE SEARCH OF RELATED WORK

F. RP 2680 MITIGATION OF IASCC IN CORE ENVIRONMENTS

WATER PURITY REQUIREMENTS



HYDROGEN WATER CHEMISTRY REGIME IS A FUNCTION OF BOTH OXYGEN CONTENT AND WATER QUALITY

7/9

- GENERAL ELECTRIC COMPANY -

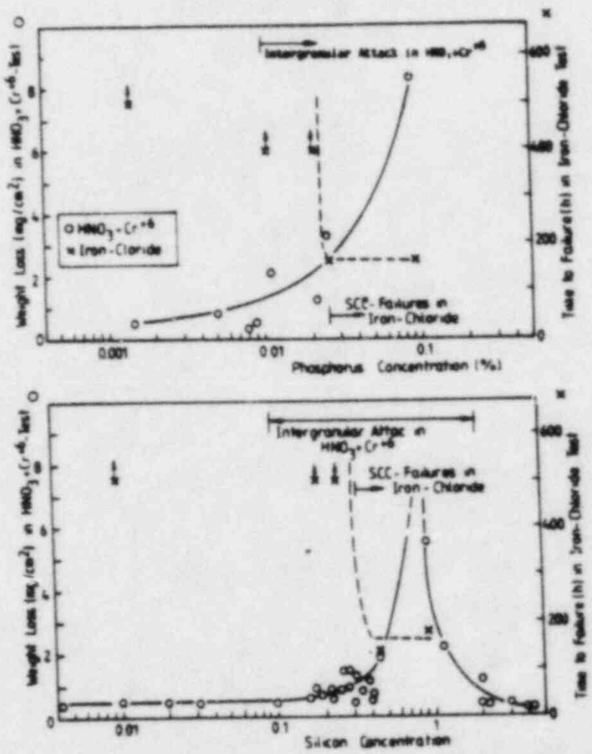


FIGURE 12 - Effect of Iron and Phosphorus on the Corrosion Resistance of Non-Sensitized High Purity Stainless Steels in Boiling $HNO_3 + Cr^{+6}$ and High Pressure Water with Iron Chloride (from /36/ with additional data of /37/)

[redacted]

CONCLUSIONS-CONCLUSIONS-CONCLUSIONS-CONCLUSIONS-CONCLUSIONS-CONCLUSIONS
++

1. IASCC IS PHENOMENA WHICH IS CLOSELY RELATED TO IGSCC OR JUST ANOTHER ASPECT OF IGSCC WHICH IS AMPLIFIED BY THE NEUTRON IRRADIATION.

2. ALTHOUGH IASCC HAS BEEN SEEN IN COMPONENTS OF BOTH TYPES OF REACTORS, BWR'S APPEAR TO BE MORE SUSCEPTABLE TO IT BECAUSE OF ITS UNIQUE ENVIRONMENT.

3. STAINLESS STEEL AND NICKEL BASE ALLOY COMPONENTS APPEAR TO BE THE ONLY MATERIALS EXPERIENCING THIS PHENOMENA

4. H2WC HAS THE POTENTIAL OF BEING ABLE TO MITIGATE IASCC JUST LIKE IGSCC, HOWEVER, THE WHOLE STORY HAS NOT BEEN MADE CLEAR YET AND THE CRITICAL EXPERIMENTS HAVE NOT BEEN PERFORMED. THESE TESTS ARE NOW IN THE FORMULATION STAGE AND CRITICAL EQUIPMENT FOR IN-REACTOR TESTS HAVE NOT BEEN DEVELOPED AND TESTED.

5. COMPONENTS WHICH HAVE EXPERIENCED IASCC TO DATE HAVE BEEN;

A. FUEL CLADDING
B. DRY TUBES
C. CONTROL ROD COMPONENTS

6. COMPONENTS WHICH ARE EXPOSED TO RADIATION LEVELS WHICH HAVE BEEN KNOWN TO CAUSE IASCC,BUT WHICH HAVE NOT YET EXPERIENCED IT, INCLUDE;

A. SHROUD
B. TOP GUIDE

7. SAFETY ISSUES HAVE BEEN ADDRESSED BY THE MANUFACTURERS, EPRI AND THE UTILITIES AND SO FAR NO ONE HAS BEEN ABLE TO IDENTIFY A PROBLEM WHICH HAS ANY SIGNIFICANCE. IN ADDITION, THE NRC HAS BEEN MADE AWARE OF THE ISSUES AND HAS NOT CONSIDERED THE PHENOMENA A PROBLEM AS YET.

XXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXXX

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REVIEW OF

MATERIALS ENGINEERING BRANCH

BUDGET AND RESEARCH PROGRAM

-

FOREIGN INTERACTIONS

ACRS METAL COMPONENTS SUBCOMMITTEE

SEPTEMBER 4-5, 1985

C. Z. SERPAN, JR.

PRIMARY SYSTEM INTEGRITY

	CURRENT BUDGET FY 1985	BUDGET REQUEST FY 1986	FY 1987
	($ IN THOUSANDS)		
REACTOR VESSELS	9010	9150	8950
- FRACTURE MECHANICS VALIDATION (HSST)	5820	5130	5230
- IRRADIATION AND TESTS	2250	2800	2600
- SURVEILLANCE DOSIMETRY	940	1220	620
- PLANT LIFE EXTENSION	--	--	500
PIPING STEAM GENERATORS	5684	6450	5250
- INTEGRITY OF PIPING	4079	5100	4500
- STEAM GENERATOR INTEGRITY	1605	1350	750
NONDESTRUCTIVE EXAMINATION	2956	2300	3800
- UT, A/E OF PIPING AND VESSELS	2856	2150	2600
- EDDY CURRENT INSPECTION OF S.G. TUBING	100	150	200
- PLANT LIFE EXTENSION	--	--	1000
TOTAL PRIMARY SYSTEM INTEGRITY	17650	17200	18000

MATERIALS ENGINEERING BRANCH

KEY SAFETY ISSUES

- o AN UNDERSTANDING OF THE PROPERTIES AND BEHAVIOR OF REACTOR MATERIALS DEGRADED BY AGING IS NEEDED FOR LICENSING DECISIONS (FY 1989).
- o LIMITS OF ACCEPTABILITY MUST BE ESTABLISHED FOR OPERATING REACTOR SERVICE OF CRACKED AND DEGRADED PIPING (1987).
- o LIMITS OF ACCEPTABILITY MUST BE ESTABLISHED FOR OPERATING REACTOR SERVICE OF CRACKED AND DEGRADED STEAM GENERATOR TUBING (FY 1986).
- o METHODS FOR DETECTION AND SIZING OF FLAWS IN VESSELS, PIPING, AND STEAM GENERATOR TUBING MUST BE VALIDATED AND IN SOME CASES IMPROVED (FY 1988).

PRIMARY SYSTEM INTEGRITY

REACTOR VESSELS	FY 1985	FY 1986	FY 1987
	$9.0M	$9.1M	$8.9M

SAFETY PROBLEM

AGING DEGRADATION OF VESSEL AND PRIMARY SYSTEM

- o IRRADIATION EMBRITTLEMENT
- o DEGRADATION OF FRACTURE TOUGHNESS
- o CRACK GROWTH
- o PREDICTION/EVALUATION METHODS

REGULATORY USE

- o TO EVALUATE THE ACCURACY AND FEASIBILITY OF APPLICANT'S SUBMITTALS ON PRESSURIZED THERMAL SHOCK

- o TO PROVIDE VALIDATION OF THE PTS RULE AND SCREENING CRITERION TO EVALUATE SUBMITTALS ON REVISION OF PRESSURE-TEMPERATURE CURVES FOR STARTUP AND SHUTDOWN

- o TO EVALUATE SUBMITTALS ON VESSEL FLUX REDUCTION THROUGH ADOPTION OF LOW LEAKAGE CORES

- o TO PROVIDE BASIS FOR NRR DECISIONS ON OPERATION OF REACTORS AND COMPONENTS IN THE PRESENCE OF CRACKS

4

PRIMARY SYSTEM INTEGRITY

PIPING - STEAM GENERATORS	FY 1985	FY 1986	FY 1987
	\$5.7M	\$6.4M	\$5.2M

SAFETY PROBLEM

- INTEGRITY OF CRACKED AND DEGRADED PIPING
- INTEGRITY OF CRACKED AND DEGRADED STEAM GENERATOR TUBING
- CORROSION AND CRACKING BY THE COOLANT ENVIRONMENT

REGULATORY USE

- FORM THE BASIS FOR NRC EVALUATION OF THE ASME IWB-3640 RULES FOR EVALUATION AND ACCEPTANCE OF FLAWS IN BWR PIPING
- EXPERIMENTAL BASIS FOR FORMULATION AND ACCEPTANCE OF THE RULE ON LEAK-BEFORE-BREAK IN PIPING
- FORM THE BASIS FOR NRC ACCEPTANCE OF FIXES (SUCH AS WELD CLAD OVERLAY) AND REPLACEMENT MATERIALS (SUCH AS 316NG) FOR LONG-TERM OPERATION IN BWRs
- FORM THE BASIS FOR UPGRADED STEAM GENERATOR TUBE PLUGGING CRITERIA AND FOR IMPROVED IN-SERVICE INSPECTION PLANS

PRIMARY SYSTEM INTEGRITY

	FY 1985	FY 1986	FY 1987
	$3.0M	$2.3M	$3.8M

NONDESTRUCTIVE EXAMINATION

SAFETY PROBLEM

- ACCURATE DETECTION AND SIZING OF FLAWS IN CARBON STEEL VESSELS AND NOZZLES, IN CAST STAINLESS STEEL ELBOWS AND IN BIMETALLIC JOINTS

REGULATORY USE

- FORM THE BASIS FOR RECOMMENDATIONS FOR UPGRADED ASME CODE OR REG. GUIDE REQUIREMENTS FOR INSPECTIONS
- TO EVALUATE QUALIFICATION OF PERSONNEL, PROCEDURES AND EQUIPMENT FOR UT OF VESSELS AND PIPING
- AS A BASIS TO REVIEW APPLICANT'S PROCEDURES FOR PLANT INSPECTIONS, AND TO JUDGE ACCEPTABILITY OF RESULTS
- TO PROVIDE INDEPENDENT THIRD PARTY CHECKS OF FLAW DETECTION AND SIZING

MEBR FOREIGN INTERATIONS

EXCHANGE AGREEMENTS - ACTIVE

- o FRG - MATERIALS, HDR, RES. ENGINEER
- o U.K. - NDE, DOSIMETRY, FRACTURE MECHANICS
- o SWITZERLAND - FRACTURE MECHANICS, MATERIALS, RPV, PIPING
- o BELGIUM - DOSIMETRY, ANNEALING

o STEAM GENERATOR GROUP PROJECT

STUDY RETIRED STEAM GENERATOR -
FRANCE, ITALY, JAPAN CONTRIBUTE FUNDS

o ICCGR COOPERATIVE PROGRAM

INTERNATIONAL CYCLIC CRACK GROWTH RATE -
USA, EUROPE, JAPAN POOL RESEARCH, COORDINATE PROGRAMS

o GUNDREMMINGEN

EVALUATE IN SITU IRRADIATED PV STEEL

FRG - MPA STUTTGART MACHINES AND TESTS OF RPV STEEL

UK - D_2O IRRADIATIONS OF ARCHIVE STEEL.
RADIATION DAMAGE ANALYSIS

o CSNI

PWG-3 - FRACTURE MECHANICS, STRESS ANALYSIS, NDE

PISC-III - EVALUATE ADVANCED NDE METHODS
STUDY NDE ON REAL FLAWS IN REAL ENVIRONMENTS

-7-

MEBR FOREIGN INTERACTIONS

o DOSIMETRY

COOPERATIVE PROGRAM IN NEUTRON DOSIMETRY

BELGIUM - VENUS, TRANSPORT CALCULATIONS

UK - NESDIP, RADIATION DAMAGE ANALYSIS

FRG - RADIATION DAMAGE AND SPECTRUM ANALYSIS, GAMMA HEATING

ASTM/EURATOM DOSIMETRY SYMPOSIA

o PIPING IPIRG

PIPING TESTS UNDER EXTREME CONDITIONS UNDER FORMATION NOW.

(4)

DRAFT REGULATORY GUIDE 1.99, REVISION 2
RADIATION DAMAGE TO REACTOR VESSEL MATERIALS

PURPOSE: GET REVISION 2 OUT FOR PUBLIC COMMENT

I. SAFETY SIGNIFICANCE

- o NORMAL OPERATION
- o TRANSIENTS
- o FUNCTION OF R.G. 1.99 AND APPENDICES G & H IN REGULATING FRACTURE PREVENTION OF R.V.

II. COMPARISON OF REVISION 2 TO REVISION 1

- o BETTER DATA BASE
- o BETTER ANALYSIS
- o ACCEPTANCE BY TECHNICAL COMMUNITY

III. TECHNICAL BASIS FOR REVISION 2

- o AMALGAM OF NRC AND EPRI SPONSORED RESEARCH
- o PLOTS OF RESIDUALS DEMONSTRATE DATA FIT
- o BASIS FOR ATTENUATION FORMULA

IV. VALUE/IMPACT

- o WHAT PLANTS ARE AFFECTED AND HOW MUCH?
- o PWRS-SAFETY IMPACT
  - o TRANSIENTS
  - o NORMAL OPERATION -- PNL ANALYSIS
- o COSTS -- HEATUP
- o LTOP
- o BWRS -- HYDROTEST

V. RELATIONSHIP TO PTS RULE

VI. IMPLEMENTATION - CRGR REVIEW NOT COMPLETE

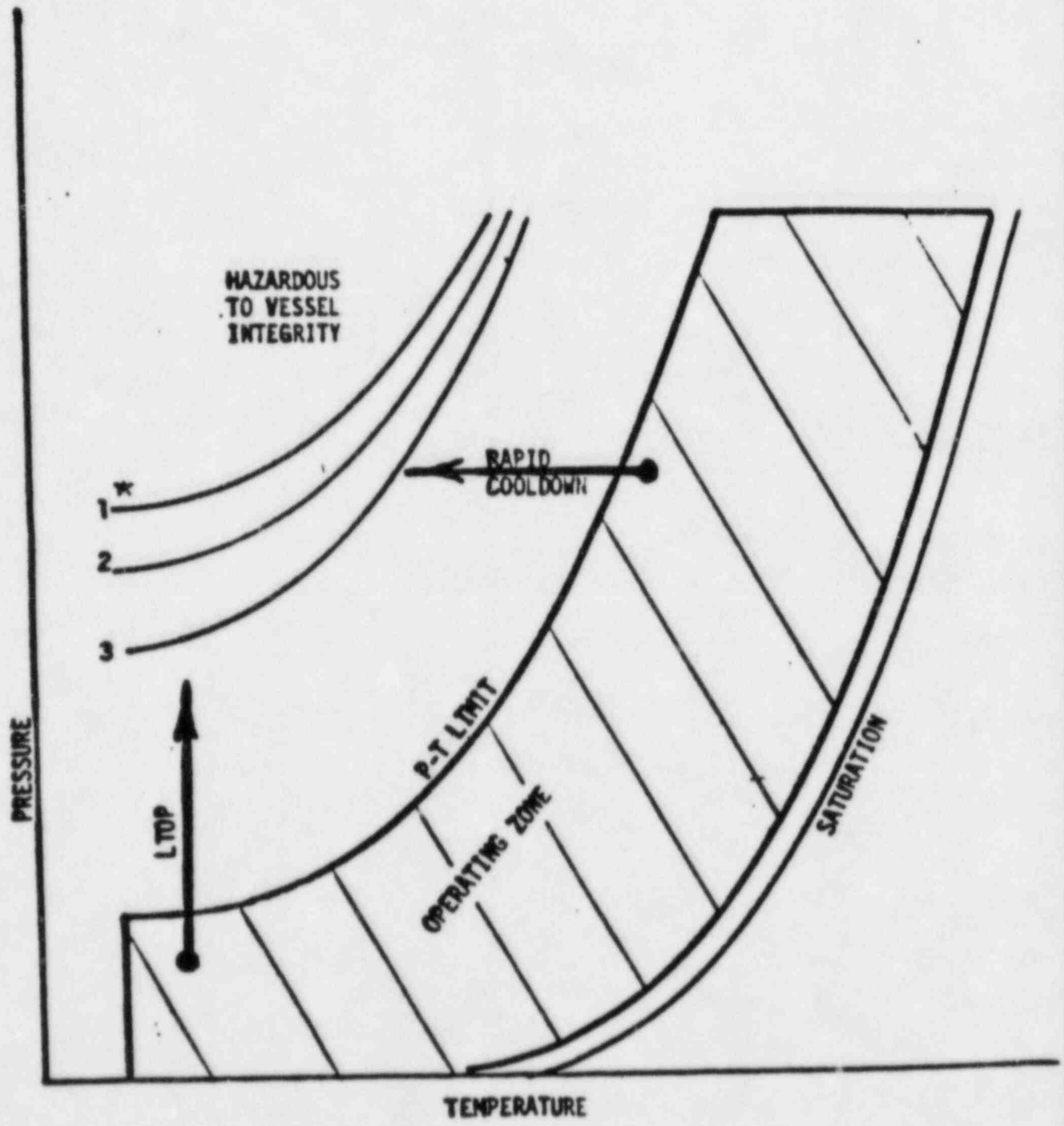


FIGURE 1 SCHEMATIC PRESSURE-TEMPERATURE DIAGRAM

*Note: The three curves represent different cooling rates, Curve 1 being the slowest.

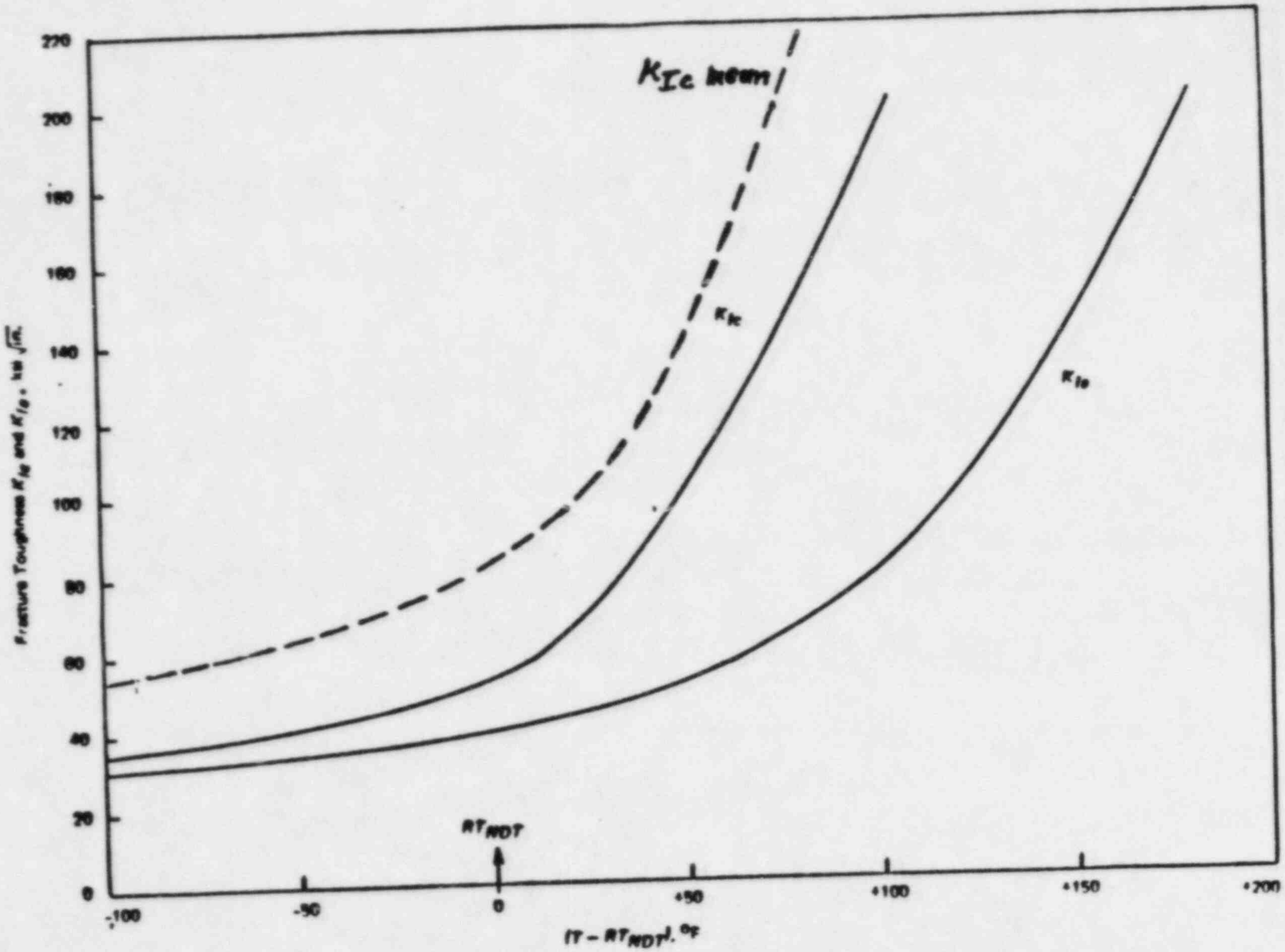


FIG. A-4200-1 LOWER BOUND K_{Ia} AND K_{Ic} TEST DATA FOR SA-533 GRADE B CLASS 1, SA-508 CLASS 2, AND SA-508 CLASS 3 STEELS

FIGURE 5 REFERENCE TOUGHNESS CURVES FROM SECTION XI OF THE ASME CODE. THE K_{Ia} CURVE IS THE SAME AS THE K_{IR} CURVE FROM SECTION III, APPENDIX G.

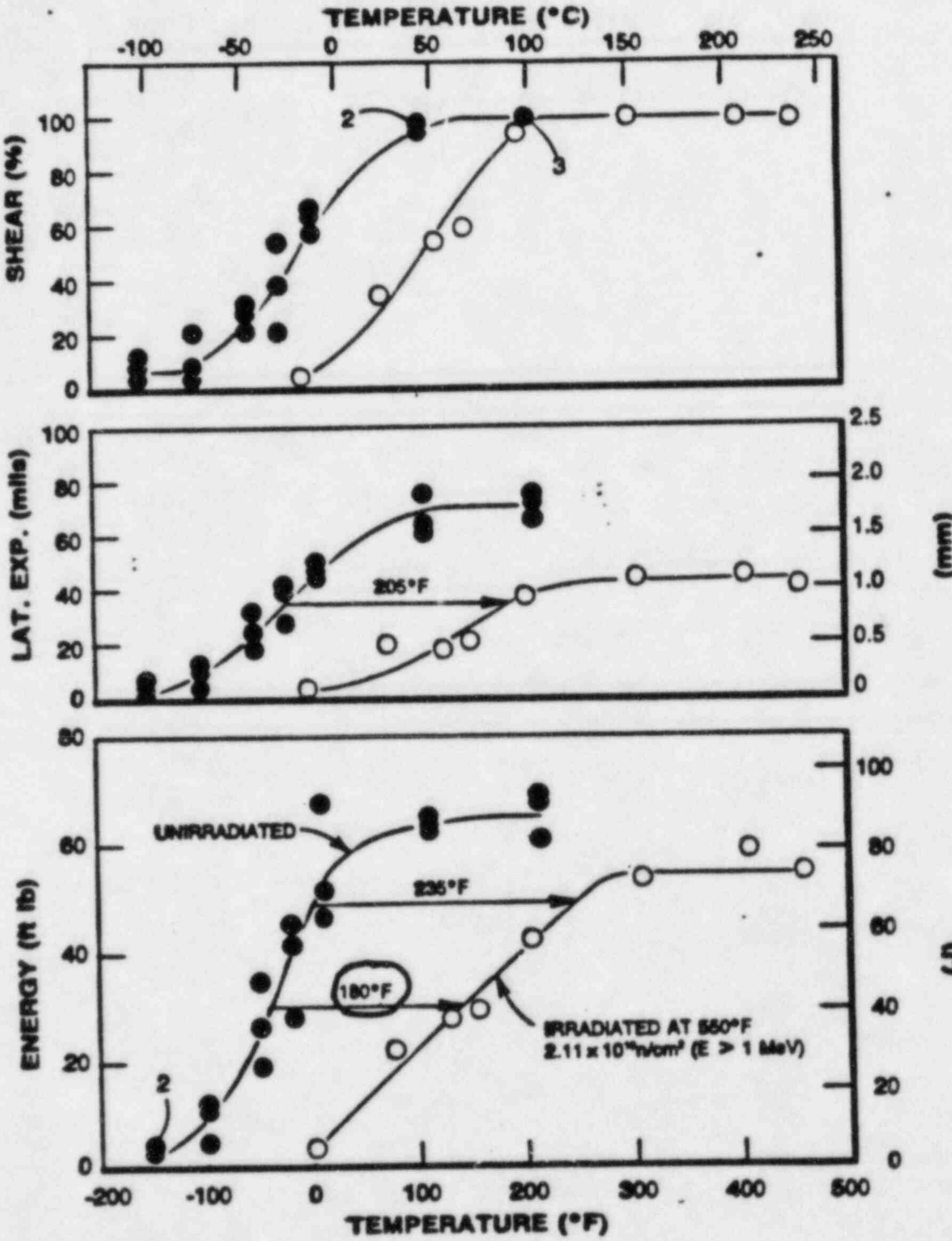


FIGURE 5-3. CHARPY V-NOTCH IMPACT ENERGY FOR THE POINT BEACH UNIT NO. 1 PRESSURE VESSEL WELD METAL

5

$$\Delta RT_{NDT} = (CF)\,(F)^{0.28 - 0.1 \log F}$$

CF = Chemistry Factor (Copper And Nickel)

- o Table I For Welds
- o Table II For Base Metal (Plates and Forgings)

F = Fluence, n/cm^2 (E>1MeV)

- o Figure 1

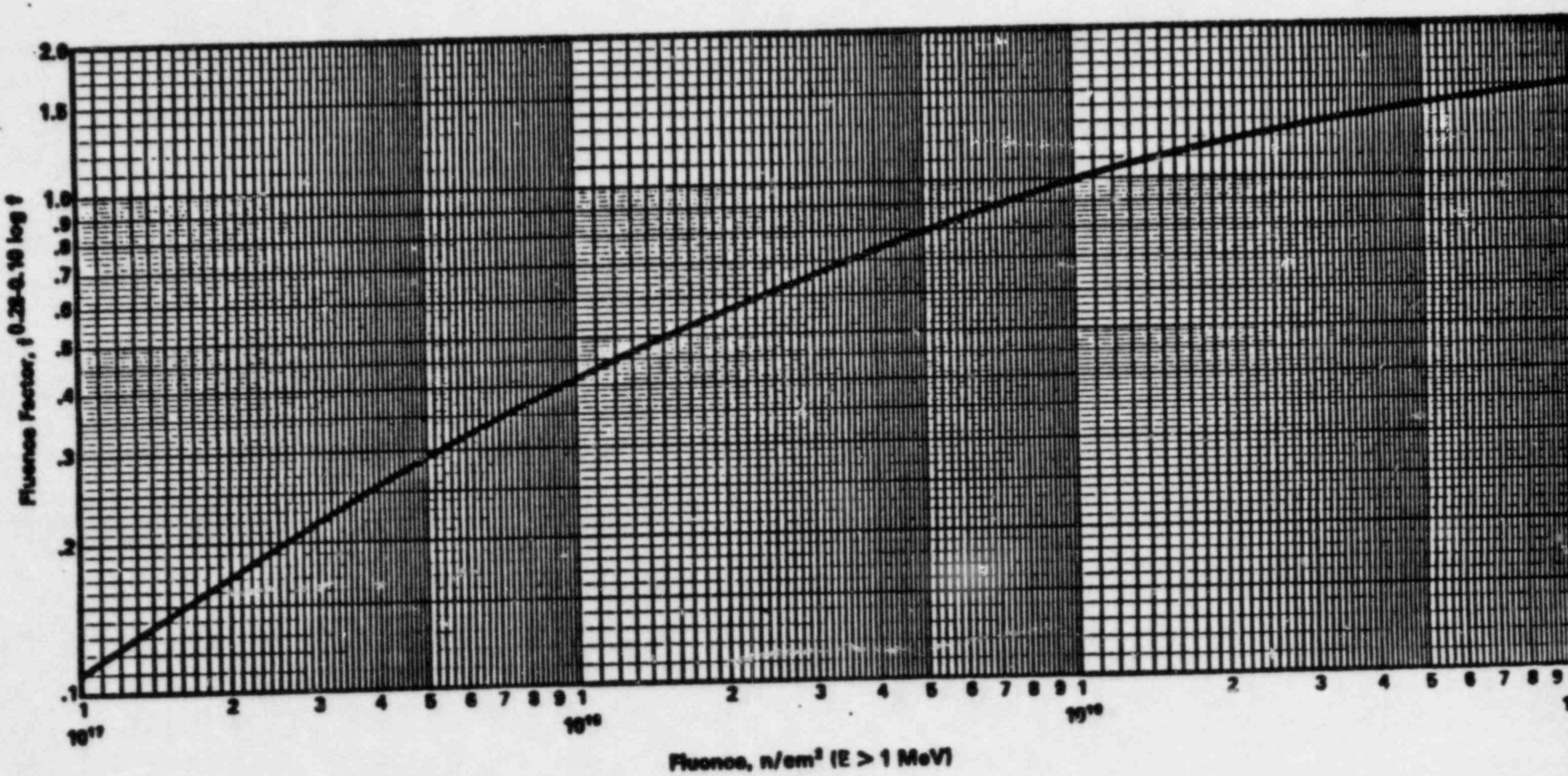


FIGURE 1 FLUENCE FACTOR FOR USE IN EQUATION 2, THE EXPRESSION FOR ΔRT_{NDT}

RG 1.99 Rev.2

Improvements In Revision 2 Over Revision 1

- Surveillance Data Base - 8 Years Accumulation
- Regression Analysis Gave Mean Curves And A Measure Of Scatter
- Chemistry Factor Includes Copper and Nickel
- Separate Factors For Welds And Base Metal
- Amalgam Of NRC- And EPRI-Sponsored Research
- Adoption Expected For ASTM E900 And ASME Code, Section XI
- Attenuation Equation , MARGIN
- Use Of Plant Surveillance Data

GUTHRIE

- POWER REACTOR SURVEILLANCE DATA ONLY
- CHARPY 30 FT LB SHIFT FROM SURVEILLANCE REPORTS
- FLUENCE FROM HEDL ANALYSES

ODETTE

- POWER REACTOR SURVEILLANCE DATA ONLY WITH SOME ADDITIONAL BWR DATA
- CHARPY 30 FT LB SHIFT FROM REFITTED CHARPY DATA, USING TANH FUNCTION
- FLUENCE FROM SURVEILLANCE REPORTS

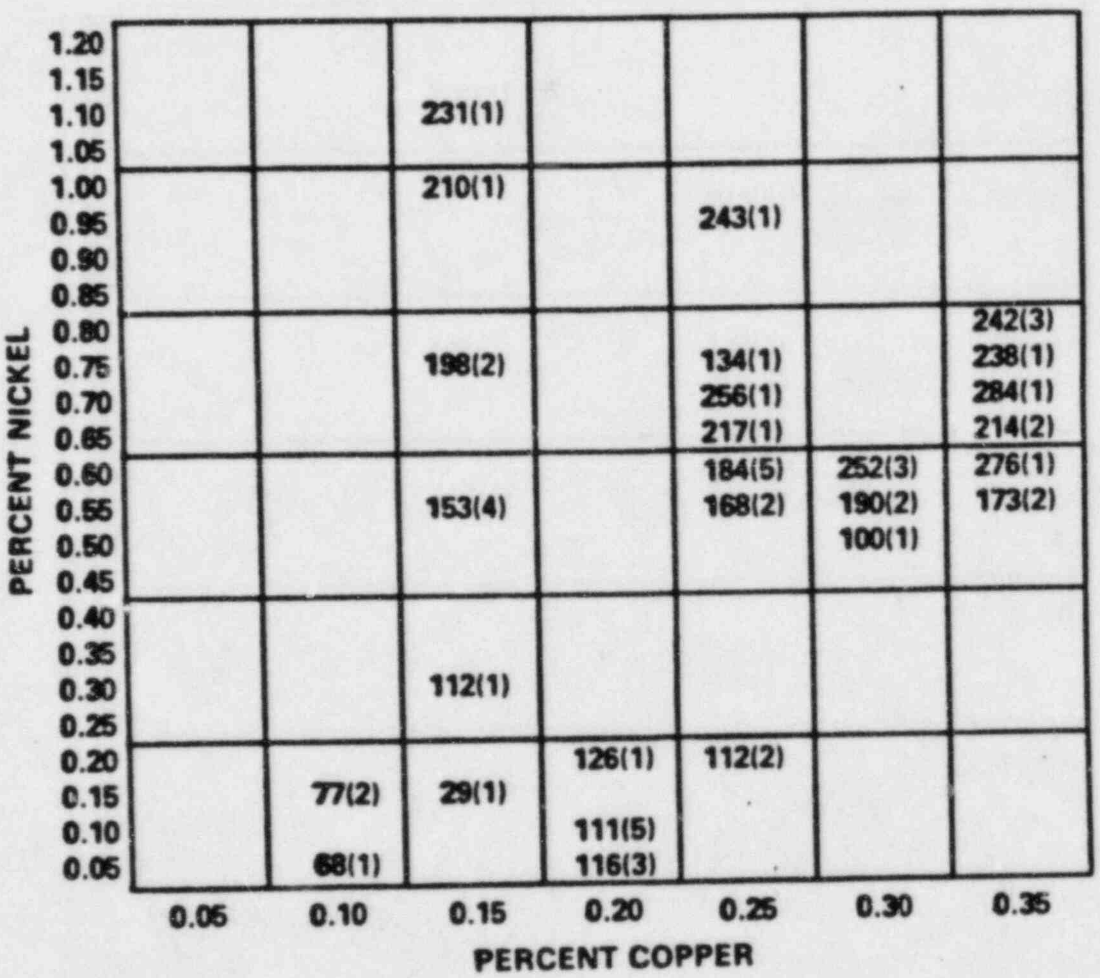


FIG. 1 DISTRIBUTION OF GUTHRIE'S WELD DATA BASE IN TERMS OF COPPER AND NICKEL CONTENT

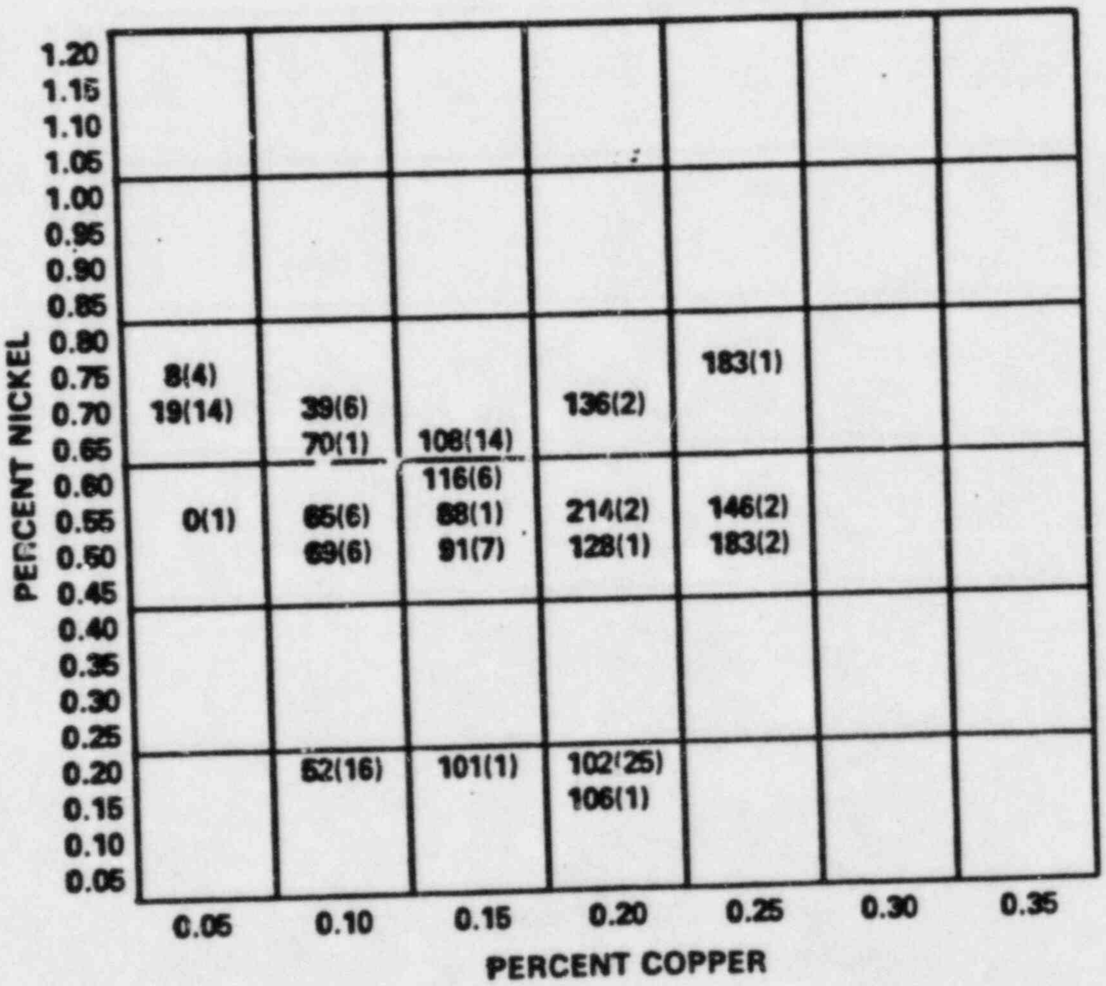


FIG. 2 DISTRIBUTION OF GUTHRIE'S BASE METAL DATA BASE IN TERMS OF COPPER AND NICKEL CONTENT

CONCLUSIONS BY GUTHRIE AND ODETTE

(1) SEPARATE CORRELATIONS ARE NEEDED FOR WELDS AND BASE METAL

(2) THE EXPRESSION SHOULD BE THE PRODUCT OF A CHEMISTRY FACTOR AND A FLUENCE FACTOR

(3) THE ELEMENTS IN THE CHEMISTRY FACTOR SHOULD BE COPPER AND NICKEL

(4) THE FLUENCE FACTOR SHOULD PROVIDE A TREND CURVE SLOPE, WHEN PLOTTED ON LOG-LOG PAPER OF ABOUT 0.25 TO 0.30 AT 10^{19} N/CM^2, AND IT SHOULD BE STEEPER AT LOWER FLUENCES AND FLATTER AT HIGHER FLUENCES

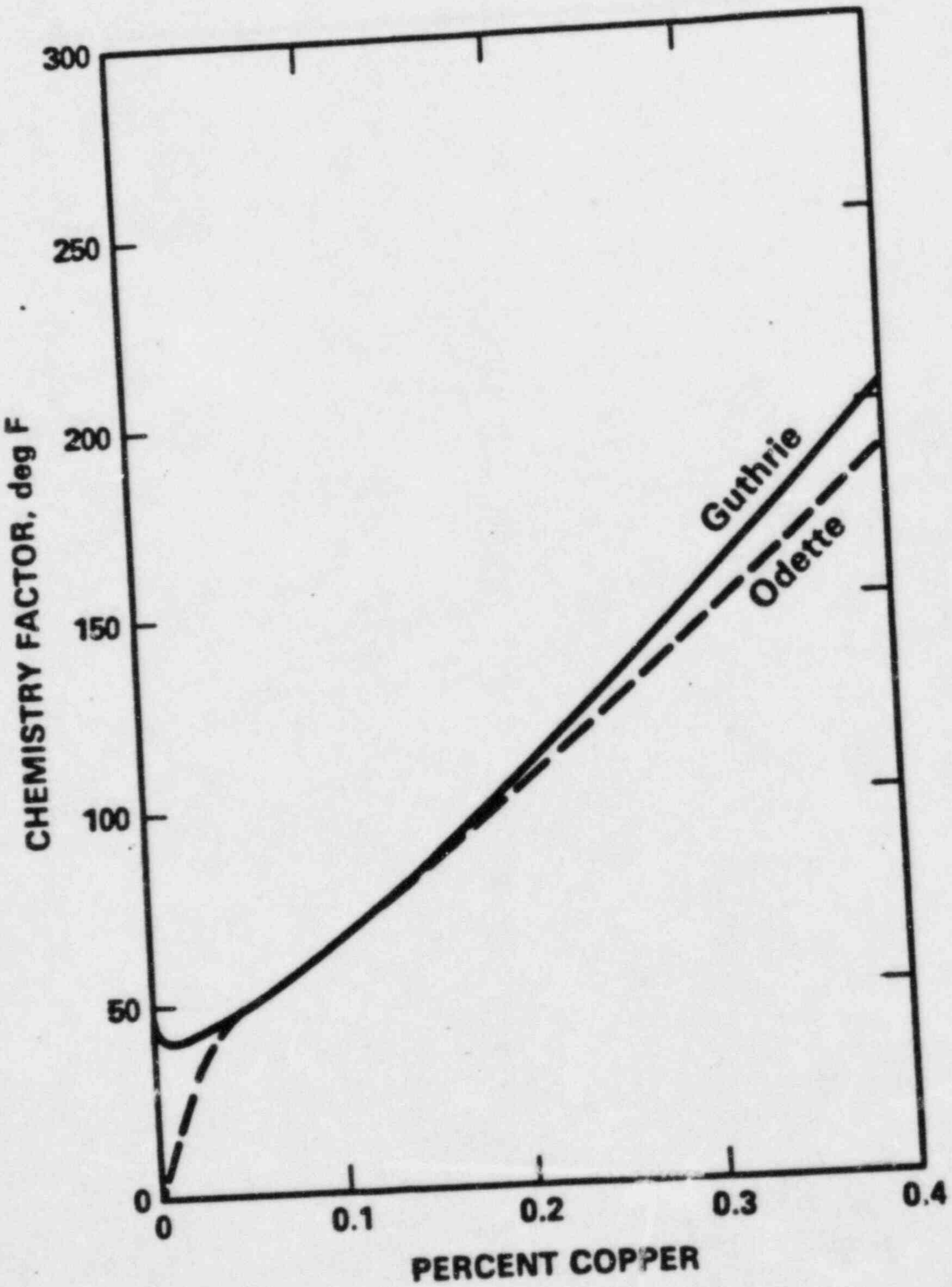


FIG. 3 COMPARISON OF GUTHRIE AND ODETTE STUDIES ON THE EFFECT OF COPPER ON THE CHEMISTRY FACTOR -- WELDS WITH 0.2 PERCENT NICKEL

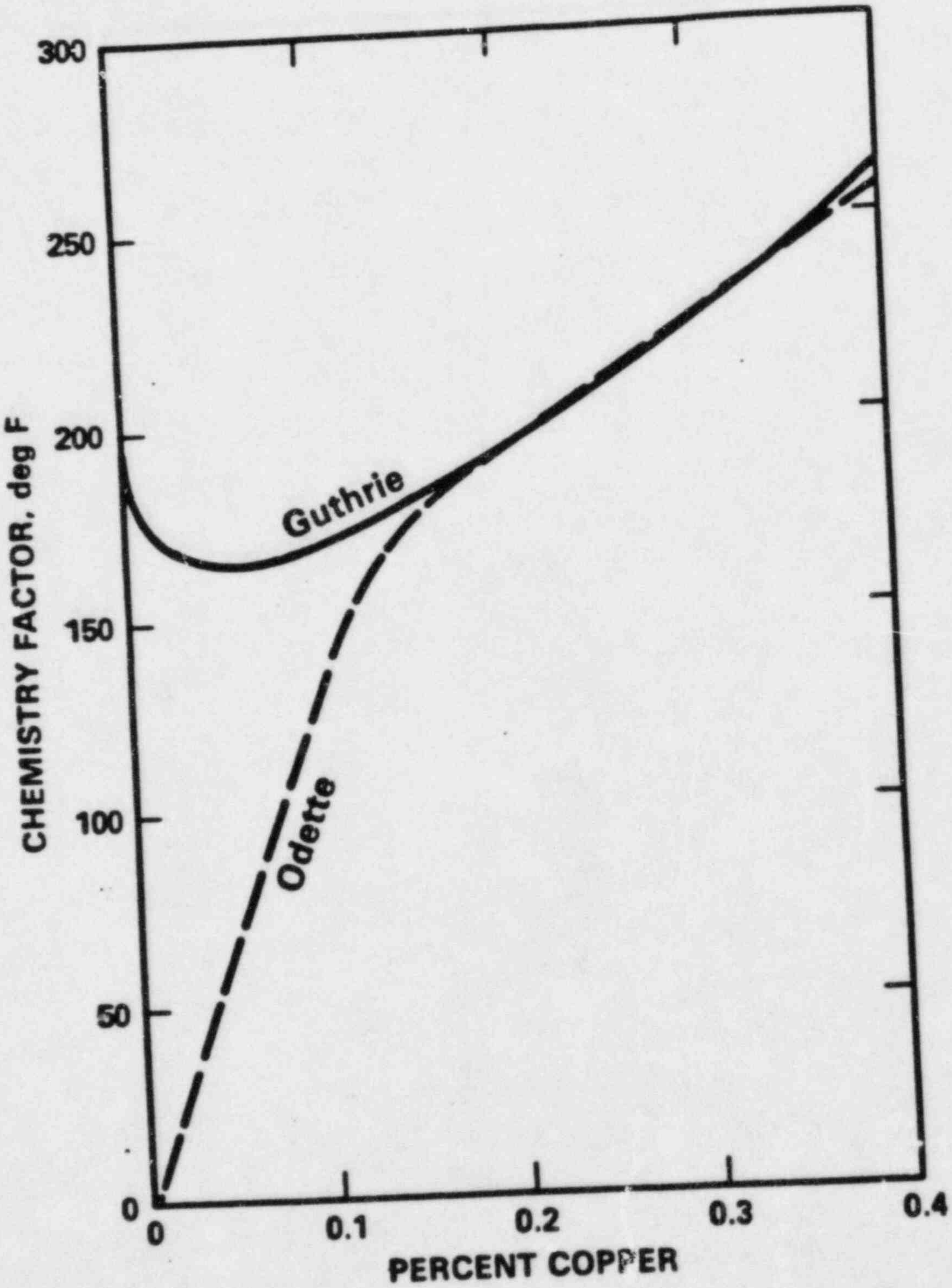


FIG. 4 COMPARISON OF GUTHRIE AND ODETTE STUDIES ON THE EFFECT OF COPPER ON THE CHEMISTRY FACTOR -- WELDS WITH 0.8 PERCENT NICKEL

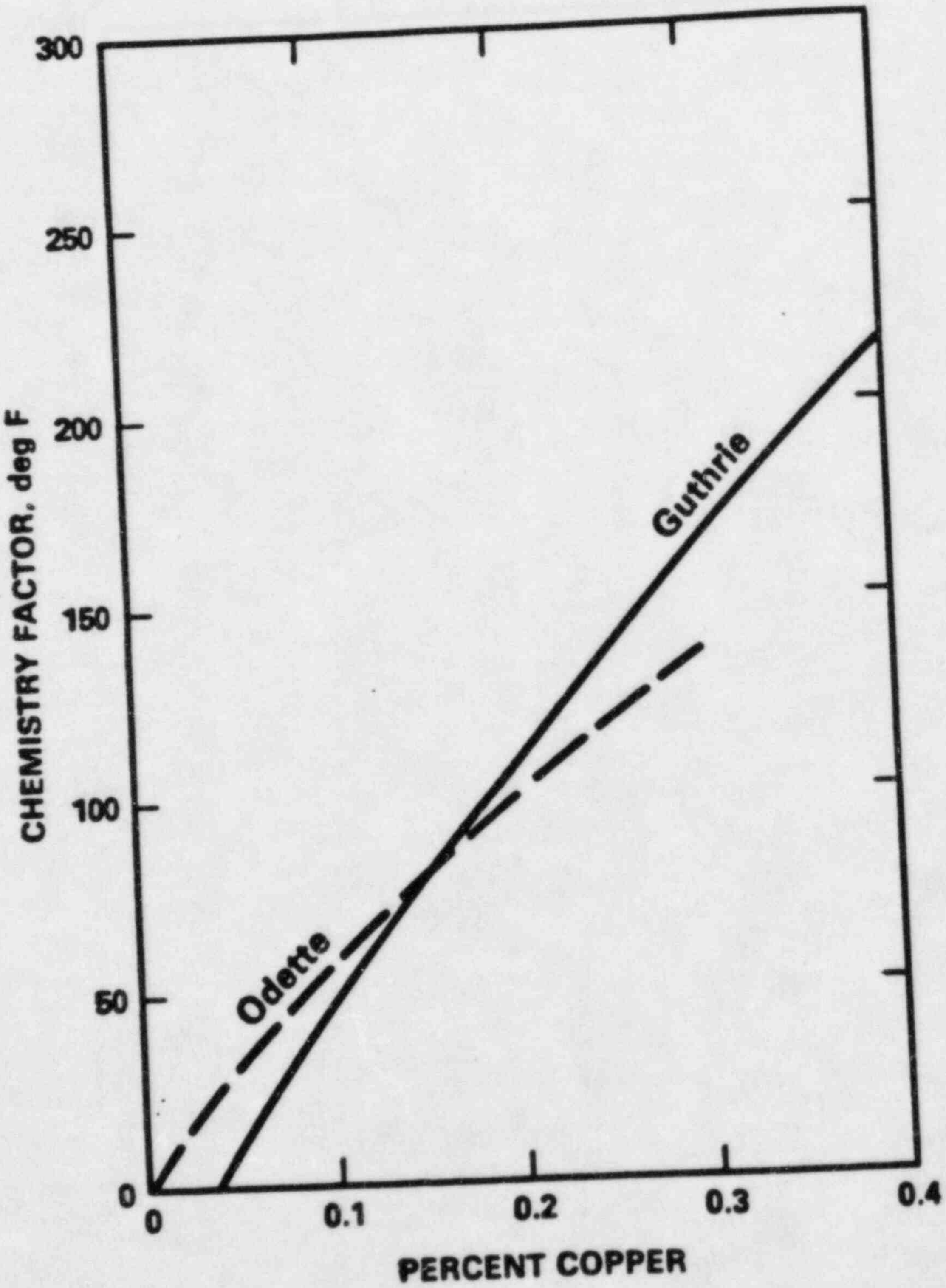


FIG. 5 COMPARISON OF GUTHRIE AND ODETTE STUDIES ON THE EFFECT OF COPPER ON THE CHEMISTRY FACTOR -- BASE METAL WITH 0.2 PERCENT NICKEL

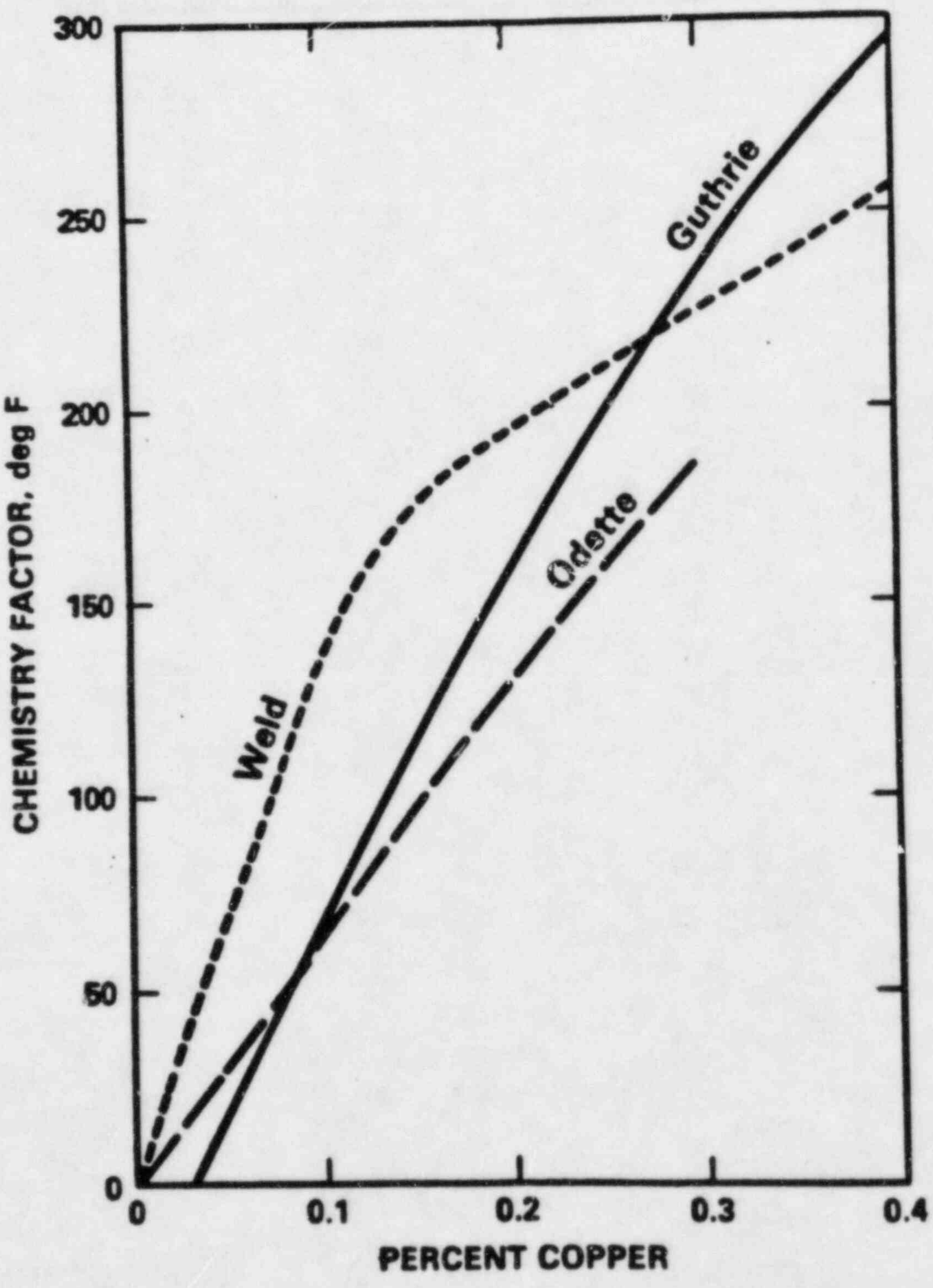


FIG. 6 COMPARISON OF GUTHRIE AND ODETTE STUDIES ON THE EFFECT OF COPPER ON THE CHEMISTRY FACTOR -- BASE METAL WITH 0.8 PERCENT NICKEL

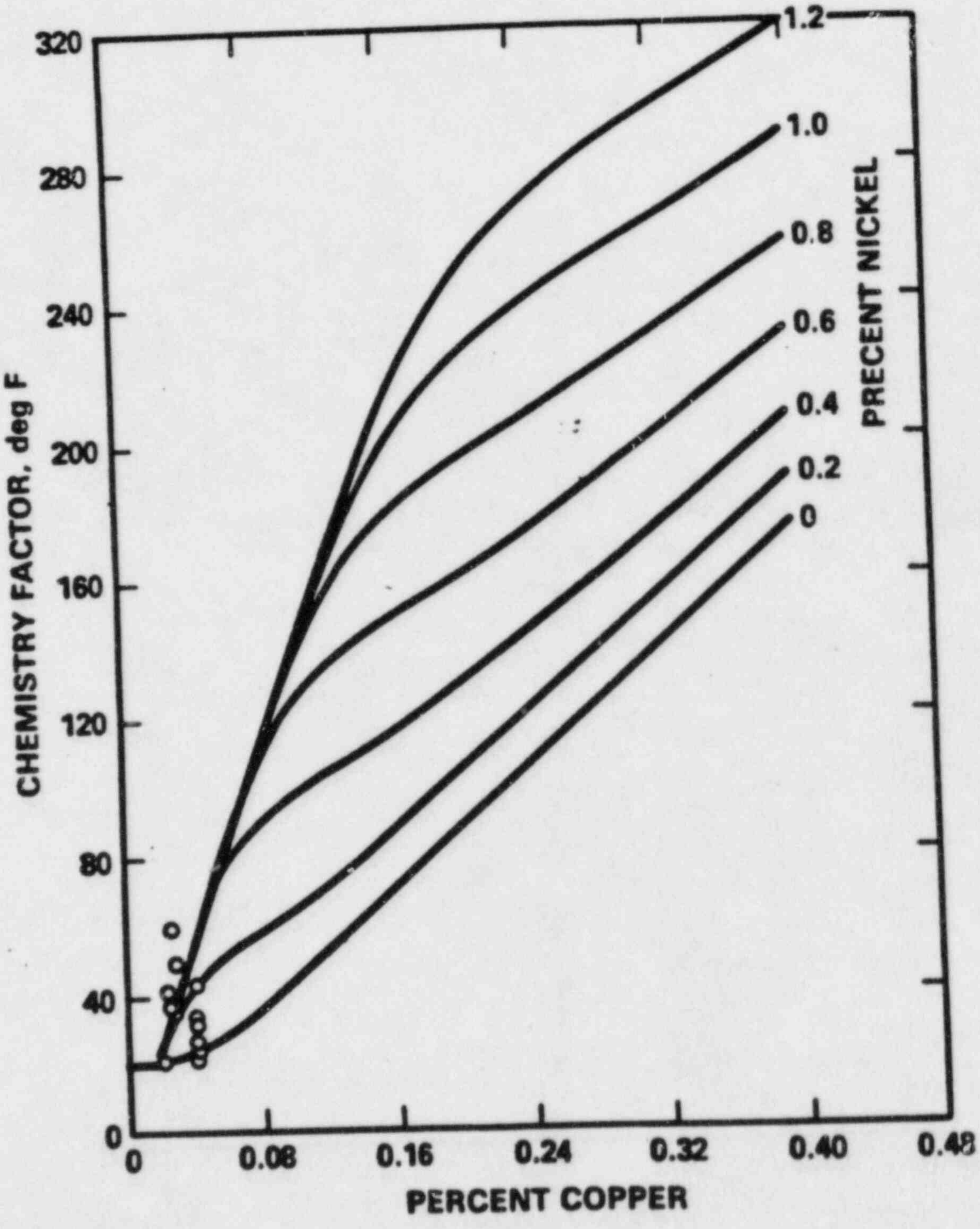


FIG. 7 TEST REACTOR DATA FOR LOW COPPER WELDS, NORMALIZED TO 10^{19} n/cm^2, COMPARED TO THE CURVES FOR CHEMISTRY FACTOR VERSUS COPPER FROM REVISION 2

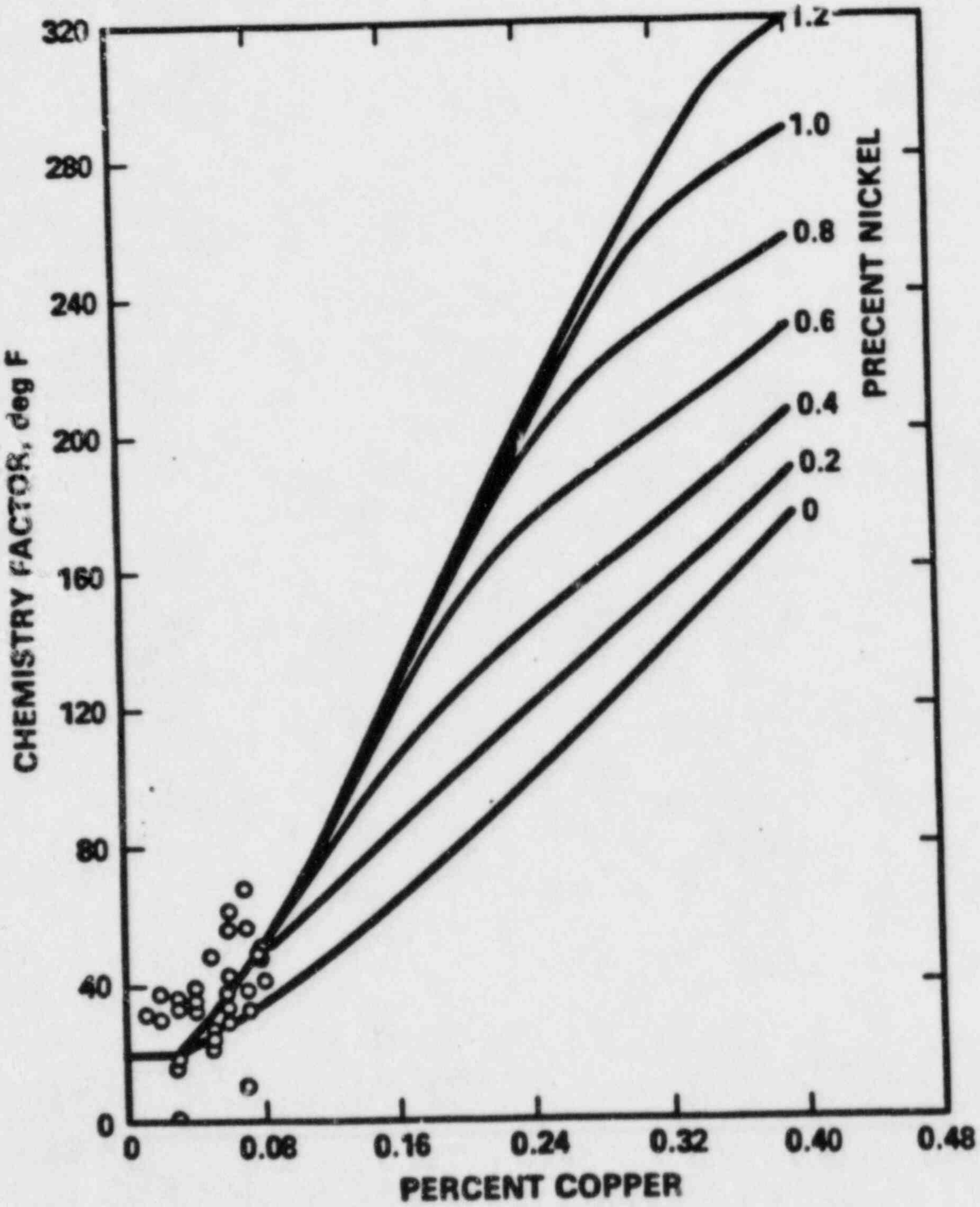


FIG. 8 TEST REACTOR DATA FOR LOW COPPER BASE METAL, NORMALIZED TO $10^{19} n/cm^2$, COMPARED TO THE CURVES FOR CHEMISTRY FACTOR VERSUS COPPER FROM REVISION 2

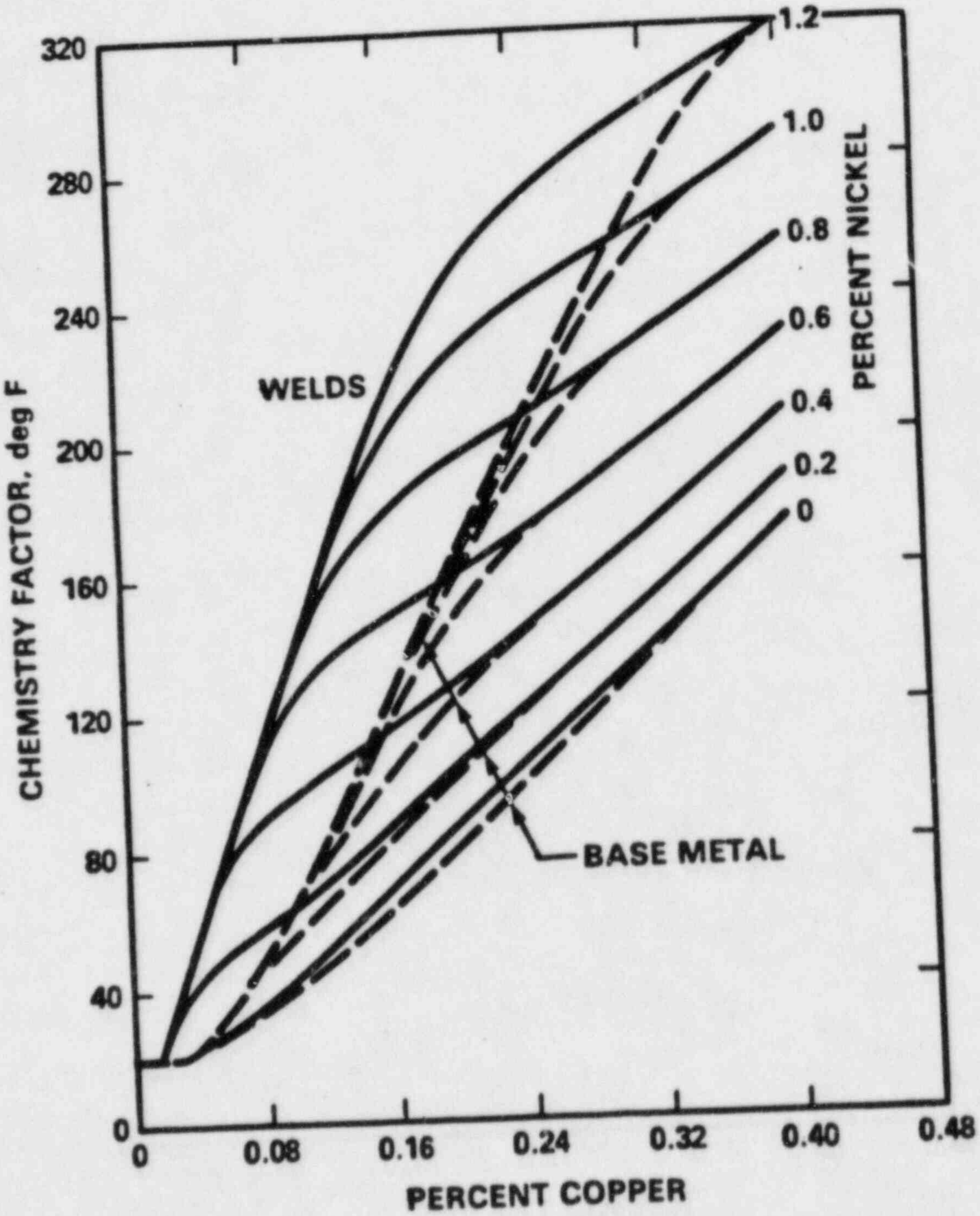


FIG. 9 COMPARISON OF CHEMISTRY FACTORS FOR WELDS AND BASE METAL, GIVEN IN REVISION 2

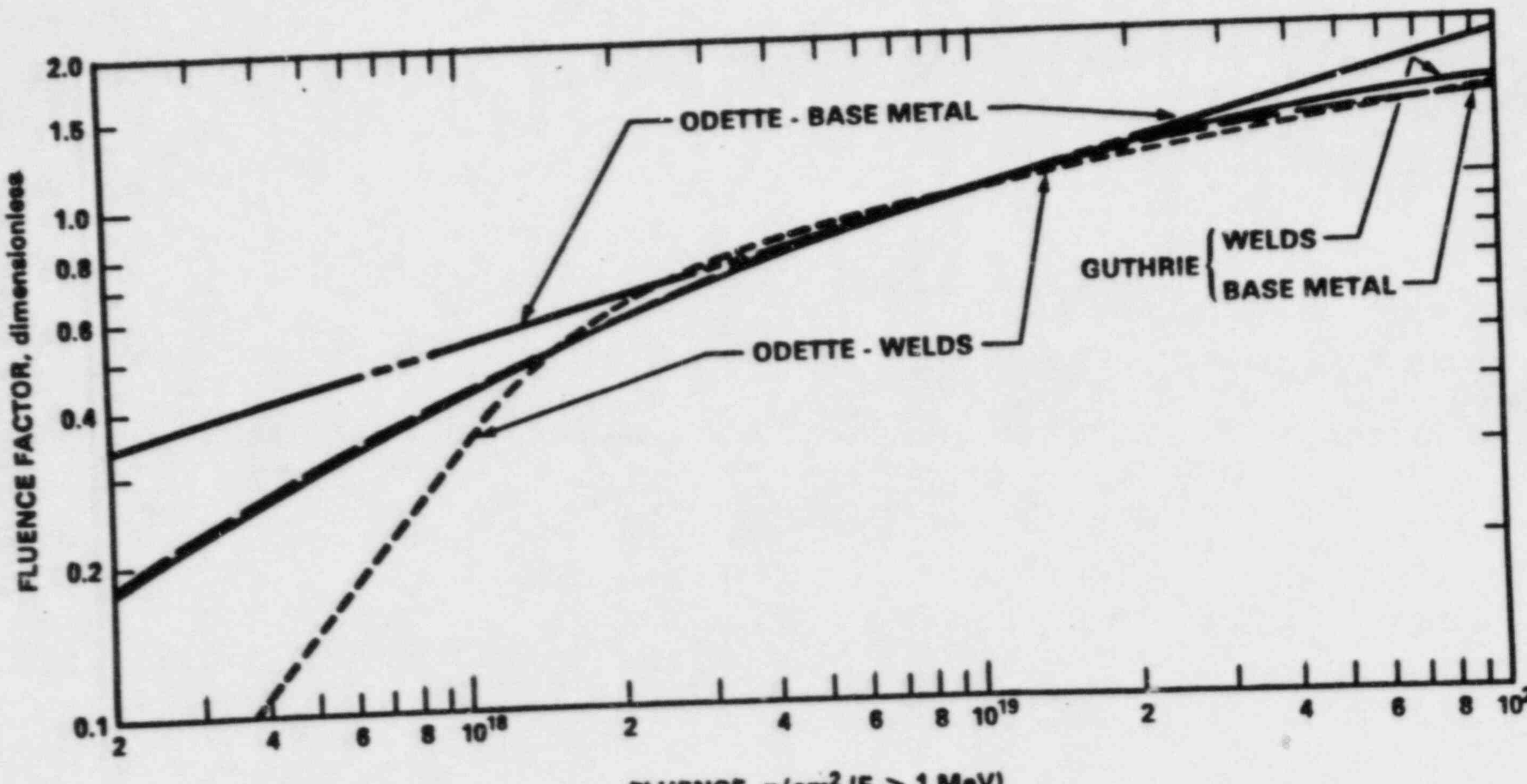


FIG. 10 COMPARISON OF FLUENCE FACTORS

20

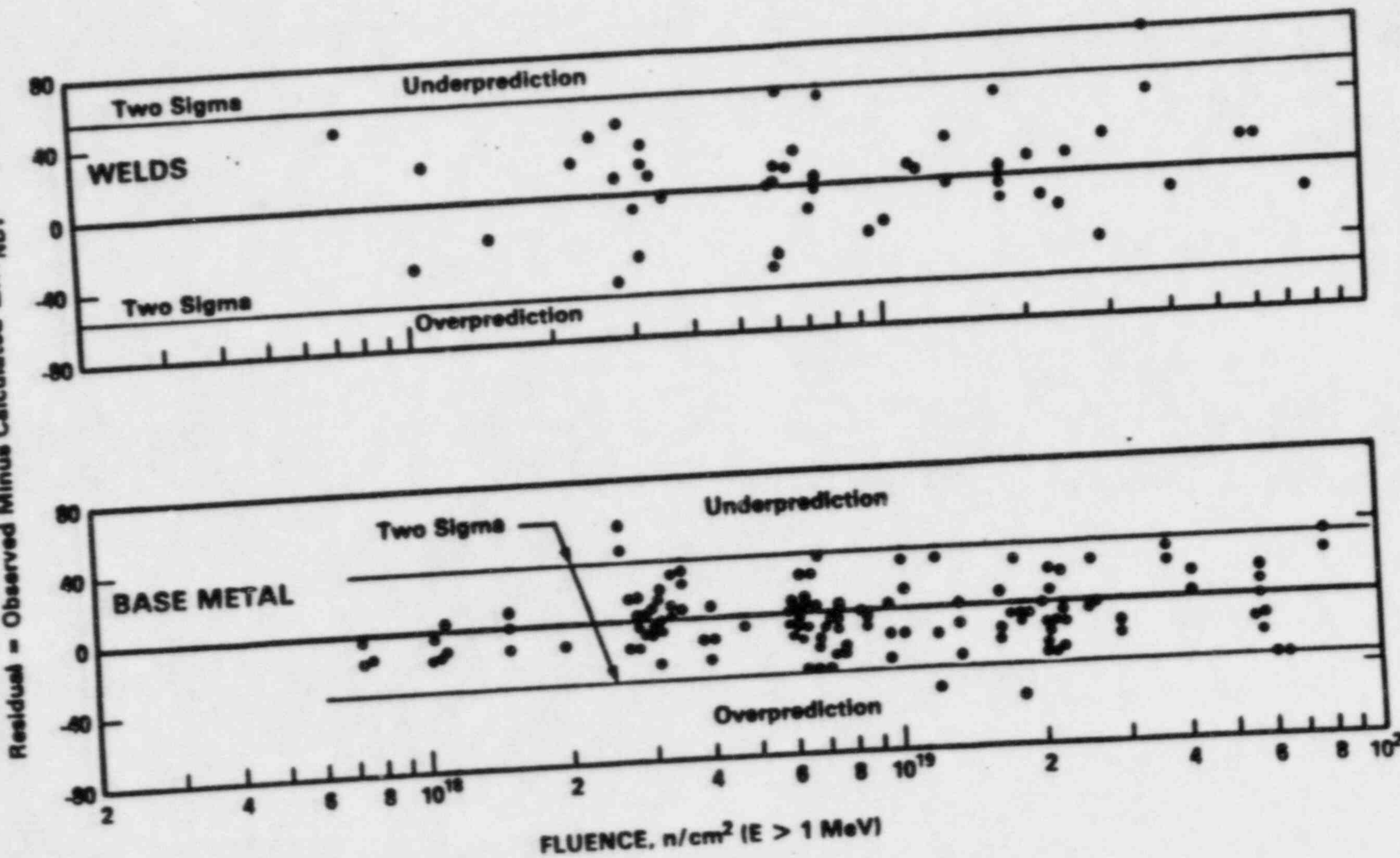


FIG. 11 PLOTS OF RESIDUALS VERSUS FLUENCE FOR 51 WELD AND 126 BASE METAL DATA POINTS

21

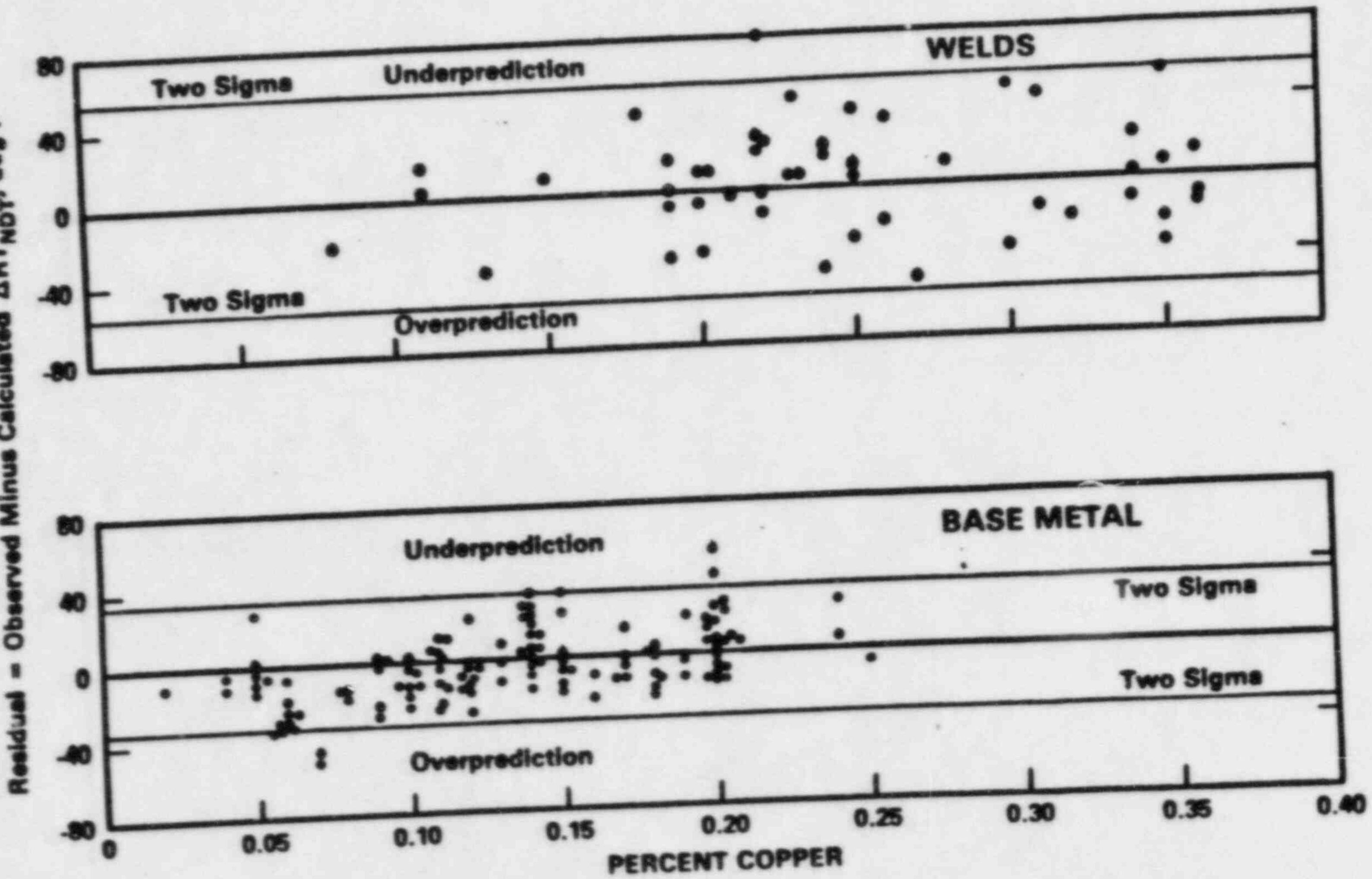


FIG. 12 PLOTS OF RESIDUALS VERSUS COPPER CONTENT FOR 51 WELD AND 126 BASE METAL DATA POINTS

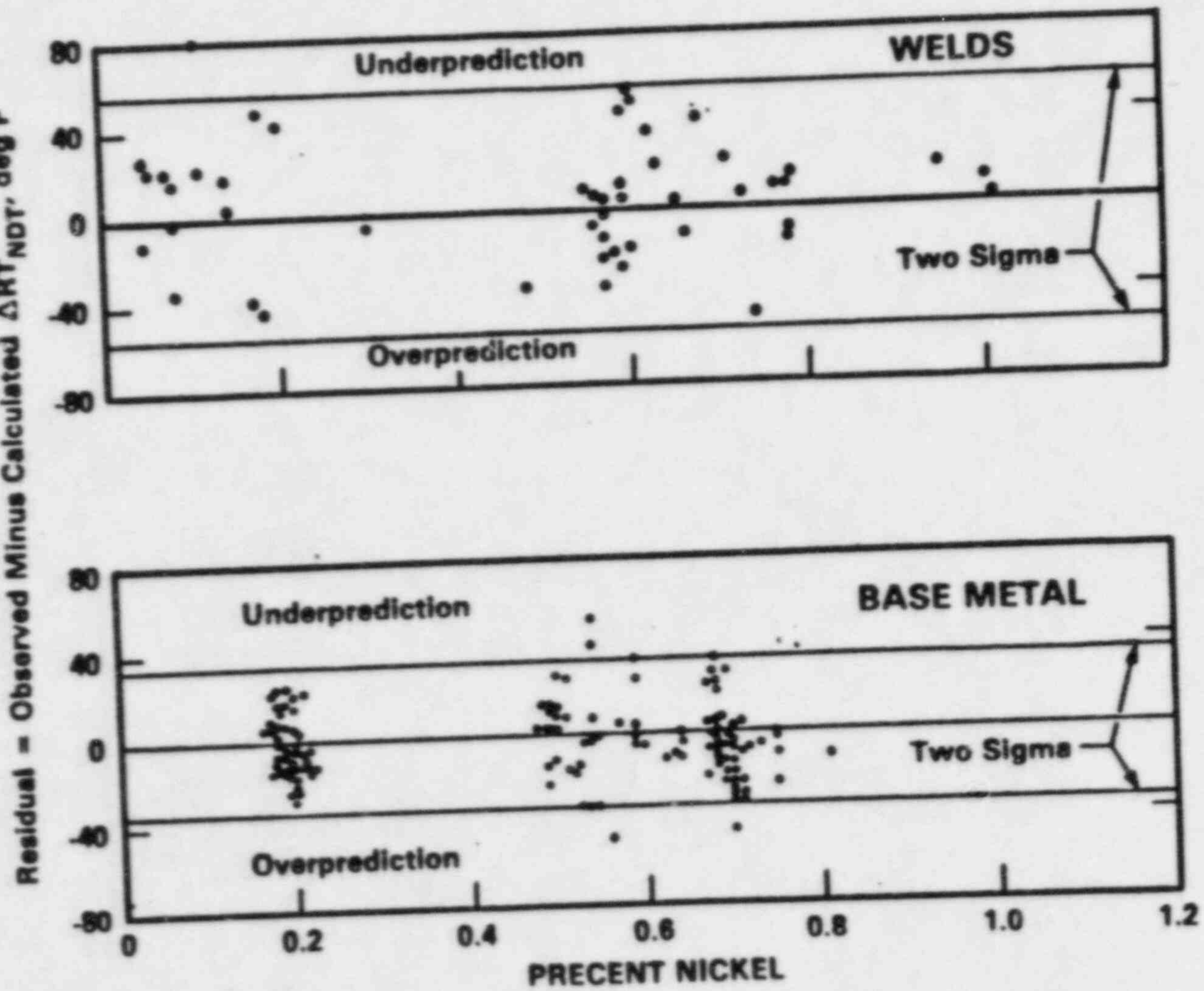


FIG. 13 PLOTS OF RESIDUALS VERSUS NICKEL CONTENT FOR 51 WELD AND 126 BASE METAL DATA POINTS

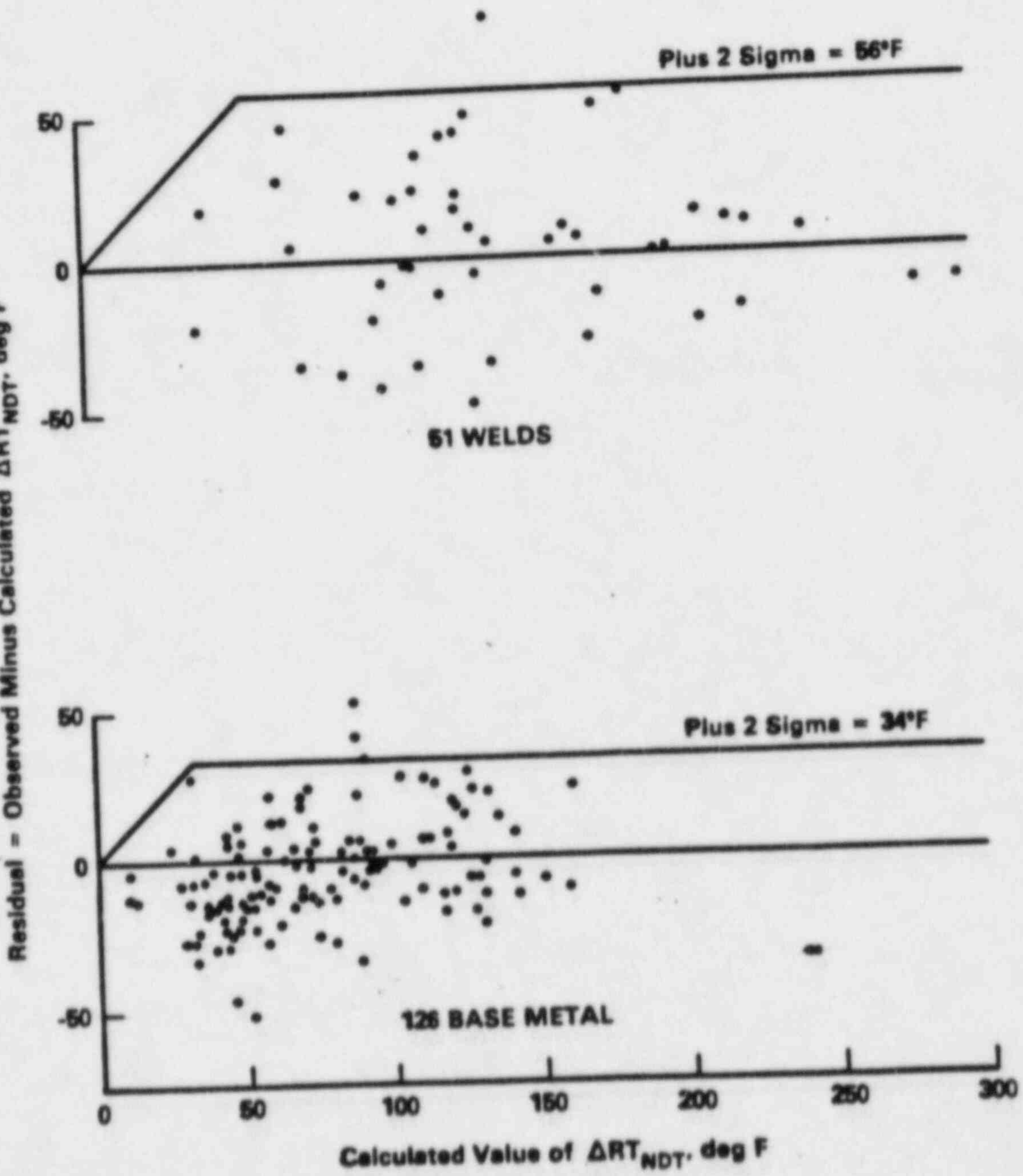
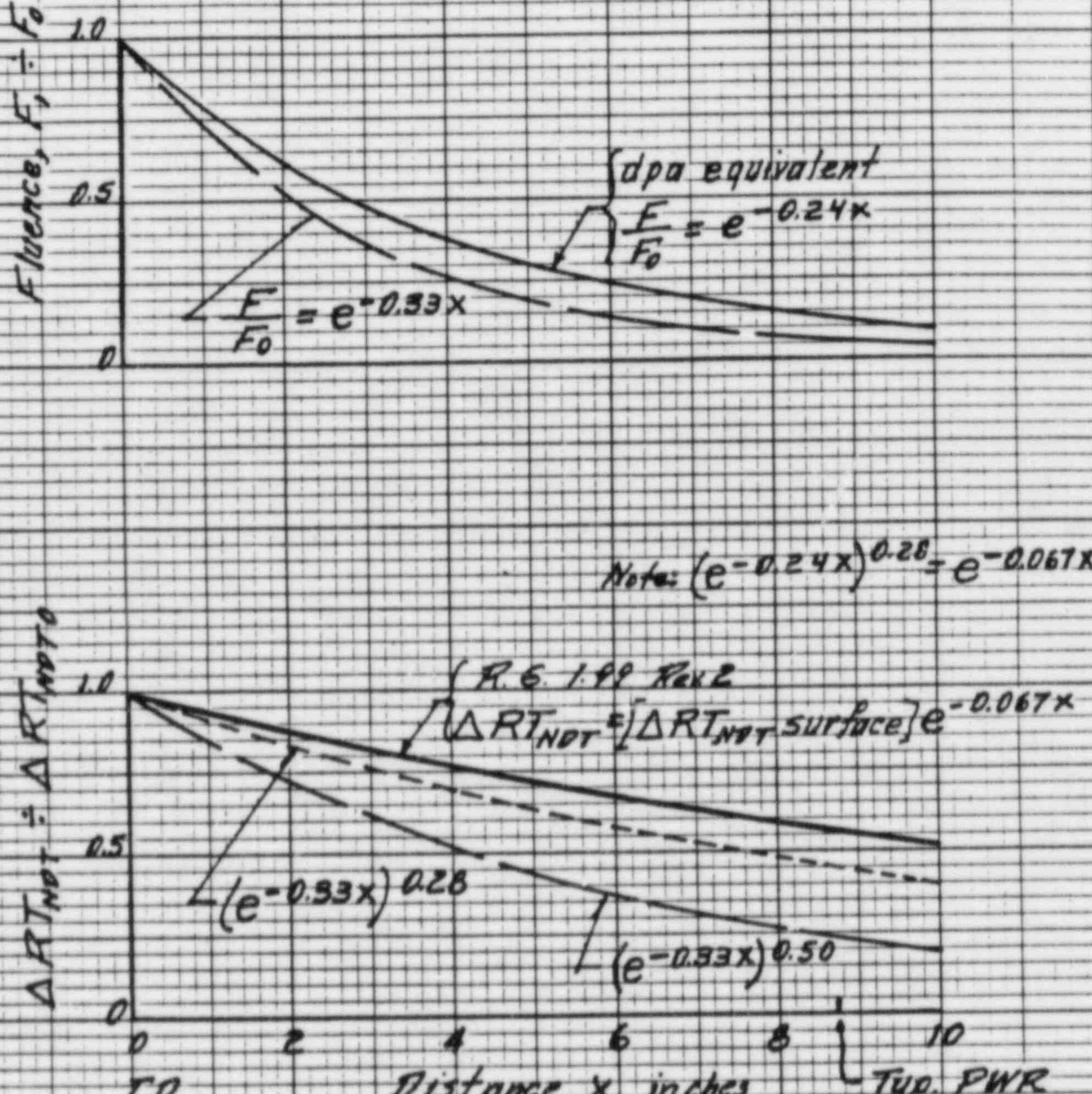


FIG. 14 PLOTS OF RESIDUALS VERSUS CALCULATED VALUE OF ΔRT_{NDT} FOR BOTH WELDS AND BASE METAL



ATTENUATION OF ΔRT_{NDT} AND FLUENCE THROUGH THE VESSEL WALL

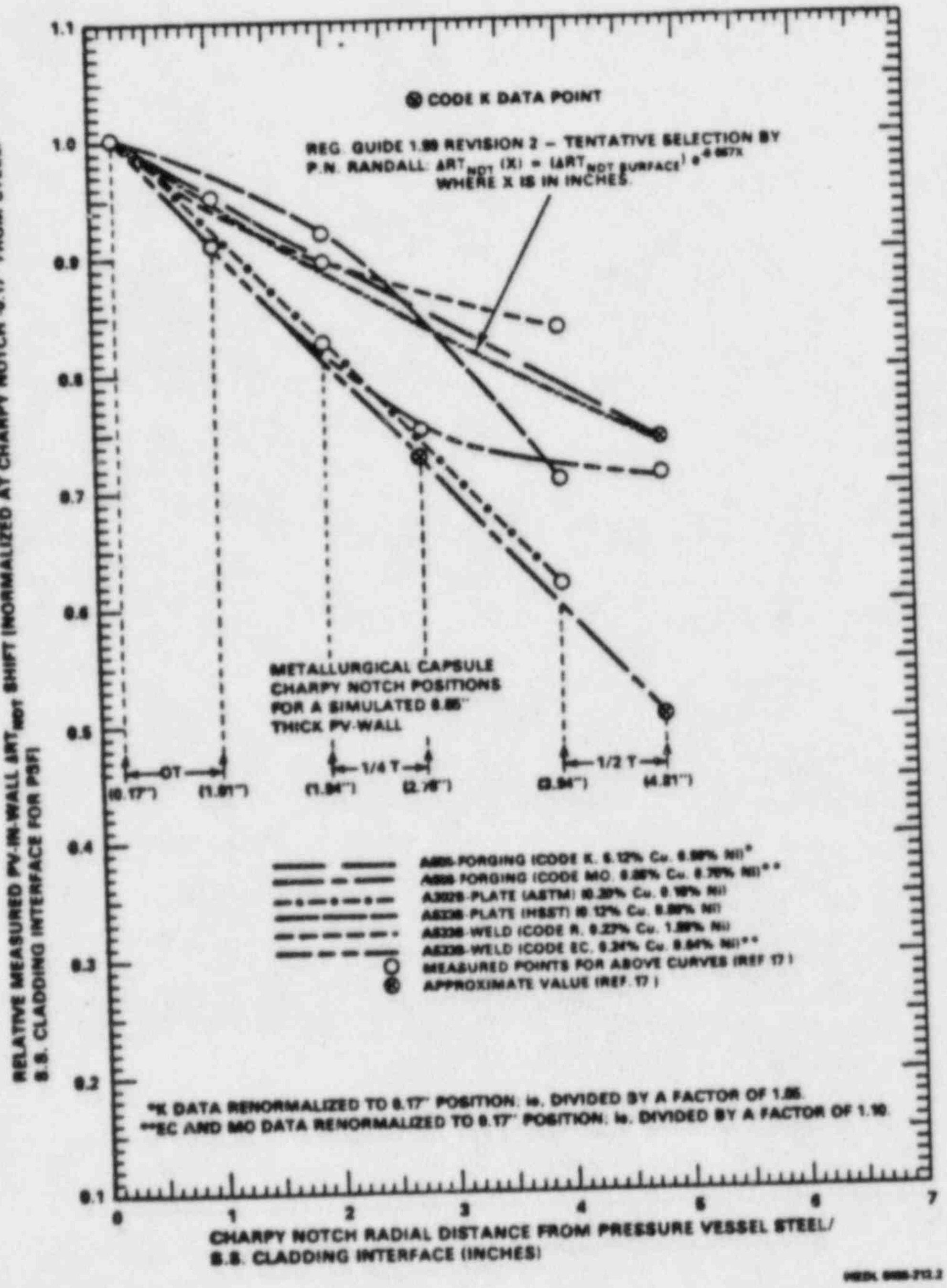


FIGURE 3. PSF Experiment Measured PV-in-Wall ΔRT_{NDT} Shift Gradients for Two Forgings, Two Plates, and Two Weld Materials.

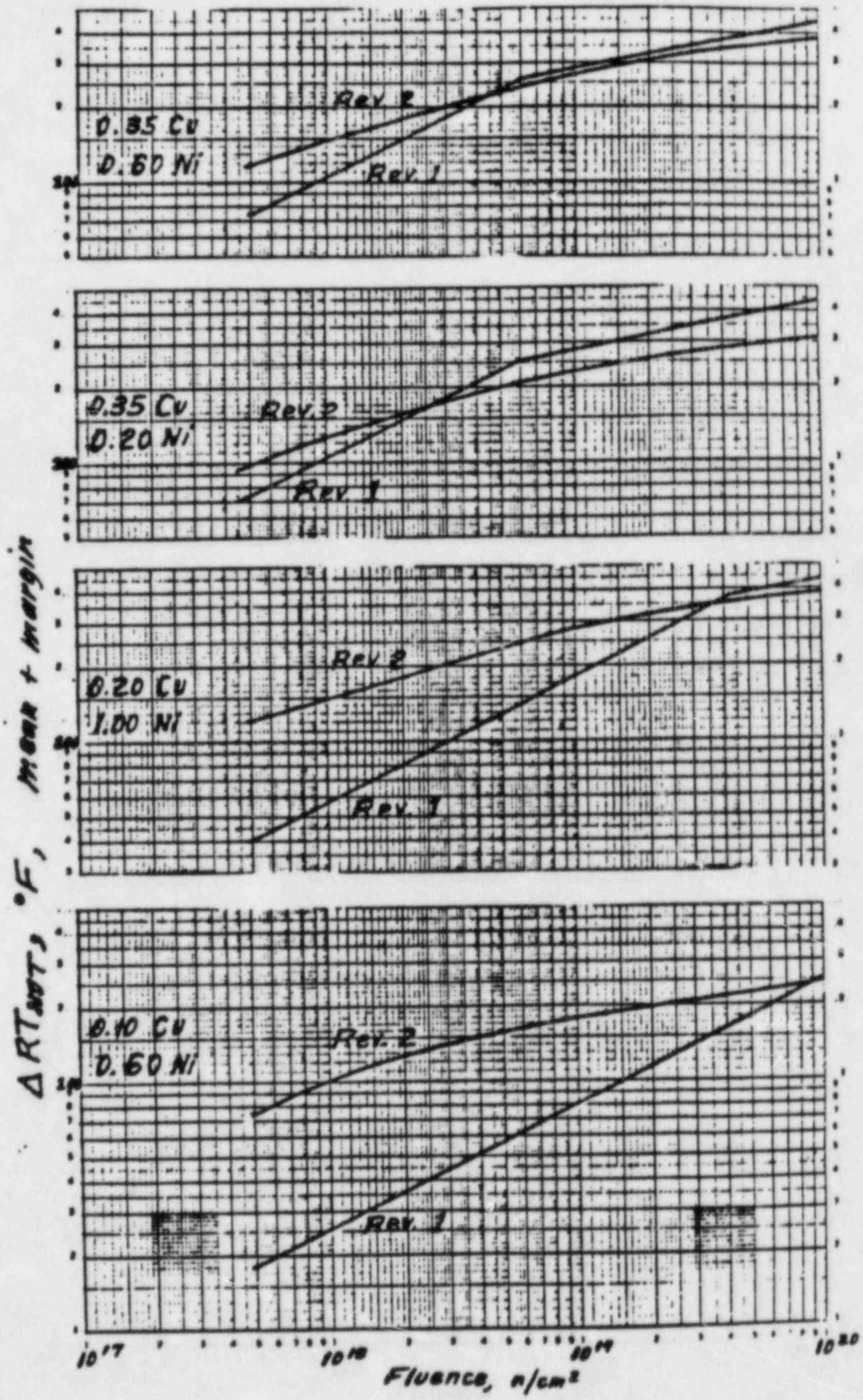
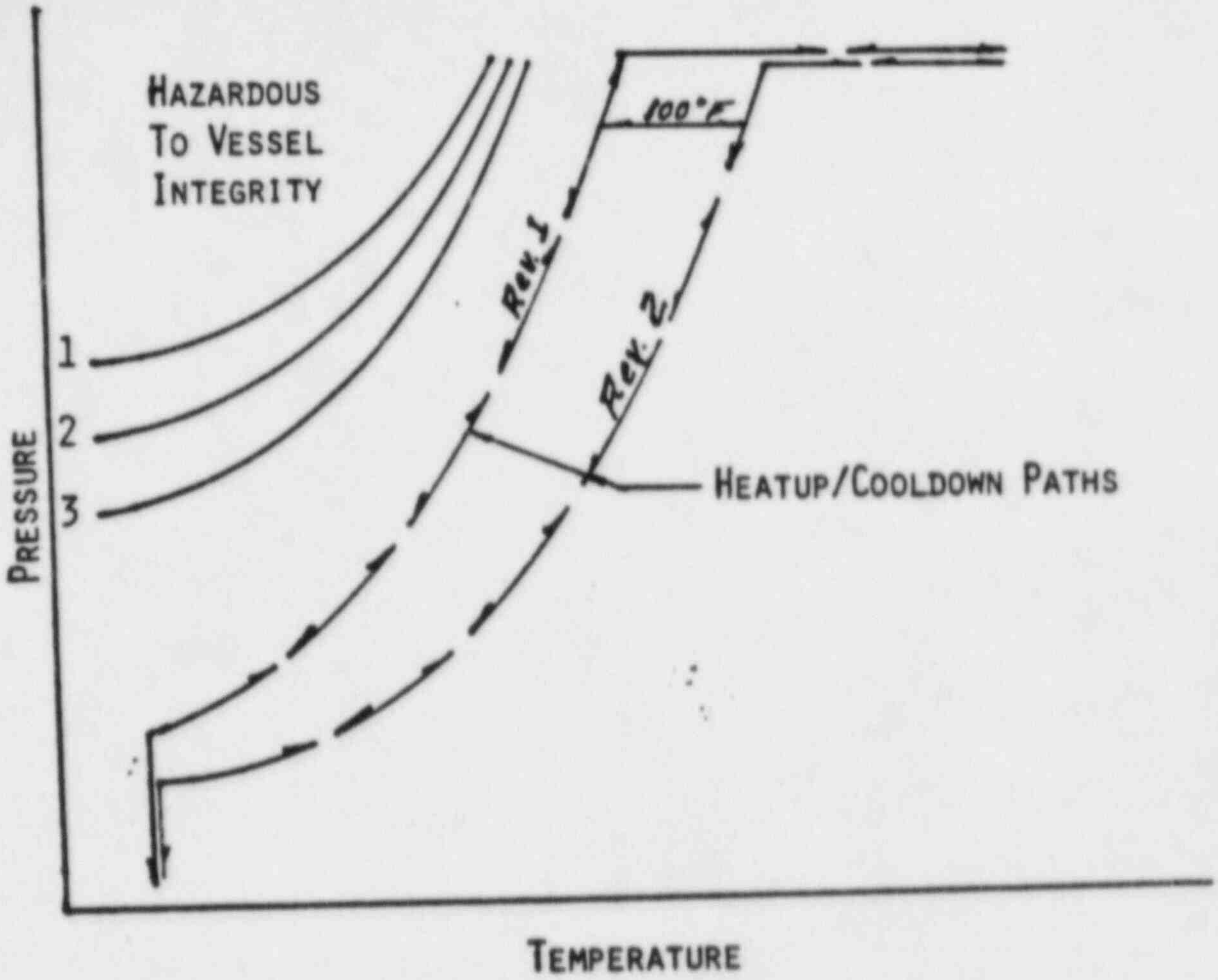

0.35 Cu
0.60 Ni
Rev 2
Rev 1
0.35 Cu
0.20 Ni
Rev 2
Rev 1
0.20 Cu
1.00 Ni
Rev 2
Rev 1
0.10 Cu
0.60 Ni
Rev 2
Rev 1
ΔRT_{NDT}, °F, mean + margin
Fluence, n/cm²
10¹⁷
10¹⁸
10¹⁹
10²⁰

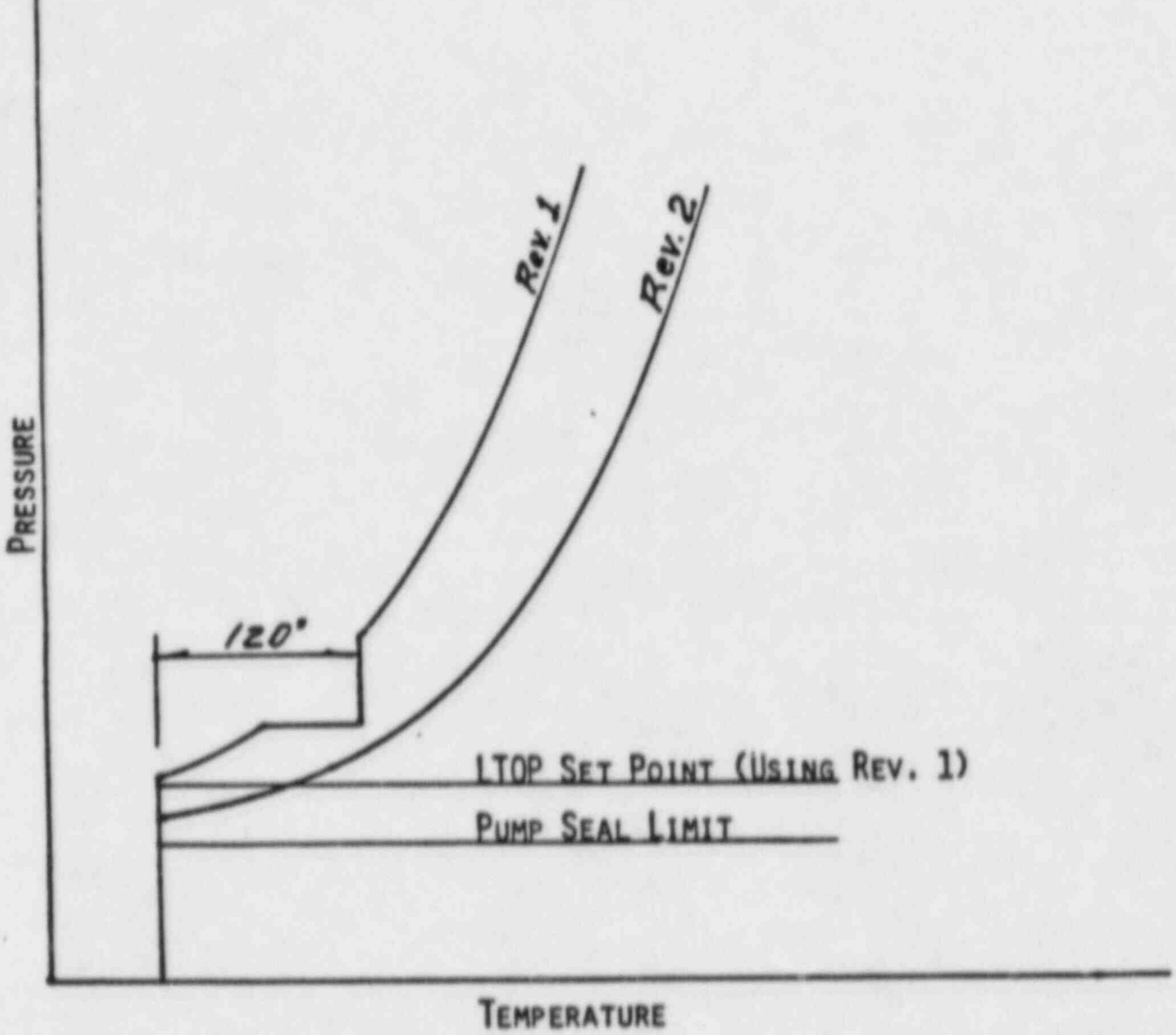
TABLE 1. SUMMARY OF THE CHANGES IN PRESSURE-TEMPERATURE LIMITS EXPECTED TO RESULT FROM A CHANGE FROM REVISION 1 TO REVISION 2 OF REGULATORY GUIDE 1.99

Effect of Change from Rev. 1 to Rev. 2	Operating Reactors PWR	Operating Reactors BWR	Operating Reactors Total	Plants Undergoing Licensing PWR	Plants Undergoing Licensing BWR	Plants Undergoing Licensing Total
Ratchet 50-100°	4*	4	8			
Ratchet 20-50°	16	17	33	32	12	44
No Change (±20°)	23	7	30	3	0	3
Benefit 20-50°	7	1	8			
Benefit 50-100°	1	0	1			
Benefit 100-150°	1	0	1			
Totals	52	29	81	35	12	47

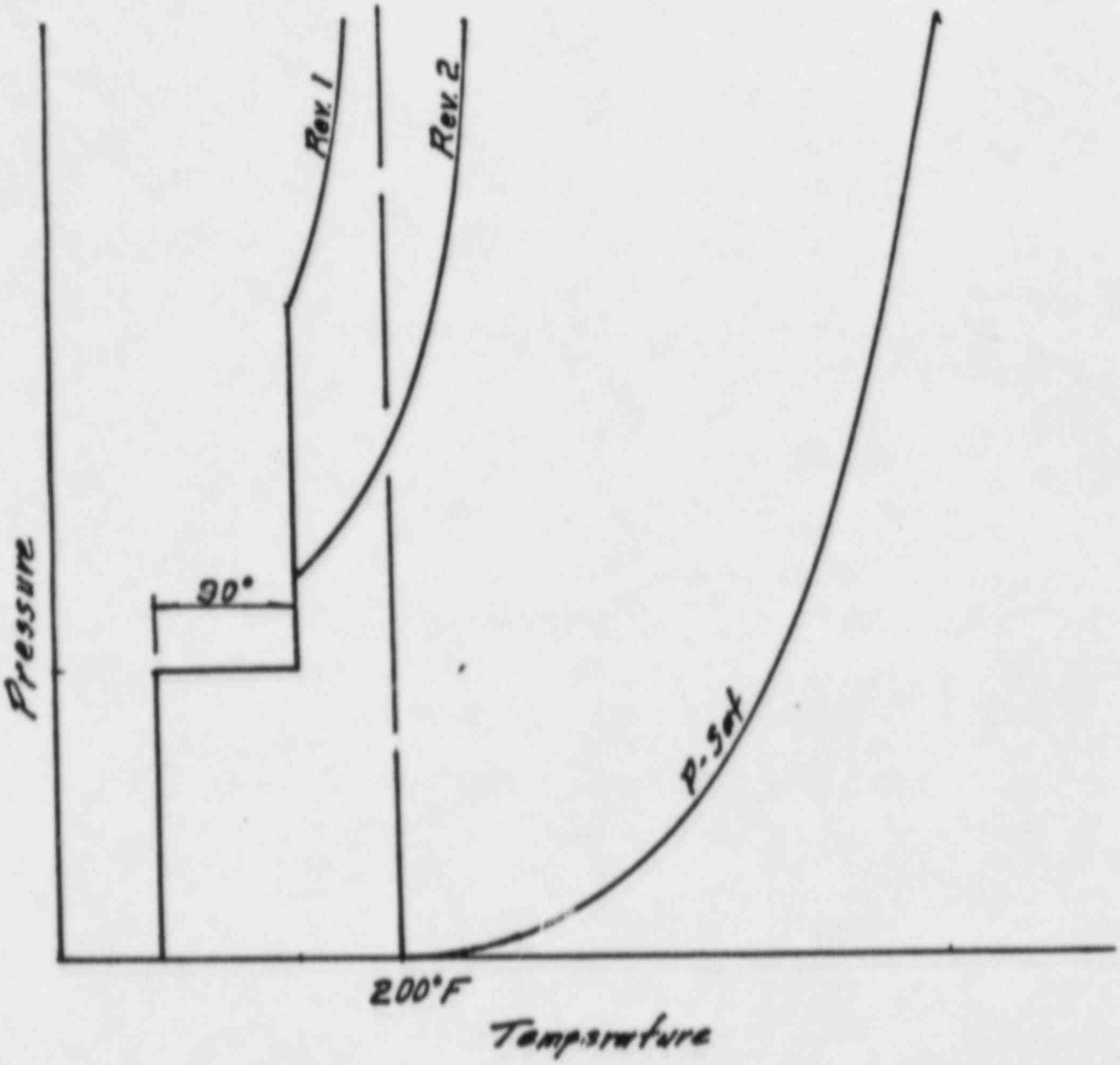
*Values in the table are number of plants.



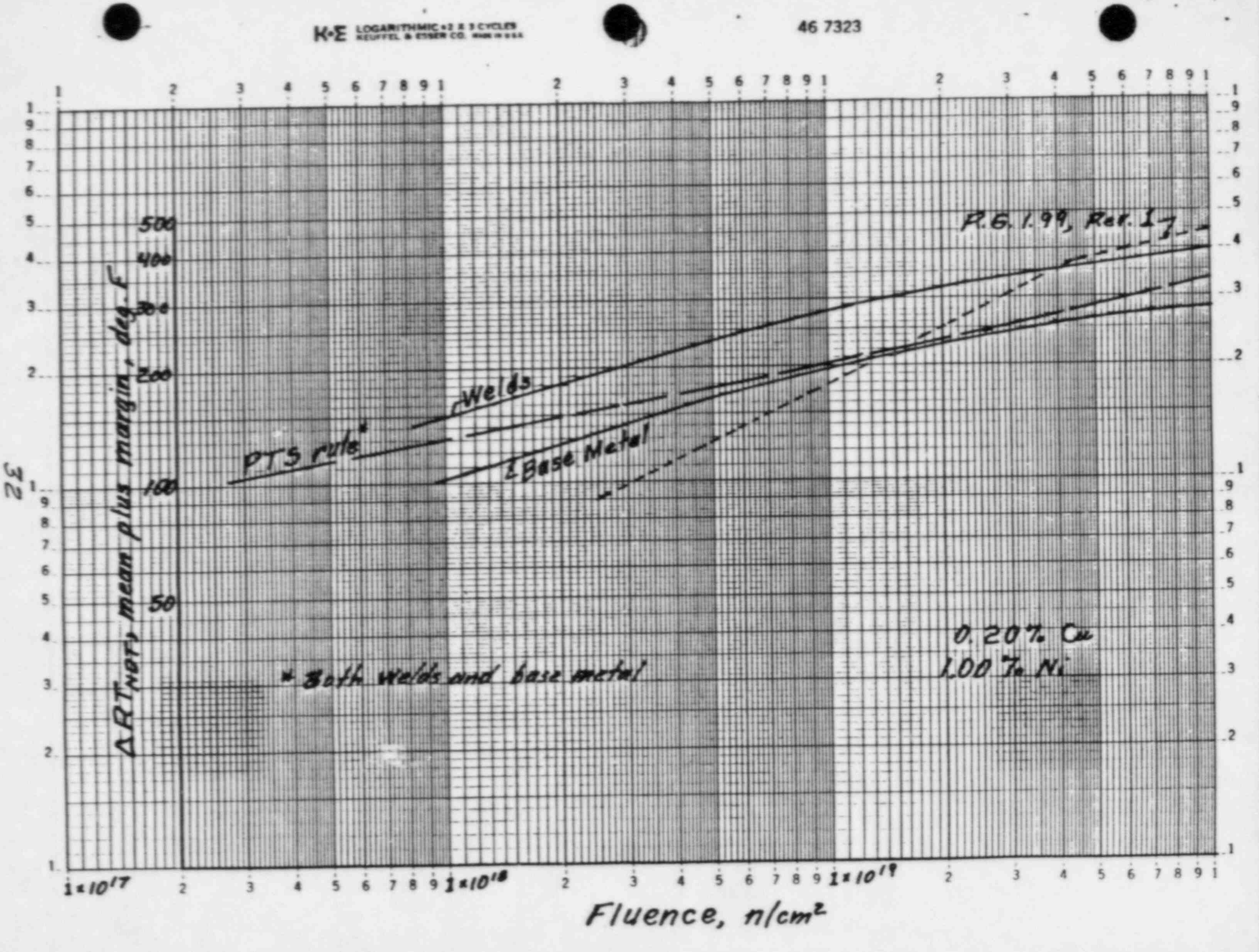
- Increase in Probability of Vessel Failure ← Person - Rem Public Exposure
- Delay in Startup ← Cost of Replacement Power
- Other Costs
- Costs Avoided
- PNL Estimate: $3500-5600 Per Person - Rem Avoided
- Staff Estimates (CAG etc.) $900-1400

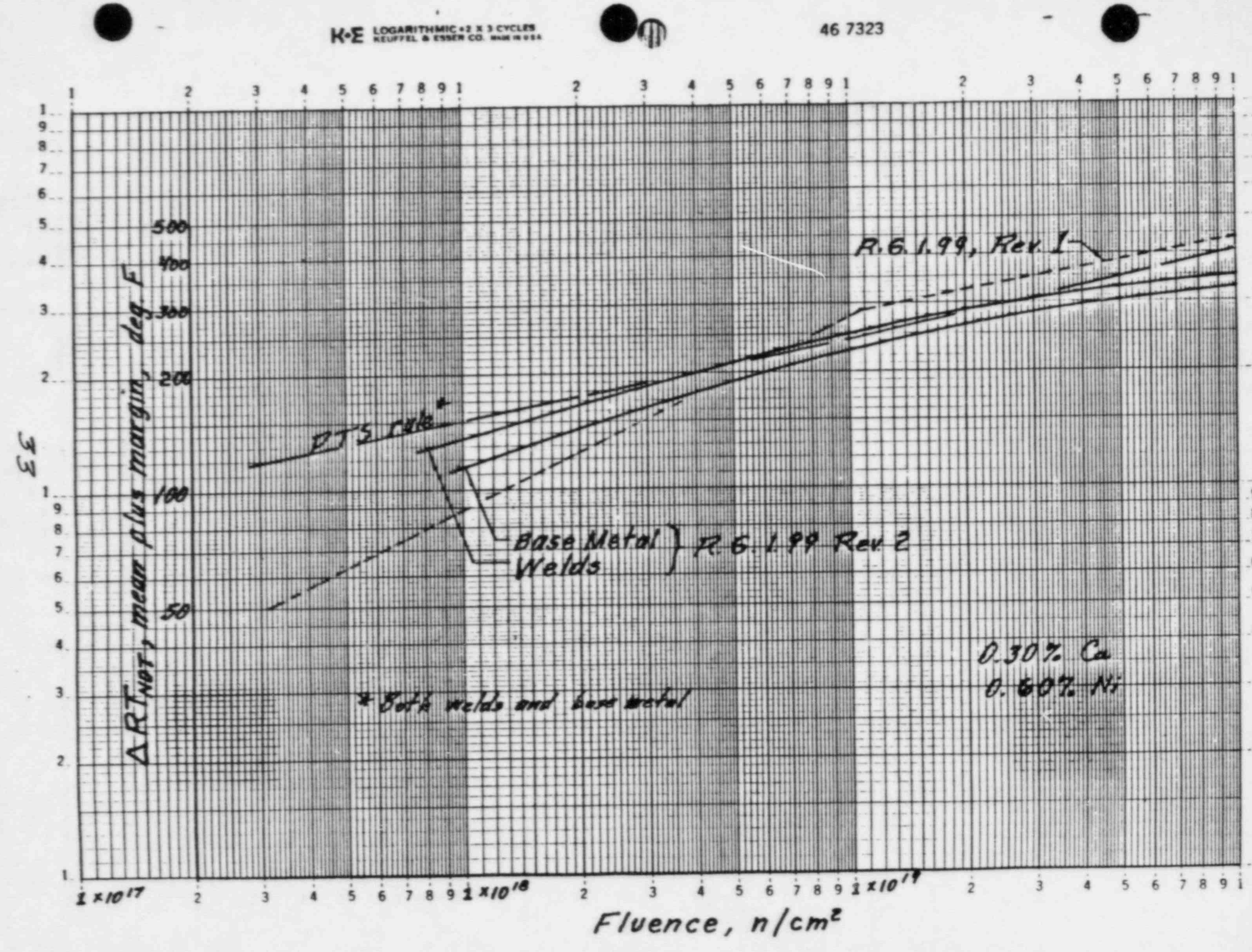


Potential Impact of Revision 2 on the LTOP Set Point for Pressurized Water Reactors



POTENTIAL IMPACT OF REVISION 2 ON THE PRESSURE-TEMPERATURE LIMITS FOR HYDROTEST OF BOILING WATER REACTORS


K·E LOGARITHMIC 2 X 3 CYCLES KEUFFEL & ESSER CO.
46 7323
ΔRTNDT, mean plus margin, deg. F
500
400
300
200
100
50
R.G. 1.99, Rev. 1
Welds
PTS rule*
Base Metal
0.20% Cu
1.00% Ni
* Both welds and base metal
1×10^17
1×10^18
1×10^19
Fluence, n/cm²
23


K·E LOGARITHMIC 2 X 3 CYCLES KEUFFEL & ESSER CO.
46 7323
R.G. 1.99, Rev. 1
PTS rule*
Base Metal
Welds
R.G. 1.99 Rev. 2
0.30% Cu
0.60% Ni
* Both welds and base metal
ΔRT_{NDT}, mean plus margin, deg F
500
400
300
200
100
50
1 x10^17
1 x10^18
1 x10^19
Fluence, n/cm²

RELATIONSHIP TO PTS RULE

- o ISSUANCE OF REVISION 2 FOR PUBLIC COMMENT IN NO WAY AFFECTS THE PTS RULE

- o COMMENTS REQUESTED ON THE EFFECT OF REVISION 2 ON PLANTS FOR NON-PTS PURPOSES

- o COMMENTS ALSO REQUESTED ON THE EFFECT OF REVISION 2 ON CALCULATED PTS RISK AT PLANTS, ASSUMING THE REVISION 2 CORRELATION, IF JUSTIFIED, WOULD AT SOME FUTURE TIME REPLACE THE RT_{PTS} CORRELATIONS.

- o FOLLOWING RESOLUTION OF COMMENTS,

  1. RE-EVALUATE THE CONSERVATISM OF THE PTS RULE, AND

  2. IF JUSTIFIED, AMEND THE RULE, AND

  3. PUBLISH REVISION 2 IN FINAL FORM

IMPLEMENTATION LETTER TO UTILITIES

P-T LIMITS AND ANALYSIS OF ANTICIPATED OPERATIONAL OCCURRENCES

1. ALL SUBMITTALS (CP, OL OR OPERATING REACTORS) AFTER (EFFECTIVE DATE OF REV. 2) WILL BE REVIEWED PER REVISION 2.

2. PLANTS ARE TO SUBMIT REVIEW OF P-T LIMITS WITHIN 3 YEARS.

REVISION OF UPPER-SHELF ENERGY (USE) TREND CURVES

- o P-T LIMITS NOT AFFECTED
- o TRANSIENT ANALYSIS NOT AFFECTED, EXCEPT ATWS
- o USED IN CP REVIEWS OF BELTLINE MATERIALS
- o USED IN ANALYSIS OF 50 FT LB REQUIREMENT (APP. G)
- o CONTRACTOR'S EFFORT TO UPDATE WAS INCONCLUSIVE

A

T-11

OVERVIEW OF NRC RESEARCH PROGRAMS IN NDE

FOR PRESENTATION TO
ACRS METAL COMPONENTS SUBCOMMITTEE

SEPTEMBER 4, 1985

BY
JOSEPH MUSCARA
MATERIALS ENGINEERING BRANCH
DIVISION OF ENGINEERING TECHNOLOGY
OFFICE OF NUCLEAR REGULATORY RESEARCH

NRC PROJECT BRIEF

FIN #B2088

A. E./FLAW RELATION FOR INSERVICE MONITORING OF REACTOR PRESSURE BOUNDARIES

NRC ADMINISTRATION: ENGINEERING TECH. DIV.
MAT'L. ENGR'ING BRANCH
DR. J. MUSCARA, PROG. MNGR.

PNL ADMINISTRATION: ENGINEERING PHYSICS DEPT.
P. H. HUTTON, PROJECT MNGR.

AE/FLAW RELATIONSHIPS

PROGRAM PLAN

- DEVELOP RELATIONSHIPS IN LABORATORY
- TEST ON A STRUCTURE AND REFINE
- DEMONSTRATE ON A REACTOR SYSTEM

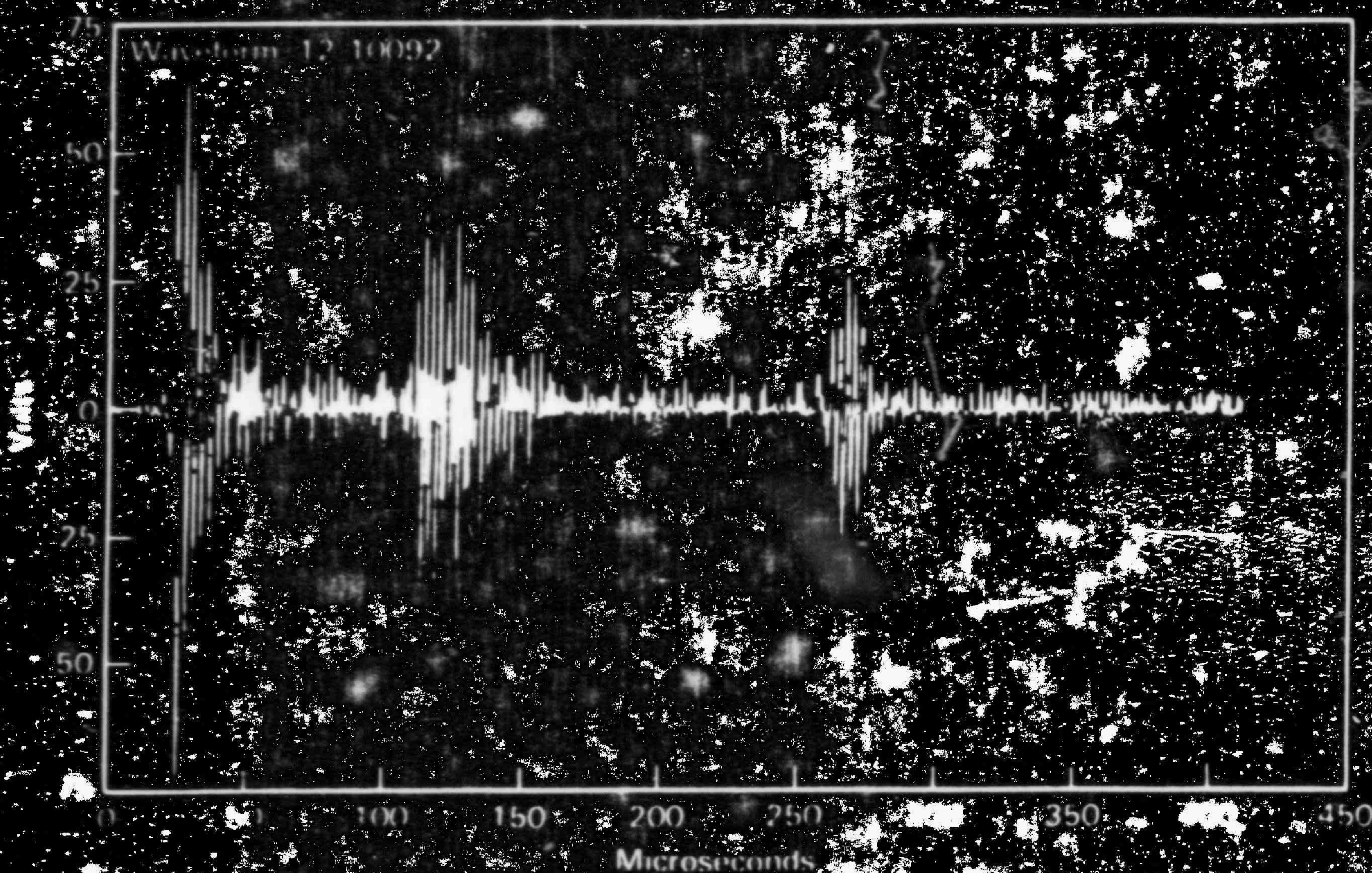
AE/FLAW RELATIONSHIPS

STATUS

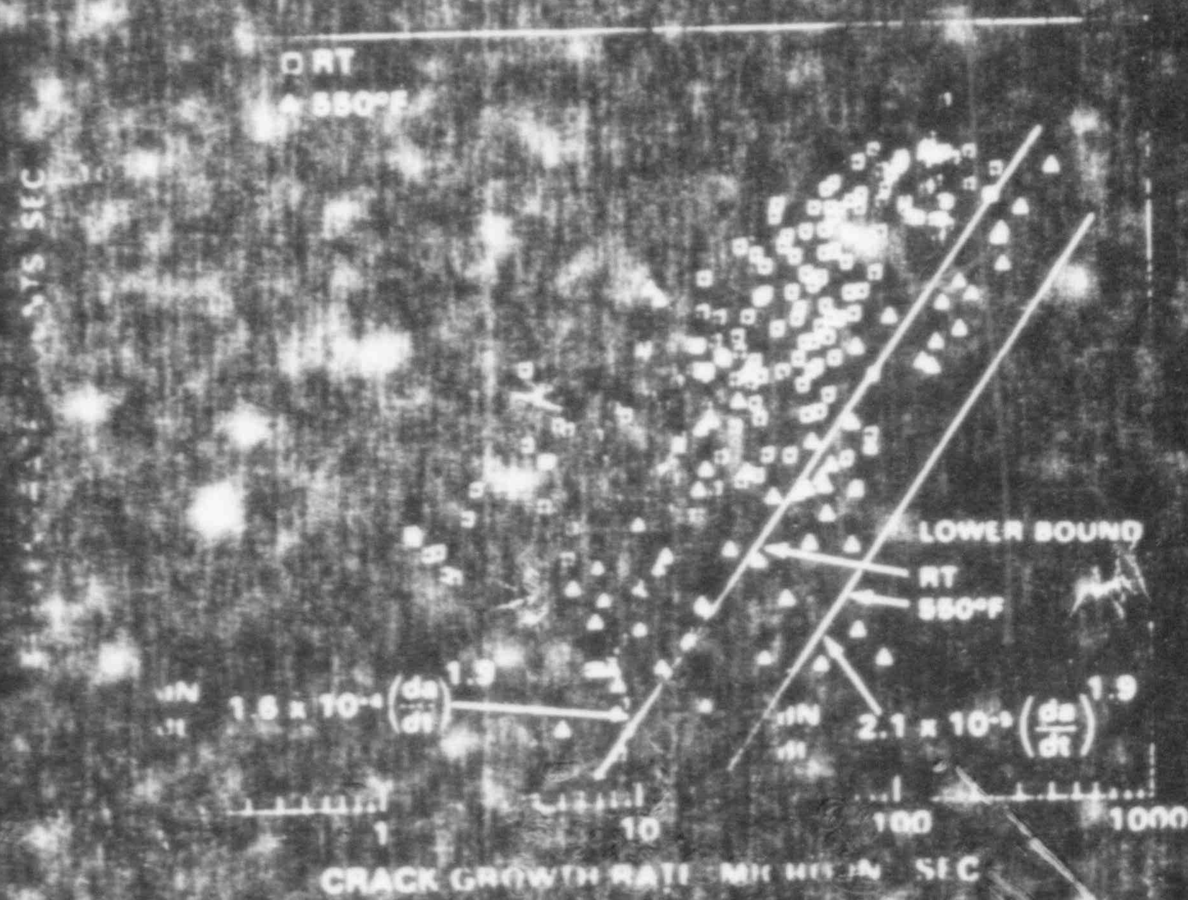
- COOLANT FLOW NOISE PROBLEM SOLVED
- CRACKING BY FATIGUE & BY STRESS CORROSION CAN BE DETECTED
- AE SIGNAL IDENTIFICATION METHOD DEVELOPED
- AE DATA INTERPRETATION METHOD DEVELOPED
- INSTRUMENT SYSTEM FOR REACTOR MONITORING IS AVAILABLE
- PREPARED FOR STARTUP & OPERATIONAL MONITORING AT WATTS BAR #1 REACTOR


AE/FLAW CHARACTERIZATION
COOLANT FLOW NOISE VS TEMPERATURE & PR
- WATTS BAR UNIT 1
VOLTS-PEAK
psig
AE SYST

Three Pulse Waveform Produced Crack Growth AE



Experimental AE/Fatigue Crack Growth Relation



AE/FLAW RELATIONSHIPS

REMAINING WORK

- ON-LINE AE MONITORING AT WATTS BAR
- ON-LINE AE MONITOR PIPE CRACKING
- DEVELOP IGSCC/AE RELATIONSHIPS
- TEST CRACK GROWTH RATE EFFECTS
- FINALIZE AE DATA INTERPRETATION METHOD
- ESTABLISH APPROVED ASTM STANDARD
- GAIN ASME CODE ACCEPTANCE

T-12

AE/FLAW RELATIONSHIPS

END PRODUCT

- DEMONSTRATED TECHNOLOGY WITH IMPLEMENTATION METHODS
- STANDARDS & CODES TO GUIDE APPLICATION AND REGULATION
- TECHNOLOGY TRANSFER

AE LEAK MONITORING OF LWRS

FIN NUMBER A2250

LABORATORY: ARGONNE NATIONAL LABORATORY

OBJECTIVE:

TO DEVELOP AND ASSESS ACOUSTIC TECHNIQUES FOR LEAK DETECTION AND CHARACTERIZATION IN REACTOR COOLANT SYSTEMS

AE HAS THE POTENTIAL FOR:

- MORE SENSITIVE LEAK DETECTION
- LOCATION CAPABILITY
- DISCRIMINATION CAPABILITY
- LEAK RATE DETERMINATION

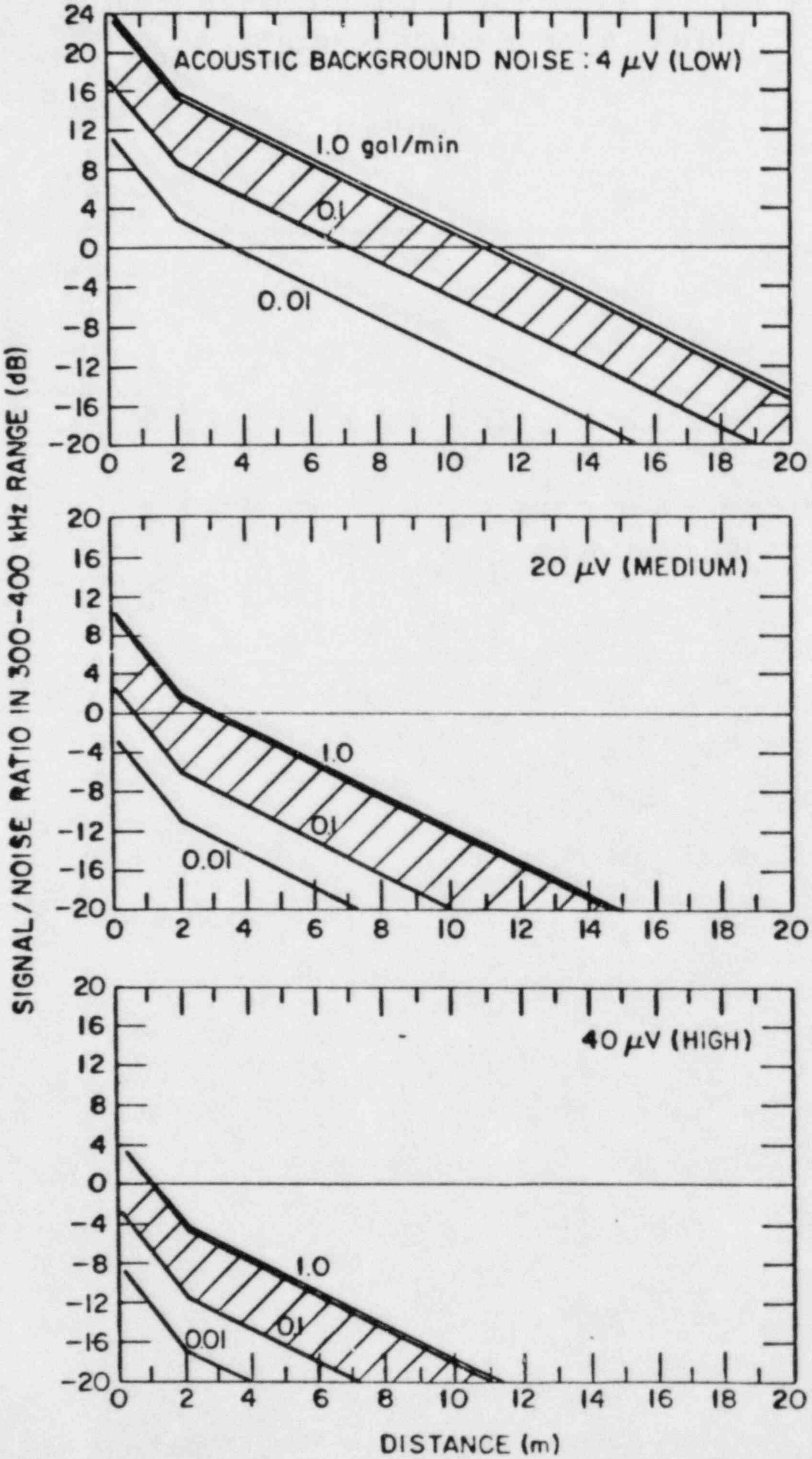
Laboratory Studies at Argonne

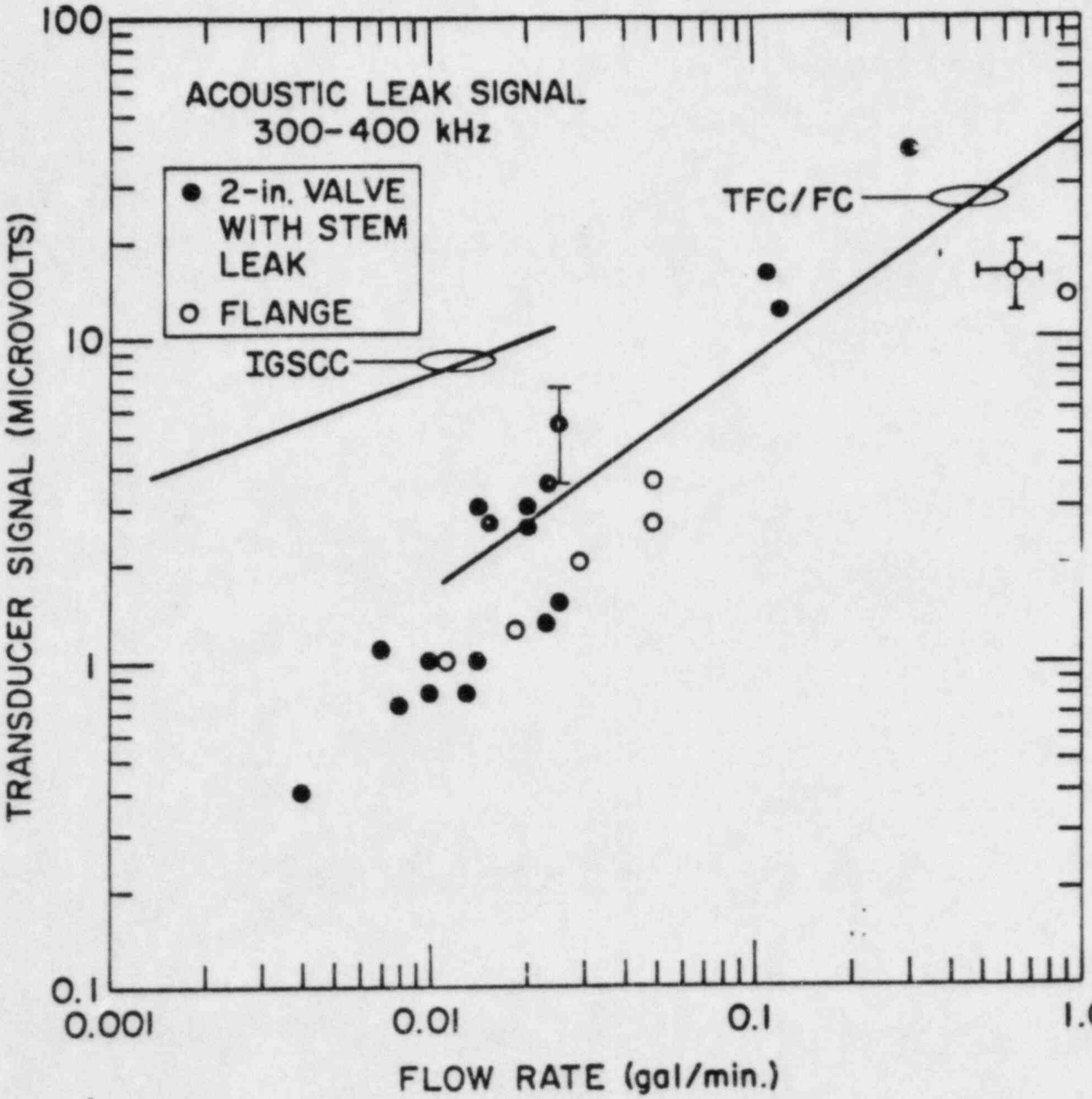
- Laboratory established to assess adequacy of acoustic methods to detect, locate and size leaks
  - Studies carried out with field induced cracks
  - Acoustic background data acquired from existing reactors
  - Acoustic leak detection system installed by utility was reproduced by ANL and evaluated under laboratory conditions
  - Digital continuous acoustic monitoring system using advanced signal processing technology developed for field application

- Laboratory facility was also used to evaluate moisture sensitive tape

Status of Acoustic Monitoring Analytical Capability

- Leaks can be detected, located and probably sized with commercially available acoustic emissions sensors and advanced signal processing
- Leaks rates as small as 0.005 gpm can be detected at 1 meter in low acoustic background noise (1.0 gpm at 1 meter in high background noise)
- Acoustic spectral analysis can be used to distinguish leaking IGSCC from other types of leak sources
- Leaks can be located by cross-correlation analysis of acoustic signals
- IGSCC leak rates probably can be estimated from analysis of acoustic signals in 300-400 khz range


ACOUSTIC BACKGROUND NOISE : 4 μV (LOW)
1.0 gal/min
0.1
0.01
20 μV (MEDIUM)
1.0
0.1
0.01
40 μV (HIGH)
1.0
0.1
0.01
SIGNAL / NOISE RATIO IN 300–400 kHz RANGE (dB)
DISTANCE (m)



ACOUSTIC LEAK SIGNAL
300-400 kHz
2-in. VALVE WITH STEM LEAK
FLANGE
TFC/FC
IGSCC
TRANSDUCER SIGNAL (MICROVOLTS)
100
10
1
0.1
0.001
0.01
0.1
1.0
FLOW RATE (gal/min.)

Future Efforts

- Laboratory studies
  - Complete software development for GARD/ANL acoustic leak detection system
  - Carry out experiments with large leaks from cracks (1 gpm or greater)
- Carry out field tests with GARD/ANL acoustic leak detection system
  - During startup and early operation of watts bar (in collaboration with PNL)
  - With midwest reactor under construction employing electronically generated acoustic signals to study wave propagation and establish feasibility for reactor environment
- Prepare NRC regulation guide for continuous acoustic monitoring (1987)

PROGRAM: DEVELOPMENT AND VALIDATION OF A REAL-TIME SAFT-UT SYSTEM FOR INSERVICE INSPECTION OF LWRS

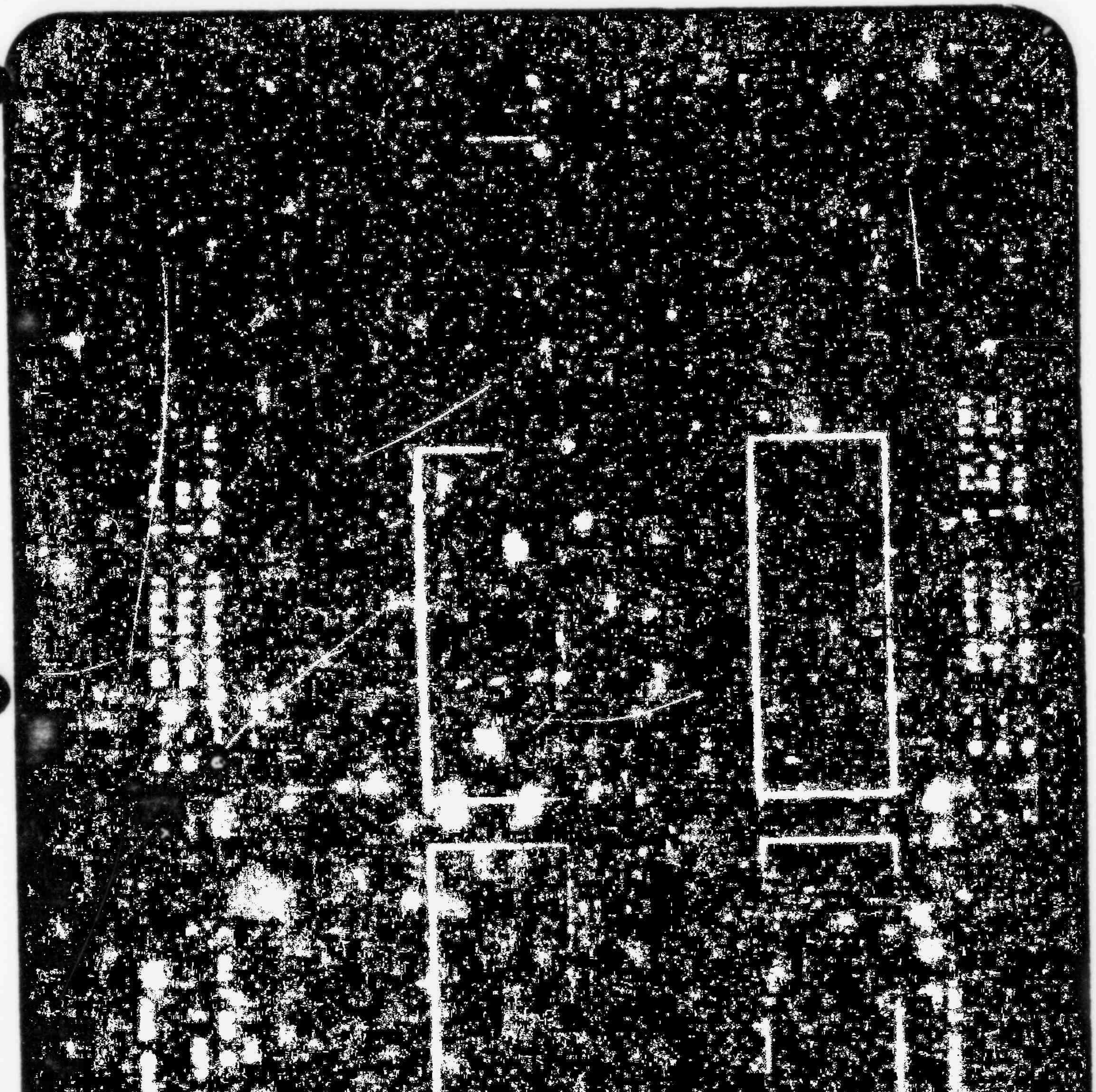
LABORATORY: BATTELLE, PACIFIC NORTHWEST LABORATORY

OBJECTIVES:

1) ENGINEER AND EVALUATE A REAL-TIME SAFT-UT IMAGING SYSTEM FOR INSPECTION OF REACTOR PRIMARY SYSTEM

2) DEVELOP ENGINEERING DATABASE TO ESTABLISH CODE AND REGULATORY ACCEPTANCE OF SAFT-UT IMAGING TECHNIQUES

- UNIVERSITY OF MICHIGAN - DEVELOPED FUNDAMENTALS OF SAFT-UT AND DEMONSTRATED IN LABORATORY
- BATTELLE PNL (FALL 1982) - DEVELOP SAFT-UT INTO A FIELD USABLE TECHNOLOGY
- DEVELOP ANGLE-BEAM SAFT-UT TECHNIQUE
- SAFT ALGORITHM OPTIMIZATION
- ROUND-ROBIN - PISC II (NOVEMBER 1983)
- PRIMARILY FOCUSED ON THE IGSCC PROBLEM
- DEVELOP TANDEM SAFT TECHNIQUE
- FIELD EXPERIENCE - DRESDEN UNIT 3 (FEBRUARY 1984)
  VERMONT YANKEE (JULY 1984)
- ON-SITE SAFT PROCESSING (JUNE 1985) - COMMONWEALTH EDISON (CHICAGO)
- COMPLETE SAFT-UT FIELD SYSTEM INTEGRATION



Bottom Tandem Result

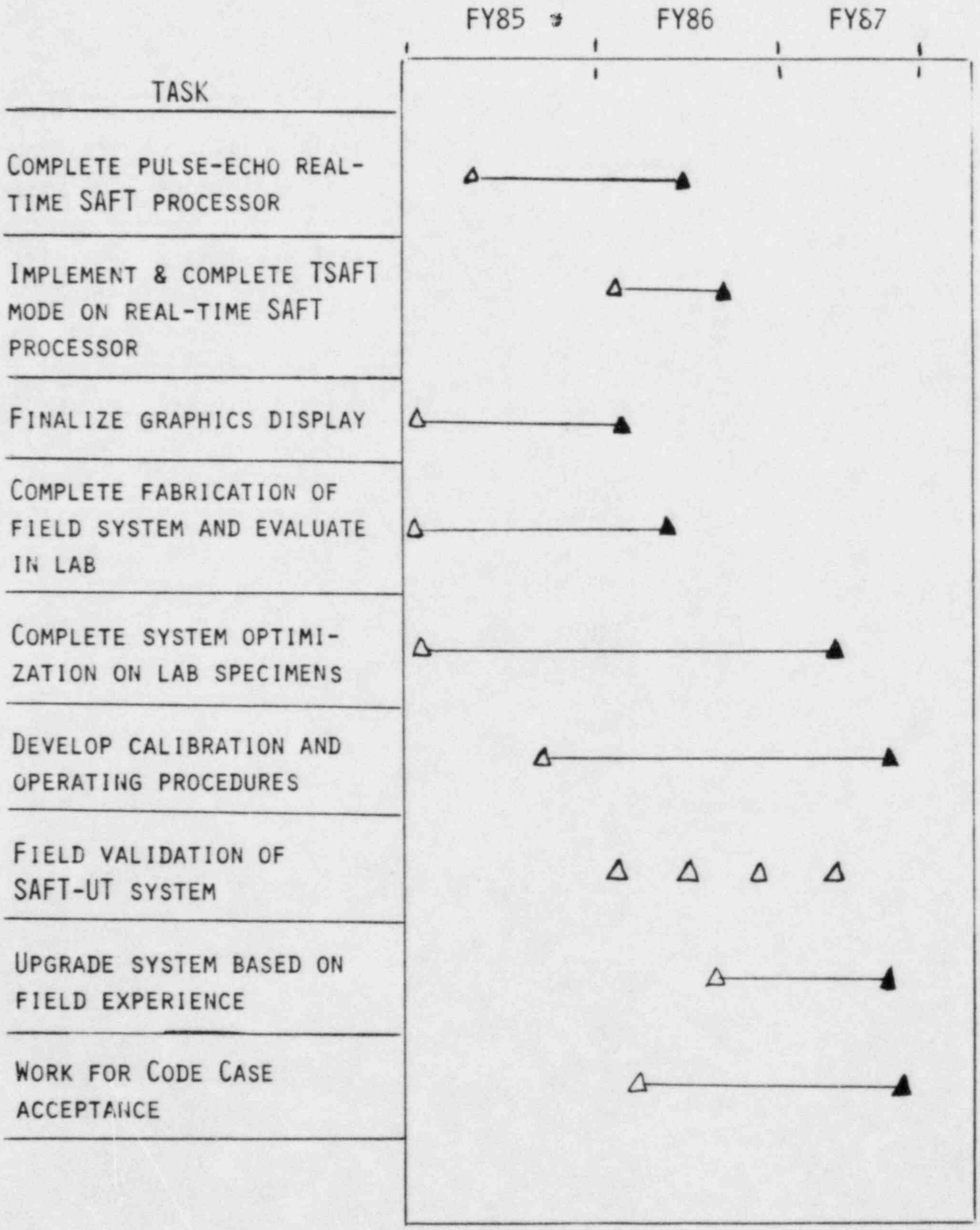
REAL-TIME SAFT-UT FIELD SYSTEM FEATURES

- o MODULAR SOFTWARE
- o COMMERCIALLY AVAILABLE COMPONENTS
- o IMPLEMENTS PULSE-ECHO AND TANDEM SAFT TECHNIQUES
- o REMOTELY OPERATED
- o FLEXIBLE
- o PORTABLE

REMAINING TASKS

- o COOPERATIVE EFFORT WITH INDUSTRIAL ORGANIZATION FOR TECHNOLOGY TRANSFER
- o COMPLETE LABORATORY TESTING ON OTHER LWR COMPONENTS FOR SYSTEM OPTIMIZATION
- o FIELD VALIDATION TESTS
- o UPGRADE FIELD SYSTEM BASED ON FIELD EXPERIENCE
- o WORK FOR CODE CASE ACCEPTANCE
- o FINAL REPORT

REMAINING TASKS ON SAFT-UT PROGRAM



INTEGRATION OF NDE RELIABILITY AND FRACTURE MECHANICS (NDE/FM)

FIN NUMBER B2289

LABORATORY: BATTELLE, PACIFIC NORTHWEST LABORATORY

OBJECTIVE:

TO QUANTIFY THE RELIABILITY OF PRESENT INSERVICE INSPECTION TECHNIQUES FOR PRIMARY SYSTEMS COMPONENTS AND TO ESTABLISH MEANS FOR OBTAINING IMPROVEMENTS IN THE RELIABILITY OF INSERVICE INSPECTIONS. THIS INCLUDES:

- DETERMINE EFFECTIVENESS AND RELIABILITY OF INSERVICE INSPECTIONS PERFORMED ON COMMERCIAL LIGHT WATER REACTOR PRIMARY SYSTEM PIPING AND PRESSURE VESSELS

- USING PROBABILISTIC FRACTURE MECHANICS ANALYSIS, EVALUATE THE IMPACT OF INSPECTION UNRELIABILITY ON SYSTEM SAFETY

- EVALUATE RELIABILITY IMPROVEMENTS OFFERED BY ADVANCED INSPECTION TECHNIQUES

- BASED ON MATERIAL PROPERTIES, SERVICE CONDITIONS, AND INSPECTION UNCERTAINTIES, FORMULATE RECOMMENDATIONS FOR CODE REVISION AND REGULATORY INSPECTION REQUIREMENTS TO ASSURE SUITABLY LOW FAILURE PROBABILITIES

NDE/FM PROGRAM ACTIVITIES

- o PIPE REMOVED FROM OPERATING PLANTS -- MONTICELLO
  - UNDERGOING DECONTAMINATION AND EVALUATION VERSUS ISI RESULTS
- o CAST STAINLESS STEEL ROUND ROBIN -- PISC III
  - 18 TEAMS FROM EUROPE AND THE U.S. EXAMINED 15 CCSS SPECIMENS AND THE RESULTS ARE SIMILAR TO THOSE FOUND IN THE PREVIOUSLY REPORTED PNL WORK -- DATA AND ANALYSIS ARE IN PROGRESS SO THESE MUST BE REGARDED AS PRELIMINARY RESULTS.
- o UT EQUIPMENT INTERACTION STUDY
  - DEVELOP CRITERIA FOR ACCEPTANCE AND RE-QUALIFYING OF UT EQUIPMENT
- o DEVELOPMENT OF ADVANCED INSPECTION CRITERIA
  - REVIEW REASONS FOR INSPECTION AND DEVELOP SAMPLING PLAN, INSPECTION FREQUENCY AND ACCEPTANCE REQUIREMENTS TO MEET OBJECTIVES OF INSPECTION
- o MINI-ROUND ROBIN ON STAINLESS STEEL
  - QUANTIFY IMPROVEMENT IN CAPABILITY FOLLOWING PERFORMANCE DEMONSTRATION REQUIRED IN IEB 83-02
  - QUANTIFY DIFFERENCES BETWEEN INDIVIDUAL AND TEAM PERFORMANCE
  - QUANTIFY DIFFERENCES IN PERFORMANCE IN DETECTING LONG CRACKS VERSUS SHORT CRACKS
  - QUANTIFY DEPTH SIZING CAPABILITY

  STATUS: THE ANALYSIS TO DATE IS VERY VERY PRELIMINARY AND NO DEFINITIVE CONCLUSIONS CAN BE DRAWN, TWO TEAMS ARE SCHEDULED TO PARTICIPATE IN SEPTEMBER. IN GENERAL, RESULTS SEEM TO BE SIMILAR TO PIRR RESULTS.

- o PRESSURE VESSEL ACTIVITIES
  - THESE ACTIVITIES ARE PRIMARILY THE TRACKING OF THE PISC II WORK TO SEE WHAT NEEDS TO BE DONE SO THAT DUPLICATION OF EFFORTS DOES NOT OCCUR.

INSPECTION OF WELD OVERLAY REPAIRS

OBJECTIVE:

EVALUATE THE EFFECTIVENESS AND LIMITATIONS OF ULTRASONIC INSPECTION OF WELD OVERLAY REPAIRS.

PROVIDE RECOMMENDATIONS FOR INSPECTION PARAMETERS THAT WILL PROVIDE BEST EXAMINATION RESULTS.

DELIVERABLES:

AN INTERIM REPORT HAS BEEN PREPARED AND TRANSMITTED TO NRC STAFF.

A FINAL REPORT WILL BE DEVELOPED BY DECEMBER 1986.

INSPECTION OF WELD OVERLAY REPAIRS

INTERIM REPORT:

- o BASED ON
  - LITERATURE SEARCH
  - EXPERIMENTAL DATA ON DISTORTION OF ULTRASOUND PROPAGATING THROUGH WELD OVERLAY
  - REVIEW OF DATA FROM WELD OVERLAY INSPECTION DEVELOPMENT PROGRAMS OF EPRI

- o CONCLUSIONS:

1. CONVENTIONAL SHEAR WAVE EXAMINATION NOT EFFECTIVE.
2. DETECTION AND SIZING OF IGSCC LESS THAN 50% OF ORIGINAL WALL THICKNESS IS UNRELIABLE.
3. LONGITUDINAL WAVE PROBES WITH INCIDENT ANGLES BETWEEN 40° AND 70° PROVIDES BEST RESULTS. MOST SUCCESSFUL INSPECTION RESULTS BETWEEN 2.0 AND 4.0 MHZ.
4. SURFACE PREPARATION OF WELD OVERLAY IS REQUIRED FOR EXAMINATION.
5. SURFACE CONDITION SHOULD BE:
   - o RMS SURFACE ROUGHNESS SHOULD BE 250 MICROINCH OR LESS
   - o SURFACE WAVINESS MUST BE NOT GREATER THAN 0.06 INCH RADIAL DEVIATION FROM PEAK TO VALLEY POINTS WITHIN A 1" X 1" SURFACE AREA.
6. TANDEM AND OR CREEPING WAVE PROBES WERE MORE ACCURATE FOR DETERMINING REMAINING LIGAMENT ASSOCIATED WITH DEEP IGSC CRACKS.

INSPECTION OF WELD OVERLAY REPAIRS

RECOMMENDATIONS:

1. WELD OVERLAY JOINTS SHOULD BE EXAMINED WITH LONGITUDINAL WAVES USING AT LEAST TWO DIFFERENT INCIDENT ANGLES IN THE 40° TO 70° RANGE AND SEPARATED BY A DIFFERENCE OF 15° (E.G., 45° AND 60°).

2. INSPECTORS SHOULD DEMONSTRATE THEIR CAPABILITY TO DETECT FLAWS IN WELD OVERLAID JOINTS.

3. WHERE POSSIBLE, EACH HEAT-AFFECTED ZONE SHOULD BE EXAMINED FROM BOTH SIDES.

4. TANDEM AND/OR CREEPING WAVE PROBES SHOULD BE USED TO ESTIMATE THE REMAINING LIGAMENT OF IGSC CRACKS SUSPECTED OF ENTERING INTO THE WELD OVERLAY MATERIAL.

5. AN ACCEPTANCE CRITERIA FOR FABRICATION FLAWS SHOULD BE DETERMINED.

QUALIFICATION OF ULTRASONIC INSERVICE INSPECTION

A. BACKGROUND

1. SEVERAL RESEARCH STUDIES HAVE SHOWN DEFICIENCIES IN UT/ISI

   - o NATIONAL STUDIES - PNL PIPE INSPECTION ROUND ROBIN
     - EPRI SIZING STUDY
     - CONCLUSIONS OF NUREG-1061
   - o INTERNATIONAL STUDIES - PISC I, II AND III

2. FIELD EXPERIENCE

   - o RESULTS OF NINE MILE POINT NUCLEAR STATION UNIT 1
   - o TEAMS PASSING I&E BULLETIN 83-02 STILL HAVE INCONSISTENT RESULTS (DRESDEN, VERMONT YANKEE, HATCH, ETC.)

B. BASIC ASSUMPTIONS/ELEMENTS FOR DEVELOPMENT OF QUALIFICATION CRITERIA

- o CURRENT REQUIREMENTS INADEQUATE
- o "BLIND TEST" DEMONSTRATIONS
- o ADDRESS PERSONNEL, EQUIPMENT, PROCEDURES
- o UNIQUELY ADDRESS SPECIFIC APPLICATIONS
- o ANNUAL TRAINING COURSES TO ADDRESS CURRENT PROBLEMS AND TECHNIQUES
- o STRENGTHEN WRITTEN EXAM REQUIREMENTS TO COVER SPECIFIC UT/ISI APPLICATIONS
- o PASS/FAIL BASED ON DESIRED PERFORMANCE STATISTICS
- o SPECIFY TOLERANCES ON EQUIPMENT PERFORMANCE
- o APPLIES TO ALL UT/ISI
- o ALL CRITERIA PROVIDED SO ANYONE CAN SET UP QUALIFICATION PROGRAM

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QUALIFICATION ACTIVITIES

- o FOR SEVERAL YEARS NRC HAS BEEN SPONSORING RESEARCH AT PNL TO DEVELOP CRITERIA FOR THE QUALIFICATION OF THE INSPECTION OF LIGHT WATER REACTOR COMPONENTS.

- o JUNE 1983 - A WORKSHOP WAS HELD WITH THE NUCLEAR INDUSTRY TO REVIEW AN INITIAL DRAFT ON THE NEEDS AND METHODS OF QUALIFICATION.

- o OCTOBER 1984 - A SECOND DRAFT DOCUMENT WAS REVIEWED BY THE NRC STAFF; GENERAL AGREEMENT WAS REACHED ON THE NEEDS AND CONTENT OF THE DOCUMENT.

- o NOVEMBER 1984 - THE SECOND DRAFT DOCUMENT WAS PRESENTED AT A WORKSHOP WITH INDUSTRY.

- o THE OVERWHELMING POSITION OF THIS WORKSHOP WITH THE INDUSTRY WAS THE RECOMMENDATION FOR AN AD HOC TASK GROUP TO GENERATE RECOMMENDATIONS USING THE DRAFT DOCUMENT, ON QUALIFICATION FOR ASME SECTION XI. THE GENERAL VIEW WAS THAT PRESENT PERFORMANCE WAS NOT ADEQUATE AND IMPROVEMENTS WERE NEEDED.

- o AUGUST 1985 - THE AD HOC TASK GROUP PRESENTED THEIR RECOMMENDATIONS TO ASME SECTION XI SUBGROUP ON NDE.

- o NRC DOCUMENT BEING PREPARED FOR PUBLICATION AS A NUREG REPORT IN FALL 1985.

- o IN THE FUTURE THE PNL WORK WILL FOCUS ON SUPPORTING THE ACCEPTANCE OF THE AD HOC TASK GROUP'S RECOMMENDATIONS AND ON THE DEVELOPMENT OF ADDITIONAL APPENDICES TO COVER PERFORMANCE DEMONSTRATION TESTS FOR ALL INSPECTED COMPONENTS.

- o IT SHOULD BE RECOGNIZED THAT THE INDUSTRY HAS BEEN WORKING VERY HARD IN THE ASME AD HOC TASK GROUP TO GENERATE THE QUALIFICATION CRITERIA.

Eddy Current Examinations of the Retired from Service Surry Steam Generator

Post-Service Baseline Examinations

- 3000+ Tubes
- Zetec MIZ 12 Multifrequency System
- Intercontrole IC3FA Multifrequency System

Summary of SGTIP Baseline Inspections

	Two Detections Size Difference 10%	Two Detections Size Difference 10%	Disagree on Type of Detection	
Team A Size	232	73	14	7
Team B Size	190	174	20	5
Team A 20% < Size < 40%	156	111	44	11
Team B 20% < Size < 40%	125	40	31	2
Team A 0% < Size < 20%	88	41	63	0
Team B 0% < Size < 20%	88	44	390	0

Results of Post Service Baseline Examinations

- One team sized defects larger (~5%) on average
- One team called considerably more <20% defects
- A number of instances where both teams detected an eddy current indication, which one team analyzed as a defect and the other team as a nondefect artifact (such as a conductive deposit)
- Several indications in each size range where one team had no signal
- A generally high correlation on data acquisition (i.e. both teams had eddy current signal information at a large majority of locations)

Selection of Round Robin Tubes

Purpose: Establish a representative subset of samples within the steam generator which could be repeatedly examined by a number of teams within a reasonable time.

- **Statistically sufficient to enable a 90% probability of detecting a 10% disagreement in detection**
- **Representative of all regions of the generator and all potential defect locations/types**
- **Bounded by the reasonable limits for conducting destructive validation studies**
- **Utilized information from the baseline examinations plus information from historical generator records and from secondary side characterization efforts**
- **A double stratification statistical technique, taking into account defect size categories and sizing agreement categories from the two post service baseline examinations**

Actual Composition of Round Robin Tube [illegible]

Tubes with Indications: 240

	Indications	Two Detections Size Diff < 10%	Two Detections Size Diff [illegible]	Disagreement on Detection
Size Category	40	36	36	32
	[illegible]	30	30	29
	20	16	16	16

[illegible] Tubes 80

Round Robin Examinations

- **Data Acquisition and Analysis Round Robin**
  Five teams using Zetec MIZ 12 and DDA 4 Analyzer

- **Data Analysis Round Robin**
  Eight teams analyzing same MIZ 12 generated data tapes using DDA 4 Analyzer

- **Data Analysis Round Robin (French)**
  Eight teams analyzing same Intercontrole IC3FA generated data tapes (in Europe)

- **Advanced Technique Examinations**
  Use of alternate NDT techniques (ultrasonics, profilometry), or advanced eddy current developments (special probes, new instruments)

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SGGP EC Round Robin

Team	Total Number Indic[illegible]ions	Total [illegible] Indic[illegible]
A	287	
[illegible]	202	20[illegible]
C	292	226
[illegible]	257 &	255
[illegible]	259	222

[illegible] Ca[illegible] Within 3 [illegible]ches Deleted

SGGP EC Round Robin

Numbers of Indications in Each Size Category

	Size Category		
Team	<20%	20-40%	>40%
A	45	90	121
B	27	76	89
C	24	82	120
D	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	81	[illegible]

SGGP EC Round Robin

Two-Team Comparisons of Common Detections

	Fraction Agreement by Team				
	A	B	C	D	E
A	1	0.61	0.76	0.77	0.75
B	0.78	1	0.79	0.79	0.76
C	0.86	0.70	1	0.88	0.85
D	0.74	0.60	0.75	1	0.72
E	0.86	[illegible]	0.87	0.86	[illegible]

Data Analysis Round Robin Team

- J.A. Jones Applied Research Company
- Rochester Gas and Electric Company
- Comb[illegible]neering.
- Bab[illegible]ilcox
- W[illegible]house
- [illegible] Incorporated
- Conam Inspection Services
- [illegible] Testing Laboratories

Zetec · Baseline used [illegible] tubes

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Analysis Round Robin
Detection Agreement [illegible]

[illegible]	F	G	H	J	I	K	L	M	[illegible]
F	1	0.83	*	0.79	0.77	0.69	0.61	0.6	[illegible]
G	0.87	1	*	0.57	0.26	0.27	[illegible]	[illegible]	0.38
H	*	*	*	*	*	*	*	*	*
J	[illegible]	0.34	*	1	0.79	0.70	0.77	0.28	0.29
I		0.59	*	0.75	1	0.14	0.40	0.58	0.77
K		0.87	*	0.87	0.86	1	0.59	0.56	0.68
L		0.63	*	0.74	0.75	0.80	1	0.63	0.80
M		0.34	*	0.33	0.42	0.33	0.28	1	0.40
[illegible]		1.00	*	0.72	0.72	0.82	0.80	0.52	1

4.28.[illegible]

H = no location info

Numbers of Defects* Reported by Analysis Round Robin Teams

Team F	227
Team G	379
Team H	355
Team I	227
Team J	604
Team K	184
Team L	179
Team M	313
Zeller Baseline	244

*Multiple defects removed

4 20 85

Advanced Techniques Demonstrated

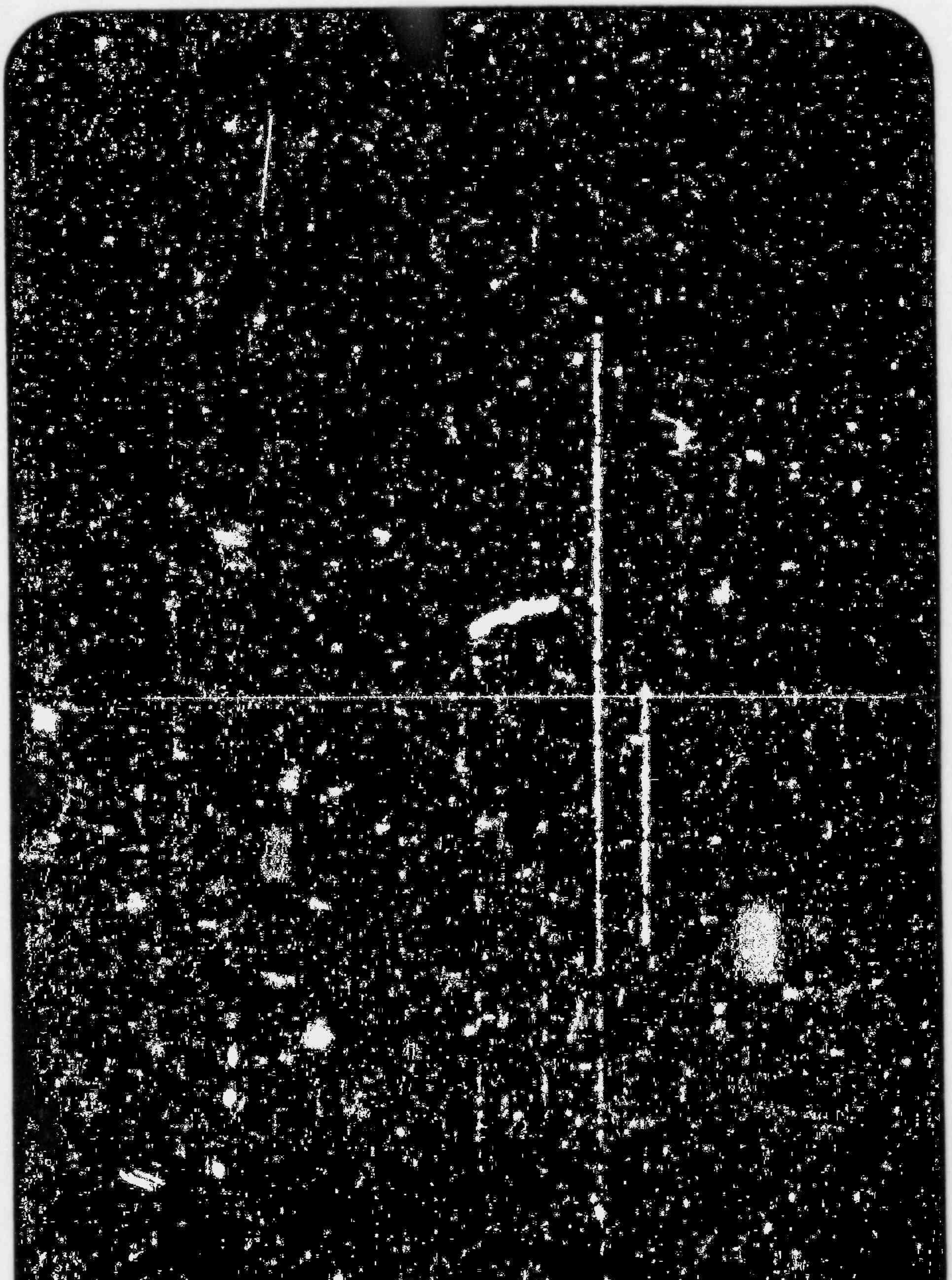
- NRC - Oak Ridge Multiparameter Eddy Current System
- EPRI-J.A. Jones NDE Center - 1) Zetec MIZ 12 with double mix to eliminate conductive deposits, support plates, dents and 2) Zetec MIZ 18 digital multifrequency system (high signal saturation potential) and 3) special IGA 8X 1 probe
- Mitsubishi Heavy Industries - Japanese multifrequency eddy current system plus examinations with special U-bend and tube sheet crevice probes
- UTL-KWU - German multifrequency system plus rotating point eddy current probe and rotating ultrasonic probe
- Intercontrole/Framatome/CEA - Demonstration of new French ultrasonic and rotating point eddy current imaging systems
- Babcock and Wilcox - Profil 360 profilometry
- Zetec - Zetec MIZ 15 profilometry

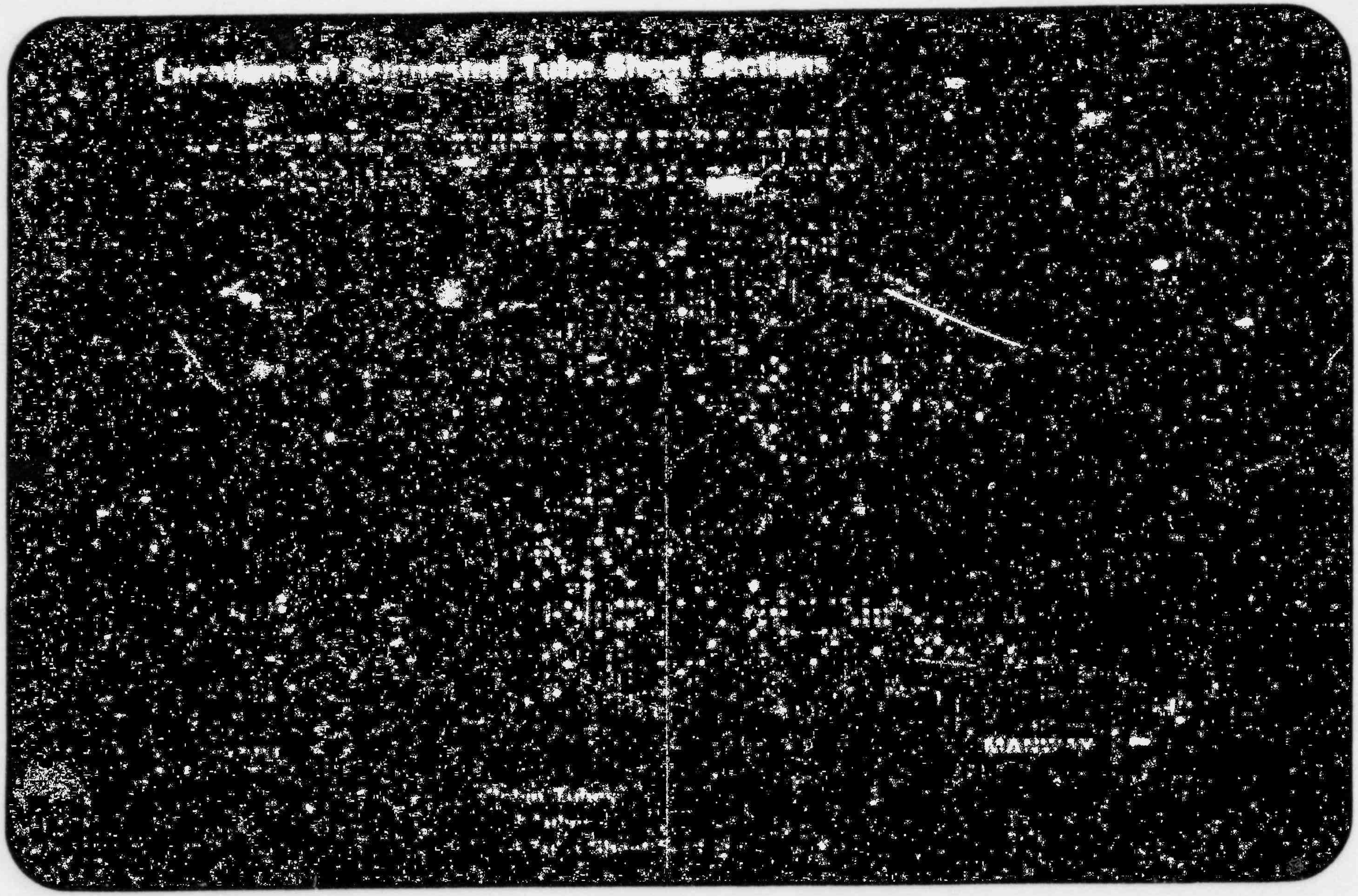
Advanced Technique Results (Non Validated)

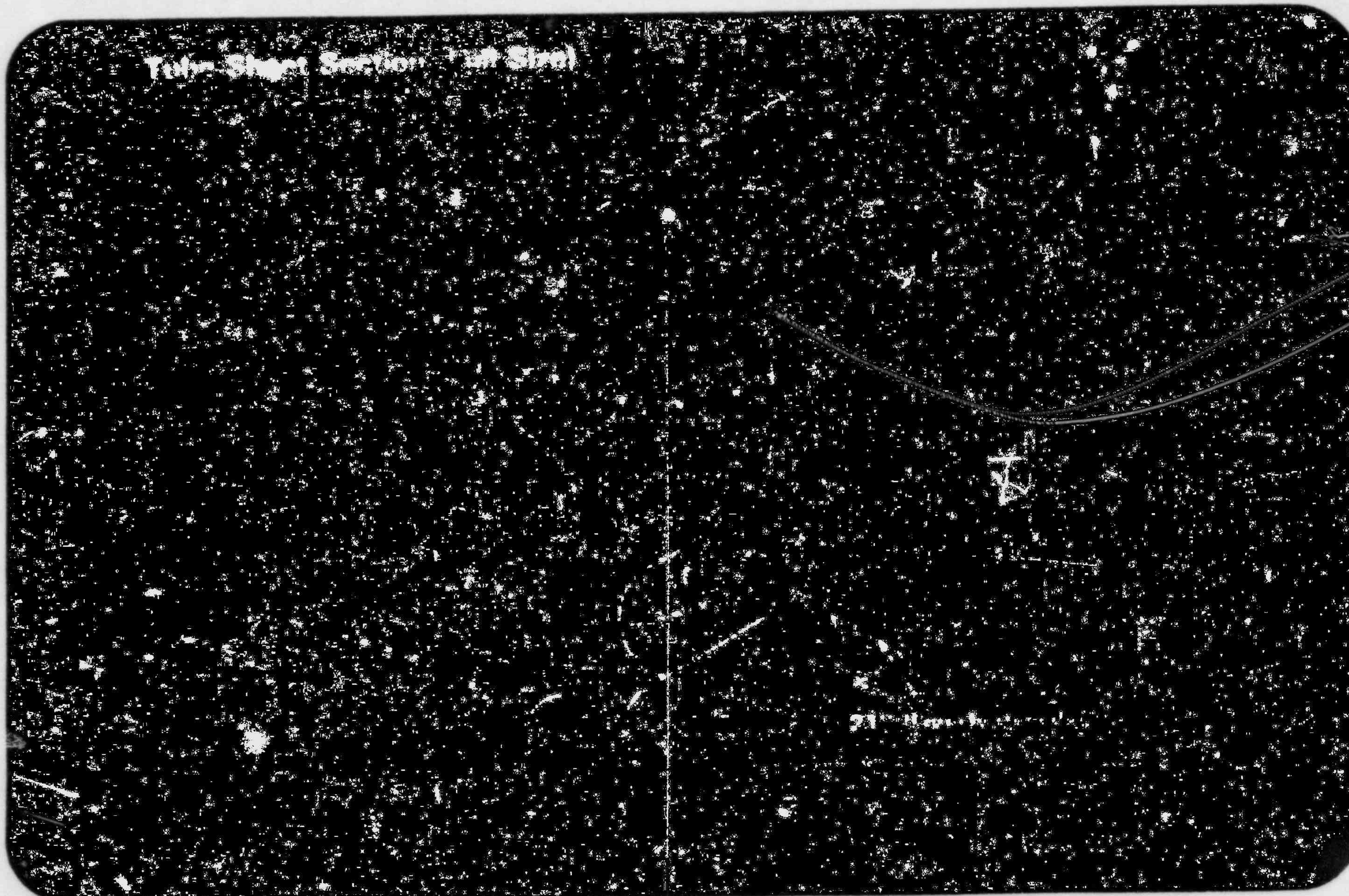
- Japanese examination found several U-bend indications not reported by other teams
- Japanese reported IGA, only reported by one other team
- KWU used ultrasonic test sizing to modify eddy current size results originally indicated, increasing depths by ~ 10%
- KWU ultrasonic system only one to define pitting above top of tube sheet, initial validation shows this system was much closer on sizing
- Profile 360 system identified a range of dent profiles to allow investigation of strain as a condition for tube plugging, versus tube constricture

Validation

- Tube sheet section removals
- Tube support plate removals
- Tube pulls
- Post removal nondestructive examination
- Post removal destructive metallographic examination




Locations of Suggested Tube Sheet Sections


Tube Sheet Section

Summary of Current Status

- There appears to be a relatively wide range in reported results from standard practice eddy current examinations
- Most of the difference in reported results appears to be due to inconsistency in the analysis of data
- Initial validation studies indicate that all eddy current teams undersized the depth of pitting above the top of tube sheet. All teams reported an indication. The removed tubes, with approximately 85% through wall pitting, remained in service at the end of generator life
- Initial nondestructive examination after removal of the pulled hot leg specimens has indicated IGA (up to 80% of wall) near the top of the tube sheet and 3" below the top of tube sheet. This is currently being metallographically determined, however only one eddy current team indicated any IGA in this specimen
- Ultrasonic examination gave an apparent improved (and accurate) sizing of the pitting defects

Future Activities

- Remove all round robin tube sections containing a reported defect indication (825 sections)
- Nondestructively characterize then selectively destructively characterize defected tube sections to determine:
  - A) The reliability of nondestructive techniques in detecting tubing defects
  - B) The accuracy of nondestructive techniques in sizing tubing defects
- Test selected defected tubes for remaining tube integrity via burst testing at operating conditions
- Using NDE reliability and accuracy information and remaining tube integrity models, develop models for steam generator in service inspection to ensure safety
- Provide inputs into tube plugging criteria