UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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In the matter of:	:
COMMONWEALTH EDISON COMPANY	: Docket Nos.: 50-454, 50-455
(Byron Nuclear Station, Units 1 and 2)	:

- x

10th Floor Conference Room
Nuclear Regulatory Commission
Maryland National Bank Building
Bethesda, Maryland

Wednesday, July 7, 1982

The deposition of EMMETT MURPHY and JAI RAJ RAJAN, witnesses called by counsel for the Intervenors, convened at 9:07 a.m., pursuant to agreement, taken before Ann Riley, a Notary Public in and for Montgomery County, Maryland.

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P R O C E E D I N G S

MR. GOLDBERG: As I indicated in my July 2nd letter to Diane Chavez of DAARE/SAFE, this deposition of Emmett Murphy and Jai Raj Rajan is being conducted pursuant to Ms. Chavez' June 29th telephone request.

I understand from Ms. Chavez that she has contacted the local Illinois parties concerning this deposition, including the legal representative for the Rockford League of Women Voters, who is not in attendance at this deposition.

I agreed to make Mr. Murphy and Dr. Rajan available for deposition concerning their June affidavit on DAARE/SAFE Contention 9-C, despite the fact that the formal discovery period has closed.

There is obviously greater latitude normally accorded the topics for deposition, but I would expect that the deposition will focus on the affidavit for which these gentlemen are prepared to respond.

Any evidentiary objections I may make are for the record. I will not instruct the witnesses to decline to answer on the grounds that I have interposed an evidentiary objection, and I would not encourage discussion on those objections, nor do I anticipate a necessity for inviting any ruling by the presiding officer.

I understand also under the rules and practice that objections are not waived if they are not made, and I will try not to disrupt the orderly flow of the deposition unnecessarily.

And with that, I have nothing further.

MR. JENKINS: Gentlemen, the purpose for our gathering here today is to just get some information. We are not going to be asking any trick questions or anything like that. All our questions are going to be in the area of your expertise.

MR. GALLO: You have to talk louder.

MR. JENKINS: I'm sorry.

If there are any questions I ask that you don't understand, please just tell me and I will rephrase the question or ask it in a different manner.

Now I've never asked questions of two witnesses at once, and I'm not sure where your areas of expertise overlap. So if either of you feels more qualified to answer than the other, then, please, that person answer the question. And if the other person has some information to add to that, please go ahead and add that information at the end of the previous person's discussion.

Whereupon,

EMMETT MURPHY

and

JAI RAJ RAJAN

were called as witnesses by counsel for the Intervenors and, having been first duly sworn, were examined and testified as follows:

E X A M I N A T I O N

BY MR. JENKINS:

Q Gentlemen, a couple of brief questions relating to technical expertise in the licensing of these plants.

First of all, what areas of technical expertise are required for a thorough understanding of steam generator problems?

A (Witness Murphy) I'll do this one first, Jai, and then you add to it.

For a thorough understanding of steam generator problems, it requires a knowledge of a variety of technical disciplines; mechanical, materials, chemistry, and systems. Understanding of how the reactor plant works.

Q Which of these areas do you individually have expertise in?

A I personally am knowledgeable regarding the history of steam generators to date, surveillance programs to monitor steam generator performance and inspection techniques. I have considerable background in structural mechanics, and I have been involved in the review of steam generators, primarily operating steam generators for the past three years.

Q Dr. Rajan?

A (Witness Rajan) My involvement has been primarily in the area of mechanical engineering aspects of the steam generator internals modifications, and I have been involved in development of plugging criteria for degraded steam generator tubes. And my involvement has been for a number of years in this area, off and on.

Recently, the last two years, has been more than the previous years.

Q What is you gentlemen's specific authority and expertise with reference to Byron's SER?

A (Witness Murphy) In terms of authority -- I'm not sure I understand the question. Speaking for myself, I felt qualified to respond to the contentions set forth and there were others that could have also responded, were I not involved. I'm not quite sure I understand the full breadth of the question.

MR. GALLO: Can we go off the record?

(Discussion off the record.)

WITNESS RAJAN: I don't know if I can call it an area of expertise, but I did review the steam generator design for the Byron-Bradewood plant, as I did for several other plants, and our review was primarily within the mechanical engineering aspects of the steam generator side.

This includes the internals, primarily, and does not get involved with the systems aspects.

BY MR. JENKINS:

Q Could you define for me what is an unresolved safety issue?

A (Witness Murphy) One, an issue for which perhaps the safety significance is not fully understood, and one where perhaps some question exists as regards to whether additional regulatory action or study, perhaps, should be devoted to this particular issue.

Basically I think an unresolved safety question is one of whether or not the Staff is -- we have sufficient -- sufficient regulatory control over the situation, over the issue.

Q Are there any -- I'm sorry.

A (Witness Rajan) I have nothing to add to that.

Q Are there any safety issues which might be described as unresolved that might be unacceptable? I'm not asking for specific safety issues, but is that contemplated within the definition of a safety issue?

A (Witness Murphy) In terms of -- I'll go on and I'll answer your question in a moment.

Q Okay.

A But let me mention the fact that at the NRC, we have a specific organization assigned to perform unresolved safety issue studies, and I'm sure they can give you a much more polished and perhaps a better philosophical feel for what these -- how these issues interrelate to existing regulatory approaches to reactor plants, or nuclear plants.

Would you mind repeating the question? I'll answer your question.

Q Right. Are there any unresolved safety issues which might be deemed unacceptable over the life of the plant?

A With regards to steam generators, in my opinion, there are areas that certainly deserve further study by the Staff. These studies are in progress, but in my opinion, pending the outcome of these studies, we have sufficient -- we have sufficient requirements and checks and balances in place to ensure the safe operation of the unit.

In my opinion, it is important that we not sit on our present position indefinitely. I think it's important that we continue to study the matter and identify any of the areas where improvements should and can be made.

Q Okay. I want to discuss briefly some of the history of tube degradation problems.

How many Westinghouse reactors are currently operational, approximately?

A On the order of 40.

Q How many of that number have had tube problems of some sort?

A I think I have seen -- somebody has actually gone down a list and counted it, a number of approximately

40, so therefore my first answer to -- maybe change my first answer a little bit. We must have a bit more than 40 operating Westinghouse -- no, wait a minute.

MR. UDELL: There are 49 PWRs.

WITNESS MURPHY: I think you are correct. I don't think we have that many Westinghouse plants. I did bring a document, if it were necessary I could physically run down a list of plants and count the number involved.

Let me just say that, one, the number of PWR -- operating PWR plants is well known, and the number of plants that have reported anywhere from minor to severe degrees of tube degradation has also been reported in NRC reports. I think we are talking on the order of in excess of 40 PWR systems, total.

A major share of those, off the top of my head, 60 percent, might involve Westinghouse plants, and the vast majority of PWR plants have reported anywhere from very minor degrees of tube degradation ranging to severe degrees of tube degradation.

BY MR. JENKINS:

Q What proportion is that? Would you repeat that?

Approximately?

A (Witness Murphy) Does 60 percent sound reasonable to you, Jai?

Let me refer you to NUREG 0886. I don't think it serves any purpose for me to speculate.

Q Please do.

A NUREG 0886 provides a listing of all operating PWRs, and a summary of what kinds of tube degradation have been experienced in each of these facilities, ranging from very minor degrees of tube degradation to the most severe forms of tube degradation.

Q Would you refer now to that document and tell me the number of Westinghouse plants that have had tube degradation problems?

A I count 29.

Q Okay. And how many Westinghouse reactors are there?

A Operating reactors?

Q Yes.

A Nine, eleven -- I count 31. That doesn't say I may not have missed one or two.

Q Are you aware of any problems since the publication of that document?

A Yes.

Q And how many?

A I'd have to give some thought to that answer. Westinghouse plants -- I'm not aware of any plants which have not experienced -- previously experienced degradation, now having experienced degradation, with the exception of McGuire. McGuire, I think, since the publication of this document, has reported some very minor degrees of -- some degree of tube degradation as a result of the preheater vibration problem.

Q So virtually all of the plants have experienced some sort of problem; is that fair to say?

A Yes, ranging from very minor degrees to very severe degrees.

Q One of the measures that your affidavit suggests to minimize tube degradation problems is AVT water chemistry. How many of the Westinghouse reactors employ AVT?

A All but two.

Q How many of this number have had some type of tube degradation problem, to the best of your knowledge?

A Well, to the best of my knowledge, the vast majority have had tube degradation problems.

Q Your affidavit lists, along with AVT, three other measures to deal with tube integrity problems. Have these measures been employed at the other Westinghouse units as well?

A Let me refer to my affidavit.

Regarding item 2, improved controls and monitoring secondary water chemistry, I'm really not the one to address what steps, what specific steps the industry in general has taken in regards to operating plants to improve secondary water chemistry control.

Plants that have -- well, in-service inspection requirements in accordance with various criteria, plants which have been licensed in the past several years, have been subject to the requirements of the standard technical specifications, in addition to the requirements of -- the criteria of Regulatory Guide 1.83 and the ASME Code.

The standard tech specs represent a general upgrading of steam generator tube surveillance requirements with respect to Regulatory Guide 1.83 and the Code.

Q And the fourth item there, limiting allowable primary to secondary leakage rate?

A Leakage rate limits are specified in technical specifications. The newer plants, subject to the standard technical specifications, have the most stringent limits on -- tech spec limits on primary to secondary leakage.

Let me just add there that where a plant runs into severe difficulties with corrosion and so forth, it is not at all uncommon for the Staff to require much more stringent limits on allowable primary to secondary leakage, as a special license condition, which are lower than the actual tech spec limits.

Q Are there any plants that you are aware of that have employed all four of these measures?

A Yes. In saying yes, it is based upon my understanding that many of the newer plants have implemented improved controls and monitoring of secondary water chemistry. So my response to that would be that some of our more recently licensed plants would have implemented all four of these items listed.

Q What is the record of tube problems in these plants?

A Mixed. One thing to keep in mind is that the newer plants have had the least amount of operating experience and short-term trends may or may not be indicative of long-term trends. Just looking over a list of plants which we have licensed in the past -- oh, since 1980, none of these plants have experienced any extensive degradation, tube degradation problem.

Q But they have still experienced some degradation problems?

A Some have experienced some degradation problems. Sequoyah 1, for example, has experienced minor amounts of denting. Salem Unit 2 has not reported -- Sequoyah 1 was issued a license in September 1980. Salem Unit 2, it was issued a license in April 1980, and has not reported any degradation to date.

That is not to say that they don't have any degradation. It just isn't significant enough to be reportable to us. Neither of these two plants have plugged any tubes as of last February.

Q Are these plants at 100 percent operation?

A There is nothing in the steam generators that would preclude their operating at 100 percent. I can't speak to other constraints.

Q You earlier mentioned the imposition of license conditions where problems have emerged. Instead of imposing license conditions, why didn't you simply revise the technical specifications?

A That is sometimes done, too, where we feel that a common vehicle for changing the requirements is to revise the tech specs.

Another vehicle that has been used in the past has been to place special license conditions on the plant which govern it, which, you know, are the governing requirements, as long as the conditions are in effect.

Q In your affidavit you state, and I quote:

"Several steam generator design features are employed at Byron to limit the regions where deposits could tend to accumulate and possibly cause corrosion."

What are these design features?

A One that immediately comes to mind is the elimination of the tube to tube sheet crevices. These crevices -- they've added a design feature to enhance the coolant flow, secondary coolant flow across the tube sheet to minimize the buildup of sludge deposits on the tube sheet. A traditional location of corrosion problems in steam generators has been within the area of sludge accumulation.

Those are two that come to mind immediately.

Q Are these features employed in any other plants?

A Tube sheet crevice -- tube sheet crevices have been eliminated at a handful of domestic plants, operating domestic plants. None of these plants have, to my knowledge, experienced any tube degradation, because the crevice has been eliminated. They are not subject to the crevice corrosion that earlier plants had been. There have been no reported difficulties for domestic units as a result of eliminating the tube sheet crevice.

Q That was my next question.

A With regards to the second part of the question, I'm not sure that we have any significant or any significant amount of operating experience yet.

Q In February 1982, the NRC issued a report entitled "Steam Generator Status Report." This report chronicled some of the problems of tube integrity, and in section 3 it states, and I quote:

"Short-term solutions to one problem may create other problems. Conversion from phosphate to AVT water chemistry, which minimized wastage and stress corrosion cracking but was followed by denting, is a case in point.

"Finally" -- I'm still quoting -- "it should be noted that the majority of plants under review for operating licenses have SGs of similar design to those currently in operation, so that the potential for SG tube degradation exists in these plants as well."

Is this steam generator at Byron similar to those currently in operation?

A It's basically similar. I believe one of the Byron units employs a D-5 steam generator.

Q We'll go into that. I just want to go into the history first.

A They are basically similar.

Q Based on the history of tube degradation problems, what is the probability of a tube degradation problem emerging at Byron within the operational life of the plant?

A I would say based upon previous experience, there is a high probability that they will encounter somewhere between minor and significant amounts of corrosion.

It's very difficult to project whether these will be minor problems or severe problems.

Q What is the probability of tubes needing to be repaired? Can you speculate on that?

A There is a possibility -- oh, repaired? You mean plugging?

Q Right.

A I would say there is a likelihood that Byron might expect to have to plug tubes during the life of its plant.

Q Is this a high likelihood?

A Based upon previous experience, it would seem yes.

Q After how many years is this likely to occur?

A Five years. Just a ball-parkish number.

Q Is it possible that a tube problem could emerge in the first year of operation?

A Possible.

Q How long is the vendor's warranty period?

A I don't know.

A (Witness Rajan) To this I would like to add that plugging of a few tubes should not be put out of perspective. It really does not affect the operation of the steam generator, and also it does not, in my judgment, constitute a safety problem. So if a few tubes need to be plugged during the early life of a plant, it should not be blown out of perspective.

Q Mr. Rajan, does plugging or sleeving the tubes require any worker exposure as a result of the work exposure?

A I do not have the numbers with me, but obviously during the early stages the exposure is less. As it goes to -- after the plant has been in operation for a longer period of time, and also the techniques have now been continuously refined to the point -- the plugging techniques, that is, that exposure is limited to a minimum, and plugging of a few tubes is within acceptable limits.

MR. JENKINS: Just a moment, please.

(Pause.)

BY MR. JENKINS:

Q I want to go on and talk about a comparison between the different models of steam generators. I have reviewed the schematics of the D-2, D-3, and the D-4, D-5 model, and other than the size of the preheater, I couldn't note any differences. Could you describe how the models are different?

A (Witness Rajan) Well, primarily the D-4, D-5 preheater section has a flow pattern in it, in which the flow, as it enters from the main feedline, hits an impingement plate and is directed downwards, and then it curves upwards again. This is referred to as a counterflow type.

The reason for the preheater is to -- the object of a preheater is to increase the flow within that region so as to increase the efficiency of the steam generator, and this is one way of doing it.

In the D-2 and D-3 type, primarily the flow hits an impingement plate and then is split almost half and half upwards and downwards, but the basic idea in both the preheater designs is to create a region of flow in which there is mixing and high rate of heat transfer.

Q So, then, other than the design of the preheater, there is no significant difference between the designs?

A I would say that it is a significant difference in one sense: In the D-4, D-5 case, the impingement plate directs the flow downwards, and in that sense its effects on the steam generator tubes would be different from those that the tubes would experience in the D-2 and D-3 model, because in that case there is a high cross-flow component. The flow, after hitting the impingement plate, goes upwards and towards the tubes.

A (Witness Murphy) It's my recollection that D-5 steam generators employ stainless steel support plates as opposed to carbon steel support plates of earlier designs. With stainless steel support plates, we would expect the D-5 steam generators to be less susceptible to denting.

Q Your affidavit discusses flow-induced vibration and subsequent wear of tubes and preheater. You state, and I quote:

> "The tube excitation mechanism appears to be a combination of a threshold type of fluid elastic instability and turbulent buffeting."

Would you please define those two terms?

A (Witness Rajan) In fluid elastic instability type of phenomenon, if a tube is excited by a range of frequencies, it responds to a very narrow band of frequencies. And when those frequencies are -- when those frequencies excite the tube, this narrow band of frequencies, the tube goes into a rather violent mode of vibration.

In turbulent buffeting, on the other hand, the tube responds to the exciting forces at all frequencies and as the power level increases, the buffeting forces increase, and therefore the tube vibrations increase, and in such a case the vibration of the tube increases with the power level.

Whereas, if the tube is excited by turbulent -- by fluid elastic instability, it is possible it may not experience violent oscillations and vibrations at other power levels, except at which it goes into resonance.

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Q What happens if there is vibration and subsequent wear of tubes in the preheater?

A Some vibration is obviously acceptable. It is only when the vibrations of the tubes -- as a matter of fact, all tubes in the steam generator vibrate to cross-flow velocities and axial velocities. So that is some level of vibration which is acceptable and will not create any damage over the life of the plant.

However, if the tube, for some reason, goes into a violent state of vibration, only then the wear rates -- the wear rates as a result of the tube hitting the support plates-- the wear rates increase to such levels that degradation is rapid and can lead to a tube failure in a shorter period of time than it is designed for.

Q Has this ever happened in an emergent situation?

A (Witness Murphy) What?

Q An emergent situation.

A Emergent?

Q Creating an emergency.

A We have not, to my knowledge, had a tube failure rupture, or a rupture event, rather, as a result of tube vibration. There have been some large leaks as a result of tube vibration, but not one which we formally classified as a tube rupture event.

Q Are you aware of any that may have occurred in foreign facilities?

A Not as a result of vibrations, that I'm aware of.

A (Witness Rajan) Are you talking of a leak, or are you talking of a severance?

A (Witness Murphy) Emergency -- the emergency -- we're speaking of rupture events.

A (Witness Rajan) You're talking of severance of a tube. I agree with Emmett.

MR. JENKINS: Just a moment.

(Pause.)

BY MR. JENKINS:

Q Mr. Murphy, you can go ahead. I was going to ask you about Ginna. You can go ahead and make your statement on the record.

A (Witness Murphy) Okay, we'd like to amend our previous answer to say at Ginna the tube that ruptured at Ginna did not rupture directly as a result of vibration. Vibration did play a role.

A (Witness Rajan) But the primary mode of failure was another mechanism.

A (Witness Murphy) Vibration did play a role in transferring -- in the overall failure scenario which was initiated by foreign objects and through a complicated sequence of events, led to excessive wear and pressure burst of the failed tube.

A (Witness Rajan) We were responding, or I was responding primarily in terms of the flow-induced vibration phenomenon and the fluid elastic and the turbulent buffeting type of phenomena. But as Emmett pointed out, at Ginna, vibrations had a role in tube failures, although not the primary cause of failure.

Q What was the nature of the problem with the Ringhals plant in Sweden, if you're familiar with that?

A Primarily flow-induced vibrations acting on the tubes in the vicinity of the preheater section. And as a result the tubes impacted against the support plate, the holes in the support plates, and there was a high rate of wear in those regions. And I believe one tube was degraded to the point, or worn down to the point that it started to leak.

Q Was there any warning of this leakage?

A As far as I know, it was not a failure that resulted in the severance of the tube. When the tube wall wore down to the point that there was a throughwall area of leakage, that obviously provided a warning to shut the plant down, because of the high leakage. And in that sense, there was warning.

Q But there was no warning of the problem emerging or arising?

A (Witness Murphy) [illegible] not prepared to answer whether or not the leak may have grown from very insignificant amounts slowly to something that was detectable at plant shutdown, or whether the tube was penetrated in one instant leading to a sudden small leak, if you will, we can't answer that question without checking.

It was a relatively small leak, though, when you are comparing it to Ginna or Point Beach or something like that.

Q Has there been a design fix for this problem in the D-2, D-3 model steam generators?

A (Witness Rajan) In my judgment, there is a design fix which is in an advanced stage of test and evaluation, and from what results that I have seen, I am convinced that this problem will be taken care of.

Q Has this design fix been implemented in any plants in operation?

A No. It has -- the first domestic plants during which it will be implemented are scheduled in September or October.

Q Describe the nature of this design fix.

A There are a number of modifications in the preheat section that are being -- they have been finalized, and these involve -- the basic change is the replacement of the impingement plates by a manifold, internal manifold, which has two double walls and a large number of holes in both the plates, and the object of this impingement -- internal manifold is to produce a uniform flow velocity at the first row of tubes beyond the inlet region, which is the 49th row.

And as a result of this uniform velocity, the flow-induced vibrations and the turbulent buffeting on these tubes, it has been shown has dramatically decreased.

Q How long would it take to retrofit this to existing operational plants?

A I do not remember the exact periods involved, but the time required for tooling and installation is within weeks.

Q Will this design fix totally eliminate all problems with tube degradation?

A No, this fix is aimed at the flow-induced vibration phenomenon that is experienced in the D-2, D-3 models only. It is not expected to, for example, have any effect on corrosion -- stress corrosion cracking or other denting or other problems.

Q I see. Okay. That's a good clarification.

Will these design fixes be applied to the steam generators used at Byron?

A No. As I pointed out,this fix is directed at the model -- is designed only for the D-2 and D-3 steam generator design -- steam generator models and will therefore not be applicable for Byron.

MR. JENKINS: Excuse me just a moment.

(Pause.)

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BY MR. JENKINS:

Q As the denting and corrosion problems occur in plants, as they get worse in the plant, is it necessary to reduce the operating capacity of the plant to lessen the possibility and the probability of new ruptures occurring?

A (Witness Murphy) Would you repeat the last part of the question, please.

Q Yes. Is it necessary then to decrease the operational level of operating of the plant in order to reduce the possibility of rupturing?

A Is it necessary to reduce the rated power of the plant to reduce the potential for accidents?

Q Right.

A No. Once you get into large-scale plugging of steam generators, ultimately you may reach the point where you have physically removed such a substantial part of your heat transfer area, your available heat transfer area, that you do affect your capability to produce heat and power, but does not affect the propensity for -- the level of plugging, as Jai Rajan pointed out earlier, does not affect the potential for a rupture or a failure.

Q How did Westinghouse test its D-2, D-3 steam generators prior to its use in a nuclear facility?

A (Witness Rajan) Basically, the design is verified on a computer model which simulates the flow phenomenon and the thermal hydraulics within the tube -- within the steam generator, and this is done with the aid of several highly sophisticated computer codes.

In addition to this, some verification is obtained with scale models.

Q How did those test results correlate with in-plant experience?

A This particular phenomenon, which occurred at Ringhals and McGuire, was obviously not predicted by the computer codes and the scale modeling tests.

Q In your expert opinion, why do you think there is a disparity between the predicted results and the actual results?

A Well, for one thing, we have to recognize that this phenomenon is occurring only in a limited area of the preheat section, and is not something that the conventional thermal hydraulic codes used in the design of the steam generator.

They would predict -- as I pointed out earlier, there is a high degree of turbulence and this was purposely created to extract more heat in that region.

So, with this high degree of turbulence, some tubes were excited into resonant modes which was not -- which had not been anticipated.

Q Your affidavit states that Westinghouse will do "extensive analyses and tests, including large scale

model tests" on the D-4, D-5 generators.
How large a scale?
A I believe it's the -- the preheat section is being -- it's a two-thirds scale model, the largest one. Although it's entirely possible to obtain meaningful data from smaller models.
Q Does NRC plan to empirically verify these tests?
A What do you mean by empirically, empirically verify?
Q Well, are you planning to run your own tests, or are you planning to look at the data and put it into your own model?
A As far as I know, there are no such plans.
Q What is your level of confidence, based on your expert opinion, that Westinghouse's results can be extrapolated to actual use experience?
A Now are we talking of which?
Q D-4, D-5.
A D-4, D-5? I have not seen any results of model test data on D-4, D-5 yet, although I understand that Westinghouse is in the process of conducting analyses and test data for the D-4, D-5 models in a similar fashion

as they have done for the D-2 and D-3.

My understanding is that these will be available by the end of the year.

A (Witness Murphy) I think it's important to note that once it was recognized they had a problem in the preheater, that they were able to determine -- they were able, through analysis and tests, to demonstrate how the wear mechanism worked. With the ability of hindsight they were able to demonstrate, yes, you know, you certainly would expect a wear process to take place as a result of vibration.

So they have been able, by analysis and test, to say, yes, under these conditions you will get wear and vibration, and it explains the wear patterns that we are actually observing in the field, and being able to do this gives you confidence that they have a model where, if they adjust certain parameters, they have the tools necessary to evaluate the effect of these parameters on the overall performance of the preheater section.

A (Witness Rajan) To this I would add that one nondomestic plant has been instrumented which has a model D-4 design, and data obtained from that is providing useful information as to which modifications would be effective, and which may not be of that great utility.

Q Is this domestic or foreign?

A One foreign plant.

Q Let's jump ahead to that. Are you aware -- you're referring to the KRSKO plant in Yugoslavia?

A Yes.

Q Are you aware of the current operating status of that plant?

A I am aware that it has had about 2000 hours of operation, approximately, and there was negligible -- there was no detectable degradation found as a result of measurements in the tubes that normally would be affected.

The instrumented data from that plant indicated that there is a higher level of tube vibration than is expected. So while there seems to be evidence that this problem is there, there is no evidence of degradation of the tubes as such -- detectable level of degradation of the tubes as such, so far.

Q Eddy current testing, I understand, is capable of determining whether thinning has occurred only past 20 percent; is that correct?

A (Witness Murphy) That's not exactly correct. At the support locations where the degradation is occurring, we would be a little hard-pressed to say what the threshold of detectability is, but I think it is something less than 20 percent. Certainly 20 percent, I believe, should be detectable.

Q Is 2000 hours of operation in the KRSKO plant sufficient to establish whether you have reached that threshold of being able to determine if thinning has occurred?

A (Witness Rajan) Well, definitely it indicates one thing, that the mechanism of degradation is not so severe that it would manifest itself in a short while. But it certainly does not preclude degradation if the operation were continued.

Q If repairs are an eventuality at Byron, which I believe is a fair assessment of your earlier statement --

A (Witness Murphy) Plugging? Plugging repairs?

Q Right, plugging or sleeving repairs. -- would their retrofit be a radioactive task?

MR. UDELL: I'm sorry, excuse me for a minute.

(Discussion off the record.)

MR. JENKINS: Okay, I retract that question. We got an answer to it previously.

BY MR. JENKINS:

Q Let me ask the question this way:

Is a design fix inevitable at Byron, do you think?

A (Witness Rajan) Are we talking -- a design fix -- okay, now we are talking of the tube vibration problem?

Q Right.

A I believe there would be some fix. Now what that exact fix would be, it's not -- there are several options available, and I don't know exactly what form that fix will take.

Q Okay. Now if the plant starts up before you are able to establish that design fix, would that then be a radioactive task?

MR. GOLDBERG: Excuse me. I don't know if the witnesses understand the question, but what would be a radioactive task? Starting up the plant without implementing a vibration fix?

MR. JENKINS: Doing the design fix after the start-up of the plant.

MR. GOLDBERG: Okay.

WITNESS MURPHY: It would seem reasonable to assume that installation of a fix after start-up would involve some amount of occupational exposure.

BY MR. JENKINS:

Q Okay, as a summary question here, in light of the variety of tube integrity problems, in light of Westinghouse's track record in testing, and in light of NRC's own observation that solutions to one problem may create other problems, why are you recommending that Byron be permitted to operate before ultimate resolution of these various issues?

A (Witness Murphy) I'm not sure that we've --

A (Witness Rajan) Let me say this: That it is our -- that we anticipate that the fix will be in place before Byron goes into operation.

A (Witness Murphy) But we may want to discuss this. I understand there is some question on this point which we may want to discuss later, but the Staff has not made its conclusions regarding the preheat -- regarding the preheat problem and what an acceptable basis for start-up of the plant will be.

The Staff has committed in its SER to review it, the generic problem, as it relates to Byron, and it has not concluded -- and in the SER, that a fix is a necessary condition for start-up. It has not made any conclusion whatsoever as yet. The Staff must make a conclusion and a finding before start-up. We would issue, to the extent that -- if it were to turn out that a fix could not be implemented prior to start-up, then the Staff would have to evaluate the acceptability of a program for interim operation pending a fix.

MR. GOLDBERG: Mike, I would interject, your question was a broad summary question, and if I understood the answer, when the term "fix" is used, are you talking about now a fix for the tube vibration phenomenon?

WITNESS RAJAN: That's precisely what we are talking about.

MR. GOLDBERG: So I don't know if you got an answer to your question. Their remarks were devoted to a position of tube vibration. I don't want to unduly confuse the process, but I think they confined their answer to vibration, their position on tube vibration.

MR. JENKINS: I'm satisfied with the response.

MR. GOLDBERG: Okay. Fine. Fine.

MR. JENKINS: Hold on just a moment, please.

(Pause.)

BY MR. JENKINS:

Q When a steam generator tube cracks, ruptures or leaks, what is the potential for radiation to escape to the environment?

A (Witness Murphy) When a tube leaks or ruptures, there is a potential for radiation to escape to the environment. I am not the one to -- I cannot provide any sort of expert testimony regarding the amounts of doses, offsite doses as a result of leakage or ruptures.

Q Let me ask it this way:

How large a leak would be necessary before radiation could leak to the environment?

A I can only give you my nonexpert understanding, and that is that you will get some amount of radiological release with leakage. But once the -- once radioactive water gets into the secondary, the pathways are available for the radioactivity to get into the environment.

Q Have there been any tube degradation problems which resulted in radiation leaks?

A Oh, I think -- again I think if you want to discuss radiological releases to the environment, perhaps I think you need a different --

Q Different set of experts?

A -- different set of experts.

A (Witness Rajan) The answer is yes, but we are -- I don't think we can quantify the releases.

Q Okay.

A The answer to your question is yes. Yes.

A (Witness Murphy) We'll say that the Staff does have regulations regarding acceptable offsite releases, and that these are enforced.

Q Well, now, I have a number of other questions here relating to radiation leaks and safety design and so forth, and I understand that you are not experts in this area, so if you could just -- I want to run through them and see if I can get an answer to the best of your knowledge, and just advise me if I'm going out of bounds here.

Are you familiar with the safety response systems in the event of a ruptured steam generator tube? And if so, could you please describe it?

A Jai, do you want to take a stab at that? I'm not, in terms of how the plant is brought to a cold shutdown, in event of a rupture event, I'm not prepared to comment. I cannot comment on that.

A (Witness Rajan) I am also not a systems man.

Q Is there a potential for any multiple failures, for example, the pilot operated safety valve sticking open? Is there a potential for that to occur during a tube rupture event?

A (Witness Murphy) Once you have a tube rupture event, and you go into emergency shutdown situation, I would have to assume that there is something -- there is no reason why something else couldn't necessarily go wrong during the shutdown.

Q Do you know if this has ever happened in multiple failures of some sort?

A I would assume that it has. I would refer you to an interesting document to review in that regard, would either be the Ginna report put out by the Staff that described the shutdown in excruciating detail. Any problems that were experienced with valves and the like are all in there.

There have been -- we have had a number of these rupture accidents. The first three are described in very -- in great detail in a NUREG report entitled "Evaluation of Steam Generator Rupture Events." I don't know the NUREG number right off the bat. And the Ginna event was analyzed separately in a more recent NUREG.

Q Do you know if there is any potential for secondary to primary tube leaks under certain accident conditions?

A Under certain accident conditions, if you have a tube failure, there is a potential for secondary to primary tube leaks.

Q What could that result in? What could that leak do, do you know?

A Ultimately if one were to have excessive secondary to primary leakage, you could affect your capability to adequately cool the core. It's an ultimate consequence of excessive leakage.

Q Now my next question --

A Well, not ultimate, but it's a consequence.

Q My next question I think you probably have a little bit more expertise in. Is there a potential for more than one steam generator tube to fail at or about the same time?

A During normal operation, steady state normal operation, I cannot imagine that we would get more than a single tube failure. Ruptures obviously have happened, tube ruptures have happened four times in this country. They have been single failures. I cannot imagine -- to get more than one tube failure, you need a triggering event, you need a transient of some sort.

Otherwise, under normal steady state conditions, the rupture will occur, assuming the degradation is out of control and not being adequately surveilled, it will occur in a random fashion.

Q But doesn't flow-induced vibration affect many tubes at once?

A Yes, but you would not expect that each tube would be degraded to exactly the same degree, such that the failures would occur simultaneously.

(Pause.)

Q Okay. Again another summary question here relating to safety and so forth.

In your expert opinion, do you agree with a statement of Mr. Harold Denton that there is no way to ensure that tube leaks will never happen?

MR. GALLO: I'm going to object to that question. Did you answer it?

WITNESS MURPHY: No, I didn't answer it.

MR. GALLO: There is no foundation that Harold Denton made such a statement. Do you want to try to do that?

MR. JENKINS: I think I'll just withdraw the question. It was more of a fun question, anyway.

MR. GALLO: All right. Well, then, I have a fun objection.

(Laughter.)

MR. JENKINS: Could I hear that?

BY MR. JENKINS:

Q Now I have a series of questions that are a bit more technical than what we've gone into so far.

Off the record.

(Discussion off the record.)

BY MR. JENKINS:

Q Okay, I have some questions about what to me at least are technical questions relating to corrosion.

What, besides deposition of corrosive products in the tube and tube-supporting annulus, contributes to tube denting?

A (Witness Murphy) What besides deposition of --

Q Deposition of corrosion products in the annulus.

A Neither Jai nor myself is well versed in the dynamics of corrosion. I can speak only in general terms. The dynamics of corrosion is not information I need to have to do my job. But magnetite -- the corrosion products you are referring to is magnetite, and it's the corrosion product that results from corrosion of the carbon steel support plates. The corrosion products aren't carried to these crevices from elsewhere in these plants. The corrosion product is a result of corrosion leaks in the support plate itself.

A (Witness Rajan) The denting phenomenon essentially consists of these carbon steel support plates that react adversely in a certain environment, and as a result of their interaction they put excessive stresses on the tubes, and they also can cause cracking within the support plate itself. And if the denting progresses, it progresses unchecked. Then the plate itself can be broken into smaller pieces, and that's an advanced stage of denting.

Q Is it necessary for the annulus to be completely filled with corrosive products for denting to occur?

A (Witness Murphy) Yes.

Q Has tube denting occurred in any plants using AVT water chemistry?

A Yes.

Q What design features dealing with tube denting have been used at other plants, and to what extent have you evaluated their effectiveness?

A For new generation steam generators, the support plate designs will be used employing different materials, stainless steel, different tube hole designs will be used to reduce the potential for denting.

Unfortunately, to my knowledg e -- well, we don't have any Westinghouse steam generators with these new features on line as yet, so we have no operating experience. So that answers the question.

Q What, in your opinion, is the combined effect of reduced water flow velocity and increased secondary water temperature and pressure relative to promoting corrosion at the annulus?

A   I think to answer that question, you'd want to refer to our corrosion specialists.

Q   Okay. What, in your opinion, is the cause of the flow-induced vibration problem?

A   (Witness Rajan) Flow-induced vibrations are a result of high cross-flow velocities. Either the feedwater or steam flow.

Q   And this is true of all the D-4, D-5 models?

u4 A   Yes.

Q   Have you been able to verify the Westinghouse findings alluded to in your affidavit that quote:

> "Vibration response in the preheater section is negligible for main feedwater flow rates, up to about 70 percent."

A   This is primarily based on data obtained at KRSKO. In that plant, some tubes were instrumented and their vibratory characteristics were obtained at different power levels, and the data from that plant seems to bear this out.

A   (Witness Murphy) Let me add something else:

Even if the work Westinghouse has done -- and what they have recorded has provided us with a certain amount of confidence that they are pursuing the right approach -- but even if they're wrong, if they developed a problem, it will be discovered probably most likely through eddy current inspection, or perhaps small leaks in the case of Ringhals, and if necessary, additional actions can be taken as the need arises.

I'd just like to make that point.

Q Mr. Rajan, what is your definition of a negligible vibration response?

A (Witness Rajan) Well, a negligible vibration response would be such that it would not cause -- it would not cause a wear of the tube as a result of its impacting with a support plate.

Q Is it possible that testing and power escalation process at Byron might fail to detect any vibration problems in the steam generators?

A If the tubes are not instrumented, then obviously there is no way to detect any flow-induced vibration.

Q Are you going to require instrumentation?

A Well, we are assuming that their fix will be available for Byron, and if we are convinced from the data that is provided to us that the fix is adequate, then we may not require instrumentation.

But in answer to your question, we have not ruled the option out that it may be instrumented.

A (Witness Murphy) Let me add, one, we do have another vehicle, that is eddy current testing. What the program will be for eddy current testing, of course, has not been reviewed by the Staff as it applies to Byron. But instrumentation, internal instrumentation, is one method by which one might detect the onset of vibrations. Eddy current testing is another.

Q What is the significance of determining the optimum combination of main to auxiliary feedwater flow rates, preheated water temperatures, tube support design, and tube length between supports that would result in a tube oscillation rate equal to the natural frequency of the tubes?

A (Witness Rajan) I'm not in a position to respond to that. I can see what you are asking, but I don't have the answers to it.

Q Is this part of the SER?

A I'm sorry?

Q Is this evaluation part of the SER?

A Not within the scope of the review that we conducted in the mechanical engineering branch.

Q What is the significance of determining the effects on dented steam generator tubes of a natural frequency drop by a factor of four to eight?

A (Witness Murphy) Would you mind repeating the question?

Q Sure. What is the significance of determining the effects on dented steam generator tubes of a natural frequency drop by a factor of four to eight?

A What is the --

MR. CHESNUT: Are you talking about dropping the frequency or the magnitude of the vibrations, or what?

MR. GOLDBERG: Excuse me, Steve. The witness wants the question clarified.

WITNESS MURPHY: Why is natural frequency dropping in a dented tube?

BY MR. JENKINS:

Q If you do drop the natural frequency, what is the effect on a dented tube?

A (Witness Murphy) How do we drop the natural frequency? The natural frequency is a property of the tube and its supports. It's a property of the tube system.

Q How is that affected by the denting?

A (Witness Rajan) It would tend to make it stiffer, if anything.

Q Is this a part of the SER? Is this something that is evaluated in the SER?

A (Witness Murphy) For new plants, not typically. Generally speaking, you know, denting per se does not adversely affect the -- operating experience does not indicate that denting per se adversely affects the dynamic response of the tube. If you have very severe denting, you get support plate cracking. For tubes near the periphery of the bundle you might effectively lose your lateral support.

A (Witness Rajan) Let me add to this. In the denting phenomenon, the support plate tends to crimp the tube and the supports and this results in a much stiffer system than one normally would have, when the tube is free to oscillate within the support plate holes.

So, as a result of denting, whatever happens to the natural frequency of the tube is not likely to be a matter of concern.

A (Witness Murphy) That's generally -- for most of the tube bundle, that is the situation. In row 1 and row 2 you have a peculiar situation where cracking can lead to islanding, the islanding effect, whereby effectively you are losing that lateral support.

A number of plants have run into this situation, and have therefore found it necessary to reanalyze the dynamic response of the tubes. We are now making the assumption of no lateral support at these support plates.

Q To what extent have you compared -- I'm sorry.

(Discussion off the record.)

BY MR. JENKINS:

Q To what extent have you compared AVT parameters, monitoring and control systems at Byron with those in other plants?

A (Witness Murphy) We personally have not done this. This again would be within the cognizance of our corrosion specialists who have responsibility for reviewing secondary water chemistry controls.

Q Are you aware if those other individuals have recommended any changes that Commonwealth Edison should incorporate in the Byron plant?

A They have made an evaluation which is described in the SER of the Byron secondary water chemistry program. There is also generic activity ongoing, both on the part of the industry and the NRC.

Q What has been, in your opinion, the significance of condenser leakage as a contributing factor to tube degradation?

A It's had a significant effect.

Q To what extent will this continue to be a factor?

A I believe it will be over the long term -- it will become a decreasing factor, primarily for the reason that to implement the improved secondary water chemistry controls and monitoring, it will be necessary to more closely monitor and control the performance of condensers to achieve the objectives.

Q Are you aware of any specific changes that have been required of Commonwealth Edison in its condenser materials and designs in its condensate clean-up system?

A This particular area did not fall within our area of cognizance. The condenser materials and so forth have been evaluated and are discussed in our SER.

Q Do you know when you will be issuing NUREG 0844?

A Is that Task Action Plan A-3, A-4, A-5? Then the answer is yes. No, I do not.

Q Do you have any prediction on the impact of any requirements that may be required -- you will be issuing it by January 1st of 1983; correct? I believe that's in your affidavit?

A I don't think so.

Q I read that somewhere.

A Let me check. I don't think I would have said that.

Q Well, then, are you familiar with any proposed requirements under that NUREG 0844?

A Yes. But, you see, this is -- at this point -- an internal -- we have a draft report that is being intensely reviewed and critiqued at this very moment, and the sponsoring organization -- for you to understand -- if you wish information regarding the exact status of the program and where it's going, I think you'd have to interview somebody from the sponsoring organization for the TAPS report, the generic issues organization in NRR.

Q Well, it's hard for us to go a hearing in which one of the methods of evaluating the success of the steam generator is its in-service inspection requirements, without knowing what some of those requirements are.

Can you describe some of the proposed requirements and what will be the impact?

A I'm not sure that it's really appropriate for me to do so, because right now the various recommendations are being proposed internally by the Staff and are being debated internally and discussed internally, and I cannot predict how this is going to come out, necessarily.

I can offer some judgment on that viewpoint, but I don't think it's really appropriate for me in this forum, because it might be prejudicial to the proceeding. So I think the Staff is doing an intense review right now and it is not for me -- I'm not the right person who should comment upon the status of the program. I think I'd be overstepping my areas.

MR. GOLDBERG: Let me try to ask a question, because I think your question and answer were two different things.

I understood you first to want some kind of broad indication of what the Staff proposals were in the prospective Staff document on Task A-3, 4 and 5. And then you switched and confined your answers to, I gather, what the proposed in-service inspection program was for Byron.

And correct me if I'm wrong, Mr. Murphy, is not that program discussed in the Staff SER?

WITNESS MURPHY: I was not addressing myself to in-service inspection requirements for Byron. I guess I was addressing what I thought was the question, what is the status of the TAPS issue, and where are we going with it, what will our recommendations be, and I can only respond that it's in progress and being reviewed very intensely by the Staff.

MR. GOLDBERG: Maybe we can get from the generic to the specific, because I think the witness is a little confused. I was confused whether you were talking about generic recommendations or specific plans for Byron.

BY MR. JENKINS:

Q Well, let me ask it this way:

Can you confirm for the record whether any generic requirements will have an effect on the plant capacity performance or in any other respect of the proposed requirements in this document on the Byron plant?

A (Witness Murphy) The Staff is considering -- has under consideration a number of recommendations-- it is part of A-3 and it is part of other generic reviews -- a number of recommendations that may have some effect on surveillance requirements, methods for improving the performance of the steam generators, the corrosion performance. We have a number of these things under consideration.

Q Have you required, or is it possible that these new requirements may require Commonwealth Edison to install radiation monitoring equipment at potential release points in the event of a steam generator tube rupture?

A I don't know the answer to that question.

Q What changes in Commonwealth Edison steam generator operational procedures or design in secondary water chemistry monitoring and control systems will be necessary to comply with this NUREG 0844?

A Well, I have no way to answer that question, because I don't know how it's going to end up. The generic recommendations are under -- being reviewed right now. I can't predict how it's going to end up.

MR. JENKINS: I have just a couple more questions more, but let me go off the record for a minute.

MR. MURPHY: Let me make one comment, just for the record:

Task A-3, A-4 and A-5 is a generic ongoing activity. It is not the-- there is a separate generic activity ongoing right now directly as a result of the Ginna incident, but it's getting into areas that were initially addressed by A-3, A-4 and A-5. So we have this generic program ongoing, too.

(Discussion off the record.)

BY MR. JENKINS:

Q First of all, I'm going to refer to a telegram from Mr. Goran Mandeus of the Swedish Nuclear Power Inspectorate, and I have a copy of this, Mr. Gallo, if you'd like. It's a telegram addressed to Mr. Joseph LaFleur of the Office of International Programs. It's titled "Urgent Telegram," and it says:

"This message should reach the persons who will be in telephone contact. . ." concerning the Almarz plant, and so forth, and I quote here from the second to the last page:

". . .it has been recognized that the curve has been computed for undamaged tube with adequate support in all baffle plates and not damaged tubes as the ones in Ringhals 3 with substantially increased clearance in the support plates and thus possibly possessing a larger free oscillating length."

MR. GOLDBERG: Mr. Jenkins, I'm not sure the witnesses have located your reference here, but can I ask that they be given a few minutes to familiarize themselves with this document?

MR. JENKINS: Sure.

MR. GALLO: Why don't we take a recess so we can read this thing?

MR. JENKINS: Fine.

(Recess.)

BY MR. JENKINS:

Q You have had a chance to familiarize yourself with the telegram from Mr. Mandeus. Could you describe, in your opinion, what is the significance of a drop in the natural frequency by a factor or four to eight?

MR. GOLDBERG: Are you referring now to a statement in the document, Mr. Jenkins?

MR. JENKINS: Yes.

MR. GOLDBERG: Could you just refer to that for me?

BY MR. JENKINS:

Q I will ad the last three sentences prior to paragraph 4:

"If the support in one or two plates is lost partly or in whole, the natural frequency may drop substantially. Approximately by a factor of four to eight. According to the theoretical model cited by Westinghouse, the threshold fluid velocity for instability drops by the same factor or down into the region of what can be described as idling power for the plant."

My question is: What is the significance of a drop in the natural frequency of four to eight?

A (Witness Rajan) I think that's a very large drop. When we are talking of a drop of the natural frequency of four to eight, we have to assume that the support plates are no longer effective at at least two locations, so that you have a much larger length of the tube now free to oscillate in its natural mode.

I cannot visualize losing support at least in support plates in order for that to happen, but if one were to make that assumption, then obviously the tubes would then have a different natural frequency in response to fluid elastic vibrations, but also be substantially different.

Q Go ahead.

A (Witness Murphy.) The Staff has considered the fact that with some wear of the tubes, there may be some effect on the rate of wear, that the rate of wear may not be constant -- that may not remain constant, as you wear away the surface of the tube.

In our monitoring and following of McGuire, we have taken the consideration into account in reviewing and approving of their interim operating program.

Q If this scenario were to occur, what would happen if the plant were not dropped to idling power?

A (Witness Rajan) Are we assuming --

Q The drop in frequency rate by a factor of four to eight.

A Is this drop being assumed for just one or two tubes, or a whole bunch of tubes?

Q Why don't you assess the response to that question for both circumstances?

A (Witness Murphy) Let me -- we have developed between Ringhals and Almarz and McGuire a considerable degree now of operating experience. We have a good idea of qualitative -- an idea of the qualitative relationship between operating at higher power levels and what effect it has on the observed wear rates.

In the case of McGuire, we have been -- McGuire has been performing steam generator inspections very frequently, on the order of every couple of months, something of that -- two or three months, something of that frequency. And based upon what is observed regarding the amount of degradation or the incremental degradation that's taken place since the last inspection, that experience is factored into our evaluation of the next short period of operation, in that we would not -- we do not predict that during each succeeding interval of operation, that the wear will be excessive or exceed allowable limits during that period. And even if it did -- which we don't expect -- but even if it did, we would -- the likely consequence is a small leak, but we don't expect that to be the situation.

Q Let me repeat the question, just so we can get that on the record:

If this scenario did occur, what would be the effect if the plant were not dropped to idling power following that scenario?

A I guess my response is that we'd eventually be shutting down for steam generator inspections, we'd observe the degradation had proceeded beyond what we had anticipated, and we'll take appropriate corrective action.

At worst, I would expect that we'd get a leak and that would precipitate the corrective action.

Q Mr. Rajan?

A (Witness Rajan) My response would be that if the natural frequency of certain tubes were to change by this order of magnitude that has been postulated here, there would be excessive -- there would be excessive vibrations for those affected tubes, and the damage, if it were to occur in those tubes would be at the supports which are affected, and as Emmett pointed out, there would be -- these would be detected by eddy current measurements.

Q Earlier you gentlemen stated that you thought there was not a great probability of multiple tube leaks, but would you say that leaks --

A (Witness Murphy) Multiple tube failures, gross failures.

Q I'm sorry, I misunderstood that.

A We do have occasion to experience multiple tube leaks, simultaneous leaks.

Q Is there any increased safety risk that occurs from that?

A From multiple tube leaks?

Q Yes.

A No. The leak rate limit, the tech spec leak rate limit for Byron, has been set such that if you have a throughwall crack or leaking crack which is leaking at less than the technical specification leak rate limit, that the length of the crack is smaller than the length of the crack that it would take to result in a tube rupture under postulated main steam line break conditions.

The fact that the leaks are occurring at less than the leak rate limit will provide assurance that these leaks would not result in any rupture or gross leakage under accident conditions.

Q For the record, would you -- excuse me just a moment.

(Pause.)

BY MR. JENKINS:

Q Well, then, would you say that ruptures of vibrations -- ruptures resulting from vibrations in the D-4, D-5 model, the probability of those ruptures occurring is greater than at plants without preheaters?

I'm sorry, let me withdraw that question. We're having problems with it.

ou5

(Pause.)

BY MR. JENKINS:

Q Two very quick questions here for the record:

Would you compare Westinghouse's record with that of other vendors in problems of tube degradation?

A (Witness Murphy) They've all had tube degradation. Sometimes the problems tend to be unique to a particular vendor, but they have all had tube degradation.

A (Witness Rajan) The nature of the problems may be different, but degradation is not confined to just Westinghouse steam generators.

Q Okay. Relative to one another, are any of the vendors better or worse in terms of the seriousness of the tube degradation problems?

A (Witness Murphy) Well, you know, one can make the qualitative observation that four plants -- steam generators in four units have been replaced as a result of tube degradation.

Q These are Westinghouse plants?

A Westinghouse plants. And additional replacement activities are scheduled for certain Westinghouse facilities.

A (Witness Rajan) One, and possibly two.

A (Witness Murphy) Pardon?

A (Witness Rajan) One, and possibly two.

A (Witness Murphy) Yes.

Q My very last question:

In your expert opinion, would you say that it is possible that a steam generator tube will rupture at an operating plant in the future?

A Possible.

(Discussion off the record.)

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E X A M I N A T I O N

BY MR. GALLO:

Q Gentlemen, do you have a copy of the contention in front of you? It's the steam generator tube integrity, 9-C. I take it you have it? Is that right?

A (Witness Rajan) Yes.

Q Are you able to identify that the paper that counsel gave you contains Contention 9-C?

A (Witness Murphy) Yes, the paper that we have just been given contains that particular contention.

Q Is that the contention, Mr. Murphy, that your affidavit addresses?

A Yes.

Q How about you, Dr. Rajan?

A (Witness Rajan) Yes.

Q The second sentence of the contention states that -- refers to the previous sentence and indicates that certain problems indicated in the first sentence, and now I am quoting:

> ". . .constitutes a hazard, both during normal operation and under accident conditions."

Do you see that sentence in the contention?

A (Witness Murphy) Yes, sir.

A (Witness Rajan) Yes.

Q Mr. Murphy, what does the term "under accident conditions" mean to you as used in this contention?

A (Witness Murphy) It means whether -- it means, for example, a main steam line break transient or LOCA transient could precipitate or initiate a tube failure.

Q It's not referring or -- strike that.

Is it referring to a situation where a steam generator break -- I'm sorry, steam generator tube break might cause an accident in and of itself?

A I think that the statement is general enough, perhaps, to encompass that also.

Q But it's also including, if I understand your testimony, a situation where during the course of an accident, the accident phenomena, if I can use that phrase, caused a steam generator tube rupture. Is that correct?

A Yes. The tube rupture, I guess, is an accident in and of itself. Secondly, there are other accidents which one must be sure that you don't run into a situation where another type of accident could precipitate a failure.

Q All right, fine.

Dr. Rajan, do you agree with that statement -- that interpretation of this part of the contention?

A (Witness Rajan) Yes, I agree with this, and I would add to the accident scenario an earthquake event, for example, a seismic earthquake, which can precipitate a steam line break.

Q Okay. Now it's not clear to me which of you gentlemen is expert in the area of what I will call what we've just been talking about, I'll call it accident analysis involving steam generator tubes.

Mr. Murphy, are you an expert in that area?

A (Witness Murphy) In terms of accident analysis, it depends on what exactly what you mean. I have a solid background regarding what it takes to fail a tube. I know pretty well what it takes to fail a tube. Regarding the systems aspects of the shutdown transient, I am not expert on that matter.

Q Well, how about the -- do I take it from the last statement you made that as far as the effects of an accident involving -- let's use hypothetically a design basis loss-of-coolant accident, the effects of that accident on a steam generator tube rupture? Is that an area of your expertise?

A It's an area I have knowledge regarding the potential for a rupture event to aggravate certain accident situations. For example, LOCA. I'm aware of studies that have been done to assess these effects.

Q And would that include the effects on systems within the reactor, including the ability to keep the core cool?

A I'm aware of some of the analyses that have been done and the conclusions which have been reached. I am not an expert on how the analyses were conducted, what the assumptions were.

Q All right, I want to ask that question again. Do you consider yourself an expert in this area as we have defined it here?

A I'm not quite sure exactly what we're -- I know what it takes to break a tube. I have a general knowledge of what may constitute excessive leakage during accident situations. This is information I must have in order to make a finding as to whether or not we have adequate assurance that we are doing enough to ensure steam generator tube integrity.

Q I guess that's the answer you gave me previously. Really, what I wanted to know is whether or not you're an expert in terms of the consequences of steam generator tube failure in a design basis LOCA and its effect on reactor systems operation?

A No, but I am knowledgeable regarding some of the conclusions that have been derived from such studies.

Q Essentially you have acquainted yourself with the work of others; is that correct?

A That's correct.

Q What about you, Dr. Rajan? Are you an expert in this area?

A (Witness Rajan) No, I am not. I am aware of the forces and the stresses that would cause -- that would act on steam generator tubes during a LOCA event, and a seismic event.

I am not, however, an expert on how other systems would be affected during a LOCA event, and also how the LOCA would -- a LOCA event would affect the coolability of the core, for example.

Q Who is within the Staff? Either one of you can answer that.

MR. GOLDBERG: Mr. Gallo, let me suggest this: If you have some questions which you want answered, these witnesses haven't shown any reluctance to indicate where they are an expert or ill-equipped to provide the answers. Why don't you ask the question?

MR. GALLO: I am asking the questions. He is doing a fine job of telling me the answers to the questions I'm asking. They have now established they are not experts in the area I am inquiring in, and I am asking who is in the Staff.

MR. GOLDBERG: But you haven't made any particular inquiry. You have outlined some --

MR. GALLO: Do you have an objection? If so, state it. I want to get on with my cross-examination here. All right?

WITNESS MURPHY: Okay. Studies of the effects of tube ruptures on accidents have been done under the heading of the Task Action Plan. The task manager for that program is Jack Strosnider.

BY MR. GALLO:

Q Can you spell that for me?

A (Witness Murphy) S-t-r-o-s-n-i-d-e-r.

Q He's the task action plan manager?

A Yes. Task manager, I think, was the formal description. He managed and coordinated the studies done by a variety of organizations and outside consultants having to do with the effects of ruptures on LOCAs and main steam line breaks, et cetera, and so he could refer you to the specific people who did the analysis.

Q I understand. All right. But now to your knowledge, do you know of any individual within the Staff who is an expert on this accident analysis area that we have been discussing here? I mean to your knowledge? I recognize that if Mr. Strosnider were here, perhaps he could tell us, as well, but he's not.

A Yes, I think I know of guys that are fairly knowledgeable in this area.

Q Can you name them for me?

A Chris Parcheski.

Q Can you spell that one?

A No, sir. It starts with a P.

(Laughter.)

Q What's his first name?

A Chris.

Q Can you pronounce his last name again?

A Parcheski.

Q All right, that's close enough.

A I believe he was very involved in the LOCA study, the one that was done to evaluate tube ruptures concurrent with LOCA.

Q Is that the one that was performed out at Idaho?

A Yes.

Q Anybody else that comes to mind?

A A person by the name of Akstulewicz.

(Laughter.)

I think his first name is Frank, but that's not for sure. He was involved in the -- was involved in the evaluation of tube ruptures concurrent with main steam line break.

Q Could you try that name on me again, please?

A Akstulewicz.

MR. CHESNUT: I can give you the spelling of that name if you need to know the name. A-k-s-t-u-l-e-w-i-c-z.

BY MR. GALLO:

Q Is that gentleman a member of the NRC Staff?

A (Witness Murphy) Yes.

Q Is that it?

A Those are people that I know for sure that were involved. People I think -- a person I think had considerable involvement is Pasedag, Walt Pasedag.

Q That I can spell.

How about you, Dr. Rajan?

A (Witness Rajan) In my judgment, in answer to your question, I don't have any other names besides these, but my feeling would be that there would be more than one person who would be involved in the kind of study that you are looking for, and these people probably would be from the Reactor Systems Branch and the Accident Analysis Branch.

Q Let me tell you where I'm going with my questions. If I look at your affidavit, and I'm limiting myself to just what's in your own affidavit, let me ask a preliminary question:

Am I correct in concluding that beginning with page -- I'm sorry, beginning with paragraph 5, through the end, which I believe is paragraph 12, that that represents a joint statement by both Mr. Murphy and

Dr. Rajan?

A (Witness Murphy) Don't we address that up front? 8 through 12.

Q Say again?

A 8 through 12.

Q Paragraphs 8 through 12 are your testimony; is that correct?

A It was sort of a joint prepared testimony.

Q Oh, 8 through 12. And what about paragraphs 5 through 7?

A That more or less represents mine. That does represent my testimony.

Q I see. All right.

Now in reviewing paragraphs -- well, strike that.

Paragraphs 8 through 12 appear to be talking about a particular problem which has been referred to here as the flow-induced vibration problems; is that correct, Dr. Rajan?

A (Witness Rajan) That's correct.

Q Paragraphs 5 through 7 are addressing, I guess, the generic question of steam generator tube integrity; is that correct, Mr. Murphy?

A (Witness Murphy) Yes.

Q And this is primarily your testimony; correct?

A Yes.

Q Where in these three paragraphs or any place else in the affidavit, for that matter, do you discuss the accident aspects of Contention 9-C?

A Item 5 is intended to address our assessment of the requirements that have been imposed to prevent tube failures.

Q So paragraph 5 deals -- is intended to deal with routine operation and tube failure under accident conditions, as well; is that correct?

A The approach, the regulatory approach to date to preventing tube failures through normal operation or accidents, is to surveil the tubes, inspect them regularly, periodically, to remove those from service that are excessively degraded within our acceptance criteria, and to reinforce these requirements with stringent leak rate limits during normal operation.

In addition, we -- the Staff has been requiring plants during the licensing process to implement improved controls in secondary water chemistry. This is, in a nutshell, the regulatory approach to preventing failures during normal operation,and ruptures. When --

Q I'm sorry, go ahead. Go ahead.

A These are requirements that the plant starts up with. It's not at all unusual that as problems do occur in service, for the Staff to impose additional requirements, with the express purpose of preventing or minimizing the potential for tube ruptures during normal operation and accidents.

Q When you used the term "tube rupture," do you mean an instantaneous failure?

A In the context I've been using, yes.

Q And is that also true when you use the term "tube failure"?

A In that context, yes. But one of the things we looked for, and the NRC does monitor the operating experience, you know, at operating facilities -- and one of the things we look for is leak experience. Also the number of tubes found to be degraded during routine periodic inspections.

In other words, we don't wait for a tube rupture to occur before we get involved and impose additional requirements. So degradation -- if degradation occurs or if leaks occur, of course, this then enhances any concern one might have about potential for excessive degradation.

Q Now I guess the confusion I have by reading this paragraph -- strike that.

By reading paragraph 5, how am I supposed to know that it addresses both accident conditions and routine operation? What is in paragraph 5 that gives me that clue?

A Well, my answer to that would be the second sentence of paragraph 5. I refer here to steam generator tube integrity problems. It's not explicitly stated here, but it's assumed to be understood that we are concerned with tube integrity during normal operation and during accidents.

Q I understand that. Part of my confusion here is if I look at these four steps, if I can use that -- or I guess that's the wrong phrase -- four factors or approaches that might be taken to deal with steam generator tube integrity problems, I'm confused as to whether they are mitigative measures or preventive measures. Can you clarify that for me?

A Items 3 and 4 are intended to -- let me start from the beginning.

Items 1 and 2 are intended to reduce the potential for corrosion or degradation of the tubing.

Q Reduce, but not prevent?

A Hopefully it is an ultimate objective to prevent. In a practical sense, right now, certainly one seeks to minimize any potential for corrosion.

Items 3 and 4 --

Q Wait a minute. Wait a minute. Let's get back to 1 and 2. Are they preventive measures or simply mitigative measures? I thought you were going to tell me they were mitigative measures, meaning they don't prevent corrosion, but necessarily try to control it. But you switched on me. Maybe I misunderstood you. Is my question clear? I'll repeat it, if it's not.

A The question is whether these measures are intended to eliminate corrosion?

Q Items 1 and 2, yes.

A As opposed to whether they are intended to minimize the corrosion?

Q Yes. What do you expect?

A I expect some amount of corrosion during the life of the plant.

Q What do you expect in terms of results from employing all volatile secondary water treatment and improved controls and monitoring of secondary water chemistry?

A All volatile treatment, secondary water treatment, should minimize, if not eliminate any concerns, regarding phosphate wastage, corrosion of the steam generator tubes.

Q Has that been the experience so far, that the NRC has seen at operating plants?

A AVT chemistry has been very successful in arresting existing wastage problems and preventing new wastage problems from developing at plants which have not operated --

Q Mr. Murphy, you and I are going to be here a long time if we don't get a reconciliation. I'm trying to get an answer to the question. You switched from "eliminate" to "arresting" to "mitigate" to "reducing," and I'm trying to segregate those terms.

Now I thought you were telling me that all volatile secondary water treatment was successful -- well, I won't try to characterize what your testimony is. So tell me again whether or not all volatile secondary water treatment is considered, in your opinion, to be a mitigative measure or a preventive measure in terms of steam generator tube integrity problems, as you use it in your testimony.

A You're limiting the question to all volatile treatment?

Q Yes. I'm taking it a piece at a time now.

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A With regard to phosphate wastage, I'm not a corrosion specialist. I expect it is a preventive measure which addresses that particular phosphate wastage problem.

Obviously the treatment is intended to address corrosion problems in general, and I would expect that in that sense it is a mitigative treatment.

Q Fair enough.

How about the next item, improved controls and monitoring of secondary water chemistry? Is that, in your opinion, a mitigative or a preventive measure in terms of steam generator tube integrity problems, or perhaps both, as you previously testified?

A Mitigative.

Q I'm sorry?

A In my opinion, this is a mitigative approach.

Q All right. Fine.

Now looking at items 3 and 4, I thought in one of your previous answers you were putting items 3 and 4 in a different category from items 1 and 2.

A They are.

Q Is that true?

Could you explain?

A These items will neither mitigate nor prevent -- well, let me withdraw that comment.

Well, these items will not directly mitigate nor prevent corrosion problems. They will provide a warning that you have problems and will warn the utility that it must, if it wants to save the steam generators, it better take some corrective actions to slow down the process or prevent it.

But beyond that, these last two items, items 3 and 4, are intended to detect a situation where the tubes have become excessively degraded, and for those tubes which are excessively degraded, they must be repaired, either plugged or sleeved, or whatnot.

Item 4 is -- we have a one-two punch approach here to ensuring tube integrity.

Item 3 deals with regular periodic in-service inspections.

Item 4 is an additional very important method or -- not method, but provides considerable added assurance that on top of the periodic inspections, that the tube integrity is not becoming excessively degraded.

Q Is item 4 pertinent to the question of tube rupture?

A Yes.

Q Can you explain how it's pertinent?

A Two ways:

One is the occurrence of leaks may be an indicator that corrosion is proceeding at a higher rate than anticipated; that perhaps a sufficient allowance for additional incremental corrosion pre-inspection

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hasn't been provided for in the plugging limits. Leakage may be indicative of additional tubes which represent -- which are incipient leakers.

Q What do you mean by the term "incipient leakers"?

A About to leak, 90 percent throughwall, or 95 percent. But in a practical sense, then, leakage in a sense is an early warning signal.

Secondly, any leakage rate, regardless of its number, provides -- regardless of what you set it at, provides some additional measure of assurance against tube failures. But the limit, which I understand will be specified for Byron, has been set to assure that if a given tube is leaking at the leakage rate limit under normal operating conditions, that if you were to suddenly go into main steam line break, that the crack length involved would not be sufficient length to result in a tube rupture, or a gross tube failure, or a significant leakage during the accident condition.

Q How does a limit leakage rate warn us of incipient leakers?

A In general, I would say that where you have a tube with a defect that's gone all the way 100 percent throughwall --

Q Wait a minute. I thought an incipient leaker was one that wasn't all the way through.

A Well, the answer to the question was why does the occurrence of a leaker tell me we have incipient leakers.

Q I'm sorry I interrupted. Go ahead.

A Based upon experience, whenever we have leakers, in all probability we have tubes where similar degradation has proceeded, at least part throughwall, often considerably part throughwall. You generally have some secondary side corrosion. It typically affects many tubes in the specific region of the bundle, not just one bundle.

So there will be many tubes behaving similarly. A leaker will represent the tube that has been degraded the most.

Q What you are telling me is that if there are leakage symptoms, that this is an indicator, perhaps, that there may be other tubes in the steam generator that might be subject to bursting and testing ought to be done, eddy current testing or other surveillance ought to be done to check it out? Is that what you are telling me?

A No. But as I indicated in the second aspect of the response, the leakage rate limit is intended to assure that individual tubes won't rupture. So if you've got detectable leakage that's less than the tech spec limit, I'm not suggesting that we're in a situation that you may fail tubes even if you go into an accident.

It does suggest to anybody monitoring the performance that the steam generators are degrading, that a number of tubes may be involved, even beyond the leakers, and that certainly you want to keep on top of the situation.

In the case of the Regulatory Staff, we want to perhaps address ourselves to whether or not the plugging criteria remain adequate for the corrosion process that is taking place; whether or not the frequency of inspections that are specified in the tech specs remain adequate for the situation we are actually experiencing at the plant in question.

Q Well, will a steam generator tube leak before it bursts? Aren't those two inconsistent phenomena?

A The term -- the word "burst" in the context of steam generators generally refers to --

Q As you and I have defined it already, what the word "burst" means.

A Gross tube failure, you're talking about the gross tube failure.

Leaks, when I speak of leaks, I'm generally speaking of local failure. Failure might even be the wrong word. Local penetration of the tube wall, where you get relatively small amounts of leakage, on the order of 1, 2 gpm or less.

Q All right. But will a steam generator tube that is the subject of a gross failure, will that leak before it incurs the gross failure?

A Operating experience indicates that generally in the vast majority of the cases, that will be the case, but not absolutely always.

Q All right. Are you familiar with the steam generator tube failure incident at Point Beach back in 1975?

A Yes, but if you're going to ask me whether or not -- yes -- well, I have some familiarity with it. It's been a while since I reviewed the circumstances.

Q Do you know whether or not that involved a gross failure of the steam generator tube?

A It was on the order of 100 or 125 gpm, or something like that.

Q Do you know whether or not that leaked before that happened -- the failure occurred, rather?

A I haven't reviewed it recently enough to say.

I believe it was leaking at some rate, but I can't --

Q I'm talking about the particular tube now, not elsewhere in the system.

A One -- well, I don't know for a fact without checking the circumstances what the prior leakage history was for that unit prior to the rupture.

Q Are you familiar with the steam generator tube rupture that happened at Surrey in 1976?

A It happened, yes, it was about 80 gpm.

Q Was that a gross steam generator tube failure incident?

A It's generally classified as one of the gross rupture events, yes.

Q Do you know whether or not that tube leaked before it failed in a gross manner?

A As I recall it -- and I'd have to check the facts again -- there was some initial leakage prior to the failure. I don't recall how much. I don't recall the specifics. I'd have to research that.

Q Okay.

A There are two other rupture events, of course, which did not involve prior leakage. Prairie Island and

Ginna. And the reasons these did not involve prior leakage is because we were dealing with a wall thinning phenomenon and not cracking. If you have a general wall thinning, you can get -- you can lose enough wall thickness over enough of an area of the tube, such that there will be no tell-tale leakage prior to the event.

Q Was that wall thinning due to corrosion or some other problem at Ginna and at Prairie Island?

A Both occurrences involved mechanical wear or abrasion of the outer surface.

Q There was some foreign material or something inside the steam generator that was wearing on these tubes; is that correct?

A Yes.

Q But that wasn't the case at either Surrey or Point Beach, was it?

A No.

Q Now as I understand this regime as you have described it in paragraph 5, that you have this leak rate limit and you have something called a plugging criteria, and you have an inspection interval, and the idea is to coordinate all three so that you plug all tubes before you reach a point where the tube walls are so thin, they might burst or leak or whatever; is that correct?

A That's correct.

Q And I'm interested to -- well, I guess let me ask Dr. Rajan:

Can you tell me briefly just what the plugging criteria are?

A (Witness Rajan) The plugging criteria are -- they are outlined in Regulatory Guide 1.121, and basically the criteria -- there are three criteria:

One is, number one, that the tube will not reach the yield point, the tube material will not reach the yield point during normal operating pressure differentials.

The second criteria that has to be met is that the margin to failure or margin to burst will have a factor of safety of three against normal operating pressures.

In other words, if the burst pressure is 3 delta P, then the normal operating pressure should be no more than delta P.

Or, putting it the other way around, if the normal operating differential is delta P, then the burst pressure should not be more than 3 delta P -- no less than

3 delta P.

And the third criteria is that there should be an adequate margin to burst under accident conditions and pressure differentials and loads.

A (Witness Murphy) Also we built into the plugging limit allowances to account for eddy current error, and incremental corrosion between inspections.

Q Is there a generic yield for the first criterion which you characterized as the yield point of the material itself, the steam generator tube material? Is there some throughwall thickness that establishes that yield point?

A (Witness Rajan) Yes. Based on a very large number of tests with the different types of defects, I believe for -- of course, this would depend on the dimensions of the tube. Different steam generators have different diameters and wall thicknesses. So this would differ from tube to tube.

But in general it can be said that approximately 25 percent of the tube wall -- if there is a 25 percent of the tube wall remaining, the yield point would not be reached under normal operating conditions.

Q All right, now, what about under accident conditions?

A Under accident conditions, the minimum wall thickness has been -- well, depending on the location of the defect in the steam generator.

For example, the tubes that are located near the U-bend regions, they would be subjected to higher loads than the tubes that are near the tube sheet and the supports.

So if one were looking for defects near the supports, near the tube sheet -- and here again the figures differ from steam generator to steam generator -- approximately 25 percent of the tube wall would be adequate.

Q So basically the criteria are the same whether it's normal operation or -- at least for these particular tubes you're describing, the criteria would be the same whether it's normal operation or under accident conditions; is that correct?

A (Witness Murphy) I can speak to a series of D-1 steam generators.

Q Wait a minute. I want to get an answer from Dr. Rajan, and then you can add to it.

A (Witness Rajan) The numbers work out to be about the same, but they are based on different analyses, totally different analyses.

Q I see. It's just a coincidence?

A It's just a coincidence. And here again, as I pointed out, we have to look at a specific model and a specific tube wall thickness to determine the minimum wall requirement.

Q Mr. Murphy, do you want to add to that?

A (Witness Murphy) No.

Q When is this analysis normally done, Dr. Rajan?

A (Witness Rajan) The Licensee makes a commitment to abide by the requirements of the reg guide prior to operation of the plant.

However, during operation, when the specific tubes are being plugged, specific analyses may be done for those tubes.

Q I see.

A (Witness Murphy) I'd like to expand on that, perhaps.

Q Sure.

A Standard technical specifications contain in parentheses, plugging limits, which I think are generally about 40 percent -- which are 40 percent for Westinghouse steam generators. So this plugging limit is shown on the standard tech specs with an asterisk. The asterisk provides guidance for the Applicant in terms of how he might go about justifying a different limit.

The limit is -- the structural characteristics of a tube is a function of its geometry, and there are only a few different tube geometries out there. We have lots of plants, but we have just a few categories of different tube geometries. All Model Ds are the same, all Model 51s are the same, and once a supporting structural analysis has been performed for a plant with a given type of tube, that analysis is generally valid for other separate plants.

Q I see.

A So, as far as I know, individual plants don't keep resubmitting the same analysis over and over again.

Q What is the inspection interval under the tech specs, if you know? I mean, let me explain where I'm coming from.

Dr. Rajan has explained the tube plugging criteria. So if I understand what he told me correctly, we have to inspect the steam generator tubes from time to time to make sure if there's any degradation, we catch them before they reach the yield point; is that correct? That's the objective; is that right?

A That's the objective.

Q That tells me as layman that we have to inspect at some reasonable interval related to whatever the rate of corrosion might be?

A That's correct, yes.

Q Can you tell me what the inspection interval is?

A Typically plants are required to perform under their tech specs steam generator inspections every 12 to 24 months. There are provisions, depending upon the steam generator performance, how well they performed, how free of problems they've been, for extending the interval for inspections for longer periods.

Q When those inspections are conducted, do they sample by eddy current testing a segment of the steam generator tubes? Do they do 100 percent testing?

A Yes. The initial inspection sample is a percentage of the tubes, depending upon the -- the results of this initial sample inspection can fall into one of three categories, ranging from essentially good to bad.

If you are in the good category, no further sampling is required. If you are in the bad category, additional sampling is required. You may go through several sampling stages.

Eventually you may get thrown into 100 percent inspection of steam generators.

Q How does the Staff know that a 12 to 24-month interval is sufficient, inspection interval?

Let me strike that and ask the question better.

How does the Staff know that a 12 to 24-month interval is sufficient to identify any steam generator tubes that may be approaching the yield point, so that they might require plugging?

Does that clarify it for you?

A Yes. First, let me state that the Staff is generally aware of the condition in terms of the general condition of a plant, whether it's got an extensive corrosion problem, whether it's occurring at a low or high rate.

Q It's an unfair question. Let's limit it to a plant that is just beginning to operate, like Byron. Let's take that kind of clean plant. Can you answer the question in that context?

A Yes.

Q How do you know that 12 to 24 months is a proper interval?

A Based on operating experience, with one exception, we don't run into significant wall penetrations by corrosion generally within the first -- well, one or two cycles of operation. If there is a corrosion process taking place, you will see the early stages of it during your eddy current inspection.

Q Well -- go ahead, I'm sorry.

A There is at least one corrosion phenomenon, the primary side corrosion, stress corrosion cracking phenomenon, that can occur quickly. This particular corrosion problem could conceivably occur during the first cycle of operation. It has been observed that way for Model 51 steam generators.

Q What's the phenomenon?

A The so-called U-bend cracking phenomenon. Non-denting-related U-bend cracks. It's often called the the tangent point cracking problem.

Q This is a problem you say that is only applicable to Model 51 steam generators?

A It's only occurred there, to date.

Q You said there was one exception to this. Is this the exception you are talking about?

A Yes. Yes.

Q Did this particular phenomenon result in a gross tube failure at some plant?

A That particular phenomenon has not ever resulted in a gross tube failure.

Q All right.

A We have had dozens -- tens, or perhaps dozens, of leaks as a result of this phenomenon and they have all been very small leaks.

Q Does the Staff change the inspection interval depending on plant experience, in terms of corrosion problems, or steam generator tube integrity problems that might be identified during an eddy current inspection?

A Yes. Turkey Point 3 and 4 and Surrey Units 1 and 2 ran into extensive and very severe denting. The Staff imposed requirements for performing -- see, first performing steam generator inspection every three months. This was relaxed to every six months. For a number of years, these four units were required nominally to operate for six months between inspections, although the Staff did consider on a case basis extensions of two months or four months. As I said, on a case basis.

Q All right.

A There are other examples, as well, where we have imposed additional inspections.

Q Is it fair to say you start out 12 to 24 months, and as experience dictates, you either lengthen it or keep the same interval, or make it shorter? Is that it?

A The tech specs already make provision for lengthening the inspection interval, if you have real good experience, and we generally don't -- we've never been requested to relax those criteria. But we have intervened to require more frequent inspections than required by the tech specs.

Q I guess the only other question I have in this particular area, is which do you select? Is it 12 or 24, or a range, or what would go in the Byron tech specs? Do you know?

A What would go in the Byron tech specs are what we have written into the standard tech specs, and as memory serves me, I believe they are required to do a periodic inspection every 12 to 24 months. Something -- we are talking about -- and the precise length of the interval is, you know, the Licensee will select that, depending upon his schedule.

Q All right. Now returning to paragraph 5, the reason I have asked these questions is that I would have expected to see in paragraph 5 an item 5 in parentheses dealing with some sort of analysis of steam generator tube failure during the course of an accident similar to the one that we briefly referred to, that was performed by Idaho with respect to steam generator tube rupture effects on a LOCA.

Can you explain to me why you don't deal with the consequences of that situation in your testimony?

A With the requirements that we have --

Q These four items.

A Yes. I believe that pending the outcome of our ongoing generic programs, that with these programs, that we have reasonable assurance against --let me remove the term "reasonable assurance," because it applies to something else, but I believe that there is a --

Q Well, why don't you use the language in your affidavit in paragraph 7?

A It's used in a different context.

Q All right. Go ahead. Sorry to interrupt.

A I believe that we have -- there is a very, very low likelihood of gross tube failures during accidents.

Q That's because of the tube plugging criteria, the in-service inspection and the in-service inspection interval, and the other controls for monitoring, and water chemistry treatments; is that right?

A That's correct. That's correct, but let me correct one thing. I don't wish to get unduly carried away. I said very, very unlikely. I mean to say I consider it very unlikely that we would have a rupture during -- in the event that we did run into a major design basis accident. I do believe that there are a lot of -- with the extensive degradation that we have observed throughout the industry, the tube ruptures that we have had during normal operating conditions, not during a transient, that there is sufficient cause for the Staff to take a close look at the regulations to see that they adequately address the problems that do indeed provide sufficient assurance against a rupture event occurring, both during normal operation and during accident conditions.

Q Is the Staff doing that?

A Yes.

Q Why isn't it in your testimony some place, then?

A Well, it is. I looked upon my testimony as sort of an expansion of testimony to the SER. The Task Action Plan, the generic safety issues, something that is being reviewed by the Staff.

Q So in paragraph 2 of your affidavit, you essentially -- I guess what you have done is adopted and incorporated by reference the information and material that's in the Safety Evaluation Reports for Byron; is that correct?

A Yes. And I think it so states. Yes.

Q To your knowledge, has the Staff determined that it is necessary to -- strike that. Let me see how I want to phrase this.

We have been using the term "design basis accident." There are a number of accidents that the Staff requires evaluation for before a nuclear power plant can be licensed to operate; isn't that correct?

A I'm sorry, do you mind repeating the question?

Q Sure. We've been using the term "design basis accidents" during our discussion here, and I just wanted to establish that we are on the same wave length, that there are a number of so-called design basis accidents which Applicants must analyze and the Staff must review before a plant can be licensed to operate. Is that correct?

A That's correct.

Q I take it that the Staff has not determined that an accident situation involving an established design basis accident, coupled with a concurrent tube failure, should rise to the dignity of being called a design basis accident, in and of itself; is that correct?

A It has not, to my knowledge, issued a formal conclusion to that effect. It is certainly something that is under discussion and consideration at this moment.

Q All right. I guess it's the lack of that discussion and consideration that has triggered me to ask this line of questions. Perhaps it is in the Safety Evaluation Report and I just didn't notice it. Could you point me to where that discussion might be in the Safety

Evaluation Report?

A It's a lead sentence, in Appendix C.

Q Can I look over your shoulder? I'm sorry, it's Appendix what?

A Appendix C, or whatever they call it.

Q Yes, I've got a copy of that. Go ahead.

A Addressing the primary concern of tube integrity. The primary concern --

Q Where are you reading, what page?

A Page 9, page C-9.

Q Okay, give me a chance to catch up with you. Okay.

A The section is entitled "Westinghouse Steam Generator Tube Integrity," and we begin the discussion by stating the primary concern is the capability of steam generator tubes to maintain their integrity during normal operation and postulated accident conditions.

Q All right. Is that the extent of the discussion on the occurrence of steam generator tube failure under postulated accident conditions? I don't see any discussion on that subject on either page C-9 or page C-10. Maybe I just haven't seen it there. It might be there and I just missed it.

A No, it's the -- that sentence describes the overall objective of our regulatory approach, you know, the secondary water chemistry, the surveillance requirements, the plugging limits and the leak rate limits.

Q I know, but we got to this page through a series of questions of which I ultimately asked you, and I thought you said the SER might contain a discussion of the Staff's judgment in the consideration of the need to deal with steam generator tube failure under accident conditions, and we are now looking at page C-9 and C-10, and I'm asking you where that is.

A Let me first refer again to the contention. As I say, the SER and my testimony describes what we consider to be the rationale for --

Q Well, I'm prompted to say that the answer to my question is no, there isn't any discussion in either your testimony or the SER. But look at page C-9, and one, two -- the third paragraph, last sentence of the third paragraph. I'll read it:

"The tubes and tube sheet are analyzed in WCAP 78-32 and confirmed to withstand the maximum accident loading condition."

A Uh-huh.

Q Does that sentence get to the point I am trying to elicit here?

A I don't believe so.

Q It does not?

A You know, it goes without saying that the Westinghouse and the Applicant are designed -- are required to design the plant, including the steam generators, to meet all design loadings, including those which occur during a design basis accident or a faulty condition. There are requirements they must satisfy. They must demonstrate, in accordance with established rules, that they can sustain all normal operating or accident or transient conditions.

Q Do those analyses assume a steam generator tube failure during the course of the design basis accident?

A No.

Q All right.

A But that wouldn't -- you are getting into a different area, you are getting into in terms of systems, does the systems response consider that situation, and the answer is no. That is not a limited loading situation for a given steam generator tube.

Q Mr. Jenkins was --

MR. GOLDBERG: Dr. Rajan had something he wanted to add.

MR. GALLO: I'm sorry, go right ahead.

WITNESS RAJAN: In an appendix to this WCAP 78-32, they did consider the steam line break event also.

BY MR. GALLO:

Q Say that again.

A (Witness Rajan) In an appendix to the WCAP 78-32, that we just referred to, the analysis to do a steam line break accident was also considered.

Q I see.

When you say that, do you mean a steam generator tube failure was -- the consequences of that failure on the accident was considered in WCAP?

A Essentially the WCAP considered the effects of LOCA loads, loss-of-coolant accident loads on the steam generator tubes, and it determined to what extent degraded tubes can withstand the dynamic LOCA loads, and in an appendix they also considered these facts of steam line break accident loads on steam generator tubes.

Q Well, then, Mr. Murphy, maybe I misled you or wasn't clear with my question. Perhaps this sentence does address the very point that I was driving at. The sentence on page C-9 that refers to 78-32.

A (Witness Murphy) You're --

Q I'm sorry?

A The way I've been interpreting your questions, there were two situations one might want to consider:

One, whether or not we design the plants to prevent tube failures, and the answer to that question is yes. And there are established rules for that.

This sentence alludes to the fact that these particular components discussed here have been designed and analyzed to withstand accident conditions.

But now we're going beyond that, and we're saying let's assume that corrosion takes place and becomes extensive, and let's assume that routine surveillance, leak rate limits and so forth, in a particular instance didn't work. I mean it didn't successfully prevent the rupture. Okay? And what then? That's a different -- that's a little different aspect.

Q That's the part I'm trying to focus on.

A We designed the plants to avoid that situation. Now, if it happens, okay, and then the question is -- the question then is so what? Is this a concern? This is a different consideration entirely.

Q In your judgment, in your opinion, is it unnecessary to consider those situations in circumstances that you just described, because of the four factors on paragraph 5 of your testimony will essentially provide reasonable assurance that you're not going to have that kind of problem?

A Not exactly. I think, as I said before, we've got to look into this situation further, and we are as part of the unresolved safety issue alluded to here, and it's part of another study which is also going on at the same time. It's a related study. I think we have to take a close look at and understand the effects of ruptures and the consequences of LOCA, or main steam line break or anything else we might want to postulate. We have to -- in view of these findings, we will have to reexamine, perhaps reevaluate our existing requirements in light of these findings. But I know that these studies are ongoing.

You know, I'm familiar with the steps that are being -- that this matter is being pursued, and pending resolution of these items, of these issues, I believe that the requirements that we have in place, will be putting into place for Byron, are sufficient to provide reasonable assurance of public health and safety.

MR. GOLDBERG: Can we go off the record for a moment?

MR. GALLO: I just want to ask one question, and then we can go off the record.

BY MR. GALLO:

Q I guess maybe to some extent, Mr. Murphy, I've been unfair to you. If either Mr. Parcheski or Strosnider or Pasedag were with you on the panel, perhaps they could provide the insight that I'm striving for. Is that a fair statement?

A (Witness Murphy) Well, the -- you haven't really asked me about the mechanics of what specifically is it about steam getting into the primary from the secondary that causes the fuel to heat up, and when do we have to start worrying about the fuel melting and all this kind of stuff. They would be able to address that and tell you how they evaluated that.

Q You're correct, I haven't done that. I'm basically trying to determine in my own mind whether or not your affidavit, as you have testified, is complete, and I'm having a hard time reaching that judgment, because there is no discussion of the point that we have been debating here for some time.

Let's go off the record.

(Discussion off the record.)

(Recess.)

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MR. GALLO: I've just got one question left on the accident discussion, and then I'd like to go on to a new subject.

BY MR. GALLO:

Q Mr. Murphy, Mr. Jenkins gave me a document prior to the start of this deposition entitled "Steam Generator Status Report, February 1982, U.S. Nuclear Regulatory Commission."

I understand it was obtained from the NRC under a Freedom of Information Act Request by the Intervenors in this case.

I am asking you if you recognize that document, and have you come across it in your work in dealing with steam generator tube problems?

A (Witness Murphy) Yes.

Q Can you tell me who developed it, or wrote the document, and the circumstances for its development?

A Yes. Jack Strosnider. And the raw material behind this report was prepared under the -- as part of the Task Action activity.

Q I see.

Do you work for Mr. Strosnider?

A No, Mr. Strosnider is in a different office of the NRC.

Q I see.

A I used to work with him a couple of years ago.

Q Did you have any involvement in the preparation of that document?

A Of this one? No, I did not.

Q How about you, Dr. Rajan?

A (Witness Rajan) As far as formally, no. There may have been some input from me on the implementation of Reg Guide 1.121, which I don't see here, so I would say the answer is no.

MR. GALLO: Can we go off the record?

(Discussion off the record.)

MR. GALLO: Let's go on the record.

BY MR. GALLO:

Q Mr. Murphy, while we were off the record, I asked you whether or not you recognized this document as the complete report, and you indicated that perhaps it might be a preliminary version. Would you clarify that for the record, please?

A (Witness Murphy) I simply cannot say whether or not it is a final document or not. I'd have to read it in detail to know whether or not this was the one. Or better yet, to check with the issuing organization to make sure this is the proper -- this is the final report. I would assume that it is, judging from the date, but --

MR. GALLO: All right. Subject to that check, I'd like to have this document marked as Applicant's Exhibit -- no, strike that, marked as Murphy/Rajan Deposition Exhibit No. [illegible] and I'll give it to the reporter to mark for that purpose. But as I understand, during an off-the-record discussion, that Mr. Goldberg, on behalf of the Staff, will be kind enough to obtain a copy of this document that does not have the markings and underscoring that is on this document, which I understand had been performed during the review by DAARE/SAFE people.

(The document referred to was marked Murphy/Rajan Depo. Exhibit No. 1, for identification.)

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MR. GALLO: That's all I have on this point, unless there is any other comment.

MR. GOLDBERG: My only comment is one of clarification. I assume you are making it a -- why don't you just mark it for identification? I'm not sure how to make it an exhibit, unless these gentlemen are, you know, responsible for adopting its contents or you just want it --

MR. GALLO: That's the infirmity that exists. If I or anybody should try to offer it into evidence, that objection is there. I just want it a part of the deposition record.

MR. GOLDBERG: Okay.

MR. GALLO: That's all. But I'm not addressing myself to the question of whether or not it's an admissible document in the present form.

MR. GOLDBERG: Right.

BY MR. GALLO:

Q All right, let's change the subject.

Dr. Rajan -- let's see, I'll have to find your affidavit -- I believe that you have described, in answer to Mr. Jenkins' questions, the flow-induced vibration problem that you address in paragraphs 8, 9, 10, 11 and 12. Is this the problem that occurred at McGuire?

A (Witness Rajan) In paragraph 8 I do talk about Model D-2 which is the steam generator model for the McGuire Plant. And in paragraph 9 I describe -- I make a statement that D-4 and 5 were used -- are used at Byron, and I go into some of the differences between the Byron steam generator and the Model D-2 and D-3.

Q And did this phenomenon, flow-induced vibration phenomenon, occur at a foreign plant, too?

A Yes, it did.

Q Was that KRSKO? Or some other plant?

A Well, KRSKO has the D-4 steam generator, very similar to the one being used -- proposed for Byron.

Q Well, did the phenomenon occur there at that plant?

A Yes.

Q And how is KRSKO spelled, for the reporter's benefit?

A K-R-S-K-O.

Q All capital letters?

A All capital.

Q Any other foreign plants besides KRSKO?

A There are two other plants involved in this phenomenon. One is Ringhals in Sweden, and the other one is Almarz in Spain. And both of these have D-2 and D-3 type steam generators which are somewhat different in their preheat design than the Byron.

Q Now, as I understand it from your testimony, Westinghouse has developed a generic program to deal with this problem; is that correct?

A Yes, sir.

Q Are you familiar with their program?

A Yes, I am.

Q I believe you testified that you looked at some data that was taken at the KRSKO plant with respect to the evidence of vibration?

A Yes, sir.

Q And on the basis of that data, have you determined that a 70 percent power level is about right where the phenomenon might not be seen?

A , I would say that is a preliminary conclusion I have reached.

Q All right. Now you say in paragraph 10 that Westinghouse is evaluating modifications to the auxiliary feedwater system and you described one of those modifications.

To your knowledge, is Westinghouse considering modifications in addition to the one you described in paragraph 10?

A That's correct, they are considering several approaches and these may be used in combination or individually.

Q Do you know whether or not Westinghouse is recommending any of these approaches for implementation?

A No, they have not. They have not finalized their recommendations as to which approach or combination of approaches they will adopt for Byron, or for domestic plants.

Q Do you know when they might do that?

A My understanding is that the schedule for this is they have made -- they have made a preliminary presentation on what these options are.

Q To whom?

A To the NRC Staff, and they are currently doing the analyses and testing and evaluation of the various options, and my understanding is that by November or December they will have finalized the test results and data for Staff review, and at that point we will proceed on the acceptance or nonacceptance of those options.

Q Will the Staff approve one or more of those options? Is that what will happen?

A I can only predict at this point.

Q I'm not asking you if in fact you will, but is an approval, up or down, down the road, is t at what you plan?

A Yes.

Q So Westinghouse will come in with their program, Staff will review it and approve those aspects that it finds acceptable; is that a fair statement?

A I think it should be clarified that we are aware -- we are aware in more than a general way of what is being considered. All we are waiting for is hard data, and results of analyses. We are aware in a fairly --we have a pretty good idea at this point as to how Westinghouse is approaching this problem and what the most likely fixes are going to be. So we have a fairly good idea at this point.

Q What hard data do you need?

A Well, the hard data would consist of -- it could consist of, for example, model test results and it could also consist of stress analysis results of some of the fixes that they are proposing. And it could also result -- well, thermal hydraulic analyses and the results.

So we are aware of the fixes in a general way, but we have not reviewed the documented information yet.

Q I guess I neglected to ask you. What fixes are you aware of besides the one you described in your paragraph 10?

A There are several fixes. One of them is the addition of flow-straightening veins that will be attached to the impingement plate, and the effect of this would be to reduce the turbulence and make the flow uniform in that region.

Another is the change of the flow restricter device from one which has three holes to one which has a larger number of holes.

And then they are also considering sleeving of the tubes and the supports to stiffen the tubes in that region, and I forget, but these are the major other options besides the change in the aux feed system.

Q Now is the purpose of these fixes to reduce the flow of the water so that the vibrations don't occur? Is that it?

A No, it's not -- the object is not necessarily to reduce it, but the object is to reduce the turbulence in the flow.

Q And one way to do that is as suggested in your paragraph 10 in your testimony; is that correct?

A In 10, I talk about actual reduction of flow to the main feed.

Q I see. Well, but you've got 30 percent coming from another source; is that correct?

A That's right.

Q The effect of the two is to provide as much water as the current design has now?

A Yes. Yes.

Q So by creating two sources, I assume we deal with the turbulence problem you are talking about?

A That's right.

Q Now I think you have testified that you expect the Westinghouse analysis on their potential fixes and recommendations in October or November?

A That is the general timeframe that has been discussed.

Q Are you going to personally be involved in the review of these analyses?

A Yes.

Q Do you have any estimate of how long the Staff's review might take?

A Well, we have consultants assisting us in the Staff review, and generally we can complete this in short order. I cannot give a timeframe.

Q Well, what do you mean by short order? Don't give me a specific date, but just ball park.

A Within weeks.

Q Within weeks?

A Within weeks, yes.

Q Now you answered Mr. Jenkins' question that way. Are we talking 52 weeks, 100 weeks, or can you do any better than that? I don't want to press you unduly, but the reason I asked the question is on the bottom of page 4 of the testimony, you say it is anticipated that Westinghouse will have completed its generic program to select the most effective combination of auxiliary feed and/or steam generator modifications to enable installation and Staff review prior to start-up of Byron. And I am trying to probe to find out the basis for that statement.

We now know that Westinghouse -- we expect something from Westinghouse in the October-November time-frame, and now how long is the Staff going to take?

A Let me clarify that again, that what we are expecting from Westinghouse is not going to be a surprise, for example. It's something that has been discussed.

Q All right.

A And we are aware generally of what is there. What they will come up with is field data and test data from scale model testing. So essentially it will be a confirmation of what we feel are going to be the fixes, and the review for that should not take too long, if the end results do indeed conform with what we expect from them.

Q I see.

So these are in the nature of confirmatory studies and analyses?

A That's -- I think that would be correct.

Q So you feel that you know enough now that you could draw the conclusion that this problem of flow-induced vibration can be resolved prior to start-up of the Byron facility?

A We have gone farther along with the D-2 and D-3 fixes. We have reviewed the flow model test data and the analyses, and based on what has been accomplished there, we feel that these are very promising avenues, and these are very promising methods of approaching this problem. So we do feel that an adequate fix will be available, and we would like to see confirmatory results and analyses along these lines.

Q Is the supplement to the SER that -- strike that.

Somewhere in the materials there is a statement with respect to the flow vibration problem. The Staff intends to address it further in the supplement in the SER; am I correct in that?

A (Witness Murphy) Yes.

Q Is the purpose of that to deal with the results of these confirmatory studies?

A That particular part of the SER was meant to address either one, the fix or any other alternative approach that the Applicant would propose before start-up. The SER was written in a general way before, you know, we knew -- before we had much information from the Applicant or from Westinghouse regarding where they were going with this.

Our anticipation, as expressed in the testimony here, was based upon our understanding of what Westinghouse's schedule is for completing its design review and what they call their generic modification selection program.

Q And is that the October-November timeframe that Dr. Rajan mentioned, or is that a different timeframe?

A Yes. Yes. Yes. This was a date they gave us at a meeting here in May.

Now we can't -- we certainly can't make any conclusion regarding whether or not Westinghouse can meet its schedule, or we cannot speak for the Applicant, who may not choose to buy this modification.

Q What happens if the unexpected occurs and the analysis is either not completed by the time of start-up of Byron or it shows it's unacceptable to the Staff? What happens then, in terms of licensing the Byron facility? Dr. Rajan?

A (Witness Rajan) It would seem to me that if we do not find acceptable fix, we could limit the operation of the Byron plant to somewhat less than 100 percent power. That is one of the options that is obviously available.

Another option might be that it might be delayed in the extreme situation, the operation might be delayed.

A (Witness Murphy) Well, you know, there are factors -- we have a lot of things to consider. I think -- I believe that from a strictly technical standpoint, forgetting about questions like ALARA and so forth, I believe that a satisfactory technical basis could be arrived at, in terms of justifying an operating program for Byron. Based upon tne experience we have acquired day to day at McGuire, and based upon the experience overseas and what our knowledge of the Westinghouse analysis and test results regarding the causes of the problem are.

Q The Staff believes that -- I guess it is your present understanding and belief that the vibration problem doesn't occur below 70 percent. Would that be the power level you'd select if you were going to limit start-up of Byron, something less than 100 percent of full power?

A (Witness Rajan) At this point it would be conjecture, but that could be. That could be an option. We will have to examine the KRSKO data in far greater detail and make a determination as to what level of power operation would be safe.

Q But you're telling me at that time you'd have to review the data to see if 70 percent was still the correct number?

A That's correct.

Q What's happening at McGuire? Are they limited in some fashion presently?

A They are, they are limited.

Q Can you tell me what it is?

A They are limited to 75 percent power.

Q At the present time?

A At the present time.

A (Witness Murphy) This is following their start-up from the present outage?

Q Do you want to speak up?

A This was an aside. It should be off the record.

MR. GOLDBERG: If the witnesses want to confer before giving an answer, they are entitled to.

(Discussion off the record.)

MR. GALLO: All right, let's go on the record.

BY MR. GALLO:

Q Do you want to clarify, Mr. Murphy, Dr. Rajan's statement or testimony that McGuire is operating under a 75 percent power limitation?

A (Witness Murphy) McGuire has operated at different times under either a 50 percent or 75 percent power limitation. They are currently shut down, and the Staff is evaluating what it considers to be an acceptable program for future operation beyond this outage.

Q Do you know whether these limitations, these power limitations, were imposed by the Staff, or were they voluntarily assumed by the utility?

A (Witness Rajan) They made a recommendation and we reviewed the data and the analyses which formed the basis for proposal, and then we allowed them to continue for a certain period of time for 75 percent, up to 75 percent power, and as Mr. Murphy pointed out, they completed that period of operation recently, and now they are shut down.

A (Witness Murphy) They originally at one point last spring proposed a period of operation at 75 percent power. We found they provided insufficient justification for that power level, and limited them to 50 percent operation. They resubmitted their basis, and we bought off on it, based upon our review, the second time around.

Q You mean you agreed with the higher --

A We agreed with their justification, that they had reasonable justification, based upon their resubmittal on a technical basis.

Q For what level?

A 75.

A (Witness Rajan) 75.

MR. GALLO: I just want a minute here to look.

(Pause.)

MR. GALLO: Okay, I'm finished, Steve.

MR. GOLDBERG: Okay. I'd just like to note for the record that it's 1:05 p.m., and we have exceeded the allotted time limit, with the indulgence of the witnesses, and at their further indulgence, I am going to ask some further questions, but I would urge the parties to attempt to bring this to a rapid conclusion thereafter.

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E X A M I N A T I O N

BY MR. GOLDBERG:

Q   To the extent I address a question to Mr. Murphy and, Dr. Rajan, you wish to add a comment, or vice versa, please feel free to do so.

First, I wonder, Mr. Murphy, can you briefly describe your relevant educational background regarding your testimony or affidavit on Contention 9-C?

A   (Witness Murphy) I will. My education, I have an M.S. and a B.S. degree in the field of engineering. I got the B.S. in aeronautical engineering, and M.S. in civil. My specialty being structural engineering.

I have worked for the Bettis-Tye Power Lab for six years, engaged in the analysis and design of core structurals, the Naval Reactors Program. I was not involved at that time specifically with steam generators. This was primarily in the area of core structurals.

Since joining the NRC in July of 1979, I have been involved exclusively in the review of steam generator operating experience, surveillance programs, repair programs, et cetera, anything having to do with the integrity of the steam generator tubes, particularly with regard to operating reactors, I have been involved in those reviews.

With regards to my association with steam generator problems, that association goes back three years.

Q I think you have combined an answer that I was going to ask. I was also going to ask professional background, but I think you have given that in addition to educational background.

Let me ask you, what is your role in the Staff consideration of resolved safety issue A-3, which if I'm not mistaken, is the steam generator tube integrity task?

A For the past several weeks, at least the past two weeks, and probably for the next few days, we have been commenting extensively on the current draft --

Q I'm sorry, I'll take that answer, but I think it's not to the question I asked. I said what is your personal role in that effort?

A My personal role has been that for the past several weeks, has been to provide my comments to the draft report.

Q Are you part of that Task A-3?

A I'm not part of the task, per se. I have made contributions to the report.

Q In what areas?

A In the areas of steam generator inspection programs, procedures, sleeving, and surveillance in general.

Q These are subjects in which you have been engaged since you joined the NRC in, I believe, 1979, you say?

A That's correct.

Q I wonder, Dr. Rajan, if you can give me your relevant educational and professional background.

A (Witness Rajan) I have a B.S. in physics-chemistry, and another B.S. in civil engineering, with a major in hydraulics, and a Master's in structural mechanics, and a Ph.D. in fluid mechanics. I have worked for six years with the Naval Research Laboratory in their piping programs for nuclear submarines, and since '74, I have been with the Nuclear Regulatory Commission, and besides other things, been the principal reviewer in the mechanical engineering branch for problems of steam generators related to the mechanical engineering branch scope of review.

Q You have a prominent Staff role in consideration of this so-called flow-induced vibration phenomenon?

A I'm sorry, can you repeat that?

Q Yes. Do you have a prominent Staff role in terms of the consideration, further Staff consideration of the developments in this flow-induced or mechanical tube vibration problem?

A That's correct.

Q Mr. Murphy, there were several questions that Mr. Jenkins asked, all regarding past, present and anticipated steam generator tube integrity concerns.

I wonder if you can tell me just generally on the basis of your experience at what level of tube degradation does it become significant from a public safety standpoint?

A (Witness Murphy) Okay. That particular -- I interpret your question to mean how much -- how much leakage would it take during an accident before we got severe consequences. And that particular issue is addressed in some detail in the document -- the February 1982 document that has been made part of the record about 20 minutes ago. This is the one prepared by Jack

Strosnider in the Office of Research.

Q Okay, let me clarify. I'm not sure it's been made part of the record. It's been identified, and will accompany the transcript of this deposition.

I guess what I'm saying is there was a line of questions about what I think was described as overall steam generator tube problems, and I'm trying to get some kind of understanding of this, the magnitude of the problem, from the standpoint of public safety.

In other words, does a steam generator tube problem equate to a public safety problem, and where is the line drawn, based on your experience?

A That's a very complex issue. In my opinion, based upon what I know, I think that without proper controls and regulation, that steam generator tube degradation could ultimately lead to severe problems. The whole issue in terms of what the concerns are, and how we should be approaching these concerns is under -- you know, it's under study by the Staff, but I believe that based upon everything known to me of the analyses -- based upon my understanding regarding preliminary analyses concerning the consequences of an accident with ruptures and so forth, that our current regulatory approach is adequate for this interim period before we issue our final generic conclusions.

Q Let me ask it a little differently:

You indicated, I think, in response to some questions by Mr. Jenkins that you couldn't preclude some steam generator tube integrity problems over the expected lifetime of Byron; is that correct?

A It was -- I couldn't preclude corrosion.

Q Okay, we'll confine ourselves to corrosion.

A Or degradation in general.

Q Okay. What kind of measures give assurance that this isn't going to be a public safety problem? Should we be concerned about this?

A In my opinion, the current regulatory approach, which we have discussed quite extensively up to now, I think provides reasonable assurance for the public health and safety but, you know, I also believe that it's necessary for the Staff to reexamine all the relevant issues concerning the steam generators, both for what the safety concerns are, and I think we have to study and make the finding that the current regulatory approach is satisfactory.

Q Let me get at it differently:

I gather that --

A Let me take away the word "satisfactory." Whether or not it should be approved further.

Q This corrosion-related steam generator tube integrity phenomenon is considered generally an unresolved safety issue; is that correct?

A Yes.

Q And that is part of an ongoing Task A-3; is that correct?

A Yes.

Q To which you referred.

Now you have also taken a position, I believe, in the SER and in your testimony that notwithstanding those ongoing efforts to which you have continually alluded, that to quote you on paragraph 7, page 2, of the affidavit, that Byron can be operated before resolution of the above, which I assume is this particular generic issue, without undue risk to the health and safety of the public.

A That's correct.

Q Why don't you just, at the risk of redundancy, then tell me why.

A It's not really on the basis of -- well, the major basis for this finding is my assessment of operating experience. The fact that surveillance requirements have proven generally successful in preventing rupture occurrences, though not absolutely. That where we have had rupture occurrences, that these have not resulted in unacceptable consequences. The results of the consequences have not been severe. That even if we were to have a tube rupture assumed to occur concurrently with the design basis accident, the only Staff studies -- the conclusions of Staff studies that I'm aware of all indicate that the results would not be unacceptable.

That last item does not factor directly into any safety evaluation which we customarily prepare. It is something that I -- that provides me with some added assurance.

Q Go ahead, if you want.

A That's enough for now.

Q By the way, Dr. Rajan, if you want to add anything, go ahead at any time; not that I'm inviting a response now.

You indicated that as part of your answer to my last question, you alluded to the steam generator tube ruptures that have occurred in the past; am I correct?

A I'm sorry?

Q In the past, there have been past instances. I believe in response to questions by Mr. Jenkins, you identified four instances in which there have been steam generator tube ruptures on domestic reactors; is that correct?

A That's correct.

Q Did any of these ruptures result in impermissible releases to the environment, to your knowledge?

A To my knowledge, none of these rupture events resulted in unacceptable releases to the public, or to the environment.

Q By unacceptable, do you mean --

A To my knowledge, no requirements, 10 CFR 100 or otherwise, have been violated. The radiological consequences of the rupture events which have occurred, have been evaluated in detail by the Staff, and have been reported upon and documented.

Q Has the Applicant, to your knowledge, done a steam generator tube rupture accident analysis, as part of its application?

A The answer to that is yes.

Q Was it done as a design basis accident?

A The tube rupture event?

Q Yes.

A It's my understanding that that is a design basis condition.

Q Okay. I believe there were earlier questions about the necessity to consider a steam generator tube rupture coincident with some other significant design basis accident, such as a LOCA, and you indicated by -- well, I'm not sure what you --

A That's not a design basis accident.

Q It's not presently then a Staff requirement for design basis accident, and you're not making any recommendation today whether or not it should be; is that correct?

A No, I'm not. It's something under evaluation by the Staff.

Q And it's under evaluation, I gather, as part of the overall review of steam generator tube integrity problems generally; is that correct?

A Yes. But as this report that's been identified, this February 1982 report, discusses, the analyses that I'm aware of that have been performed to examine the consequences of tube failure concurrent with an accident indicate that the consequences of a tube rupture, a single tube rupture during accident conditions will not result in unacceptable consequences, whether we're talking about a LOCA or a main steam line break, or what-have-you.

Q The February 1982 document you are referring to is the one that Mr. Gallo earlier marked as Deposition Exhibit 1?

A That's correct.

Q Let me talk about this mechanical tube vibration problem for a moment. I guess I'll direct my comments, then, to Dr. Rajan.

I wonder -- first of all, the Ginna accident which was referred to earlier, I believe it was your testimony that that did not result from this flow-induced problem; is that correct?

A (Witness Rajan) That is correct.

Q I wonder if you can distinguish for me the

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difference between the Model D-2, D-3, and Model D-4, D-5 Westinghouse type steam generators from the standpoint of both their susceptibility to tube vibration problems and, secondly, their amenability to corrective modifications.

A The one thing in common with D-2 and D-3, on the one hand, and D-4 and 5, is that both are preheat type steam generators. The difference is that in the D-2, D-3, the flow is split upwards and downwards, as it emerges from the feedwater nozzle.

In the D-4 and D-5, on the other hand, the flow is directed downwards, and this design is referred to as the counterflow type, in which the flow is directed downwards, and then it goes upwards again to a series of baffles.

The fact that the flow does not impinge on the tubes directly as it comes from the feedwater nozzle, in my judgment, somewhat reduces the possibility of flow-induced vibrations in the D-4, D-5 model. Although let me emphasize that this whole area of preheat region is an area of turbulence, and the first row of tubes has evidence from data at KRSKO to experience unacceptable flow-induced vibration.

So we do recognize that there is a problem, but in my judgment, the magnitude of the problem is somewhat less severe in D-4, D-5, as opposed to the D-2, D-3.

Q Let's assume for a moment that contrary to present expectation, the corrective modifications are not available and satisfactory at the time that Byron is ready to operate.

In questioning by Mr. Gallo, I think you gave some alternative measures that were available. I wonder if you can just briefly summarize what posture you think you'll find yourself in if that's not the case.

A Well, at the end of the review of the options -- of the analyses and test results from Westinghouse, if we do conclude that it has not been adequately demonstrated that the flow-induced vibrations have been eliminated or reduced to within acceptable limits, then at that point we will have, in my judgment, two options:

One of them would be again based on the results of the test data and operating plant experience at that point, we can limit the power at Byron to less than 100 percent. And until such time that a fix can be found.

The other option, of course, would be to delay,

delay the start of the plant until a fix has been -- acceptable fix has been found.

Q Okay. Both of these alternatives -- some kind of power restriction and delay -- would you say it's fair to characterize those as primarily imposing an economic burden on the Applicant, as distinct from representing a safety problem to the public?

Is my question unclear?

A Our decisions would primarily be based on what the effect of the fix would have on the safety.

Q Okay. So clearly, then, the objective of some kind of alternative plan, then, would be to ensure public safety during some period of operation or defer operation if it was found that it could not be satisfactory?

A That's right.

Q So just to follow up, then, the absence of final corrective modification at the time Byron may be prepared for power operation does not mean that there is not some alternative measure by which public safety can be assured pending some ultimate corrective modification or fix; is that not correct?

A Yes.

MR. GOLDBERG: Hold on for one second.

(Pause.)

BY MR. GOLDBERG:

Q Mr. Murphy, let me ask you one or two particular questions about some areas in which you were examined by Mr. Gallo.

One of the measures identified in your affidavit on page 2, paragraph 5, to minimize the onset of steam generator tube integrity problems, is utilization of all volatile secondary water treatment; correct?

A (Witness Murphy nodding.)

Q Does all volatile treatment chemistry -- excuse me, chemistry control add phosphates to the steam generator?

A (Witness Murphy) No.

Q Does it then prevent phosphate wastage on new plants such as Byron?

A Well, you don't have the ingredients for phosphate wastage, as I understand.

Q So is it your belief and is it your testimony today that these -- I'm sorry, strike that.

Okay, a second component of the second measure that you identify as designed to minimize steam generator tube problems, is improved controls and monitoring of secondary water chemistry; is that correct?

A Yes.

Q Is it your belief that improved chemistry controls will reduce corrosion to at least controllable levels?

A Yes. Yes.

Q Okay, how does both the all volatile treatment chemistry control and other improved chemistry measures protect the steam generator from corrosion?

A The chemistry AVT affects the -- AVT is a method for treating the secondary water, for scavenging the oxygen, for gauge control, and its function is to minimize corrosion problems relating to the secondary coolant.

I cannot go into it any deeper than that, because quite simply corrosion is not my specialty. The dynamics.

Q Okay, Mr. Murphy. As I understood your answer to questions by Mr. Gallo, you referred us to the statement made in Appendix C, Section A-3, containing the Staff review of the steam generator tube integrity problem which you adopted as a principal response to questions concerning, I guess, the hazard of steam generator tube failure problems coincident with other accidents; am I correct?

A Yes.

Q And that statement -- and let me just read you the statement, I realize you may not have it in front of you. The statement is that the primary concern is the capability of steam generator tubes to maintain their integrity during normal operation and postulated accident conditions. Is that correct?

A That's right. And therefore it pertains to the hazard presented by various degradation mechanisms like stress corrosion cracking and so forth.

Q And if we look to Contention 9-C, it cites the steam generator tube integrity problem stemming from corrosion cracking and denting and fatigue, and goes on to submit that this constitutes a hazard, both during the normal operation and under accident conditions.

A You're asking me to respond to that, and the response is indicated in item, I believe, 5 of the affidavit -- of the testimony, which states that we have implemented -- requirements have been established to keep these degradation problems from becoming a hazard to the public.

Q Okay. Realizing the limitations, perhaps, on your knowledge and experience in this area -- and, Dr. Rajan, if you have anything to add, I'm just going to kind of ask you the question that maybe Mr. Gallo stopped short of, but in the context of that particular contention, do you have an opinion about what kind of incremental risk is posed by the addition of the steam generator tube rupture to a design basis accident?

Let's maybe stick with one of the worst, a design basis LOCA.

A I'm not sure I understand the question. You are saying -- does the question assume a tube rupture concurrent with an accident?

Q Assume that you have a design basis accident, let's say it's design basis LOCA, one of the most severe. What incremental contribution would a steam generator tube rupture have to the severity or public risk stemming from that kind of an accident?

MR. GALLO: I'm going to object to the question. The witness has already testified he is not an expert in that area. I just make that for the record. Go ahead.

MR. GOLDBERG: That's your objection.

WITNESS MURPHY: Well, I think it's safe to say that tube rupture during a design basis accident aggravates the severity of the accident. I don't think there is any doubt about that.

The question is whether -- how much does it aggravate the accident. The Staff is -- other organizations are pursuing studies in this area. The document that has been identified, the internal -- I guess it's classified as an internal NRC document, addresses the preliminary findings, or the findings of those analyses.

BY MR. GOLDBERG:

Q Unless you have anything to add, Dr. Rajan --

A (Witness Rajan) The only thing I would say, that these studies have been pursued and we are aware of some of the results of these studies.

My understanding is that during a loss-of-coolant accident, several -- a rupture of several tubes can be tolerated.

Now, the exact number I'm not aware of.

A (Witness Murphy) Well, the exact number, according to the existing analyses, is 1300 gpm. This is more than the leakage that you would expect during a LOCA, even if you assumed the tube completely double-ended during a LOCA, which is an unlikely failure mechanism. But this amount of leakage considerably exceeds the expected leakage in the amount of the double-ended failure of a tube during LOCA conditions.

Q You are acquainted with existing Staff analyses that document that point?

A Like I say, I have testified here that I am acquainted and have read the results and conclusions, and am familiar with the results and conclusions of the analysis.

MR. GOLDBERG: Okay. I have no further questions.

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MR. JENKINS: I have three, very brief, that were raised, but before we go into those questions, I would like to note for the record Mr. Connell's presence here, which I do not believe was noted in the record.

MR. GALLO: Who? Mr. Connor?

MR. JENKINS: Mr. Connell? Is that your name?

MR. GALLO: Connor.

MR. CONNOR: Connor.

MR. JENKINS: Connor. I'm sorry.

MR. GALLO: I noted that when he entered the room. You'll find that in the transcript, I'm sure.

MR. JENKINS: I'm sorry.

MR. GALLO: He's from Westinghouse, in any event.

MR. JENKINS: All right. Very good.

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RE-EXAMINATION

BY MR. JENKINS:

Q The first question, you have noted that the inspection frequency typically is 12 to 24 months. Why is McGuire inspected more often, every two months?

A (Witness Murphy) Because we are concerned that with the vibration, or potential for vibration, that they have, that if we -- because of uncertainties that exist regarding the exact wear rates that take place at 50 percent power, 75 percent power, what-have-you, we feel that much more frequent inspections like are being performed are needed to ensure that the wear will not proceed beyond acceptable limits, before these tubes are reinspected.

Q But I guess I don't understand; if you would normally inspect every 12 to 24 months, why are you inspecting in this circumstance every two months?

A (Witness Rajan) The rate of flow-induced vibrations in certain groups is rather rapid, and if we allow the plant to operate for the 12-month interval, which it is normally scheduled for, it is quite likely that several tubes would reach their plugging limit prior to that, completion of that operation period, and may begin to leak, perhaps.

Q Are you saying that McGuire has some unique problems?

A That's correct.

A (Witness Murphy) All Model Ds have unique problems.

A (Witness Rajan) All Model Ds have unique problems.

Q You averted to U-bend cracking. Now that is something that I have never been familiar with before. Would you just briefly describe what that is?

A (Witness Murphy) It's described in excruciating detail in NUREG 0886. There are two U-bend cracking phenomena.

One is caused by -- is a direct result of denting in the upper support plate.

The other is not related to denting.

The denting-related U-bend cracking led to the Surrey rupture in 1976. The phenomenon is believed to be well understood now. There are a lot of tell-tale -- there are some tell-tale indicators that a plant may be approaching the condition where one must be concerned about such a failure, and sufficient surveillance -- steam generators can be surveilled to see that these conditions don't exist. Or if they do exist, you can take preventive measures.

The nondenting-related cracks have resulted in a number of small leaks at some units. I'm not sure to tell you just where to go. It appears to be related to -- the occurrence of the cracks is related in part to the residual stresses in the U-bend as a result of the bending process.

Q What class of incident was given to the Ginna accident? As I understand it, incidents are classified somehow in a numerical basis.

A I can't answer that question.

A (Witness Rajan) I'm not aware of it.

MR. CHESNUT: We have two or three different types -- at least two different kinds of classifications. I don't know if you are familiar with the numbering system or --

MR. BUNCH: Yes.

MR. GOLDBERG: Mr. Chesnut is not sworn.

MR. JENKINS: But these gentlemen said they don't know.

MR. JENKINS: Mr. Gallo?

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R E - E X A M I N A T I O N

BY MR. GALLO:

Q Has the Staff, Dr. Rajan, conducted any analysis to determine the likelihood of a gross tube failure occurring the course of a design basis accident?

A (Witness Rajan) Are you talking in terms of a probability number now?

Q Yes.

A No, sir, I have not.

Q Are you aware of anyone on the Staff that might have performed such analysis?

A I am not aware of anyone.

Q How about you, Mr. Murphy?

A (Witness Murphy) The analyses which have been done have determined the probability of tube ruptures during accidents, assuming that you have -- that you have degradation in the steam generator.

These analyses were done for the purposes of evaluating various sampling plans versus the tolerable number of tube failures during an accident.

Q Is it possible -- I'm sorry, go ahead.

A Insofar as I know, there has been no risk assessment evaluating the probability of a tube rupture at any plant if it had a main steam line break.

Q Do either of you know whether or not it is possible to perform such analysis? Just offhand, the uncertainties seem to be so great that it would seem to be difficult to perform an analysis. Do either one of you have an opinion on that point?

A (Witness Rajan) I would think that it is possible to make an analysis although the uncertainties about the accuracy of such an analysis would be rather high.

A (Witness Murphy) Jai and I have been recently discussing such a risk assessment, but this is -- to our knowledge, there is no -- there has been no -- well, maybe I'd better -- I'm not aware of a specific risk assessment regarding the potential for steam generator tube ruptures during an accident.

Q Okay. Two short questions:

You have been referring here today to design basis accidents and the occurrence of a gross steam generator tube failure during the course of such an accident. What ones are we talking about? We mentioned a design basis loss-of-coolant accident which I guess for a

PWR would be the double-ended guillotine break. Is that correct?

A Yes.

Q And the main steam line break, is that another design basis accident that we would be concerned with?

A (Witness Rajan) Yes.

A (Witness Murphy) Yes.

Q And I think, Dr. Rajan, you mentioned an earthquake situation. Are you talking about SSE?

A (Witness Rajan) Yes, that would be a loss-of-coolant accident in conjunction with a safe shutdown earthquake, in conjunction with an SSE.

Q How about a break to a feedwater line, or is that not a PWR problem?

A It is a PWR problem.

Q Would that be another design basis accident?

A That would be another design basis accident.

Q Are there any others?

A Well, for the steam generator, I would say that these are the main design basis accidents, or the bounding design basis accidents.

Q Just one last question:

Out of curiosity, if I wanted to go find a list of these accidents, where would I find them? Are they in a Standard Review Plan some place, identified as design basis accidents, or a reg guide, or regulation?

A (Witness Murphy) The -- maybe you can correct me if I'm wrong, Jai, but the regulatory guide provides guidance.

For example, Regulatory Guide 1.121 provides guidance on this sort of --

A (Witness Rajan) That lists these accidents.

MR. GALLO: That's all I have. Thank you.

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R E - E X A M I N A T I O N

BY MR. GOLDBERG:

Q Dr. Rajan, the accidents you have just identified for Mr. Gallo and called bounding, those are all fairly low probability accidents, aren't they?

A (Witness Rajan) They are.

Q Coupled with a steam generator tube rupture, they would be even lower; is that correct?

A That's right. That's absolutely right.

MR. GOLDBERG: Okay. I have no further questions.

(Whereupon, at 1:55 p.m., the deposition was adjourned.)

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EMMETT MURPHY

____

JAI RAJAN

STATE OF MARYLAND :

COUNTY OF MONTGOMERY :

I hereby certify that the above-named witnesses, EMMETT MURPHY and JAI RAJAN, personally appeared before me and signed and subscribed to this deposition.

Notary Public,
Montgomery County, Maryland

My Commission Expires:

STATE OF MARYLAND :

COUNTY OF MONTGOMERY :

I, ANN RILEY, a Notary Public in and for Montgomery County, Maryland, do hereby certify that I reported in Stenomask the deposition of EMMETT MURPHY and JAI RAJAN, the witnesses herein.

I further certify that the foregoing pages are a true and correct transcription of the testimony given.

I further certify that the deposition was transcribed either by me or under my personal supervision.

I further certify that I have no interest, financial or otherwise, in the outcome of this proceeding.

Given under my hand and seal of office, this the 9th day of July, 1982.

Ann Riley
Notary Public,
Montgomery County, Maryland

My Commission Expires:

July 1, 1986

Murphy / Rajan Deps. # 1
7-7-82
(Ann Riley)

STEAM GENERATOR STATUS REPORT

FEBRUARY 1982

U.S. NUCLEAR REGULATORY COMMISSION

I. Problem Defintion

A. Summary of Tube Degradation

Degradation of steam generators (SG) manufactured by each of the three pressurized water reactor (PWR) vendors has resulted due to a combination of steam generator mechanical design, thermal hydraulics, materials selection, fabrication techniques, and secondary system design and operation. In the early and mid-1970s, Westinghouse (W) S.G. experienced caustic stress corrosion cracking, and W and Combustior Engineering (CE) S.G.s experienced tube thinning (wastage). These modes of degradation were due to difficulties encountered with phosphate secondary water chemistry. Because of these difficulties, most W and all CE plants converted to an all volatile (AVT) secondary water treatment. Although this conversion greatly reduced the occurence of stress corrosion cracking and wastage, other degradation modes including denting (deformation of the S.G. tubes due to corrosion of the carbon steel support plates) began to occur.

Babcock and Wilcox S.G., which have a significantly different design from W or CE and have operated exclusively with AVT water chemistry, had relatively good operating experience in their early years of operation. Nevertheless, they have experienced numerous tube leaks. The principal modes of degradation in B&W units have been fatigue crack growth, confined primarily to limited sets of tubes located on the open inspection lane, and more recently erosion-corrosion and primary side intergranular attack.

To date, many different forms of steam generator degradation have been identified including: stress corrosion cracking, wastage, intergranular attack, denting, erosion-corrosion, fatigue cracking, pitting, fretting, and support plate degradation. One or more of these forms of degradation have affected at least 40 operating PWRs and have resulted in extensive S.G. inspections, tube plugging, repair, or replacement. Recently, foreign Westinghouse S.G.s of the same design as McGuire have experienced tube wear associated with flow induced vibration due to a new integral preheater design. References 1, 2, and 3 present detailed discussions of domestic S.G. operating experience.

The economic impact of steam generator degradation has been significant. Approximately 23% of non-refueling outage time has been attributed to steam generator degradation. The cost of such outages in terms of replacement power alone is very high. However, perhaps the greatest financial costs incurred to date are those associated with steam generator replacement. Replacement of the Surry Unit 1 and Unit 2 S.G.s cost approximately $200 million, including cost of makeup power. Replacement of the Turkey Point S.G.s, currently in progress, will cost an estimated $460 million. NRC staff time involved with these activities is estimated at 6000 manhours for Turkey Point (which included time for a hearing) and 3000 manhours for Surry. Less radical operations also incur significant costs. Recent tube sleeving operations at San Onofre involved repair of approximately 7000 degraded tubes at a cost of $70 million. Proposed sleeving of 3000 tubes at R.E. Ginna has an estimated cost of $20 million.

B. Safety Significance

The safety significance of S.G. tube integrity can be divided into three categories: tube failures under normal operating conditions; tube failures concurrent with postulated accident conditions; and personnel exposure associated with S.G. inservice inspection (ISI), repair, and replacement.

The majority of the S.G. tube failures that have occurred under normal operating conditions were small stable leaks sometimes requiring plant shutdown, inspection, and corrective actions, but for the most part small enough (e.g., below technical specification leak rate limit) that operations continued until a scheduled shutdown. However, four significant S.G. tube ruptures have occurred in domestic PWRs since 1975. These events occurred on February 26, 1975, at Point Beach Unit 1, September 15, 1976, at Surry Unit 2, October 2, 1979, at Prairie Island Unit 1 and on January 25, 1982, at R. E. Ginna. The first three of these events were evaluated in NUREG-0651, "Evaluation of Steam Generator Tube Rupture Events." The report includes an evaluation of system response, operator action, and radiological consequences during the three events. The leak rate associated with these events ranged from about 80 gpm to 390 gpm. The conclusion of the report is that no significant offsite doses or systems inadequacies occurred during the tube rupture events analyzed. However, the potential for more significant consequences was recognized and a number of procedural recommendations were made to correct the deficiencies that were noted. The present disposition of each of the recommendations is discussed in a recent memo to Commissioner Bradford from W. Dircks (Ref. 4). The present design basis for assuring that plants are acceptably protected against S.G. tube rupture events is a postulated double-ended rupture of a single S.G. tube. This assumption is intended to provide a bounding leak rate for a spectrum of rupture geometries in a single tube and a spectrum of smaller leaks in multiple tubes within a single S.G. The consequences of multiple tube failures, in excess of the design base, * have not yet been rigorously studied. Rapid degradation between inspections of a large number of tubes could create the potential for multiple tube failures in the event of a plant transient or failure of a single tube and the accompanying jet impingement and tube whip could cause failure of additional tubes. Furthermore, the potential for complicating circumstances involving multiple equipment failures such as the stuck open PORV during the Ginna incident and possible steam bubble formation in the primary * system have not been evaluated. Another concern is ruptures in multiple S.G.s. In this event, unless the plant can be rapidly depressurized and * brought onto Residual Heat Removal, there is the potential to continuously lose emergency core cooling water outside of containment. The above concerns are being addressed as part of the TMI Action Plan. Item I.C.1 in the TMI Action Plan addresses S.G. tube failures coupled with other failures (such as a stuck open safety relief valve in the secondary system), ruptures of multiple tubes, and simultaneous ruptures in multiple S.G.s. The purpose of this effort is not to expand the plant design basis but to assure that operator emergency procedures provide proper guidance for safely controlling the plant during these types of events. Although rigorous analyses of many of the scenarios postulated above have not been completed, ISI, leak rate limits, and tube plugging requirements are intended to guard against such occurrences (See Section II). In summary, the consequences of S.G. tube ruptures under normal operating conditions have been small; however, such events can present a significant challenge to plant operators and safety systems.

During postulated accident conditions, such as main steam line break (MSLB), feedwater line break, or LOCA, the S.G. tubes are subject to increased pressure differentials and possible pressure waves (e.g., subcooled decompression phenomena) and vibrational loadings. These loads increase the potential for failure of degraded S.G. tubes which could exacerbate the accident sequence. In the event of MSLB, failed S.G. tubes would provide a leakage path from the primary to secondary system and several potential leak paths for radioactivity to the environment would then exist. In the event of a LOCA, the core reflood rate could be retarded by steam binding. This phenomenon is associated with a cold leg break, in which reflood of the core requires displacing steam generated in the core through the hot leg, the affected steam generator, and out of the cold leg break. S.G. tube failures would create a secondary to primary leak path which aggravates the steam binding effect and could lead to ineffective reflooding of the core. Analytical and experimental evaluations of this phenomenon are contained in References 4 and 5. Large MSLBs and LOCAs are considered extremely low probability events, but are postulated as bounding conditions. More realistic events might include small and intermediate size MSLBs or LOCAs. Although these postulated accidents pose a less severe challenge to S.G. tube integrity, tube rupture(s) leading to or following such events could have serious consequences. This is particularly true if fuel damage has occurred as in the case of Three Mile Island.

The final area of concern is the radiation exposure of personnel involved in S.G. inspection, repair, and replacement. Reference 3 presents a summary of data on S.G. related personnel exposure for selected plants from 1974 to 1980. In recent years, as much as 25% of some plants annual occupational exposure has resulted from routine S.G. inspection and maintenance and as high as 60% for S.G. replacement. Recent tube sleeving operations at San Onofre incurred 3500 man rem exposure and similar operations are planned for other plants.

II. Regulatory Approach

The NRC approach to assuring S.G. tube integrity under all operating conditions is based on inservice inspection (ISI), primary to secondary leakage rate limits, and preventive tube plugging requirements. Guidance for performing ISI is provided in R.G. 1.83, "Inservice Inspection of S.G. Tubes," and plant technical specifications include requirements for ISI. Typical plant specifications require periodic inspections of 3% of the S.G. tubes in the plant and augmented ISI in the event tube degradation is detected. Required frequency of inspection is generally flexible enough to allow inspections to be performed concurrent with refueling outages. Certain incidents such as tube leakage require unscheduled ISIs. Furthermore, many plants with extensive degradation problems have licensing amendments imposing higher frequency and larger size inspections. The ISI requirements were developed largely through a combination of engineering judgement and operating experience. More rigorous statistically based ISI programs have been developed as part of Unresolved Safety Issues A-3, A-4, and A-5 (see Section V). The purpose of the required ISIs is to determine if tube degradation is occurring in the S.G., assess the rate of tube degradation based on results of successive inspections, and identify those tubes requiring plugging or repair.

Primary to secondary leak rate limits are an extremely important requirement for ensuring safe S.G. operation. Some forms of tube degradation have been observed to degrade tubes beyond the prescribed plugging limit during the interval between inspections. Technical Specification primary to secondary leak rate limits requiring shutdown, ISI, and corrective actions provide protection against unacceptable levels of degradation between inspections. Many serious conditions of tube degradation have been detected by monitoring of primary to secondary leakage and subsequent inspection. Primary to secondary leak rate limits exist in each plant's technical specifications. The bases for these limits are twofold. First, the leak rate limit ensures that the calculated dosage contribution from tube leakage will be limited to a small fraction of the allowable limits in the event of a S.G. tube rupture or MSLB. Second, the leak rate limit is intended to correspond to a defect size that would not be expected to result in tube rupture under normal or postulated accident conditi

Finally, degradation limits for tube plugging exist in the plant Technical Specifications. Criteria for establishing the tube plugging limits are presented in R.G. 1.121, "Basis for Plugging Degraded Pressurized Water Reactor Steam Generator Tubes." These criteria require that the plugging limit include margins for eddy current testing error and continued degradation between inspections. Thus, it is important to have a good estimate of the rate of degradation based on successive ISI results and an understanding of the degradation phenomena.

The primary focus of the current NRC philosophy is directed at maintaining primary system integrity. This is accomplished primarily through the requirements described above for ISI, leak rate monitoring, and tube plugging. In a sense, it is directed at treating the symptoms and not the cause of S.G. degradation, which lies primarily in secondary system design and operations. This philosophy has been debated extensively, but the current position regards eliminating the problem at its source as an industry responsibility.

III. Current Corrective Actions

An effective solution to S.G. tube degradation problems would require major changes in S.G. mechanical design, thermal-hydraulics, materials selection, fabrication techniques, and changes in the secondary system design and operation. Elimination of S.G. degradation requires a systems approach integrating all of these considerations. There are no simple corrective actions. This is particularly true for those plants which have significant operating time and have experienced S.G. degradation. Design changes in operating S.G.s that would be necessary to eliminate degradation problems are virtually impossible. For example, tube to tubesheet crevices already contaminated with corrosive environments are virtually impossible to clean, carbon steel support plates cannot be replaced with more corrosion resistant materials, and residual fabrication stresses cannot be removed. Thus, corrective actions may prolong S.G. life, but tube degradation is expected to continue in operating plants. Once the secondary system is contaminated by an aggressive environment it is difficult to reverse the adverse affects. For example, caustic stress corrosion cracking and wastage, due to residual phosphate water chemistry conditions, still continue in some plants long after conversion to AVT water chemistry.

Several corrective actions, however, have been proposed and are in use. These fixes include such actions as tube sleeving, sludge lancing, soaking and flushing, reduced operating temperatures to slow corrosion, boric acid injection to arrest denting, support plate modifications to retard denting, S.G. replacement, and improvements in secondary system design and operation. Secondary system improvements include prompt correction of condenser in-leakage, condenser retubing, removal of copper based alloys from the secondary system, and addition of demineralizi systems. An industry constituted secondary water chemistry guidelines committee, under chairmanship of EPRI, is developing generic chemistry limits and operating guidelines. NRR has been in contact with this committee for the past year and will review a copy of the draft reports prior to issue. Chemical cleaning has also been proposed but has not been implemented due to uncertainties regarding its longer-term affect on S.G. integrity. Industry efforts are currently underway to eliminate these uncertainties and chemical cleaning may become a viable option in the near future. These fixes have met with varying degrees of success, but none of them is a panacea. Furthermore, short term solutions to one * problem may create other problems. Conversion from phosphate to AVT water chemistry, which minimized wastage and stress corrosion cracking but was followed by denting, is a case in point.

Finally it should be noted that the majority of the plants * under review for operating licenses have S.G.s of similar design to those currently in operation, so that the potential for S.G. tube degradation exists in these plants as well.

IV. NRC, Industry, and Foreign Research and Development Activities

NRC's steam generator research program addresses improved eddy current inspection techniques for steam generator tubing, stress corrosion cracking of steam generator tubing and evaluation of tube integrity.

The objective of the eddy-current program is to upgrade and improve eddy-current inspection probes, techniques and associated instrumenta for inservice inspection of steam generator tubing to improve the ability to identify and characterize tube defects. Specific objectives include improving defect detection and characterization as affected by tube diameter and thickness variations, tube denting, probe wobble, tubesheet and tube support interference, and defect location and type.

The stress corrosion cracking program is developing data and models which will be used to predict the stress corrosion cracking initiation and service life of Inconel 600 steam generator tubing. The testing program includes variables which influence stress corrosion cracking such as temperature, stress, strain and strain rate, metallurgical structures and processing, and ingredients in the primary and secondary coolant.

A steam generator, with service induced degradation will be used for the validation of the accuracy and confidence limits of nondestruct inspection instrumentation and techniques; burst and collapse tests on field degraded tubes to validate tube integrity models; and for developing data for validation of previously developed stress corrosion cracking predictive models, chemical cleaning and decontamination, dose-rate reduction and secondary side characterization. In addition, statistically based sampling models for inservice inspection programs will be confirmed and/or improved utilizing the first ever confirmed data base.

There are many ongoing programs addressing S.G. issues at EPRI, most of which are sponsored by the S.G. Owner's Group, and the rest by EPRI itself. The programs address the following areas: (1) chemistry and corrosion, (2) materials selection and testing, (3) thermal hydraulic and structural testing and analysis, and (4) nondestructive examination (NDE). Efforts in the chemistry and corrosion area are directed at examining the causes of corrosion related degradation such as denting, intergranular attack, and stress corrosion cracking, and identifying potential fixes such as alternative secondary water chemistry treatments. Materials selection and testing efforts are directed at characterizing and evaluating the suitability of alternative tubing and S.G. materials. This includes consideration of new heat treatments for tubing and compatability of S.G. tubing with structural materials. Testing and analysis in thermal hydraulics and structures is directed at secondary side S.G. design and performance and their effect on S.G. tube integrity. The EPRI nondestructive examination programs focus on development of improved inspection techniques. These techniques include multiple frequency/multiparameter eddy current testing, automatic eddy current signal analysis, profilometry for quantifying dent configuration and strain levels in dented tubes, and methods for evaluating the condition of the tube support plates. In addition, EPRI has established the NDE Center in Charlotte, NC, dedicated to providing good NDE techniques, and effectively transferring research and development results to the industry.

Research and development activities underway on steam generators outside the USA are being funded at high levels in several countries. The Japanese are conducting a very large program with emphasis on thermal/hydraulics, and also on water chemistry and tube testing. To date, we have received little information on the progress or results of their programs. The French have work underway on eddy current NDE, crevice chemistry, and decontamination. There is work underway in Sweden on water chemistry. The Germans have work underway in eddy current NDE, and at KWU on primary side decontamination and secondary side cleaning; however, German steam generators are tubed with Incolloy 800 so much of their research is less relevant to ours. Finally, the Italians have underway a large program which will allow them to make new designs to avoid current and possible future problems.

V. Long Term Approach

A. Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity

In 1978, the NRC established Unresolved Safety Issues A-3, A-4, and A-5 (USI) regarding degradation in W, CE, and B&W steam generators, respectively. A draft report, NUREG-0844, presenting the proposed NRC staff resolution of these generic safety issues has been prepared and is currently being reviewed by NRR management prior to transmittal to the Committee for Review of Generic Requirements and the Commission and publication for public comment. The report integrates technical studies in the areas of systems analyses, inservice inspection (ISI), and tube integrity to establish improved criteria for ensuring adequate tube integrity and safe steam generator operation under all conditions.

In the systems analyses portion of the report, the consequences of steam generator tube failures during normal operation and postulated loss-of-coolant and main steam line break accidents are evaluated. The evaluation considers predicted fuel behavior, emergency core cooling system performance, radiological consequences, and containment response. The results of the systems analyses lead to proposed criteria for establishing a tolerable level of steam generator leakage during postulated accidents. ISI techniques are then evaluated, and statistically based ISI programs presented which, if implemented, would provide additional assurance that no more than the tolerable level of tube leakage, defined by the systems analyses, would occur during normal or postulated accident conditions.

In the tube integrity portion of the report, the behavior of degraded tubes during normal and postulated accident conditions and tube plugging criteria are evaluated. Proposed changes in operating procedures and design changes to minimize tube degradation are also identified.

Implementation of the proposed requirements and criteria developed in the program for resolution of the USI are not expected to totally eliminate S.G. degradation. The intent of the proposed requirements is to establish a logical approach to evaluating steam generator tube integrity and ensuring safe steam generator operation. The draft NUREG-0844 recommends criteria and requirements that can be used to evaluate current and future degradation programs in steam generators. The establishment of maximum allowable steam generator tube leak rates during postulated accident conditions and associated tolerable number of defective tubes is a major contribution to the evaluation of steam generator tube degradation problems. It provides objective criteria against which steam generator tube integrity can be evaluated. Similarly, the development of statistical ISI programs provides a rational, scientific basis that can be used to establish and evaluate ISI requirements that will ensure the above criteria are satisfied. Results from NRC S.G. research programs are expected to lay the experimental basis for many of these criteria.

In keeping with the NRC's current and past philosophy on this issue, the proposed regulatory requirements developed in the draft report focus on ISI programs and techniques and tube plugging criteria. <u>The primary responsibility for attacking the problem at its source and eliminating S.G. degradation is the industry's.</u> However, several of the requirements proposed in NUREG-0844 are intended to promote industry efforts in this area. For example, one requirement is to ensure that all operating plants have implemented an approved secondary water chemistry monitoring and control program. This is a requirement in the most recent version of the NRR standard review plan for licensing of new plants. In addition, this type of program has been implemented at some but not all operating plants. Under this requirement, it is the industry's responsibility to .tablish specific water chemistry limits and effective monitoring techniques. This will ensure that each utility at least considers the importance of secondary system water chemistry and puts in the effort to develop a comprehensive water chemistry program. Similarly, ISI requirements for condensers are proposed. These requirements will hopefully reduce the frequency of condenser in-leakage and encourage utilities to improve condenser performance. Use of noncopper based alloys when retubing condensers and feedwater heaters is also a requirement. Additional requirements are proposed for plants in the preoperating license stage and many recommendations for operating and future plants are made. The intent of the proposed requirements as stated in the report is to leave primary responsibility for correcting the S.G. problem in the hands of the industry, to allow the industry flexibility in addressing the issue, but at the same time, to strongly encourage proper industry actions.

B. Comprehensive NRC/Industry Program

The preceeding review has attempted to summarize the status of the S.G. issue at this time. As indicated, the NRC has many ongoing efforts to address this multifaceted problem. However, to date, joint NRC and industry cooperative efforts on this issue have not been extensive. This is due largely to the different focuses on the issue. NRC is primarily concerned with requiring adequate ISI and corrective actions to ensure primary system integrity, while the industry has been concerned with developing fixes to prolong S.G. service life and reliability. NRC and industry efforts have been primarily complementary in nature. However, to the extent that reliability implies safety and vice-versa the NRC and industry efforts are synonomous. Therefore, the staff is pursuing the development of a joint NRC and industry program to address both near-term and long-term actions required for continued safe operation of steam generators and ultimate resolution of the S.G. degradation problem. The intent is to evaluate the degree to which the NRC can expand its role in prevention of tube degradation and work with the industry to solve this problem. Efforts to determine the feasibility of this type of cooperative program have been initiated and proposals for a joint NRC and industry program will be presented in a later document.

REFERENCES

1. Eisenhut, Liaw, Strosnider, "Summary of Operating Experience with Recirculating Steam Generators," NUREG-0523, January 1979.

2. Liaw, Strosnider, "Summary of Tube Integrity Operating Experience with Once-Through Steam Generator," NUREG-0571, March 1980.

3. SECY-81-664, "Information Report - Steam Generator Tube Experience," from W. J. Dircks to the Commissioners, November 24, 1981.

4. Memorandum for Commissioner Bradford from W. J. Dircks, Status of Recommendations Made in NUREG-0651, "Evaluation of Steam Generator Tube Rupture Events," to be transmitted.

5. EG&G Idaho, Inc. Report TREE-NUREG-1213 (NUREG/CR-0175), "Investigation of the Influence of Simulated Steam Generator Tube Ruptures During Loss-of-Coolant Experiments in Semiscale MOD-1 Systems," May 1978.

6. EG&G Idaho, Inc. Report CAAP-TR-032, "Steam Generator Tube Rupture - Effects on a LOCA," November 1978.