OFFICIAL TRANSCRIPT OF PROCEEDINGS

Agency: Nuclear Regulatory Commission

Title: Public Workshop on Technical and Policy Considerations for Nuclear Power Plant License Renewal: Reactor Pressure Boundary

Docket No.

SESSION 2

LOCATION: Reston, Virginia

DATE: Monday, November 13, 1989 **PAGES:** 1 - 83

DF03
0/1

ANN RILEY & ASSOCIATES, LTD.
1612 K St. N.W., Suite 300
Washington, D.C. 20006
(202) 293-3950

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

- - - - -

PUBLIC WORKSHOP ON

TECHNICAL AND POLICY CONSIDERATIONS

FOR

NUCLEAR POWER PLANT LICENSE RENEWAL

- - - - -

Session 2

Reactor Pressure Boundary

- - - - -

Sheraton Resort Hotel

11810 Sunrise Valley Drive

Reston, Virginia

Monday, November 13, 1989

1:15 p.m.

SESSION LEADERS:

J. Richardson

L. Shao

PARTICIPANTS:

James A. Norberg, NRC

Matthew F. Cinadr, NYS-DPS

Edgar I. Landerman, Consultant

Charles Z. Serpan, Jr., NRC

I. Manoly, NRC

Joseph McCumber, Yankee Atomic Electric Company

L. R. Katz, Westinghouse

Cedric L. Child, Yankee Atomic Electric Company

William D. Hinkle, Yankee Atomic Electric Company

Vik Shah, INEL

Peter Stancavage, GE

Jim Chardos, GPU-N

L. B. Marsh, NRC

R.E. Nickell, EPRI

Peter D. Hedgecock, Nutech Engineers

Jim Perrin, Public Service Electric & Gas

Bill Corwin, Oak Ridge National Laboratory

Gilbert Zigler, SEA/ASME

John Nyquist, Detroit Edison - Fermi

PARTICIPANTS, Continued:

Charles W. Allison, National Board

Jim Higgins, MPR Associates

Johnnie B. Cannon, Oak Ridge National Laboratory

Terry R. Woods, TVA

Wayne Pavinich, TENERA

Paul Norman, NRC

Herb Massie, Westinghouse

Peter Bloch, NRC

John Bromgren, Commonwealth Edison

Rhonda Doney, Combustion Engineering

Ron Chrzanowski, Commonwealth Edison

Michael Mayfield, NRC

Joe Muscara, NRC

Art Deardoff, Structural Integrity Association

P R O C E E D I N G S

MR. SHAO: Good afternoon. Ladies and gentlemen, the microphones aren't working so I have to shout. Anyone who wants to speak will have to shout too. Anybody who wants to speak will have to identify themselves so that it can be recorded.

My name is Larry Shao. I'm the director of engineering, Office of Research. Today I'm the chairman of Section 2, Reactor Pressure Boundary. My co-chairman, James Richardson, is the director of engineering technology, Office of Nuclear Reactor Regulation.

Before we start questions, let me give you a short introduction related to the issues relating to primary pressure boundaries.

[Pause.]

MR. SHAO: As you all know the primary system pressure boundary must have very high reliability for the operational life of a nuclear power plant. It is necessary for reactor core cooling during shutdown and serves as a barrier to the release of fission products during accidents.

The principal components of the primary pressure boundary, listed here, mainly are the reactor vessels, steam generators, the pipings, pumps, and valves. Also, there is some instrumentation. Maintaining the high reliability of these components for the license renewal period is the thrust of this workshop.

Let me briefly discuss some of the issues related to these components.

[Pause.]

MR. SHAO: Reactor vessel materials in the region become brittle by neutron irradiation and thus load carrying capabilities -- it loses its toughness and when it loses its toughness, the reactor vessel is most susceptible to operating in accident loadings. The method for calculating thinning and embrittlement is provided by so-called Reg Guide 1.99, Revision 2.

Some of the older reactor vessels have high copper and nickel contents, and they are most susceptible to embrittlement. The four factors that are most important to embrittlement -- and these are the four factors, are copper, nickel, neutron fluence, and the irradiation temperatures.

In addition to embrittlement, the thermal stresses can also cause cumulative damage to the vessels. Both the reactor vessel internals, the irradiated, stretch, corrosion, cracking, may be an area of concern for certain vessel internals and core support structures.

We have seen many points of degradation. Here is a list of the degradations. The first one is firesign stress corrosion cracking. Mainly it results from religious stresses and occur at a new band and road transition in the tube area.

I understand the French have all kinds of problems with this area.

The secondary size stress core cracking occurs at the two sheet crevices and two support plate intersections. The fatigue crack corrogation was caused by frozen in use. The tube denting, which is regressing in diameter is caused by the corrosion of the carbon steel support plates. The intragranular attack, the corrosion attack of grain boundaries usually occurs at low flow areas. Flooding and wear occur the support plate and the anti-vibration member areas.

Pitting and wastage occurred in the sludge part region above the two cheese. Lastly, steel tube was prepared up causing a tube failure at Los Anna. The two-plugging failure was initiated also by primary stress corrosion cracking.

This is a list of potential degradation in piping that should be considered in license renewal. The BW piping has experienced intragranular stress growing cracking. IGSCC is mainly caused by the combination of sensitized material, religious stresses and high oxygen content and impurities.

For the cast stainless steel, the tests at Fermaton, Argonne National Lab and Western National Lab indicate the cast stainless steel piping loses load carrying capability when aged for a long time at operating temperatures. Thermal fatigue also may result from the poor mixing of hot and cold fluids.

Erosion corrosion is a flaw resisting general corrosion of carbon steel piping and it has occurred in Surry and to some extent at Trojan.

In the pump area, the pump shafts experience high cyclic alternate stresses which cause cumulative material damage. Mechanical removal of bearing material may occur due to long-term pump operations. The heat, humidity, flow, pressure and vibration may cause degradation in pump seal gasket and packing.

Erosion and corrosion of pump internal are caused by local high flow turbulence and chemical attack. The loosening of parts and distortion of certain compartments may result from long-term pump operations.

In the valve area, the valve disk and disk connection hinges experience high cyclic stresses and impact loading due to valve operation and flow excitations. This closure impacts may cause mechanical removal of material at valve seats. The heat, humidity, valve pressure and motion may cause seal leakage and insulation failure.

The full pressure set for valve actuation may drift but that may not be an aging problem. It's a current problem. Local high flow turbulence and chemical attack may cause erosion and corrosion of valve internals. Distortion of internal parts may result from long-term operations. Also, the mechanical removal of material at the valve stem and worm gear may occur due to abrasion.

The valve, disk and seat may be locked which may interfere with opening or closure. For motor operated valves, the bearing may be worn or broken due to long-term operation. The binding torque switch or limit switch may cause losing control of valve stem motion. So these are the lists of potential degradations we have seen now and we will see more in the next 20 years if the plants get an extension.

Do you have any questions on this?

If not, we go to the --

MR. CINADR: One question on pumps. Just as a topic, pumps just generally wear out and they do not deliver as they once did. These are all kind of catastrophic -- these are end-of-the-light problems, but there's another category somewhere between where it's just a worn out pump and it will not perform as it used to. Is that something that can be found or considered?

MR. SHAO: Hopefully, we'll find out from the IST program. We have an IST program, an inspection program, and this IST program should find out whether a pump is worn out or not.

MR. RICHARDSON: I would add to that. As the pump degrades in its performance, as Larry points out, that ought to be picked up in the normal operation of the plant and that loss of pump performance ought to be detected and the pump maintained and refurbished or replaced or whatever.

But I think we would consider that more in the normal surveillance --

MR. CINADR: Normal.

MR. RICHARDSON: Yes. Not something associated with life extension or anything. It's just the normal wear-out of pumps, as you point out. They don't deliver fluid at the rate that they were intended to and, when that happens, then, as part of the maintenance program, they would have to be refurbished.

MR. SHAO: For the reactor vessel piping, we have the ISI program. For the pumps and valves, we have the IST program. Hopefully, those would be picked up by these programs.

MR. CINADR: Thank you.

MR. LANDERMAN: Are you asking for comments now?

MR. SHAO: Yes.

MR. LANDERMAN: The name is Ed Landerman. You mentioned on the casting, casting the steel, if they lose maybe -- making a comment on your comments. You said they lose load bearing capability. I would say that maybe that's the way you're wording it, but that is not what happens. They certainly do not lose their load bearing capability. They certainly have those comparable properties that they had originally.

What you're talking about is a change in consideration of the toughness of that material.

MR. SHAO: But suppose you have a regular toughness, I don't know, 300 or whatever it is. After 10 years or 20 years, constant thermal embrittlement, you don't have 300 anymore.

MR. LANDERMAN: You have the same load bearing capability.

MR. SHAO: Originally, you have a bigger margin. That's okay. Suppose in your design margin you go to one, you may be in trouble. The original design margin may be ten. Even if you lose some toughness, you're still okay.

MR. LANDERMAN: I just want to make the comparison that this is no difference in surveillance tests of reactor vessels. We know there are changes in properties.

MR. SHAO: I would say the same thing about reactor vessels, but for other things, it may be different. You don't lose any capability. Some material, like -- suppose you have stainless steel piping. It can be there 20 years and your toughness would be the same.

MR. LANDERMAN: Okay. It's just a comment on that term. I think the analogy is more with changes of embrittlement to reactor vessels.

MR. SHAO: Reactor vessels are no different.

MR. LANDERMAN: Than stainless steel.

MR. RICHARDSON: I think another way of saying it is stainless steels that age are more susceptible to catastrophic failure through brittle failure.

MR. SHAO: The reactor vessel and cast iron steel are hopefully the same. Anymore questions?

MR. CORWIN: Bill Corwin, Oak Ridge National Lab. Two comments. One, I notice there was no mention of thermal, long-term thermal aging with respect to the reactor pressure vessel. I understand that's not currently within our codes, but I know that the British, in their licensing, are including up to a 30 degree C shift, even for modern reactor materials for long-term thermal aging in their pressure vessel.

Just a comment; I hadn't heard that specifically addressed and I didn't know if that was going to be considered in the life extension process.

Another comment is with respect to the adequacy of Reg Guide 199. I don't recall exactly how high the fluents limits are in that, that are explicitly addressed by 199, but some of the higher fluents plants may very well run off the top end of the scale in 199 and that's -- the higher fluents regions are something which may need to be reinvestigated relative to the rulemaking.

MR. SHAO: That's true, because we may not have the data for real high fluents.

MR. KATZ: I noticed you left off the list two components; pressurizers and CRDM. Is there some reason for that?

MR. SHAO: No particular reason. I just say these are the key components.

MR. LANDERMAN: One other comment. It falls within the reactor vessel and piping, which is a highlight, and that is -- it seems to me it should be highlighted. It's a similar metal combination. Everywhere we've talked about similar metal, concerns about casting, reactor vessel. An area of concern, as it should be considered, would be the dissimilar metal welds that are part of all these components or most of these components.

MR. SHAO: You're saying maybe bimetallic welding.

MR. LANDERMAN: Bimetallic, yes.

MR. SHAO: But bimetallic welding hopefully would be picked up also in the methodology, also in the fatigue stresses, with differential expansion and also methodology.

Any more questions?

MR. SMEDLEY: When you're talking piping, are you talking mainly out of the control room?

MR. SHAO: I'm talking about piping, pressure on all pipes. It can be any piping. If you allow thermal stress on the surge line. Recently, we found in many plants they have some kind of high thermal stresses which was not analyzed in the original FSAR.

MR. RICHARDSON: I might say to you that if you're going to ask questions or are going to make comments, if you would stand up and identify yourself and speak very loudly because it has to be picked up by these microphones up front to be recorded.

MR. SHAO: Any more questions?

[No response.]

MR. SHAO: If not, why don't you start.

MR. RICHARDSON: Okay. Good afternoon. My name is Jim Richardson. We've put together some seven questions that we as the staff have to you as industry and citizens to help us put together the rule. These are certainly not totally inclusive of all the questions that need to be raised in the area and I'm sure there are questions that you have or comments that you have that go beyond these.

The way I would like to operate this is, first, I would like to hear your comments on these specific questions and then, after we get through these seven questions, I'm sure some of you may have comments outside of these specific questions and we would welcome those comments.

But, first, I would like to get through these seven specific questions.

MR. SHAO: Do you think it's a good idea to have a summary of each question?

MR. RICHARDSON: A summary of each question.

MR. SHAO: Well, after we go through each question, we have a summary.

MR. RICHARDSON: Okay. The first question deals with the surveillance program in the reactor vessel and, of course, Appendix H of Part 50 of the Federal Regulations have set forth the surveillance requirements. However, those requirements are based on the assumption of a 40 year life.

The question that we're raising is what additional requirements should be set forth and implemented to account for life beyond 40 and how should the surveillance program be adjusted to account for that, considering the fact that surveillance capsules have been removed on a periodic base with the assumption that end-of-life comes in 40 years.

Is there someone that would like to address that question?

MR. SMEDLEY: I'm Rick Smedley.

This Appendix H thing is right in Appendix G, which means it's in the Reg Guide 199, rev. 2, which is quite conservative. Now, with modern calculation methods of reactor pressure, do you foresee the requirements being more exact, for instance, the Reg Guide 1.154 analysis and the NUREG 4744 analysis, one having to do with the steam criteria and the other having to do with the upper shelf?

MR. RICHARDSON: As I understand your question -- or your comment, and your question to us is, do we anticipate a more exact type of analysis for licensing renewal.

MR. SMEDLEY: I'm just interested in the 50 foot-pound criteria. Some of us are going to exceed the steam criteria of 5061. CE just came up with a new method of analysis, and I think they put in a proposal, but I'm not sure. I can foresee everybody having to go with this system. Have you considered that in this?

MR. SHAO: The 50 foot-pound criteria is being changed anyway. We know it's considered in the regulation. If you have some new data, you can always submit to NRC a topical report. We can look at it.

MR. SMEDLEY: What are you going to change it to? That's what we want to know.

MR. RICHARDSON: Well, you're really getting into an area that I want to avoid in this workshop, and that is operational problems.

MR. SHAO: It is an operational problem.

MR. RICHARDSON: There is certainly a sharp distinction. We want to confine the rulemaking, as was stated this morning, to those phenomena that are peculiar to the aging phenomena beyond 40 years. I don't mean to be avoiding your questions.

MR. SMEDLEY: But I also think that you should consider before 40 years. If we go over 20, then sure as hell we're going to be beyond.

MR. RICHARDSON: Well, I guess I would like to turn the question around, if I may. Do you think that the NRC should tighten its rules and go for a more stringent analysis? Does that make sense from your perspective?

MR. SMEDLEY: I don't think you have to tighten the rules. I think, as was indicated, you may have to make the rules more realistic.

MR. SHAO: We realize that 50 foot-pounds is from many years ago. Also, when you do a Reg Guide 1.99 revision 2, and you cannot do the steam criteria, you do it more realistically, fracture mechanics, perhaps, and you always have a way out. It doesn't mean you're stuck right there.

Fifty foot-pounds, we would entertain anything that's more realistic, and for pressure and thermal shock, we can always develop more realistic standards.

MR. MARSH: My name is Tad Marsh. I'm from the Staff, the mechanical engineering branch.

With respect to low-temperature compression protection, there's many things the licensees can do to address pressure maximum, pressure limits, as the fluence increases on the vessel, just one of which is the adjustment of the Appendix G curve. There are other operational means they can use to reduce pressure.

The CE topical report -- if there is one -- or at least the CE methodology, and some of the others that we've heard of, use not just adjustments in the Appendix G curve, but also operational things that can be done with respect to the PORB set point, with respect to using the RHR systems, so that you indeed have a bigger operational window in which to operate. Should it be life extension, or if you have problems even before life extension.

MR. RICHARDSON: Do we have other comments on this particular question that we have of you?

MR. PAVINICH: I'm Wayne Pavinich, from TENERA.

I think the question is talking about the surveillance program, not what to do once you have an embrittlement problem, so I think people are missing the boat a little bit. ASTM, E-10, who had jurisdiction on E-1-85, which tells you how to run a surveillance program, which I believe is referenced in Appendix H, is already taking a look at that, a draft standard, or an outline of a draft standard for review.

MR. RICHARDSON: Okay. Thank you.

Other comments on this question? Can you offer assistance in us addressing this question?

MR. LANDERMAN: Ed Landerman, consultant.

I'm not sure whether Appendix H really addresses annealing. It does certainly cover time dependency, but this would be a new time basis on any of it. The comment is whether it's covered under Appendix G, but it's maybe something that should be considered.

MR. SHAO: This is a valid point. Right now, I think we see annealing as a viable method to restore the original toughness, and I think we need to have criteria on annealing, so right now we are trying to develop some criteria. But we don't have a criteria now.

MR. HEDGECOCK: Pete Hedgecock, from NUTECH.

Building a little bit on what Wayne Pavinich said, there are a number of us here from ASTM committee E-10. With respect to the annealing, we did revise the standard E-509 several years ago, and there is some further revision. We need some guidance in that.

We also have a reconstitution standard, for reconstituting specimens, because one of the problems that you face as you go on in life is that you're running out of sample material which is valid, upon which to refine many of these analyses that you were speaking of earlier. So ASTM is undertaking some of that work already, which I hope will be of help to you.

MR. RICHARDSON: Question: What is the time frame for completion of those standards?

MR. HEDGECOCK: E-509 is already on the books and is due for revision, I think, a year after next. I think it's E-12 or -13 that's the reconstituted specimen standard. That was issued about a year ago. We need some feedback and practical experience on that, which will probably lead to revision before the normal five-year period.

MR. RICHARDSON: I guess another question I have related to that: Is there a relationship being developed with some of the foreign countries that are having the same type of experience? I'm thinking particularly of the Soviet Union.

MR. HEDGECOCK: Yes. As a matter of fact, Neil Randall, of your staff, who's a member of that subcommittee, mentioned this at our last meeting. We were hoping to get an update on some of the Russian experience. You mentioned earlier some of the factors influencing the degree of embrittlement, and among them were copper and nickel, as the constituents of the seats and weldments. The question is about some of the European steels, where they have high phosphorus and sulfur levels. We have asked the NRC representative to go back to the work that you're doing at Oak Ridge and see if you can come up with some correlations on these elements. We have some correlations on those, and they were submerged into copper and nickel being the most significant elements.

There's some question that the European data now have to be looked at with respect to the --

MR. SHAO: The Russian reactor has a lot of phosphorus in it.

MR. HEDGECOCK: They have high phosphorus, in European steels. We don't.

MR. RICHARDSON: Thank you.

MR. KATZ: Larry Katz, Westinghouse.

On the subject of annealing -- I'm the chairman of the special working group, plant life extension, ASME. We have moved ahead on an initiative on an annealing procedure which is on the docket for the subgroup for repairing and replacements, based on E-509. That is actively under way in that group, where they're trying to write a draft procedure based on that.

MR. SHAO: Is there an NRC member in this group?

MR. KATZ: Yes, there are.

MR. CORWIN: Bill Corwin, Oak Ridge, again.

A comment that ties a lot of this together: Yes, there are standards that ASTM either has or is developing, and certainly ASME as well, on how to anneal the vessels, and how one can follow the material properties of the vessels through an anneal. It all hinges on the availability of adequate numbers of surveillance specimens, which have seen the operating history.

Even with reconstitution, it's very likely that there will be an inadequate base for a number of the older plants, even if they were willing to reirradiate, starting from scratch. A lot of times there's not the archival material available with which to perform the proper irradiation.

One of the concerns that's undoubtedly going to fall back on NRC is, what type of substitutions can be made, or if data are not available, how does one make the judgement calls where it's just simply impossible to get an adequate amount of surveillance data. That's going to be a key problem as plants either are annealed or simply go through extended lifetimes.

MR. RICHARDSON: Any comments or last questions?

MR. SHAO: Do you want to take the next question.

MR. RICHARDSON: Well, I guess if I were to summarize the question, one; I heard that the national standards ASTM is undertaking this problem and looking at it. However, as Bill points out, we're still going to come up short on surveillance samples, and how does one deal with that? It's a nagging problem that we're going to have to face.

Annealing is certainly one approach that's going to have to be considered. The NRC and the industry are going to have to struggle with an annealing criteria. What is an acceptable annealing criteria?

In fact, as you anneal, how much recovery do you get and how do you measure that? Then what do you do post-annealing with respect to samples? Do you insert new sample? Do you have -- is there archival material available to start a new sample program that starts the annealed condition.

These are problems, of course, that we struggle with, but the feedback I'm getting from you is a focus on improved standards and look at annealing. Those are the two main points that I got out of it.

MR. SHAO: The next question is related to cast stainless steel. You know, maybe I didn't use the right phrase, but the cast stainless steel, after operating for a long time, has a tendency to lose its toughness. That's one issue.

The other issue that's not listed here is; it's very difficult to inspect cast stainless steel, so really there are two issues. It may lose the original margin, and it's very difficult to inspect, so how should we tackle this issue during life extension or license renewal. Can somebody say something?

MR. HEDGECOCK: Pet Hedgecock from Newtech. On the issue of cast stainless steels and their embrittlement with aging, it is very much a compositional matter and determined by the chemical composition, the length of time and the temperature at which it's exposed, but there are ways of culling out a large percentage of the subject components, given that you have some history on these materials.

You can show, by fracture mechanics, what kind of flaw they would tolerate which might turn out to be a through-wall thickness, leak-before-break issue as opposed to a fracture problem. I think you can put the problem in perspective in an individual component through some such analysis.

Then, of course, if you are still faced with your second problem which is the inspection itself, and there is some work going on, I believe. You've probably read all the reports that I have, and there is some hope for improved inspection techniques, plus ultrasonics isn't the only way one needs to inspect cast stainless. One has radiography which is not a small undertaking, but it's being done on comp bodies, in particular.

I think you can handle the problem by history of the material, taking the research data that's been done in this country and in Europe and Japan on aging and embrittlement of cast stainless steels. Then looking for the susceptibility of the particular component and the amount of embrittlement that you showed by fracture mechanics could be of concern.

If you've done that, of course, you can relate back to what kind of flaw size you should be looking for, and hopefully your research work on NDE technique development will allow you to decide what you can inspect for realistically.

It doesn't offer a solution, but at least there's a methodology there, I think, that could be followed.

MR. SHAO: What if some of the valves are made of cast stainless steel?

MR. HEDGECOCK: Again, if you look at the valves, I suspect you'll find that a lot of the bodies are subjected to much lower stresses than the piping and the elbows, for example, therefore, their flaw tolerance is probably much greater in term of the fracture mechanics analysis. The only concern one would have on pump bodies is the extensive amount of post-casting weld repair that may have been done, leaving fairly high residual stresses.

Again, this should be looked at on a fractured cast basis.

MR. LANDRUM: Ed Landrum. I still want to make the relationship between reactor vessel surveillance and changes in properties of the castings. Certainly, there are a number of attempts and work on getting that kind of data, but I think it's minimal compared to the number of plants out there.

I could say, originally, on reactor vessels, the intent was to use a correlation monitor material and not an induced surveillance and get a lot of data. I believe we don't have sufficient data of aged stainless steel castings at the temperatures of operation.

Certainly, the analyses can be done, but I believe it's with limited information. When you asked what uncertainties and what measures, and I'm saying there's a need for additional data on the effect of aging. We're talking about a time dependency. We're talking about another 20 years. We're talking about a lot of parts being taken out of service.

It seems to me that that's the kind of measures that the NRC ought to be able to implement, to get that data and have a feedback cycle, just like you do with the surveillance -- the reactor vessel surveillance stuff.

MR. SHAO: Any other comment?

VOICE: I read your question slightly differently. I said, I read your question slightly differently. I think the question you're asking is; is a change in fracture is sufficient to change your inspection, your current inspection requirements? I think that there are bounding analyses would tell you that the answer is no.

A loss in toughness, the worst case that we would expect, wouldn't change your present Section 3 inspection reports. Therefore, you would probably need to do very little in the way of monitoring the -- fracture toughness. What you really need to do is to determine that the boundary of toughness which we now use, which is like a fracture of 2 deteriorations, is correct.

Parenthetically, I'm not sure I understand why this is a life extension issue, as opposed to a curve point.

MR. SHAO: I don't read the question the way you read it, but this is one potential solutions. Any other questions or any other comments?

[No response.]

MR. SHAO: Okay, let me summarize this question here. Regarding cast stainless steel, the general feeling is that there is not sufficient data. Maybe some work needs to be done in this area. The problem is not only on the amount of toughness and maybe the difficulty inspection, however, we can look at some of the components out of service, see how they behave and we can do some fracture mechanics, do some kind of bonding analysis. Hopefully we don't have to change the inspection intervals.

MR. RICHARDSON: The third question has to do with the in-service inspection and in the in-service testing programs. As you heard this morning, at least on a preliminary look, it is our intention to exclude the in-service inspection and the in-service testing programs from license renewals since this is an ongoing program that hopefully will detect the degrading mechanisms and take corrective action as the plant progresses through its life, not only for life extension but as the plant exists today.

The question that we're asking here is, in your opinion, do the current in-service inspection and in-service testing programs adequately pick up and detect the aging mechanisms that may be critical to safe operation of those components and materials? Are the ISI programs and the IST programs that we have on the books or coming on the books, are they sufficient to pick up the aging degradation such that they do not have to be modified or addressed in licensing renewal.

MR. KATZ: I'm Les Katz of Westinghouse. There are three groups looking at that question in the industry at the moment. ASME, special working group, collects for the pressure boundary, O&M for IST and IEEE 3.4 for electrical and instrumentation. The ASME pressure boundary PLEX group has been in business for about three years and we so far have not recommended any specific changes in the code to enhance inspections.

However, we do have a lot of things on the agenda where people are still searching for those and one of the things that didn't occur, there has been, I think, as a result of the findings of the pilot studies, a new ASME group looking into core structures considerably. There is a subgroup on core structures currently underway as a result of recommendations on that issue.

There are some additional inspections being looked at that came directly out of the pilot studies and one of those is maybe more enhanced inspection of surge line for the thermal inspecting and so on, inspection of that. The modern initiatives that have moved forward were in the repair area. There has already passed a repair procedure for steam generator 2 plugging -- I'm sorry -- sleeving -- which is new, and as I mentioned earlier, there is a reactor vessel annealing procedures underway and we're continuing to look for issues and either writing them off or making recommendations so as far as I'm concerned, that's a question that still is unanswered, totally.

MR. SHAO: What about the ISI of reactor vessel internals? Do they require additional inspections for their license? Core reactor vessel internal as a whole.

MR. KATZ: I don't think that you're a direct problem is associated with life extension. I don't see them that at all. I think and the ASME believes that the treatment of core structures has not been adequate in Section 11, totally adequate, and it's been emphasized more by the work that came out of the pilot studies -- not safety issues but probably economic issues in regard to understanding when these things may need repair and so on. I don't think it's strictly a PLEX issue at all on core structures.

MR. RICHARDSON: Bill?

MR. CORMAN: Bill Corman. I think there are a couple of aspects one needs to look at in this. A lot of the current inspection techniques look at geometric changes, looking at flaws, cracks, thinning, whatever. There's also the focus that could be applied if it is deemed necessary and necessary in this case I think is an economic justification or how close are you coming to the margins and that is whether or not you can look at changes in properties nondestructively or very closely.

I understand that there was a meeting in conjunction with the recent older reactor safety meeting where a number of experts from around the country got together and looked at innovative ways of nondestructively examining properties in situ and that may be something that ends up being required. Again, it's a question of how much damage can you tolerate. You need to only look at the geometric changes or do you also need to try and assess the property changes in some of these materials like the aged stainless.

MR. RICHARDSON: Thank you.

Other comments, responses to those questions?

Well, if I were to summarize what I heard from the two gentleman, one would be that a number of bodies have been formed and are looking at these very questions. The ASME, PLEX group, IEEE, O&M, are looking at these very issues and the jury is still out as to whether the ISI, IST programs need to be changed. I would read that not so much in terms of life extension as changes needed in the program as they exist today for operating reactors.

So it may not be as much a life extension issue as the need for improvement in ISI, IST in general. That's at least the flavor that I got from what your response was. The second aspect is, it may be helpful to look at some innovations that may allow us to look at changes in material properties as opposed to the traditional way of just looking at geometric changes in the materials and their behavior in a non-destructive way that may give us insights regarding the aging degradation mechanism associated with materials and this may be worthwhile looking into and again, the jury is out but it may have promise on the horizon.

Okay. Thank you.

MR. SHAO: The next issue is weld overlay. For those people who are not familiar with this subject, it is a lot of IGSCC piping for repairing these cracks. Instead of replacing the piping, they put a weld overlay. We believe this, although it's safe for current operation, whether this weld overlay is safe for another 20 years, mainly because the examination is very difficult and when you put on weld overlay, the stress pattern is changed. For some people, they'll even worry about seismic analysis because some plants have so many weld overlay, the stiffness of the piping is changed.

So any comment on this?

MR. HEDGECOCK: This is Pete Hedgecock. One comment came from the previous question. The answer there was that the jury is still out and there are improved techniques being developed. One would hope that they would be applied to weld overlays as they develop. That doesn't answer the second part about the stress patterns, I realize.

MR. SHAO: Aside the stress pattern, there are two types of stress pattern. One is localized stress ride underneath the weld overlay. The other stress pattern is general stresses under seismic loading or dynamic loading, whether the stiffness of the piping is changed or not.

MR. PAVINICH: Wayne Pavinich of TENERA. I believe that when you do a weld overlay, you have to update your piping analysis. That's a requirement right now. I think it's done as a matter of routine; am I correct?

MR. SHAO: Have you done that?

MR. PAVINICH: Does anyone disagree with that? Is that true? I thought you had to know what the stress patterns are and I'm not sure you know what the stress patterns are if you do weld overlay.

MR. SHAO: But usually, we feel -- when you do one or two overlays, it's very localized and you do a stress analysis in this local area and --

MR. O'DONALD: Bill O'Donald. Just one comment about the weld overlays is that something that should be a cautionary thing. When you do a weld overlay because of shrinkage, you can create residual stresses at other welds that have previously been stress improved and sometimes those welds will then crack because the residual stresses are in tension in those welds and it's a good idea to take a look at the system and stretch your leads to the other welds again that are put under tension when you put in a weld overlay.

MR. SHAO: Okay. Dr. McDonald is also the inventor of the mechanical device which also -- stress in crack piping. That -- is also true of NRC to take care of cracking the pipe.

MR. DEARDUFF: Art Dearduff, Structural Integrity Associates. I believe that NUREG 0313 today requires when a weld overlay is applied that you do go back and look at the effect of the shrinkage stresses which result on the rest of the piping system and the other weld overlays that are on the system. So it is being addressed today from the standpoint of the piping integrity.

MR. SHAO: The shrinking stress is one thing. The stiffness is another thing.

Also, you change the stiffness of the piping.

What do you people thing, the weld overlays should continue for another 20 years or -- get rid of it and replace piping?

[Laughter.]

MR. LANDERMAN: You qualify the word by ending is difficult and I'm sure there are a number of people out there who will keep trying to make that less difficult.

MR. SHAO: -- do a better piping analysis. Okay, let me summarize this.

We can include the weld overlay design by doing two things, by improving NDE technology and do a better piping analysis to take care of the residual stresses, weld shrinking stresses and stiffness effect.

MR. RICHARDSON: The fifth question we had was a multi-question, on a number of issues here. And I will summarize them as a group and then ask that you comment on any or all of them.

The first one being that it is the recognition that NDE technology over the years has changed, and we have gotten better at it.

The question is for plants that for years have started out with an NDE technology that by today's standards was not very sensitive, should we require a rebaselining, using the better techniques that are available today? That is the first question of Question 5.

The second being, should the in-service inspection intervals and the extent of the sampling remain the same, should they be increased or should they be decreased as a licensing renewal issue?

Third, considering that with aging, in particular with materials such as cast stainless steel, there is a loss of toughness. Therefore, should our flaw acceptance standards, our acceptance criteria if you will, be modified to recognize and account for the loss of toughness?

The next question, again, recognizing the uncertainties in the level of degradation and in the effectiveness of our NDE techniques, should we impose as a condition of license renewal a requirement of continuous or near continuous surveillance or NDE techniques during the extended life?

Now, all of those questions have some relationship to each other. There is also some disparity in those questions. I invite your comments on any or all of those parts of Question 5.

MR. KATZ: Les Katz, Westinghouse.

We very recently addressed in the special working group the issue of rebaselining, one of the issues that came up, because many of the reports that we read on plant life extension, some of the early work, and we had recommended such a rebaselining. We put together a task force that looked at that, and they came back with a resounding recommendation that we do not recommend any kind of a rebaseline inspection, and I think for good reason.

First of all, with regard to less efficient techniques, the NRC in 10 CFR 50.55(a) does require that the program be rated every ten years, and therefore it would be hoped that new techniques would be picked up as a result of that. We also see absolutely no reason for going back to 100 percent inspection, which would be very high radiation effect, on personnel, would be very bad on availability, and in our view would have gained very little new information with regard to the flaw in the root.

So we made a resounding recommendation that we hope sticks, that says that no rebaseline should be required. We amended rules of Section 11 so that what you do in the first 40 continues on exactly the way it has been.

With regard to ISI intervals, we have also looked at that and although we didn't look at them in any real scientific approach, we feel as if the intervals and the sampling is about right, has been successful in covering things, and should also continue.

I won't address the toughness with aging, because we haven't look at it.

One other point I wanted to make was that with regard to monitoring, it just was a new change in the code which will be published I think in the next edition which talks about the use of acoustic emission, and does allow that, in lieu of the enhanced inspections the code now requires when a flaw is accepted by analysis. So we have already made some inroads. And it is not specifically related to life extension. It is a now thing which I think is very good.

MR. SHAO: If I understand you right, you recommend acoustics emission?

MR. KATZ: It is in the final review process in ASME, and it is allowed now. It is an option to more, to an enhanced inspection program which is also required if you have to accept by analysis.

MR. RICHARDSON: But that is really not a licensing rule question. It is a general improvement in the rule.

MR. KATZ: It is a milestone in that it is the first monitoring technique that I know of in Section --

MR. SHAO: Also, you recommend, suppose you extend life for 20 years, two more inspection, every ten years?

MR. KATZ: We have, yes, the group, the task people worked on that, recommended that it stay just the way it is.

No increase, no decrease, but just about the way it is.

MR. LANDERMAN: Ed Landerman. Yes. My comment falls between 3 and 5. I've commented on the dissimilar metal wells, and somewhere it doesn't fall out of all these questions. But it does fall out in the sense of, I don't know whether you want to call it rebaseline, but certainly a close look at an evaluation of that inspection, because it is probably the most difficult. We are talking about inconel or stainless inspection. And so whether you want to call it a new baseline with new techniques or an assessment of where we are, I think there is a need to look at that.

MR. SHAO: You mean --

MR. LANDERMAN: Well, a dissimilar metal well, inconel or stainless.

MR. RICHARDSON: Are you getting at that as a specific requirement for license renewal?

MR. LANDERMAN: It is currently in Section 11. You raised the question here about less efficient, do we need an adequately, you know, are current inspections adequate. I am just raising that question, to assess that.

MR. RICHARDSON: But again, that may be a question that is related to operational plans, not necessarily a license renewal issue in itself.

MR. LANDERMAN: Whether it is an additional issue as effective aging, I don't know.

MR. RICHARDSON: I see. Okay. Thank you.

Other comments, responses to this three-part question?

[No response.]

MR. RICHARDSON: Well, to summarize this, the feedback that I get is, at least one arm of industry that is represented within the code activity is saying there is no need to rebaseline in general, that the ten-year inspection interval where programs must be updated to encompass new technology as reflected by the code is sufficient. There is no need for going back to a 100 percent inspection. You pay a dear price for that in terms of exposure and you get little benefit from it, and that the ISI intervals, the ten-year intervals and other intervals that are employed, are about right. They seem to be working. So far, there is no reason to believe that those intervals shouldn't be about right for beyond 40 years.

And a final note, that it would be good to take a look at these bimetallic welds, that although I don't see that as a necessary, necessarily as a license renewal issue, but certainly something that needs to be paid attention to in the process. Okay?

MR. SHAO: This morning somebody raised some question on fatigue requirements. The next question is how should we treat fatigue in license renewal?

As you probably know, the ASME curve, the ASME fatigue curve is mainly based on virgin materials. During the operating life the material has seen a lot of harsh water environment and high temperatures but there is one saving grace -- in the ASME curve they use a factor of 2 on stress and 20 on cycles. For that margin we feel maybe it's good for 40 years, but what about for 60 years? Should we use a different curve?

Another question is what happens if the accumulative damage factor reach one, then what do you do about it?

So I invite some comments how to tackle this question.

Dr. O'Donnell?

MR. O'DONNELL: I have some viewgraphs, Larry. May I show them?

MR. SHAO: [Nods affirmatively.]

MR. O'DONNELL: I'm Bill O'Donnell. I'm Chairman of the ASME Sub-group on Fatigue so I take the blame for a lot of the fatigue methods that are in the code, and the NRC came to our committee and pointed out that our curves are based on data in error and their research over the past 10 years including about 10 NUREGs shows that if you tested reactor water you get a lot less fatigue life because of the accelerated crack growth that you get due to environmental effects.

Of course there is the International Crack Growth Committee has done a lot of work on that same subject.

The NRC has also pointed out that they feel that the fatigue life, the fatigue properties are lower as elevated temperatures and that we didn't take that into account adequately in our design curves and they have raised a number of issues, such as the load sequence effects -- perhaps we're not calculating the usage factors correctly and it also is true that our curves are like 30 years old and they need to be updated.

As a result of that and also some criticisms that the code doesn't do anything for weldments that is very worthwhile and that the practice in the industry is that some people are using fatigue strength reduction factors for weldments, for metallurgical nudge effects and other people are not and they feel that the code should make that practice more uniform.

There are a lot of issues that the NRC has raised about the adequacy of the design curves and we are working right now to upgrade them.

Another one is that for the early materials we only go to 10 to the sixth cycles, whereas vibrations take you to 10 to the tenth cycles and besides cycle thermal mixing fatigue takes you to 10 to the tenth cycle so the curves need to be extended for burning materials for their cycles.

As a result the Subgroup on Strength of Weldments and the Subcommittee on Design and others worked up the program which I approved the fatigue design curves that are in the code and the Pressure Vessel Research Committee is the research arm of the ASME code and they are trying to get this program to upgrade these curves underway and it's like a three year program that KPRC has just started.

I have a couple of viewgraphs just to show you the kinds of things.

[Slide.]

MR. O'DONNELL: This is a graph on a NUREG NRC did showing that sure enough at higher temperatures the fatigue properties are actually lower, whereas there had been the assumption that in the low cycle regime when the material was more ductile at higher temperatures the curve should be higher. In reality the data shows that for ferritic materials and possibly for austenitics as well the curve is actually lower in the low cycle regime than higher temperatures, contrary to common sense. It has to do with thermal strain agents.

MR. SHAO: The question is -- I know the curve is low -- the question is the factor of 2 on stress and 20 on cycles.

MR. O'DONNELL: Yes, that's a good question and let me address that with this next curve.

[Slide.]

MR. O'DONNELL: This curve shows that if you take pressure vessel steel A-Clad 33, and you assume that the cladding protects it against early crack initiations -- after water -- but that once you get an underclad crack that you lose the integrity of the cladding, you can see where the assumptions are very important in all of this. This is why we need national consensus standards involved in making these assumptions.

If you make the assumption that the reactor water accelerates the crack growth once the crack is initiated, you go from this type of failure code here to this lower failure curve. This is the mean failure code for those assumptions.

If you lock at the difference in life it's like a factor of 5 or so.

Now the code put in a factor of 2 for environmental effects so if we were to construct a curve, just a design curve based on this, we would reduce the 20 to a 10. That would give you a factor of 2 but unfortunately this difference is like a factor of 5 so that if you make those assumptions and you put environmental effects into the curve, you would have to reduce the curve quite a bit, so this is why we are trying to put together this PVRC ASME consensus standards approach to make sure the assumptions and the factors of course include the scattering of the data, the size effects and the surface finish effects in addition to the environmental effects.

To get this lower curve we took the crack growth rates from section 11 for reactor water. Section 11 is working on improving their -- and this is an illustration of what section 11 is looking at right now for environmental effects on crack growth rates. They are factoring in the cyclic rate effects into this whole thing.

The reason I wanted to take a few minutes to present those curves is to make sure that you know that the ASME code and the PVRC are trying to update these fatigue curves and they are trying to put environmental effects into them and they are trying to account for aging effects and to run the curves out to high enough cycles so that we can do vibrations and high cycle thermal fatigue.

MR. SHAO: Any other comments?

MR. KATZ: One other addition to the ASME worth mentioning, in section 11 there is a task group on fatigue who are in the process of writing a white paper on fatigue and they are looking at, you know, what to do about the fact that you may run on usage factor -- and one of the recommendations to come out of that group is that -- it's not firm yet but one of the preliminary recommendations is that indeed if one views actual transients rather than design phases they probably become free and I think the background incorporated somehow into these options and code, that would be a big help.

MR. SHAO: Anything else?

MR. CINADR: Does the NRC consider the usage factor of recalculations? In other words, that indicates something as you approach one, how many times can you --

MR. SHAO: If you want to sharpen your pencil --

MR. CINADR: Does that tell you something?

MR. SHAO: If this gentleman tells you that you need I training and you get into trouble and you want to use X training and you have some data to back it --

MR. CINADR: If everyone records that -- this is no problem.

MR. RICHARDSON: I think from a personal point of view
-- if I discover somebody out there sharpening the pencil to push because they are pushing the cumulative usage factor toward one I am going to start getting concerned just because of the great uncertainties associated with the loads that go into the fatigue calculations and the fatigue properties themselves.

I start getting worried when we need to sharpen our pencils. That would certainly be of concern to me.

MR. SHAO: When you do a design calculation at that time the specification gives you a 10 pound cycle -- if you're realize maybe only --

MR. RICHARDSON: I understand that. It would just cause me to take a closer look and challenge it at least so that I really in my heart of hearts agree with what they are doing.

MR. CINADR: And there will be more of that as time goes on.

MR. RICHARDSON: Absolutely. I think our job is going to get tougher and tougher.

MR. SHAO: Okay, let me summarize this.

MR. DEARDUFF: Art Dearduff, Structural Integrity Associates.

I guess I take issue a little bit with your comment about looking at anyone who wants to sharpen the pencil because you have got to realize that in the original design of these plants the criteria was that the usage factor be less than one and most of these components were designed by the low bidder and so when you get this stress analysis and you tilt the founding transients and he could take all of the predicted transients from the plant, divide that by the allowable cycles for his one analysis and he got usage factor of less than one, his job was done. He didn't get paid any more to do additional analysis to show that that usage factor was really zero in many cases, so with just a little bit more going back and looking at the original design analysis you can show that it is not a concern even though the stress report reported a number which many times has made its way into technical specifications --

MR. RICHARDSON: Well, I didn't mean to imply that I or the NRC would in fact reject a design that had a usage factor near one, not at all. It would cause me to look a little harder -- when one starts to approach a usage factor of one it just causes me to want to look the calculations over a little closer than I would with a calculation that came in with some very conservative back of the envelope calculations that come out with a cumulative usage factor of .1. I turn the game up a little bit.

MR. SHAO: I think your point is well taken but a lot of things now is done based on boundary analysis -- very conservative and your usage factor, less than one, that's it, so that in the license renewal you have to sharpen your pencil if you want to get out of it.

MR. DEARDUFF: Another observation is that industry continues to address fatigue through inspections. As you are well aware, there's been a number of actual fatigue failures which have been observed or cracking has been observed in plants.

MR. SHAO: -- well documented, backed by data analysis or you do it with good data and actually we'll look at it.

MR. DEARDUFF: I just wanted to say though that we have to look at the fact we can't just look at the fatigue usage curve, the usage curve here. There should be a balanced program of analysis with existing curves combined with inspections --

MR. RICHARDSON: Of course, of course, I understand.

MR. DEARDUFF: -- and you shouldn't just ask them to use a new peak usage curve.

MR. SHAO: Any other questions?

[No response.]

MR. SHAO: Okay. Let me summarize this.

On the fatigue requirements, there was a concern with each of the fatigue curves because the fatigue curve was based, actually the fatigue curve was based on virgin material, not this material being faced with different kind of involvement in high temperatures but now ASME PVRC has a three year program to try to adjust these issues.

Also, there was some questioning related to whether the original analysis shows that the cumulative damage factor reaches one then they want to look at the cycle, look at the analysis, look at the assumptions when they can sharpen the pencil and do a better analysis to show the cumulative damage factor of less than one.

MR. RICHARDSON: The last of the questions that we thought about before coming to this workshop is sort of a change of pace. We've been talking about non-destructive examination and analysis. Sort of the last question is, are there any tests that we ought to require to demonstrate operability in terms of license renewal?

Now, of course, there are tests under the IST program and others that require some testing. What we're asking is, is there a need for additional tests to demonstrate operability, vis-a-vis licensing renewal?

MR. SHAO: Well, I think the integrity is a form of operability. I would say integrity/operability.

MR. RICHARDSON: I guess a pipe becomes inoperable, it collapses. Any thoughts?

MR. HEDGECOCK: Are you including tests to determine material condition to actually measure some form of aging in the material?

MR. RICHARDSON: Why not?

MR. HEDGECOCK: Okay. If you include those, there is some work going on in ASTM, some work at Oak Ridge National Lab on an indentation method which is not non-destructive, but is very mildly invasive, and from it, one hopes that we'll be able to produce actual mechanical properties. There's been some correlation on irradiated materials and papers published already. This might be applicable to other materials where the aging phenomena are not just related to the radiation.

MR. RICHARDSON: Is this analogous to a hardness test?

MR. HEDGECOCK: It's very much like a hardness test. You just get a lot more information out of the same process and derive correlations --

MR. RICHARDSON: More than just field stress?

MR. HEDGECOCK: Yes.

MR. RICHARDSON: Okay. Yes, over here.

MR. McCUMBER: This is Joe McCumber with Yankee

Atomic. I think, in general, as to one of the questions, I think, that came up earlier as far as rebaseline, I think any type of test we feel should be done and technically required based on the degradation you're concerned with cannot be tied, necessarily, to the extended life, or licensing, or anything like that.

MR. RICHARDSON: Yes. I think we fully recognize that there are, from time to time, the necessity to perform the test to demonstrate operability. A recent generic letter that we have on the streets really addresses that in spades, and that's not what we're talking about. It is those tests above and beyond the routine tests that would be associated with license renewal, to either rebaseline, whatever, to assure ourselves that this is operable for the next increment of life. Yes, your comment is very valid.

Well, I guess I would summarize it by the one comment that we got, was that the ASTM is working on some quasi-non-destructive techniques involving indentation to determine mechanical properties, and that this may prove to be a useful tool in licensing renewal applications.

Well, we've come to the end of the seven questions that occurred to us. This list is far from exhaustive, as I said earlier.

We're at break time, or very near break time, and what I'm going to suggest is that we take the break, but before we take the break, can I get a feel for -- are there those among you that would like to raise additional issues in this general area of primary system integrity in terms of life extension, keeping in mind that the purpose of the workshop is to hear from you where you think we are going too far, or where we're not going far enough. Are we straying off of the path of the real central issue of assuring safety for license renewal in light of the philosophy that you heard this morning, and that is the assumption that the licensing basis, the current licensing basis is sufficient for extended life, and we need only concentrate on those issues that deal with aging degradation?

So I guess I would like at least a show of hands of those that would like to extend the discussion, or have something to say beyond what you've said in response to these few questions that we've put forth.

(Show of hands.)

MR. RICHARDSON: Okay. So it seems like we may have a half a dozen or so.

MR. CHARDOS: Jim Chardos of GPU. It would seem to me to make sense, after the break, to maybe go through the speakers that have signed up to give some kind of a presentation, and then based on what you hear there, you might want to expand this discussion about the current licensing basis, age-related degradation. I know part of what I'm going to talk about, and maybe some others, will touch on that subject of licensing basis, age-related degradation, the significance thereof, and so on.

MR. RICHARDSON: Okay. So let's take a break, and we'll be back at three o'clock.

MR. SHAO: There are a couple of persons requesting to comment. The first one is James Chardos, representing the NUMARC/NUPLEX Working Group.

MR. CHARDOS: Good afternoon. On behalf of the NUMARC/NUPLEX Working Group, I'd like to make a presentation on pressure boundary and then detail, to some extent, the reactor vessel for PWRs.

I'd like to, before I get started, mention that NUMARC Working Group and NUMARC itself will develop and has developed answers for those seven questions, so we will put something in writing in the next week or two, within the period allotted us.

[Slide.]

MR. CHARDOS: We had talked earlier today about the reg guides and SERs and Frank Gillespie mentioned words about industry reports from the -- reports from the industry being reviewed as topical reports or possibly part of the reg guide.

From our point of view, it's not mandatory that a reg guide be developed for the industry reports that we contemplate at this time. As it mentions, topical reports would be sufficient given some form of an SER endorsing those reports.

A guide will be acceptable with some caveats here. In terms of getting something that's together, a benefit from analysis or something that's developed by consensus within the Working Group, we've got those reports put together that can be used for that reason. If the conceptual rule -- and we talked a little earlier about this in terms of the rule versus what's in the register, in terms of being together on two points, one of which is the credit for ongoing programs that the NRC has in terms of inspection and enforcement.

It only requires review for those efforts that have significant age-related degradation. So those are kind of like two caveats for usage, as well as we talked before and others have about reg guides and not holding up the two lead plants, given their submittals are in the June and December timeframe of 1991.

So that kind of gets back to the schedule we saw earlier today in terms of reg guides, and then some more reg guides. There were two submittal dates of reg guides. And we, as an industry, would back that given it doesn't, as a last concern here, effect the lead plant submittals and eventually their review and hopefully receiving a renewed license.

[Slide.]

MR. CHARDOS: In terms of specific reports which we believe address the pressure boundary, these are the four that we have that just reflect the pressure boundary. As you can see, there's the PWR reactor vessel; the PWR reactor vessel and then the BWR and the PWR pressure boundary reports.

These are four of ten reports that NUMARC/NUPLEX plans on eventually submitting between now and the summer of next year, with the express purpose of using these as benchmarks for the industry and for the NRC. So these four are four that reflect on the reactor coolant pressure boundary, two of which deal with the reactor vessels.

As you see, the BWR reactor vessel has been submitted to the NRC in October of this year, and the other ones, these three have schedules which we've already previously submitted to the NRC which are all between now and the June/July timeframe of next year.

[Slide.]

MR. CHARDOS: Next, I'll kind of go through quickly -- I have an outline of industry report and get back to the licensing basis and those kinds of issues, as we talked before.

The process for an industry report and kind of the format of the two we've already put in, the BWR reactor vessel and the PWR containment, follow the following format. In terms of a total group for the containment or the reactor vessel, you determine which components are, in fact, safety related and safety significant.

With that grouping of components or subcomponents, you now describe all the plausible degradation mechanisms that apply to this particular component or subcomponent. It's worth noting here that in the proposed rule, a list -- a number of degradation mechanisms, and with or without inclusion in the rule and definitions thereof, for each of these reports, you need to look at the mechanisms that, in fact, effect the component or subcomponent. Some of them will not and do not.

The next bullet kind of follows to that in terms of determining which ones are potentially significant to those that, in fact, are applicable to the component. And then the last two talk about those that are potentially significant, determine if the existing programs for inspection and for testing and analysis as currently implemented are adequate to bound the age-related degradation within acceptable limits.

So that tells you for particular degradation mechanisms if the existing programs are adequate and bound to degradation within limits, then you don't have a problem. For those, which is the last bullet, where you may have significant degradation beyond established limits, then you either put together a program for aging management.

So that's kind of a -- if you look at a format of the report, the last section of the report picks up the last bullet and tells people, utilities, which ones they think are significant, that may be outside established limits and may require each particular utility to put together a program to address those degradation mechanisms.

[Slide.]

MR. CHARDOS: For the four reports I talked about earlier, we've tried to, as shown here, display the appropriate mechanisms or degradations that apply to each of those four. If you look at the BWR reactor vessel, we've listed the six that we believe are, in fact, at play here and need to be addressed.

For the PWR reactor vessel, those are the ten or so that apply. Then down below, for both the PWR and the BWR pressure boundary, there's those five. Once again, these groups of five, ten or six all come, to some extent, from that list in the proposed rule of degradation mechanisms that, in fact, need to be addressed when you go through and review a component or a system. So those are the mechanisms that will be addressed for these reports.

MR. SHAO: May I ask a question?

MR. CHARDOS: Certainly.

MR. SHAO: Why is creep a problem? Is it the high temperature creep?

VOICE: He didn't say it was a problem.

MR. CHARDOS: No. I said that's something we need to address or review and if, in fact, it's bounded, then it would be not a problem. I just mentioned that. Therefore, for creep, the answer would be temperatures not high enough;

therefore, not a problem. But you need to at least review it to make sure that's the case and not just dismiss it offhand. You ought to be safe by addressing it as not addressing it.

MR. SHAO: Am I right, to have creep, you have to have over 150 degrees --

MR. CHARDOS: Correct, but --

VOICE: Could you repeat that question?

MR. SHAO: The question is he lists creep as an aging mechanism. My reaction is the temperature in the PWR is not in the creep range. If you're not in the creep range, why consider creep, then?

MR. CHARDOS: Only from a completeness point of view so that, in fact, was addressed and dismissed as opposed to leaving it open and not addressing it.

MR. SHAO: But if you do have a creep there, I'd like to know it.

MR. NICHOLS: I'm Bob Nichols from EPRI. The report that is complete, which is the BWR pressure level report, gets rid of creep completely. It doesn't even treat it as a plausible aging degradation mechanism. The report which is in the process of being developed, the PWR vessel, and so far it's carried that along as a plausible mechanism. It may not be there when we get through with the final review. Is that clear?

MR. SHAO: Yes. That's why I was very surprised to see creep here.

MR. NICHOLS: It may not be there when we finish the final review.

MR. CHARDOS: All right. Those degradations not evaluated in the BWR reactor vessel and the P and B reactor coolant systems and are listed in the outline of the review -- we talked before about creep and shrinkage, service, wear, and chemical and biological effects.

So in these cases, these three were not as plausible. So we're talking about mentioning being plausible. At this point, for the PWR, we're carrying it along as a possible plausible, but in this case, in those three, we've said no, they're not plausible and so, therefore, they're not carried.

It's kind of a degree of conservatism here, because how far do you carry it along in the report before you say it's not plausible and, therefore, I will not consider it.

[Slide.]

MR. CHARDOS: At this point, I'd like to just simply address, if I could, the BWR reactor vessel age-related degradation. Now, this is a report I mentioned that we have submitted in October of this year and we talked about -- earlier, there were two groups of mechanisms, degradation. Going back to the previous slide, the last bullet was for those of which you needed to put in place an age-related management or management program to address age-related degradation for certain issues.

This last bullet here mentions that there are, in fact, three of those mechanisms which require a plant specific management methodology during the license renewal period. And these three, as the report outlines, is stress, corrosion, cracking of attachment welds to the reactor vessel; fatigue of the CRD or control rod drive return line nozzles for the BWR-2 reactor vessels; and neutron radiation embrittlement of some worst case vessel weldments is a concern for a few applicants.

And that last one really is, to some extent, a current day problem for maybe two or three. It may, in fact, be a 40 to 60 or 40 to 70 year problem for two or three also. So it's kind of a limited concern for two or three between now and age 40 and two or three or so after age 40 in the license renewal period.

But these, once again, are the three that fall out, which a licensee or an applicant would need to put together put a plant-specific program to address these three. That's really the significance of this particular slide. These are three, but they're reviewing all of them for the PWR. These are the three for which they have, after review, fallen out that need to be addressed. They could possibly be outside of established limits.

We talked before about being within bounded limits, outside limits, and inside limits.

MR. SHAO: Is the reactor containment considered in this program or a separate program?

MR. CHARDOS: In a separate report for both BWR and PWR. At the end, I'll kind of go through the report we're putting out, in terms of submittals, and the BWR reactor vessel internals and the PWR reactor vessel internals are two more reports that were not part of the pressure boundary.

MR. SHAO: What about reactor vessel supports?

MR. CHARDOS: That's in the reactor vessel itself. The supports are covered in the reactor vessel.

[Slide.]

These are mechanisms which have established programs for inspection and testing for which the degradation is bounded within acceptable limits. We talked before about being bounded within limits and those that may be outside established limits.

These here, namely neutron irradiation, vessel belt line weldments, in the majority of the BWRs -- and for a PWR, LPCI nozzles -- stress corrosion cracking of CRD stop tubes and neutron flux holders, fabricated from stainless steel and subject to welding stress, fatigue of PWR reactor vessel feedwater nozzles, stress corrosion cracking, and fatigue of nozzle safe ends, depending on the geometry, and material selections, fatigue damage of reactor vessel studs, and fatigue damage of reactor vessel support skirts.

These mechanisms have been reviewed, and the report mentions that these are covered and are bounded within existing programs for inspection and testing, and are therefore not a problem. They have been reviewed and been found to be not a problem.

So this is kind of the second set of mechanisms that were reviewed and found not to be a problem.

MR. BLOCH: What's the criterion for "not a problem"?

MR. CHARDOS: In terms of established limits -- take fatigue on the feedwater nozzles for the skirt. There are limits established for various mechanisms.

MR. BLOCH: The initial design safety margins were what?

MR. MARSH: So you evaluated out to the extended life, and the cumulative fatigue limits were not exceeded. Is that correct?

MR. SHAO: Yes.

MR. CHARDOS: On the PWR, you used to have cracks. It's in NUREG that periodically, of course, you inspect. There's an internal and an external inspection requirement. I think it's NUREG 0619 that requires you periodically, every so many outages, to look at the external PT, internal PT, so that's an established program. You're talking about temperature monitoring and those kind of things that address fatigue of feedwater nozzles.

MR. MARSH: You put on thermal sleeves?

MR. CHARDOS: Right. Thermal sleeves.

So that's a program that's under way, and it's covered.

MR. MARSH: Notwithstanding the words that the code uses to look at the fatigue usage factor, the fatigue usage curves themselves, right?

MR. CHARDOS: Correct.

MR. MARSH: You're saying you need the curve.

MR. CHARDOS: Correct, and that's separate, as we heard before.

MR. MARSH: So your conclusion may be changed, depending on the code.

MR. CHARDOS: Yes. True.

MR. MARSH: Do you have a mechanism for that, to review your conclusions?

MR. CHARDOS: Well, as part of the NUMARC/NUPLEX working group, we have PNCS representation on the working group, and we have people sitting on both groups, and so we have feedback from one to the other, so we don't lose sight --

MR. SHAO: Let me ask you a generic question. Do you have any designs relating to the vessel? How do you tell generically that everything's okay. How do you know you're being independent?

MR. CHARDOS: Pretty much, in the report, if you look at it and you see "feedwater nozzle" and "feedwater sparger" and "thermal sleeves," you go through the various thermal sleeve designs, whether it's single-ring, double-ring, welded, forced, and you do --

MR. SHAO: But how do you know the design located is the worst design? Different geometry, different diameter, different thickness, different thermal mixing, how do you know the design you are looking at is the worst design?

MR. STANCAVAGE: My name is Peter Stancavage, from GE.

We did not investigate all 30 feedwater nozzles. We looked at about 12 of them and took the worst of those, and then we supplemented that by the inspection programs that plants are undertaking and confirmed that cracking is not going on.

MR. SHAO: But how do you create another 60 years now?

MR. STANCAVAGE: Well, extension programs.

MR. SHAO: Another 20 years?

MR. STANCAVAGE: Yes.

MR. SHAO: But you need 60 years.

MR. STANCAVAGE: Well, the extension programs say, every three cycles, one has to inspect feedwater nozzles, and that inspection program would continue throughout the service life of the plant, whether that's 40 years or 60 years or 80 years. The basis for concluding -- at least on the feedwater nozzle -- that fatigue is well managed is inspection more than analysis. Support is good at the other end.

We looked at a number of other configurations, that the usage factor, based on existing codes, would always be less than one for 80 years.

MR. SHAO: But you analyzed the geometry and came to this conclusion?

MR. STANCAVAGE: Yes.

The report is based on assumptions, and each individual plant has to verify that those assumptions are true.

MR. SHAO: What you're saying is that this is a public report, and each licensee submits his own findings of the bounding.

MR. CHARDOS: Right. Absolutely.

MR. SHAO: Does that mean you're bounding -- that's very conservative.

MR. NICKELL: Bob Nickell. It should be pointed out that one of the issues that fell through to requiring a plant-specific management was the CREL nozzles on the BWR IIs. That happened to be one that fell outside the limits and had to be treated as a special case. It required that additional plant-specific management program. That's an example. You can't cover everything by bounding analysis always.

MR. SHAO: If you have a lot of margin, you can do that. But if you don't have enough margin, it could be very difficult.

MR. CHARDOS: So you've really only extended it 20 years, right?

MR. SHAO: Yes. The life expectancy.

MR. CHARDOS: For conclusions on the PWR reactor vessel and the PWR and BWR pressure boundary, we mentioned before that those are reports that were submitted on a previously agreed upon schedule within the next six months or so, and those reports will have detailed evaluations, conclusions, similar to the ones that I've mentioned here for the BWR reactor vessel, and those will in fact be submitted and reviewed and, hopefully, approved by the NRC.

As I mentioned before, we're putting out 10 topical reports. The four that I mentioned were the BWR and PWR reactor vessel and the P- and the B- pressure boundary. Some of the other ones that either had been submitted or will be submitted are the PWR containment, which went in in the summer of this year, in August. We had the BWR containment and containment cables and class I structures, and the PWR and BWR reactor vessel internals. Those make up the 10 reports that the industry plans on submitting.

MR. SHAO: The reactor vessel supports are here.

MR. CHARDOS: The reactor vessel supports for the BWR reactor vessel are in the report. The PWR reactor vessel will address its vessel supports. The pressure boundary should cover that. We have to take a look at the scope.

MR. SHAO: What about the steam generator supports?

MR. CHARDOS: I have to take a look at that. I'm not exactly sure if that's covered or not. I'm not sure if the steam generators are specifically covered in the pressure boundary for the PWRs. I'm not exactly sure. I'll have to get back to you on that.

MR. SHAO: The steam generators may have an aging problem, because the temperature fluctuates.

MR. CHARDOS: We'll take a look at that.

That's all I have.

MR. MARSH: I have a question.

MR. CHARDOS: Yes.

MR. MARSH: Back in the earlier slide, you talked about designating some systems and structures as safety significant. What does that mean? What are you saying?

MR. CHARDOS: As defined, a component whose failure would affect the function of the system or the component to do its safety function.

MR. MARSH: Now you're talking about active machine: pumps and valves, for example.

MR. CHARDOS: Or support plates.

MR. MARSH: So "safety significant" doesn't extend to active machinery? It's only for passive machinery?

MR. CHARDOS: Both.

MR. MARSH: It does cover pumps and valves?

MR. CHARDOS: Yes.

MR. MARSH: I didn't hear any words discussing those.

MR. CHARDOS: I think pressure boundary IRs --

MR. MARSH: That's all in the pressure boundary portion of it. I'm talking about pumps and valves, their capability to perform their job. Active machinery.

MR. CHARDOS: Specifically, like main coolant pumps.

MR. MARSH: Not main coolant pumps. I'm not sure if that's safety significant. Safety injection pumps, high-pressure coolant injection pumps, check valves, those kinds of things.

MR. CHARDOS: Currently, I don't believe we have any reports contemplated to cover that.

MR. BURTON: My name is Rich Burton. I'm with EPRI.

The 10 IRs that Jim is going to go over shortly, to show you what they are, are basically only in safety class I systems, in containments, for instance. I realize there is a bunch of others we discussed. The screening methodology or methodology to identify something to evaluate for license renewal has a process such that all systems are evaluated using a very similar basis, looking at effective programs, significance, and so forth, for all systems. They are not generic reforms in a plant, the valves and RHR and LPCI systems, and so forth.

MR. CHARDOS: Anything else?

[No response.]

MR. CHARDOS: Thank you very much.

MR. SHAO: Anyone from Northern State Power want to comment? Anybody from Northern State Power?

The next speaker will be from Yankee Atomic, Cedric Child.

MR. CHILD: My name is Cedric Child. I'm with Yankee Atomic Electric and we're currently doing the lead plant PWR analysis on our plant -- on our Yankee Plant out in Roe, Massachusetts.

Roe is a pressurized water reactor. As mentioned previously, the pressurized water reactor pressure boundary consists of the reactor vessel, which would include the control rod drive mechanism housings, the main coolant system piping and the main coolant system components such as the pumps, the valves, the primary cite of the steam generator and the pressurizer.

Now, all these components are safely Class I components, so, a discussion which really hasn't arisen here as to significance of safety or significance to safety doesn't really apply for the reactor pressure boundary components. They're all safety Class I, they all have significant safety requirements.

Now, these -- the next step in the plant life renewal process is to evaluate these components against degradation mechanisms and maybe we should step back here we talk about degradation mechanisms because the degradation mechanisms that we're analyzing for are really nothing new. They've been identified, they're generally well understood based on years of experience.

We've had some discussion here earlier about possible changes undergoing in the understanding of these degradation mechanisms. The quote that came quite often was the jury is still out on these mechanisms. We're continually understanding the relationship between the environment and the materials and these interactions will continue to develop. These understandings will continue to develop as we go along. This has been a factor of the operating plants and there's no reason to expect this to change at all just because we're going into a plant license renewal.

However, the industry and Jim Chardes present the concept of the industry reports, particularly in the reactor pressure boundary area we are renewing the unique aspects of degradation mechanisms and how they are tied in in the license renewal area.

We've seen lists of degradation mechanisms. You've seen the list of degradation mechanisms that were listed for the IRs. We feel that we should -- as part of our evaluation we'll be looking at age related mechanisms and looking at the significant mechanisms in the plant license renewal area.

Now, in addition to the industry work that's going on, that is, the industry reports, each plant will then demonstrate their applicability to the industry report on a plant unique basis and that's a point that I think got a little muddled a little while back that we will be looking at the IRs. The IRs -- the industry reports -- are doing the best they can to encompass on a generic basis the relationship between degradation mechanisms and materials but it will be up to the individual plants to show that the conclusions reached in the industry reports are applicable.

As an example of a degradation mechanism, let me just touch on fatigue, because that's a primary one in this area and I might be stepping into it here but fatigue is a mechanism that's always been considered in plant design. What has changed has been the complexity of supporting analyses. This complexity has increased with the development of codes. The original B-31-1 codes, for instance, had implicit within it an assumption that the plant would go from -- would go through a full temperature cycle some 7,000 times. With the introduction of ASME 3 we're now doing explicit calculations for design cycles.

Generally speaking our system, at least the reactor pressure boundary system, has been very resistant to fatigue problems. Where we've been explicitly designing for fatigue we have not found -- we have not had many problems develop in that area.

MR. SHAO: Can I ask a question on this?

MR. CHILD: Sure.

MR. SHAO: Since Class I is B-31-1 --

MR. CHILD: Right, B-31-1, yes.

MR. SHAO: B-31-1 power has fatigue in a general sense, okay, but they don't have a local discontinuity area with fatigue?

MR. CHILD: That's right, that's right. The --

MR. SHAO: Then how do you really know that what is the really fatigue at, let's say, the local discontinuity area?

MR. CHILD: Well, the point I was making was that fatigue was considered. It wasn't considered to the degree it is today.

MR. SHAO: How do we know that for licensing for the local continuity area there's no fatigue problem?

MR. CHILD: Well, what we're going to have to do is still understand the differences between these codes and make sure that these differences are understood and accounted for. There's nothing new here that wasn't brought up already.

MR. SHAO: Yes, I know, but for 40 years people say, well, maybe you don't have to do a detailed analysis, maybe it's okay. Now, for another twenty years how do we know the additional, the additional damage because you don't really, we don't really have a good detail analysis, how do we know this, this local area is okay?

MR. CHILD: Well, let's consider my next slide then. What are the options open to us in the area of fatigue? I'm not saying we're going to dismiss fatigue. We still have to consider fatigue, but let's look at the options that may be open to us.

First of all, if you are a plant that's done a detailed Section 3 analysis, you can re-do your analyses to include the additional cycles that you will see for the renewal period.

MR. SHAO: That's the easy one.

MR. CHILD: That's the easy one.

Now, if you're a plant that does not have a current design, let's say, current design basis under Section 3, there are some options open to you. One is to demonstrate a similarity with a plant that has done Section 3 plants -- has done a Section 3 analysis. If you can show similar geometry and similar loading then you can reference to this Section 3 analysis and determine where they have found areas of high fatigue usage.

MR. SHAO: Their temperature had to be the same?

MR. CHILD: That's right, that's right, because your cycles and your geometry would have to -- you would have to show a similarity in loading and as well as geometry and if so then you can use the trends that their analysis has shown to demonstrate where you're going to get high fatigue usage and when you know those areas you can then do the detailed analyses.

The point I'm making is that there need not be an exhaustive Section 3 analysis. You can by benefit of history, by considering areas of geometry differences, you might want to look at those particular areas and do a detailed analysis.

There are other areas where you can economize. We talked about using the actual plant transients to better quantify the frequency of events.

MR. SHAO: Detailed analysis or not?

MR. CHILD: You would only need to do the detailed analysis for those areas where you have had high fatigue usage, where you by one way or another feel that there will be high fatigue usage and there's another area that's being looked at now and is being discussed under Section 11 through the various ASME code committees, and that's to look at a crack growth potential.

MR. SHAO: Okay, has anybody done an analysis dealing with sharpening your pencil and you have to change your operating procedure?

MR. CHILD: You're asking me is there any place?

MR. SHAO: Yes.

MR. CHILD: Where they have sharpened the pencil?

MR. SHAO: Sharpen the pencil -- you have to change your operating procedure or --

MR. CHILD: I'd guess I'd have to defer to someone like Art Dearduff.

MR. DEARDUFF: Art Dearduff. Inherently there have been quite a few of these things that have gone on where cracks have been found in plants and so inherently that means that its usage is greater than one and analyses have in many cases have shown inclement life, no cracks.

MR. SHAO: Yes, but this is after the fact.

MR. DEARDUFF: It's after the fact but it's --

MR. SHAO: If many years ago they found cracks, and the changed the design, but if you had found anything before the cracks, you found it, you do a good analysis, you do a detailed analysis and you found that the damage --

MR. CHILD: May I try to re-phrase the question?

MR. SHAO: Yes.

MR. CHILD: Has designed -- has a fatigue analysis ever influenced the design?

MR. DEARDUFF: I'm certain it has. You know, the MSSS vendors have been doing fatigue analysis, well, let's say, even before that when they were designing --

MR. SHAO: I'm thinking for operating plants. For plants who are designed for B-31-1, if somebody re-do that analysis using Section 3 and find out that fatigue damage is too high.

MR. CHILD: Has there been an instance where someone has done B-31-1 and had to do, I mean, re-do?

MR. DEARDUFF: Yes.

MR. SHAO: Right.

MR. DEARDUFF: People had to re-do fatigue calculations on the surge line using Section 3 techniques and that influenced some designs in some plants, so the answer is yes, after the fatigue was found --

MR. CHILD: Until plant life renewal came along those plants then had a basis in B-31-1, had no reason to re-do it through Section 3 except to solve particular problems.

MR. SHAO: B-31-1 never found a surge line --

MR. DEARDUFF: Neither did Section 3.

MR. SHAO: If you do it right.

MR. CHILD: Do you wish to discuss this more?

MR. BLOCH: Go ahead.

MR. CHILD: I did bring up the crack growth potential in which you consider the possibility of a crack just below the level of detection and you determine the time it would take for that crack to propagate. Now, you can couple that with your surveillance program as necessary, or if necessary. You may find that the crack will not propagate, that it would relieve itself and that's an alternative method to doing a fatigue type analysis, or to do an analysis to show the effects of fatigue.

Then finally, as an add-on, as another possibility which wouldn't replace, but would be in addition to these methods above, there are various fatigue monitoring systems being looked at now. The point that I wish to make here is that there are several methods available for managing an age-related degradation.

The methods can be summarized as: further analysis to demonstrate that the projection degradation is acceptable through the renewal period, or demonstrating the current programs are adequate to assure the degradation mechanism does not impact safety. For the renewal period, you may want to go to some procedural enhancement, and that's where the question of trending is a possibility as a means of assessing the effects of a degradation mechanism.

You can modify operating practices and in may cases, you can consider component replacement or refurbishment. In managing degradation mechanisms, programs should only be required for components important to safety. That's not an issue here when we're talking about reactor pressure boundary, but in a general sense, as a message that Yankee wants to make in this workshop, is that we should only be required to institute programs for components important to safety.

No additional programs should be required for those components that are already covered by existing programs, and finally, any rule that's set up, should allow a flexibility for managing degradation.

MR. BLOCH: Does it follow that a program is adequate for the first forty years, continues to be adequate after that?

MR. CHILD: Does it follow that a program adequate --

MR. BLOCH: We have a program that is now in existence that is adequate for the first forty years, but questionable beyond that?

MR. CHILD: Well, you make a demonstration that the components -- that the existing programs are satisfactory for the remaining -- for the license renewal period. If the existing program -- maybe I've gotten cause and effect a little mixed up here, but if the existing program is satisfactory, there's no need to impose additional programs. Let me express it that way.

So, additional administrative controls are only applicable to special actions necessary to manage age-related degradation in support of licensing renewal and then such action should become license commitments for the license renewal. That concludes my presentation. Yes?

MR. MANOLY: The scope of the program we're talking about on safety; I don't think that's consistent with the NUMARC documents. The NUMARC document does not say --

MR. SHAO: Safety related.

MR. MANOLY: No, it doesn't say that. There is a distinction between new position and the NUMARC position on the screening.

MR. CHILD: Well, I really don't want to get into a discussion of screening here. It's not really applicable. Screening is not really applicable when you talk about the reactor pressure boundary. That was my point from the start.

MR. MR. MARSH: We have plans for those check valves that form part of the pressure boundary itself. That is, that section that isolates inside the pressure boundary from outside the pressure boundary.

I'm talking not about the quality now specifically to this and the other moveable parts that form a part of that boundary.

MR. CHILD: Well, I think what we would have to do is, we would have to -- that's getting into the question of operability and integrity of the valves. We would expect to undergo a program to review our current valve inspection techniques and periods of inspection and see if that's adequate or if that can continue for the remainder of life. Joe?

MR. McCUMBER: This is Joe McCumber again from Yankee. In general, a lot of this discussion came up in the fluids one this morning. We look at any fluid system as having a couple of functions; one being the pressure boundary and the other being operability. The operability function, from what we've seen so far, the programs in place are doing a pretty good job, we feel, in detecting or managing aging.

The concerns, the concerns with pins and disks and all of that have been identified and there are several programs to disassemble and inspect as required. So, we're hoping, at least from what we've seen right now, we think that that would be the case. There may be augmentations or enhancements required, but, you know, we will look at them individually.

MR. MARSH: Do I hear you saying that you're going to be systematically looking at programs that you have that are required and make sure that they are disassembly programs and comparability programs; that they are adequate?

MR. CHILD: Yes.

MR. MARSH: You mean systematically go through all of those programs?

MR. CHILD: That's right.

MR. BURKE: Rich Burke, EPRI. The IR -- and I could be wrong here -- on TRAC being tentative. I believe this is on the pressure boundary, and someone tell me if I'm wrong -- someone from General Electric or from Westinghouse or from B&W. However, the programs that answered this yesterday, in other words, the operability function would be part of the methodology to evaluate plant equipment for license renewal for all those affected programs.

Those types of things, where all these other committees are now involved with now looking at the operability. There are so many problems going on, we decided not to address those within these -- these are pressure boundary reports, looking at the integrity of that pressure boundary.

Those programs would be considered effective if implemented as part of this methodology doctrine.

MR. MARSH: These valves are not just your normal operability valves. These are pressure boundary valves. You've got to be sure that you cover these check valves and MOVs.

MR. BURKE: Pressure boundary integrity is specifically being looked at, PIVs, light water reactors, pressurized water reactor. The reactor cooling system. It does not look at a scope timed to 35 seconds of an MSIB or the spring packs, and so forth. It doesn't look at things like that in this particular industry report.

Those types of things that you're addressing would be part of the methodology for license renewal, to look at the functions independently, but not as part of this report. The answer is that it's just not part of the scope.

MR. MARSH: Okay.

I've got an itch that hasn't been scratched yet. PIVs have been a problem within the industry, not necessarily for BWRs, but for PWRs. We've seen numerous cases where they've not been functional, and there ended up being leaks to the outside of the containment. What program is going to ensure that these are adequate?

MR. BURKE: What particular program are we working on?

MR. MARSH: Be as specific as you can. Tell me how you're going to make sure that the PIVs of the primary pressure boundary are going to stay adequate for the extended life.

MR. BURKE: The PIVs specifically --

MR. MARSH: Those are the valves that I'm talking about: the pressure valves and the isolation valves.

MR. McCUMBER: Again, Joe McCumber from Yankee Atomic.

We look at any pressure boundary valve in two ways. The pressure boundary will be looked at based on the material construction, and the environments that it's within. The level at which we will look at it will depend on how harsh those two things combine. If there is a concern, we will look at it closer.

The operability, again, we look at the package of programs that are looking at that function right now and make sure that valve does what it needs to do. Right now, there seems to be significant program coverage on the safety injection valves, looking at the types of things you've brought up, as far as leakage and the valve discs. They've been issues that have been identified over the last 10, 20 years. I think we are covering them. I think they may get more coverage than some other valves, depending on the service conditions, the environmental conditions.

MR. CHILD: I'm getting the suspicion that what you're saying is that the current testing method for these valves are not adequate in today's -- I mean, it's not a PLEX issue; it's an operating issue that you're raising.

MR. MARSH: It may be partly that, but I don't want to say it's only a current possibility issue, because I think for the extended life we have to make sure that the degradation mechanisms won't appear later on.

MR. McCUMBER: Did I get closer to an answer?

MR. MARSH: Yes. You were closer that time.

You've got a program where you systematically do the simple check valves, and sometimes routine is bad.

MR. McCUMBER: Well, I think in general that's not done today. For the important valves, it is beginning to happen today. There are valves that are being disassembled. Many plants have instituted programs that look at critical valves based on their safety importance and their environmental condition, and disassembling them, looking for the problem.

MR. MARSH: Is that going to be a plant-specific issue, or is this an industry issue?

MR. CHILD: Well, I would think it would be -- if it's indeed an industry issue, it should be looked at on an industry basis.

MR. MARSH: Well, it is an industry issue.

MR. BURKE: The answer is yes.

We at EPRI put each of those concerns in and identified the industry concerns. You can find the answer to your question right here. It is a very expensive check valve text program. It includes long-term deteriorations, if there are any.

We could spend a lot of time on this, but I'm not sure it's appropriate.

MR. MARSH: I think I've got enough to chew on. Thanks a lot.

MR. RICHARDSON: Anything else?

[No response.]

MR. RICHARDSON: Our third and last speaker who has requested some time for presentation is Edgar Landerman.

MR. LANDERMAN: I think I've made my point.

MR. RICHARDSON: You've made your point?

MR. LANDERMAN: Yes.

MR. RICHARDSON: That, then, brings to an end those that have asked for time. I guess we will open it up for anybody else that would like to make comments in this particular area, that is, primary pressure boundary.

VOICE: I'd just like to comment on Mr. Child's presentation. I thought it was very excellent. I'm particularly to hear again that not only are we considering license renewal, but I think some of us -- or at least I did -- got the wrong impression today that what is happening during the 40 years doesn't really make that much difference. It really does. The operation, the maintenance, the inspection, the testing, the monitoring of all those things, are in license renewal. I think he made that very clear. That also includes the record-keeping, certification of the documentation. I think all of those things are very important, and I was glad to hear him speak up.

MR. RICHARDSON: Good. Any other comments?

MR. ZIGLER: I'm Gil Zigler. I'm with the ASME subcommittee on vibration monitoring. We've been working on the vibration monitoring program for the primary site of PWRs and BWRs since 1976. One of the basic problems that we've been having -- and we would like just to make this point one more time -- is this question of operating industry records, training information, and documentation, which have not been maintained by operators, vis-a-vis one that has a very good, close -- whatever that may be -- monitoring program. It's an issue that we have been trying to tackle within our subcommittee, and we find it an interesting point to debate.

I would hasten to add quickly that the NRC should carefully look at this question about the differentiation between a plant that has a known, proven track record, vis-a-vis a plant that does not have a known, proven track record.

We have several analogies. In our subcommittee, we have been using the analogy of the aircraft.

MR. RICHARDSON: Thank you.

Any others?

[No response.]

MR. RICHARDSON: If not, I want to say -- and I'm sure I speak for Larry -- we appreciate your participation and comments. I understand that there will be some written comments submitted. We appreciate those. That ends this session, and the workshop will reconvene tomorrow morning at 8:30. It will be three sessions going on, so consult your program.

Thank you very much.

[Whereupon, at 4:05 p.m., session 2 was concluded.]

REPORTER'S CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission

in the matter of:

NAME OF PROCEEDING: Session 2 Public Workshop

DOCKET NUMBER:

PLACE OF PROCEEDING: Reston, VA

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

Mark Handy

MARK HANDY
Official Reporter
Ann Riley & Associates, Ltd.

①

Bill OD Chm. ASME Subgroup on Fatigue Strength & Coordinator of ASME Code Program to Develop Improved Fatigue Design Basis

The Purpose of this ASME/PVRC Program is to update the fatigue curves in the ASME Code, extend them to high cycles to include rapid thermal cycling (mixing) and mechanical vibrations, potential load sequence effects and improved usage factor evaluation methods. The major goal of the Program is to include aging and environmental effects with special considerations as needed for weldments.

The ASME Code, including the Subgroup on Strength of Weldments, & the Subcommittee on Design have approved a 3 year development program being implemented by their research arm, the Pressure Vessel Research Committee (PVRC). The PVRC will be actively soliciting both technical and financial support for this program from the industry.

Fig (1) shows a comparison of the new general fatigue failure curves, being developed for ferritic steels with data obtained in an NRC sponsored research program. ~~NRC is active in the development of new fatigue design curves in the ASME Code and they have pointed~~ NRC has pointed out that the low cycle end of the curve is significantly lower at higher temperatures. Indeed, this is born out by statistical analyses of available data.

Fig (2) shows the latest results of an analysis of reactor water environmental effects on the fatigue life of reactor pressure vessel steels. This curve uses [illegible]

(2)

the ferritic steel from reactor water effects on crack initiation. However, once an underclad crack is initiated, the integrity of the cladding is assumed to be compromised, and reactor water effects were assumed to increase the crack growth rates per Section XI of the ASME Code. Of course the design curve used in the Code includes a factor to account for environmental effects. Thus, when environmental effects are explicitly taken into account, these factors should not be superimposed. The ASME Subgroup on Fatigue has tentatively agreed to reduce the factor on mean failure life from 20 to 10 when reactor water effects are explicitly taken into account. This factor must account for scatter in the data, surface finish and size effects.

Fig. (3) illustrates the more advanced environmentally assisted crack growth rate model being developed by the International Crack Growth Committee and the Metals Properties Council Task Force on Crack Propagation. This approach includes cyclic rate and mean stress effects. It can also be applied to develop fatigue life curves which include reactor water environmental effects.

We are looking for support for this ASME Code activity and Pressure Vessel Research Committee program.

~~on this effort which is critical to [illegible]~~

(3)

Having the numerous "how safe is safe enough" issues decided by national consensus standards committees has been found to be the ideal approach to setting such acceptance criteria.

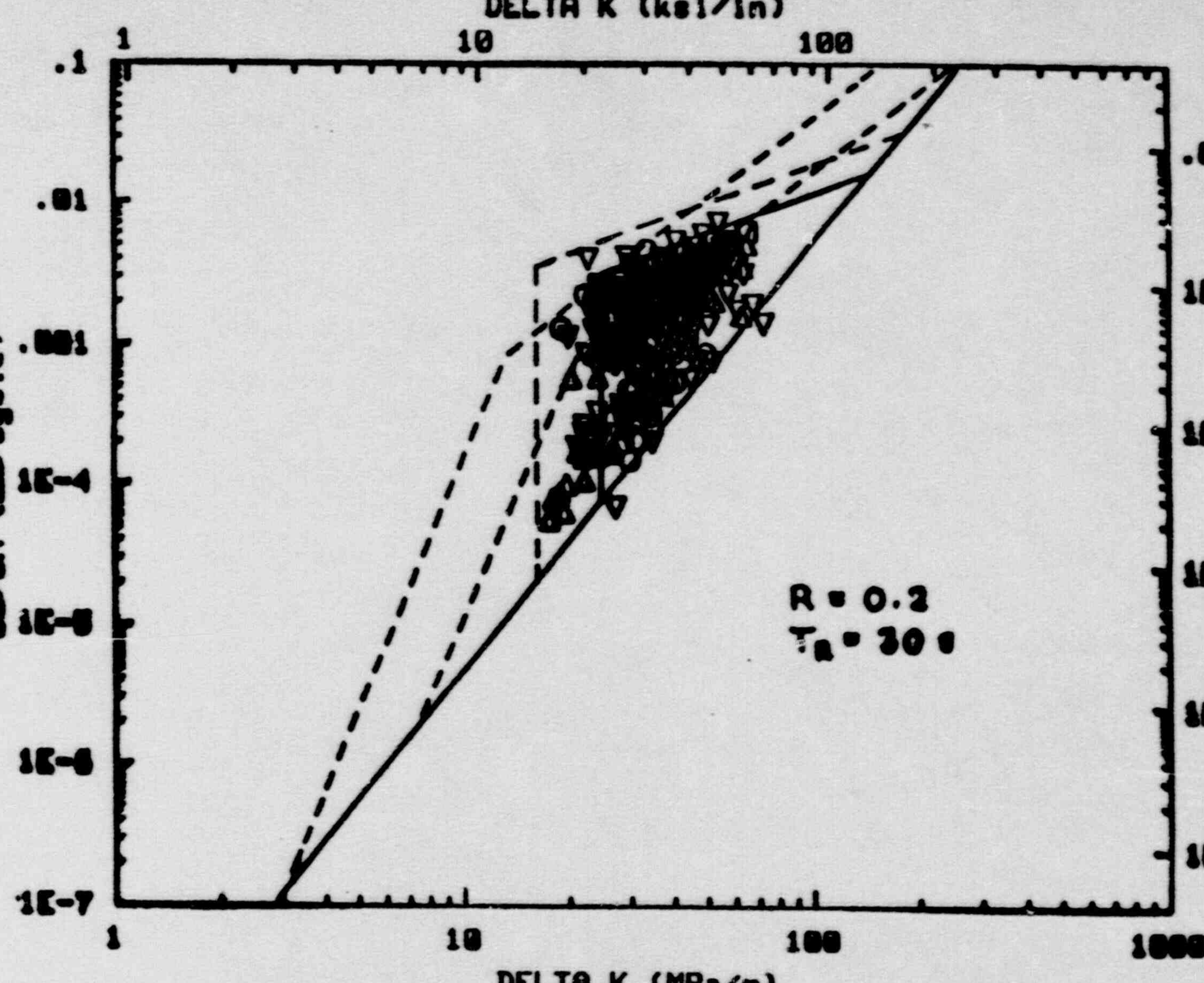

FIG. 3 NEW ENVIRONMENTALLY ACCELERATED CRACK GROWTH MODELS IN CONVENTIONAL da/dn VS. ΔK DOMAIN

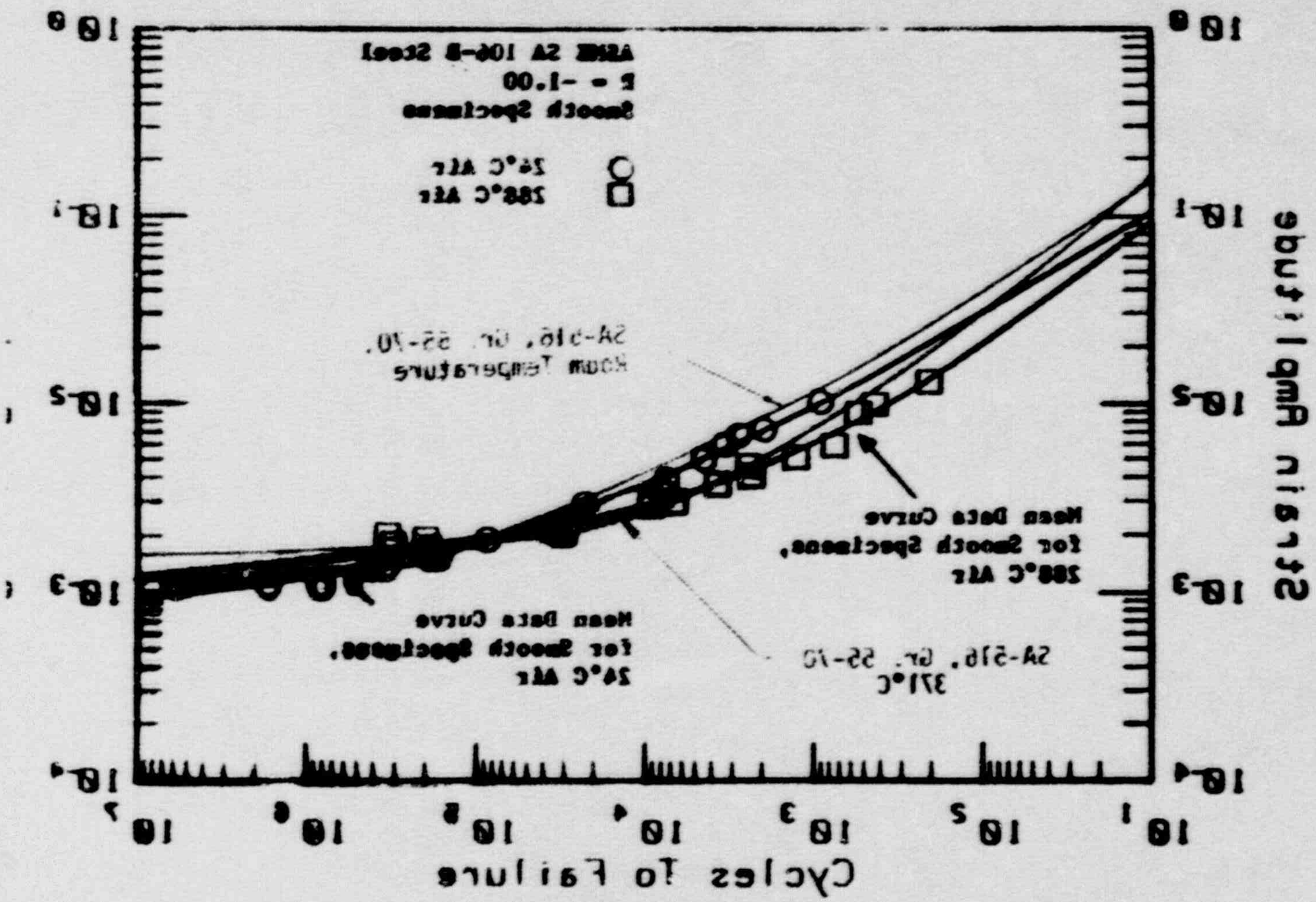

Fig. 28 Strain-life plots for smooth specimens of SA 106-B steel in air at 24°C (76°F) and 288°C (550°F).

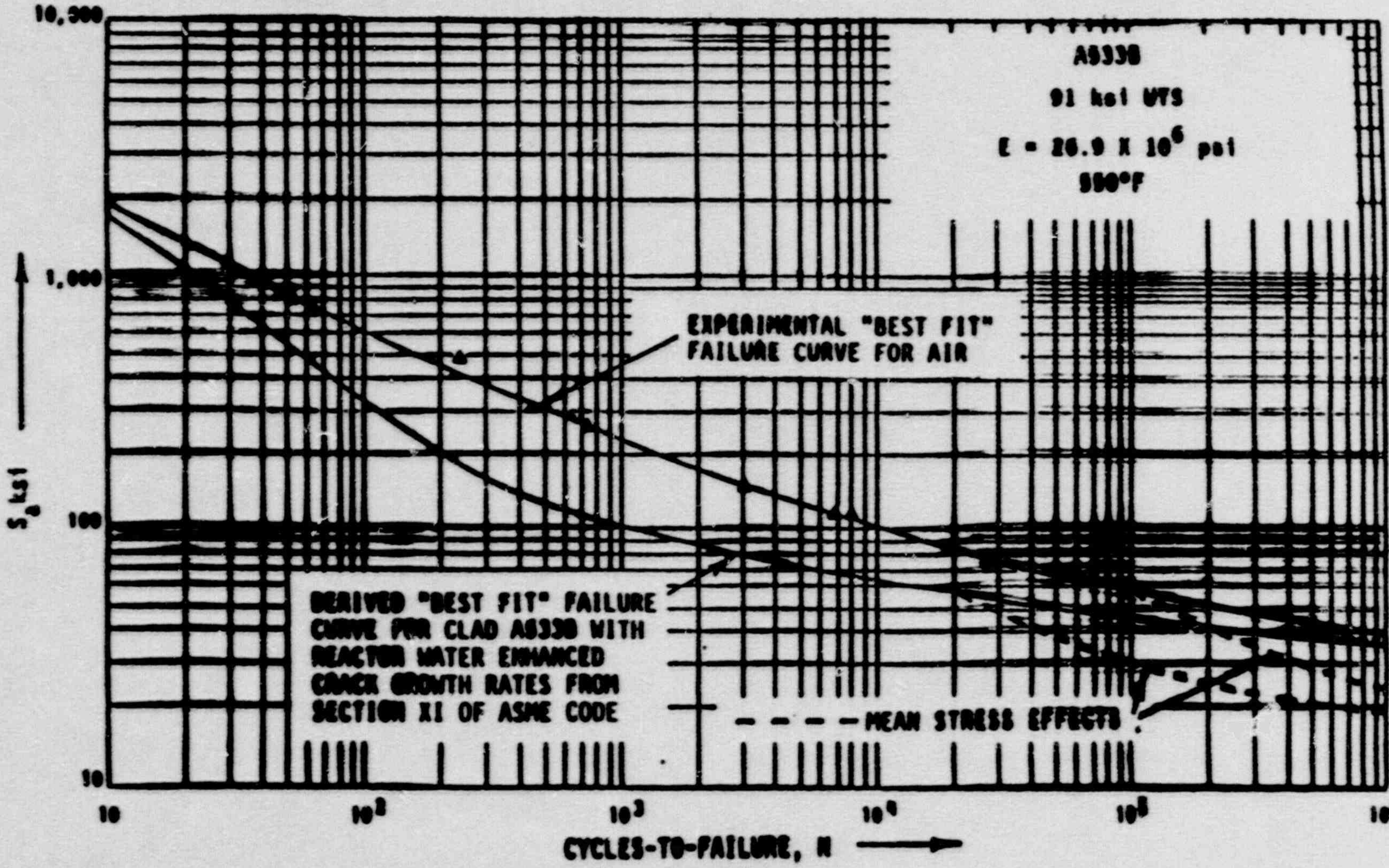

FIG. 2 A533B PRESSURE VESSEL STEEL FATIGUE FAILURE CURVE CORRECTED FOR REACTOR WATER CRACK PROPAGATION COMPARED WITH FATIGUE FAILURE CURVE IN AIR AT 550°F

Handy
NRC
11-13-89

NRC WORKSHOP ON LICENSE RENEWAL
NOVEMBER 13-14,1989
SESSION 2

PRESENTATION
ON
REACTOR PRESSURE BOUNDARY

Tape 11 A

BY
CEDRIC L. CHILD
YANKEE ATOMIC ELECTRIC COMPANY

REACTOR PRESSURE BOUNDARY
- PWR -

REACTOR VESSEL
MAIN COOLANT SYSTEM PIPING
MCS COMPONENTS

ALL SAFETY CLASS I COMPONENTS

DEGRADATION MECHANISMS

- IDENTIFIED AND GENERALLY WELL UNDERSTOOD BASED ON YEARS OF EXPERIENCE
- UNDERSTANDING OF ENVIRONMENTAL/MATERIALS INTERACTIONS CONTINUE TO DEVELOP
- INDUSTRY REVIEWING UNIQUE ASPECTS OF DEGRADATION MECHANISMS IN THE LICENSING RENEWAL ARENA
- EACH UTILITY WILL DEMONSTRATE APPLICABILITY TO INDUSTRY STUDIES ON A PLANT UNIQUE BASIS

FATIGUE

- HAS ALWAYS BEEN CONSIDERED IN PLANT DESIGN
  COMPLEXITY OF SUPPORTING ANALYSES
  INCREASED WITH DEVELOPMENT OF CODES

- SYSTEM DESIGN GENERALLY RESISTANT TO FATIGUE PROBLEMS

CURRENT FATIGUE OPTIONS

- REANALYSIS TO INCLUDE THE ADDITIONAL LICENSE RENEWAL PERIOD
- SHOW SIMILARITY TO PLANTS WITH SEC III ANALYSIS TO DETERMINE REGIONS OF HIGH USAGE. DO DETAILED ANALYSES OF THESE REGIONS
- USE ACTUAL PLANT TRANSIENTS TO BETTER QUANTIFY THE FREQUENCY OF EVENTS
- DETERMINE CRACK GROWTH POTENTIAL AND AUGMENT THE PLANT SURVEILLANCE PROGRAM AS NECESSARY
- CONSIDER FATIGUE MONITORING SYSTEMS

MANAGING DEGRADATION MECHANISMS

- PROGRAMS SHOULD ONLY BE REQUIRED FOR COMPONENTS IMPORTANT TO SAFETY

- NO ADDITIONAL PROGRAMS SHOULD BE REQUIRED FOR THOSE COMPONENTS ALREADY COVERED BY EXISTING PROGRAMS

- RULE SHOULD ALLOW FLEXIBILITY FOR MANAGING DEGRADATION

SEVERAL METHODS AVAILABLE FOR MANAGING DEGRADATION

- FURTHER ANALYSIS TO DEMONSTRATE THAT THE PROJECTED DEGRADATION IS ACCEPTABLE THROUGH THE RENEWAL PERIOD
- CURRENT PROGRAMS ARE ADEQUATE TO ASSURE DEGRADATION MECHANISM DOES NOT IMPACT SAFETY
- FOR THE RENEWAL PERIOD, PROCEDURAL ENHANCEMENT MAY BE NECESSARY
  - e.g., Trending
- MODIFICATIONS TO OPERATING PRACTICES
- COMPONENT REPLACEMENT OR REFURBISHMENT

ADDITIONAL ADMINISTRATIVE CONTROLS

- <u>ONLY</u> APPLICABLE TO SPECIAL ACTIONS NECESSARY TO MANAGE AGE RELATED DEGRADATION IN SUPPORT OF LICENSE RENEWAL

- SUCH ACTIONS BECOME LICENSE COMMITMENTS FOR LICENSE RENEWAL