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NUCLEAR REGULATORY COMMISSION

Title:	Entergy Nuclear Operations, Inc. Indian Point Nuclear Generating Station
Docket Number:	50-247-LR and 50-286-LR
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UNITED STATES OF AMERICA

U.S. NUCLEAR REGULATORY COMMISSION

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BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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In the Matter of:	: Docket No.
ENTERGY NUCLEAR OPERATIONS, INC.	: 50-247-LR
(Indian Point Nuclear Generating	: 50-286-LR
Station, Units 2 and 3)	: ASLBP No.
	: 07-858-03-LR-BD01

Tuesday, November 17, 2015

Doubletree Tarrytown

Westchester Ballroom

455 South Broadway

Tarrytown, New York

BEFORE:

LAWRENCE G. MCDADE, Chairman

MICHAEL F. KENNEDY, Administrative Judge

RICHARD E. WARDWELL, Administrative Judge

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P R O C E E D I N G S

8:31 a.m.

CHAIRMAN MCDADE: This hearing will come to order. We will continue with discussion on Contention Number 25. Before we get started however, there was one administrative matter that I forgot from yesterday that I'm embarrassed that I forgot. Looking at the Statement of Position of Entergy, this is Exhibit 615, Page 33, third paragraph down. First word of the paragraph is stricken from the record.

MR. KUYLER: Your honor, could you say again the page?

CHAIRMAN MCDADE: Page 33, third paragraph.

MR. KUYLER: Yes, your honor.

CHAIRMAN MCDADE: We are striking from the record as grossly inappropriate the first word of that paragraph. Are you ready to proceed?

ADMIN. JUDGE WARDWELL: What about the homework assignments? Should we start with Dr. Lahey first and then we'll assume Entergy has something in regards to the data? If not, we can give you more time if you need it. But let's start with Dr. Lahey.

DR. LAHEY: So, he wanted these.

ADMIN. JUDGE KENNEDY: Well, we don't what these are, Dr. Lahey.

CHAIRMAN MCDADE: Yes.

DR. LAHEY: Oh, they're -- you asked me to --

CHAIRMAN MCDADE: Dr. Lahey, excuse me. If you're going to talk, you've got to be sitting with the microphone. Otherwise, it's going to get lost for the record.

DR. LAHEY: Okay.

CHAIRMAN MCDADE: And we can get someone --

DR. LAHEY: Richard Lahey. Your honor, you asked me to give you the references that I had cited yesterday. So I have copies of them for you.

CHAIRMAN MCDADE: Well, could you just tell us what the cites are?

DR. LAHEY: What the references are?

CHAIRMAN MCDADE: Yes.

DR. LAHEY: Okay. There's three of them. One of them is Kanaski, the other one is Arai, and the other one is Korth, et al. And these had to do with low cycle fatigue versus high cycle fatigue and the effect of this on failure for testing components that were irradiated.

ADMIN. JUDGE WARDWELL: And were these previous exhibits that you had submitted?

DR. LAHEY: Yes. They were in my

testimony, but Judge McDade asked me if I would -- I thought it was my homework assignment if I would make him a copy of them, so I did.

MR. SIPOS: And Judge, this is John Sipos, we have the exhibit numbers if you would like.

ADMIN. JUDGE WARDWELL: And that's all we needed.

MR. SIPOS: Would that be helpful?

CHAIRMAN MCDADE: Yes.

ADMIN. JUDGE WARDWELL: That's what I think we were trying to imply is we just needed the cites for those, we didn't need the cites.

MR. SIPOS: So the first one, New York State 564 is Arai, A-R-A-I. Another one is Riverkeeper 152 and that's Korth, K-O-R-T-H. And the third is NRC 177 and the first author is Kanasaki, K-A-N-A-S-A-K-I.

CHAIRMAN MCDADE: Okay. Thank you, Mr. Sipos.

ADMIN. JUDGE WARDWELL: And thank you, Dr. Lahey, for digging out those specific ones. We only really needed the cites though. Sorry for the misunderstanding, we appreciate your effort.

DR. LAHEY: Okay.

ADMIN. JUDGE WARDWELL: Entergy, is there anything you'd like to offer in regards to the data that demonstrates the effect of, I believe it was looking at the effects of embrittlement on fatigue in some fashion or fracture toughness.

DR. LOTT: Yes. My understanding was there were two questions that came up yesterday related to data. That being one of them, the irradiation effects on fatigue. And there was another question about high fluence properties in general and the survivability of materials at high fluences. As to the question of the effects of radiation on fatigue, I must admit, I've been reliant on NRC NUREG/CR-6909, which in Section 1.32 has a discussion of irradiation embrittlement and fatigue.

Let me say that I have a difficult time with the word synergism when we talk about these particular relationships. I understand that all irradiated materials are embrittled and the question to me is whether or not irradiation also has effect on the strain life of the material or on the fatigue resistance of the material. I don't see that as a synergism, that's just a question of irradiation embrittlement and irradiation fatigue life.

I can give you a summary of what I think it says in NRC 6909. It does reference the papers by

Korth and Harper. I'll point out that, that data in that paper is basically generated for fast meter conditions, it's not at a temperature that we would generally think would be applicable for PWR interactions. The paper by Arai was actually reporting on irradiation embrittlement work that was done at Westinghouse. I'm not sure that there's any fatigue data in that paper at all, I don't recall any.

And the paper by Kanasaki, I'll point out I'm a co-author on that paper, so we're certainly aware of it. It is the one paper that, and we can put the data -- do you want to discuss the data or just note that the data is there? That is the paper that does show that at low strain amplitudes, below I believe it's 0.6, the data that was tested of PWR relevant conditions all showed an increase in fatigue life with irradiation. We're not to discussing the CUFs in Contention 26 yet, but when we do, I think we'll find that there's, A, a limited number of materials of the internals which have CUF values calculated are irradiated. And those materials, I do not expect to have a large strain amplitude. So I think the data in the Kanasaki paper is directly relevant to the conditions we're talking about.

ADMIN. JUDGE WARDWELL: Okay. When you

first started off referencing a NUREG was it?

DR. LOTT: It's CR-6909 and I'm sorry, I'll find you the --

ADMIN. JUDGE WARDWELL: And is there an exhibit number. If you just --

DR. LOTT: Yes, there is.

ADMIN. JUDGE WARDWELL: -- have an exhibit number, that would suffice because then we can find it through the --

MR. STEVENS: Your honor, Jerry Stevens of the Staff. That's New York State 490 Alpha.

ADMIN. JUDGE WARDWELL: Okay.

DR. LOTT: Yes.

ADMIN. JUDGE WARDWELL: Thank you.

DR. LOTT: Of which Mr. Stevens is an author.

ADMIN. JUDGE WARDWELL: Okay. Is there anything else you'd like to offer in regards to sources of data backing your claims from yesterday?

DR. LOTT: Well, there is, in addition, the issue of high fluence properties. There was some question in our mind whether the question was high fluence data on fracture toughness or high fluence data on the yield stress of materials. My observation again would be that, when we come to the very highly irradiated components in the plant, particularly the baffle former bolts, and you look at strategies for dealing with the baffle former bolts, anything that we observe with a crack, we assume has failed. So we never actually use a fracture toughness value for very highly irradiated materials because we have this failure assumption in our acceptance criteria.

So if you want to know about those materials at very high fluences, I would recommend to you and I'm going to find it here, it's Exhibit Entergy 000646, MRP 210, has in there in Figure 1-4 a plot of yield strength versus neutron fluence that goes out to fluences data certainly greater than 90 dpa, actually one data point closer to 120 some dpa. Which would be well beyond what we would expect the fluence on the vast majority of the reactor internals ever to see.

ADMIN. JUDGE WARDWELL: Thank you. And again, is there a -- did you reference an exhibit number in your --

DR. LOTT: Yes, I believe I did. ENT 646.

ADMIN. JUDGE WARDWELL: Oh, ENT, okay. Thank you.

MS. BRANCATO: Your honors, this is Debra Brancato from Riverkeeper. Dr. Hopenfeld has indicated to me he has some helpful input to this discussion as well if you care to entertain some testimony from him at this time.

CHAIRMAN MCDADE: Well, probably not at this time, but before we leave the topic.

MS. BRANCATO: Okay.

ADMIN. JUDGE WARDWELL: Okay. So let's continue on with some questioning and dealing with Contention 25 --

CHAIRMAN MCDADE: But before you go on, Dr. Hopenfeld is, as I said yesterday, sometimes things move and we hear an awful lot of things. Take a card, write down the comment to remind yourself, and you will be testifying later. Thank you.

ADMIN. JUDGE WARDWELL: And I'll refer again to Entergy's Exhibit 616, their testimony, Answer 174, Pages 113 to 114. During its technical review of MRP 227, the NRC Staff specifically requested additional information on how the program accounts for synergistic effects. And I guess I'll ask again for Entergy, how does your AMP look at and include actual analyses which have addressed the change in fatigue strength as a function of varying degrees of embrittlement of the specimen that occurs with time?

DR. LOTT: Well, again, to me there are two questions in mind there. One is when we talk about embrittlement, if we're talking about the loss of fracture toughness, we calculate or look at lower bound fracture toughness curves based on the dose of the material. So we're using what we believe to be bounding fracture toughness curves and then analyze the crack that has grown. So there's certainly an interaction there between the tolerance of the material and the embrittlement and the fatigue crack growth, they're tied into the same calculation.

ADMIN. JUDGE WARDWELL: But that's a proposal of what you would do if a crack was observed, right? And until a crack is observed and as part of the development of your AMP, you haven't done any analysis in regards to that, is that correct?

DR. LOTT: Well, we've certainly done analysis to show that we do not expect to see cracks occur in these materials whatsoever. Again, the CUF would be a good example of that. We've looked at extensive analysis of IASCC, particularly in the highly rated components we generated a fairly complex model of the reactor internals to look at the aging of those reactor internals and predict where IASCC might occur. The only place that we actually predicted it would occur was in the baffle former bolts, which we discussed. There's a fairly detailed process for that.

In our processing and recommendations for inspections, we certainly looked at susceptibility to multiple degradation mechanisms as part of that and we based our inspection program on determination of the effect of those mechanisms. So as far as we're concerned, we're obviously inspecting for irradiation fatigue, IASCC, or SCC, all three of which are cracking mechanisms and the inspection program doesn't care which one of those caused that, it's just simply looking for that. So I think in our prioritization of inspections, we certainly have looked at that. We've looked at that in the design of our program in general.

ADMIN. JUDGE WARDWELL: I'm just trying to think of whether I'll wait until 26 to discuss this further or not, but I'll bring it up again now, I guess, because we did talk about it yesterday and I want to fix again, I guess with you, Dr. Lott. I'm trying to get a grasp on the handle between the peak strength and fatigue durability. As I heard our discussion yesterday, I can understand why someone may say embrittlement won't come into effect really until there's a crack due to excessive peak strength. You've got to get past the peak strength and then embrittlement comes into play to a certain degree. But fatigue durability doesn't rely on loads that exceed the peak strength. It's a repetitive loading that causes the types of cracks, is that correct?

DR. LOTT: Yes.

ADMIN. JUDGE WARDWELL: And so, again, I'm back to the question of where is anyone looked at whether or not that fatigue durability is influenced by embrittlement and to what degree is it?

DR. LOTT: Well, again, this is, at least from my perspective, I don't necessarily -- to me, embrittlement and fatigue life, well, again, embrittlement is not a property in and of itself. The properties are yield stress, ultimate stress, ductility measurements, such as total elongation, fracture toughness. And I would see, again, as another issue, would be the question of what's the impact on the S-N curve, the number of --

ADMIN. JUDGE WARDWELL: Well, can't we use fracture toughness as an indication of embrittlement?

DR. LOTT: It's an indication that the material has been irradiated and irradiated materials have a decrease in that. But it does not necessarily indicate how the material will survive under fatigue conditions. So basically, we have a test for that, just like we have a test to determine yield stress. That is the fatigue life test to generation of the S-N curve. We actually take a specimen, strain it at some strain amplitude, we put it on repeated cycles, and count the number of the cycles it takes to fail the specimen or to have a load drop in the specimen actually. So those fatigue life curves, again, are behind basically the calculation of the CUF factors based on a design curve, which is bounding to all of the measurements of fatigue life.

ADMIN. JUDGE WARDWELL: Let me ask you this, if I did a fatigue test, whatever the fatigue test might be, if I try to fatigue it, as a specimen that's not irradiated whatsoever, and I repeated that test under different degrees of radiation exposure, what would you expect the results of the fatigue test to do? Remain the same, improve, or degrade?

DR. LOTT: That's exactly what we did in the Kanasaki paper that, again, I'm a co-author on.

ADMIN. JUDGE WARDWELL: Okay.

DR. LOTT: We looked at that. We looked at that at some fairly high fluences. So, if you'll -- the fluences are in two different papers reported in different units. If you'll take my word for it, in the Kanasaki paper the data is about 20 dpa. At 20 dpa, if you look at the yield stress or the fracture toughness curves, you'll see that there's a sharp decrease in the fracture toughness of the material, there's a large increase in the yield stress and, yet, the fatigue life in those specimens that we tested for Kanasaki, the fatigue life of that material got longer. So the fatigue life improved at the same time that the fluences were such that the yield stress would increase and the fracture toughness decreased.

ADMIN. JUDGE WARDWELL: Thank you. Dr. Lahey, do you have any other types of cites or evidence in your testimony that differs from what they have just expounded upon and/or what would you believe would be the change in that fatigue property as the materials are irradiated?

DR. LAHEY: Right. This is Richard Lahey for New York State. I don't disagree with anything he said except he didn't say everything. If you look at the Korth paper, which admittedly was for high pressure or higher temperature conditions for the reactor, what it says is for high cycle fatigue where you have larger amplitudes, you can have a reduction by a factor of one half of the cycles to failure for irradiated materials, irradiated up to 1.1 times ten to the 22 neutrons per centimeter square.

It is inconclusive. The reason people say it's inconclusive is there's other data that shows if you have high cycle fatigue or have low amplitude, then it strengthens and you're in the region where things get better. Here, things get significantly worse. So all the real experts, the people who do research on this, say the same thing. We don't really have any good data for light water reactor conditions and it's sorely needed. And this is why in the sustainability program they're doing those kind of tests. So I don't disagree with that at all.

What I am concerned with is there's evidence, admittedly it's not perfect, but there's evidence it can have a significant degrading effect. And then the question is, what do you do in the meantime until you can definitively tell and quantify the effect? Do you just ignore it and keep looking until you get a crack? Or do you put on some sort of factor to account -- put a cushion in to account --

ADMIN. JUDGE WARDWELL: In your cite that you just gave us, would that temperature have a big difference and is that temperature representative of what might be experienced in a PWR at IPEC?

DR. LAHEY: It's somewhat higher than the temperature that you would have in a light water reactor for sure. Therefore, it's not a perfect data set.

ADMIN. JUDGE WARDWELL: And would that likely have an influence on the data results, the extra temperature?

DR. LAHEY: It's hard to say.

ADMIN. JUDGE KENNEDY: Dr. Lahey, this is Judge Kennedy. I'm curious now that I've heard from Dr. Lott, we've got two conflicting views of the data, one for slightly different conditions. It appears that Dr. Lott's paper addressed the right conditions and addressed different levels of irradiation. How would you challenge his paper? You've offered up the higher temperature data, but what would you say to the data that he's presented? I mean, I recognize it's one of the three references you provided.

DR. LAHEY: Right. Well, my understanding -- I mean, he's the author, so --

ADMIN. JUDGE KENNEDY: But I think you're --

DR. LAHEY: -- he has a little advantage there, but --

ADMIN. JUDGE KENNEDY: I take it you're

offering his same paper for a different conclusion.

DR. LAHEY: Yes. I mean, it is one of the few papers that addressed fatigue and irradiation. And as I understood the focus of his paper was more into what happens once you get a crack and how does it propagate and the initiation of the crack, rather than the cycles to failure.

ADMIN. JUDGE KENNEDY: But isn't that what he just said? That the number of cycles to failure increases with the irradiation of the sample material?

DR. LAHEY: Well, that's what --

ADMIN. JUDGE KENNEDY: That the fatigue life went up?

DR. LAHEY: It depends entirely on the amplitude. It depends on the amplitude of the fatigue cycle.

ADMIN. JUDGE KENNEDY: Are you suggesting that the amplitude in his test data or in his calculations using the test data are not the right characterization of the amplitudes that would be present at Indian Point Units 2 or 3?

DR. LAHEY: I mean, I don't -- I mean, he should say what the purpose of his test was. But as I understand it, it wasn't to specifically address that.

ADMIN. JUDGE KENNEDY: And maybe this is a more general question about the term high cycle/low amplitude and low cycle/high amplitude has been pushed around here. I'm not sure sitting up here, are both applicable to the operating conditions at Indian Point? Is one grouping, I don't know if it was -- is high cycle/low amplitude more applicable to Indian Point? Or is low cycle/high amplitude more applicable? Or are they both? And is there a hole in the data as Dr. Lahey is suggesting?

DR. LAHEY: Well, as I under -- this is Richard Lahey again. As I understand high cycle, it's things like flow induced vibration, turbulence induced, flow induced vibration. This is not a great concern. It is in the steam generators and what that might do to fretting and things like that, but it's not in the primary side. It's more low cycle and many of the kind of transients they have would in my opinion give a larger amplitude. So it's more like low cycle, larger amplitude fatigue.

CHAIRMAN MCDADE: Okay. Dr. Lahey, let me interrupt for a second here. The document that you referenced was the Korth paper, K-O-R-T-H.

DR. LAHEY: Right.

CHAIRMAN MCDADE: And that was Riverkeeper

152. That paper was originally presented back in June of 1974. Is that data still valid? I mean, isn't there something more recent that you can address us to as far -- you're talking about the absence of data here and that particular study is more than 40 years old.

DR. LAHEY: Well, that's correct. But nobody that we could find in a literature search has really systematically done that and as a consequence that's why they took it on in the Light Water Reactor Sustainability Program. I agree there's data needed.

CHAIRMAN MCDADE: Okay. So that particular study, although it's more than 40 years old, it's your position that since then there has been no significant work that has generated more informative data?

DR. LAHEY: I haven't been able to find it if there has been. And I don't know anybody else that has. All the -- this has been discussed by the NRC because they had input, like I'm giving you, from some of their experts at Argonne and they took all that input in, looked at this data, the other data, and then decided it's inconclusive and so we'll do our inspection program, but not do anything separate in terms of putting a penalty factor in. But that's their job, in my view, as regulators. That's their opinion.

ADMIN. JUDGE KENNEDY: I guess -- this is Judge Kennedy. I'm still curious about low cycle/high amplitude and high cycle/low amplitude. Is this worth discussing in more detail? Is it relevant to the metal fatigue for reactor vessel internals?

DR. LOTT: Well, I think probably what's relevant to this particular discussion is the limits that are in the Kanasaki paper in general, which is this 0.6 percent strain amplitude. We believe those conditions were chosen such that they would be relevant to reactor internals, that was the intent of the testing in the first place. And, again, as I indicated, did a quick check of the CUF values that are reported for the reactor internals and those that are in components that also see irradiation, which is only a fraction of the total. And having surveyed those, I believe that we'll find that all of the strain amplitudes are within the limits that are in the Kanasaki paper. If you look at the number of cycles to failure in Figure 8 of that paper, you'll see that in some cases they are as few as 1,000 cycles. So it's not like this is thousands of cycles per year kind of numbers that we're talking about.

ADMIN. JUDGE KENNEDY: All right. Thank

you.

ADMIN. JUDGE WARDWELL: Moving on from that, Entergy's testimony, Exhibit 616, Answer 144, Pages 93 to 94, present eight age relating degradation mechanisms. And they include stress corrosion cracking, irradiation assisted stress corrosion cracking, wear, fatigue, thermal aging embrittlement, irradiation embrittlement, void swelling and irradiation growth, thermal and irradiation enhanced stress relaxation, or irradiation enhanced creep.

They then go on to say that for each of these eight mechanism, MRP 227 identifies the resulting aging effect, which will then be managed through inspections under MRP 227-A guidelines. Notably, in most cases, the key effects are cracking, dimensional changes, or wear, but in all cases, as explained below, the inspections specified in MRP 227-A are designed to detect potential aging effects applicable to each reactor vessel internal component regardless of the underlying mechanism.

And I guess I'd address this to Entergy, whoever would like to answer, if you say that in most cases the key effects are cracking, dimensional changes, or wear, but given that the effects are managed, not the mechanism, what are the key effects in the other minority cases and are they detectable by your AMP? I'm addressing your throw away statement in this testimony that in most cases the key effects are cracking, dimensional changes, et cetera. What I'm asking is, what about in those other cases that aren't included in that, what are those key effects because it is true you are claiming to be monitoring for effects and not mechanisms?

DR. LOTT: I may have to take a minute to think about this one. I suspect that, that is, as you suggested, a throw away sentence that we were probably just overly cautious. I don't --

MR. DOLANSKY: Dr. Lott?

DR. LOTT: Yes.

MR. DOLANSKY: This is Bob Dolansky with Entergy. Perhaps one example would be the internals hold down spring.

DR. LOTT: Okay.

MR. DOLANSKY: We're actually taking measurements of the hold down spring, we're not looking for cracking or wear. We're going to take dimensional measurements of the hold down spring and use that to determine whether that is acceptable. Is that --

DR. LOTT: Yes.

MR. DOLANSKY: Would that be one example?

DR. LOTT: That would be a good idea.

ADMIN. JUDGE WARDWELL: Okay. So in that particular case then for that spring, you're doing some actual measurement rather than just observing for a crack?

MR. DOLANSKY: Correct.

ADMIN. JUDGE WARDWELL: Do you have any other examples of this?

MR. DOLANSKY: Give me one --

ADMIN. JUDGE WARDWELL: And let's just assign that as a homework assignment so you don't break out in beads of sweat panicking --

DR. LOTT: Right.

ADMIN. JUDGE WARDWELL: -- trying to find everything you possibly can at this moment, I know the feeling all too well being on that side of the table often. Let's just -- when you feel comfortable, get back with a response if there's any other examples you can give in regards to those. Because I think it is somewhat important because, again, you are monitoring for the effects and I want to make sure we don't have some giant hole of something else that's out there that we would like to be able to track.

DR. LOTT: May I say that we did go through this FMECA process which looked at each component and asked ourselves what are the appropriate failure modes and consequences? So we certainly did look at that and asked ourselves the questions in the process.

ADMIN. JUDGE WARDWELL: So it sounds like your reference to most cases, you were referring to in most of the internals, we are looking for some type of strain and deformation or a crack or something in that neighborhood.

DR. LOTT: Yes. Well, we --

ADMIN. JUDGE WARDWELL: And in some of your -- and then other cases, you are doing something else besides looking for some sort of strain.

DR. LOTT: Yes. We tried to identify in each case what the effect was that we thought we were looking for in the prescribed inspection. So if it was a wear inspection, we would be looking for loss of material or evidence of wear on the surfaces. If it was something that was subject to a cracking mechanism, we would say, we're looking for cracks. And this is where the question of irradiation embrittlement would come up as well.

ADMIN. JUDGE WARDWELL: Yes. And I kind of view wear as a strain, it's not really a strain, but it's a loss of the dimension if nothing else.

DR. LOTT: Loss of --

ADMIN. JUDGE WARDWELL: A change in a dimension.

DR. LOTT: Yes. Opening of a gap, displacement of a component one with respect to the other in some small way. We tried to find the places where that was most evidence where we could see it.

ADMIN. JUDGE WARDWELL: Is it not true though that your AMP is based on the fact of monitoring for effects and not trying to deal with the mechanism that caused those effects?

DR. LOTT: Yes. I think that was the instruction of the Aging Management Program in general.

ADMIN. JUDGE WARDWELL: And, NRC, would you agree that, that's the motive behind the inspection program for the reactor vessel internals AMP?

DR. HISER: This is Allen Hiser with the NRC. Yes, that's correct. It's monitoring, inspecting for aging effects. Mechanisms create effects and they're important to understand what aging effects you need to manage, but the mechanisms themselves are not managed.

MR. STROSNIDER: This is Jack Strosnider for Entergy. If I could just add there, if you want a citation on that, you can go to the Statements of Consideration that were issued when the License Renewal Part 54 was issued in 1995. And it explicitly discusses this notion of managing effects rather than mechanisms. And I think part of the reason for that is to recognize that the effect doesn't care what synergisms, if there are any, that are happening. If you see a crack, you see a crack and whatever contributed to it, contributed to it and then you need to take the right corrective actions. But I just wanted to get you that citation in case you want to look at that.

ADMIN. JUDGE WARDWELL: How does one observe embrittlement?

MR. STROSNIDER: This is Jack Strosnider of Entergy. You asked -- is that question directed at me?

ADMIN. JUDGE WARDWELL: Yes.

MR. STROSNIDER: Okay. So the discussion in the AMP is that embrittlement is not directly observed, but it is managed through the detection of cracks. And, as we've been discussing, if you find a crack, you then need to assess it considering the material properties that would be associated with whatever level of embrittlement that component has experienced. So it's an indirect approach. There is not an embrittlement meter, if you will, that you can go in and look. You have to do this indirectly and that's what's laid out in the program.

ADMIN. JUDGE WARDWELL: Thank you.

DR. LOTT: May I add? This is Randy Lott. In MRP 227, when there were components that were subject to irradiation or thermal embrittlement, we tried to note them in that way. There was effects listed and then there would be the conditional note that says, this effect should be -- aging management for irradiation embrittlement, thermal embrittlement would be required in this component. So we would mark those places where this would be a concern.

ADMIN. JUDGE WARDWELL: Okay. Thank you. Dr. Lahey --

CHAIRMAN MCDADE: Okay. If I could, just -- and, Dr. Lahey, as I understand your position is that given the fact that you can't directly observe or monitor embrittlement, that there is a significant change, even in the absence of cracks, that could set up the reactor vessel internal for failure in the event of a shock load.

DR. LAHEY: The reactor vessel internals, is that what you're talking about? Yes, I agree with what they said that, that's what they're monitoring. They're not looking at incipient failure, they're looking at failures that have occurred. And I'm concerned with the mechanism of microcracks, there are plenty of cracks, but they're microcracks, and how this weakens the material in the event that you have an impulsive load on this material.

CHAIRMAN MCDADE: Okay. And it's your view that there's no specific inspection technique currently available that would be able to identify the effects of embrittlement prior to failure and that, in your view, the only way to adequately manage the effect of aging is to have a reasonable replacement or repair system. Am I correct in summarizing --

DR. LAHEY: Yes. I think you are correct. I mean, the problem is, if we had all the data that we really need to have, we wouldn't be having this discussion. We would know what to do, the NRC would be requiring it, Entergy would be complying with it, and everything would be fine. But we don't. We have some fragmentary data, which indicates concerns, and so how do you deal with that? To me, the easiest way to deal with it, for things like bolts, they're relatively easy to replace, would be just replace them. Get rid of the problem rather than try to calculate how many bolts can be failed and still keep running and that sort of thing. I think that's a very dangerous game, you're playing with fire when you do that.

CHAIRMAN MCDADE: Okay. And let me address this in turn to Dr. Hiser and then to Dr. Lott, it's your view, Dr. Hiser, that in the absence of cracking, there is no reason to believe that you are on the verge of failure, even in the event of a shock load, with these reactor vessel internals and that, therefore, the inspections that are currently available are adequate to monitor the aging of these reactor vessel internals? Is that correct?

DR. HISER: This is Allen Hiser. Yes, that is correct.

CHAIRMAN MCDADE: Okay. And, Dr. Lott, do you agree with what Dr. Hiser just affirmed?

DR. LOTT: Yes, I do.

CHAIRMAN MCDADE: Okay. So that's basically the difference of opinion here between the NRC Staff, Entergy, and the position of New York's witness, Dr. Lahey, as you see it Dr. Hiser?

DR. HISER: If I could just -- this is Allen Hiser. If I could just elaborate a little bit because --

CHAIRMAN MCDADE: Please.

DR. HISER: -- some of the discussion yesterday related to ductility of the materials at high fluence and a couple of the exhibits, New York 487 shows data, they're up to about 70 dpa that have measurable fracture toughness, which is indicative of ductility, NRC 209 has data that are up to about 12 dpa, that again show reasonable fracture toughness. So there is still ductility in the material at these fluence areas of interest. And, again, there also is, my understanding, there was no exhibits provided to, no data that we've seen that would indicate that fatigue weakens a material or a component in the absence of cracks.

ADMIN. JUDGE WARDWELL: And what types of fluences do we expect after 60 years of operation?

DR. HISER: My understanding is for the internals on the upwards of 75 dpa for the maximum for the baffle bolts. If you go beyond that area, go beyond the baffle assembly, they're much lower. So the baffle bolts --

ADMIN. JUDGE WARDWELL: But the ones you just quoted were well below that, weren't they? You say it was over --

DR. HISER: The data in New York State 487 is upwards of 60, 70 dpa.

ADMIN. JUDGE WARDWELL: Okay. Thank you.

CHAIRMAN MCDADE: Okay. And, Dr. Hiser, we're talking perhaps about different things. We've been talking about cracks. Dr. Lahey mentioned microcracks, cracks that are there, but are not observable given current inspection techniques. Do you agree that there's probably microcracks in many of these reactor vessel internals, such as the baffle former bolts?

DR. HISER: This is Allen Hiser. As the CUF gets much closer to one, I think the likelihood increases that you could have microcracks. But I think the impact of those microcracks on the fracture response of the component is negligible. I think that's been demonstrated through many tests.

CHAIRMAN MCDADE: So the degradation in fracture toughness would be minimal in your view?

DR. HISER: Well, I think the effect of the degradation of fracture toughness would not be significant. There may be reductions in fracture toughness, but the impact of that in the presence of even microcracks is not significant.

CHAIRMAN MCDADE: In your view, would not be of consequence?

DR. HISER: Yes, that's correct.

CHAIRMAN MCDADE: Dr. Lott, do you agree with that?

DR. LOTT: Yes, and I believe that the data that Dr. Hiser just cited basically shows you that these components can survive with actual cracks, macrocracks, measurable cracks, not just the microcracks that are suggested here. Certainly if we can withstand cracking and we can demonstrate stability of the component with a crack in it, concern about microcracks does not, to my view, seem to be important.

CHAIRMAN MCDADE: Okay. And this is the data in New York State 487 and NRC 209, correct, Dr. Hiser?

DR. HISER: Yes, that's correct.

CHAIRMAN MCDADE: Okay. Dr. Lahey?

DR. LAHEY: All right. I want to try to clear up something we talked about yesterday and here. Dr. Kennedy brought up the question, have I invented new loads, do I need new loads to show that these things can fail or not? And my answer was, no, the existing type of accidents and seismic events are sufficient. What's happened, if you go back in history, is there was a point in time when we were real concerned about design basis LOCA event and the decompression wave that went in and what that would do to the structures. Because in a particular instant, you can have atmospheric, essentially atmospheric pressure on one side and several thousand PSI on the other and that would give you a very large impulsive load.

So codes were written to address that, method of characteristic kind of codes, like the WHAM code that Stan Fabic wrote in Westinghouse, and detailed analysis was done that showed for ductile structures, they can withstand it. And we confirmed that in the Loft experiment, which was an experiment which we ran a simulated loss of coolant accident. And so that concern was mitigated and as a consequence, now people do the analyses using codes such as RELAP and TRAC, those kind of codes, and they really smear out this type of shock. They don't give you the kind of shock loads that you would get if you tracked the wave, the rarefaction waves, throughout the vessel.

So I'm concerned with the real shock loads. It's time, in my view, to go back and take a look at this again with degraded materials. These are weakened materials, you can't have lots of microcracks in there and not have it weakened. It depends on how much weaker it is, that's just due to fatigue. And then irradiation makes it weaker yet, makes it more brittle, more subject to failure. So that's the real difference. That's why when I say shock loads, I may mean something quite different than what they're talking about with shock loads because they're talking about the normal safety analyses using these big system codes, which really are intended to look at the inventory of the liquid and the coolability and that sort of stuff. They're not very good at giving instantaneous loads, either the thermal or pressure loads.

CHAIRMAN MCDADE: Okay. But according to Dr. Hiser and Dr. Lot, even in the situation where you have macrocracks, observable cracks as opposed -- there still would not be sufficient degradation in order to create a real risk, a significant risk of failure. And they rely on, I believe it was, what was it, New York 487, which was an Argonne Lab study from 2010. Somewhat more recent data than the one that you had cited. Does that not alleviate your fears that this material would continue to be sufficiently robust?

DR. LAHEY: That's not what the Argonne

experts are saying. When they give input, they're saying very similar things to what I'm saying. They have the same kind of concerns about the lack of data and the effect of it and what it may imply. And they're, of course, hoping to get funding to run more experiments. And I agree, more experiments are needed.

CHAIRMAN MCDADE: Okay. So you don't disagree with the study, it's just what you take away from that Argonne Laboratory study is different from what Dr. Hiser and Dr. Lott take away from it.

DR. LAHEY: Apparently. I mean, I don't agree with what he said. I think a degraded structure is inherently weak and more subject to failure.

CHAIRMAN MCDADE: Okay. But what we're dealing with here is basically a professional disagreement. We're looking at the same data and you're interpreting it, it creates more concern in your mind as to the potential for failure than what was expressed by Dr. Hiser and Dr. Lott. Is that correct?

DR. LAHEY: I suppose.

CHAIRMAN MCDADE: Well, I'm not trying to -- this is new material for me and I'm trying to make sure that I understand the relative positions here and what the basis of the disagreement is. It seems to me that it is an interpretation of the data, the significance of the data that's available and the data that isn't available. And that you seem to be very concerned with an absence of data that leaves questions in your mind. Am I correct?

DR. LAHEY: Well, not entirely. I'll tell you why I feel the way I do is I'm on a science council for a program called CASL, which is based in Oak Ridge. It's a very large program, which is funded by DoE, to develop computational capability for nuclear reactors. It involves many national labs and many universities. One of the members of the board was also an executive or the person in charge of the Light Water Reactor Sustainability Program, so I've had ample opportunity to talk to the people who are working in that program. And I know from the comments that I have there's people who would like to think, that's just for operating out beyond where we're talking about. We're talking about 60 years, they're going on to 80 years.

But in nobody's mind is there a sharp demarcation. Those are the same concerns, they have the same concerns that I do, and they're working on it. And we're now looking at how to take the code that we've developed under CASL, which is a three dimensional neutronics, thermohydraulics, fuel, water chemistry, crud deposition, everything, code, and apply it to these kind of issues for relicensing of nuclear reactors. So there are certainly things you can do, like the water chemistry, what is the water chemistry during transience and how does that affect FN?

So these are things we'll do in the next -- I think we're going to get to in the next issue. But it's not -- I haven't just made this stuff up. I mean, these are valid concerns by people who are working in it, some of them are even working under NRC funding. So, I think it's an honest professional disagreement and nobody has the perfect data set right now to say, here's the answer. But there's significant concerns about these type of things.

CHAIRMAN MCDADE: Okay. Thank you, Dr. Lahey.

ADMIN. JUDGE KENNEDY: Dr. Lahey, since you responded to my design basis question -- this is Judge Kennedy. I'm just curious, it seems like the scenario that you painted in terms of analysis methodologies would be just as applicable to non-irradiated material shock loading analysis as irradiated material shock loading analysis. Are you -- am I correct in that? Do you have the same concern with non-irradiated material analysis, shock loading, as you do with the irradiated material shock loading analysis?

DR. LAHEY: I didn't explain it very well. Originally, that was the concern and so they were more ductile materials. These were newer reactors and so the concern was, will they maintain a coolable geometry during these type of events? And so, I think that question has been settled and now the issue is, given an intact geometry, can you cool them with emergency core cooling engineered safety systems? The new thing is, now you have degraded structures, which you never had to deal with before, highly degraded both from fatigue and irradiation, and it's time to now relook at that. Because they can be, not only deformed, they can be failed and relocated and then the concern is core coolability.

ADMIN. JUDGE KENNEDY: I guess I thought I understood you to say that there was a problem with the analysis techniques that are being utilized to analyze the shock loadings, that they were deficient. Is that not the case? Is that not what you were trying to say?

DR. LAHEY: The tools that were developed and used previously were focused on tracking depressurization fronts and their effect in terms of loading. They're rather expensive to run method of characteristic type codes, but they give you good answers. Since then, people have concluded that, at least for ductile geometry, the core will stay intact, so given this geometry, the geometry is assumed, now we can use codes that do not do that, they're more control volume kind of codes, like RELAP and TRAC. So when you do the analyses for plants like Indian Point, you don't look at the deformation of the core, you have a certain geometry. What you look at is where's the liquid, how's the cooling, what's the peak clad temperature, that sort of thing, to see if you're in the compliance with the safety regulations.

ADMIN. JUDGE KENNEDY: All right. Thank you.

ADMIN. JUDGE WARDWELL: Getting back to the inspections where we were, do you have any criticisms, Dr. Lahey, of their approach of trying to monitor for effects rather than mechanisms? Strictly this issue, not in regards to your overall issue, but as an expert witness before us, do you agree that monitoring for effects is a good approach rather than trying to monitor for any mechanism?

DR. LAHEY: The answer to that, your honor -- this is Richard Lahey for New York State. As I tried to say yesterday, I think the program that they have in place, which is inspection based, is a pretty good program. So I'm all for it. I just don't think it addresses all the real concerns.

ADMIN. JUDGE WARDWELL: You've said that, thank you. In your testimony, 482, Page 17, Lines 6 through 17, you talk about the rather complex and interacting metal degradation mechanisms associated with fatigue, irradiation, and corrosion interact is still an area of active research. And you point to a DoE, USNR, in conjunction with various other national laboratories that have recently embarked on a program to understand and resolve issues related to these interacting and synergistic effects. And you reference NUREG/CR-7153, which is entitled Expanded Materials Degradation Assessment, or EMDA --

DR. LAHEY: Right.

ADMIN. JUDGE WARDWELL: -- Aging of Core Internals. And in that particular NUREG, do you know the age of the study of how far out they were looking? Was it within the license renewal period or was it beyond that, to your knowledge?

DR. LAHEY: That report is part of the

Light Water Reactor Sustainability Program. So they're looking beyond the design life of 40 years, they're going out farther. They're actually charged with seeing if you can get out to 80 years. But the phenomena has no sharp demarcation at any point in time. So they're looking at it all the way out.

ADMIN. JUDGE WARDWELL: But any conclusions they reach, if they're making them in regards to how it is out to 80, that would be certainly different than if it was out to 60, wouldn't it?

DR. LAHEY: Yes, but if they find phenomena that is of concern at 50, they will communicate that and it will be relevant to what we're talking about here.

ADMIN. JUDGE WARDWELL: And did they have any conclusions from that report that related to out to 50 or 60 years that demonstrate or support your positions?

DR. LAHEY: Dr. Busby is one of the authors of that report and one of the things that I recall from that report was he was greatly concerned about irradiated assisted stress corrosion cracking and what the impact of that may be.

ADMIN. JUDGE WARDWELL: Okay, thank you. You go on, in fact on that same page and extending over to Page 18 with lines 17 through 22 and then 1 through 2 on the top, that the federal government has also embarked on a fairly large research program. And this is what you've termed -- or not you -- yes, you call this Light Water Reactor Sustainability Program, which includes research into whether the different materials and light water reactor components can continue to perform their intended functions during the period of operations. This report that you cite, which is Exhibit New York State 485, was in August of 2014. And, again, what were the periods of years that they were looking at under that particular document?

DR. LAHEY: From now until operating out to 80 years.

ADMIN. JUDGE WARDWELL: Okay, thank you.

DR. LAHEY: I mean, it's being done at various national labs and they put out monthly newsletters on where they stand on various things.

ADMIN. JUDGE WARDWELL: Okay. Thank you. NRC's testimony, 197, Answer 124, Page 75, states that, "Entergy has also implemented a low leakage core design for IP2 and IP3 prior to 30 calendar years of operation, which reduces the potential for irradiation driven aging mechanisms, such as the IASCC, the IE void swelling, and ISR." I guess I'll go to Staff considering this was your exhibit. I knew what all those other acronyms were for, but what about the ISR?

MR. POEHLER: This is Jeffrey Poehler of the Staff. So ISR is irradiation assisted stress relaxation or irradiation stress relaxation.

ADMIN. JUDGE WARDWELL: Irradiation stress relaxation, is that what the ISR is for?

MR. POEHLER: Correct.

ADMIN. JUDGE WARDWELL: Okay, thank you. How is this low leakage core design achieved?

MR. POEHLER: This is Jeffrey Poehler of the Staff. So, basically the core design is such that you have fuel assemblies that have higher, I guess, levels of burn up or depletion or placed around the outside of the periphery of the core so that the newer fuel assemblies are concentrated more towards the middle of the core.

ADMIN. JUDGE WARDWELL: And what does this do for you supposedly?

MR. POEHLER: It reduces the, I guess, the leakage of -- it reduces the fluence levels.

ADMIN. JUDGE WARDWELL: That's what I was wondering. Okay, thank you.

CHAIRMAN MCDADE: I'm sorry. It releases the -- I just didn't hear.

ADMIN. JUDGE WARDWELL: It reduces the fluence --

CHAIRMAN MCDADE: Fluence.

ADMIN. JUDGE WARDWELL -- or the flux actually, so over time the fluence will be less.

ADMIN. JUDGE WARDWELL: Dr. Lahey, do you have any comments on the low leakage core design?

DR. LAHEY: No. No, I understood what they were doing and why. I mean, it's certainly helpful to the concern we talked about yesterday with the core plates and the pressure vessel to try to reduce the fluence.

ADMIN. JUDGE WARDWELL: Okay. Thank you. Entergy's Exhibit 616, testimony, Answer 182 on Page 120, failure of a component without a pre-existing crack is governed by the mechanical properties of the material, the yield strength, the ultimate strength in particular. Irradiation increases the yield and ultimate strengths, and we talked about this. I guess my question for Entergy, does this increase in strength with irradiation occur without any bound? I mean, will it continue on forever, the longer you irradiate it, will it continue to gain strength --

DR. LOTT: Okay. This is Randy --

ADMIN. JUDGE WARDWELL: -- keep on going

unlimited?

DR. LOTT: This is Randy Lott for Entergy. I think it might be clearer if you examine the exhibit I offered this morning about yield stress in material versus fluence. In general, most of the changes that happen to the mechanical properties of the material happen within the first five to ten dpa. So they actually happen early in life and saturate such that there are much less changes. That's true both of the increase in yield stress and the ultimate stress, the decrease in ductility. Most of the action happens, I would say, at less than ten dpa. Which, for your highly irradiated -- ten dpa may be the end of life fluence for some components, for others it's as much as 60, for others it's one. It just depends on where the component is.

ADMIN. JUDGE WARDWELL: And are you referring to all components or just the reactor vessel internals?

DR. LOTT: The internals. The internals are the only ones that are going to see enough neutron exposure to exceed one dpa. We talk about milli-dpa when get out to the reactor pressure vessel.

ADMIN. JUDGE WARDWELL: Okay. Thank you. And recognizing that these microcracks exist, I guess are you claiming those still aren't caused by irradiation or where would those microcracks be occurring from?

DR. LOTT: The microcracks, as I understand the argument, are one of the precursors to crack formation in the fatigue specimens. So, as where it was offered into evidence, and I think this is also explained in NRC NUREG-6909, at least in the draft version, they're sort of stages in the process of forming an observable crack and one of those early stages is the microcracking stage. So, again, yes, it's a form of -- I mean, obviously, if there is a limiting number of cycles, then something must be changing over the course of time. It's been a struggle to identify those things in these materials, but this microcracking and some of these other small micro-structural changes are just the evidence of accumulating fatigue in material.

ADMIN. JUDGE WARDWELL: Okay, thank you. Does irradiation decrease the resistance to the crack propagation?

DR. LOTT: Okay. Let me ask, I want to clarify here. When we talk about crack propagation, are we talking about crack propagation due to loading or crack propagation due to corrosion cracking or crack propagation due to fatigue? There are multiple kinds of ways that, that term might be used.

ADMIN. JUDGE WARDWELL: Answer for each situation.

DR. LOTT: Okay. So, I've created my own question now, didn't I?

(Laughter.)

DR. LOTT: In terms of mechanical properties, we talked and I think Dr. Hiser talked earlier about the fracture mechanics mechanisms that we look at or fracture toughness values that we use. And in general, those toughness values are measured in terms of J-resistance curves. That would tell you effectively how much work it takes to advance a crack in the material mechanically. And the slope of that J-resistance curve gives you an idea of what the, again, the resistance the material is to crack advance. And in general, as the irradiation goes up, that number goes down. So there is some decrease in the resistance. The good news, however, is if you're even measuring resistance, you're into ductile failure, not into the brittle failures we had before.

Fatigue, I think we talked about the effect of radiation on fatigue earlier and it's our contention that where we have relevant data, that data says that the resistance to fatigue initiation is decreased. Fatigue propagation, I think, follows similar kinds of rules, but it is a different mechanism. I think we'd have to go back -- I'd have to do a little more research if you wanted to know about that.

ADMIN. JUDGE WARDWELL: But by different rules, I mean, does irradiation decrease that resistance to crack propagation? Does a crack propagate faster when it's been irradiated under fatigue cracking?

DR. LOTT: Not necessarily. I think we'll get back to the same kind of data that we discussed in terms of the initiation. And I'd have to go back and review that data for you in detail. And, again, we talked about -- so there's fatigue stress corrosion cracking, again, that effectively is irradiation assisted stress corrosion cracking, that's the concern we have is that there will be crack formation and growth due to irradiation.

ADMIN. JUDGE WARDWELL: Okay. Thank you. Dr. Lahey, do you believe that irradiation decreases the resistance to crack propagation once a crack forms from whatever mechanism?

DR. LAHEY: Yes, definitely the crack will propagate faster.

ADMIN. JUDGE WARDWELL: Thank you. Entergy's Exhibit 616, Answer 114, Page 71, says RVIs have no pressure retaining function. A pressurized thermal shock, PTS, transient, therefore, does not subject the RVI components to the sustained membrane stresses characteristics of the effects of a PTS event on a reactor pressure vessel itself. I guess my question to Entergy is, why wouldn't the internals still feel the pressure wave from a PTS transient if one occurred?

DR. LOTT: Well, and again, I'm not sure if someone else from Entergy panel wants to -- I'll step forward first, I guess, and they can help me out as I go along. Effectively, the pressurized thermal shock is a repressurization, it's not necessarily -- it's a long time in developing. It's not, I don't believe, the same kind of process we're talking about here. And one of the key elements of it is that you've cooled down the vessel at a time when the internal pressure on the vessel, the membrane stresses, remain high because you've not seen the depressurization of the system. So, the pressure itself in the vessel creates the baseline high loads that the thermal shock challenges. I'm not sure I've -- perhaps somebody can help me out with my words.

MR. AZEVEDO: Yes, this is Nelson Azevedo from Entergy. What Dr. Lott said was correct, the repressurization during a PTS event is not an instantaneous event like a pipe break, we have a fraction wave traveling through the system. This is a repressurization that takes some time to repressurize the system.

ADMIN. JUDGE WARDWELL: But even so, wouldn't the vessel internals feel however small gradual change there is regardless? And I'm trying to understand your statement that, as I assume you're trying to imply on this Answer 114, is that we don't have to worry about PTS because it's not a pressure retaining -- these aren't pressure retaining components, but they're in and amongst the pressurized area and so why wouldn't it still feel that change, whatever shock that does occur?

MR. AZEVEDO: This is Nelson Azevedo for Entergy. And, yes, that's correct, but the way I visualize at least, for a pressure boundary component, like the reactor vessel, you have 2,200 pounds on the inside and essentially zero on the outside. So you have that whole delta P across the component. When you're talking about a reactor vessel internal, like a column, it's pressurized the same amount all the way around, so there's no pressure differential. That's why they're differentiated between pressure boundary components and vessel internals.

ADMIN. JUDGE WARDWELL: Thank you, that's helpful. And, Dr. Lahey, do you agree with those statements? Any disagreement?

DR. LAHEY: I was somewhat surprised when I got comments on my concerns about thermal shock, significant thermal shocks, because I was not worried about pressurized thermal shocks, that's a pressure vessel phenomena. I was worried about cold water, for example, coming in and some of the internal structures would then be suddenly changed in temperature, it would hit, it would shock the surface, it would try to contract, it would crack, it could fail. That was my concern.

ADMIN. JUDGE WARDWELL: Sure. And I just want to verify that you don't have any disagreement with the pressurized thermal shock specifically that we just discussed now.

DR. LAHEY: No, I have no disagreement with it.

ADMIN. JUDGE WARDWELL: Okay, great.

DR. LAHEY: I wasn't concerned with it.

ADMIN. JUDGE WARDWELL: Just want to make sure they're not trying to spoof me over here with some voodoo.

DR. LAHEY: No.

ADMIN. JUDGE WARDWELL: That's why I'm going to you to see if you can agree with those small points. That's why I go back and forth. But I understand your comments about the other thermal shock.

DR. LAHEY: Okay.

ADMIN. JUDGE WARDWELL: That was a different issue. Thank you. Entergy's Exhibit 616, Question and Answer 185, Page 122. The Question 185 says, do accident loads need to be considered as a contributor to the effects of aging on reactor vessel internals? And the Answer in 185 says, no, aging is a gradual, long-term degradation of a component resulting from sustained environmental conditions, that is applied loads and residual stresses. And then goes on to talk about the ASME codes and Staff Guidance.

While I can understand why the loads aren't a contributor to aging, I just want to make sure that it's clear that I haven't confused this by saying that still those loads, those design basis loads, are considered in your AMP in regards to any evaluation that you might be doing to look at any changes in strength or whatever else occurs during the aging process. Is that correct? And I'll address this to Entergy, I guess, because it was your testimony.

DR. LOTT: This is Randy Lott for Entergy. Yes, what we were trying to say, I think, is exactly what you have said that it does not contribute to the condition that you would observe at any time of the components, but if we're looking at the ability of the components to withstand an accident mode, that's a different question.

ADMIN. JUDGE WARDWELL: You're still --

DR. LOTT: It's not defined as the aging.

ADMIN. JUDGE WARDWELL: You still will consider those loads --

DR. LOTT: Right.

ADMIN. JUDGE WARDWELL: -- design basis loads. And aren't the design basis loads -- do not the design basis loads also include LOCAs?

DR. LOTT: Yes.

ADMIN. JUDGE WARDWELL: Thank you. Entergy's testimony, 616, Answer 185, Page 122, the ASME Code Section 3 compares accident loads such as large break LOCAs and large main steam line breaks to the stress allowables to ensure that the affected components remain capable of performing their intended safety function during and after the event. And so I want to just confirm again with you that doesn't the degraded age strength of the PEO determine these stress allowables in your analysis?

DR. LOTT: I -- go ahead.

MR. AZEVEDO: Yes, this is Nelson Azevedo for Entergy. The stress allowables are directly obtained from the ASME code. So based on what material the component is made out of, the ASME code specifies what the allowables are, we don't get to choose those.

DR. LOTT: Yes, and I think that what you're talking about is basically the design section of the code, right? So it's --

MR. AZEVEDO: Right. So this would be discussed in ASME Section 3, the original analysis, and the analysis has been revised since then.

ADMIN. JUDGE WARDWELL: So isn't it taking advantage of the full virginal strength of this material if it's part of your -- or, I guess not, if it was an allowable.

MR. GRIESBACH: Yes, this is Tim Griesbach for Entergy. You're right. The original Section 3 design process uses ASME minimum yield strength and code allowables in evaluating the margins that are intended to be there. What we've described previously, those tensile strengths and yield strengths may actually increase due to the irradiation and aging process. That's not taken into account, you still use the code required properties of the virgin material when you do those analyses. They're still valid, in fact.

ADMIN. JUDGE WARDWELL: But if you observed a crack and now you're trying to evaluate where you go from there, does this statement not apply to those types of analyses that you may or may not perform as part of your corrective measure for the observation of that crack?

MR. AZEVEDO: Yes, this is Nelson Azevedo for Entergy. The design process, the ASME Section 3 specifically that we're talking about right now, does not allow cracks. So cracks are not allowed during the design phase. If you find cracks, then you evaluate them either under ASME Section 11, which is the operating version of the code, or another NRC approved methodology. But the original design that we're talking about, stress calculations do not allow cracks.

ADMIN. JUDGE WARDWELL: Thank you, that's helpful. Let's move now on to talk a little bit more about these inspections. Yes, Dr. Lahey, would you like to -- I hadn't gotten back to you after telling you why I was getting back to you all the time.

DR. LAHEY: This is Richard Lahey again. I think that was a great question because it really captures a fundamental difference in our view, or my view and the view that they have expressed. I'm concerned with having degraded properties and the ability if you have high enough strain to fail those properties, those components in core. And I don't believe it's adequate just to do a safety analysis using ductile materials. I don't believe it's adequate at all. That's the fundamental difference in our view.

ADMIN. JUDGE WARDWELL: Okay. Thank you.

CHAIRMAN MCDADE: Excuse me. Judge Wardwell, before we move on, I've got a question, perhaps going back a little bit, that I need some clarification on. Maybe a little bit off point, but Dr. Lahey had raised an issue at Pages 45 and 46 of his testimony, about issues at Davis Besse. And in the response, in Answer Number 102, by Entergy, you indicated that -- dismissed it by saying, it does not appear that conditions similar to those which led to the degradation at Davis Besse are present at Indian Point. But there isn't any explanation as to why the conditions are different or how the conditions are different. This is your Answer 102 on Page 59 of your testimony. Anybody from Entergy can address what the differences are between the conditions at Davis Besse and Indian Point and why you believe the concern of Dr. Lahey is unwarranted?

MR. AZEVEDO: Yes, this is Nelson Azevedo for Entergy. The events at Davis Besse, I'm assuming you're talking about the corrosion that the reactor vessel had --

CHAIRMAN MCDADE: Yes.

MR. AZEVEDO: -- at Davis Besse.

CHAIRMAN MCDADE: And I realize that's covered by a different AMP, but --

MR. AZEVEDO: Yes, so the events at Davis Besse occurred because the reactor vessel had penetrations, which are LI600, cracked and they were undetected for a period of time. And that caused leakage onto the reactor vessel head, which eventually corroded the base metal itself. Indian Point does not have this issue and one of the reasons is because we do inspections every outage. We inspect those penetrations every outage and we have not found any cracks to date. So we know there is no leakage going on similar to what happened at Davis Besse.

MR. DOLANSKY: This is Bob Dolansky with Entergy. Additionally, we not only inspect the penetrations for cracking, but we also do what's called a bare metal visual inspection on the top of the head, on the outside surface of the head, where we go around every penetration and look visually and make sure that there's no evidence of any corrosion. So we actually look for where cracking could start inside the head and then we also verify by a bare metal visual inspection that there is in fact no corrosion going on like there was at Davis Besse.

CHAIRMAN MCDADE: Okay. But, Dr. Lahey, am I correct, what your concern was, was the mechanism that caused the cracking as opposed to the failure of inspecting?

DR. LAHEY: Well, I'm concerned about what happens internally and what happens if it's a through-crack. So it's heartening to hear that they do those types of inspections. I think that's very responsible. But there are some welds in the inside that they can't do full inspection of and if they weaken for various degradation mechanisms, like stress corrosion cracking or whatever happens, you can have a concern, particularly for some of the stub tubes associated with the control rod drives. So this is one of the concerns that we had.

CHAIRMAN MCDADE: Okay, but are we concerned about the mechanism that caused the cracking? As I understand it, at Davis Besse, there was an issue that many of these locations were either inaccessible to inspection or very difficult to inspect and that they didn't inspect and that's why the problem was able to reach the level that it had. Here, Entergy is indicating that they do have an inspection program that identifies these potential problems and are able to ensure that there has not been cracking. What I'm concerned with is the mechanism that would have caused the cracking in the first place, not the ability to identify it, but the mechanism that would cause the cracking. And would there be any difference in the situation leading to that mechanism between Davis Besse and Indian Point?

DR. LAHEY: Well, one of -- this is Dr. Lahey again. One of the things in the conclusions in the document that you asked about before, which was authored by Dr. Busby, was that irradiated assisted stress corrosion cracking was still one of the greatest concerns that we have. And in fact, EPRI had put that out as a statement about a year before as well. So it is an issue, I mean, that's the mechanism as I understand it that's of most concern and since you can't do full inspection inside, it remains a concern.

CHAIRMAN MCDADE: Okay. When --

MR. STROSNIDER: This is Jack Strosnider for Entergy, if I could just comment on this. First of all, there could be differences in the susceptibility to cracking based on who manufactured the vessel head and the specific configuration. I can't speak to that without a lot of details, but there could be differences. But I think the important thing to recognize is that GALL says that the potential for cracking does exist and that's why these inspections are done.

So, if you look at the overall framework, the GALL report doesn't say you're not going to see that kind of cracking, it says, it's a potential and you need to go look for it and they're doing two different types of inspections. And, by the way, at Davis Besse, they had several outages of opportunity to identify the problem that was occurring there, but the inspections weren't being done that needed to be done.

CHAIRMAN MCDADE: And all I'm trying to find out is just to make sure I understand, when you use the term, does not appear to have conditions similar, are we talking about only the fact that you have an effective inspection program, and specifically your reactor vessel head penetration inspection AMP, that it is effective versus the one at Davis Besse? Or is there something different about the reactor itself where when you use the term conditions similar which would result in cracking as opposed to just your ability to identify it?

MR. AZEVEDO: Yes, sir. It's Nelson Azevedo for Entergy. There are differences between the way the reactor vessel heads were fabricated for Davis Besse versus Indian Point. Davis Besse's reactor vessel head was made by B&W. Our heads were made by Combustion Engineering. So the process they used to install the penetrations and the way they strength the penetrations, the way the welding was done, was different. Also, the materials were different. They're both LI600, but the Davis Besse materials were B&W tubular products, ours are Huntington alloy materials.

And if you look at the OE, at the operating history for the leaks, essentially all the leaks occurred in heads made by B&W. If you look at Oconee, Davis Besse, most of the other leaks were by B&W heads. Again, ours is different. So they're both LI600, so they're both material susceptible, but I feel that our material is much less susceptible than the Davis Besse heads. Also, Oconee --

CHAIRMAN MCDADE: Excuse me. Why and is there anything here relevant that carries over to the reactor vessel internals, which we're focusing on?

MR. AZEVEDO: Yes, that's what it was going to say. As far as the irradiation on the upper head, there's no -- the fluence is very, very low, less than one times ten to 17. And, no, I don't think there's anything that carries over to the reactor vessel internals.

CHAIRMAN MCDADE: Okay. Dr. Lahey?

DR. LAHEY: Yes, I misspoke, I should have said primary water stress corrosion cracking. He's absolutely correct.

CHAIRMAN MCDADE: Okay, thank you. Dr. Wardwell?

ADMIN. JUDGE WARDWELL: Entergy's testimony, 616, Answer 139, Page 86, states at Table

5-2 of Attachment 2 to NL12-037, and that Attachment 2037 is the Inspection Plan, so this is Table 2 of the Inspection Plan, specifies a required timing of the first inspections and subsequent intervals for the primary components in the reactor vessel internals AMP. For most components, the first planned inspection at Indian Point are scheduled for two refueling outages from the beginning of the PEO, i.e., the Spring of 2016 for IP2 and the Spring of 2019 for IP3.

You go on to state in Answer 142 of Page 92 that the NRC Staff, in a safety evaluation for MRP 227-A, acknowledged the justification for the timing of the initial PEO and subsequent inspections and found the inspection intervals acceptable, and referencing the SE that Staff put out for MRP 227, which I believe is Entergy's Exhibit 230. And I guess I'll start with Entergy. What's your technical basis for justifying not performing the first inspections during the first refueling outage?

DR. LOTT: This is Randy Lott for Entergy. As I think someone said earlier in the day, we did a lot of evaluations for these components and we did not identify any component where degradation, shall I say, fell off the edge of the table. That it was a gradual process, it was a process that seemed to be appropriately managed. We wanted to see that it was integrated into -- that there were baseline conditions set up within the program. And so we thought within the first two years of PEO was appropriate.

It also gave a chance to coordinate some of these examinations with the ASME Section 11 examinations, gave some flexibility to that schedule, which I think is important to the implementation of these exams. It's proven to be -- and we've already done a number of baseline exams and I think we've shown that our number of findings have been extremely low and it seems an appropriate response.

ADMIN. JUDGE WARDWELL: But I guess I don't understand the timing need. I mean, the application was submitted in 2007, so they knew this was coming up. Why did they need more time to get ready for the first inspection besides the first refueling outage after the PEO started?

MR. DOLANSKY: This is Bob Dolansky with Entergy. To perform the MRP 227-A inspections, we'll remove the component called the core barrel. We also do that when we do the ten year ISI inspections. So we wanted to do both of those together. And the ten year ISI inspection, which inspects the reactor vessel itself, the welds that make up the reactor vessel, that's required on a ten year frequency for ASME Section 11. So the reason we're doing it during the second outage of the PEO is to allow us to do both of those together so we only have to remove the core barrel one time.

ADMIN. JUDGE WARDWELL: NRC Staff, would you like to comment on why you are convinced through their justification that the second refueling outage was adequate?

DR. HISER: This is Allen Hiser. The Staff -- I think some of the bases relate to the expectations of degradation and the desire to allow for a higher likelihood that degradation would be detectable. Some of the mechanisms, the analyses by Westinghouse, it led to MRP 227, the degradation that they -- the levels that they were using relate to 60 year fluences and things like that. At 40 years, the fluences aren't to that level. So I think we just thought that it was reasonable to delay the baseline inspections into the PEO.

As Dr. Lott mentioned, plants have done inspections so far, so far there have been very limited, if any, indications of degradation identified at all. So I think that reinforces -- if the inspections that had been done at other plants indicated problems, then I think the NRC would have likely pushed to accelerate the inspections at Indian Point and similar plants.

ADMIN. JUDGE WARDWELL: Back to Entergy. I believe your Table 5-2 and 5-3 of your Inspection Plan shows inspection intervals of ten years for many components. Were there any other shorter intervals incorporated into your Inspection Plans beside the ten year cycle that you can remember?

MR. DOLANSKY: No.

ADMIN. JUDGE WARDWELL: And what's the justification for what seems like a pretty long interval between inspections considering the importance of which you're placing on these inspections?

DR. LOTT: This is Randy Lott for Entergy. Again, there were two issues. One was coordination of schedules because we felt that, that was actually important, it's a fairly difficult operation to remove the core barrel and it's not something people want to do lightly. And I have to point out that it was our feeling that, particularly as an industry, a single inspection gives us information about a wide variety, for instance, the 800 baffle former bolts in the plant, there's factors of three and four in the fluences, so we see lead components, we see a whole range of fluences in a single inspection. As we gather data, that gives us a much broader data base.

And I'll say that we looked at in particular one of the components that had us most concerned were the baffle former bolts themselves. That, as Dr. Hiser has kind of alluded to here, that drove a lot of our inspection schedule decisions. As well, we found because of the assumptions in our analysis and the way that the analysis was put together that the rate of degradation of the baffle former bolts was actually slower in the last part of our irradiations and that it took at least, I think it was 25 effective full power years of operation before we had a reasonable number of predicted failures such that we thought there would be actually something to see. We have experience on baffle former bolt inspections from the 1990s when things were done at lower fluences and very little is seen in any of those plants. So, again, we just felt that it gave us the best opportunity to collect data and that we would get a robust data base from the industry.

MR. STROSNIDER: This is Jack Strosnider for Entergy. I'd also like to add to this and what was mentioned a minute ago that I think it's helpful if you look at this from a fleet perspective. There's a lot of plants doing these inspections at different times and they have similar designs, similar environments. So if something shows up in the operating experience, in accordance with the Program Element 8, Operating Experience portion, that's going to be reviewed and evaluated and if it required a change in the inspection frequency, then they could do that. So there's a lot more data than is coming just from the inspections at this point.

CHAIRMAN MCDADE: Okay. If I can clarify in my own mind here, so the difference between the initiation and propagation of a crack, do you have sufficient data to determine, for example you inspect today, tomorrow a crack is initiated, it's another ten years before you inspect again. At the rate of propagation of that crack, do you have data that would indicate how long it would take after initiation for a crack in the bolt to become problematic?

DR. LOTT: Well, again, our presumption in the bolts was that any bolt that had a crack would fail. So we basically did not do a calculation of crack propagation in the bolts. We basically said, if it's cracked, it's gone. We do have --

CHAIRMAN MCDADE: Well, what I'm getting at, Dr. Lott, is you do the inspection today, you don't see any cracks in a baffle former bolt. You now don't inspect it for another ten years. You don't know when during that period of time a crack may initiate. If it initiates nine years nine months after the inspection, there's no reason to believe there will be significant propagation by the time you inspect the next time.

My question is, is there any way of knowing, for example, if the crack initiates a day, a week, a month after the inspection of whether it is subject to failure within that ten year period before it's inspected again? Is there data that would lead you to believe that the propagation would be at a rate slow enough that it would not be problematic prior to the time of the next inspection? Do you understand where my --

ADMIN. JUDGE WARDWELL: And if it helps you, because this was my question I was going to ask, don't relate it to the bolts, relate it to anything. What's to say that a crack that is just ready to be initiated when you inspected it, but hadn't occurred yet, or at least wasn't large enough to be visible, and the next day, it became visible, you don't inspect it now for ten more years, how do we know there won't be some catastrophic type of result associated with that crack over that ten year interval? What gives you confidence that, that interval is sufficient enough to still be within the range of it being able to maintain its intended function?

DR. LOTT: I guess in response to that, I would point out that in preparation for these examinations, we're working with Entergy to develop inspection acceptance criteria. Those inspection acceptance criteria have built into them, again, how large a crack would be allowable and that includes an allowance for crack growth. So that would be starting with a crack, that means there is an allowable size. I would suggest to you that if a crack were to initiate now, it would be less than that allowable size because there's ten years of growth in the acceptance criteria. So as long as --

ADMIN. JUDGE WARDWELL: But what is the allowable size in your acceptance criteria? I thought it as soon as a crack appears, you have to take some -- that is --

DR. LOTT: Well, we need to --

ADMIN. JUDGE WARDWELL: -- that isn't an acceptance criteria.

DR. LOTT: Well, that would trigger an engineering evaluation. What I'm talking about is the engineering evaluation, the basis for the engineering evaluation.

ADMIN. JUDGE WARDWELL: Yes, but the ten years is passed, you're not doing that evaluation, you're not doing anything on it, but the crack is already there and off and running. And what's to lead you to believe that it will maintain its intended function for ten years as this crack propagates?

DR. LOTT: Well, again, within the evaluation, we started with a finite crack length and allowed it to grow ten years and showed at the end of ten years it would still be acceptable. That's part of the engineering evaluation.

ADMIN. JUDGE WARDWELL: And you know that for all the RVIs that they will still be able to maintain their function as a crack initiates, assuming that the criteria for evaluating it occurs the day after you inspect it, when you didn't see anything, and you do nothing during that period, do you have enough data to comfort yourself that it won't reach some -- it will still be able to maintain its intended function ten years later?

MR. DOLANSKY: I don't think -- this is Bob

Dolansky with Entergy. I don't think I can say right now that I've reviewed and looked at that for every component. I mean, I know that -- what Dr. Lott is saying, Westinghouse is doing work for Entergy right now coming up with acceptance criteria for the inspection. I don't know for every single component what that acceptance criteria is. I do remember for some of the components where there is an acceptance criteria, where there's an acceptable crack length, that it says crack below this length are acceptable. That means that they have done the calculation out to ten more years before we would inspect it again and determined that, that's acceptable. But I can't say right now if -- I'd have to go back and -- I don't think we have -- we don't have the final acceptance criteria at this time. I'd have to go ask Westinghouse if they've --

ADMIN. JUDGE WARDWELL: You've got acceptance criteria in your AMP, don't you?

MR. DOLANSKY: The methodology and -- yes. Acceptance criteria, but not the detailed plant specific where we look at the plant specific loads, the plant specific licensing basis, where they actually did the engineering evaluation. You'd have to ask --

ADMIN. JUDGE WARDWELL: Well, we've got more questions, we'll get into the acceptance criteria in more detail a little bit later. But I guess I'd like to turn to Staff in regards to, what did you review that comforts you that you could accept this program knowing that there's a potential for a crack to occur the day after an inspection was finished and it wouldn't be looked at again for ten more years that it would still maintain its intended function?

DR. HISER: This is Allen Hiser with the NRC. I was not involved in the specific review of MRP 227 and their program is based on 227. In general though -- and there's nothing in the SCR that really describes the basis for acceptability, the ten year reinspection interval. In general, I think there's an expectation that given the knowledge that we have of crack growth rates and things like that, that a flaw that could initiate the day after the inspection as you mentioned, that it would not have sufficient time to propagate to where it could cause a failure of that piece, maybe not even the assembly, but of that individual piece, before the next inspection.

ADMIN. JUDGE WARDWELL: And there's data to support that or is that just based on, because that's what we've doing all along?

DR. HISER: I don't believe it was, that's what we've been doing all along. I believe there was more consideration to it. Fundamentally though, it came, my guess is that it came down to engineering judgment. There is nothing in the SCR that really provides a roadmap to that, but I think given our knowledge of crack growth rates, evaluations that have been done for similar components, for example in BWR internals, that, that was found to be a reasonable reinspection interval to provide reasonable assurance of the integrity of the RVI.

ADMIN. JUDGE WARDWELL: Some of these inspections have been going on as part of your current licensing basis, haven't these, for some of these internals?

DR. HISER: That's correct.

ADMIN. JUDGE WARDWELL: And what's the frequency for those?

DR. HISER: Those would be every ten years.

ADMIN. JUDGE WARDWELL: That's what I thought. With the ten year interval, wouldn't it mean that these inspections basically are going to occur just once over the PEO?

DR. HISER: Well, they would occur twice. One would be, say, at year 43 roughly and then at year

53. So they would occur twice, every ten years during the PEO.

ADMIN. JUDGE WARDWELL: Well, you've got a 20 year PEO period. That's going to be the end of the --

DR. HISER: Right. So --

ADMIN. JUDGE WARDWELL: -- 53 is going to be the end, you're going to shut down then. You're going to inspect them as you shut down and tear it down?

DR. HISER: Year 43 would be the first inspection under this program. Year 53, approximately, would be the second inspection. So it would be two times during the PEO.

MR. DOLANSKY: This is Bob Dolansky with Entergy. Just to clarify, the first inspection would be within two outages of the beginning of the PEO.

ADMIN. JUDGE WARDWELL: Right.

MR. DOLANSKY: That's the first. And then the second one --

ADMIN. JUDGE WARDWELL: Oh, that's what you're -- I'm calling the first -- that's what I call a baseline inspection. The first subsequent interval inspection will occur --

MR. DOLANSKY: Ten years.

ADMIN. JUDGE WARDWELL: -- 53 and then, so you have only have one interval inspection is what I'm saying beyond this baseline inspection.

MR. DOLANSKY: Okay. We count the baseline as the first. We say that we're doing the MRP 227-A exam in Spring of 2016 at Indian Point 2.

ADMIN. JUDGE WARDWELL: That's your baseline, correct?

MR. DOLANSKY: Yes, but we --

ADMIN. JUDGE WARDWELL: Right.

MR. DOLANSKY: We treat that as our first 227-A exam.

ADMIN. JUDGE WARDWELL: Fine. Semantics and that's -- I understand the difference between our discussion. Okay.

CHAIRMAN MCDADE: Okay. And I understand the, I think, genesis for the ten year period of the difficulties and what is involved in the inspection. And trying to become sanguine that the ten year inspection is adequate. And I was using the term crack propagation, Dr. Hiser, you used the term crack growth rates. Where do you look to determine the anticipated crack growth rates for these kinds of materials? To satisfy you that a ten year period is adequate?

MR. POEHLER: This is Jeffrey Poehler of the Staff. So MRP 227-A has guidance on crack growth rates to be used for engineering evaluations and that's in Chapter 6 of that report. Also, the industry plans to use WCAP-17096, which is under review by the Staff and we mentioned that yesterday. But that provides methodologies for engineering evaluations when degradation is found. And that report gives guidance on which crack growth rates to use. But they are referencing industry accepted crack growth rates that have been developed for -- well, I probably shouldn't say because it's proprietary at this point.

CHAIRMAN MCDADE: Okay. But basically, as I understand what you're saying, is that you look right now to MRP 227, the crack growth rates that are projected there. That your review suggests those are valid and you're willing to rely on those crack growth rates --

MR. POEHLER: Right.

CHAIRMAN MCDADE: -- and your review of the validity of those crack growth rates in determining that a ten year period is sufficient to ensure that these reactor vessel internals will maintain their intended function.

MR. POEHLER: The Staff believes or it's our opinion that the crack growth rates that they're recommending would be conservative for pressurized water reactors, given the --

CHAIRMAN MCDADE: Would be or are?

MR. POEHLER: Are.

CHAIRMAN MCDADE: Okay. Okay, Dr. Lahey, do you have some input here with regard to crack growth rates and MRP 227 and the validity of the ten year period?

DR. LAHEY: Yes. Your honor, we have the same concern as you expressed in your questions. Two inputs that I would give to this discussion is there's been publications which indicate the baffle former bolts, when you do ultrasound inspections, you're unable to detect up to or below 30 percent through-crack. So that means you could already start out with a significant weakened bolt at that point. And I want to remind you that on top of all of this, this is an inspection based program and I agree with the concerns that you have raised, but on top of this, at any time during this event, if you have some of these highly degraded structures subject to shock loads, you can fail them. That's the concern that I have. Because, as you know, I'm focused, the bottom line is on the safety of the plant. And the safety of the plant means to be me core coolability.

CHAIRMAN MCDADE: It means core --

DR. LAHEY: Coolability.

CHAIRMAN MCDADE: -- coolability? Coolability, you're saying?

DR. LAHEY: Yes. Once you lose an intact geometry, all bets are off as to core coolability. And by far the most vulnerable reactor vessel internals are these baffle bolts.

CHAIRMAN MCDADE: Which of course Entergy and the Staff represent can have a 50 percent failure rate without impacting the integrity.

DR. LAHEY: Well, my understanding of how that conclusion was drawn was really based on the kind of loads you get during steady state operation and the redundancy to hold them in place. But during accident loads, if you have significant loads, you can unzip the rest of your bolts. That's what I'm worried about.

CHAIRMAN MCDADE: Okay.

MR. STROSNIDER: This is Jack Strosnider for Entergy. I just should comment that those bolting patterns that were analyzed in the WCAP, and I don't remember the exact number right now, that they're based on accident loads.

DR. HISER: This is Allen Hiser with the Staff.

ADMIN. JUDGE WARDWELL: What was the --

DR. HISER: I can fill in. Entergy 654 and 655 are exhibits that are, at least in one case, is an NRC approved report that looks at the bolting patterns under accident loads.

ADMIN. JUDGE WARDWELL: Okay. Well, we'll talk about baffle former bolts in more detail this afternoon too. But before that, I do want to ask the Staff in regards to what's their understanding of the current acceptance criteria that's in the AMP as it stands now?

DR. HISER: This is Allen Hiser with the Staff. The acceptance criteria are provided in the RVI Inspection Plan for Indian Point. I think there maybe was some confusion with some of the discussion earlier. The acceptance criteria that are under development by Westinghouse would be an engineering analysis option under corrective actions. So it's -- maybe if we say inspection acceptance criteria are in the Inspection Plan and they are definitive.

ADMIN. JUDGE WARDWELL: And what do they state there?

DR. HISER: I think in general for cracking, it's no cracking. Any identified cracking is subject to corrective actions.

ADMIN. JUDGE WARDWELL: Thank you.

CHAIRMAN MCDADE: Okay. Let me -- Dr. Hiser, mentioning that any identifiable cracks, we've talked quite a bit over this morning and yesterday about microcracks. The existence of microcracks that could be present even during an inspection, but not identified. Does the possibility of those microcracks affect your consideration of this, of the frequency of inspection?

DR. HISER: No, I don't believe so. Because a microcrack would be subsumed under the inspection -- or depending on the size of the microcrack. I mean, microcracks, if they're below the inspectability limit of the NDE method would not -- clearly you would not be able to detect those. And I don't believe that microcracks would have a significant impact on the integrity of the RVI components, sort of as a starting point. So with the analyses --

CHAIRMAN MCDADE: But isn't the microcrack basically the crack initiation and then you have the propagation. So the question is, being able to identify where you are on that spectrum at the time of the inspection. The clearer picture you have, the better way to follow it. Now, you made reference to the crack growth rates in MRP 227, does that take into consideration that at the time of the inspection there may well be significant, and perhaps maybe not use the word significant, but microcracks that are approaching visibility, but not yet visible?

DR. HISER: Yes. This is Allen Hiser of the Staff. When the report was reviewed for acceptability, my expectation is that the consideration was along the lines of what kinds of flaws could be missed by the inspection given the knowledge that we have of crack growth rates. Was it likely that there would be a challenge to functionality at the end of the ten year reinspection interval? Based on that analysis, be it -- and my guess is this may have been an engineering judgment based that there was not thought to be a significant concern. There was reasonable assurance that the RVI would still maintain their functionality.

CHAIRMAN MCDADE: Okay. Thank you, Dr. Hiser.

ADMIN. JUDGE WARDWELL: Entergy's testimony, 616, Answer 209, Page 140. The results of the IPEC inspections will be available to the NRC Staff for onsite inspection. Are these results also publically available?

MR. DOLANSKY: This is Bob Dolansky for Entergy. These inspection results are put into a report and supplied to EPRI for the whole industry and EPRI rolls that up. I don't know if that's publically available. I don't think it's publically available, but it's available to the industry for sure.

ADMIN. JUDGE WARDWELL: So publically it's not available?

MR. DOLANSKY: I don't believe so.

ADMIN. JUDGE WARDWELL: Okay, thank you.

MR. POEHLER: And, your honor, I would just like to add to that. And that report is also submitted to the NRC Staff for our information. So, that's in the MRP 227 implementing process that they will do that. So we will get a chance to review what the operating experience has been and if there's been any trends in failures, then we'll know about it.

ADMIN. JUDGE WARDWELL: And is that available to the public or is it considered proprietary information?

MR. POEHLER: I don't know the answer to that offhand, I can check.

ADMIN. JUDGE WARDWELL: Do that, would you please? And I think now would be a good time to take a break.

CHAIRMAN MCDADE: Okay. Would anybody need more than ten minutes? Okay, it's now -- why don't we come back at 10:40.

(Whereupon, the above-entitled matter went off the record at 10:26 a.m. and resumed at 10:44 a.m.)

CHAIRMAN MCDADE: We're back on the record.

ADMIN. JUDGE WARDWELL: Let's turn to some inspection actions now. NRC's Exhibit 197, testimony, Answer 80, Page 51, says inspection techniques include ultrasonic UT testing, EVT1 enhanced visual examinations, and VT3 visual examinations. And so, considering it's NRC's testimony I'm referring to, I'll let them answer. Could you explain each type of test and its applicability for the various reactor vessel internals?

MR. POEHLER: Okay. This is Jeffrey Poehler of the Staff. So, just to clarify the question, you want us to explain the applicability of each type of inspection technique?

ADMIN. JUDGE WARDWELL: I want you to explain what's a UT test, what's an EVT1 test, and what's the VT3 test.

MRP: So UT test is an ultrasonic examination. And it's basically using sound to detect cracks in a material. Do you want a detailed explanation or --

ADMIN. JUDGE WARDWELL: How do they show up? How does a crack show up? So, you put an --

MR. POEHLER: So you have a transducer that generates ultrasonic sound. It has to be in contact with the material and it puts sound into the material and you get echos back from the material basically that are detected by either the same transducer or a separate transducer. And those are processed electronically such that you get a signal. So if you have a discontinuity, like let's say in a bolt, you have a partially cracked bolt, that's going to reflect the sound back and be detected and it'll be processed by the electronics such that you can determine the location of that discontinuity.

And with certain ultrasonic techniques, you can create images, you can image the discontinuity. So ultrasonic is used throughout the nuclear industry for piping weld exams to detect cracking and also in vessels, in large structural welds. So it's a very established technique for welds and it's also for bolting, such as baffle former bolts, that's been used -- it was first used back in the 1990s when baffle former bolt cracking was first detected.

ADMIN. JUDGE WARDWELL: So that's a type of test that's used for the baffle former bolts?

MR. POEHLER: It's a pretty well established technique for bolts.

ADMIN. JUDGE WARDWELL: So is it used for the clevis bolts that we'll talk about later also?

MR. POEHLER: It is not specified for the clevis bolts at this time. It could be, but it's not --

ADMIN. JUDGE WARDWELL: Okay.

MR. POEHLER: -- has been determined not to be necessary.

ADMIN. JUDGE WARDWELL: So is the baffle former bolts the only one it's used for in regards to reactor vessel internals?

MR. POEHLER: The only component in a Westinghouse design reactor internals that it's currently used for is the baffle former bolts.

ADMIN. JUDGE WARDWELL: Okay.

MR. POEHLER: You also asked about visual examination.

ADMIN. JUDGE WARDWELL: How about the enhanced visual examination, EVT1, what is that composed of?

MR. POEHLER: Well, so enhanced --

ADMIN. JUDGE WARDWELL: Well, let me ask one other question about the UT. So is this a sensor that you put in at the time you do the inspection or are these permanently mounted so you can turn it on? Or how are the mechanics of this, the logistics of this achieved?

MR. POEHLER: Right, it's a sensor that has to be put in. It's put in for the time of the inspection and there's special tooling to access the bolting. And so it's not permanent.

ADMIN. JUDGE WARDWELL: Okay, thank you.

CHAIRMAN MCDADE: What percentage of the baffle former bolts are accessible to the UT inspection?

MR. POEHLER: For the UT inspection, essentially 100 percent of them are accessible. There are minimum coverage requirements in MRP 227-A, so you can credit 100 percent -- you can take credit for an examination if you exam 75 percent of a population in general. And that would include both the accessible and inaccessible members of the population. But, to my knowledge, I don't think there are any major obstructions that would prevent you from accessing all the baffle former bolts.

CHAIRMAN MCDADE: To Entergy, are 100 percent of the baffle former bolts accessible to the ultrasound inspection?

MR. DOLANSKY: Yes. This is Bob Dolansky with Entergy. We believe 100 percent will be accessible. We won't know for sure until we actually get out there and do the exam, but based on all of our drawing reviews and so forth, we expect to get 100 percent of all the bolts and we expect, there's 832 of them, we expect all 832 to be accessible.

CHAIRMAN MCDADE: So unless somebody changed the design since you last looked at it?

MR. DOLANSKY: Well, there could -- in theory, there could be -- let me give a little bit of background. The technique to perform this inspection is a submarine that goes down underwater, it's done underwater. It basically docks up against the baffle plate and there's a special head that goes in and inspects the bolt. So although all the bolts should be accessible, we're not 100 percent sure we can reach every one with this tooling technique. So that's why there's a possibility that we wouldn't get every one.

But other Westinghouse plants have got 100 percent, we expect to get 100 percent, but there could be some interference that would limit us, but only a very small number.

CHAIRMAN MCDADE: Okay. Thank you.

ADMIN. JUDGE WARDWELL: Enhanced visual examinations?

MR. POEHLER: Yes. This is Jeffery Poehler of the Staff. So enhanced visual examinations are specified for examining welds for cracking. Or anytime you're looking for cracking specifically in MRP 227-A. For example, the core valve girth welds in a Westinghouse design, reactor internals, you use enhanced visual testing. And that's a visual examination that has a fairly stringent detection or resolution requirement. So you have to be able to -- the way that they test this in situ is that they have to be able to identify a character, like a letter, letter C or A or O, and they have to identify that letter and the letter has a 0.04 inch size on the card. So it's a pretty small letter. So that's how they determine the resolution is adequate in the environment that you're going to do the testing.

ADMIN. JUDGE WARDWELL: And what's this card?

MR. POEHLER: It's like a plaque --

ADMIN. JUDGE WARDWELL: A plaque, a little plaque?

MR. POEHLER: Yes, it's a --

ADMIN. JUDGE WARDWELL: Okay.

MR. POEHLER: It's something that -- yes.

ADMIN. JUDGE WARDWELL: Is it on each internal or is it just a test plaque for you to start off to verify that you've got enhanced --

MR. POEHLER: It's just a test plaque. You could -- analogous to calibration standard.

ADMIN. JUDGE WARDWELL: Calibration coupon --

MR. POEHLER: Right.

ADMIN. JUDGE WARDWELL: -- shall we say?

MR. POEHLER: Right. And so then you have to -- so that's basically how that's qualified. And you would do that before you start the examination.

ADMIN. JUDGE WARDWELL: This is a camera that you put --

MR. POEHLER: Yes, remote --

ADMIN. JUDGE WARDWELL: -- in there, it's not someone's eyeball that you're calibrating.

MR. POEHLER: I think they used to -- the submarine type delivery system, or I think for the core valve girth weld, they have special tooling to go up under the thermal shield to deliver the camera. So those are used for when you're specifically looking for cracks such as stress corrosion cracks, IASCC, irradiation assisted stress corrosion cracks, in welds or other components. Other than bolting, it would not be used for bolting because of the location. Where the cracks are would be at the head, the shank area in the bolt, which you're not going to be able to look at side-on. So that's why ultrasonic is used for those bolts. So you also asked about --

ADMIN. JUDGE WARDWELL: VT3, visual examination.

MR. POEHLER: So VT3 is another visual examination technique. The main difference between VT3 and EVT1 is that it is -- VT3 has a slightly lower resolution requirement and it's used for general mechanical and structural conditions. So you're looking for gross failure, like broken components, broken bolts, and other distortion in structures such as it's used for the baffle former assembly to look for effects of void swelling. But the VT3 visual is only specified in MRP 227-A when it's for either a redundant population of components or components that have been changed to be highly flaw tolerant, such that they could tolerate cracks. So that's VT3.

ADMIN. JUDGE WARDWELL: And VT3 implies there's a VT1 and a VT2 and where did they go and are they ever used at Indian Point?

MR. POEHLER: Yes. VT1 is not used for, and Entergy may correct me, but I don't believe it's used for any of the Westinghouse RVI exams. But in some other designs it's used for looking for gaps. But VT1 is also a more -- would have a higher resolution requirement than a VT3. But it's an examination that's called out in ASME code, so it's defined in there. But it's not used very much for the internals. And VT2 is another type of visual examination where you specifically look for leakage of pressure boundary components. So that's not applicable and they don't use that for reactor internals.

ADMIN. JUDGE WARDWELL: In New York State's testimony on 482 at 62, lines 3 through 8, Dr. Lahey criticizes the use of VT3 in visual inspections as inadequate for use in inspections for cracking, stating that there are significant shortcomings of this technique to detect material cracking, degradation, or wear prior to failure, as illustrated by the visual detection of only seven out of 29 fractured clevis insert bolts at the Westinghouse PWR in 2010. Why I don't I turn to -- well, I'll stay with you. So, why do we have any confidence in VT3 for any components and what components beside the clevis bolts is this technique used and what's it's track record for the other components?

MR. POEHLER: Well, for example, it's used for the baffle former assembly, for the general examination for the assembly for distortion due to void swelling. So that wouldn't be something you would use -- it's the most appropriate for looking for that type of aging deformation.

CHAIRMAN MCDADE: Okay. Well, let me jump in here a second and to Entergy, in your Exhibit 616 at Page 87, your Table 1, you list the various items to be inspected and how you're going to be inspecting them. Is there any difference in the, for lack of a better phrase, degree of difficulty in the EVT1 and the VT3? I mean, it appears that the EVT1 gives you greater resolution, more information. Is there any reason why you use the VT3 for certain items, like baffle edge bolts, as opposed to the EVT1?

MR. DOLANSKY: This is Bob Dolansky with Entergy. An EVT1 is looking for cracking. A VT3 is not looking for cracking. So, for example, the baffle edge bolts, if the cracking is down in the bolt, a visual inspection is not going to see it. So if you do an EVT1 on the baffle edge bolts, it's not going to give you any more information than a VT3 would and it is in fact a more difficult exam to do because you have -- the character requirements are more stringent and --

CHAIRMAN MCDADE: I'm sorry, I just didn't hear. The what requirements?

MR. DOLANSKY: The character card requirements are more stringent and you have to inspect at a certain speed. You typically use video enhancement to do that inspection. So, for the baffle edge bolts, since you can't see the area where they would be cracking anyway, an EVT1 doesn't buy you anything. What a VT3 tells you about baffle edge bolts is that if you look at all the edge bolts, if you saw that there was -- the two plates where they go together are shifted or moved, a VT3 is very good for that. You're not --

ADMIN. JUDGE WARDWELL: But I thought you used the UT for baffle bolts.

MR. DOLANSKY: For baffle former bolts, a UT is used.

ADMIN. JUDGE WARDWELL: Yes.

MR. DOLANSKY: But a baffle edge bolt --

ADMIN. JUDGE WARDWELL: Oh, okay, sorry.

MR. DOLANSKY: -- it's not.

ADMIN. JUDGE WARDWELL: Sorry.

MR. DOLANSKY: A VT3 is used for the baffle edge bolts. Did that answer your questions, your honor --

ADMIN. JUDGE WARDWELL: Yes.

MR. DOLANSKY: -- about the differences?

MR. AZEVEDO: Your honor, this is Nelson Azevedo. Maybe I can add a little bit to that. A VT1 is done up close. So, for example, if you're looking for deformation of a component, see if a component is bent, if you're up real close to a component, you may not be able to see if the component is bent. So you actually, if you step back, you can be further from the component, in a lot of cases, you actually get a better assessment of what the component is experiencing rather than be within a couple inches of the component. So that's another reason why sometimes VT3 are used versus VT1.

CHAIRMAN MCDADE: Sort of why I take my glasses off to thread a needle?

MR. AZEVEDO: If that's the example you want to use.

CHAIRMAN MCDADE: Do you thread needles?

(Laughter.)

MR. AZEVEDO: Occasionally.

CHAIRMAN MCDADE: Sorry. Staff, do you thread needles?

(Laughter.)

ADMIN. JUDGE WARDWELL: Back to the clevis insert bolts, and I'll direct this to, I guess I'll stay with Entergy, it doesn't seem like the seven out of 29 fractured clevis insert bolts that were detected by this process is a very high percentage of success. Well, let me rephrase that, you look confused. New York State claimed that only seven out of 29 fractured clevis insert bolts were detected in 2010 at a Westinghouse PWR. Are you aware of that and, if so, why is VT3 a successful inspection technique for seven out of 29?

MR. DOLANSKY: This is Bob Dolansky with Entergy. We definitely were aware of that. There was a Technical Bulletin that was issued as a result of the clevis insert bolting. I would say that the VT3 is what initially found it at the other plant. I think it was sufficient, they did find it, they didn't find every one, but it showed them that there was something going on and then they did additional exams to determine that. To me, with the clevis insert bolts, and Dr. Lott can step in and help me with this, but the clevis insert bolts don't have any safety function. So there's no -- the fact that they cracked and that they detected that with VT3 shows that, as part of the Section 11 program, shows that, that is in fact working in my estimation. Do you want to add anything about that?

DR. LOTT: I don't think we ever believed or contended that the clevis insert bolts, the VT3 inspection would identify failed clevis insert bolts. What we were worried about, as Mr. Dolansky said, was that the location and the securing of the clevis in the log on the reactor pressure vessel -- and the Technical Bulletin that he referred to, which is Westinghouse Technical Bulletin 14-5, is ENT Exhibit 656, just for the record.

And we looked very closely at the clevis insert bolts in that case and made additional recommendations about inspections that would help us determine what we thought was the key feature, which was the clevis being still in place. Once the reactor is loaded, the lower core plate is in, the clevis is effectively locked in place and the safety function of the clevis, as long as it's there when you start up, there's no place for it to go when you're basically running the reactor. We can go through this in more detail if you want to, but we'd probably have to pull up some of the diagrams.

ADMIN. JUDGE WARDWELL: Yes. I think I at least have the picture of that. How did you determine that eventually there was 29 -- or how did Westinghouse determine that there were 29 fractured bolts, not just the seven that were detected by the VT3?

DR. LOTT: Well, Westinghouse's involvement in this program, we had advised the utility that, when they observed the first, or you told me the number seven, broken bolts, advised them that they should think about replacing the bolts, not because of a safety concern, but our concern was that if that were to become dislodged in the refueling cycle, it would be extremely expensive and difficult to put it back in place, it's a precision fit part. So therefore --

ADMIN. JUDGE WARDWELL: This is the clevis itself, the insert itself?

DR. LOTT: The clevis itself. The clevis insert is specifically machined to match the keys in the core barrel.

ADMIN. JUDGE WARDWELL: And if I understand it correctly, these bolts merely hold that in place until that superstructure or whatever you call it, the --

DR. LOTT: The core barrel.

ADMIN. JUDGE WARDWELL: -- the core barrel coming in and --

DR. LOTT: Yes, once the core barrel is engaged --

ADMIN. JUDGE WARDWELL: -- resting on that, which --

DR. LOTT: -- in the key, it's effectively restrained by multiple factors that we can go into if you want to. And so, therefore, we determined it wasn't a safety concern, but that it might be in the best interest of the utility to replace those bolts because if it were to come loose, it would be incredibly difficult to replace.

ADMIN. JUDGE WARDWELL: Are either the -- any of these bolts, whether it's the clevis or the baffle edge or the baffle former bolts, if they fail, do they become dislodged and have a potential to impact the geometry, if you will, and the coolability of the core itself? Or a function of any of the other vessel internals?

DR. LOTT: Well, one of the things and one of the advantages of the VT3 examination is that it allows you to determine whether the lock bars, which hold the heads of all the bolts in place, lock bars that are locked across the heads, are in place. And as long as those lock bars are in place, there's no way for the bolt to escape. In particular, the clevis insert bolts are, again, geometrically constrained. Once they're in place, there's no way for that head to escape from this very small tight dimensions of the object.

ADMIN. JUDGE WARDWELL: Well, couldn't the bolt itself crack and fall away from the head and go down below?

DR. LOTT: Well, it's threaded in and it can't -- there's nowhere for it to go except for out past the head.

ADMIN. JUDGE WARDWELL: Okay, thank you. While we're on it, Dr. Lahey, would you agree with the statements that were made in regards to the clevis bolt inserts and do you have comments about the seven out of 29?

DR. LAHEY: Well, our concern was it was not a very effective non-destructive testing technique and the accuracy of it wasn't very effective. I'm not very concerned about the safety aspects of the clevis bolts, but the technique itself had some deficiencies.

ADMIN. JUDGE WARDWELL: Thank you.

CHAIRMAN MCDADE: Okay. If I could, just to clarify. We've talked about these clevis bolts and not having a safety consideration. If you could elaborate for me, as I understand from that, Dr. Lott, you referred to the Westinghouse Technical Bulletin, which was Entergy Exhibit 656, and it talks about maintenance risk due to the complexity and cost of repair and the required level of contingency planning. That inability to remove the core barrel or need to replace the insert to reestablish customized design gaps. Could you explain to me, what really -- elaborate a bit on the implications of a failure of these clevis bolts?

DR. LOTT: Sure. The clevis inserts themselves are actually, they're positioned -- they locate these keys and they're relatively high tolerance components, such that if it -- when the reactor is actually built, they're precision machined to match. And then they're shrunk fit into the lugs in the component, so they're frozen, put in -- so they're a tight fit. That's one of the reason they don't come out is because they're shrunk fit in place. And in order to do that, to repeat that hand operation in the plant and relocate it into the exact location where it was in the first place if the bolts were to come loose and you wanted to even put the same one back in, much less replacement, is a very difficult job. So if we were to lose it, it would be hard to replace. Is that --

MR. DOLANSKY: Dr. Lott, can I just add something?

CHAIRMAN MCDADE: Hard to or impossible and then --

DR. LOTT: I don't think I would say impossible to replace, but, again, this was a maintenance concern, not a safety concern. So it was really a matter of what did it make sense for the utility to do at that point?

MR. DOLANSKY: This is Bob Dolansky with Entergy. Just to try to clarify a little bit, when you refuel the reactor vessel, basically you do everything remotely, so you're not unbolting things. Basically, everything slides together and slides apart. So when you're removing the core barrel, the clevis insert basically is like a key and there's a key on the core barrel that just slides in there. Once it's in there, there's no safety function, it has no safety function.

But if I as the owner am planning to refuel the reactor vessel and remove the core barrel and that clevis insert moves, then I cannot get that back in. That's very expensive for me to then try to get that fixed, just so I can get it back in, it has no safety function, but just to be able to put it back together. So that's why it's a commercial issue and not a safety issue. And that's why what Dr. Lott is saying is, if three out of six bolts or four out of six bolts failed, but two are still holding it, I can get those four that are failed out and put new ones in without that thing shifting. Once it shifts, to try to get it lined up exactly again, underwater, would be very, very difficult. It could be done, absolutely could be done and would be done, but it just would be much more difficult. So for us, it makes much more sense to try to get those things replaced before it causes us a commercial issue. Does that help?

CHAIRMAN MCDADE: Yes. So basically what you're saying is, it's not an inability to remove the core barrel, it just makes it a much -- the degree of difficulty increases significantly?

MR. DOLANSKY: Getting it out or putting it back together. Putting it back together, if it's cocked, it wouldn't slide, it's a very close machine tolerance. The core barrel may not slide back into the key if it were totally -- if it had moved.

CHAIRMAN MCDADE: Okay. Thank you.

ADMIN. JUDGE WARDWELL: Entergy's testimony, 616, Answer 188, Page 124, states that the basis for the adequacy of these inspection techniques is described in the companion document to MRP 227-A, which is MRP 228. This document describes the standards to be met by each specific examination method. The standards in MRP 228 reflect the latest information and regulatory documents, such as NUREG/CR-6943, which addresses visual examinations, including remote visual examinations, and describes the characteristics of flaws to be detected in nuclear reactor components. In particular, such critical characteristics as the crack opening displacement. And I guess I just want to verify, is 228 an exhibit in this proceeding, Entergy, as far as you know?

DR. LOTT: I believe it is.

MR. DOLANSKY: I believe it is.

ADMIN. JUDGE WARDWELL: Okay, thank you.

MR. KUYLER: Your honor? MRP 228 is Exhibit Entergy 645.

ADMIN. JUDGE WARDWELL: Thank you. I guess I'll ask Dr. Lahey, have you provided any evidence challenging these details on the basis for the development of Entergy's Inspection Plan or that Entergy's Inspection Plan does not meet all the requirements of MRP 227?

DR. LAHEY: Are you talking about other than the inability to detect the clevis bolts?

ADMIN. JUDGE WARDWELL: Right. Just --

DR. LAHEY: I would have to look back, I don't know off hand.

ADMIN. JUDGE WARDWELL: Okay. Thank you.

CHAIRMAN MCDADE: Excuse me one second, and perhaps I just got confused. Did you say, MRP 227?

ADMIN. JUDGE WARDWELL: Yes.

CHAIRMAN MCDADE: Okay. Because 645 is EPRI 228.

ADMIN. JUDGE WARDWELL: Right. That was my first question.

CHAIRMAN MCDADE: Thank you. I'm just -- thank you.

ADMIN. JUDGE WARDWELL: Okay. We switched MRPs on you. NRC 197, testimony, Answer 183, Page 105, states that we disagree with Mr. Lahey's concern regarding synergistic effects because, one, primary components are to be inspected under Entergy's program are those components which are most likely to be effected by synergistic effects, if they exist. Two, that Entergy's AMP inspects for and will detect cracking whether a single aging mechanism or multiple or synergistic aging mechanisms contribute to cracking.

Three, embrittlement alone will not cause failure without the presence of a crack and the inspections performed by Entergy's AMP are sufficient to detect cracking. Four, Dr. Lahey has not identified any tests or operating experience which demonstrates that synergistic effects are significant for PWR RVIs and existing laboratory test data on synergistic effects is inconclusive. And, five, the industry reactor vessels internal program in which Entergy is participating is a living program which shares operating experience among all PWR licensees and, thus, any occurrence of an unexplained or accelerated degradation due to synergistic effects will be identified and adjustments to the industry guidance and the Entergy AMP will be made to ensure continued integrity of the RVI across the fleet.

I think with this statement, I'd like to start off fixing one other point again. And this is this statement that embrittlement will not occur without a crack. And I'll address this to Entergy.

Does this statement mean that embrittlement won't have any affect until cracking is caused by some other mechanism or is this a statement that says that embrittlement won't alone cause failure without a crack? Meaning that embrittlement can cause a crack, but it won't result in failure unless there is a crack demonstrated from the embrittlement?

DR. LOTT: This is Randy Lott for Entergy. The statement, I believe, is saying that if a material is embrittled, that fact alone will not result in the failure of the component, it needs to be combined with a crack and a load that would challenge the stability of the component.

ADMIN. JUDGE WARDWELL: With embrittlement, can embrittlement eventually cause a crack in and of itself?

DR. LOTT: No. It's not identified as one of the crack causing mechanisms in the component.

ADMIN. JUDGE WARDWELL: What is the effect of embrittlement then?

DR. LOTT: What is the effect? The change in the -- for me, the discussion of embrittlement is a discussion of the change in properties that happen when the material is irradiated. And we've listed them several times before, it's the increase in yield stress, the decrease in ductility, the decrease in fracture toughness, those combined effects.

ADMIN. JUDGE WARDWELL: Is the resulting effect?

DR. LOTT: Those are -- yes. If I had a test specimen, those are measurable, objective things I could measure.

ADMIN. JUDGE WARDWELL: So regardless --

DR. LOTT: I can't do that test non-destructively.

ADMIN. JUDGE WARDWELL: So regardless of how long or how much fluence there is at a location, a piece of metal there will not exhibit a crack due to just embrittlement causes? Let's say you have a test coupon and --

DR. LOTT: If I had a test coupon and I were to just -- we certainly irradiated lots of specimens to the fluences that we've talked about lots. We've irradiated specimens to the fluences that we've talked about here, the 60, 70 dpa, and they do not just spontaneously crack.

ADMIN. JUDGE WARDWELL: Okay.

DR. HISER: And, your honor, this is Allen Hiser of the Staff. I just want to make sure, the Staff did not imply that embrittlement can cause cracking. The statement that you read just says embrittlement cannot cause failure unless there is a crack.

ADMIN. JUDGE WARDWELL: Correct.

DR. HISER: So the Staff did not imply embrittlement causes cracks.

ADMIN. JUDGE WARDWELL: Right.

DR. HISER: Okay. Just to make sure.

ADMIN. JUDGE WARDWELL: Yes. I didn't mean to imply that you implied that.

DR. HISER: Okay.

CHAIRMAN MCDADE: Okay. And --

DR. HISER: Maybe I inferred that from you.

CHAIRMAN MCDADE: And either Dr. Hiser or Dr. Lahey, let me -- and, again, I want to make sure I understand this before we go off to try to make a decision here. Embrittlement is a mechanism, an effect of embrittlement is a decrease in ductility. And it's Dr. Lahey's position that, that decrease in ductility makes the item more susceptible to cracking. Dr. Lahey, is that correct?

DR. LAHEY: The effect of embrittlement is -- I mean, fundamentally, the difference is no crack no problem. I mean, there are people who believe no crack no problem, to me that was the quotable quote from yesterday. I don't believe that. I believe with or without a crack, if you have an embrittled structure and you hit it with a significant impulsive load, you can fail the structure. And that's the concern.

CHAIRMAN MCDADE: But that's -- again, the concept there is the ductility has decreased because of the embrittlement and because of that decrease in ductility, the item is more susceptible to cracking under stress with a load?

DR. LAHEY: Well, the fracture toughness, how the material can retard crack growth, is decreased. So cracks can grow faster if it's embrittled. So we don't disagree on that. What we disagree on is, do you need a surface crack or not? Is the material weakened or subject to failure if it's embrittled? That's really the crux of it. One thing that --

CHAIRMAN MCDADE: Also, and again maybe we'll get into this later, but you used the word failure and there's an inspection failure and then there's a failure of intended use. The failure of intended use involves cracking. But as I understand it, there would be cracking before you had a catastrophic failure, where it would no longer serve its intended use. I'm just trying to follow in my mind the mechanism along. And bear with me here because I'm plowing new ground for me, I realize that this is old ground for all of you. But I had thought from yesterday that ductility decrease made the item, a bolt for example, more susceptible. I thought that the fracture toughness actually increased with the embrittlement. Do you disagree? Did I --

DR. LAHEY: It depends, what do you mean by fracture toughness? If you mean the ability of a crack to propagate, which is the way I think it's being used, then it's not correct. I mean, there is another mechanism in which the strength of the material is increased by irradiation induced embrittlement.

CHAIRMAN MCDADE: Okay. And I guess what I'm confused then is the sort of the inter-reaction of these terms and what these terms represent. So if, and again, bear with me here, the strength increases, I'm not really sure what you mean by strength, I had taken strength to be synonymous with fracture toughness. And that the ductility, basically the ability to bend and come back is something entirely different. And although the ductility was decreasing with embrittlement, that there was a strength increase and there was not a decrease in fracture toughness. Dr. Hiser, is that the Staff's position? Or please educate me here.

DR. HISER: This is Allen Hiser with the Staff. What neutron fluence does to stainless steel in particular or the internals is you increase the fluence, the yield strength increases, the ultimate strength increases, the fracture toughness is reduced, and that's normally what we consider neutron embrittlement. It makes the fracture toughness decrease. In addition, the ductility decreases. And that's the bendability that you spoke of. So the fracture toughness does get reduced by neutron fluence.

CHAIRMAN MCDADE: And please explain to me very briefly if you can exactly what you mean by fracture toughness --

DR. HISER: Fracture toughness --

CHAIRMAN MCDADE: -- as opposed to strength.

DR. HISER: Well, I guess to separate them maybe in two ways. Strength is in particular relevant if you have an uncracked component. The failure of that component will be directly related to the strength of the material. So if you have an embrittled component or an irradiated component that has no cracks, it will fail at a higher load than one that is unirradiated. And going back to yesterday's discussion, you can always have a higher load and cause that uncracked component to fail.

For the fracture toughness, that really relates to the response of a component that has a crack in it. So, for example, if you have two components that are identical, one is irradiated, one is not, they both have the same size of crack, they both are subjected to the same loads, the one that's irradiated will have a lower fracture toughness, the crack will grow more readily in that material than in the unirradiated component. So that is where the fracture toughness comes into place. If you have a crack to start with.

CHAIRMAN MCDADE: Okay. And, Dr. Lahey, do you agree with that?

DR. LAHEY: Yes, sir. I tried to say that, he said it better than I. And I do want to bring another thing up --

ADMIN. JUDGE WARDWELL: Before you go on to that other thing, I want to clarify something else you said to fix this point, if I might.

DR. LAHEY: Yes.

ADMIN. JUDGE WARDWELL: And that is, you used the phrase that if it's embrittled and then it's hit with a significant load. Would your same result happen if it was hit with design basis loads? Do you believe those are significant loads to cause the issue that you're dealing with?

DR. LAHEY: Yes, many of them are. Many of them are and --

ADMIN. JUDGE WARDWELL: That's all I want to know is that it doesn't have to be an excessive load, you believe this will happen under design basis loads. Is that correct?

DR. LAHEY: That's correct.

ADMIN. JUDGE WARDWELL: Thank you.

DR. LAHEY: The difference between what you've heard in the past and what I'm saying is, is that when people have said that they can withstand design basis loads, if you look at the EPRI document and see what they've done, they've applied those in a static way, not in an impulsive way. And I tried to show you in my little cartoon yesterday that you can have significant difference between static and dynamic loads. I want to return to the document where Mr. Lott was an author of. This is the Icone paper that you had asked about. And in that one, the difference from the one we had talked about in more detail --

CHAIRMAN MCDADE: Do you have an exhibit number on that?

DR. LAHEY: Do you know the exhibit number? Oh, yes -- NRC 000177.

CHAIRMAN MCDADE: Thank you.

DR. LAHEY: Okay. In there is a statement which is a little different from what I heard before. This is light water reactor conditions and it says that the increase in tensile strength because of irradiation causes a, in the high cycle fatigue region, low amplitude, an increase in the strength capability, less failure, and a decrease of ductility results in a decrease of fatigue life in the low cycle fatigue region. So this is entirely consistent with a high pressure or the high flux reader reactor data, which was at a higher temperature.

They go on to say that the strain amplitude in their test was rather small, 0.6 percent, so they only saw the increase, not the decrease. And we're worried about the decrease, because that's an indication if you have enough amplitude, you can cause the thing to fail early and if you have a larger amplitude because of a shock load, it can have a catastrophic failure. So I wanted to bring that quote to you. That's on Page 2 under the Effect of Fluence, is the heading.

DR. LOTT: And may I, your honor? I just want to say that, that statement about the high strain amplitude data or the low cycle data reflected primarily the limits of what was tested in this program. And we were aware when we wrote the program of exactly the data that Dr. Lahey has suggested here. And just wanted to be cautious about how far we extended the findings of this paper. And I would stick by that. I will also point out, and I think it's again something we may get to under the next Contention, that we don't really expect to see these large strain amplitudes in the reactor internals, at least the reactor internals that have significant irradiation effects.

MR. STROSNIDER: Your honor, this is Jack Strosnider for Entergy.

ADMIN. JUDGE WARDWELL: I think we're going to move on now, thank you.

MR. STROSNIDER: Going to move on? Okay.

ADMIN. JUDGE WARDWELL: Yes. Back to my original question, which I really don't want to reread again, but it was all those -- the statement that NRC disagreed with your concerns and pointing out that --

one of the things they pointed out was primary components to be inspected under Entergy's program are those components that are most likely to be affected by synergistic effects if they exist. Do you agree that the primary components will be those most likely to be affected by synergism?

DR. LAHEY: Yes. I don't disagree with that.

ADMIN. JUDGE WARDWELL: Thank you.

DR. LAHEY: Their difference between how I view it and how they view it is, is the cracking required?

ADMIN. JUDGE WARDWELL: But if cracking does occur, wouldn't that be a result of whatever mechanisms created that? And so the synergism would be built into that observed crack, would it not?

DR. LAHEY: I don't disagree what that either. But remember, what I'm concerned with is either before or after any of the significant surface cracking. If you have an event which loads it impulsively, you can fail the structure. And then once you fail the structure, depending on what it is, it can lead to an uncoolable geometry. That's what I'm concerned with.

ADMIN. JUDGE WARDWELL: Okay, thank you.

I think you may have said this, but I'm going to ask it again because maybe you didn't, so I want to make sure we do cover this space. Are you concerned with the interval of the inspections or do you feel that's adequate based on the operating experience to date?

DR. LAHEY: You're going back to the ten year --

ADMIN. JUDGE WARDWELL: Right.

DR. LAHEY: -- interval?

ADMIN. JUDGE WARDWELL: Exactly.

DR. LAHEY: It seems a little long, but I'm more interested in a baseline inspection, a very thorough baseline inspection as they go into the period of extended operation, because otherwise, an inspection process doesn't make a lot of sense if you don't know where you start. So I think the sooner the better they do that. And then after that, the interval I think would depend on the kind of things that were discussed. What would be the implications of failure and how long do you have to take action before you can have a big problem?

ADMIN. JUDGE WARDWELL: Right now, and I believe they're planned for 2016 and 2019, if I was correct in my memory. Do you have any comments in regards to what you would recommend and what's your justification for recommending anything earlier than -- well, it's not going to happen before 2016 anyhow, but the 2019 one?

DR. LAHEY: Well, we had recommended some time ago, but time goes on, so the sooner the better. But if that's the sooner they can do it, so be it.

ADMIN. JUDGE WARDWELL: Thank you. I guess you, and as I heard you say, you're more concerned with the extent of that inspection program rather than necessarily any timing between the small years between 2016 and 2019?

DR. LAHEY: I think that's correct.

ADMIN. JUDGE WARDWELL: Thank you. Let's move on now to talk a little bit more about the preventative actions, the corrective actions, and acceptance criteria. Entergy's testimony, 616, Answer 203, Page 136 to 137, Entergy has undertaken or will implement several types of preventative actions to manage the effects of aging on reactor vessel internals at Indian Point, including implementing the IPEC water chemistry control program, replacing the split pins at IP2, committing to replace IP2 split pins again in 2016, using the fatigue monitoring program and addressing the action levels Number 8 in the safety evaluation for MRP 227-A, tracking plant transience and cycles, thereby assuring that fatigue usage from actual plant transience does not exceed ASME code design limits, and implementing neutron flux reduction measures to minimize the neutron fluence on the reactor pressure vessel, which in turn will minimize radiation induced aging effects at high fluence locations within the RVIs. I guess I'd ask the Staff, do you agree that these are considered preventative maintenance activities that are designated by GALL to be part of a consistent AMP?

DR. HISER: This is Allen Hiser with the Staff. The water chemistry program clearly is a preventative measure that's implemented as part of their AMP. The replacement of split pins, again, clearly putting in new material would prevent the accumulation of degradation that had occurred with the old pins. Fatigue monitoring is, I'm not sure if I'd call it preventative, but it clearly is an appropriate measure to ensure that the fatigue life is adequately monitored during the operation of the plant. So that would -- I'm not sure if I would call it preventative, but it -- prevent, mitigate, or minimize the effects of aging, I'd say, yes, it's within that umbrella. And finally the flux reduction, again, would help to minimize the aging effects on the vessel internals.

So I would agree with each of these.

ADMIN. JUDGE WARDWELL: Okay, thank you. In New York State's Exhibit 496, Attachment 1, which is the AMP, Page 5, the AMP states that the reactor vessel internals program is a condition monitoring program that does not include preventative actions. So I guess I'd ask, and I'll start with Entergy, doesn't that kind of contradict what you've stated in Answer 203 that I quoted from above where you're taking credit for a bunch of preventative actions?

MR. COX: Could you repeat that question?

ADMIN. JUDGE WARDWELL: Yes. The previous quote I had from your testimony of which I just asked Staff to respond to, which was Answer 203, you listed a number of preventative measurements that you were taking credit for. Your AMP states that, on Page 5, that the reactor vessel internals program is a condition monitoring program that does not include preventative actions. Well, it seems like you just took credit for a bunch of preventative actions and then you're saying in your AMP, it doesn't include them.

MR. COX: This is Alan Cox for Entergy. I think it's really a matter of semantics and how you describe things. What was meant by the statement in the reactor vessel internals program was that the water chemistry controls were included in another AMP. I mean, there is a separate AMP that applies not just to reactor vessel internals, but to water chemistry on the primary side as a whole. So that AMP goes beyond just reactor vessel internals. It's treated as a separate program, but it is referenced from the preventative action section here and it says that we do have preventative actions in that AMP that will apply to the reactor vessel internals. It's not a part of the reactor vessel internals AMP in the way it was described because it's a program that covers a whole lot more areas than just reactor vessel internals.

ADMIN. JUDGE WARDWELL: Okay, thank you. I guess that explains it as best you can. And I guess I'll turn to Staff. That Criteria 2 of GALL does require preventative actions. So don't there need to be some preventative actions within each individual AMP for each of the different components that are addressed by different AMPs, i.e., doesn't the reactor vessel internals have to have preventative actions associated with it in order to be consistent with GALL?

DR. HISER: This is Allen Hiser of the

Staff. For the vessel internals program, that is correct. Water chemistry is an essential element of the program. Otherwise, the inspection types, frequency, scope, et cetera, would be different, because there's an expectation that there is a program that's helping to minimize the effects of aging.

ADMIN. JUDGE WARDWELL: Would the, likewise, the activity in regards to the fuel that reduces the fluence, if you will, but the way it's configured also be considered one of these preventative measures that you're taking account of and taking credit for in your evaluation of whether or not Criteria 2 is met for the RVIs?

DR. HISER: This is Allen Hiser of the Staff. I think the flux reduction is not explicit within the GALL AMP, but it is explicit in MRP 227 that it was one of the three criteria used for demonstrating that a plant was bounded by the report. And based on that, it is sort of implicit to the GALL AMP that, that is necessary. Again, if flux reduction was not implemented at a plant, than it would be inappropriate for them to use MRP 227 because that is one of the fundamental assumptions because under the aging evaluation, it's in the report, and to demonstrate the adequacy of aging management.

ADMIN. JUDGE WARDWELL: And in your SER, have you specifically addressed each of the criteria, including Criteria 7 for corrective actions and 8 for confirmation process dealing with assuring preventative and corrective actions?

MR. POEHLER: This is Jeffrey Poehler of the Staff. Yes, we have specifically addressed each of the ten elements of the GALL Aging Management Program, as modified by the LRISG.

ADMIN. JUDGE WARDWELL: Okay, thank you. Entergy's testimony, 616, Answer 146, Page 95, states that once a defect is discovered, its ability to withstand fatigue and combinations of both normal and accidental loads is evaluated by either fracture mechanics analysis or a structural analysis, i.e., an engineering evaluation, using the lower bound fracture toughness, i.e., the evaluation assumes a bounding level of embrittlement of the material.

Thus, the program has compensated for any inability to directly determine the level of embrittlement through a conservative assumption employed during evaluation of inspection findings. Thus, reasonable assurance that the effects of aging will be adequately managed is provided without the need for direct observation or measurement of the level of embrittlement. I guess my question to Entergy is, how was this bounding level of embrittlement, as you referred to up above, determined?

DR. LOTT: Basically, it was determined based on published data on fracture toughness and, effectively, analysis. Which is in -- I'm going to look to my colleagues for the reference numbers, 727? And published by the NRC in the NUREG process.

MR. GRIESBACH: 7207?

DR. LOTT: 7207, I'm sorry.

MR. GRIESBACH: Also MRP 210.

DR. LOTT: Yes. Maybe I should let Mr. Griesbach talk.

MR. GRIESBACH: This is Tim Griesbach from Entergy. That data and the evaluation method that would be used as an example is given in MRP 210. That's an EPRI MRP program document.

ADMIN. JUDGE WARDWELL: And you believe that's an exhibit in this proceeding?

DR. LOTT: Yes.

MR. GRIESBACH: Yes, it is.

ADMIN. JUDGE WARDWELL: Okay. Thank you.

DR. LOTT: 646.

ADMIN. JUDGE WARDWELL: And I assume I'll hear what the number is shortly from my ace searchers of what that is.

MR. KUYLER: Your honor, MRP 210 is Entergy Exhibit 646.

ADMIN. JUDGE WARDWELL: Oh, thank you so much. I'd also like to -- in that statement, to refresh your memory from three minutes ago, the program has compensated for any inability to directly determine the level of embrittlement through a conservative assumption. And what is this conservative assumption employed during the evaluation of the inspection findings to which you refer that ensures that the program has compensated for any inability to directly determine the level of embrittlement? And where might that be documented?

DR. LOTT: Again, that's documented and it's required based on the procedures in WCAP-170986, which was our internal, we'll be calling methodology and data requirements for determinative engineering evaluations. What we -- again, the bounding value is based on a fracture mechanics analysis where we're looking at this reduced fracture toughness, which is a property of the material. In most cases, it's done on the basis of a linear elastic evaluation, even though we've already testified that it appears that there's some ductility left in these materials, it assumes very low levels of ductility. And it bounds the existing data on fracture toughness of material as a function of fluence, for stainless steel alloys in general.

ADMIN. JUDGE WARDWELL: Thank you. Dr. Lahey, why hasn't this approach covered much of some of your uncertainties that you are concerned with in regards to the potential failure modes of what might take place under operational conditions?

DR. LAHEY: Does your question -- this is Richard Lahey from New York. Does your question also include the previous discussion with the replacement of split pins and that sort of thing?

ADMIN. JUDGE WARDWELL: Sure, if you want to add that --

DR. LAHEY: Okay.

ADMIN. JUDGE WARDWELL: -- into it, you can talk about that also.

DR. LAHEY: Those steps, I believe, are quite prudent. We particularly like the replacement of a degraded component and would highly encourage the replacement of other degraded components, in particular the baffle bolts, which are the most vulnerable. The other part of your question, I guess, relates to the loads and are the loads which are applied to determine the integrity or the time it will take for a component to failure, if they are appropriate loads, if they are impulsive loads, which would have the maximum effect on causing a failure. If they're absolutely correct, then I have no problem with it. So far, I haven't seen that.

ADMIN. JUDGE WARDWELL: I think I was more concerned with the bounding level of embrittlement that was selected for this analysis and the conservative assumption employed in evaluation of the inspection findings.

DR. LAHEY: Okay. Maybe you can -- one of the exhibits we have, New York State 000495, could we bring that up and we talk about it?

ADMIN. JUDGE WARDWELL: What page number would you like?

DR. LAHEY: It's Page 3.

ADMIN. JUDGE WARDWELL: Okay.

DR. LAHEY: Right. So this is a --

ADMIN. JUDGE WARDWELL: Well, wait just a second until we get that --

DR. LAHEY: Oh, it's not up yet?

ADMIN. JUDGE WARDWELL: It's not up in front of me at least.

DR. LAHEY: Oh, it's up here.

ADMIN. JUDGE WARDWELL: The ON/OFF button being in the wrong position. We all -- none of us have it, do we?

MR. HARRIS: Your honor, this is Brian Harris for the Staff. We don't have it over here either.

ADMIN. JUDGE WARDWELL: You don't have it either?

MS. SUTTON: Neither does Entergy, your honor.

MR. SIPOS: Your honor, for New York, we have it up over here.

(Laughter.)

ADMIN. JUDGE WARDWELL: You seem pretty smug about that, Mr. Sipos.

MR. SIPOS: It's a rare feeling.

(Laughter.)

ADMIN. JUDGE WARDWELL: You going to rent us out a little sneak views of this or --

MR. SIPOS: I could turn the monitor towards your honors direction if you like.

ADMIN. JUDGE WARDWELL: For the right price?

(Laughter.)

ADMIN. JUDGE WARDWELL: These aren't connected in series are they, like my Christmas lights? Here we go. We've got a winner.

(Laughter.)

DR. LAHEY: Can you see it? Is it up on your screen now?

ADMIN. JUDGE WARDWELL: It is, thank you.

DR. LAHEY: Okay.

ADMIN. JUDGE WARDWELL: Thank you for waiting.

DR. LAHEY: Over there is it okay?

ADMIN. JUDGE WARDWELL: Is everyone happy?

MR. HARRIS: Yes.

MR. SIPOS: Yes, your honor.

DR. LAHEY: Okay. So this is a plot that keeps getting passed around. It's in Gary Was's classic book on nuclear metallurgy, it's in a lot of EPRI reports, a lot of U.S. NRC reports. It came from an individual and there's certain assumptions made in it in terms of how you calculate the fluxes, the neutron fluxes. And, in particular, what components you're choosing. But it gives you a pretty good estimate of the type of fluences you're going to see. So on the upper curve is the fluence for greater than one million electron volt neutrons, which are the damaging ones. And on the lower scale is the corresponding displacements per atom, how many times each atom is knocked out of its lattice. So you can see, if you look at the end of PWR life extension, it's expected to be greater than 100 dpa or greater than ten to the 23 --

ADMIN. JUDGE WARDWELL: And what are the years for, do you know, that they assume for this life extension?

DR. LAHEY: This is the 20 year life extension.

ADMIN. JUDGE WARDWELL: Okay. So the 60 years total?

DR. LAHEY: It's the type we're talking about.

ADMIN. JUDGE WARDWELL: Okay, thank you.

DR. LAHEY: So you can see there's significant, absolutely significant fluences and damage at that point. And of course, that is where you have to be really concerned because if you look at the data on when bad things happen in terms of embrittlement and all these other issues we've been talking about, you have to get to a displacement per atom of about one or so before bad things start to happen. Then it drops off pretty fast, it really goes down very fast.

So once you start getting out here, you're really in a region where you have to be very careful because you're on what I call the bathtub curve, if you know what I mean by that. You're way out on the other part of the bathtub curve where you're starting to wear things out, you're really beating them up and they're failing. So, hopefully this gives you some insight into what the concerns are, just as a way to benchmark yourself.

The NRC uses the criterion for significant embrittlement, as I understand it anyway, you folks can correct me if I'm wrong, but about one times ten to the 21 for fluence, when for stainless steel you start to get significant embrittlement. And other people use like 6.7 times ten to the 20, but it's of that order of magnitude. So you can see, by the end of this thing we're a thousand times greater than the onset of that kind of damage.

DR. LOTT: Your honor, may I just -- a moment ago, I had trouble coming up with a reference for you in terms of limiting fracture toughness values. I wanted to point that actually those limiting values, when I think about it, are in MRP 227, Section 6, as well as in the other documents we described. I'd also like to perhaps amend a little bit to my statement and say that, one of the other conservatisms in our analysis has been throughout, we tend to apply peak fluence values to entire components. So we don't necessarily -- and when we're doing evaluations, when we've identified components, we have looked at 60 year fluences and looked at the peak location on that and there could be a large gradient in fluence across the component. So I would just suggest that, that's an additionally conservative assumption we've made about determining the fluence on a component.

ADMIN. JUDGE WARDWELL: Okay, thank you. Do you have any other comments that you'd like to add in regards to this figure that Dr. Lahey has presented and his comments associated with it?

DR. LOTT: I mean, I think I've seen this document many times before as well. It's just a schematic that I think is consistent with actually many of the assumptions we've made here in terms of -- I think Dr. Lahey was talking about the threshold values for irradiation embrittlement of seven times ten to the 20, that's exactly the value we used in our evaluations, that's approximately one dpa. So, I think that we're on the same -- we've dealt with and understand these concerns and that our testimony has demonstrated that.

ADMIN. JUDGE WARDWELL: Clarify for me, will you, if you are using seven times ten to the 20 or about one dpa, as what you assume as a conservative fluence value, how does that relate to what you may actually be experiencing, which is two orders of magnitude higher than that by the end of the life extension?

DR. LOTT: Well, first of all, what we're looking at is the threshold for what's the lowest value. I mean, so basically what that determines is whether a component has any irradiation embrittlement concern at all. Above that, all the way up to a million, presumably, it has susceptibility. Our first goal of the threshold values was to determine the lower limit.

ADMIN. JUDGE WARDWELL: So it's conservative because it's on the low side, encompassing much more numbers of --

DR. LOTT: Right.

ADMIN. JUDGE WARDWELL: -- potential of --

DR. LOTT: Right, so yes. That's --

ADMIN. JUDGE WARDWELL: -- internals.

DR. LOTT: -- what the concern is and we

identified components with potential irradiation embrittlement. And our basis for that was a relatively low value, the onset of embrittlement, consistent with the curve I showed you earlier this morning on yield stress.

ADMIN. JUDGE WARDWELL: Yes. Okay. Thank you.

ADMIN. JUDGE KENNEDY: Dr. Lott, this is Judge Kennedy. The baffle, let's see if I'm getting this right, the baffle bolts are, at least components I identify on this chart, at somewhere around ten to the 22, PWR baffle bolt failures?

DR. LOTT: Yes. Some --

ADMIN. JUDGE KENNEDY: Are there --

DR. LOTT: Yes.

ADMIN. JUDGE KENNEDY: Are there reactor vessel internal components that experience fluences beyond that value?

DR. LOTT: Yes, there are. I mean, it's not like that's a fall off the cliff value. When we have -- and have reported seeing IASCC and baffle bolts in operating plants, I think that, that value is basically trying to demonstrate that it happened roughly at that fluence.

ADMIN. JUDGE KENNEDY: Okay. All right,

thanks.

CHAIRMAN MCDADE: Is this the baffle former bolt or the baffle edge bolt or is this all baffle bolts that they're talking about here?

DR. LOTT: Okay. There are basically three different kinds of baffle bolts. There's the baffle -- if you're familiar with the baffle structure, they're plates that surround the core and there's plates behind them, horizontal plates, that hold them in place. The bolt between the baffle plate and the horizontal plate is a baffle bolt. Where the two plates come together, along the edge, along the seam, there may be bolts that go from baffle plate to baffle plate, not baffle plate to the former. Those are the baffle edge bolts. They basically seal up the gap between the two plates to keep water from jetting through there. Not all plants even have baffle edge bolts.

And we've done evaluations to determine that in terms of holding the baffle together, in terms of an accident type scenario, we don't take credit for the baffle edge bolts at all. So it's, again, a conservatism in our assumptions when we do these acceptable bolting patterns. So baffle edge bolts. And then there's barrel former bolts, which are from the round barrel, which is the outside container of the entire internals, that the horizontal plates, the other side of them, attach to the barrel. So barrel former bolts, baffle former bolts, and baffle edge bolts.

MR. KUYLER: Your honor, there's a diagram in Entergy's testimony, Exhibit 616, Page 54.

CHAIRMAN MCDADE: Okay, thank you. And thank you, Dr. Lott.

ADMIN. JUDGE WARDWELL: NRC's testimony, 197, Answer 179, Pages 102 to 103, the reactor vessel internals AMP is structured to managing the aging effects such that the intended function of the components will be preserved during the period of extended operation from 40 to 60 years. It accomplishes this task by establishing an Inspection Plan for the relevant components that it's able to identify potential aging effects prior to any loss of function through appropriate schedules and conservative acceptance criteria. And so I'll ask Entergy, does Table 5-5 of the Applicant's Inspection Plan, and that's New York State 496, Attachment 2, contain these acceptance criteria?

MR. DOLANSKY: This is Bob Dolansky with Entergy. Yes.

ADMIN. JUDGE WARDWELL: Okay. And does it not state that the examination acceptance criteria for visual examination is the absence of the specific relevant condition?

MR. DOLANSKY: Yes.

ADMIN. JUDGE WARDWELL: And now the tougher part, what are examples of this relevant condition? And where would one find that?

MR. DOLANSKY: For an EVT1, it would be a crack-like indication.

ADMIN. JUDGE WARDWELL: So it varies by your inspection technique generally, rather than by component?

MR. DOLANSKY: And by the component. In other words, EVT1 is typically looking for cracking, but a VT3 -- this is Bob Dolansky for Entergy. VT3 could be looking for either wear or it could be looking for a dimensional change, like void swelling, something like that. So the acceptance criteria would depend on what you're looking for and the method that you're using.

ADMIN. JUDGE WARDWELL: And where is that documented anywhere in regards to the various components so that we could turn to that and it would say, yes, it's any indication of cracking or it's any indication of a dimensional change or --

MR. DOLANSKY: Give me one moment, please.

ADMIN. JUDGE WARDWELL: If you want to get back to us --

MR. DOLANSKY: Okay. If you -- I'll give a cite, one second. New York State 496, Letter 12-037.

ADMIN. JUDGE WARDWELL: Okay.

MR. DOLANSKY: Table 5-5.

ADMIN. JUDGE WARDWELL Yes, that's what I was referring to.

MR. DOLANSKY: That gives -- you'll see the actual -- under examination acceptance criteria in the table. For instance, for the upper core barrel cylinder girth welds, the specific relevant condition is a detectable crack-like surface indication.

ADMIN. JUDGE WARDWELL: Okay. So where ever we see that, that's what it means. As soon as you detect cracking, you're going to take some corrective action.

MR. DOLANSKY: Right. So our inspection procedure, that's the people actually doing the inspection use a procedure, their acceptance criteria for a recordable indication would be that.

ADMIN. JUDGE WARDWELL: Okay. Thank you.

MR. DOLANSKY: You're welcome.

ADMIN. JUDGE WARDWELL: That helps. Just for completeness, Dr. Lahey, any comments in regards to that acceptance criteria and where it's found and the adequacy of it?

DR. LAHEY: This is Richard Lahey. I'm sorry, I don't have that document in front of me and I don't recall it.

ADMIN. JUDGE WARDWELL: Okay. You have no comment?

DR. LAHEY: So I can't really answer right now.

ADMIN. JUDGE WARDWELL: If you do later on, if you do want to look at it later on and have some comments --

DR. LAHEY: Okay.

ADMIN. JUDGE WARDWELL: -- remind me of it and we'll be glad to. I want to make sure you have a chance to anyhow.

DR. LAHEY: Thank you.

ADMIN. JUDGE WARDWELL: New York State's testimony, 482, Page 79, states that, I believe that the most vulnerable reactor pressure vessel internals need to be carefully identified and repaired or replaced prior to the extended operation since it is beyond the current state of the art to perform realistic and accurate calculations on relocation of failed RVP internals and the resultant potential for core blockages or degraded core cooling. And I guess I'll ask you, Dr. Lahey, that while I understand the basis for your position in regards to this replacement, does not this run counter to the whole idea of managing aging? And doesn't it to a large degree try to circumvent all of the regulations that are geared towards aging management as opposed to prescriptive replacements?

DR. LAHEY: That's a very interesting question. In my opinion, aging management is indeed important. But when you get components that look like they're vulnerable and can fail and you're not able to determine with any precision what the effect of that might be, then I think the prudent thing to do is to replace those components. I've spent ten years of my life trying to calculate where things go and it's very hard to do, very difficult to do, there's too many possibilities.

But the one thing I know for sure is, once you lose an intact geometry, you've got big problems. So, anything that will preserve that, I think we're way ahead of the game and, to me -- do you know what a make-buy decision is? You decide is it cheaper to make something or buy something? This is like a make-buy decision. Is it cheaper to calculate and inspect and go through litigation and all that or is it cheaper just to replace it and have the problem go away? And, for many of these things, I think it's cheaper and much more prudent to just replace it.

ADMIN. JUDGE WARDWELL: The -- I forgot my question now because I was going to change it in regards to your last comment. But with your description of the details which basically are in the AMP and the inspection program that's in there and your applauding of it as not the complete what's needed, but not really a lot of substantive disagreements with the approaches and the extents of what they're doing, given what we've heard in regards to the conservative assumptions, why isn't there a fair degree of reasonable assurance that something, if it does go amiss, would be detected before that intended functionality of the RVIs was lost?

DR. LAHEY: This is again Richard Lahey from New York State. When you say conservative assumptions, are you referring to embrittlement alone now or --

ADMIN. JUDGE WARDWELL: I'm talking about all that we've heard about --

DR. LAHEY: All of the above?

ADMIN. JUDGE WARDWELL: -- to date in regards to our questioning and what's in their testimony in regards to the decisions they make as they prepared their AMP and then how they're implementing their AMP.

DR. LAHEY: Right.

ADMIN. JUDGE WARDWELL: Does that not give one reasonable assurance that if something does go awry, it's not going to try to stop anything that might go awry, but if it does, isn't there a reasonable assurance that it would be detected prior to the RVI losing all of its intended function, even though it may crack or even do some other things?

DR. LAHEY: Not in my opinion. If you have sufficient degraded components that can lead to a destruction of intact geometry, there's a reasonable chance that you can have an unexpected accident. Accidents, by definition, you don't expect them, but they happen, or an earthquake happens just when you don't want it. And then what will that lead to? If it can lead to degraded or materials which fail and relocate, then I'm very concerned about it. Because I just know you can't calculate the consequence of that.

So if you can identify materials that are highly irradiated, and we can, and highly fatigued, and we can, and they are things that could lead to destruction of a coolable geometry, then you should take action, the sooner the better. Don't wait for something bad to happen. So that's why I'm very keen on do what you're doing with the -- have Entergy do what they're doing with the split pin, replace it because it's degrading. Replace the things that have a significant effect on the safety of the reactor. There's not that many. I mean, we haven't talked about pressure boundary components yet, but we will, tomorrow I guess. And there's only a few that are really, really crucial. And there's only a few things in core that are really, really crucial. And, happily, these are things that aren't that hard to replace.

ADMIN. JUDGE WARDWELL: Thank you, Dr. Lahey.

DR. LAHEY: That's why --

ADMIN. JUDGE WARDWELL: Thank you.

DR. LAHEY: -- I feel that way.

ADMIN. JUDGE WARDWELL: Entergy, have you addressed anywhere or evaluated or how did you handle this potential relocation of failed RVIs and the resultant potential for core blockages and degraded core cooling?

MR. DOLANSKY: This is Bob Dolansky with Entergy. When we do an analysis that looks at components, one of the requirements is that we maintain core coolability and core geometry. I mean, that's ultimately what we're trying to do. We want to make sure that, that core -- that's the basis of the whole thing is that the core stays coolable and the geometry stays -- that it maintains core geometry. I mean, that is what we do, that's the exact requirements that we analyze for.

DR. LOTT: Yes, this is Randy Lott. And I think, again, we kept coming back to it, but the acceptable baffle bolting pattern analysis is a good example of what we're doing. In that analysis, effectively, you're looking to see that there are enough baffle bolts to keep the baffles from moving, interacting with the fuel, crushing fuel grids, or interfering with the ability to drop control rods. Which is really always our concern, it's maintaining core coolability and being able to insert the control rods, shut down the reaction. That's basically the definition of the safety requirement.

MR. DOLANSKY: Right. And I just want to clarify one thing that Dr. Lahey said. We are replacing split pins. We're not planning to replace baffle bolts. Replacing baffle bolts is not some no nevermind, easy thing to do. They have lock bars welded on, it's an extremely difficult area to get to. Split pins were more designed that they might have to be replaced. So the replacement of the split pins is a little easier, it's more straightforward, something that can be done much more easily.

It's easy to sit here and say, just replace the baffle bolts. Actually replacing baffle bolts, although it can be done, is not a simple things. There's a lot of dose involved with that, there's a lot of possibility of loose parts, and there's consequences to replacing things that aren't bad. So, the way we look at is, we will go out and do these inspections on the baffle bolts using a technique that's very good, difficult tooling that was developed just to get us a better UT exam will additionally -- when we do that inspection, as Dr. Lott said, we're going to use an acceptable bolting pattern analysis that ensures that if we find degraded bolts that we can maintain core coolability, core geometry. If we do that inspection and we find that there's very, very few bolts that are degraded, to me, it doesn't make any sense to go out and just wholesale replace baffle bolts. There's too much danger and risk in doing that. If the bolts are good, I don't see any reason to do that.

ADMIN. JUDGE WARDWELL: Okay, thank you.

CHAIRMAN MCDADE: Okay. And also, just as our discussion moves on for the rest of the week, I mean, we neither have the authority or interest in micromanaging the way that Entergy does its operations. You have to make certain business decisions. Our function is just whether or not the plans that you have put forward provide reasonable assurance that these items will maintain their intended function for the period of extended operation.

So, certain issues of whether or not as a matter of policy you replace, that's outside the scope of what we're looking at. Again, the scope of what we're looking at is whether or not the plans that you have put forward provide that reasonable assurance with regard to intended function, period. So, it's not as wide -- it's not just an open-ended discussion. We're trying to focus in on what we have to decide here. Judge Wardwell?

DR. LAHEY: Yes, and your honor, can I say something? What we just heard from Entergy is the most encouraging thing I've heard in the last eight years. I'm very happy to hear it because it sounded like they're going to do something that makes a lot of sense, look at the integrity of it, and if it's not a big issue, then don't do it, if it is a big issue, presumably they will entertain that as a possibility to replace. Up until now, all I heard was, question, do you agree with Dr. Lahey on anything and the answer is, no. Everything, no, no, no.

(Laughter.)

DR. LAHEY: And so this was encouraging to me.

ADMIN. JUDGE WARDWELL: Thank you, Dr. Lahey. NRC's Exhibit 197, Answer 134, Page 82, states that the Staff found Entergy's AMP met the Staff's guidance for corrective actions because detected conditions not satisfied in the examination acceptance criteria will be processed through the plant's corrective action programs. And I guess I'll start with the Staff considering it's your exhibit. What's your understanding of how the plant's corrective action program works and interacts with the AMP for reactor vessel internals?

DR. HISER: This is Allen Hiser of the Staff. The corrective action program would be initiated if there's an inspection finding that exceeds the inspection acceptance criteria, which we've discussed previously. I believe it's Table 5-5 of the RVI Inspection Plan from Indian Point. The corrective action program would require that the condition be assessed, I guess in maybe two or three different ways.

One is, is the condition that was identified, that condition needs to be resolved, is it acceptable maybe through engineering evaluation, is repair required, is replacement required? So that would be one path that would be followed by the corrective action program. A second would be a consideration of expansion of the inspections consistent with the AMP. So that would follow. Consideration of reinspection interval would be a part of that evaluation as well. So the corrective action program should consider pretty much any aspect that's relevant to the finding itself, be it for that finding or other components.

ADMIN. JUDGE WARDWELL: Okay, thank you.

DR. HISER: And that is required in Appendix B to 10 CFR Part 50, so it is a regulated activity, a regulated program of the plant that the NRC inspects periodically.

ADMIN. JUDGE WARDWELL: Thank you for your understanding of the plant's corrective action. I'll turn to Entergy and can you now explain how your plant's corrective action program actually works and interacts with the AMP or just state what differences you might have with what Dr. Hiser just presented?

MR. AZEVEDO: Yes, this is Nelson Azevedo for Entergy. What was just discussed is correct, that's what we do. So we find an indication then we put in our corrective action program to determine whether we have to repair or replace it or whether it's acceptable. We also do extended condition inspections if it's warranted. So those are the things we look at.

MR. DOLANSKY: I just want to -- this is Bob Dolansky with Entergy. Just to add on something I said earlier where I said Westinghouse was developing acceptance criteria for us. The acceptance criteria in the plan are the just evidence of a crack-like indication. That would be in the inspection procedure. We're additionally, right now, getting additional acceptance criteria that if there was a crack and it could be a certain size, that's what's being developed. So I misspoke a little bit before, that's not the acceptance criteria in here. That's the --

ADMIN. JUDGE WARDWELL: As it exists now.

MR. DOLANSKY: -- acceptance criteria after it enters our corrective action program.

ADMIN. JUDGE WARDWELL: Okay, thank you. Ready to rock and roll.

CHAIRMAN MCDADE: Okay. This may be a good time to break for lunch. Do you have anything before we break?

ADMIN. JUDGE KENNEDY: No, I do not.

CHAIRMAN MCDADE: Okay. The question is, how long we break for lunch? And I'm not sure if there's any difficulty in people getting lunch within a relatively short period of time. I would propose to come back at 1:30. Is that going to give enough time for people to get lunch and take care of what they need to? NRC?

MR. HARRIS: Yes, your honor.

CHAIRMAN MCDADE: Entergy?

MS. SUTTON: Yes, your honor.

CHAIRMAN MCDADE: New York?

MR. SIPOS: New York, yes, your honor.

CHAIRMAN MCDADE: Riverkeeper?

MS. BRANCATO: Yes, thank you.

CHAIRMAN MCDADE: Okay. Do any of the witnesses perceive a problem with restarting at 1:30? Apparently not, we're in recess. We'll come back --

DR. LAHEY: No, your honor.

CHAIRMAN MCDADE: -- at 1:30. Thank you.

(Whereupon, the above-entitled matter went off the record at 12:19 p.m. and resumed at 1:34 p.m.)

CHAIRMAN MCDADE: The hearing will come to order. There are a couple of, or a few matters I guess, out there that I just wanted to address before we move forward. I believe there was a question, Mr. Dolansky, to you, as to whether or not you could provide other examples of monitoring of aging effects not observable by inspection. Were you --

MR. DOLANSKY: Just the one. Just the one that I gave.

CHAIRMAN MCDADE: Okay. And there was a question to Mr. Poehler about an EPRI report and whether or not that was a publically available report.

MR. POEHLER: Yes, this is Jeffrey Poehler of the Staff. That report is publically available and it's Exhibit NRC 000208, A through E, there's five parts.

CHAIRMAN MCDADE: Okay. And what is the

EPRI report number?

MR. POEHLER: It's an MRP letter actually, MRP --

CHAIRMAN MCDADE: Okay.

MR. POEHLER: -- Letter 2014-09.

CHAIRMAN MCDADE: Okay. Thank you.

MR. POEHLER: You're welcome.

CHAIRMAN MCDADE: And, Dr. Lahey, you were asked with regard to questions on Table 5-5 of New York 496. Did you have an opportunity to review that?

DR. LAHEY: No, your honor, I haven't been able to get access to it yet.

CHAIRMAN MCDADE: Okay. At our next break, we will make sure that you get access to that.

DR. LAHEY: Okay. Thank you.

CHAIRMAN MCDADE: Mr. Sipos, it looked like you had a question.

MR. SIPOS: I had a procedural question, your honor. And it concerns scheduling of New York's second expert, Dr. David Duquette, who is an expert on Contention 38. And I was wondering if the Board could provide any guidance as to when you would like to -- or when the Board might start Contention 38? And it's purely a logistical question as to when --

CHAIRMAN MCDADE: No, I understand.

MR. SIPOS: -- to tell him to get in the car and --

CHAIRMAN MCDADE: And my best estimate at this point in time would be our start time on Thursday morning. That we're still, it's Tuesday afternoon, we're still working on 25, 26 we should get started maybe later today, should run most if not all of tomorrow. And a lot of what had been in 38, is in 38, we have touched in on the testimony on 25 and will touch on some more in the testimony of 26. So, from my standpoint, and I haven't, you just asked the question, I haven't discussed it with my colleagues, I would, for planning purposes, plan on starting on 38 on Thursday morning at 8:00.

ADMIN. JUDGE WARDWELL: Yes, I guess we just talk out loud. Yes, I would concur, in fact, to the point that even if we got done 26 early tomorrow, it would probably still be later in the day, that we could just not plan on starting 38 until Thursday morning.

MR. SIPOS: That's very helpful, thank you.

CHAIRMAN MCDADE: Dr. Kennedy?

ADMIN. JUDGE KENNEDY: That's fine with me.

ADMIN. JUDGE WARDWELL: And do you agree with that, that we just wouldn't --

CHAIRMAN MCDADE: Yes. I mean, I think for planning purposes, we're certainly not going to get done with 26 early tomorrow. So, if we did finish with 26 before 6:00, I don't think anybody would complain if we leave it at 5:50 instead of 6:00. And having Dr. Duquette get here for a relatively short period of time --

ADMIN. JUDGE WARDWELL: You want to check with the other parties to make sure they're comfortable with that too? If we waste a couple hours tomorrow afternoon by not starting 38, is that --

CHAIRMAN MCDADE: I mean, I don't think it'll be wasting a couple of hours, I think it'll be a few minutes, if anything. And we may still be on 26. You all can gauge how well we are keeping a schedule. Ms. Sutton?

MS. SUTTON: Yes, your honor. That's fine, we don't have any issues with that.

MR. HARRIS: The Staff has no issues with that, your honor.

CHAIRMAN MCDADE: Okay.

MS. BRANCATO: Riverkeeper has no issues, thanks.

MR. SIPOS: Thank you.

CHAIRMAN MCDADE: Okay.

ADMIN. JUDGE WARDWELL: Well, I have no more questions on 25.

MR. SIPOS: We'll start 38?

CHAIRMAN MCDADE: There's no questions on 26?

MR. SIPOS: All right, let's start 38.

CHAIRMAN MCDADE: We're in recess until Thursday morning.

(Laughter.)

ADMIN. JUDGE WARDWELL: Now, we're going to start looking at some specific materials and components associated with the RVI and starting off with control rods and J-groove welds. Entergy testimony, 616, Answer 98, Page 56, the Inspection Plan NL12-037 Attachment 2 at 62 through 64, provides a complete and corrected list of the RVI subassemblies at Indian Point and breaks those subassemblies down to their constituent components. New York State's testimony, 482, Page 13, states the control rods and the associated components are very important RPV internals and their integrity is an extremely important safety concern. In my opinion, omitting the control rod assemblies and associated fittings from an RPV internals Aging Management Program is a serious and indefensible omission.

Entergy's testimony, 616, Answer 99, Page 56, states that control rods are not subject to aging management review for two reasons. First, they perform their intended function with moving parts or a change in configuration. Thus, as the NRC Staff concluded, the control rods are active components not subject to aging management review. Second, control rods are considered consumables and the NRC has excluded from the license renewal review process those components that are subject to replacement based on a qualified life or a specified time period. And I guess I'd ask Dr. Lahey, do you now agree that control rods are not subject to aging management review?

DR. LAHEY: I agree that's the rule that has been put in place. I still have the concern that, because I'm looking at everything through the prism of reactor safety, so I have the concern that if you have a significant shock load, you can fracture these highly, highly embrittled structures and they will relocate in some way and you don't know how. And they don't care if they're moving or not, I mean, they're going to be at risk. So I think the reason for the rule is because of what I call silos. Things are being thought of quite separately, rather than in an integrated way.

ADMIN. JUDGE WARDWELL: Thank you.

CHAIRMAN MCDADE: But, Dr. Lahey, as I understand the position of Entergy and the NRC Staff, these are consumables. Therefore, they don't need an aging management plan, you are of necessity going to be replacing them at set intervals. So, what would an aging management plan consist of since you're already going to replace them?

DR. LAHEY: No, I understand that's why they view it the way they do. What I'm saying though is, let's say a week before they're going to replace them and they're in really bad shape in terms of embrittlement, you have an event which causes them to fail. This causes a big problem. So, I personally believe that when you look at aging management, you ought to look at all the things that affect the safety of the plant, the safety of the plant during the extended operation.

CHAIRMAN MCDADE: But the handling of those control rods would be no different during the period of extended operation than during the original license. They'd be subject to the same replacement criteria, would they not? So why does this have to do with the aging management and whether or not there should be a period of extended operation as opposed to just the current operating license of the plant?

DR. LAHEY: Well, I'm concerned with what happens during the period of extended operation. And this is a possible event that can happen, it's no different than some of the static components. And just because you can replace them doesn't mean that they couldn't be at risk if you have an event which causes a significant shock load.

CHAIRMAN MCDADE: Okay. But my question is, why would that be any different during the period of extended operation than it is during the original licensing period?

DR. LAHEY: Oh, I misinterpreted your question. It wouldn't, not for the control rods themselves.

CHAIRMAN MCDADE: Okay. Thank you. Dr. Wardwell?

ADMIN. JUDGE WARDWELL: Entergy's testimony, 616, Answer 100, Page 57, "the control rod guide tube assemblies, including the guide plates (CRGTs) and the lower flange welds are subject to aging management review and the effects of aging on these components are managed through the RVI AMP." And they're citing again NL12-037 Attachment 2 at 4 to 5 and Attachment 1 at 6 to 8. And those two attachments, if we remember, are the Inspection Plan and the AMP, respectively and that's New York State 496. Dr. Lahey, therefore, is incorrect when he asserts that Entergy has claimed that the guide tubes, plates, pins, and welds associated with control rods are not RVIs. Dr. Lahey, do you now agree that the control rod guide tube assemblies, including the guide plates and the lower flange welds are in fact subject to aging management review and part of the AMP for RVIs?

DR. LAHEY: Yes, I believe they are.

ADMIN. JUDGE WARDWELL: Okay.

DR. LAHEY: And they should be.

ADMIN. JUDGE WARDWELL: Thanks. New York State's testimony, 482, Page 45, Lines 3 through 9, because of geometric considerations, many pressure water reactors, including IP2 and IP3, cannot meet the U.S. NRC's required minimum coverage for the non-destructive testing of so-called J-groove welds. And, thus, the integrity of these important CRD stub tube welds cannot be directly confirmed by inspection. I'll start with you, Dr. Lahey, and what do you mean by the CRD stub tube welds?

DR. LAHEY: Well, this is one of the issues we talked about this morning, when we were talking about the possibility of a leakage forming and boron and water and forming boric acid and that sort of thing. So this is the weld on the inside. And as I understand it, they cannot get a complete inspection of this.

ADMIN. JUDGE WARDWELL: Okay. And what does that CRD stand for or --

DR. LAHEY: Oh, I'm sorry.

ADMIN. JUDGE WARDWELL: No, that's true, I appreciate what you've answered so far, that helped. But I also want to know what is the --

DR. LAHEY: Control rod drive.

ADMIN. JUDGE WARDWELL: Got you. Got you, okay. Thank you. Entergy's testimony on 616, Answer 101, Page 57 and 58, the reactor pressure vessel head penetration nozzle welds, sometimes referred to as J-groove welds, are not RVIs or even part of the reactor vessel pressure, but instead part of the reactor pressure vessel head. Aging effects applicable to the J-groove welds on the, and this must mean control rod drive M, head penetrations are managed under the reactor vessel head penetration inspection AMP. So, Entergy, what is the M of the control rod drive stand for?

MR. COX: The M -- this is Alan Cox for

Entergy. The M stands for mechanism.

ADMIN. JUDGE WARDWELL: Okay, mechanism. And, so, back to you, Dr. Lahey, and I think we covered this morning, do you now agree that these are managed under a different Aging Management Program and are not really part of this Contention?

DR. LAHEY: Well, they are reactor vessel internals. I mean, they're in the vessel, they're subject to aging. How they want to deal with it is not a great concern to me, as long as it's dealt with.

ADMIN. JUDGE WARDWELL: Good, thank you. Let's move on to a couple of materials we're dealing with. The first one we'll deal with is wrought Austenitic stainless steel. Entergy's testimony on Exhibit 616, Answer 105, Page 61, says that the majority of the Indian Point reactor vessel internal components are fabricated from wrought Austenitic stainless steels. Even though an increase in strength and decrease in toughness do occur when exposed to neutron irradiation, these materials retain their resistance to fast fracture within the operating temperature of interest for PWRs. The only exceptions are wrought Austenitic stainless steel materials with high amounts of cold working, and that's greater than 20 percent of cold working, but these materials are not present at Indian Point. And I think my first question for Entergy would be that, what do you mean by this cold working and what percentage do you have here at Indian Point?

DR. LOTT: Cold working is basically the rolling of the material or the stretching of the material, the working of the material to increase its yield strength. The more you -- for instance, you would take a plate and reduce the area by rolling it, the higher the strength becomes. In general, the reactor internals are not cold worked beyond about 15 percent, they're controlled. That's one of our screening criteria that we have been in fact evaluating for.

ADMIN. JUDGE WARDWELL: Okay. Thank you. And, Dr. Lahey, do you have any issues with those statements made in their original testimony or what was -- as amplified here?

DR. LAHEY: Are you asking me if I agree with the percentage of cold work they have or --

ADMIN. JUDGE WARDWELL: Yes. And with that, why they don't see the issue with the Austenitic steel that is exhibited when cold working is greater than 20 percent.

DR. LAHEY: But my understanding is, maybe

I misunderstood the statement you read, it was focused on that stainless steel is not subject to stress corrosion or embrittlement if it's not cold worked. Did I misunderstand that?

ADMIN. JUDGE WARDWELL: No, this dealt mostly -- well, let me repeat it again. The majority of the reactor vessel components are fabricated from wrought Austenitic stainless steel. Do you agree with that?

DR. LAHEY: I agree with that, absolutely.

ADMIN. JUDGE WARDWELL: And even though an increase in strength and decrease in toughness do occur when exposed to neutron irradiation --

DR. LAHEY: Okay.

ADMIN. JUDGE WARDWELL: -- these materials retain their resistance to fast fracture within the operating range of interest for PWRs.

DR. LAHEY: I don't agree with that. I mean, certainly, there are components that we've talked about that are highly embrittled and do not satisfy that.

ADMIN. JUDGE WARDWELL: Okay. And they say that the only exception are those wrought Austenitic stainless steels with high amounts of cold working, greater than 20 percent. And would you agree that if you had cold rolling steels greater than 20 percent, they wouldn't necessarily show an increase in strength or decrease in toughness and retain their resistance to fast fractures within the operating temperatures?

DR. LAHEY: No, I would not.

ADMIN. JUDGE WARDWELL: You would not? So you think they still would retain theirs even if you did have cold working --

DR. LAHEY: Yes, I think if they're highly embrittled, they are subject to fracture.

ADMIN. JUDGE WARDWELL: Right. So then you would agree that, that is an exception from their abilities to resist fast fracturing?

DR. LAHEY: I guess. I mean, I think if you have a material that's stainless steel and it has fluence above a certain level, and we talked about that level this morning, it becomes embrittled and then if subjected to the right kind of load, it can in fact fail. And we can talk about dimple failure, fracture versus --

ADMIN. JUDGE WARDWELL: We will in a minute.

DR. LAHEY: -- to me that's semantics, it failed.

ADMIN. JUDGE WARDWELL: Thank you.

CHAIRMAN MCDADE: A quick question, this is probably going to seem very basic to you all. We've been talking about Austenitic stainless steel, what does that mean? What is Austenitic stainless steel? Dr. Lahey or Dr. Lott or anybody can offer what that definition is as opposed to just stainless steel.

MR. GORDON: This is Barry Gordon from Entergy. Austenitic stainless steel is a -- it's named after the fellow who discovered the micro-structure. It's a face centered cubic structure. Austenitic stainless steel is not magnetic, a magnet will not stick to it, and it can only be strengthened by cold working. You can't heat treat it like you can some other alloys, like low alloy steel, you can only strengthen it by cold working it to some extent.

CHAIRMAN MCDADE: What are the different characteristics of Austenitic stainless steel as opposed to just your typical stainless steel? Are there significant differences?

MR. GORDON: Well, there is a ferritic stainless steel and there's a martensitic stainless steel. Martensitic stainless steels can be hardened and that's what -- like you have a stainless steel cutlery at home, those are martensitic stainless steels. And they'll stick, you have a magnetic board and you can put your knives on it. But Austenitic stainless steel is not a magnetic material.

CHAIRMAN MCDADE: Okay. For our purposes right here in the way that it reacts, not how you put it together, not how you work it, but how it reacts over time, is there any specific distinction with it being Austenitic stainless steel?

MR. GORDON: It's a very ductile material, it can be embrittled, like Dr. Lahey said, by irradiation, and it can be strengthened by cold working also, and also by irradiation.

CHAIRMAN MCDADE: Okay, thank you.

ADMIN. JUDGE WARDWELL: As a follow-up --

MR. GRIESBACH: Your honor, this is Tim Griesbach --

CHAIRMAN MCDADE: I'm sorry, who?

ADMIN. JUDGE WARDWELL: A follow-up question on that, which is extremely important to cover. When I buy some stainless steel fittings and stuff, sometimes a magnet will stick, to say, washer fenders and sometimes it won't. Now, it won't be strong, but some of them there's nothing and other times there is and, yet, it's supposed to be a stainless steel. What am I getting?

MR. GORDON: It really depends. There's

six families of stainless steels and there's also a new family called the super stainless steels that have more alloying elements in it. If you go to like a Best Buy --

ADMIN. JUDGE WARDWELL: Yes, I'm going to an Ace Hardware.

MR. GORDON: -- that's supposed to be a stainless steel refrigerator and you can bring a magnet if you want -- you really probably want Austenitic stainless steel and that's probably what you want in your sink also. But if you can bring a magnet along, you can see if it's really Austenitic or maybe ferritic stainless steel.

ADMIN. JUDGE WARDWELL: But, yet, I've bought these fender washers --

MR. GORDON: Yes.

ADMIN. JUDGE WARDWELL: -- and some of them it won't stick and other times it'll stick a little bit, it won't stick heavy, but it's not going to be like a --

MR. GORDON: Well, some of them, if you cold work Austenitic stainless steel, you get this diffuseness reaction, which you get martensite in the Austenitic stainless steel. Part of it will be transfer -- and you can measure this magnetically.

And this means you're -- it's a way of even measuring the amount of cold working you have by how much martensite you have in the Austenite.

ADMIN. JUDGE WARDWELL: So when I bring my magnet to my local Ace Hardware dealer, I can impress him, finally --

MR. GORDON: Yes. I mean --

ADMIN. JUDGE WARDWELL: -- instead of looking like a fool in the hardware store with my magnet go, oh, well, that one's had some cold working and this is pure --

MR. GORDON: Right.

ADMIN. JUDGE WARDWELL: -- Austenitic steel.

MR. GORDON: That's right. And steel comes --

ADMIN. JUDGE WARDWELL: Thank you.

MR. GORDON: -- these manufacturers make a stainless steel looking thing, but it's actually made out of carbon steel, but they just put a finish on it. So bring your magnet.

ADMIN. JUDGE WARDWELL: I thought we weren't going to get anything out of this hearing and I have been proven wrong.

(Laughter.)

ADMIN. JUDGE WARDWELL: Especially when I tell him, gee, I wonder if this is a cast Austenitic stainless steel or a CASS. Which brings us a nice segue into NRC's testimony, 197, Answer 161, Page 92, where the A/LAI, Action Item 7 requires an applicant or a licensee to perform a plant specific analysis of cast RVI components to demonstrate the components will remain capable of performing their intended functions during the period of extended operation. I guess my question for Entergy is that citing your testimony on Answer 175, Page 114, does not the Action Level 7 require that this analysis account for the potential loss of fracture toughness of the components due to both thermal embrittlement and irradiation embrittlement?

DR. LOTT: Give me a minute to get organized here. Can you give me the citation again?

ADMIN. JUDGE WARDWELL: Sure, the citation that it was citing was your testimony, Answer 175, Page 114 in response to that. I'm just asking, does not the A/LAI Number 7, which again you see in MRP 227 or in the Aging Management Plan, I may be able to find that, but does not that require, A/LAI 7, does that not require that the analysis account for the potential loss of fracture toughness of the components due to both thermal embrittlement, or TE as we'll call it, and irradiation embrittlement, IE as we'll call it?

MR. DOLANSKY: This is Bob Dolansky for Entergy. The answer's yes.

ADMIN. JUDGE WARDWELL: Say again?

MR. DOLANSKY: The answer is yes.

ADMIN. JUDGE WARDWELL: Okay. It does --

MR. DOLANSKY: Yes.

ADMIN. JUDGE WARDWELL: -- require that, okay. New York State in their testimony, 576, Page 5, Lines 1 through 17, and through Page 6, Lines 11 through 20, that in regards to NUREG/CR-7184, New York State contends that the following observations support previous opinions and testimony. And one of those, the first observation is that for cast materials, synergies may exist between TE, the thermal embrittlement, and the irradiation embrittlement. And, two, embrittled cast materials were observed to experience transgranular brittle cleavage and ductile tearing. And I guess the first question I will have for Entergy is, what does transgranular brittle cleavage and ductile tearing look like and how are they formed and when are they formed and when are they an issue?

DR. LOTT: Okay. We talked earlier about Austenitic stainless steels and ferritic stainless steels, or ferritic steels in general. For instance, a reactor pressure vessel steel is a ferritic steel. It's subject to a brittle fracture at low temperatures that cuts across the grain because it's a very flat surface, it's essentially a cleaving of the crystal structure in the grain, a very brittle failure. That does not tend to happen in the Austenitic stainless steels where the failures remain ductile, they fail by stretching of the material and eventually finding small dimples or ruptures in the material and pulling it apart. So there's a difference in the fracture process between a cleavage failure, which is a very low ductility, lower shelf reactor pressure vessel kind of failure, and the failures we see normally in stainless steels.

ADMIN. JUDGE WARDWELL: But say again, what is that transgranular brittle cleavage look like? It looks like a cut surface or a sheared surface or --

DR. LOTT: Yes. There's no deformation on the surface, it's very clearly flat and sometimes stepped.

ADMIN. JUDGE WARDWELL: So it just looks like a break in the surface?

DR. LOTT: Yes, it's a very -- it's a clean break, I guess is the best way to say it.

ADMIN. JUDGE WARDWELL: A crevice in a rock type of thing?

DR. LOTT: What's the best way to describe this?

ADMIN. JUDGE WARDWELL: Or a crack in a rock?

DR. LOTT: Yes. Certain rocks you would fail by cleavage, that's true.

ADMIN. JUDGE WARDWELL: And so what is ductile tearing? Is that what the --

DR. LOTT: Ductile tearing is the manner in which a crack in a ductile material would advance. So when this begins to fail, you begin to slowly -- it would continue to deform or continue to show some -- reconnected and you'd form small voids that would grown into little pockets that would give you these dimpled rupture effects on the surface of the specimen.

ADMIN. JUDGE WARDWELL: And so usually you'd say this ductile tearing is associated with dimples at the surface?

DR. LOTT: Yes. You can tell by looking at the surface that it's failed in this manner, under a microscope, not necessarily -- well, you might be able to tell by eye.

ADMIN. JUDGE WARDWELL: Okay. And this dimpled surface will appear before or after the tearing? I mean, can't you just tell by the split in the tearing or do you see the tear or --

DR. LOTT: Well, you can tell, obviously, by the load -- if you're tearing this part and you're measuring the load displacement, you can see that it's continually taking new force to pull it apart. Whereas a brittle fracture, a cleavage fracture, will obviously be real sudden.

ADMIN. JUDGE WARDWELL: So a transgranular brittle cleavage would be a definitive break, is that a fair assessment? Where the ductile tearing, you may not see a separation of the material, it just may be a necking down and possibly this dimpling on the surface that you talk about?

DR. LOTT: Well, eventually you will form, after it's necked down, you'll begin eventually to form cracks and those cracks will have these dimpled ruptured surface. We're talking about the fracture surface of the specimen, when it separates and you look at it, that's where you'd see the dimples.

ADMIN. JUDGE WARDWELL: Okay, thank you.

MR. STROSNIDER: Your honor, this is Jack Strosnider from Entergy. And maybe this will help you visualize it. When you have a cleavage fracture, it's flat because when it's transgranular, it's going along the atoms. Basically it's a shiny, when you look at that surface when it fails, it's shiny because it's very flat. When you have the ductile failure, because of the dimpled, because you have to break little pieces of material, it's got a duller surface and you can see it's not the flat sort of surface that you would see in a cleavage fracture.

ADMIN. JUDGE WARDWELL: But both of them are failure surfaces though that you're looking at?

MR. STROSNIDER: Oh, yes.

ADMIN. JUDGE WARDWELL: Okay.

MR. STROSNIDER: It comes apart.

ADMIN. JUDGE WARDWELL: Great. Thank you. Now, do you agree with New York's statements in regards to the observations from this NUREG 7184 that the Cast materials synergies may exist between thermal embrittlement and irradiation embrittlement and, two, the embrittled Cast materials were observed to experience these two types of failures?

DR. LOTT: I'm not sure that I -- I don't think I do agree with the term synergism. I think that has a meaning to me that I don't see proven in any of the data. I do agree that irradiation embrittlement and thermal embrittlement may both -- either one may happen in a Cast material.

ADMIN. JUDGE WARDWELL: Either one or both?

DR. LOTT: Well, certainly a material can be subject to both these conditions at the same time. It's just not clear that they interact to me synergistically.

ADMIN. JUDGE WARDWELL: Okay. Dr. Lahey, I want to ask you, did the NUREG 7181 use the term synergy between them or how did you -- what did they say in regards to the reference, I guess, to your first statement that for Cast materials synergies may exist between TE and IE?

DR. LAHEY: Your honor, this is Richard Lahey, New York. I would actually have to go back and look at it to see if it used that word. This word is used often, in fact it was one of the things that Judge McDade gave me as a homework to read something and it asked about the synergy between these two effects for cast stainless steel. And so it's not something I made up. It may or may not be in NUREG 7184, I'd have to check.

ADMIN. JUDGE WARDWELL: So that could be

your wording in regards to interpreting what they said --

DR. LAHEY: Yes. I mean, my --

ADMIN. JUDGE WARDWELL: -- in regards to the interaction between the TE and the IE?

DR. LAHEY: My personal belief is that you certainly can have both of them and the effect can be larger than each one separate. That's what I mean by synergy.

ADMIN. JUDGE WARDWELL: And say that again, what you mean by synergy?

DR. LAHEY: Synergy means that they reinforce each other and the combination is more than each effect separate.

ADMIN. JUDGE WARDWELL: But wouldn't it be more than just the sum also?

DR. LAHEY: Yes.

ADMIN. JUDGE WARDWELL: Because that would be the synergy --

DR. LAHEY: That's what I meant, I'm sorry.

ADMIN. JUDGE WARDWELL: -- you take this TE, you take IE and yes it's more than either one of them --

DR. LAHEY: Right.

ADMIN. JUDGE WARDWELL: -- but likewise,

it's more than the sum of the whole.

DR. LAHEY: Exactly.

ADMIN. JUDGE WARDWELL: Okay. Thank you.

DR. LOTT: Your honor --

ADMIN. JUDGE WARDWELL: Yes.

DR. LOTT: -- I think in that document, 7184, I have some notes on it here. It basically did show that irradiation, relatively low doses 0.08 dpa, can trigger a response in the material that similar to thermal embrittlement, these materials that are normally subject to thermal embrittlement. And what happened effectively was that thermal embrittlement by itself would produce an effect, the irradiation produced a similar effect without the thermal embrittlement, materials that were combined showed about the same. So that's why I'm concerned that they were not necessarily a synergistic effect. Yes, there was an effect of thermal embrittlement, yes, there was an effect of irradiation embrittlement, but to me, the suggestion that it's a synergistic effect would say that it could be larger combined than it is individually.

ADMIN. JUDGE WARDWELL: Okay thank you.

CHAIRMAN MCDADE: Are you saying that it isn't or you don't know?

DR. LOTT: I just don't -- it doesn't meet my definition for what I would call synergistic.

CHAIRMAN MCDADE: Okay. As I understood that, the definition you would use is that it's larger than the sum of the individual parts.

DR. LOTT: Right. If I had both, in the specimen if I had both irradiation and thermal embrittlement, the decrease was about the same as it was just due to irradiation and thermal embrittlement alone. I think that's Figure 142 in the document. That's probably 14.2, my handwriting is very bad, I'm sorry.

ADMIN. JUDGE WARDWELL: And, Dr. Lahey, I will leave you with this comment, that if you would like to explore 7184 and come up with the wording that they use that would support the synergistic effects of this, meaning that there's some words in there that state that the TE and the IE amplify one another to the point that the effect is greater than the sum of the two, then feel free to present that to us --

DR. LAHEY: Yes, sir.

ADMIN. JUDGE WARDWELL: -- whenever you have a chance to review that further within the context of why we're here this week.

DR. LAHEY: Yes, sir, I will. In fact

though, I've read that in a number of Argonne reports from researchers at Argonne.

ADMIN. JUDGE WARDWELL: Well, I'm mostly interested in -- I'm only interested in this regard in --

DR. LAHEY: Okay.

ADMIN. JUDGE WARDWELL: -- regards to this as applied to 7184. Thank you.

DR. LAHEY: Thank you.

CHAIRMAN MCDADE: Okay. And, Dr. Lott, you referenced Figure 142? Is that what you said?

DR. LOTT: We're looking it up right now. Can we get back to perhaps in a --

CHAIRMAN MCDADE: Sure.

ADMIN. JUDGE WARDWELL: Entergy's testimony, 616 again, Answer 106, Page 64, although stainless steel and nickel alloy RVI materials are also subject to irradiation embrittlement, they do not undergo a ductile to brittle transition or fail by brittle cleavage, even though the neutron exposure levels are much higher than those of the vessel. However, at fluences above the MRPA 175 screening threshold, it is recognized that these Austenitic stainless steels will experience decreases in fracture initiation toughness and in the resistance to ductile tearing. These effects have been explicitly considered in the MRP 227-A guidelines and in the RVI AMP implementation at Indian Point. And I guess my question for Entergy, I just want to confirm that your use of the word brittle cleavage is the same as this transgranular brittle cleavage that we were talking about earlier?

MR. STROSNIDER: Your honor, this is Jack Strosnider for Entergy. That's correct, but --

ADMIN. JUDGE WARDWELL: Thank you.

MR. STROSNIDER: -- I think the point here that needs to be understood, the point that's being made, we continue to discuss about highly embrittled materials. These Austenitic materials are embrittled and there is a reduction in fracture toughness. The thing you need to understand is when they test that fracture toughness, it still shows ductility, it's not failing by cleavage fracture. They're developing what's called a J-R curve, which requires some ductility in order to fail it. So, I just want to put this highly embrittled in context. There's embrittlement, but there's still ductility in these specimens, they're not failing by cleavage fracture, they don't go through the transition that the ferritic materials go through. It's very important in understanding how these materials can respond to various loads, including the accident loads.

ADMIN. JUDGE WARDWELL: And in this Answer 106 on Page 64, you say that they do not undergo ductile to brittle transition or failure by, I'm inserting now, the transgranular brittle cleavage, correct?

MR. STROSNIDER: Right.

ADMIN. JUDGE WARDWELL: So they don't fail by brittle cleavage, but they do still have resistance to ductile tearing, is that correct?

MR. STROSNIDER: Jack Strosnider for Entergy. Yes, that is correct.

ADMIN. JUDGE WARDWELL: Thank you. So in summary, the aging effects of RVIs do not include transgranular brittle cleavage, but do include ductile tearing. But how about some of the other effects that we've talked about? And I think they include cracking, dimensional changes, wearing, dimpling, stress relaxation, and void swelling cracking. Do they exhibit those types of effects? And we can stay with you, Mr. Jack, I can't see your name so I can't --

MR. STROSNIDER: Jack Strosnider for Entergy. Yes, they do. Those are the type of effects that are identified in the program and managed by the program.

ADMIN. JUDGE WARDWELL: Okay, thank you. Entergy's testimony, 722, Answer 8, Page 5, says the existing research also suggests that combined thermal aging and irradiation of representative CASS materials does not appear to lower toughness below what is expected for thermal embrittlement alone. And I guess I'd ask, what's the basis for this statement? Of anyone from Entergy that would like to respond.

DR. LOTT: Let me first respond by correcting my previous figure number. It's Figure 98 on Page 142. I had them switched in my notes.

ADMIN. JUDGE WARDWELL: Okay. Thank you.

DR. LOTT: And in fact, that figure I think actually makes the same point that he just said and that I said earlier, which was that the effect of thermal embrittlement and irradiation embrittlement in these high ferrite materials, and we haven't talked about necessarily high ferrite and low ferrite materials, these relatively high ferrite materials do show thermal aging and irradiation caused a similar effect.

MR. GRIESBACH: Your honor, this is Tim Griesbach from Entergy. I think one thing that we should point out is that the materials we're talking about at Indian Point in the lower support columns are known to be low in delta ferrite. It's that delta ferrite content that causes the thermal embrittlement that we keep referring to and the materials in these plants are known to not be susceptible to thermal embrittlement based on their low delta ferrite. So, by saying that, they should also not be affected by the synergistic effects of thermal and irradiation embrittlement.

ADMIN. JUDGE WARDWELL: Okay. Let me, I guess -- I'm getting swamped by too much information here, I think. And then diversion over to one chart and then I didn't know whether they answered my question or not is the way I interpreted it. You're testimony, Entergy's testimony, 722, Answer 8, Page 5, says the existing research also suggests that the combined thermal aging and irradiation of representative CASS materials does not appear to lower toughness below what is expected for thermal embrittlement alone. And my question to you is, what is the basis for that statement?

DR. LOTT: And, again, I think it's illustrated -- I was trying to answer your question with my previous answer. In the data that was shown, it shows exactly what --

ADMIN. JUDGE WARDWELL: And I'm sorry, I can't --

DR. LOTT: The data that's shown in that figure shows that you start -- and again, for three different materials, all high ferrite materials, where thermal embrittlement would potentially happen, they were either irradiated or thermally treated and the decrease in toughness in the J-integral toughness was basically the same in both cases. In other words, the thermal embrittlement and the irradiation embrittlement and the three of them together all gave very similar results.

ADMIN. JUDGE WARDWELL: But the cast materials that we're dealing with aren't high ferric are they?

DR. LOTT: The cast materials we're dealing are low ferrite, so we would not expect to see thermal embrittlement, nor would we expect to see any large amount of irradiation embrittlement by the same mechanism.

ADMIN. JUDGE WARDWELL: So why does that figure, which deals with high ferric, provide a basis for your statement that the existing research suggests that combined thermal aging and irradiation of representative CASS materials did not appear to lower toughness below what is expected for thermal embrittlement alone?

MR. STROSNIDER: This is Jack Strosnider for Entergy and let me see if I can clarify this.

ADMIN. JUDGE WARDWELL: Sure.

MR. STROSNIDER: I think the first part of the answer that Dr. Lott gave is a generic discussion about the laboratory data --

ADMIN. JUDGE WARDWELL: Fine.

MR. STROSNIDER: -- which includes high delta ferrite materials. When we talk about the material at Indian Point, in particular the lower columns, they are low delta ferrite not susceptible to thermal aging, and, therefore, this -- what you see in the generic data is really not applicable at Indian Point because of the specific material that's at Indian Point. So a generic part of the response and a plant specific part of the response.

ADMIN. JUDGE WARDWELL: And so the plant specific, you've made a statement that, again, I'll read for the third time. The existing research also suggests that combined thermal aging and irradiation of representative CASS materials, which is what you have at Indian Point, correct?

MR. STROSNIDER: Read that again, please?

MR. GRIESBACH: This is Tim Griesbach. The --

ADMIN. JUDGE WARDWELL: CASS material, you have -- you're dealing with CASS material. For those CASS materials at Indian Point, do not appear to lower toughness below what is expected for thermal embrittlement alone. That was your statement. My question is, where did that come from?

MR. DOLANSKY: This is Bob --

ADMIN. JUDGE WARDWELL: What is the basis for it? Point me to a graph that demonstrates not what it isn't, I want to see a graph for what it is or some other discussion of why you came up with that statement or how you came up with that statement.

MR. DOLANSKY: This is Bob Dolansky with Entergy. I believe the graph that Dr. Lott pointed to contains CF-8 material and that is --

ADMIN. JUDGE WARDWELL: Whoa, what's CF-8 material? Now we've got another material.

(Laughter.)

MR. DOLANSKY: -- which that is what we have at Indian Point.

ADMIN. JUDGE WARDWELL: That's CASS?

MR. DOLANSKY: That's CASS and it's --

ADMIN. JUDGE WARDWELL: Great.

MR. DOLANSKY: -- low delta ferrite.

ADMIN. JUDGE WARDWELL: Yes, great.

MR. DOLANSKY: That is, I believe, and Dr. Lott can correct me if I'm wrong, but that figure that he gave you contains the low delta ferrite CF-8 material that's CASS that we have at Indian Point.

ADMIN. JUDGE WARDWELL: Okay. And you're referring to Figure 98 on Page 142?

MR. DOLANSKY: Yes.

ADMIN. JUDGE WARDWELL: Okay. Because by the time we get with putting the record together and look at the transcript, we want to make sure we have it.

MR. DOLANSKY: This figure here.

ADMIN. JUDGE WARDWELL: And so it's on our screen now?

DR. LOTT: Yes.

ADMIN. JUDGE WARDWELL: Okay. Right? Is that the one you're referring to?

MR. DOLANSKY: Yes.

DR. LOTT: Yes.

ADMIN. JUDGE WARDWELL: Okay. So how does that support this statement that CASS materials do not appear to lower toughness below what is expected for thermal embrittlement alone?

DR. LOTT: Well, I'm not sure what part of your question I'm missing. I mean, if you look at the data here, both irradiation and aging produce a similar decrease in toughness and the material that is both irradiated and aged has a very similar, particularly in the CF-8 material, is a similar toughness. Now, in the CF-8 materials in Indian Point, we don't expect to see these decreases due to thermal aging because the CF-8 material there is low ferrite.

MR. COX: This is Alan Cox with Entergy. And let me give you my layman's interpretation of what this drawing shows. It shows --

ADMIN. JUDGE WARDWELL: Sure, just get a little closer to your mic though because I can hear you better.

MR. COX: The tall bar on the graph is the material at the beginning and then the arrows show --

ADMIN. JUDGE WARDWELL: And we're looking at the blue tall bar, which we're going to focus only on the CF-8, is that correct?

MR. COX: Sure. I'm just -- the response is similar in all three cases. But if you --

ADMIN. JUDGE WARDWELL: Sure.

MR. COX: -- the center one, the blue tall bar is the starting condition. The dotted line that says irradiation shows the decrease in the fracture toughness as you irradiate the material. The dotted line that says aging is the thermal aging. And you can see it lowers -- both of those effects cause a drop in the fracture toughness. The third bar, you see another irradiation on the dotted line that goes from the shorter blue bar to the pink/red colored bar in front, that's the additional effect, that's basically the combined effect of the --

ADMIN. JUDGE WARDWELL: Got you.

MR. COX: -- thermal aging and then the irradiation. And you see it has a slightly lower value than the irradiation alone, a little bit lower than the thermal alone, but it's certainly not greater than the sum of the two effects.

ADMIN. JUDGE WARDWELL: And what is CF-3 material?

DR. LOTT: It's just a different specification of cast stainless steel.

ADMIN. JUDGE WARDWELL: Okay. So all of these are cast stainless steel materials?

DR. LOTT: Yes.

ADMIN. JUDGE WARDWELL: Dr. Lahey, any

comments on how these graphs apply to what CASS materials appear to do in regards to lowering toughness below what is expected for thermal embrittlement alone when you combine thermal aging with irradiation of those?

DR. LAHEY: Just a request for a little bit more information from Entergy. My understanding is the delta ferrite in IP2 is like 14.6 percent, is that correct?

ADMIN. JUDGE WARDWELL: Well, why don't I ask the question. You tell me what you're interested in and -- I want to make sure you're not --

DR. LAHEY: Well, the screening criteria is 15 percent and I'm trying to understand if that's what they mean by low delta ferrite, the 14.6.

ADMIN. JUDGE WARDWELL: What do you mean by low delta ferrite?

MR. AZEVEDO: Yes, your honor. This is Nelson Azevedo for Entergy. It is true that both Units 2 and 3 have low delta ferrite as measured by the screening criteria 15 percent. So they're both below 15 percent.

ADMIN. JUDGE WARDWELL: And do you have any idea how far below?

MR. AZEVEDO: Yes. Unit 2 is 14 and

change, Unit 3 is a little bit lower, like 11, 12.

ADMIN. JUDGE WARDWELL: Okay. Thank you. Dr. Lahey, any other comments or --

DR. LAHEY: Yes. My comment is I wish my colleague Dr. Duquette was here because this is his field, he's a world class metallurgist. And he would have a lot of comment on this and all I can give you is secondhand information because this is not my field.

ADMIN. JUDGE WARDWELL: But he's not even a witness for this Contention, so even if he was here, he would be a spectator.

DR. LAHEY: Okay. Well, I can tell you what he would say if you want to hear that.

ADMIN. JUDGE WARDWELL: No. No, I'd like to hear --

CHAIRMAN MCDADE: Just tell us what you would say.

ADMIN. JUDGE WARDWELL: -- what your professional interpretation might be of anything, but only from what you're comfortable saying in regards to your professional background.

DR. LAHEY: Well, concerning the data that we have on the screen, the data speaks for itself. If this is good data, that's what it is.

ADMIN. JUDGE WARDWELL: Okay. Thank you.

CHAIRMAN MCDADE: Okay. It may speak for itself, but it doesn't speak loud enough for me to understand it. Okay. We start off --

ADMIN. JUDGE WARDWELL: You're never going to understand this chart.

(Laughter.)

CHAIRMAN MCDADE: Okay. The low delta ferrite, start from the premise that it's below 15 percent, it's 14 or 11 percent. How does that inform your conclusion here?

DR. LAHEY: Well, if you want me to give you some secondhand information, then I can do that. But I can't give you any firsthand information other than looking at the graph and what it shows to me is there's obviously no synergism shown in this data.

CHAIRMAN MCDADE: Okay. There's no synergism shown in this data, but from your experience, do you believe when you have the low delta ferrite that you would expect synergism?

DR. LAHEY: Not necessarily. I think it depends on how the duplex structure is arranged. If it's arranged in a certain way, it can have a different effect than if it's arranged in a different way. And unfortunately the only way you know that is with a destructive examination. So, I think you do have to rely on data that shows, here -- when I am talking about synergism, there's a number of other experiments, not necessarily for these type of percentages of delta ferrite, which do show synergism. But it may be, apparently is, for the materials used in Indian Point, this is the result you get.

CHAIRMAN MCDADE: So you're saying that based on data that you have observed, if the percentage was not 11 percent, but was 20 percent or 40 percent, it would have a different effect on your conclusion with regard to synergism?

DR. LAHEY: That's my understanding of the reading that I've done, yes.

CHAIRMAN MCDADE: Okay, thank you.

ADMIN. JUDGE KENNEDY: Dr. Lahey, this is Judge Kennedy. If I looked at the material CF-3, look at the data there, does that show more synergism than CF-8 does?

DR. LAHEY: Well, if I understand the way they're doing it, it would show less. The one on the right, the little pink or red or whatever color that is on the right is lower than the one on the left. So it's a little odd that the thermal aging plus irradiation would give you lower, but it depends on the level of irradiation, I suppose.

ADMIN. JUDGE KENNEDY: All right, thank you.

ADMIN. JUDGE WARDWELL: Yes. I guess that raises a question I hadn't really come up with. What is this J-integral that we're plotting on the Y axis?

DR. LOTT: It's a measure of the ductile fracture toughness.

ADMIN. JUDGE WARDWELL: So that's the measure of the toughness?

DR. LOTT: The toughness being the resistance to crack initiation. And, again, I'll use that word, let me qualify. In a J-integral test, you would start with a crack specimen and you would basically be measuring how much it would take to reinitiate and grow that crack.

ADMIN. JUDGE WARDWELL: To reinitiate and what that crack?

DR. LOTT: Grow it.

ADMIN. JUDGE WARDWELL: Grow that crack. So what do low values of J mean? That it's not susceptible to cracking or susceptible to crack or --

DR. LOTT: It's not the susceptibility to cracking, it's susceptibility -- the impact of cracking on the ability to maintain a load.

ADMIN. JUDGE WARDWELL: On the ability -- so what are --

MR. STROSNIDER: This is Jack --

ADMIN. JUDGE WARDWELL: -- the low values mean?

MR. STROSNIDER: This is Jack Strosnider for Entergy, let me see if -- first of all, in terms of the J-integral and what it is, it can be related to the amount of energy that's required to allow the crack to tear through the material, the preexisting crack. It can be related to the energy for that. So the lower the J value, the less energy it takes. But this J-integral approach was developed specifically for ductile materials. If the material is not ductile, if it's going to fail in the cleavage mode that we talked about earlier, there's a different measure that's used for that. So, in this case, it's related to energy, the lower that value, the lower the energy. But in every case, it's showing some ductility.

ADMIN. JUDGE WARDWELL: And so, while the CF-8 shows that we reach the same level of the J-integral with irradiation as we do with aging and irradiation, the CF-3 material shows that we end up with a lower value with the combination of the aging and the irradiation --

MR. STROSNIDER: Right.

ADMIN. JUDGE WARDWELL: -- than just irradiation, which means less energy is needed to grow the crack. Am I interpreting that correctly?

DR. LOTT: I supposed you could interpret it that way. I must admit, I think it's within the scatter in the data, within the accuracy of the data. To make that conclusion would be difficult in my mind.

CHAIRMAN MCDADE: Okay, Dr. Lott, excuse me. Could you either move yourself closer to the microphone or the microphone closer to you?

DR. LOTT: Okay, sorry.

CHAIRMAN MCDADE: Okay, thank you.

ADMIN. JUDGE WARDWELL: So you think that the data is plus or minus almost 100 percent because it seems to be about half of what it is before and if that's the noise, then that's accuracy of what we're dealing with here in this graph? Can you see how I reached that conclusion?

DR. LOTT: Yes, I see how you reached that conclusion. It's hard for me to answer that question.

ADMIN. JUDGE WARDWELL: Fine, thank you.

DR. LOTT: Again, I think it's important to point out that basically the materials that we're dealing with, we don't expect to see large amounts of thermal embrittlement at all. Now, these materials are higher in ferrite and we would expect to see that happen. I think all materials measured at least at levels greater than 20 percent.

ADMIN. JUDGE WARDWELL: Boy, you had me right to the very end. These materials, what are you referring to as these materials?

DR. LOTT: The materials in this graph, the CF-3, the CF-8, and the CF-8M. The particular materials that were tested.

ADMIN. JUDGE WARDWELL: But all of these -- now we're back to where we started I think.

DR. LOTT: I'm sorry.

ADMIN. JUDGE WARDWELL: What is the ferric content of these materials on this graph?

DR. LOTT: The ferrite content in all three cases I believe is greater than 20 percent, measured.

MR. DOLANSKY: Your honor, maybe I could help a little bit. This is Bob Dolansky with Entergy. CF-8 can have a range of delta ferrite. Indian Point 3 has CF-8 materials. We actually went and did the research, pulled our CMTRs, our delta ferrite was below 15 percent at both IPEC units. That doesn't mean that all CF-8 is below 15 percent. Does that help?

ADMIN. JUDGE WARDWELL: And so, if I looked at the rest of this --

MR. COX: This is Alan Cox --

ADMIN. JUDGE WARDWELL: -- Exhibit 488B, is that what it is? And that is an 83 page report, it would tell me what the ferric content of this is?

DR. LOTT: I believe so.

MR. COX: Judge Wardwell, this is Alan Cox for Entergy. The paragraph immediately above this graph says that all the samples tested were high delta ferrite samples. And the report may say somewhere what that is in terms of a number, but it does say it's high delta ferrite.

DR. HISER: Dr. Wardwell, the information you're looking for is in New York State 488A, Page 5, Table 1, provides chemical composition of the materials. And for the CF-3, it's measured 24 percent, CF-8 measured 23 percent, and the 8M is 28 percent.

ADMIN. JUDGE WARDWELL: So would -- now I'm going to go back to Entergy. Thank you for that, Dr. Hiser, from the Staff. Back to Entergy, you would call these samples that were used to generate this figure in 488 high ferric content samples?

DR. LOTT: And perhaps I could offer further explanation. In the evaluation not in the reactor internals, but in other reactor components, particularly piping components, the guidelines are effectively to consider thermal embrittlement in materials containing more than 20 percent ferrite. So, in our minds I guess, that is the threshold. I'm not sure there's an absolute threshold, but certainly there's no requirement in analyzing piping materials to consider the loss of toughness due to thermal aging for materials that are less than 20 percent ferrite.

ADMIN. JUDGE WARDWELL: So this graph is not representative of the CASS materials we have in the reactor vessel internals at Indian Point?

DR. LOTT: No, because we would not in that case expect to see thermal embrittlement, or certainly not this level of thermal embrittlement.

ADMIN. JUDGE WARDWELL: So, which gets me back to the original question I had.

DR. LOTT: Well, I guess I could only suggest that the only data had to offer one way or the other on the original question, which was is there a synergistic effect, was this data. I don't see, again, in this data -- what it suggests to me quite frankly is that the properties of the duplex steel, this material that has 80 percent Austenite and 20 percent ferrite or whatever, is affected by the embrittlement of the ferrite phase. There are two ways that you could end up embrittling the ferrite phase. One is by thermal aging, the other is by irradiation. This would indicate that it doesn't really matter what you do to embrittle that phase, once you have it embrittled, you have a similar effect on the material itself. But that's an interpretation --

CHAIRMAN MCDADE: Okay. Following --

DR. LOTT: -- the best I can give you.

CHAIRMAN MCDADE: Following up, Dr. Lott, using this chart. The delta ferrite for the CF-8, 23 percent based on the testimony that Dr. Hiser just gave, when you look at the difference between the effect of irradiation and aging plus irradiation, you have very little difference, minimal, drawing your conclusion that there is no synergistic effect or minimal synergistic effect. In the Indian Point situation, we have, at IP3, the delta ferrite is approximately 11 percent. So this chart suggests to you there is no synergistic effect, but the fact that there's a lower delta ferrite, meaning it would be even less susceptible to the heat aging, would indicate to you that there would be even a lesser impact on the metal used at Indian Point 2 and 3 at 14 and 11 percent?

DR. LOTT: Yes.

CHAIRMAN MCDADE: Am I --

DR. LOTT: Yes. I --

CHAIRMAN MCDADE: I just want to -- I repeat this just to make sure that I'm hearing what you're saying and there's not something --

DR. LOTT: I believe you've well interpreted what I'm trying to say.

CHAIRMAN MCDADE: Okay, thank you.

ADMIN. JUDGE WARDWELL: But then I'll go back to my original question based on your original statement in your testimony, which this is where I'm trying to get to. This chart does not apply to Indian Point materials, correct?

DR. LOTT: No, we have no measurements on the Indian Point materials.

ADMIN. JUDGE WARDWELL: The statement that was made on your testimony, 722, Answer 8, Page 5, you've made the statement, the existing research also suggests that combined thermal aging and irradiation of representative CASS materials, and I assume you mean representative of what's there at Indian Point, does not appear to lower toughness below what is expected for thermal embrittlement alone. And so my question is, what is the basis for that statement? And if you're going to use this figure, I'd like to know why you can use this figure to extrapolate for something that isn't representative of what's at Indian Point.

MR. COX: Let me take a shot at that and then Dr. Lott can chime in. But I think the way that I would look at that graph is that if you have little thermal aging, I mean, you can imaging the aging line on the graph would show much less of a decrease than what you show there. So it seems very reasonable to say if you have a lot of irradiation embrittlement, a lot of thermal embrittlement, and you see very little difference in the combined effect than you do from the irradiation effect, if you reduce the thermal aging embrittlement, like you would expect to see from material with a low delta ferrite, you would have no reason to expect a larger difference when you combine irradiation with the thermal.

So, I mean, I think this is not the exact same material, but knowing what we know about the behavior of the material under irradiation and thermal embrittlement, we would expect the same result. And that is very little change when you combine the two effects since the thermal embrittlement would be even less of an effect, there would be very little change from the combined effect and what you see from the irradiation effect alone.

DR. LOTT: Can I request -- I offered this slide because I thought it was going to quickly address your issue, obviously it did not. If we could have some time to look at the answer to your question and come back to you with it, because I think it's going to require us to go through more than one document in order to put the answer together.

ADMIN. JUDGE WARDWELL: That's fine.

DR. LAHEY: Your honor --

ADMIN. JUDGE WARDWELL: Yes.

DR. LAHEY: -- could I say, if I understand this graph correctly now, the CF-3 shows the synergism that you are asking about, whereas the CF-8 does not. And so I think that's what they're talking about, but for a lower percentage. But over here you can clearly see the synergism in terms of the integral or the J, it puts the onset of the crack occurs.

ADMIN. JUDGE WARDWELL: In your opinion?

DR. LAHEY: Well, if I believe the heights, you go down with irradiation, you go down with thermal aging, and if you add on the irradiation, you're significantly below than just each one.

CHAIRMAN MCDADE: But what you're saying, and again, I'm repeating this to make sure I hear what you're saying and understand it, is that the CF-3 shows the synergy even though the CF-8 does not on this graph.

DR. LAHEY: It appears so.

CHAIRMAN MCDADE: Okay. And would you explain the rationale for, theoretically, why the CF-3 would show apparently some synergistic effect whereas the CF-8 did not? Dr. Lott?

DR. LOTT: Well, I think there's two issues here. One, as I indicated before, there's a fair amount of scatter in any measurement of toughness and in CASS materials in particular. So I'm not sure about the -- I'm not trying to cut these numbers that fine. And, again, I'm not sure that -- and I don't want to get into a semantic argument about the meaning of synergism, but again, these effects are not larger than the combined individual effects. The effects seem to be similar for a large portion of the behavior.

ADMIN. JUDGE WARDWELL: But to also be sure that I'm clear here that my question had nothing necessarily to do -- my question had nothing to do with synergism. My question had to do with where is the basis for your statement that you made in your testimony? And if it's a professional estimate, that's fine. If it's a wild guess, that's fine. If it's based on something else, I would like to hear about it. Thank you.

Entergy's testimony, 722, same Answer on the same Page, because it goes on to state, we may have to pull this back too, that given the ongoing research in this area, the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) developed conservative screening criteria to identify components that are potentially susceptible to the effects of such mechanisms. And let me ask you, are these such mechanisms the thermal embrittlement and the irradiation embrittlement and the combination of the two? Is that what is meant by the such mechanisms?

DR. LOTT: Yes, I believe that's true.

ADMIN. JUDGE WARDWELL: And what is that conservative screening criteria that EPRI developed? Was that the 15 percent screening criteria or is it some other criteria?

DR. LOTT: The screening criteria that EPRI has proposed I believe is that 20 percent value with a fluence level of about one dpa.

ADMIN. JUDGE WARDWELL: You tailed off at the end and --

DR. LOTT: I'm sorry, my voice is tender. No, I do not believe it's the 15 percent. I believe the EPRI proposal is 20 percent. I might have to check to see if that number is not in our file testimony. I'll have to look and see.

ADMIN. JUDGE WARDWELL: What screening criteria are we talking about that EPRI is proposing? A screening for what?

DR. LOTT: Basically a screening for thermal and irradiation embrittlement that's used to identify components that would be -- again, we'd have to look at the effect of -- when we go to the -- first of all, when we identified in our screening process originally for MRP 227, we identified all of the cast materials as potentially susceptible to thermal embrittlement because we did not have any ferrite contents for them. And so we couldn't say if they were even above or below 20 percent.

Since that time, we've done a large amount of working with customers such as Entergy and we've identified that no materials that are greater than 20 percent in the Westinghouse internals components that are cast stainless steel. So, that would be -- but that's not in our -- it's a conservatism, because we've never undid the questions about thermal embrittlement. And, in fact, that's what the EPRI concerns are about is the, what are the threshold values for identifying thermal and irradiation embrittlement? Should there be a different fluence level for determining irradiation embrittlement susceptibility in cast stainless steels?

ADMIN. JUDGE WARDWELL: And do you need to time to also --

DR. LOTT: Let me locate that --

ADMIN. JUDGE WARDWELL: -- determine what that EPRI screening criteria is?

DR. LOTT: Yes, right. Yes, let me --

ADMIN. JUDGE WARDWELL: Can I ask this question of, I don't know who answered it before, who stated it before, but about a half hour ago when we started on these two simple questions that I thought we were going to zoom by --

DR. LOTT: Yes.

ADMIN. JUDGE WARDWELL: -- this 15 percent of ferric content was brought up as a screening criteria. Is that a different screening criteria or did I hear that wrong or where did that come from?

MR. GRIESBACH: This is Tim Griesbach from Entergy. Molybdenum also plays a big part in this thermal embrittlement of cast materials. So there is --

ADMIN. JUDGE WARDWELL: How are you simplifying this discussion?

(Laughter.)

MR. GRIESBACH: There are two different screening criteria depending on the molybdenum content. If it's less than one half percent or greater than one half percent. If it's greater than one half percent --

ADMIN. JUDGE WARDWELL: Of molybdenum?

MR. GRIESBACH: -- of molybdenum and high or low delta ferrite, there are different criteria. And then there's a separate criteria based on fluence also. So we will produce that for you and hopefully that will simplify things, especially as it applies to the Indian Point materials.

ADMIN. JUDGE WARDWELL: Can I get one of my questions out of the way beforehand, and that is, what was this 15 percent we brought and should I just forget about that and you'll trump it with more specific --

MR. AZEVEDO: Yes, your honor --

ADMIN. JUDGE WARDWELL: I see Mr. Azevedo warming up to the old mic.

MR. AZEVEDO: This is Nelson Azevedo for Entergy. Yes, so let me see if I can explain real shortly. The 20 percent came from EPRI, that's EPRI developed. The 15 percent came from the NRC.

ADMIN. JUDGE WARDWELL: And the 20 percent is ferric content?

MR. AZEVEDO: Ferrite content, right. So EPRI proposed 20 percent, the NRC used 15 percent.

ADMIN. JUDGE WARDWELL: Dr. Hiser, do you agree that you used 15 percent and EPRI used 20 percent for a screening criteria?

DR. HISER: Yes, that's correct.

ADMIN. JUDGE WARDWELL: Okay.

DR. HISER: Fifteen percent is for irradiated cast stainless steel.

ADMIN. JUDGE WARDWELL: Okay. Great. And so you will still get back to us with the molybdenum and anything else you want to add to that, for whatever you want to? Great. Moving on to other welds, New York State testimony, 576, Page 4, Line 8 through 23, and moving through to Page 5, Lines 1 through 17, states that in regard to NUREG/CR--7185, New York State contends that, one, cast and Austenitic stainless steel welds have a duplex structure and may experience thermal embrittlement, which may increase the hardness and tensile strength of a material, but decrease ductility, fracture toughness, and impact strength of cast materials and Austenitic stainless steel welds.

Two, IE is a concern for CASS components for fluences greater than two times ten to the 20, and that's N per square centimeters, so I believe that's neutrons per square centimeter, allegedly equivalent to ten displacements per atom, or dpa, and irradiation makes cast materials and Austenitic stainless steel welds more susceptible to irradiation assisted stress corrosion cracking, or the IASCC. And, three, that IASCC increases the crack growth rate of cracks induced by stress corrosion cracking, but there is allegedly virtually no data above the ten dpa, although some reactor vessel internal components may experience several hundred dpa.

And, fourth, that there is possibly synergy between TE and IE, although the report stresses the need for more information to develop reliable failure curves. And, five, that TE could make the welds associated with the pressurizer spray nozzles vulnerable to seismic and thermal pressure shock loads. Let me start with Dr. Lahey who made this statement. To your knowledge, is a pressurizer spray nozzle an RVI component covered by the RVI AMP within the list of those components?

DR. LAHEY: Not --

ADMIN. JUDGE WARDWELL: Or you don't know for sure?

DR. LAHEY: It's not a reactor vessel internal at all. It's an external pressure boundary.

ADMIN. JUDGE WARDWELL: Okay.

DR. LAHEY: And I never called it an RVI.

ADMIN. JUDGE WARDWELL: Okay.

DR. LAHEY: Also, there was a typo in my thing, it's not ten dpa. If you look at that chart we showed before, it's more like 0.3 and that was pointed out by Entergy --

ADMIN. JUDGE WARDWELL: Okay.

DR. LAHEY: -- and rightly so. So, other than that, the new information to me that I didn't have when I expressed this concern was the composition of the weld rods that they actually used. They used 308 weld rods for stainless steel 304 and 309 weld rods for stainless steel 316. And this does give a duplex structure, but the net effect is more like less than ten percent. So it's likely not subject to thermal embrittlement. And what I was made aware of based on the feedback from Entergy or NRC, I don't recall which one, was that Dr. Chopra had a very high delta ferrite content in his welds. And that made the difference.

ADMIN. JUDGE WARDWELL: So is your net conclusion that you don't believe this is an issue now?

DR. LAHEY: I don't believe the thermal embrittlement of those particular welds is an issue, if in fact that is how they did their welds.

ADMIN. JUDGE WARDWELL: Okay. And I'll turn to Entergy, as just represented by Dr. Lahey, is that how the welds were done? Or do you have some other nuances to discuss in regards to how the welds were performed?

MR. AZEVEDO: This is Nelson Azevedo for Entergy. No, your honor, what's been said is correct.

ADMIN. JUDGE WARDWELL: Okay. Thank you. So there was a list that I have here, without reading through it I guess -- if you don't believe that welds are now an issue, you're comfortable with the welds that they were performed as represented by Entergy?

DR. LAHEY: Well, the welds we were just

talking about were the welds associated with the pressurizer spray nozzle. There may be concern with the welds inside of the reactor vessel internal --

ADMIN. JUDGE WARDWELL: Okay.

DR. LAHEY: -- component welds, like the core barrel.

ADMIN. JUDGE WARDWELL: Okay, good. So let's continue then. I guess I'll go with Staff now. Are the pressurized spray nozzles to safe end welds considered part of RVIs?

MR. POEHLER: This is Jeffrey Poehler for the Staff. No, they are not considered part of RVI.

ADMIN. JUDGE WARDWELL: And are they handled under some other Aging Management Program?

MR. POEHLER: Yes.

ADMIN. JUDGE WARDWELL: Do you agree that there is a decrease in ductility, fracture toughness, and impact strength of cast materials and Austenitic stainless steel weld? And, if so, does this drive any changes need to thermal embrittlement screening criteria or other aging management procedures?

MR. POEHLER: Can you clarify -- under what conditions are you referring to? Combined irradiation and thermal exposure or one or the other?

ADMIN. JUDGE WARDWELL: One or the other.

MR. POEHLER: In cast Austenitic stainless steel, yes, we agree there is a decrease in fracture toughness that can occur to either irradiation or due to thermal aging.

ADMIN. JUDGE WARDWELL: Maybe I should back up a bit because I'm a little bit confused. What welds are left that are part of the RVIs? Are there a whole host of them or are there only a few isolated ones that part of RVIs or --

MR. POEHLER: Are we discussing welds or castings?

ADMIN. JUDGE WARDWELL: Well, I believe it was welding of cast material.

MR. POEHLER: Yes, there are some welds in cast materials, such as the lower support columns.

ADMIN. JUDGE WARDWELL: Such as the what?

MR. POEHLER: The lower core support columns.

ADMIN. JUDGE WARDWELL: And are those -- do you handle those as separate or do you handle them within the component itself, the support column itself?

MR. POEHLER: We handle those as part of the component itself.

ADMIN. JUDGE WARDWELL: In regards to those welds, though -- well, let me ask Dr. Lahey. In regards to welds as cast material, given the representation of Entergy on how those welds were performed, is it your understand that, that is also how the reactor vessel internals were welded, for any welds that were needed for reactor vessel internals?

DR. LAHEY: Your honor, what we were talking about before was welds outside of the pressure vessel using wrought Austenitic stainless steel and the type of weld rods that they used to perform that. Which was one of my concerns until I understood what they actually did. Inside the reactor pressure vessel for some of these other components, it's a different story.

ADMIN. JUDGE WARDWELL: And is that story the same story you believe applies to the welds as well as the -- that you also have for the internals themselves?

DR. LAHEY: Well, depending on the delta ferrite content, as long as it's above a certain level, you heard 15 percent, then you could have this thermal embrittlement as well as irradiation embrittlement. And there's certainly welds that can be subjected to both of those effects.

ADMIN. JUDGE WARDWELL: But if it's below the -- if it's screened out below the 15 percent ferric content, then you believe that thermal embrittlement is not an issue, it's just the irradiation embrittlement?

DR. LAHEY: Generally, yes, but there are some exceptions if you have a linkage within the weld material itself of the delta ferrite. But generally you have to have more of it to have this effect.

ADMIN. JUDGE WARDWELL: Let me turn to Staff, how would anyone handle leakage out of a weld and the impacts of that? Is there a need to evaluate potential leakage out of a weld?

MR. POEHLER: This is Jeffrey Poehler from the Staff. Are you referring to pressure boundary welds or --

ADMIN. JUDGE WARDWELL: I'm referring to welds of the RVIs that we're dealing with under the RVI AMP, whatever they are.

MR. POEHLER: Since the welds in the RVI do not serve a pressure boundary function, then leakage is not a failure criteria for those welds.

ADMIN. JUDGE WARDWELL: And in your review of the AMP and the components that are in the RVI AMP, have any of the components been welded such that the weld and the component itself have a higher ferric content than 15 percent? Or were they all screened out?

MR. POEHLER: We didn't look at the ferrite content of the welds. So weld filler metals used for Austenitic stainless steel welds, they tend to have lower ferrite content in general than cast stainless steels. So thermal aging is not generally an issue, has not generally been considered an issue for those welds. So in the screening criteria used to develop the AMP at Indian Point and MRP 227-A, they didn't screen-in thermal embrittlement for the welds.

ADMIN. JUDGE WARDWELL: How many RVIs are made out of -- are composed of cast materials?

MR. POEHLER: Basically, there's a handful of components, the lower core support columns and then at Indian Point it's only the upper portion of those columns, which they call the column cap, is cast stainless steel. You also have the lower core support forging at the very bottom of the core barrel. There may be a couple other ones, but those are the main ones.

ADMIN. JUDGE WARDWELL: Let me turn to Entergy and see if they want to clarify. Well, let's start off with, how many of the RVIs are, to your knowledge, cast materials versus the wrought

Austenitic?

DR. LOTT: I believe that we identified that in our filed testimony. I'm looking at Page 62, I think it's Question 105, and it identifies the upper instrumentation conduit supports --

ADMIN. JUDGE WARDWELL: Sorry, you're going to have speak into the mic and a little slower.

DR. LOTT: Okay. It identifies the upper instrumentation conduits and supports, upper support column assemblies, the lower support casting. That should be lower support column assemblies. Oh, no, I'm sorry, that's right. The upper instrumentation conduits, upper support column assemblies, and the lower support casting.

ADMIN. JUDGE WARDWELL: And how many of those were screened out due to ferric content? Or moly content or whatever?

DR. LOTT: In the original screening process, none of them were screened out because everything was screened in based on lack of knowledge at that time of the ferric content.

ADMIN. JUDGE WARDWELL: Okay. I'll wait, I guess, until I get to those individual components. Because I thought somewhere, it was my impression that only those upper caps of the lower supports were cast materials that were part of the aging management plan, but I don't know that. I'll have to --

DR. LOTT: But they are cast materials part of the aging management plan, but they were not necessarily -- well, first of all, most of them are not irradiated at all. So this question of irradiation embrittlement doesn't come in. And I think there was also -- some of them were screened out based on lack of structural requirement. In other words, there were just no --

ADMIN. JUDGE WARDWELL: Well, I'll be able to get to that quote once I get to it. I just can't find it and I don't want to spend time looking for it. I'll come upon it as I work my way through. But I want to get back to welds, I think. How many -- how do you handle the welds in your AMP, for both the wrought and the cast materials? Are they part of the individual component or are they considered separately?

DR. LOTT: There are some key welds that are considered separately. And those welds were, for instance, the core barrel welds, we knew we were concerned about them, they're identified separate. There are other components that contain welds and in the screening process, one of the things we identified was which components contained welds and which did not. And then the primary concern with welds was large structural welds where concern for stress corrosion cracking, not necessarily embrittlement.

ADMIN. JUDGE WARDWELL: And, so, how many of these special welds were designated as individual components when you split this out?

DR. LOTT: I'd have to go back and count, I --

ADMIN. JUDGE WARDWELL: Okay.

DR. LOTT: -- couldn't answer that.

MR. DOLANSKY: This is Bob Dolansky from Entergy. The Table 1 on Page 87 of our testimony lists the primary components and the expansion components. The primary components lists if it's a weld or not. Does that answer your question?

ADMIN. JUDGE WARDWELL: To a certain degree. I was hoping you had an approximate number of those that were there that related to split out welds, but that's fine. I don't need that right now.

MR. DOLANSKY: Okay.

ADMIN. JUDGE WARDWELL: I would like that number to get a feeling for how many there are and what's being -- so I can then refer to that table to see what's being handled for them.

MR. DOLANSKY: I could go through this table if you wanted and --

ADMIN. JUDGE WARDWELL: Yes, at some point and then just come up with a number and that will be sufficient so I make sure I see the same number that you do when I -- if it does come up in our decision writing, I'm able to refer to that and handle any discussion associated with it.

MR. DOLANSKY: Just to clarify, what I'll do the research on is for each primary component, if it's a weld, I'll tell you how many there are.

ADMIN. JUDGE WARDWELL: Okay.

MR. DOLANSKY: That's basically what you're looking for --

ADMIN. JUDGE WARDWELL: That would be great.

MR. DOLANSKY: -- to understand?

ADMIN. JUDGE WARDWELL: Thank you.

MR. DOLANSKY: Okay.

ADMIN. JUDGE WARDWELL: Yes, let's do it now.

CHAIRMAN MCDADE: Okay. It might be an appropriate time to take a short ten minute break. A couple of things. First of all, Mr. Sipos, we had a discussion with Dr. Lahey about the Table 5-5 and he hadn't had an opportunity to review it. Do you have handy New York 496, which is the document that Table 5-5 is in? If not, I've got it right here and I can provide it to you for Dr. Lahey to look at.

MR. SIPOS: Your honor, I believe we have it with us. Thank you.

CHAIRMAN MCDADE: Okay. If you don't, then at the next break I can provide it to you and --

MR. SIPOS: Thank you.

CHAIRMAN MCDADE: -- you can give that to Dr. Lahey. Another thing that I actually should have mentioned earlier, in Exhibit 616, which Entergy provided, at the beginning of it, there's a table of abbreviations, which is extremely helpful. Whoever prepared it, I really want to thank them. I just wanted to mention it, as the court reporter is going through this, if you have not reviewed it, Exhibit 616, the table of abbreviations I think is going to be very helpful to you in make sure that you -- well, okay. Then that isn't necessary, apparently he found it without my assistance. Do we have anything else we need to take up before we take a short break?

ADMIN. JUDGE WARDWELL: But I think ISR was missing from that table though, but I echo how much I did use that for both other testimony, I would go back to the Entergy abbreviations or the NRC ones, whoever had them --

CHAIRMAN MCDADE: Okay. So it's ten minutes after 3:00, we'll break until 20 minutes after 3:00. Thank you.

(Whereupon, the above-entitled matter went off the record at 3:09 p.m. and resumed at 3:23 p.m.)

CHAIRMAN MCDADE: Let's get started. The hearing will come to order.

ADMIN. JUDGE WARDWELL: Okay. Moving on to baffle former bolts, the NRC testimony, 197, Answer 80, Page 59, says, an example of an augmented inspection is the baseline volumetric examination using ultrasound testing, UT, of the baffle former bolts between 25 and 35 effective full power years, which you use the acronym with the letters E-F-P-Y, there's probably some fancy way to say that, that I don't know, within a subset examination on a ten year interval, as specified in Table 4.3 of MRP 227-A.

CHAIRMAN MCDADE: Wait, whose testimony is this?

ADMIN. JUDGE WARDWELL: I'm sorry?

CHAIRMAN MCDADE: Which exhibit are you reading from?

ADMIN. JUDGE WARDWELL: This is NRC's

Exhibit 197, A 80, Page 59.

CHAIRMAN MCDADE: NRC's?

ADMIN. JUDGE WARDWELL: Yes. And I guess I'd ask Entergy, how long -- have you initiated any UT testing of the baffle former bolts at Indian Point?

MR. DOLANSKY: This is Bob Dolansky for Entergy. No.

ADMIN. JUDGE WARDWELL: And has the industry itself and do you have data on that?

MR. DOLANSKY: The industry has performed inspections on baffle former bolts. They have -- when you say, do we have data on that, what kind of data are you looking for?

ADMIN. JUDGE WARDWELL: I'm just curious on what the experience is with any of the failure rates associated with these in other Westinghouse reactors.

MR. DOLANSKY: I would characterize it as most people who have inspected have found some degraded bolts, but not enough that they were required to replace any bolts.

ADMIN. JUDGE WARDWELL: Okay, thank you.

DR. LOTT: May I suggest, there is some data on that in Entergy Exhibit 650, which summarizes the experience to date with various reactor internal exams, including the baffle bolt exams.

ADMIN. JUDGE WARDWELL: Okay, thank you. NRC's testimony on 197, Question 222, Page 124, states, what does this operating experience tell us about the probability of cracks existing in the PWR RVIs? And the Answer says, this result summarized in the presentation indicate no cracking has been found with the exception of some cracking of bolts, about 1.5 percent of Westinghouse baffle former bolts. And there I might ask Staff, what might be the cite for this 1.5 percent cracking rate for the baffle former bolts that you state in your testimony?

DR. HISER: This is Allen Hiser of the Staff. I believe it's NRC 207.

ADMIN. JUDGE WARDWELL: Okay. And does that also give the total number of baffle bolts that they looked at in regards to this, that generated the 1.5 percent figure?

DR. HISER: If you'll indulge me, I can pull it up and review it real quickly.

ADMIN. JUDGE WARDWELL: Okay. And what are you pulling up now?

DR. HISER: From Exhibit 207.

ADMIN. JUDGE WARDWELL: Okay. As you're pulling that up, let me ask this question also, is it your understanding or do you know if this industry experience with the inspection of baffle former bolts is also documented in Appendix A of MRP 227, which is NRC 114C?

DR. HISER: It is document in the MRP 227, but that was as of several years ago. The Exhibit 207 represents data through the fall of 2014.

ADMIN. JUDGE WARDWELL: Okay.

DR. HISER: So it's probably the most recent information. And this indicates that 8,887 baffle bolts have been UT inspected.

ADMIN. JUDGE WARDWELL: And that's generated the 1.5 percent in regards to the cracking rate?

DR. HISER: Yes, that's correct.

ADMIN. JUDGE WARDWELL: And do you know what was the criteria used to say that a bolt had been cracked? Is there any parameters that was given? Does it have to be separated, loose? Is there just a crack detected? Or a visual crack would apply to that?

DR. HISER: In this case, I would expect it at least was any indication of a crack from the UT exam.

ADMIN. JUDGE WARDWELL: Dr. Lahey, it seems like this experience has shown that very few cracked or failed baffle former bolts have been detected during these examinations and in most cases no cracked or failed bolts were detected at all. Do you have any other experience that dictates that they are more of an issue --

DR. LAHEY: Well, there have been --

ADMIN. JUDGE WARDWELL: -- of these than there seem to be?

DR. LAHEY: There have been failures reported overseas as well. So, it's not been massive failures, but there have definitely been failures. While we're talking about baffle bolts, I was asked to look at Table 5-5 and if you want me to -- which includes the acceptance criteria for such things as baffle bolts. Do you want me to report on that now or not?

ADMIN. JUDGE WARDWELL: That was one of your homework assignments wasn't it?

DR. LAHEY: During the break, I thought.

ADMIN. JUDGE WARDWELL: Yes. Well, we allow homework to be done during a break --

DR. LAHEY: Right.

ADMIN. JUDGE WARDWELL: -- we don't restrict that.

DR. LAHEY: Shall I report on that?

ADMIN. JUDGE WARDWELL: Sure. This is probably as good an opportunity.

DR. LAHEY: Okay. And it has to do -- I looked at the various items in the Table 5-5 and the one that I have some concern about is the baffle bolts. And, in particular, when it talks about additional examination acceptance criteria, it's not yet specified. It says, the examination acceptance criteria for the UT of the bolts shall be established as part of the examination technical justification. So it's still sort of open-ended and since it's such an important component and has safety significance. That's a little unsettling.

ADMIN. JUDGE WARDWELL: And that's the examination criteria, is that correct?

DR. LAHEY: Yes.

ADMIN. JUDGE WARDWELL: And may I turn --

DR. LAHEY: Table 5-5.

ADMIN. JUDGE WARDWELL: Yes. Turn to Entergy in regards to that examination criteria. What does that mean in regards to the statement made in Table 5-5 for the baffle former bolts? For Entergy, anyone at Entergy who wants to answer.

MR. DOLANSKY: The examination acceptance criteria for baffle former bolts would be any indication of cracking.

ADMIN. JUDGE WARDWELL: And what is the -- is there a difference between examination criteria and inspection criteria or acceptance -- I guess we have acceptance criteria and examination criteria. Are they different or are those two sayings for the same thing?

MR. AZEVEDO: This is Nelson Azevedo for Entergy. The examination criteria is the criteria for the inspectors. So anything that exceeds the examination criteria, they must report it and then we enter into our corrective action process. The acceptance criteria or the engineering acceptance criteria, if you will, it's how many bolts, for example, can we afford to lose without impacting the ability of the structure to perform its intended safety function?

ADMIN. JUDGE WARDWELL: Great. Thank you for that clarification. And did you have anything more, Dr. Lahey, that you wanted to talk about in regards to Table 5-5?

DR. LAHEY: No. With regards to that table, that was our specific concern, the number of bolts, because that is significant.

ADMIN. JUDGE WARDWELL: And then just to

close the other item, to make sure -- you don't have any hard numbers in regards to percent of these baffle bolts that failed overseas or anywhere else that would contradict the 1.5 percent that was observed by Westinghouse?

DR. LAHEY: I've never tabulated it, no.

ADMIN. JUDGE WARDWELL: Okay. Thank you. NRC's testimony, 147, Page 47, in order to maintain the intended function, only about 20 to 30 percent of the baffle former bolts need to remain intact. And I guess I'll ask Entergy -- well, it's an NRC statement, so I'll ask NRC. What's the basis for this 20 to 30 percent figure? And if it's just that it came from the Applicant, so say, or is it something that you are familiar with in regards to the generation of these particular values?

MR. POEHLER: Right. This is Jeffrey Poehler from the Staff. It did not come from the Applicant. It is based on some -- well, there's a couple of Topical Reports, WCAP reports, where they provide a methodology for performing these types of minimum pattern analyses for the baffle former bolts. Several plants have actually used that methodology and the results came out in the 20 to 30 percent range of intact bolts needed.

ADMIN. JUDGE WARDWELL: What do you mean by, they used that and came out with a 20 to 30 percent?

MR. POEHLER: They used a preapproved -- the NRC had approved the methodology for doing these analyses and it was submitted as a Topical Report --

ADMIN. JUDGE WARDWELL: And this is --

MR. POEHLER: -- to the NRC.

ADMIN. JUDGE WARDWELL: This is an engineering analysis, it has nothing to do with inspection or anything or number of bolts, it's just -- number of bolts that have cracked or anything, this has strictly an analysis of how many are needed to maintain the intended function of the baffle, is that correct?

MR. POEHLER: Correct. And how many and in what positions, what type of patterns --

ADMIN. JUDGE WARDWELL: Okay.

MR. POEHLER: -- that would be needed to maintain the design function of the baffle former assembly.

ADMIN. JUDGE WARDWELL: And that analysis resulted in the 20 to 30 percent needed?

MR. POEHLER: When that methodology was applied for specific plants.

ADMIN. JUDGE WARDWELL: Has that -- was Indian Point one of those specific plants?

MR. POEHLER: I don't -- to my knowledge, they weren't. If they've done that type of analysis, I'm not aware that their results have been submitted.

ADMIN. JUDGE WARDWELL: Okay. Well, let's turn to Entergy. Has such an analysis been done at Entergy or do you just use that 20 to 30 percent as an accepted figure based on the analysis that has been done by others?

MR. DOLANSKY: This is Bob Dolansky from Entergy. We're having a plant specific acceptable bolting pattern analysis performed for us right now that will determine plant specific. So the methodology that -- the Topical Report is a general methodology. We then took that general methodology and put in our specific accident loads, LOCA loads, all those things, and did the analysis. Well, it's ongoing. It'll be ready before we perform the inspections in the spring.

ADMIN. JUDGE WARDWELL: Do the --

MR. STROSNIDER: This Jack Strosnider for Entergy. If I just could, this question has come up a number of time about Topical Reports. I just want to make sure people understand how that process works, if I could take just a second? When the industry sees a generic issue, such as this issue, and they know that a lot of plants are going to have to deal with it, they'll go contract the vendor, like Westinghouse or somebody, to develop a methodology to do some pilot plants to do a case study. And then they'll submit that to the NRC.

The NRC reviews that and, if they approve it with whatever approvals they make, then a specific plant, a utility, can come in and reference that report when they do their plant specific analysis. And the important part of that is that the methodology has been approved by NRC. So as was just indicated, that includes all the loading and how you do the calculations and how you demonstrate functionality. So what's happening is, at the plant specific level then is just to do the plant specific evaluation. When the NRC issues their safety evaluation, they will include in it any specific action items that have to be addressed on a plant specific basis. So I hope that's of help to you because there's a number of questions that have come up on various Topical Reports and that's how that whole process works.

ADMIN. JUDGE WARDWELL: Thank you. And, Dr. Hiser, has NRC approved this methodology for determining the number of bolts needed and the patterns for the baffle former bolts?

DR. HISER: Yes, we have. And the exhibits are ENT 654 and 655.

ADMIN. JUDGE WARDWELL: Thank you. And previously, if Indian Point is consistent with what was done before, they should arrive at -- if it's consistent with what our understanding is now, it would be 20 to 30 percent of the bolts are needed, is that correct?

DR. HISER: I would expect that to be the case.

ADMIN. JUDGE WARDWELL: I think I'll go back to Entergy though to ask this question. Do you know if that 20 to 30 percent -- does not the intended function of a component, like the baffles and the former bolts that are attached to it, include some type of, in my field I'd call it a safety factor, other people will consider it an error margin or whatever else? Isn't there some sort of safety factor or error margin applied when you estimate how many of the bolts are needed?

MR. DOLANSKY: Yes. If I could just take one minute and explain? We're getting a site specific acceptable bolting pattern analysis performed. That's going to tell us some very small number of bolts that are allowed to be degraded without requiring additional analysis. So let's just say one -- there's 832 baffle former bolts. Let's just say one, so we could find one and we would be okay. But if we find more than one, then we will have to do a thing called a real time analysis and that basically is, they take the actual place that we found the degraded bolts and they put them into the analysis.

A big part of -- it's taken roughly a year to do this analysis. They develop a computer program, they'll then take the specific bolts that we actually found degraded, put those into the computer program, run it with our specific loads and site specific requirements, and come out with, is that actual what we found acceptable or not?

ADMIN. JUDGE WARDWELL: And by acceptable, does that have an error margin or a safety margin around it?

MR. DOLANSKY: Yes, it does.

ADMIN. JUDGE WARDWELL: And my question is, do you know if the 20 to 30 percent figure of the acceptable bolts that have been done in the past, that number generated in the past, does that include that safety factor or could it possibly be that, that is what needed with a safety factor of only one?

MR. DOLANSKY: I can't answer that.

ADMIN. JUDGE WARDWELL: Okay. Staff, would you be able to answer that question?

DR. HISER: This is Allen Hiser of the Staff. In the generic methodology, Topical Report, there would be safety factors included in that.

ADMIN. JUDGE WARDWELL: So you believe that --

DR. HISER: It includes safety factors.

ADMIN. JUDGE WARDWELL: -- the 20 to 30 percent includes some sort of safety factor such that it could be even less and the thing would still hold together?

DR. HISER: If you use a safety factor of one, yes, I expect you could go to fewer bolts than the 20 to 30 percent.

ADMIN. JUDGE WARDWELL: Right. Or conversely, if the safety factor was two that they actually used, you could really go down to 10 to 15 percent?

DR. HISER: Well, safety factor is not on the number of bolts --

ADMIN. JUDGE WARDWELL: Just in general terms, just to put a handle on --

DR. HISER: Yes, that's correct. It would be a lower number.

ADMIN. JUDGE WARDWELL: The point I'm trying to get is I want to make sure that, that 20 to 30 percent doesn't pertain to a safety factor of one. And, likewise, when you do your analysis, I'm asking Entergy, will a number you have be the number that is absolutely needed in order to maintain safety with no extra margin or will it include some margin associated with it?

MR. DOLANSKY: This is Bob Dolansky with Entergy. First, when we're doing all this, we're really concerned about reactor safety, maintaining core coolability, maintaining the ability to insert the control rods. That's factored into the analysis and there are safety margins on that safety analysis.

MR. STROSNIDER: This is Jack Strosnider --

ADMIN. JUDGE WARDWELL: So when you -- no, I'd like to stay with my thoughts. When you come up with an acceptable pattern, that pattern will have some margin built into it, such that in actuality if you knew truth, you could end up with lesser number of bolts, but because we don't know truth, you're going to have some margin in there that allows for the uncertainty associated with not knowing truth in that pattern. Is that a fair assessment?

MR. DOLANSKY: Yes. And the reason why I spoke before about the real time analysis is we don't want to know, when we don't know what we actually have, come up with some bolting pattern analysis. We want to wait until we find out what we actually have out there and then run it through the program and make sure that it's robust and is perfectly acceptable. So that's why we actually run the real time analysis.

MR. AZEVEDO: Your honor, this is Nelson Azevedo. If I may add, we use the NRC approved methodologies, so the same safety margins are going to be used.

MR. STROSNIDER: And this is Jack Strosnider. Just to expand on that, typically what you're going to see in these Topical Reports when they do this type of analysis, is they're going to work to maintain the safety margins that were in the original licensing basis. So in this case, if you're talking about the structural criteria, it's going to be the ASME code factors of safety during normal and accident conditions. And when you look at the other aspect of this, we get into some of the core cooling and accident evaluations and they would need to maintain the margins that are in that licensing basis in terms of margin to core damage and that sort of thing. So they work to maintain the current licensing basis and the margins that were in those.

ADMIN. JUDGE WARDWELL: And so it's your professional opinion that the 20 to 30 percent that has historically been generated by Westinghouse at other plants includes those safety factors associated with the ASME code?

MR. STROSNIDER: Yes, it would have those margins in it.

ADMIN. JUDGE WARDWELL: Thank you. Your testimony, Entergy, 616, Answer 152, Page 99, says, to prepare for these inspections, as explained in the Supplemented SER Number 2, that the UT examination acceptance criteria for the baffle former bolts will be developed as part of the technical justification for the inspections. Page 100, Answer 154, that the examination acceptance criteria for the individual baffle former bolts will be no defect that could be detectable via UT inspections. And in parentheses, you say that detectable versus UT inspections are a defect exceeding 30 percent of the bolt cross-sectional area.

The TJ will merely demonstrate that the UT inspections at Indian Point will be capable of detecting such cracking. And I guess my first question to you is, from where did this 30 percent of the cross-sectional area come from in regards to -- I gather that is what is needed before you're able to detect it with the UT inspections.

DR. LOTT: I believe the last thing you said is basically true, that the UT inspections, a standard UT inspection, very reliably detect 30 percent. I believe that the requirement is stated in MRP 228, if I'm not mistaken, that, that minimum 30 percent is there based on the judgement of the inspectors.

ADMIN. JUDGE WARDWELL: But that's a result of what you can get out of it. That's what you need before you're -- that's the sensitivity of your UT device --

DR. LOTT: Right.

ADMIN. JUDGE WARDWELL: -- if you will. Is that correct?

DR. LOTT: Yes.

ADMIN. JUDGE WARDWELL: And, so, that means that anything under that won't be detected, is that correct?

DR. LOTT: Well, maybe not necessarily won't, but might not be.

ADMIN. JUDGE WARDWELL: The odds are it won't. I mean, there will be some that needs more than 30 percent that won't be detected also, using the same phraseology?

DR. LOTT: Yes. It's a very high probability of detection, I believe, at 30 percent.

ADMIN. JUDGE WARDWELL: And is this acceptance criteria -- has the 30 percent figure shown up anywhere in the Inspection Plan or anything else? Or is that just part of the testimony that you've created here for this hearing?

MR. DOLANSKY: This is Bob Dolansky with Entergy. I believe that 30 percent will be in the technical justification --

ADMIN. JUDGE WARDWELL: Okay.

MR. DOLANSKY: -- that will be used by -- that has to be documented and reviewed by our NDE Level III at the site before they perform the inspection.

ADMIN. JUDGE WARDWELL: This is all stuff you're going to get prepared and done prior to starting your inspection, I think you talked about earlier today.

MR. DOLANSKY: Correct.

ADMIN. JUDGE WARDWELL: Dr. Lahey, do you have any comments on what you heard in regards to the baffle former bolts --

DR. LAHEY: Yes.

ADMIN. JUDGE WARDWELL: -- inspection and --

DR. LAHEY: Yes, your honor. This was a very interesting discussion for me. I have not seen that report or the results, I'd be very interested to see it. In particular, I heard nothing about the type of loads that were applied to assure the integrity of these patterns with reduced number of bolts and would be very interested in hearing about what type of impulsive loads were used, if they were used. And what type of codes might have been used to generate these.

ADMIN. JUDGE WARDWELL: Okay, thank you.

DR. LOTT: Some of that information, I will suggest, would be available in ENT 0655, which is basically the same document that Dr. Hiser just referred to in terms of the methodology for determining acceptable bolt patterns. It discusses the types of loads that are included, particular the multi flex code and how they're transferred to the --

ADMIN. JUDGE WARDWELL: Have those loads been documented in the Topical Reports that the NRC has approved?

DR. LOTT: Yes, in great detail.

CHAIRMAN MCDADE: Okay. Dr. Lott, what exhibit did you just reference?

DR. LOTT: Entergy 655.

CHAIRMAN MCDADE: Thank you.

ADMIN. JUDGE WARDWELL: When did you start this endeavor in regards to planning for the inspections of the baffles and going through this site specific process of determining your pattern for the baffle bolts?

MR. DOLANSKY: Just the baffle former bolts --

ADMIN. JUDGE WARDWELL: Yes.

MR. DOLANSKY: -- you're asking about?

ADMIN. JUDGE WARDWELL: Yes.

MR. DOLANSKY: This is Bob Dolansky for Entergy. The contract for the inspection of the baffle former bolts was issued, I'm going to say in spring of 2015. The contract to develop the acceptable bolting pattern analysis, I think, was January. It takes a little over a year to develop that acceptable bolting pattern analysis. So I'm pretty sure it was January and the final report will be ready for us in February before our inspections in

March.

ADMIN. JUDGE WARDWELL: For Dr. Hiser of the NRC, when was the Topical Report approved by you that --

DR. HISER: I believe it was --

ADMIN. JUDGE WARDWELL: -- highlighted these inspection procedures or the whole methodology, is what I guess the best word is?

DR. HISER: This is Allen Hiser of the Staff. Is the question the methodology used to demonstrate the minimum bolt pattern?

ADMIN. JUDGE WARDWELL: Yes.

DR. HISER: That was I believe 1999.

ADMIN. JUDGE WARDWELL: So my question to Entergy will be, why haven't you started earlier in this process then in order to -- if the methodology has been outlined in regards to coming up with this pattern, why wasn't it done earlier considering your License Renewal Application was submitted in 2007?

MR. DOLANSKY: We didn't feel that we needed the acceptable bolting pattern analysis -- Bob Dolansky with Entergy -- until we were going to perform the inspection. Basically, until you perform the inspection and determine if you have any degraded bolts, the acceptable bolting pattern analysis doesn't do anything for you. You don't -- there's no value to it until you --

ADMIN. JUDGE WARDWELL: I guess that gets back to the question I asked earlier, why didn't you start to do the inspection on the bolts earlier? Which I think we've already talked about, so, okay.

MR. AZEVEDO: Your honor --

ADMIN. JUDGE WARDWELL: Mr. Azevedo, it looked like you were -- you weren't, you were just stretching or something?

MR. AZEVEDO: No, I was going to say what Mr. Dolansky just said.

ADMIN. JUDGE WARDWELL: Okay. That's a cop out. No, I believe you.

MR. KUYLER: Your honor, if I may? I believe a moment ago Dr. Lott may have misspoke. The Exhibit that I think he might have been referring to is Entergy 654, that provides the accident loads.

DR. LOTT: Yes, 655 is the non-proprietary version.

ADMIN. JUDGE WARDWELL: Okay.

DR. LOTT: Either way it works.

CHAIRMAN MCDADE: Dr. Lott had mentioned 655 and counsel is suggesting it was actually Entergy 654, is that --

MR. KUYLER: Yes, your honor.

CHAIRMAN MCDADE: Okay, thank you.

ADMIN. JUDGE WARDWELL: Moving on to the clevis bolt --

CHAIRMAN MCDADE: Actually, could I clear something up before --

ADMIN. JUDGE WARDWELL: Sure.

CHAIRMAN MCDADE: -- you move on to --

ADMIN. JUDGE WARDWELL: Yes.

CHAIRMAN MCDADE: -- a new topic, just really quickly. Mr. Dolansky, before the break, you were asked with regard -- questioned on Table 1, Page 87, regarding welds. And that particular document, it's Entergy Exhibit 616, it identifies four different kinds of welds. The upper core baffle flange, the barrel cylinder girth welds, the lower core barrel to lower support casting welds, and the core barrel outlet nozzle welds. But it doesn't have any indication as to for each of these categories, how many welds we're talking about. Are we talking about a single weld each time or are we talking about scores or hundreds or thousands?

MR. DOLANSKY: This is Bob Dolansky with Entergy. Looking at Table 87, the upper core barrel flange weld is a single weld. The upper and lower core barrel cylinder girth welds are two individual welds, an upper and a lower core barrel cylinder girth weld. The lower core barrel to support casting weld is a single weld. The only ones that are multiple welds are the lower flange welds on the control rod guide tubes, which are at the top there, the second box. That -- give me one second and if we could bring up on the screen, let's see what, number seven -- NRC 114A through C and we're actually going to be looking at 114B, Bravo, I believe. And we want Page 4-60. It'll help clarify.

CHAIRMAN MCDADE: 4-60?

MR. DOLANSKY: 4-60, yes, your honor. Yes. Stop right there, that's fine. So, this is a depiction of the control rod guide tube assembly. If you look on the bottom, so what we call the lower flange welds in the table on Page 87, if you look on the bottom of that, it shows two arrows pointing to lower flange welds, the bottom one, let's say, there's discrete, very small ribs and each of those ribs has welds. So these components have multiple welds on each component. But that's the only one that really has multiple welds on each component.

CHAIRMAN MCDADE: Okay, thank you. And I wondered, since we're at a break right here, pose a question to -- I'm sorry, Mr. Cox, did you have a --

MR. COX: Yes, I was just going to clarify that the next page of that exhibit shows the other welds that you had asked about. Mr. Dolansky indicated there were one or two welds in those cases. I just wanted to point out, those welds are around the entire circumference of the core barrel, so it's -- I mean, some of these guys could probably tell you how many inches of weld that is, but it's several hundred probably.

MR. DOLANSKY: Right. They're large -- I'm sorry, I didn't mean to imply that they were small welds. They're very large, long welds around a big robust component.

CHAIRMAN MCDADE: Okay.

ADMIN. JUDGE WARDWELL: Does one guy do them? Or do they have multiple guys?

MR. DOLANSKY: Well, if they're EVT1, they have to -- because there's criteria on scanning speed and so forth, it's actually done by special tooling. But there are people watching screens for each of them.

ADMIN. JUDGE WARDWELL: But the guy who first built it, was it one guy doing one weld and he can take ownership of that weld and say, there's my weld?

MR. DOLANSKY: Probably. Back then, probably true.

ADMIN. JUDGE WARDWELL: Okay.

CHAIRMAN MCDADE: Okay. And the other thing I wanted to do is just pose a question, not be answered right now, but to move on with Judge Wardwell's question, but to raise it for Dr. Lott and Dr. Lahey who had testified earlier, addressing New York Exhibit 488, which is a NUREG/CR-7184, it's from December of 2014. And, basically, my question is this, we've had testimony that for these low ferrite stainless steel material, they won't show a meaningful combined effect from thermal aging and irradiation.

And in that NUREG it seems to suggest, well, it states, the radiation and the reduction of fracture toughness was more significant in the unaged than in the thermally aged specimens. And it goes on to have a further discussion of that. And I'd like to, perhaps at the beginning of tomorrow, come back and discuss whether or not this supports the hypothesis of the -- Dr. Lahey's hypothesis or whether or not there's an explanation explaining why this doesn't support the synergistic effect of the thermal aging and the neutron embrittlement. But I don't want to take that up right now, I just wanted to address that particular document to you so you would have a chance to look at it this evening and perhaps you could take it up briefly tomorrow. Dr. Lahey?

DR. LAHEY: Yes, this is Richard Lahey, New York. You had asked me to look at this report, NUREG-7184, and to look at the word synergistic and I did look at this report and it's very interesting, the later version of it was redlined out. It was synergistic, but it was redlined out and combined effect was put it its place. So at one point it said synergistic and I guess the final issue would have said combined effect.

CHAIRMAN MCDADE: Okay. But you would view those two words as synonyms, but I don't want to take it up right now and I want to let --

DR. LAHEY: Right.

CHAIRMAN MCDADE: -- Judge Wardwell move on. But what I'm looking for is to whether or not, not just a particular word, but whether or not this NUREG and the data presented there, what impact, if any, does it have on the hypothesis of the synergistic effects? And, again, I don't want to take it up right now, we want to move on. Take a look at it, have Dr. Lott and Dr. Hiser take a look at it as well overnight and we can discuss it very briefly in the morning.

DR. HISER: Judge McDade, quick question on that. Allen Hiser of the Staff. What page number of 7185 were you quoting from?

CHAIRMAN MCDADE: Well, I was quoting from the abstract --

DR. HISER: Okay.

CHAIRMAN MCDADE: -- which is I think especially like a little I or a little double I --

DR. HISER: Okay.

CHAIRMAN MCDADE: -- I was just quoting from the abstract, but it then goes on.

MR. SIPOS: And your honor, for New York, I think --

CHAIRMAN MCDADE: I didn't say that, I said I was quoting from the abstract.

MR. SIPOS: Your honor, John Sipos for the record. I think you were referring to 7184. I just want to make sure that's clear for the record, rather than 7185, which I think Dr. Hiser just mentioned.

CHAIRMAN MCDADE: Wait, now --

ADMIN. JUDGE WARDWELL: You initially said 7184.

MR. SIPOS: And that is Exhibit New York 488 --

CHAIRMAN MCDADE: Hold on one second.

MR. SIPOS: -- I believe.

ADMIN. JUDGE WARDWELL: While you're doing that, I want to correct something else that you put words in Dr. Lahey's mouth and he said right and I want to verify he meant to say right to what you said. Judge McDade said that you believe that synergism and combined are the same. That isn't what I heard you say earlier today. I heard you say today synergism was more than the combined.

DR. LAHEY: My understanding of what he said and why I said right, was he indicated that they were the same as far as I was concerned. What I thought was synergism and what they called combined was the same.

ADMIN. JUDGE WARDWELL: But it's not the sum of the two components in your opinion is synergism. Synergism is more than that, is it not?

DR. LAHEY: I think it can be, yes. But it may also be the same.

ADMIN. JUDGE WARDWELL: Be what?

DR. LAHEY: Well, no, I think it can be either one.

ADMIN. JUDGE WARDWELL: You think synergism can be just the sum of the two?

DR. LAHEY: Can have two things -- synergism can mean two things going on at the same time or --

ADMIN. JUDGE WARDWELL: And it's equal to only the sum of the two individual contributions?

DR. LAHEY: It can be, I believe, and it can be --

ADMIN. JUDGE WARDWELL: Greater?

DR. LAHEY: -- more.

ADMIN. JUDGE WARDWELL: Okay. So you believe it's both the sum and/or?

DR. LAHEY: I believe so. I don't know what the author or --

ADMIN. JUDGE WARDWELL: No, what I'm asking you is what's your --

DR. LAHEY: Yes.

ADMIN. JUDGE WARDWELL: -- when you say the word synergism --

DR. LAHEY: That's my view.

ADMIN. JUDGE WARDWELL: -- do you mean it has to be more than the sum or can it be the sum or more?

DR. LAHEY: Yes.

ADMIN. JUDGE WARDWELL: No, those are choices.

(Laughter.)

DR. LAHEY: It can be either or.

ADMIN. JUDGE WARDWELL: Thank you. Moving on to see if we can confuse some other stuff or I can confuse some other stuff. I want to go back to the --

CHAIRMAN MCDADE: Well, let me clarify. I was referring to New York Exhibit 488, NUREG/CR-7184.

ADMIN. JUDGE WARDWELL: Right.

CHAIRMAN MCDADE: If I said something different, I just misspoke and I apologize.

ADMIN. JUDGE WARDWELL: Well, you said it right, Dr. Hiser said 85 and we wanted to make sure --

DR. HISER: I apologize.

MR. COX: Could I add a clarification on the report numbers?

(Laughter.)

CHAIRMAN MCDADE: Pardon?

MR. COX: I just -- this is Alan Cox with Entergy. I just wanted to point out there's two versions of the NUREG/CR-7184.

CHAIRMAN MCDADE: I was reading from the December 2014. Is that the latest?

MR. COX: There is I believe a later version. It is New York State, you said 574?

CHAIRMAN MCDADE: No, I said 488.

MR. COX: Okay. There's a 574, it's also, it's a New York State 000574, it's NUREG/CR-7184. And I think Dr. Lahey alluded to two versions of this report, so those may be the two versions. One of those may have the word synergy and the other may not.

CHAIRMAN MCDADE: Okay. Thank you, Mr. Cox.

ADMIN. JUDGE WARDWELL: Now we get to move to what we've already discussed, but I want to clarify again in regards to the clevis bolts that New York State testimony, 482, Page 56 to 57, and 56 it's Lines 20 to 23 and 57 it moves on to Lines 1 through 9. It says that failures of the clevis insert bolts apparently caused by PWSCC were detected at a Westinghouse designed reactor in 2010. Out of the 48 bolts in this reactor, 29 were partially or completely fractured, but only seven of those damaged bolts were visually detected as having failed.

Despite this high rate of failure, about 60 percent of the total bolts were damaged, and a low rate of visual detection, only 24 percent of the damaged bolts were detected, the Applicant proposes to managing the aging degradation of clevis insert bolts with visual VT3 inspections rather than preemptive replacement. And I guess I'll reaffirm again that, Entergy, do you agree with these numbers in regards to what was previously reported from Westinghouse?

DR. LOTT: I agree with the scale of the numbers, I haven't checked the actual --

ADMIN. JUDGE WARDWELL: There's no reason not to believe those numbers?

DR. LOTT: No reason not to believe those numbers.

ADMIN. JUDGE WARDWELL: And how does that low percentage, again, support the use of VT3 as an inspection, when in fact almost three-quarters of damaged ones will go undetected?

DR. LOTT: It was never our intention to inspect the bolts, it was our intention to verify the stability and location of the clevis itself. As we talked before, the clevis can function perfectly well without the bolts if it's in place. So, again, our inspection is basically -- and we made recommendations in the Tech Bulletin that we issued on this subject to basically inspect the clevis for its seating into the lug on the vessel wall and to make sure that it had not moved, that there was not undue wear on the surfaces that would indicate that it had moved or was free to move. But we did not anticipate that a visual inspection would necessarily detect cracking of a bolt.

ADMIN. JUDGE WARDWELL: But did you not also go on and testify that you're considering or evaluating whether or not to replace any of the bolts that may have been damaged?

DR. LOTT: Well, first of all, at that particular plant a number of the bolts were replaced. They were replaced, again, because of our concern about the commercial aspects, the potential that those clevises might become dislodged, which would make it difficult for them to, as we said, reinsert the barrel, restart the plant. So we certainly were advising them, and I think it explains this in the Technical Bulletin, that it might be wise to replace the bolts, but not because of a safety concern.

ADMIN. JUDGE WARDWELL: Dr. Lahey, in your testimony, 482, Page 58, Lines 5 through 9, you state that rather than taking proactive steps to replace the degraded clevis bolts prior to failure, the Applicant proposes to wait for failures to occur before taking steps to address the problem, an approach that is totally unacceptable in my opinion. Hearing what you heard now, do you still believe that your demand for wholesale replacement of these bolts are a reasonable expenditure of effort?

DR. LAHEY: Your honor, this is Richard Lahey from New York. When we had this discussion earlier today, it was my understanding that to replace a few of the bolts is straightforward, if you have a lot of the bolts, it requires a massive effort in terms of realignment. And, so, I believe based on that and the safety significance of it, it would really depend on the degree of failure of the bolts. What would make sense in any event, you don't want loose parts rattling around.

ADMIN. JUDGE WARDWELL: In regards to that last statement, Entergy, do these failures of the bolts end up with loose parts? Or are they kind of contained with the --

DR. LOTT: It's contained within the system. It's difficult for the bolt head -- first of all, there's locking bars. If there was wear through or failure of the locking bars, still the bolt head could not escape and the threaded part of the bolt can't get past the head.

ADMIN. JUDGE WARDWELL: Thank you. Just reading through here to -- Staff, is there any place within your SER or SE that you evaluated the clevis insert bolts and what is an appropriate action level, if any, for these documented --

MR. POEHLER: This is Jeffrey Poehler for the Staff. Yes, we do address that in our SER in the discussion of operating experience, which is located right after the ten element discussion. And we -- yes, we reviewed that quite extensively and we agreed with continuing to do VT3 examination as an acceptable means of managing the potential for bolt failures.

ADMIN. JUDGE WARDWELL: Okay, thank you. New York State's testimony, 482, Page 57, Line 20 through 23, and then moving over to 58, Line 1, says, the Applicant's analysis of the effects of clevis bolt failures assumes that all of the components will be functioning according to their design specification and does not consider the fact that other components may also be undergoing degradation from various interacting mechanisms.

Entergy's testimony, Exhibit 616, Answer 166, Page 107, says, Entergy is not required to assume without evidence that other components that are within the scope of the reactor vessel internals AMP or any other AMP are also degraded when it evaluates the functionality of the clevis insert bolts. And I guess considering we have the two competing statements, I'll go to Staff and ask them. What do you look for and what do you consider in regards to the potential for other components outside of the AMP, the RVI AMP, that might not be functioning along with these clevis bolts and how are they assessed? Or is there no requirement to do that and no need for safety to do that?

MR. POEHLER: This is Jeffrey Poehler from the Staff. We would not require them to assume that, for instance, the part that interfaces with the clevis insert is called the radial key and that's attached to the bottom of the core barrel, and we would not require them to assume that, that had failed or expect them to do that when doing their functionality evaluation of the clevis or the lower radial support system. And in addition to that, they're both redundant components, both the clevis inserts and the radial keys, I believe there's six. So the likelihood of a significant number of those failing at once is low. And the radial keys were not even a -- they were a no additional measure component, which means there were no inspection requirements in MRP 227-A.

ADMIN. JUDGE WARDWELL: Are there any components that do have inspection requirements that are associated with one another where you have looked at the potential failure of both in the reactor vessel internals?

DR. HISER: In general --

ADMIN. JUDGE WARDWELL: Not necessarily with the clevis system.

DR. HISER: This is Allen Hiser with the Staff. We do not require postulation of failures of other components when assessing a finding of a degraded component. That's not a part of the regulatory process.

ADMIN. JUDGE WARDWELL: And do you know any technical basis for supporting that position?

DR. HISER: I think technical basis is just Commission position that it's not required as a part of the license renewal review and that annotation of license renewal.

ADMIN. JUDGE WARDWELL: Thank you. Let's move on to the lower support columns. NRC's testimony, 197, Answer 27 on Pages 35 to 36, the fuel assemblies are supported inside the lower internals assembly on top of the lower core plate, that's LCP, and the function of the LCP are to position and support the core and provide a metered control of reactor coolant flow into each fuel assembly. The support columns transmit vertical fuel assembly loads from the LCP to the much thicker lower support casing. The function of the lower support casing is to provide support for the core. My question to Staff, considering it is your testimony, are these support columns you're referring to the lower support columns? Are they one and the same?

MR. POEHLER: This is Jeffrey Poehler of the Staff. That's correct.

ADMIN. JUDGE WARDWELL: And for these columns, what is the primary mechanism for developing flaws from aging? Is it driven by normal operating conditions?

MR. POEHLER: It would be normal operating conditions.

ADMIN. JUDGE WARDWELL: And would other operating conditions, such like seismic or LOCA events, likely be the primary contributor to service induced flaws and, if not, why not?

MR. POEHLER: No, those other events, like seismic events would not be a significant contributor because those events occur very infrequently. So the --

ADMIN. JUDGE WARDWELL: Your testimony on 197, Answer 171, Page 92, says that the Action Level 7 requires an applicant or a licensee to perform a plant specific analysis of the cast Austenitic stainless steel RVI components to demonstrate that components will remain capable of performing their intended functions through the end of the plant life. Your Answer on 163, Page 93, goes on to state that Entergy identified the only components requiring a response to Action Level 7 for IP2 and 2 are the lower support columns.

Only the upper portion of the lower support columns, known as the column cap, is made from CASS. The lower support columns are an expansion component of MRP 227-A and the associated linked primary component is a control rod guide tube lower flange welds. So, let me make sure I understand this correctly. Is this not saying to me that the only CASS materials in the population of reactor vessel internals are the upper portion of the lower support columns, i.e., what's called the column cap?

MR. POEHLER: It's not the only CASS component in the internals, or the only type of CASS component, I should say, at Indian Point. It is the one that is of most concern to the Staff because of the high irradiation levels that parts of it can experience.

DR. HISER: Dr. Wardwell, this is Allen Hiser of the Staff. Action Item 7 specifically identifies the lower support column bodies as within the scope of that Action Item. And it is due to the potential for thermal embrittlement and irradiation embrittlement of those. And those are the only ones that were identified that would be within the scope of irradiation embrittlement. So that's why it's limited only to the lower core support columns.

ADMIN. JUDGE WARDWELL: But does the Action Level 7 state that only that component is required or is it more general that it requires an applicant or a licensee to perform a plant specific analysis of CASS components?

DR. HISER: Those are specified within the Action Item, the lower core support columns for Westinghouse plants. The intent of that Action Item was to identify CASS that's within the high fluence field that could lead to irradiation embrittlement. Within the context of what was evaluated in MRP 227, that was the only generic CASS component. Referring back to Action Item 2, where plants are to evaluate differences between the plant specific configuration and MRP 227, if CASS was atypically used in a place at Indian Point different from MRP 227, then that would have been identified in Action Item 2 and also should show up in Action Item 7.

ADMIN. JUDGE WARDWELL: And so what's the criteria for being under 2 again? For an applicant to highlight whether a CASS composed material, an internal composed of CASS, that's the way I want to say it, I guess, would or would not require the additional attention brought on by Action Level 7?

DR. HISER: It says that applicants should review the information in Tables 4-12 and 4-2 in MRP 189 Rev 1, Tables 4-4 and 4-5 in MRP 191, and identify whether these tables contain all the RVI components that are within the scope of license renewal for their facilities. I think as a part of that, because material is critical, that, that would be identified as well as a difference between the plant specific configuration and the MRP 227 assumptions.

ADMIN. JUDGE WARDWELL: I guess I'm still a little confused. Are the upper portion of the lower support columns the only RVIs under license renewal that are made of CASS? And, if not, how did the others get screened out and what was the criteria for screening them out, for not being highlighted under Action Level 7?

DR. HISER: This is Allen Hiser of the Staff. For Indian Point, our understanding is those are the only CASS components that are subject to irradiation embrittlement above the threshold level used in the development of MRP 227. So they would be the only CASS components that would be subject to the potential susceptibility of thermal embrittlement and irradiation embrittlement.

ADMIN. JUDGE WARDWELL: And what is that screening criteria that would separate out those that were and were not susceptible to thermal embrittlement and irradiation embrittlement?

MR. POEHLER: This is Jeffrey Poehler of the Staff. The MRP screening criteria was one dpa.

ADMIN. JUDGE WARDWELL: Okay. And Westinghouse, is everything that was stated consistent with your approach and is, as a net result, the upper portion of the lower column supports, that is made of CASS material, the only RVI that does not meet the screening criteria of one dpa of fluence and, therefore, is part of and falls under Action Level 7?

DR. LOTT: For the most part, I believe that's true. I'm recalling --

ADMIN. JUDGE WARDWELL: Well, then the other parts I'm interested in.

DR. LOTT: I know and I'm trying to get there. I'm not -- really I'm trying to get there. And there's one component that I need to check on and that might be the upper support column, which I believe in my previous testimony indicated that it was potentially CASS. I don't know if it is or not.

ADMIN. JUDGE WARDWELL: Let's put that on your homework list for tomorrow morning then to determine whether there are any RVIs --

DR. LOTT: Yes. I would point out that there are some other CASS components in the system and that part of the evaluation procedure that led to this screening process in MRP 191 was an evaluation of the impact of those components or of those degradation mechanisms. Of course, they would have been identified for thermal embrittlement anyway.

ADMIN. JUDGE WARDWELL: Do you agree that the one dpa is the threshold screening criteria or is it not?

DR. LOTT: Yes.

ADMIN. JUDGE WARDWELL: It is?

DR. LOTT: Yes, it is. But --

ADMIN. JUDGE WARDWELL: So we're back to --

DR. LOTT: -- what I would say, what the classification process that identified components to be in MRP 227 or not also had this evaluation of the structural significance of the component. So it's possible -- and as I understand the documentation in Action Item 7, it basically said that the NRC had reviewed those decisions in MRP 191 and said that if things were appropriately dealt with there, there was not a need for functionality analysis under Action Item 7. So it grandfathered in things that were identified within MRP 191 as potentially CASS and evaluated and screened out, are not required to go through this process. And that I'm believe is true of the upper support columns.

ADMIN. JUDGE WARDWELL: Well, that's -- I'm interested in whether or not all of those that haven't either been grandfathered out by 191 or haven't been screened out due to the criteria of the one dpa are now present in or not highlighted by 7 now, the reasons why they aren't? Well, I do not want to read that question in the transcript at all.

(Laughter.)

DR. LOTT: I think I can -- I don't think that, that description you -- there are any components that meet that description that you just said.

ADMIN. JUDGE WARDWELL: I just want to have you verify that you have captured all of those that need to be captured and that the lower support column is the only one that needs to be captured by 7.

DR. LOTT: Okay. Action Item 7, that I believe is true.

MR. STROSNIDER: This is Jack Strosnider

for Entergy. I just want to say, one other parameter that may come into consideration here is the resistance of CASS material to any type of formation of any type of corrosion cracks. And we haven't talked about that, but this material, although it can permeate and we talked about that earlier, depending on the ferrite content, et cetera, but the good thing about this material is that it's very hard to crack. And I'm not sure that there's any operating experience in which we actually have seen cracking in these CASS materials. To my knowledge, we haven't, unless there's something fairly recently. So, that may also factor into how this is treated and we'll have to look at that.

ADMIN. JUDGE WARDWELL: Thank you. Back to Entergy again. The upper portion of the lower support column is made of CASS, what's the lower portion made of?

DR. LOTT: The wrought stainless steel.

ADMIN. JUDGE WARDWELL: The raw?

DR. LOTT: Wrought stainless steel. I'm sorry, I'll get closer to the microphone again.

ADMIN. JUDGE WARDWELL: I thought I heard you say raw. Okay, wrought stainless steel. Thank you. Dr. Lahey, do you have any knowledge of any potential RVIs that you believe should fall under A/LAI 7 in regards to the Aging Management Program?

DR. LAHEY: Your honor, I have reviewed EPRI 191 and I'll have to look back on it, it's in the other room. But my recollection is there were core plates, upper and lower, that were castings, but I'll have to verify that.

ADMIN. JUDGE WARDWELL: Okay. So, Entergy, you might note that also that those plates are of interest and comment on that if you would.

DR. LOTT: Okay.

ADMIN. JUDGE WARDWELL: Thank you, Dr. Lahey. Answer of NRC's 197, 163 on Page 94, states that the lower support columns for IP2 and 3 are made from type CF-8 stainless steel, which is a low molybdenum contact grade of cast stainless steel. Low molybdenum cast grades are less susceptible to thermal embrittlement than the high moly cast grades, such as types CF-8M. Entergy determined the lower support columns at IP2 and 3 were not susceptible to thermal embrittlement. And, again, do these statement refer only to the cap as being not susceptible to thermal embrittlement because the lower wrought iron has already been screened out, it's not part of A/LAI 7?

DR. LOTT: Yes. The lower cast of

Austenitic stainless steel, wrought stainless steel, is not subject to thermal embrittlement. It would only be the upper cast portion.

ADMIN. JUDGE WARDWELL: And so whenever we talk about -- whenever you reference lower support under this section of the CASS portion, you mean just the upper part? Let me rephrase that question. When you are referring to the cap of the lower support system, that's what you're referring to is that upper part that's made out of CASS, is that correct?

DR. LOTT: Yes.

ADMIN. JUDGE WARDWELL: And is that the top half of the column? Is it at the top little piece of the column? Or is it a top quarter of the column?

DR. LOTT: It's more like quarter to a third of the column. I don't think it --

ADMIN. JUDGE WARDWELL: But it's not that little, I saw a diagram of the column and there's a little rectangular piece on the top, that's not the cap, it's more than that?

DR. LOTT: No, it's more than that.

ADMIN. JUDGE WARDWELL: It's a --

DR. LOTT: Yes.

ADMIN. JUDGE WARDWELL: -- major portion --

DR. LOTT: It's --

ADMIN. JUDGE WARDWELL: -- a portion of that --

DR. LOTT: Right.

ADMIN. JUDGE WARDWELL: -- entire column?

MR. DOLANSKY: It has some length to it.

ADMIN. JUDGE WARDWELL: Yes, okay. Just so we have a feeling of it. And by screening it out as being unsusceptible to thermal embrittlement, that by necessity means that the caps cannot be susceptible to the combined effects of irradiation embrittlement and thermal embrittlement, it's just the irradiation embrittlement, is that correct? Entergy?

MR. DOLANSKY: Yes, that's correct. This is Bob Dolansky. Yes, that's correct.

ADMIN. JUDGE WARDWELL: And, Staff, as far as you're concerned, you agree with that conclusion?

MR. POEHLER: This is Jeffrey Poehler with the Staff. Yes, we agree with that conclusion.

ADMIN. JUDGE WARDWELL: Dr. Lahey, do you agree that just the upper portion of this column cap is --

DR. LAHEY: Yes, sir, that's my understanding as well.

ADMIN. JUDGE WARDWELL: Thank you. The A 163, Page 95 goes on to say that screening criteria would allow the low molybdenum cast with delta ferrite content of 20 percent or less to be screened out, that is not susceptible to thermal embrittlement. Since portions of the lower support column will have neutron fluences at the end of life greater than one times ten to the 17 neutrons per square centimeter, the Staff did not accept Entergy's screening and the Staff updated the criteria of low molybdenum statically cast CASS with a ferric content of less than 15 percent can be screened out for thermal embrittlement and any synergistic effects of TE and IE and as low molybdenum statically cast CASS is only susceptible to irradiation embrittlement at fluences greater than one times ten to the 21 neutrons per square centimeter or 1.5 displacements per atom.

Staff, could you try to condense that whole confusing thing I just read, or a bit confusing to me, in regards to how these lower columns were specifically, and specifically the lower portion of this that's called the cap -- the upper portion of the lower column, were screened out and assessed by you?

DR. HISER: Okay. This is Allen Hiser of the Staff. The top of that paragraph talks about the 20 percent or less screening for thermal embrittlement.

ADMIN. JUDGE WARDWELL: And that's for delta ferrite content, correct?

DR. HISER: Right. That's correct. And that was proposed for really CASS that was not subject to neutron embrittlement. So it was subject to low neutron fluence. With the possibility of higher neutron fluence levels combined with the potential for thermal embrittlement, we went back and looked at the screening, the 20 percent, and concluded that we should reduce that level by five percent to 15 percent. And we thought that, that was a reasonable screen to preclude the potential synergism of thermal embrittlement and irradiation embrittlement.

When Indian Point then came in with their measured ferrite values, or their ferrite values for their lower core support column caps, and it was below 15 percent, we thought it was reasonable to screen that out then from potential synergistic effects of thermal embrittlement and irradiation embrittlement. So what that leaves then for those components is only a concern with the irradiation embrittlement on the fracture toughness for those materials.

ADMIN. JUDGE WARDWELL: And the fluence screening and any molybdenum screening didn't play a part in this?

DR. HISER: The molybdenum screening did not because this was a low molybdenum material. The fluence level did from the perspective that the fluence exceeded one dpa and, therefore, the 15 percent delta ferrite screening level was implemented instead of the 20 percent that would apply for lower fluence CASS material.

ADMIN. JUDGE WARDWELL: Thank you. Dr. Lahey, do you agree that this was an appropriate screening process and that the lower column caps have been successfully screened and are being assessed in the way they should be for irradiation embrittlement alone?

DR. LAHEY: Yes, your honor. Given the criteria that's been established, it sounds like it has been properly done.

ADMIN. JUDGE WARDWELL: Thank you, sir. New York State's testimony on 482, Page 18, Lines 16 through 22, discusses a recent report prepared by Argonne National Laboratory for the U.S. NRC and acknowledges with respect to CASS materials that are used for the lower support column caps that combined effect of thermal aging and irradiation embrittlement could reduce the fracture resistance even further to a level neither of these degrading mechanisms can impart alone.

And I guess I would just ask, and we've already verified that it's been screened correctly, so really what my question is here, I want to make sure what Argonne National Laboratory work and report that you were referring to in this statement. Do you recall, Dr. Lahey? Is it NUREG/CR-7184 or is there some other one? Because you go on, on Page 20, to talk about NUREG-7184 and then even the Chopra report, the degradation of light water reactor core internal materials due to neutron irradiation, which is NUREG-7027. And I'm just trying to sort out all these reports.

DR. LAHEY: Your honor, I'd have to go back and look to know for sure.

ADMIN. JUDGE WARDWELL: Okay. Add that on to your list.

DR. LAHEY: Yes.

ADMIN. JUDGE WARDWELL: And do try to get to bed by at least 3:00 a.m. if you could. Because we're starting here tomorrow at 4:00 aren't we?

(Laughter.)

CHAIRMAN MCDADE: Thirty.

ADMIN. JUDGE WARDWELL: 4:30, I'm sorry.

(Laughter.)

ADMIN. JUDGE WARDWELL: Yes, thank you. And specifically to get back to that --

CHAIRMAN MCDADE: I digress for a second, but Judge Wardwell has stayed on Central European Time forever. So he insists that we do start early and would prefer that we start at 4:00 Eastern.

ADMIN. JUDGE WARDWELL: No, because I wouldn't be ready. I need that time for being ready.

(Laughter.)

ADMIN. JUDGE WARDWELL: It's your testimony in 482 on Page 18, Line 16 through 22, and then it's your testimony Page 20 on Lines 6 through 10 where these various -- on 16 through 22, you talk about the Argonne National Laboratory report and then you go on, on Page 20 to cite both 7184 and 7027.

DR. LAHEY: Okay.

ADMIN. JUDGE WARDWELL: NRC's testimony on 197, Answer 164, Pages 96 to 97, states that the Staff found that Entergy adequately addressed Action Level 7 based on the following. One, Entergy evaluated the CASS components of the RVIs and that is limited to the lower support column caps. Entergy screened the column caps for TE and IE using plant specific materials data and determined that the column caps are not susceptible to TE. And Staff confirmed the results of the screening using its own screening criteria.

Three, Entergy provided information on fabrication, non-destructive NDE demonstrating that preexisting flaws are unlikely to exist in the column caps. Four, Entergy provided information on expected stresses and neutron fluences for the column caps that demonstrated the service induced cracking due to irradiation assisted stress corrosion cracking is unlikely. And, five, Entergy modified its reactor vessel internals program to include a link to a lead component that is an appropriate predictor of IASCC and IE for column caps with an appropriate schedule for performing the expansion inspection if necessary.

Therefore, the Staff found that the formation provided by Entergy provides reasonable assurances that the functionality in the column caps will be maintained during this period of extended operation. Entergy goes on in their testimony and states in Exhibit 616 under Answer 197, Page 131, that in regards to the inspection criteria, the lower core barrel cylindrical girth weld reveals that it is more likely to experience this irradiation assisted stress corrosion cracking than the lower support column caps. So it is appropriate to link the two with the weld as a primary component and the LSCC as the expansion component.

And I guess I'll start with Staff on this and say that, in your approval of this as summarized in that, that I just read, you mentioned about linking to a component that is the primary component and I thought earlier, which I think I read, you stated that the control rod guide tubes lower flange welds were the primary link to the lower support column caps. And now Entergy is claiming that it's the lower core barrel cylinder girth weld is the one. Which is it?

MR. POEHLER: At this time, it's both because they added the girth weld as a plant specific primary link. So in MRP 227-A, the generic primary link for the lower core support columns is the control rod guide tube lower flange. When the Staff was reviewing the Action Item 7 information submitted by Entergy, we were satisfied with what they had given us, but we still had a lingering concern that for irradiation embrittlement, because irradiation embrittlement still applies even though thermal embrittlement was screened out, we still had a lingering concern that irradiation embrittlement wasn't appropriately predicted by the standard MRP primary link.

Because that control rod guide tube lower flange weld is a relatively low fluence component, didn't even screen-in for irradiation embrittlement or irradiation assisted stress corrosion cracking, which are two mechanisms that do apply to the lower support columns. So we asked another RAI to Entergy regarding an appropriate -- can you propose an appropriate primary link for these mechanisms? And that's when they proposed the lower core barrel girth weld. And we reviewed that information that they submitted and concluded that it was a more appropriate primary link for those mechanisms of irradiation embrittlement and irradiation assisted stress corrosion cracking.

ADMIN. JUDGE WARDWELL: Okay. Thank you. Dr. Lahey, do you have any reason to not accept those as appropriate primary links to these expansion components?

DR. LAHEY: The use of the girth weld as a proxy for the core cap, it seems an odd choice actually because of the materials used and how they would respond to thermal and irradiation effects. And I know they needed to use something because they weren't able to do an inspection of the core caps, but it just seems it's a very odd choice. And I must say when I was reading the ACRS testimony on this, they had exactly the same opinion. So I know what they did, it just seems odd.

ADMIN. JUDGE WARDWELL: And do you feel it's also odd to use the control rod guide tube lower flange welds as the other primary?

DR. LAHEY: It's not a casting. Yes.

ADMIN. JUDGE WARDWELL: So, Entergy, why have you proposed the lower core barrel cylinder girth weld as the primary?

DR. LOTT: Well, as Mr. Poehler explained, it was basically a request coming from the NRC to consider an alternative to the lower control rod guide, I can't say it either myself, the CRGT lower flange welds. And we looked at similarities in materials and we understood that -- we looked at the lower support column caps as a low ferrite material, not susceptible to thermal embrittlement. And we looked at the core barrel weld material and realized it also would have a duplex structure, a low ferrite content, and then some very similar characteristics. Both components had similar fluences and in an abundance of caution here, I think even though we probably could have demonstrated that it would screen-out for irradiation assisted stress corrosion cracking, I think we just wanted to make sure that we were looking for the most likely locations for cracks.

ADMIN. JUDGE WARDWELL: Okay.

DR. LOTT: And let me just one further -- I think the tipping point that made the core barrel the primary location for this particular situation was that based on the potential for residual stresses in the core barrel weld, we expect the stresses, particularly on the surface, to be much higher in the core barrel weld, where as we've just explained, we looked through the normal operating stresses in the lower support columns and found they're very limited. So we would expect to see cracking in these kinds of materials at these fluences first in the core barrel and later in the lower support column, if ever.

ADMIN. JUDGE WARDWELL: Okay. What about the control rod guide tubes lower flange welds, will they also show up any cracking prior to the lower support column caps?

DR. LOTT: Well, again, they have some potential to have some residual stresses and some other concerns that might put them ahead on the list. But because they weren't irradiated, we felt that this addition that Entergy has of using both as a primary component made the most sense.

ADMIN. JUDGE WARDWELL: Now, then for Staff

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CHAIRMAN MCDADE: If I could interrupt for a second. Dr. Lott, when we come back to you, if you could just state your name, otherwise --

DR. LAHEY: Okay.

CHAIRMAN MCDADE: -- Mr. Cox is going to get --

DR. LAHEY: I'm sorry.

CHAIRMAN MCDADE: -- blamed in the transcript for things that you say.

DR. LAHEY: Yes, I'm sorry.

CHAIRMAN MCDADE: Okay.

ADMIN. JUDGE WARDWELL: Take credit. Dr. Hiser, this discussion raises a question in my mind, why shouldn't these lower support column caps be elevated to a primary component? And what is the criteria that puts something as a primary component as opposed to an expansion component, if I've used the right two terminologies?

DR. HISER: This is Allen Hiser of the Staff. One of the reasons is that the column caps are a highly redundant structure. And, therefore, it would require multiple failures for the functionality of the lower core support system to be challenged. In this case, as Dr. Lott mentioned, the residual stresses in the core barrel weld create a situation where we expect that cracking would initiate much sooner than in the column caps. Therefore, we believe it's a reasonable linkage with the column caps being an expansion component predicated on inspection findings from the core barrel girth welds.

ADMIN. JUDGE WARDWELL: And why still hold with the control rod guide tubes lower flange welds? Why should that still be a primary for this? It sounds like you don't feel comfortable that it will supply any advanced notice of what might happen or substitute notice for what might happen at the lower support column caps.

DR. HISER: This is Allen Hiser with the Staff. The CRGT welds, I believe, are still a primary link for the column caps. Again, because of weld residual stresses, we would expect the likelihood of stress corrosion cracking to be higher there. So they, again, if you were to look at a hierarchy of where you expect things to occur first, the CRGT welds and the lower core barrel girth weld would be higher than the column caps.

ADMIN. JUDGE WARDWELL: Is it possible to inspect the lower support column caps?

DR. HISER: From discussions we've had with

Westinghouse, I believe it is possible, yes.

ADMIN. JUDGE WARDWELL: Okay.

DR. HISER: Our understanding is that it would be an extremely difficult inspection to do, but it is certainly possible and would be necessary pending the results from the primary inspections, if expansion was required then the plant would be required to do those expansion inspections.

ADMIN. JUDGE WARDWELL: Dr. Lahey, after hearing this additional explanation, do you have any other comments in regards to the expansion monitoring plan for the lower support column caps?

DR. LAHEY: No. I understand what they said and I understand the rationale. Just personally, I think if you're interested in core cap, it's better to do the inspection on the core cap, even if it's a more difficult inspection.

ADMIN. JUDGE WARDWELL: Okay, thank you. New York State's testimony, 482, Page 59, Lines 1 through 3, states that the Applicant, based on a lack of documented fractures of core support columns, assumed that only a limited number of columns could actually contain flaws of significant size. And I guess I'll ask Entergy, did you assume only a limited number of lower support column caps contained flaws or was this the result of any analysis or conclusion or was it just an assumption?

DR. LOTT: I'm not sure exactly where we think we -- my name is Randy Lott, I'm sorry, thank you. I'm not sure exactly --

ADMIN. JUDGE WARDWELL: Alan doesn't want to take credit for what you're saying or Mr. Cox doesn't want to take credit for what you're saying.

DR. LOTT: I'm not sure exactly where we're saying we made this assumption. We certainly did do analyses as Dr. Hiser has suggested to look at effectively the redundancy in the system, in the system of calculations that are very similar to what's done for the baffle former bolts to look at minimum requirements, and found that there was a large degree of redundancy in the system. That's all reported in -- it's a proprietary report, but it's PWROG14-048, which is ENT 667.

ADMIN. JUDGE WARDWELL: Did you have any other knowledge of any potential flaws or lack thereof in the lower support column caps?

DR. LOTT: We have no expectation of flaws.

MR. DOLANSKY: This is Bob Dolansky with Entergy. I just want to add, we also -- for the lower support column caps, when they were installed they had detailed NDE radiography performed on them, which would give very, very high confidence that there were no flaws in those when they were installed.

ADMIN. JUDGE WARDWELL: And what is that testing composed o?

MR. DOLANSKY: Basically similar to an x-ray on a person. It's an x-ray of the lower support column cap, was performed on all column caps, and determined that there was no flaws in -- it's the, I don't want to say the best, but it's an extremely good NDE technique used to verify that there's no flaws.

ADMIN. JUDGE WARDWELL: Thank you.

CHAIRMAN MCDADE: If I could jump in here quick between questions and get something clarified. A while ago, a few minutes, we were talking about, and I believe that you said, Dr. Lahey, that you thought it was an odd choice of using the core barrel girth weld as a proxy for another item. Do you recall that testimony?

DR. LAHEY: Yes.

CHAIRMAN MCDADE: Okay. And --

DR. LAHEY: For the column caps.

CHAIRMAN MCDADE: -- the control guide tube flange, was it or --

DR. LAHEY: Well, I believe we were talking about that and the cast column cap that was --

CHAIRMAN MCDADE: Okay. But what I'm trying to get at is, what I thought I heard you say is that you thought that it was an odd proxy because of the differences in the materials that they were made out of. And then I thought I heard Dr. Lott testify that part of the reason that they were chosen is because of the similarity of the materials. So, let me go to Dr. Lott, would you state for the record what components we're talking about and what they're made of?

DR. LOTT: Okay. This is Randy Lott for Entergy. We're talking about the large structural weld basically around the middle, I'll call it the belt line, of the core barrel and these column caps, which obviously we've been discussing all along, which are CF-8 material, but with a relatively low ferrite content of about 15 percent. They're castings, which means that effectively they're melt, cast, and cooled in the system, which gives us this duplex grain structure. In fact, we need some duplex structure in order to get a successful casting, so that's -- in a weld, you get a very similar micro-structure or material structure because, again, it's a material that's welded and solidified.

And you get very similar behavior in terms of duplex construction of those material, albeit sometimes a slightly lower ferrite content. So, in the sense that they're both these duplex Austenitic ferritic structures, there's some real similarities in the contents. Particularly, it seemed to us, because there was particularly low ferrite content at Indian Point, it made sense to couple up these two materials. Then, as I said, the controlling factor for us was the large potential residual stresses, which would drive much more cracking in the core barrel, much sooner, than it would in the lower support columns.

CHAIRMAN MCDADE: Okay. Thank you, Dr. Lott. And, Dr. Lahey, what were the dissimilarities that you thought made it an odd proxy?

DR. LAHEY: This is Richard Lahey from New York State. The dissimilarities are the metals themselves. The weld has typically significantly lower delta ferrite than would be in the casting. And the stresses on the structure are significantly different. As I understood it, one of the things they liked about the two were the potential for corrosion cracking. But it's a stretch, I just think it's quite a stretch to look at one and say, it's going to happen here before it happens there. So they're looking for the canary in the coal mine to decide what they might want to do on the core cap. And I, frankly, I think it's better to look at the coal mine, go down and check the core cap.

CHAIRMAN MCDADE: Okay. Thank you, Dr. Lahey.

ADMIN. JUDGE WARDWELL: Do you have any reason to believe that the cracking would not occur sooner in the lower core barrel cylinder girth weld prior to anything happening with the lower support column caps?

DR. LAHEY: Your honor, I think the stresses on the things are -- there's no reason for them to be similar. You have one loaded supporting structure, you have the other one loaded in this way. It's just -- to me it's a very odd choice. I heard what they said, I do understand why they decided to do it, but primarily I think it's driven by accessibility.

ADMIN. JUDGE WARDWELL: Thank you.

DR. HISER: Judge Wardwell, could we supplement that a little bit?

ADMIN. JUDGE WARDWELL: Yes, you may.

DR. HISER: I think you've heard a little bit -- this is Allen Hiser of the Staff. You've heard a little bit about the reasons that the Applicant thought that was acceptable. The Staff basis is sort of similar, but there may be at least two additional factors. As Dr. Lahey mentioned, the column caps will be in compression. The core barrel girth weld, the CRGT welds will not. So it's much more likely that they will exhibit stress corrosion cracking because of the weld residual stresses first and the higher membrane stresses from the operational loads.

In addition, the column caps are cast Austenitic stainless steel, which has generally been found to have a very, very low likelihood of initiating stress corrosion cracks. So, therefore, the material in the column caps is much better from a likelihood of stress corrosion cracking. The other materials have much higher stresses and so they will be much more likely to have cracking prior to the column caps. So that provides a more complete picture, I think, of the basis for the Staff finding that to be acceptable.

ADMIN. JUDGE WARDWELL: Thank you, Dr. Hiser. New York State's testimony, 482, Page 61, Lines 11 through 17, states that the reactor vessel internal components made from non-cast stainless steel will also experience the combined effects of irradiation induced embrittlement, corrosion, and other aging mechanisms. The Applicant has failed to evaluate the mechanisms that occur for many of the other important and vulnerable RVI components, such as the core baffles, the baffle bolts, and the formers.

Entergy's testimony, Exhibit 616, Answer 202, Page 135 to 136, states that these components are not made of cast materials, so they are not susceptible to thermal embrittlement. MRP 227-A and the reactor vessel internals AMP identify irradiation assisted stress corrosion cracking which, as the name implies, is actually the result of multiple underlying mechanisms itself as the aging mechanism of concern for these components, again referring to the core baffles, baffle bolts, and formers. And that these components are all designated for primary inspections under the RVI AMP and Inspection Plan. And I guess I'll start with Entergy just quickly to confirm that when you were referring to these components, you were referring to the core baffles, baffle bolts, and formers that you claim are designated as primary inspections under the RVI AMP and Inspection Plan. Is that correct?

DR. LOTT: Yes, I believe that's true.

ADMIN. JUDGE WARDWELL: Say --

DR. LOTT: Yes, I believe that's true. This is Randy Lott.

ADMIN. JUDGE WARDWELL: Thank you. And, Dr. Lahey, do you now agree that those core baffles and the baffle bolts and the formers are handled by the RVI AMP?

DR. LAHEY: It's clear that they are being inspected using the techniques in MRP 227-A, if that's what you mean.

ADMIN. JUDGE WARDWELL: Yes.

DR. LAHEY: Yes.

ADMIN. JUDGE WARDWELL: Okay. Do you have any other comments on that? And they're being inspected as the primary components, is that your understanding also?

DR. LAHEY: Yes, that's my understanding.

ADMIN. JUDGE WARDWELL: Okay, thank you. That's all I have.

ADMIN. JUDGE KENNEDY: This is Judge Kennedy. Just going back to the baffle former bolts and maybe this most recent discussion answered the question, but earlier in today's testimony, you had a colorful expression for the baffle former bolts being subject to shock loads and you referred to unzipping the rest of the bolting.

DR. LAHEY: Yes.

ADMIN. JUDGE KENNEDY: Is this still a concern of yours given what we've gone through this afternoon and all the other testimony on the baffle former bolts?

DR. LAHEY: Yes, your honor, it is.

ADMIN. JUDGE KENNEDY: Within your testimony, do you have any support for the assertion that these bolts, when subject to a shock loading, would fail catastrophically?

DR. LAHEY: If they are significantly weakened, we're talking as we get out in time and they've been irradiated significantly, they've been subjected to fatigue, they've been subjected to irradiation assisted stress corrosion cracking, and if they're significantly weakened in that way and they're subjected to a strong shock load, yes, I have a serious concern about it.

ADMIN. JUDGE KENNEDY: Is this combination of aging mechanisms all applicable to these bolts?

DR. LAHEY: I'm sorry, could you --

ADMIN. JUDGE KENNEDY: Is this combination of aging mechanisms all applicable to these --

DR. LAHEY: I believe they are, yes.

ADMIN. JUDGE KENNEDY: Entergy, do you feel that this combination of aging mechanisms would be applicable to the baffle former bolts?

DR. LOTT: Yes. I mean, to the baffle former bolts -- can you name the mechanisms again for me -- I'm sorry, I'm -- they're subject to irradiation embrittlement, they're subject to irradiation assisted stress corrosion cracking, they're subject to irradiation induced stress relaxation, they're potentially subject to void swelling. So, yes, they're -- and we model all of those components.

ADMIN. JUDGE KENNEDY: I guess the other one I heard was metal fatigue. I don't know --

DR. LOTT: Metal fatigue is, yes, certainly a possibility.

ADMIN. JUDGE KENNEDY: So is this now back to the question of how much synergism there is between all these aging mechanisms, if there is any?

DR. LOTT: Well, I mean, I will point out that our experience with baffle -- we have experience with failure rates in baffle former bolts, a fair amount. We have a fair amount of data now on baffle former bolts. So, we have -- I don't see a lot of surprises coming forward in this testimony and because we're monitoring, as we said, the effects and if we're looking for the effects on cracking, whatever combined effects are there, are there in that data base or in that operating experience.

ADMIN. JUDGE KENNEDY: Does this data include bolts subjected to higher fluences, towards the end of design life? Maybe not extended life, but design life at least?

DR. LOTT: Yes. I mean, some of these examinations have been performed as part of similar plant life license renewal applications, PEOs. So, yes, they've been towards the end of plant life.

ADMIN. JUDGE KENNEDY: So it does include some bolting that has been examined during the period of extended operation for other plants?

DR. LOTT: At or near.

ADMIN. JUDGE KENNEDY: Okay. At or near. I don't want to put words in your mouth.

MR. STROSNIDER: This is Jack Strosnider for Entergy. I'd like to come back to the first part of your question on loads for just a minute if I could.

ADMIN. JUDGE KENNEDY: Okay, sure.

MR. STROSNIDER: We talked earlier about the WCAP report that NRC reviewed and approved with regard to the methodology for establishing the bolting patterns, et cetera. That WCAP report evaluated the design basis dynamic loads associated with design basis accidents, such as the loss of coolant, et cetera. So to the extent that Dr. Lahey's concern about dynamic loads, that those loads are within the original design basis, they have been addressed. If there's something over and above what was in the original design basis, I'm not -- it's not clear, to me at least, what they are and also I would suggest that they're outside the space of license renewal. But it's just not clear what loads he's talking about that would be over and above what's in the licensing basis. And if they're within the licensing basis, they were evaluated.

ADMIN. JUDGE KENNEDY: And would those bolts that were subjected to the design basis loads include the effects of these various aging mechanisms? Consideration of the effects of these various aging mechanisms? Which seems to be the other part of Dr. Lahey's concern.

MR. GRIESBACH: Your honor, this is Tim Griesbach for Entergy. If you're asking whether those multiple aging mechanisms have been considered as part of the Aging Management Program for the baffle former bolts, as Dr. Lott said, the answer's definitely yes. They've also been identified as primary components for inspections and, to the best of my knowledge, there's been at least 19 PWR units that have performed those inspections already for MRP 227 and very few failed bolts have been found throughout those inspections.

ADMIN. JUDGE KENNEDY: Is the foundation of the strength of this bolting subject to these aging mechanisms the lack of cracking? I mean, if it hasn't cracked, it still maintains its design fatigue life or design life?

MR. GRIESBACH: This is Tim Griesbach again for Entergy. The answer is yes. Those materials undergo strengthening, in fact, due to the irradiation effects. So they are stronger than they would have been initially and still maintain the margins that were there from the beginning for that reason, if they are uncracked.

ADMIN. JUDGE KENNEDY: And we're back to the beginning. All right. Thank you very much.

DR. LAHEY: Your honor, I think I've been asked the question many times, are my loads different than your loads? They're the same loads. The only difference is how impulsive these loads are. So it has to do with how they're calculated. I haven't seen the document which gives the details on that. If they have a shock capturing routine, like adaptive grid, or they're done by method of characteristics or some way where they don't smear out things, instead of hitting it with a hammer, you're hitting it with a powder puff, then I'm happy that everything's surviving. But I haven't seen that. And I've done a lot of work in the past on RELAP and TRAC and RETRAN, so if you're using those kinds of codes, it's more like a powder puff than it is a hammer. So that's the difference.

ADMIN. JUDGE KENNEDY: Would Entergy like to respond to that? I mean, again, this is a question of how you analyze the dynamic loads.

DR. LOTT: Well, I guess --

CHAIRMAN MCDADE: Dr. Lott?

DR. LOTT: I'm sorry. This is Randy Lott. That information, I think, is dealt with in the methodology documents that we talked about earlier, and I'm sorry I don't have the number right in front of me, but I think we discussed earlier the approved methodology for analysis of the bolts, the computer programs and the process are in that document. We believe it's more than adequate and consistent with the licensing basis.

MR. STROSNIDER: This is Jack Strosnider for Entergy. I'd like to add to that, that the codes that are used for doing these analyses are codes that have been reviewed and approved by NRC as acceptable for analyzing the design basis loads. So without going into all the details of how those calculations are done, which would require a code expert, you should at least recognize that the NRC has done that type of review and concluded that these codes are acceptable for the application.

ADMIN. JUDGE KENNEDY: Thank you. Dr. Lott, I believe earlier Entergy testified that the Topical Report is an exhibit in this proceeding. Is that true? The bolting methodology?

DR. LOTT: The approved -- yes.

ADMIN. JUDGE KENNEDY: Dr. Lahey, did you get a chance to review this exhibit as part of your --

DR. LAHEY: You're talking about the WCAP report that they --

ADMIN. JUDGE KENNEDY: Yes, sir.

DR. LAHEY: -- talked about? I haven't seen it yet, I was planning to look at it this evening if I can get my hands on it. I mean, one thing that --

ADMIN. JUDGE KENNEDY: Did I miss one of your homework assignments? Is this --

DR. LAHEY: Yes, I know, it's just adding up. One thing you folks should know though is, I was very involved with the development of RELAP and TRAC. So I'm very familiar with that methodology, what it does and what it doesn't do. And let me say one last time, hopefully the last time, after we assured ourselves that these significant shock loads did not distort the core geometry, we then focused on, do you have the right mass in the right place, is the heat transfer right, could you cool the core adequately? That's what these codes are intended for. It's true, they calculate pressure versus time and temperature versus time, but they're in no way sharp shock load codes. They're not intended for that purpose.

ADMIN. JUDGE KENNEDY: All right, thank you. Appreciate it. I have nothing further.

CHAIRMAN MCDADE: Okay. Very quickly, this may have been covered, but it's sort of a question in my mind and I just want to clarify it. You've done the UT examination on baffle former bolts for the past 20 years. Yet, in the testimony, it talks about the UT examination acceptance criteria for the baffle former bolts will be developed. And the question is, what are you using now?

MR. DOLANSKY: This is Bob Dolansky with Entergy. The plants that have performed UT of baffle bolts would have that technical justification. Indian

Point has not performed UT of baffle bolts yet. So what we're getting is a plant specific technical justification for the UT of the baffle bolts, that's why we don't have it. So when we say that 8,000 bolts have been inspected, the technical justifications for those, I can't speak specifically for each plant, but I would assume each plant has one. I know that when we go to do our baffle bolts inspection, we will have a plant specific technical justification. And that's what's being developed for us now.

CHAIRMAN MCDADE: Okay. One of the concerns that we have is the differentiation between the development and implementation of a plan. That the development of the plan being something to be done that we can look at, the implementation is going to go on and is going to be monitored by the NRC during the period of extended operation if the license is granted. But is there assurance that these acceptance criteria that are to be developed will be adequate to ensure the continued operation?

MR. DOLANSKY: I believe the acceptance criteria is contained now in the AMP. This is Bob Dolansky, I'm sorry. So if we go to our --

CHAIRMAN MCDADE: I mean, here's the thing, and correct me if I'm wrong, any time you use the term will be developed, it sort of is a red flag for us. And as I understand, the term that you're using is that you basically have a procedure in existence now, but the plan is that you will be using perhaps more up to date technology down the road and that the acceptance criteria will recognize that increased data, increased knowledge at the time. But it's not that there are no acceptance criteria now, it's that just simply they're going to be updated based on newer technology, additional information. Am I incorrect in that?

MR. DOLANSKY: I want -- at the first part of your question, you said that we have a procedure now to perform baffle bolt inspections. Did I understand you correctly, is that what --

CHAIRMAN MCDADE: Let me ask the question. In the AMP as it exists right now, are there acceptance criteria for the UT examination of the baffle former bolts?

MR. DOLANSKY: No. Right now, the examination acceptance criteria for UT of the baffle former bolts shall be established as part of the exam technical justification.

CHAIRMAN MCDADE: Okay. So to Dr. Hiser, how does the NRC determine now that these acceptance criteria to be developed will be adequate to ensure the proper functioning of the baffle former bolts over the period of extended operation?

DR. HISER: This is Allen Hiser of the Staff. Our understanding is that for individual baffle bolt examinations, any indication of cracking from the UT exam indicates that, that bolt is no longer functional. So it represents a failure. I believe the analysis that Mr. Dolansky is discussing is post-inspection, when they have identified each bolt, whether it is acceptable or unacceptable, and they're looking more at the configuration whether it is acceptable. So the acceptance criteria for the final configuration post-inspection is, I believe, what they are still working on. But for each individual bolt, it's crack/no crack, unacceptable/acceptable, is our understanding.

MR. DOLANSKY: That's correct. Any indication of cracking will be considered a degraded bolt and will be put into our corrective action program.

DR. HISER: So I think -- this is Allen Hiser again. I think there's just been a confusion between acceptance criteria on a piece inspection basis versus the assembly basis after one has inspected all of the bolts. And I think that the terminology has gotten a little bit mixed up. So the inspection acceptance criteria is any crack is a failed bolt. The assembly acceptability criteria is different. That is to be determined. From the NRC perspective, that's a corrective action that is within the purview of the applicant. If they require approval of any of that, then they would come to the NRC for that.

CHAIRMAN MCDADE: Okay. But the way you've just described it, that corrective action may or may not be adequate to ensure the continued viability and the continued operation, the utility of that particular part. So how does the NRC assure itself now, and New York and Riverkeeper assure themselves now, that something that still has to be developed will be adequate? Do you understand what my concern is?

DR. HISER: I think so. The corrective actions, there are three potential paths that they could take. If they were to find degraded bolts, they could replace the bolts. Repair really is not an option in this case, but they could replace the bolts. If they wanted to continue to operate with degraded bolts, they would need to submit an analysis to the

NRC that justified their configuration that they want to continue operating with. From that perspective, we still have approval of any operation with the degraded condition. So I think from that perspective, I believe that there are sufficient controls on this.

MR. STROSNIDER: This is Jack Strosnider for Entergy. The one other thing I think we should add here is what we talked about earlier. The methodology that's being used to develop the plant specific bolting pattern, that acceptable bolting pattern, that methodology was reviewed in the WCAP report that was submitted to NRC, it's an extensive review with all the Requests for Additional Information, the back and forth, the safety evaluation written. So they're using what the NRC has looked at and concluded that, that's an acceptable methodology. You do it on a plant specific basis to make sure that if anything plant specific comes up that you've accounted for it. But you should be able to move forward with that methodology and come up with an acceptable technical evaluation for the acceptable bolting pattern.

MR. COX: This is Alan Cox --

MR. STROSNIDER: That's the whole purpose of, as I explained earlier, the Topical Report process and reviewing these methodologies up front. The NRC does not want to look at one of these -- do that same review over and over and over again for every plant. So they look at the generic methodology, if the plant can apply it within the boundary conditions that, that methodology is good for, then they can go do the plant specific evaluation and you should have confidence in it.

MR. COX: This is Alan Cox with Entergy. Let me add one more thing. I believe the Standard Review Plan provides for, when it talks about acceptance criteria, it says the acceptance criteria doesn't necessarily have to be a formal value. It can be a description of the method that will be used to establish that value. And I would say that it's not practical to do all that in advance. You're talking about finding a -- you're going to find a certain pattern of bolts where you have a failed bolt, there's 832 bolts, you can't predict or it would be impractical to try to do an analysis for every combination of which bolt might be failed where.

So you have a method established, you take the results of the inspection, you look at the bolts that are failed, and you apply that data using the method that's established and that's how you determine your acceptance criteria. It would be impractical to evaluate all of those combinations ahead of time that you could have with the inspection results.

MR. DOLANSKY: This is Bob Dolansky. I just want to add one other thing. The inspection of baffle bolts, the fact that it's been going on for years and the fact that other plants have done it means that all this stuff has been looked at before. So, I mean, it's not like a first of its kind. We're not the first ones doing it, other plants within the industry have performed these inspections, they've had NRC reviews of both during the inspection and after the outage. So not only has the Topical Report been reviewed by the Staff, but the actual implementation has been reviewed by the Staff at other plants.

CHAIRMAN MCDADE: And the preface to my question assumed that and recognized that these inspections have been going on for more than 20 years and that there has been a method established. And what I'm trying to just make sure that I want to cement in my mind is what is currently established and what, if anything, by way of methodology still needs to be developed? As Mr. Cox pointed out, I think it's relatively obvious that you can't say what you're going to do as the result of an inspection until you get the results of that inspection and then you have to assess the meaning of that.

But in developing a plan that provides assurances that you're going to have continued safe operation, you look, is the methodology used adequate to develop an appropriate response? And all I'm trying to do is cement for the record of, what is the current status of development? Again, the language you used, will be developed, which sort of, well, more than sort of, which suggests that there is still a sort of, we'll figure this out in the future, we hope. And from what Dr. Hiser has said, it isn't that and I just wanted to get as clear as possible what we currently have and what it is that still needs to be done so we can determine whether or not there exists now reasonable assurance of continued safe operation. Can you elaborate on that?

MR. AZEVEDO: Yes, your honor. This is Nelson Azevedo for Entergy. Maybe it's already clear, but just to make sure it is clear. The methodology is established, has been approved by the NRC, that's what we're using. What is going on right now is taking that methodology in developing a model specifically for Indian Point. But the methodology itself, how it is done, is established, it's been approved by the

NRC, and it's what Indian Point is using.

CHAIRMAN MCDADE: And that's part of the AMP for these baffle former bolts? Where do we look to see where that methodology is and the fact that it's been approved by the NRC?

MR. AZEVEDO: It's referenced in MRP 227, that's what Mr. Cox was telling you.

MR. COX: This is Alan Cox. Chapter 6 of MRP 227 has a discussion on evaluation of bolts and pins. And basically it says that you don't have to do the individual evaluation, you have to look at the effect on the assembly. And then Section 6.4 gives you the guidance on doing assembly level evaluations.

CHAIRMAN MCDADE: Okay. And, Dr. Hiser, if that were to change, is that something that the Applicant would be able to do and inform the NRC? Is it something that would be subject to the 50.59 procedure? Or is it something that would require a license amendment?

DR. HISER: This is Allen Hiser of the Staff. My guess is it would be subject to 50.59. Given the safety implications and -- my expectation would be that it would not pass 50.59 and would require a submittal to the NRC.

CHAIRMAN MCDADE: Okay. And once the plant specific details are worked out, what is the vehicle for NRC review of that plant specific? Do they submit it to you, is it subject to any kind of a formal review and approval? Or is it simply a situation where you would have the ability to review and comment?

DR. HISER: This is Allen Hiser of NRC. My expectation is that, that would require at a minimum a submittal to the NRC. And could involve a license amendment. But I'm not certain of that.

CHAIRMAN MCDADE: Okay. Two separate things. One, we were talking about the methodology set out in Chapter 6 of MRP 227. And we were talking about what would be necessary if that methodology were changed. My question right now is not the methodology having been changed, but rather, as Entergy explained, they are taking that methodology and are currently working to make a plant specific program. And my question is, once they have completed the details of that plant specific, how is that subject to review by the NRC?

DR. HISER: This is Allen Hiser. My prior statement really still holds. Whether it's an approved generic methodology or a plant specific methodology would require the same approval, if approval is necessary. The only difference in that case is for the plant specific methodology, we would have to review the entire methodology instead of relying on the generic approval from the generic methodology that previously was approved.

CHAIRMAN MCDADE: And if you viewed it to be unacceptable, would this be going at that point through the 50.59 procedure or is this something less than the 50.59 procedure?

DR. HISER: I'm sorry, if we found what to be unacceptable?

CHAIRMAN MCDADE: Their plant specifics. If you identified problems with the plant specifics, how does that work through?

DR. HISER: If it is a license amendment request, then they would need to modify their approach to become acceptable. In the absence of approval of the license amendment, they could replace the bolts that they found to be degraded.

CHAIRMAN MCDADE: Okay. But I -- at least I had not anticipated at the level of a -- if it's a license amendment, then it's a situation where New York gets a notice of an opportunity for a hearing and has the opportunity to challenge the adequacy. What I'm trying to get at is once they take the methodology that is in existence, that the NRC has reviewed, has approved, is satisfied with, and apply it specifically to the plant, what does the NRC do then and what is the nature of review and the public aspect of that, if any?

DR. HISER: This is Allen Hiser again. If we can just discuss for a moment here.

CHAIRMAN MCDADE: Well, we're basically at the end of the day. Do you want to answer that at the beginning --

DR. HISER: Sure.

CHAIRMAN MCDADE: -- of tomorrow rather than -- I mean, to me this is something that's important and I don't want an answer off the top of your head that I think this is the way it should be. So, why don't we leave that and take that up first thing in the morning?

DR. HISER: Okay. That's acceptable.

CHAIRMAN MCDADE: Okay. Do you have anything further for --

ADMIN. JUDGE WARDWELL: I do not.

CHAIRMAN MCDADE: Okay. I would propose that we would start tomorrow at 8:30. Before we break, NRC is there any administrative matters or other matters that you want to take up before we break for this evening?

MR. HARRIS: No, your honor.

CHAIRMAN MCDADE: Entergy?

MR. KUYLER: Yes, your honor. A few minutes ago, we were discussing the Westinghouse proprietary reports regarding the baffle former bolts and the minimum bolting pattern analysis. And I believe one of the things that Dr. Lahey said was that he had not reviewed those reports. And we just wanted to note that those have been in the record for several months and they've been disclosed several years ago. And, so, for us to look at that for the first time afresh tomorrow would be challenging, your honor.

CHAIRMAN MCDADE: Well, I mean, these are exhibits, are they not?

MR. KUYLER: Yes, they are.

CHAIRMAN MCDADE: So, I mean, if there's something that's already admitted as an exhibit that will help clarify Dr. Lahey's testimony, we will allow him to refer to that and will of course give Entergy the opportunity, if their experts need some additional time to respond. Quite frankly, so far I've been amazed at the capacity of all of our witnesses to know a rather voluminous record and to be able to testify with regard to the contents of the literally tens of thousands of pages of documents that we have. So I think you may have underestimated your experts, but in any event, we would certainly give them time and the opportunity to refresh themselves with the documents, with the reports in order to comment on Dr. Lahey's testimony.

MR. KUYLER: Yes, your honor. But I would just note that we have not seen any specific criticisms of those documents from Dr. Lahey in the past. So for him to introduce those on the very last day of the hearing for the first time, we would object to that, your honor.

CHAIRMAN MCDADE: We don't know whether he's going to criticize them. He may look at them and say, this is great. This allays my fears, if I had read this before, if I had studied it harder, I wouldn't be here. We don't know, we'll find out tomorrow. Mr. Sipos, anything before we break for tomorrow?

MR. SIPOS: Nothing further from New York at this time.

CHAIRMAN MCDADE: Ms. Brancato?

MS. BRANCATO: No, your honor.

CHAIRMAN MCDADE: Okay. Does 8:30 tomorrow pose a problem for anybody?

MS. SUTTON: No, your honor.

CHAIRMAN MCDADE: Okay, apparently not. We'll see you all at 8:30 in the morning. Thank you.

(Whereupon, the above-entitled matter went off the record at 5:36 p.m.)