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RE Indian Point Nuclear Generating Unit

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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10 CFR 2.206 PETITION REVIEW BOARD (PRB)

CONFERENCE CALL

RE

INDIAN POINT FUEL PEAK CLAD TEMPERATURE

DOCKET NOS. 50-247 + 50-286

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MONDAY

APRIL 18, 2011

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The conference call was held, Fred Brown, Chairperson of the Petition Review Board, presiding.

PETITIONERS:

PHILIP MUSEGAAS AND MARK LEYSE, representing RIVERKEEPER

PETITION REVIEW BOARD MEMBERS:

FRED BROWN, Chair, Director, Division of Inspection and Regional Support, NRR

JOHN BOSKA, Project Manager, NRR

TANYA MENSAH, PRB Coordinator, NRR

BRICE BICKETT, Region I Division of Reactor Projects

RICHARD DUDLEY, Rulemaking Branch, NRR

SHIH-LIANG WU, Nuclear Performance and Code Review Branch, NRR

PRB ADVISORS:

CHRISTOPHER HOTT, Office of Enforcement

BRETT KLUKAN, Office of General Counsel

P-R-O-C-E-E-D-I-N-G-S

1:01 p.m.

MR. BOSKA: Good afternoon. I'd like to thank everyone for attending this meeting. My name is John Boska, and I am the NRC petition manager for this petition.

We're here today to allow the petitioners, Ms. Brancato and Mr. Leyse, to address the Petition Review Board on behalf of Riverkeeper concerning their 2.206 petition dated March 28th, 2011, on the fuel peak cladding temperature at Indian Point Nuclear Generating Unit Numbers 2 and 3, which are located about 24 miles north of New York City on the east bank of the Hudson River.

I am the petition manager for the petition. The Petition Review Board chairman is Fred Brown.

As part of the Petition Review Board's review of this petition, Ms. Brancato and Mr. Leyse have requested this opportunity to address the Petition Review Board, which may also be referred to as the PRB.

This meeting is scheduled to conclude by 3:00 pm. The meeting is being recorded by the NRC Operations Center, and will be transcribed by a court reporter. The transcript will become a supplement to

the petition. The transcript will also be made publicly available.

I'd like to open this meeting with introductions. As we go around the room, please be sure to clearly state your name, your position, and the office that you work for within the NRC for the record.

I'm John Boska. I'm a project manager in the office of Nuclear Reactor Regulation, which is also referred to as NRR.

MS. SALGADO: This is Nancy Salgado. I'm a branch chief in the division of Operating Reactor Licensing, NRR.

MR. WU: Shih-Liang Wu, Nuclear Performance and Code Review Branch, NRR.

MR. KLUKAN: Brett Klukan. I'm the Office of General Counsel advisor to the PRB.

MR. BROWN: Fred Brown, Director of Division of Inspection and Regional Support within the Office of NRR.

MR. DUDLEY: Richard Dudley. I'm a rulemaking project manager from the NRR rulemaking branch.

MS. MENSAH: Tanya Mensah, 2.206 coordinator from the Division of Policy and Rule

Making, NRR.

MR. HOTT: Chris Hott. I'm an enforcement specialist in the Office of Enforcement.

MR. BOSKA: We have completed introductions in the room.

At this time, are there any other NRC participants from NRC headquarters on the phone?

Hearing none, are there any NRC participants from the Regional Office on the phone?

MR. BICKETT: Yes, this is Brice Bickett from NRC Region 1. I'm a Senior Project Engineer.

MR. BOSKA: Thank you, Brice.

Are there any representatives for the licensee on the phone?

MR. WALPOLE: Bob Walpole, Indian Point.

MR. IRANI: Adi Irani, headquarters.

MR. BOSKA: All right. Thank you.

Ms. Brancato and Mr. Musegaas, would you please introduce yourself for the record?

MR. MUSEGAAS: Sure. This is Philip Musegaas. I'm an attorney and the Hudson River Program Director at Riverkeeper.

And John, I apologize for any confusion. I will be giving our statement today on behalf of Riverkeeper. And I'm joined by Mark Leyse, but I'll

be on this call instead of Deborah Brancato. So again, sorry for any confusion there.

MR. BOSKA: All right. That's fine.

MR. MUSEGAAS: Thank you.

MR. BOSKA: Thank you, Mr. Musegaas.

Mr. Leyse, would you please introduce yourself for the record?

MR. M. LEYSE: Sure. Mark Leyse.

MR. BOSKA: All right. It is not required for members of the public to introduce themselves for this call. However, if there are any members of the public on the phone that wish to do so, please state your name for the record.

MR. LOCHBAUM: This is David Lochbaum, Director of the Nuclear Safety Project for the Union of Concerned Scientists.

MR. SIM: Hi, I am Bob Sim (phonetic), Assistant Attorney General, New York.

MR. BOSKA: All right. Welcome.

MR. R. LEYSE: Robert Leyse, citizen.

MR. WALD: My name is Matt Wald. I'm a reporter at the New York Times. I'm not a participant, I'm just listening. Thank you.

MR. BOSKA: All right. Thank you for the introductions.

I'd like to emphasize that we each need to speak clearly and loudly to make sure that the court reporter can actually transcribe this meeting. If you do have something that you'd like to say, please first state your name for the record.

For those dialing into the meeting, please remember to mute your phones to minimize any background noise or distractions. If you do not have a mute button, this can be done by pressing the keys star six. To un-mute your phone, press the star six keys again.

Please do not place this call on hold, since many phone systems play music when a call is on hold, which is very annoying for the other callers.

Thank you.

Next, I'd like to share some background on our process. Section 2.206 of Title 10 of the Code of Federal Regulations describes the petition process, the primary mechanism for the public to request enforcement action by the NRC in a public process.

This process permits anyone to petition the NRC to take enforcement-type action related to NRC licensees or licensed activities. Depending on the results of its evaluation, the NRC could modify, suspend, or revoke an NRC-issued license or take any

other appropriate enforcement action to resolve a problem.

The NRC staff's guidance for the disposition of 2.206 petition requests is in Management Directive 8.11, which is publicly available.

The purpose of today's meeting is to give the petitioners an opportunity to provide any additional explanation or support for the petition before the Petition Review Board makes an initial recommendation on whether or not to accept this petition for review.

The Petition Review Board typically consists of a chairman, usually a manager at the senior executive service level at the NRC. It has a petition manager and a PRB coordinator.

Other members of the board are determined by the NRC staff, based on the content of the information in the petition request.

At this time, I would like to introduce the Board. Fred Brown is the Petition Review Board chairman. I am the petition manager for the petition under discussion today. Tanya Mensah is the office's PRB coordinator.

Our technical staff includes Shih Liang-Wu

from the NRR's Nuclear Performance and Code Review Branch, Brice Bickett from the NRC's Region 1 Division of Reactor Projects, and Richard Dudley, from NRR's Rulemaking Branch.

We also have legal advice from Brett Klukan from the NRC's Office of General Counsel, and advice from Christopher Hott from the Office of Enforcement.

As described in our process, the NRC staff may ask clarifying questions in order to better understand the petitioner's presentation and to reach a reasoned decision whether to accept or reject the petitioner's request for review under the 2.206 process.

I would like to summarize the scope of the petition under consideration, and the NRC activities to date.

On March 28th, 2011, Ms. Brancato submitted to the NRC a petition under 10 CFR 2.206 regarding the fuel peak cladding temperature at Indian Point Nuclear Generating Unit numbers 2 and 3, which may also be called IP-2 and IP-3.

This petition is available from the NRC's public website, www.nrc.gov, from the electronic reading room under the ADAMS documents with the

accession number ML110890956.

The petitioners request that the NRC order the licensee of IP-2 and IP-3 to lower the licensing basis peak cladding temperatures of IP-2 and IP-3 in order to provide necessary margins of safety to help prevent partial or complete meltdowns in the event of loss-of-coolant accidents, also called LOCAs.

The petitioners state that experimental data demonstrates that IP-2 and IP-3's licensing basis peak cladding temperatures of 1,937 degrees Fahrenheit and 1,961 degrees Fahrenheit, respectively, do not provide necessary margins of safety in the event of LOCAs. Such data demonstrates that IP-2's and IP-3's licensing basis peak cladding temperatures need to be decreased to temperatures lower than 1,832 degrees Fahrenheit in order to provide necessary margins of safety.

Second, the petitioners request that the NRC order the licensee of IP-2 and IP-3 to determine how far below 1,832 degrees Fahrenheit the licensing basis peak cladding temperature values of IP-2 and IP-3 need to be lowered in order to provide necessary margins of safety.

Third, the petitioners request that the NRC order the licensee of IP-2 and IP-3 to lower both

of IP-2 and IP-3's licensing basis peak cladding temperatures to 1,600 degrees Fahrenheit until conservative values for IP-2 and IP-3 are determined.

Fourth, the petitioners request that the NRC order the licensee of IP-2 and IP-3 to demonstrate that IP-2 and IP-3 emergency core cooling systems, also called ECCS, will effectively quench the fuel cladding in the event of LOCAs and prevent partial or complete meltdown. Experimental data indicates that IP-2 and IP-3's ECCS may not effectively quench the fuel cladding in the event of LOCAs, if fuel cladding temperatures approached or reached IP-2 and IP-3's licensing basis peak cladding temperatures of 1,937 degrees Fahrenheit and 1,961 degrees Fahrenheit, respectively.

The petitioners also state that, although revisions to the 10 CFR 50.46(b)(1) of 2200 degrees Fahrenheit on peak cladding temperatures have been proposed in a rule-making petition, PRM-50-93, this petition has been filed separately under 10 CFR 2.206 because the concerns affect IP-2 and IP-3 and need prompt resolution to protect the lives, property, and environment of the people of New York. The safety issues raised in this petition are of an immediate nature and require prompt NRC review and action.

Allow me to discuss the NRC activities to date. On March 28th, 2011, the NRC received this petition. On April 6th, the petition manager contacted the petitioners to offer the opportunity to address the PRB, to which the petitioners agreed.

That led to this teleconference.

As a reminder for the phone participants, please identify yourself if you make any remarks, as this will help us in the preparation of the meeting transcript that will be made publicly available.

Thank you.

At this time, I'll turn it over to the Petition Review Board chairman, Fred Brown.

MR. BROWN: Good afternoon. Welcome to this meeting regarding the 2.206 petition submitted by Riverkeeper.

First, I would like to explain the purpose of this meeting. This meeting is not a hearing, nor is it an opportunity for the petitioners to question or examine the PRB on the merits or the issues presented in the petition request.

No decisions regarding the merits of this petition will be made at this meeting. Following this meeting, the Petition Review Board will conduct its internal deliberations. The outcome of this internal

meeting will be discussed with the petitioners.

At this time, I'd like to turn over to Riverkeeper the presentation to allow you to provide any information you believe the PRB should consider as part of this petition, especially reasons why this petition should be considered separately from the petition for rulemaking.

You may proceed.

MR. MUSEGAAS: Thank you, Mr. Brown. Again, this is Phillip Musegaas, representing Riverkeeper here at this PRB. I will make a brief opening statement, probably no more than five or ten minutes, and then I'll turn it over to Mark Leyse, who will explain and give a presentation about the technical basis for the petition.

I want to start out by thanking the NRC for setting up this meeting, and giving us an opportunity to present information on the petition.

Just as a little bit of a review, Riverkeeper is a non-profit, membership-supported organization. We're a 501(c)(3) environmental organization. We've been working on the Hudson River, protecting the Hudson River from pollution for about the past 40 years in one form or another.

We have been involved with Indian Point

issues, initially with environmental concerns when the plant was first licensed beginning in 1972, and then subsequently, after the terrorist attacks of September 11th, Riverkeeper filed a 2.206 petition related to safety and security issues at that time, and have been involved since 2001 on a range of safety and Nuclear Regulatory Commission issues related to Indian Point since that time.

Riverkeeper also petitioned to intervene in the NRC's license renewal proceeding for Indian Point in 2007.

Riverkeeper has significant current safety concerns about the operation of Indian Point at its current operating level.

And thank you, Mr. Boska, for spelling out our request for relief. I was going to do that, but I appreciate you reiterating that, and so I won't repeat that since you did that already.

And so I will discuss, as you requested, why Riverkeeper believes that this petition under section 2.206 should be accepted and why it's distinguishable from the rulemaking petition that has already been submitted.

The main concern here is that because Indian Point is operating at the level described in

the petition, fuel temperatures at the levels described in the petition, we have concerns that if there was a loss of coolant accident and oxidation occurred and a meltdown occurred, that the consequences of an accident would be severe.

And so I'm going to focus on the location of Indian Point, what we consider the enhanced risks of the plant's operation, given recent information, and, of course, the consequences of an accident.

Indian Point, as you mentioned already, is located approximately 25 miles from New York City, from the Bronx borough of New York City. Indian Point's about 34 miles from Times Square, which is roughly midtown Manhattan.

And within 10 miles of Indian Point, there are 300,000 people. Within 50 miles of the plant, there are approximately 17 to 20 million people, either living, or living and working, in that area.

Recently we have learned, over the past couple of years, new information about enhanced seismic risk at Indian Point. There is a 2008 Columbia University study done by a Lamont-Doherty Earth Institute seismologist, that concluded that in addition to the Ramapo fault that was already known to be near Indian Point, there was an additional fault

line.

And in addition, by looking at the history of earthquakes in the area, the Columbia seismologist concluded that Indian Point was, in fact, at risk of an earthquake up to 7.0 on the Richter scale.

Entergy, the current owners of Indian Point, have said publicly that they believe the two reactors could withstand up to a 6.1, I believe, on the Richter scale.

So we have significant concerns that the plant may not be built or designed to withstand the maximum earthquake that could be experienced in this area. And that is fairly new information.

This information from Columbia has not been considered or not been assessed to our knowledge by the Nuclear Regulatory Commission. In addition to the Columbia study, there is also a September 2010 Nuclear Regulatory Commission seismic risk study, which places Indian Point 3 at an increased risk of core damage under the core damage frequency calculations from an earthquake. And that information has not been considered in the license renewal process for Indian Point or in any other formal regulatory process that we are aware of. And so that is an outline of the seismic risks that we are concerned

about.

In addition to this, I've spoken about location, I've spoken about risk in terms of the seismic damage or the potential for an earthquake that's beyond the design basis of the plant.

I would also like to talk about the consequences again. And in our petition, Riverkeeper included and also cited, too, a 2004 report we commissioned from the Union of Concerned Scientists, Dr. Ed Lyman.

The report is called "Chernobyl on the Hudson," and that report describes how a loss of coolant accident could lead to extremely catastrophic near-term and long-term fatalities and economic and property damage in the area that would be affected by a significant radiological release from Indian Point.

And so, taken together, we have the location of Indian Point, we have the seismic risk, we have the consequences of an accident spelled out in an independent report.

Finally, I would just like to comment on the, -- and this relates back to location, Riverkeeper's ongoing concerns about the feasibility of evacuating both the 10-mile area around the plant, which is the designated emergency planning zone, as

well as the absolute lack of feasibility of evacuating a larger 50-mile zone around the nuclear plant.

And these comments are given in light of what has happened at Fukushima, Japan, and the NRC's recommendations to Americans living in Japan to evacuate to 50 miles from that plant.

And on that note, I will end my preliminary comments and turn it over to Mark Leyse. And I may give a few additional comments at the end of our presentation. But for now, I'll turn it over to Mark to give his presentation of the technical basis.

Thank you.

MR. M. LEYSE: Thank you, Phillip.

I first wanted to see, because I'm going to probably be maybe half an hour or so, I wanted to see if anyone else wanted to say anything before I started.

Okay. In such case, I will proceed.

One thing, just before I begin, I want to point out that the scenario we're discussing is a fast-moving accident, where Fukushima, I would classify as a slow-moving accident.

What we're talking about, you could have a partial meltdown that is well underway within ten minutes time after a large pipe break, but I will get

back to that.

And first, I want to start out -- that in the petition, I think the Petition Review Board should look at page 34, there's a statement that says that the Atomic Energy Commission, in responses to questions submitted by Anthony Z. Roisman, this was in the original Indian Point Unit 2 licensing hearing, Atomic Energy Commission stated that the calculated metal water reaction is negligible below 1900 degrees Fahrenheit.

Now, that is one of the major premises on which the original Unit 2 was licensed. And I want to point out that there is data from thermal hydraulic experiments that demonstrates that the zirconium-steam reaction is not negligible below 1900 degrees Fahrenheit.

I will talk about a test, it's from Thermal-Hydraulic Experiment 1, it's TH-1, and that's test number 130.

TH-1 test 130 was driven by small amounts of fission heat, such that it would simulate decay heat that would occur during a loss of coolant accident.

In TH-1 test number 130, the reactor shut down when the peak cladding temperature was

approximately 1,850 degrees Fahrenheit. And after the reactor shut down, cladding temperatures kept increasing because of the heat that was generated from the metal-water reaction.

And the peak measured cladding temperature was 2,040 degrees Fahrenheit, so there would have been a very small amount of heat that would have been added from actual decay heat.

I just want to explain, there was originally a very small amount of fission heat to simulate decay heat, so they're running at very low power, actually 0.37 kilowatts per foot, so there would have been, say, 5 percent of that value. But that wouldn't be enough to push the cladding temperature up 190 degrees Fahrenheit after the reactor shut down.

So, data from thermal-hydraulic experiments demonstrates that the zirconium-steam reaction is not negligible below 1900 degrees Fahrenheit, and that was one of the original premises on which Indian Point Unit 2 was licensed.

And that still holds true today, because what they had back then was the Baker-Just correlation. And if the Baker-Just correlation would calculate that the metal-water reaction is negligible

below 1,900 degrees Fahrenheit, so would a best-estimate metal-water correlation. What they use at Indian Point now is best estimate, and that's like Cathcart-Pawel.

So, this is a very serious problem. You have experimental data -- this was from an experiment that was conducted in the early 1980s, and you did nothing.

You have not examined that. You're still allowing these plants close to New York City to operate, and you've done nothing to address the problem that actual experimental data demonstrates that the metal-water reaction is not negligible below 1,900 degrees Fahrenheit. It's actually very substantial below 1,900 degrees Fahrenheit.

Another thing is, that information from those tests, the TH-1 tests, in 2005, NRC stated that it was actually reviewing that data from the TH-1 tests to determine its value for assessing the current generation of codes, that would be computer models such as TRAC-M, now renamed TRACE.

So in 2011 dollars, billions of dollars have been spent on LOCA research, and I think it's about time that NRC fixed the flaws in its LOCA evaluation models.

And currently, the heat transfer coefficients used in LOCA evaluation models are still not based on tests that were actually conducted with zirconium alloy bundles.

What you have are tests that were conducted with stainless steel bundles, and I will get back to it, but it's just an observation, it seems that NRC really does not like to conduct experiments with zirconium alloy bundles.

First, you have the metal-water reaction correlations. Those are based off of very tiny tube specimens instead of being based off of zirconium alloy bundles. And there is experimental data that shows that the reaction rates are different.

And then, when it comes to the heat transfer coefficients, it's the same thing. You don't have the zirconium alloy bundle. Instead, you use a stainless steel bundle, and stainless steel just does not oxidize. And the oxidation just doesn't generate heat like a zirconium alloy bundle does.

And yet, you have all this data, and you've done nothing to improve your computer models, and you're allowing Indian Point to operate in the face of all of this data that shows that there are major flaws in your models.

Now, next, I'm going to move on to something that I point out, I think this section starts on page 24 of the petition. It's regarding Entergy's best estimate ECCS evaluation calculations.

I want to point out that there is a report from the Nuclear Energy Agency Group of Experts, OECD, Nuclear Energy Agency, the title of the report is, "In-Vessel and Ex-Vessel Hydrogen Sources." I'm quoting from part one, "GAMA perspective statement on in-vessel hydrogen sources." This was published in 2001, ten years ago.

And I quote from that, "Until recently, the experimental data base on quenching phenomenon was rather scarce. The available zircaloy-steam oxidation correlations were not suitable to determine the increased hydrogen production in the few available tests, CORA, LOFT LP-FP-2."

Now, in the petition that Riverkeeper submitted, there is a lengthy discussion of the CORA tests and also the LOFT LP-FP-2 experiment. Now, those are -- they are large, integral experiments, well, especially LOFT LP-FP-2. That's a large integral experiment.

And you just get a different metal-water reaction rate than you do from your Baker-Just or

Cathcart-Pawel, which are based on experiments with tiny tube specimens, like one inch long, very small.

Anyway, because there were problems with these metal-water correlations for determining the increased hydrogen production in experiments like LOFT LP-FP-2, we argue that this indicates that the available zircaloy oxidation kinetics models, including best estimate models used at Indian Point are non-conservative for use in analyses that calculate the metal-water reaction rates that would occur in the event of a LOCA.

So, this OECD Nuclear Energy Agency paper was published in 2001. It explicitly states that there are problems with these correlations when they're applied to integral experiments.

And the NRC has done nothing, again, absolutely nothing. In fact, in 2002, Robert H. Leyse, submitted a petition. It was called PRM-50-76. And that petition addressed the fact that the Baker-Just and Cathcart-Pawel correlations are deficient because they were not developed to consider how heat transfer would affect zircaloy-steam reaction kinetics in the event of a LOCA.

And NRC denied that petition in 2005, and one of the reasons they used to justify -- one of

their statements was that there was no data available to substantiate Robert H. Leyse's claims.

Meanwhile, you know, there was this paper published in 2001, but apparently that didn't seem to matter, even though it was an OECD Nuclear Energy Agency paper.

Now, I just want to mention just to kind of examine in more detail what NRC said in its denial of PRM-50-76. It's -- now I'm quoting. This is NRC's statement.

"For the development of oxidation correlations, limited by oxygen diffusion into the metal where well-characterized isothermal -- " (that's holding the temperature of the specimen constant,) " --tests are more important than the complex thermal-hydraulics suggested by Robert H. Leyse."

"Robert H. Leyse's suggested use of complex thermal-hydraulic conditions would be counter-productive in reaction kinetics studies, because temperature control is required to develop a consistent set of data for correlation development."

"Isothermal tests allow this needed temperature control. It is more appropriate to apply the developed correlations to more prototypic transients, including complex thermal-hydraulic

conditions to verify that the proposed phenomena embodied in the correlations are indeed limiting."

"This was what was done by Westinghouse in WCAP 7665," (that's the FLECHT report), "and by Cathcart-Pawel in NUREG-17, and by the NRC in its technical safety analysis of PRM-50-76."

Now, I know I'm repeating myself, but again, here you had an integral test, and it -- like I said, published in 2001 in an OECD Nuclear Agency Paper, and it explicitly stated that the available zirconium alloy steam oxidation correlations were not suitable to determine the increased hydrogen production in the few available tests, including the LOFT LP-FP-2 experiment.

Now, I just want to touch upon Westinghouse's analysis of WCAP 7665. This would be the analysis of the four zircaloy bundle runs that were conducted in the FLECHT experiments. These are very important experiments.

The Appendix K to Part 50 is still based, for reflood heat transfer coefficients, that's based on the test with stainless steel fuel rods from WCAP 7665.

But anyway, back to the zirconium bundle runs, four of them, I point out that in the petition

that there was no -- well, I'm not sure if I mentioned it in the petition, but I've pointed it out, there was no metallurgical data from the locations of run 9573 that incurred runaway oxidation, because Westinghouse did not obtain such data.

So neither Westinghouse nor the NRC applied the Baker-Just correlation to metallurgical data from locations of run 9573 that incurred runaway oxidation.

And in its analyses, NRC did not apply the Cathcart-Pawel oxygen uptake and zirconium dioxide thickness equations to metallurgical data from those locations since, as I said, those locations were not measured.

And furthermore, it's reasonable to assume that, as in the CORA-2 and CORA-3 experiments in which local steam starvation conditions are postulated to have occurred, during FLECHT run 9573 the violent oxidation essentially consumed the available steam, so that the time-limited and local steam starvation conditions, those that cannot be detected in a post-test investigation, those would have occurred.

Now, those would have occurred at the locations where runaway oxidation did not occur. But as I pointed out, Westinghouse didn't take

measurements from the parts that actually incurred the runaway oxidation for that one bundle. They measured the parts that would have been steam-starved, and that is just not a legitimate verification of the adequacy of Baker-Just and Cathcart-Pawel for use in ECCS evaluation calculations.

And I think just the fact that NRC didn't really look into what was really going on, the fact that they weren't analyzing sections of that bundle that did incur runaway oxidation, and they were actually evaluating sections that would have most likely been steam starved.

Now, I want to point out that here we are talking about problems with metal-water reaction correlations for use in analyses that would predict the metal-water reaction rates that would occur in the event of a LOCA. But this was actually, this very same issue was raised about 40 years ago by Union of Concerned Scientists during the original licensing hearing for IP-2.

Dan Ford of UCS, for example, he pointed out that the Baker-Just correlation was, and I quote, "derived from experimental data that is completely outside of the context of nuclear systems," end of quote.

So that means, it's derived from single-rod separate effects tests, and he also pointed out that this metal-water reaction correlation had not been derived from integral, you know, large-scale integral tests.

And that's in the petition, in a section that starts at page 21. So that gives you some history, that this has been going on for quite a while, criticisms of this type of problem.

And I want to point out that Baker-Just, again, Baker-Just is not used at Indian Point now. What they presently use are best estimates, and that also involves the Cathcart-Pawel correlation.

Now, just to point out, Cathcart-Pawel, that correlation is based on data conducted with experiments in two different furnaces. And in one, the specimen was 18 inches long, but actually, only a small segment of that tube in close proximity to the thermocouple station served as the specimen. The other -- the full 18-inch length, part of that was to just hold it in place.

And then, in a different furnace, the specimen was about 1.2 inches long. So, you know, is it any wonder that you'd get these metal-water reaction correlations derived from such small tube

experiments, and then when you do a larger-scale test, like LOFT LP-FP-2, these correlations, -- they're not suitable for analyses.

And the increased hydrogen production that -- in a test like LOFT LP-FP-2 -- these correlations fall short of predicting that.

So, again, LOCA model evaluations, the models are flawed, and there's data out there that demonstrates that they're inadequate for use in analyses.

The metal-water reaction correlations are inadequate for use in analyses that predict the zircaloy oxidation rates that would occur in a LOCA, and yet, the NRC still does not do anything to fix the problem.

And now, I want to describe the type of accident that we're talking about in the petition. As I had mentioned before, Fukushima is what I would call a slow-moving accident, as opposed to this, which is a fast-moving accident.

Once there was a loss of coolant accident at either plant, within about 60 seconds, the peak cladding temperatures, they would start approaching temperatures up around 1,000 degrees Celsius, around 1,832 degrees Fahrenheit. That would happen within

about 60 seconds' time.

And what could happen at that point is that in a local area, the fuel rods could incur runaway oxidation, and then, within another 60 seconds or so, there would be a rapid temperature escalation, and it could get up to about 3300 degrees Fahrenheit, where the -- that's where zircaloy starts to melt, or zirconium alloy starts to melt. I believe the Indian Point plants actually have Zirlo in them, but that is also a zirconium alloy.

And so within a couple of minutes also there would be extensive core damage. And this is all depicted in Appendix F of the petition. There's a graph there that shows the progression of a serious accident and how this is very rapid.

And this is all based off of data like the LOFT LP-FP-2 experiment and also the CORA experiments, in which there was data that recorded these rapid temperature escalations, commencing at temperatures as low as 1,832 degrees Fahrenheit.

So just what happens, again, just in a little more detail, the loss of coolant accident, so the fuel is uncovered, and they're trying to pump water back into the core. There can be problems with pressurized water reactors, there can be steam

binding. So we're arguing that the models used at Indian Point for Unit 2 and 3, the predicted maximum temperatures of the fuel rods is 1,937 degrees Fahrenheit and 1,961 degrees Fahrenheit respectively. That's what's predicted.

But we're saying that as the fuel rods approach those temperatures, all taking place within under a minute, it could incur runaway oxidation. And then that is going to drive the temperature in the local area up to around 3,300 degrees Fahrenheit.

Meanwhile, as they're rising up, increasing rapidly in temperature, there are also other assembly components, like the control rods, they're going to be rapidly moving up behind them, maybe around 400 degrees Fahrenheit behind them.

And then when the stainless steel cladding or casing of the control rods, when that gets up to around 2,200 or so Fahrenheit, it can have eutectic reactions with Zircaloy, and that can cause liquefaction at those temperatures. This was observed on cameras in the CORA experiments.

And then you can also have a problem at that temperature, the inside of the control rod, which is silver, indium, and cadmium, that has melted at a lower temperature. But when it gets up to around

2,200 Fahrenheit, that can burst, and that can spray out onto zircaloy and also dissolve it rather rapidly, again, observed on cameras.

So we're talking about a great, substantial amount of damage that can occur within just a couple minutes' time. Again, this is in Appendix F of the petition.

So in under ten minutes' time, there would be a substantial amount of hydrogen generation from this, hydrogen from the zirconium alloy oxidation, hydrogen from the oxidation of stainless steel at higher temperatures, but primarily from the zirconium alloy. And there would also be a substantial amount of hydrogen in the containment building.

And now, I want to just point out and refer to an article -- or, not an article, it was an entry, a New York Times entry that Matt Wald wrote. It was on the Green, a blog about energy and the environment, and dated March 31st, 2011.

And there is a quote from the Indian Point spokesman, James F. Steets. So I just want to read from one sentence from Matt Wald's article.

"James F. Steets, a spokesman for Indian Point, said that Units 2 and 3 there each had two recombiners, and that one alone could eliminate all

the hydrogen in a major accident."

So I just want to point out that according to one paper, what Steets said is just simply incorrect. Now, again, this is just one paper that I have, but I want to read from that. It's Nuclear Engineering and Design, Number 230, published in 2004.

The title is, Studies on Innovative Hydrogen Recombiners as Safety Devices in the Containments of Light Water Reactors. And on page 49, I'm going to read from the abstract.

"In order to prevent the containment and other safety-relevant components from incurring serious damage caused by the detonation of the hydrogen-air mixture generated during a severe accident in light-water reactors, passive autocatalytic recombiners are used for hydrogen removal.

"These devices make use of the fact that hydrogen and oxygen react exothermically on catalytic surfaces, generating steam and heat. "

"Experimental investigations at several research facilities indicate that existing passive autocatalytic recombiner systems bear the risk of igniting the gaseous mixture due to an overheating of the catalyst elements, caused by a strong reaction

heat generation."

And then, the conclusion of the paper, on page 59, I want to read from that, and this more addresses the issue that was addressed by Mr. Steets. Actually, I'm not sure if I stated the title of Matt Wald's blog entry. That's "U.S. Drops Nuclear Rule Meant to Avert Hydrogen Explosions."

So this basically, this conclusion addresses more what Mr. Steets had said. And I quote, "Even if recombiners could be made safe against unintended ignitions, these devices cannot solve the hydrogen problem for severe accidents. Conversion rates of present systems are not sufficient for massive hydrogen release, and hydrogen transport to the recombiners cannot be assured in a sufficient way. The combination of passive autocatalytic recombiners with other existing concepts for hydrogen mitigation, for example, inerting or diluting, seems to be advisable, even if these concepts also have limitations. "

"One example is the reinforcement of passive autocatalytic recombiners by means of catalytic coated thermal insulation elements, as proposed in the THINCAT project.

"The introduction of igniters, as

discussed in the past, still seems to be very questionable, as the prediction of hydrogen distribution and combustion in the containment is at present not reliable enough to ensure safe application of this measure."

That's the end of that quote.

And now, I want to move on to talking about stainless steel thermal-hydraulic experiments, and discuss them, because so far, I've spoken about the metal-water reaction correlations, and now I want to talk about heat transfer experiments that -- in the petition, starting on page 104, there is a discussion of a stainless steel test that's commonly included as a benchmark for the validation metrics of several computer codes, and this is the FLECHT-31504 test.

And just to briefly describe that test, FLECHT-SEASET test 31504 had a rod peak power, it was 0.7 kilowatts per foot, a reflood rate of approximately one inch per second, and the PCT at the onset of reflood was about 1,585 degrees Fahrenheit, and the overall peak cladding temperature was about 2,100 degrees Fahrenheit. So, there was an increase of 516 degrees Fahrenheit in that test.

But, I point out, in the petition, that if the FLECHT-SEASET test 31504 had been conducted with a

zirconium alloy instead of with a stainless steel bundle, the test results would have been very different, and that with high probability, the maximum temperatures of the bundle, if it were a counterpart test, were done with a zirconium alloy bundle, that they would have exceeded the 2,200 degrees Fahrenheit peak cladding temperature limit, and the bundle would have, with high probability, incurred runaway oxidation.

And it's very unfortunate that NRC has actually never, never conducted a counterpart thermal-hydraulic experiment with a zirconium alloy multi-rod bundle.

This FLECHT-SEASET 31504 test is obviously very important. It's used as, like I said, a benchmark for computer codes. It's discussed at length in a recent study -- or program description of the rod bundle heat transfer facility program that's being conducted right now at Penn State. But there's never been a counterpart test conducted with a zirconium alloy bundle.

And I know I'm not supposed to ask questions at the meeting, but I mean, nonetheless, I mean, can you tell me why NRC has never conducted such a test? And if you don't want to answer that, could

you at least think about that question?

It seems pretty outrageous, frankly. Here you have computer codes, and the heat transfer coefficients that are used in them are based off of tests conducted with stainless steel bundles, and then you've never conducted counterpart tests with the same test parameters with zirconium alloy bundles.

The TH-1 tests that were conducted in the early `80s, actually, one of the parameters or conditions for those tests was that they didn't want the peak cladding temperatures in those tests to exceed 1,900 degrees Fahrenheit, and they said that was actually for safety reasons.

And I can sympathize with people who conduct tests, but the point is, the real limit is 2,200 degrees Fahrenheit, so NRC needs to conduct tests with zirconium alloy bundles with different parameters, in which the peak cladding temperature would go up to at least 2,200 degrees Fahrenheit.

But the real problem is, if you do tests like that, you're not going to have a peak cladding temperature that goes up to 2,200. You're going to have runaway, and it's going to just go up to the point where you have rapid oxidation, and you're going to have melting of the rods, etcetera.

So, again, I think it's time for the NRC, this experiment is obviously very important, the FLECHT-SEASET 31504, to conduct a counterpart test with a zirconium alloy bundle.

And I think that could be part of the technical analysis of PRM-50-93. For example, NEI has pointed out that, they say, oh, 50-93, it depends too much on the results of severe fuel damage experiments, and the results would be entirely different if that petition were based off of the results of thermal-hydraulic experiments.

Well, the real problem I had writing that is that there really is no data from thermal-hydraulic experiments conducted with zirconium alloy bundles, because you haven't done such experiments.

And in that petition, I discuss the TH-1 tests rather extensively. I do what I can to use that very limited data to draw conclusions. I state that if the initial peak cladding temperatures in those tests had been higher, that they would have incurred runaway oxidation.

But the point is, I was very limited, because there's such a limited amount of data. But if you really want to do an analysis of PRM-50-93, what you do is you just conduct a counterpart test to

FLECHT-SEASET 31504, and you see what happens. You do that with a zircaloy bundle.

And I just want to point out that here I've been talking about tests, you know, TH-1 tests from the early 1980s, and all of these FLECHT tests. Those were done about 40 years ago.

Presently, as I discuss in the petition, there is the test facility, it's the RBHT test facility. That's at Penn State. And they're conducting thermal-hydraulic experiments with Inconel 600 rods.

And I just want to read some information that's also in the petition. Let me tell you what page that's starting on. That's starting on page 116. That's where I describe the current heat transfer experiment program that uses Inconel 600 bundles.

And according to NRC's Advisory Committee on Reactor Safeguards, this is a statement made in 2010, the rod bundle heat transfer facility program was developed to address issues related to emergency core cooling, including phenomena that would affect peak cladding temperatures.

Now, as the test plan for the RBHT, the test plan points out, it states, "oxidation is not simulated in the RBHT facility tests since the

cladding is Inconel," and as the test plan also acknowledges, "Inconel will not oxidize, while zircaloy will oxidize and create a secondary heat source at very high PCTs. Zircaloy reaction can be significant at high temperatures."

So here you have ACRS, they're talking about this program, and it's supposed to be addressing issues relating to emergency core cooling, including phenomena that would affect PCTs, and yet, ACRS can't put two and two together? You know, that these tests are conducted with Inconel 600, and as the test plan explicitly states there will not be the additional heat generated from the zirconium alloy that will affect the PCTs? I don't know what to say. Actually, I don't think I'll comment on that. I think it speaks for itself.

So, currently, here we have the heat transfer coefficients used in LOCA evaluation models, they're still not based on data from thermal-hydraulic experiments conducted with zirconium alloy bundles, and there are currently no plans underway to make them based on such data.

And I want to really point out that this program, the rod bundle heat transfer facility program, it's not just like these tests are worthless.

These tests do not help public safety. It's actually, these tests decrease public safety because the results of these experiments conducted with stainless steel or Inconel 600 multi-rod bundles, they're going to lead the interpreters of tests to false conclusions.

For example, a test conducted with a stainless steel or Inconel 600 multi-rod bundle heated up to peak cladding temperatures between 1,832 degrees Fahrenheit and 2,200 degrees Fahrenheit, that is not going to incur runaway oxidation. It's not going to have much oxidation.

However, a zirconium alloy multi-rod bundle heated up to those same temperatures, between 1,832 and 2,200 degrees Fahrenheit, with high probability, that's going to incur runaway oxidation.

So what we have going on, and you can read this, it's in a ACRS meeting on heat transfer from last year, I believe it was dated October 18th, they're discussing results from this facility, the rod bundle heat transfer facility.

And they're saying that, well, because of what we see, we can actually have more power uprates. We can have more power uprates, because we see that our models are overly conservative. But it's a pure fantasy. There's no foundation.

So, like I said, all you need to do, you conduct a counterpart test to FLECHT-SEASET 31504. You do that with a zirconium alloy bundle and you'll see -- you'll find out what's wrong with your models.

And I think you should just stop funding programs like the RBHT program where they're not using zirconium alloy bundles. And I just want to point out that you're a regulator, and your job is to protect public and plant workers' safety. And you need to have realistic experiments.

And this is pretty much where I want to conclude. I just want to point out that what I have presented are generic issues, and what Phillip Musegaas focused on I think are extremely important issues that demonstrate that this petition, because of where Indian Point is located, that it also addresses plant-specific issues.

And I think it's really time that NRC has Entergy lower the licensing basis peak cladding temperatures of Indian Point Units 2 and 3 and also has Entergy demonstrate that for Indian Point Units 2 and 3, that the ECCS systems would effectively quench the fuel cladding in the event of LOCAs.

Thank you.

MR. MUSEGAAS: Thank you, Mark.

MR. BOSKA: This is John Boska from the NRC. I did want to make one comment before we continue.

MR. MUSEGAAS: Sure.

MR. BOSKA: And that is, Mr. Leyse has submitted a rulemaking petition for our rule 10 CFR 50.46 on LOCA analysis, and his rule-making petition is PRM-50-93. The NRC has accepted that petition for review, and we are proceeding with the review of that petition. So let me just make that one point.

MR. DUDLEY: On that basis, I think the NRC would also like to note that we are not ignoring the issues that you are raising in this phone call. We are, in fact, reviewing them as part of PRM-50-93 and 50-95, which is actively ongoing at the present time. That was Richard Dudley.

MR. M. LEYSE: Okay. Mark Leyse. I just want to -- yes, I really appreciate that, and I am aware of the fact that you're reviewing that, and I appreciate that.

I think just more -- because it's been so long that this problem has persisted, I think that's more of the reason why I made some of the statements that I made. But I really do appreciate that.

And as I had said, I really would advise

conducting counterpart tests to FLECHT-SEASET 31504. I think you would really learn a lot of information from that. But that's the last thing I want to say. Thank you.

MR. MUSEGAAS: Thanks again, Mark. This is Phillip Musegaas. I think that -- you know, I'll hold any further comments. I'd like to hear what the PRB has to say. Thank you.

MR. BOSKA: Thank you.

MR. R. LEYSE: Okay, this is Robert Leyse. I just came back online. If I could just have a second to talk about PRM-50-93 and the timing of its review?

MR. M. LEYSE: Mark Leyse speaking. Of course, you can.

MR. R. LEYSE: Well, I was asking permission from the guys who run the meeting.

Anyway, the fact is going back to the ACRS meeting of last October, October of 2010, it was at that meeting that I found out that NRC had changed its deadline for reviewing PRM-50-93. There's a number that I don't have available, but that review was to be completed by September of 2010.

So, Mark Leyse submitted an enforcement action, and that was thrown in as a petition, and

suddenly, there was no longer any deadline. It's all in the record.

But I never found out that the September 2010 deadline had gone away until Bajorek mentioned it at the October 2010 ACRS meeting. And at that point, I wasn't in a position to be able to talk to the ACRS.

So, I wouldn't want anybody to make too much of the fact that PRM-50-93 is under review. I mean, apparently there was review done before September of 2010, but I don't see it anywhere.

End of comment.

MR. BOSKA: All right, Mr. Leyse. Thank you for your comment.

MR. BROWN: This is Fred Brown. Our thanks to Riverkeeper for making the presentation. Let me now ask the staff at NRC headquarters, do we have any questions for Mr. Musegaas or Mr. Leyse about the 2.206 petition?

No questions in headquarters?

The region, do you have questions?

MR. BICKETT: This is Brice Bickett from NRC Region 1. None from Region 1.

MR. BROWN: Very good. Thank you.

At this time, does the licensee's representative or representatives have any questions?

MR. WALPOLE: No comments or questions, thank you.

MR. BROWN: All right. Thank you all. Before I conclude the meeting, members of the public may ask questions about the NRC's process for 2.206 petitions.

However, as stated at the opening, the purpose of this meeting is to provide an opportunity for the public to question or examine -- is NOT an opportunity for the public to question or examine the PRB regarding the merit of the petition request.

Are there any questions?

(Pause.)

Hearing none, I want to thank the petitioners for their time and energy to make their presentation to the Petition Review Board and to provide the staff with clarifying information as to the petition that you have submitted.

Before we close -- we don't have a court reporter online?

COURT REPORTER: Yes, we do. I'm online. I do have two questions.

MR. BROWN: Thank you, please.

COURT REPORTER: The first question is if the people online from Entergy could identify

themselves. I didn't catch them the first time they identified themselves.

MR. WALPOLE: Sure, it's Bob Walpole from Indian Point.

COURT REPORTER: W-A-L-P-O-L-E?

MR. WALPOLE: That is correct.

COURT REPORTER: And there was one other person from Entergy.

MR. WALPOLE: His name was Adi Irani, I-R-A-N-I, first name A-D-I, Alpha Delta Indigo.

COURT REPORTER: Thank you very much. The second question is if there is someone from the NRC staff who I could call if I have questions. I don't anticipate any now, but as I go over my notes, I might. Is there someone who can give me their phone number?

MR. BOSKA: You should call John Boska at 301-415-2901.

COURT REPORTER: 2901. Thank you very much. It will be a call within the hour or none at all.

MR. BOSKA: Thank you.

MR. BROWN: Thank you all.

(Whereupon, the above-entitled matter was concluded at 2:15 p.m.)