UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

January 30, 2017

MEMORANDUM TO: ACRS Members

FROM: Christopher L. Brown, Senior Staff Engineer **/RA/**
Technical Support Branch
Advisory Committee on Reactor Safeguards

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS METALLURGY AND REACTOR FUELS SUBCOMMITTEE MEETING ON NOVEMBER 16, 2016, IN ROCKVILLE, MARYLAND

The minutes for the subject meeting were certified on December 30, 2016. Along with the transcripts and presentation materials, this is the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc with Attachment: A. Veil
M. Banks

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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WASHINGTON, DC 20555 - 0001

MEMORANDUM TO: Christopher Brown, Senior Staff Engineer
Technical Support Branch
Advisory Committee on Reactor Safeguards

FROM: Matthew Sunseri, Chairman
Metallurgy & Reactor Fuels Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS METALLURGY AND REACTOR FUELS SUBCOMMITTEE MEETING ON NOVEMBER 16, 2016, IN ROCKVILLE, MARYLAND

I hereby certify, to the best of my knowledge and belief, that the minutes of the subject meeting on November 16, 2016, are an accurate record of the proceedings for that meeting.

/RA/	**December 30, 2016**
Matthew Sunseri, Chairman Metallurgy & Reactor Fuels Subcommittee	Dated

Certified on: December 30, 2016
Certified by: Matthew Sunseri

Advisory Committee on Reactor Safeguards
Meeting of the Subcommittee on Metallurgy & Reactor Fuels
Rockville, MD

Recent Operating Experience with Baffle Bolt Degradation
November 16, 2016

The ACRS Metallurgy, and Reactor Fuels Subcommittee held a meeting on November 16, 2016, in T2B1, 11545 Rockville Pike, Rockville, MD. The meeting convened at 8:30 a.m. and adjourned at 11:55 a.m.

ATTENDEES

ACRS Members/Staff

Matt Sunseri, Chairman for this SC
Pete Riccardella, Member
Dana Powers, Member
Walt Kirchner, Member
Gordon Skillman, Member
John Stetkar, Member
Ron Ballinger, Member

Christopher Brown, Designated Federal Official

Jeffrey Pohler, NRR
William Burton, NRR
CJ Fong, NRR
Allen Hiser, NRR
Dian Curran, Riverkeeper
Carol Nove, RES

SUMMARY

The Subcommittee received an information briefing from the staff on baffle bolt degradation and recent events. Industry discussed recent actions taking on the issue. Baffle bolts affix a set of baffle plates to the reactor core barrel via a series of baffle former plates. The baffle plates provide the radial geometry of the reactor core.. Flow holes between the baffle former plates and the core barrel are designed to allow coolant flow in an upward direction (up-flow) or a downward direction (down-flow).

The meeting transcript and slides are attached and contains an accurate description of each matter discussed during the meeting.

DISCUSSION ISSUES	
Issue	Reference Pages in Transcript
1. Discussion on the design, functions and materials of PWR internals, baffle-former assembly, and baffle-former bolts. The following points /questions were raised: 1) failure history, 2) differences material type 347 and material type 316 bolt failure, 3) geometry of the bolts.	8 - 17
2. Discussion on the potential consequences of baffle former bolt degradation. Function of baffle-edge bolts is to ensure baffle plate integrity and reduce baffle jetting. The following points /questions were raised: 1) failure of the edge bolts, and 2) stuck fuel assembly, distinguishing between up-flow and down-flow plants.	17 - 22
3. Discussions on history of baffle bolt degradation.	22 - 23
4. Discussion on US plant inspections. Two two-loop down-flow plants. Question was raised on B&W plants (up-flow).	23 - 24
5. Discussions on MRP-227-A inspection requirements for baffle-former Bolts. D.C Cook, unit 2 was discussed.	24 - 28
6. Discussion of Inspections under MRP-227-A (2011-2015).	28 - 29
7. Review of Indian Point, Unit 2, Salem Unit 1, D.C. Cook, Unit 2 degradation and corrective actions. The following points /questions were raised: 1) reasoning for not using L grade stainless steel, 2) fracture morphology, 3) damage to a fuel assembly, and 4) more on baffle jetting.	30 - 48
8. Discussion on factors influencing baffle-former bolt degradation, such as, neutron fluence, stress, down-flow versus up-flow plants, and bolt failure clustering. The following points /questions were raised: 1) manufacturing issues, 2) lubrication, types of bolt threads, 3) plugging of core barrel holes, and 4) clustering degradation plant comparison.	48 - 64
9. Discussion on bolt inspection, replacement, evaluation and re-inspection intervals.	65 - 70
10. NRC response to baffle bolt degradation. Regional inspections at plants. Questions was raised concerning a specific inspection procedure for baffle-former bolts.	70 - 75
11. Discussion on the risk informed evaluation conducted by the staff. Risk criteria of $CDF<1x10^{-3}$ and $LERF<1x10^{-4}$. Question was raised on the CDF and a LERF analysis.	75 - 85
12. Staff summary and future activities	85 – 87
13. Industry presented perspectives on U.S. plants operating experience and MRP 227 guidance. Questions was raised on an EDF (French)	88 - 105

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards
Metallurgy and Reactor Fuels Subcommittee

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Wednesday, November 16, 2016

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

SUBCOMMITTEE ON METALLURGY AND REACTOR FUELS

+ + + + +

WEDNESDAY

NOVEMBER 16, 2016

+ + + + +

ROCKVILLE, MARYLAND

+ + + + +

The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B1, 11545 Rockville Pike, at 8:30 a.m., Matthew Sunseri, Chairman, presiding.

COMMITTEE MEMBERS:

MATTHEW W. SUNSERI, Chairman

RONALD G. BALLINGER, Member

WALTER L. KIRCHNER, Member

DANA A. POWERS, Member

PETER RICCARDELLA, Member

GORDON R. SKILLMAN, Member

JOHN W. STETKAR, Member

DESIGNATED FEDERAL OFFICIAL:

CHRIS BROWN

ALSO PRESENT:

PAUL BESSETTE, Morgan Lewis

WILLIAM BURTON, NRR

GANESH CHERUVENKI, NRR

DIANE CURRAN, Riverkeeper

DAVID DIJAMCO, NRR

C.J. FONG, NRR

ALLEN HISER, NRR

MATTHEW HISER, RES

GREGORY KOLCUM, R-IV

MARVIN LEWIS (present via telephone)

SIVA LINGAM, NRR

HEATHER MALIKOWSKI, Exelon

CAROL NOVE, RES

JEFFREY POEHLER, NRR

MARY JANE ROSS-LEE, NRR

BERNIE RUDELL, Exelon

DAVID RUDLAND, NRR

BALWANT SINGAL, NRR

DONG WEAVER, Westinghouse

BRYAN WILSON, Westinghouse

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P R O C E E D I N G S

8:31 a.m.

CHAIRMAN SUNSERI: The meeting is now called to order. This is a meeting of the Metallurgy and Reactor Fuel Subcommittee, the Advisory Committee on Reactor Safeguards.

I am Matt Sunseri, chairman for this subcommittee.

ACRS members in attendance today are Ron Ballinger, Pete Riccardella, Dick Skillman, Dana Powers, John Stetkar, Walt Kirchner.

Christopher Brown is the Designated Federal Official for this meeting.

The purpose of today's meeting is for the subcommittee to receive a briefing from the NRC staff and industry regarding recent operating experience with baffle-former bolt degradation. In particular, discussions on the design, functions and materials of PWR internals, baffle-former assembly, consequences of baffle-former bolt degradation, history of baffle-former bolt degradation, factors influencing baffle-former bolt degradation, bolt inspection replacement, root cause analysis results and industry response.

The rules for participation in today's

meeting were announced in the Federal Register on November 1, 2016. The meeting was announced as an open/closed to public. A portion of this meeting will be closed in order, may be closed in order to discuss and protect information designated as proprietary pursuant to 5 U.S.C. 552b(c)(4).

No requests for making a statement to the subcommittee has been received from the public.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register notice. Therefore, we request that participants in this meeting use the microphones located throughout the meeting room when addressing the subcommittee. Participants should first identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

We have one bridge line established for interested members of the public to listen in. The bridge number and password were published in the agenda posted on the NRC public website.

To minimize disturbance, this public line will be kept in a listen-in only mode. The public will have the opportunity to make a statement or provide comments at a designated time

towards the end of the meeting.

I request that meeting attendees and participants now silence their cell phones and other electronic devices.

Dr. Riccardella has co-authored a paper on the subject matter that presents a methodology for evaluating the probability of baffle-former bolt cracking for pressurized water reactors. This methodology recently was used as part of an EPRI MRP program to address new industry findings. Dr. Riccardella will not participate in manners related to technical areas of his past contributions.

I know invite M.J. Ross-Lee, Deputy Director of Engineering in NRR to introduce the presenters and start the briefing. M.J.

MS. ROSS-LEE: Good morning. So yes, I am M.J. Ross-Lee. I'm the current Deputy Director of Division of Engineering, Office of Nuclear Reactor Regulation. We are here to present to the subcommittee. I think we'll touch on all the parts as previously introduced, talk about baffle-former bolts, what they are, what they do, some past operating experience, current operating experience, what we've done, our path forward, and what we plan to do with those.

Today's presenter is a member of my staff, Jeff Poehler. He is a materials engineers and he will be taking us through the presentation today.

In addition, beside me is Dave Rudland who is the current Branch Chief of that branch, so hopefully between us we'll be able to answer any questions you might have on our presentation. And because I know we have a number of slides, I'm going to turn it over now to Jeff and let him get started.

MR. POEHLER: Thank you, M.J. First, I just wanted to note the audience handouts, the titles came out a little dark at the top of the slide, so you might want to note on there what the title is for later reference, just so you don't confuse plants.

So I'm going to be talking about recent operating experience with baffle-former bolt degradation. What we're going to cover in this presentation and we already did introduction. First, we're going to cover design and function and materials of PWR internals, the baffle-former assembly and baffle-former bolts. We're going to talk about some potential consequences of baffle-

former bolt degradation. We're going to talk about the history of baffle-former bolt degradation, basically, the operating experience. We're going to talk about some of the factors that influence this type of degradation. We're going to talk about how the bolts are inspected and replaced, how baffle-former bolt degradation is evaluated, and how the NRC is responding to the recent operating experience. And then we're going to talk about the NRC's future activities.

So here's a couple of figures. You're going to see these again, probably. The one on the left is just a general overview of PWR internals. This is a Westinghouse-style PWR internals. The blue structure is the baffle-former assembly and it sits within the core-barrel assembly which is sort of the large cylindrical structure that is the largest component of the internals.

On the right is sort of a more detailed view, what you would see looking at the inside of the baffle-former assembly. You have a number of plates and they are attached with bolts to these horizontal plates which are called formers. And you can see that on the cross section there, the edge of the formers.

Basically, the function of the baffle-former assembly is to direct the coolant flow through the core and also provide some lateral support to the core during a seismic event or loss of coolant accident. And basically, these plates can form very closely to the outline of the core.

Here's a view looking down into an actual Westinghouse PWR which you would see when the reactor is defueled. You can see some of the openings in this. At the top would be the core barrel where you can see these two openings for the inlet, either inlet or outlet flows.

So here on the left of this figure, this is basically a plant view looking down of one-eighth of the baffle-former assembly. And the various different bolt locations are circled. Here, these are the baffle-former bolts and then you also have edge bolts which go in and connect the corners of the plates and those sort of go in here. And then on the outside -- this is the core barrel and you have barrel-former bolts which attach the formers to the core barrel.

On the right here, this is what you would see if you were inside the core looking out towards the baffle plates. This is what the

baffle-former bolt heads look like. They're inset. They're counter sunk into the plate and there is -- this is a locking bar which is welded to the baffle plate on either side and that keeps the bolt from backing out. And also, if the bolt were to fracture, it would keep the bolt head from becoming a loose part, presuming this locking bar was still in place.

MEMBER SKILLMAN: Jeffrey, let me ask this question, please. In this Figure 5, you show the baffle-former bolts, those are from the inside out.

MR. POEHLER: Right.

MEMBER SKILLMAN: You also show the core barrel to former bolt. Those are from the outside in.

MR. POEHLER: Correct.

MEMBER SKILLMAN: What is the failure history of the latter, of the core barrel to former bolt?

MR. POEHLER: Barrel-former bolts, basically, there have not been any failures of those. There might have been maybe one or two.

MEMBER SKILLMAN: But none or virtually none.

MR. POEHLER: Virtually none.

MEMBER SKILLMAN: So all the failures are on the inside out?

MR. POEHLER: Right.

MEMBER SKILLMAN: Is there a segregation of the shorter versus the longer shanks for the failures?

MR. POEHLER: You mean among the baffle-former bolts?

MEMBER SKILLMAN: Yes.

MR. POEHLER: Well, the shorter shanks have generally higher stresses. That is a factor. I'm going to talk about that a little bit later.

MEMBER SKILLMAN: I'd be curious if there is a binning of the failures of the short versus long.

MR. POEHLER: Yes, I'll talk about that a little later, but yes, that is a factor and different bolt materials tend to use a different length shank, so that is also a factor.

MEMBER SKILLMAN: Thank you.

MR. RUDELL: If I may also, one thing I wanted to point out on this particular photo because I don't think we have it, is that some baffle-former bolts have an internal hex design as

this one does with a locking bar or locking tab sometimes it's called, and welds associated with that. Some of them have an external hex with a locking bar and some of them have an internal hex with a locking washer. So between the different varieties of baffle-former bolts, there may be a half a dozen different designs. And of course, you've got to tailor, you need to tailor your examination for that particular design.

And as you can note here, they weren't made to get ultrasonic examination easily because the original requirements were not to do in-service inspection ultrasonic examination of these bolts, but industry has worked hard to develop and fairly recently demonstrated techniques to examine these bolts. But it is challenging with regards to bolt geography to do an ultrasonic examination with bolts.

MR. POEHLER: And I just wanted to note this is what -- this is like the type of bolt style that's used in the Westinghouse four-loop plant. And the bolts are about 5/8ths inch diameter shank and about 2 inches long, so they're about the size of your thumb. The heads are a little bigger.

So what are the materials used in

baffle-former bolts? In Westinghouse plants, type 347 stainless steel is used in the older Westinghouse plants. The bolt design for the 347 bolts has a sharper head-to-shank radius and shorter shank as we mentioned than in the Type 316 cold worked bolts. Type 316 cold worked stainless steel is used in newer generation Westinghouse plants and all replacement bolts that are installed to replace degraded bolts.

In other NSSS designs like Babcock & Wilcox design PWRs use Type 304 baffle-former bolts and combustion engineering plants use Type 316 annealed material. There are only two combustion engineering plants that have bolts. Most of them have a welded core shroud.

MEMBER KIRCHNER: Is this a good time to ask just from a technical basis or metallurgical basis, is there any reason why 347, have you seen any technical reasons why 347 is failing versus 316, just based on materials?

MR. POEHLER: Just as far as metallurgically-wise, I don't know that we have a good explanation of why 347 is failing more. But it is -- the operating experience clearly shows that it's more susceptible.

MR. RUDELL: Our international experience early on was with 316 bolts in the EDF plants and so I wouldn't necessarily conclude that 347 is significantly inferior material to 316 because we've seen quite a few 316 bolts fail, mostly in international units in EDF.

MEMBER RICCARDELLA: Isn't there test data and doesn't test data show difference in the two?

MR. WILSON: I don't believe there is very much test data that shows say a clear difference between the two. And I think that's the issues we run into. There's enough design differences between the bolts, say we don't have a real good one-to-one comparison across the board to make a conclusion on relative susceptibility based purely on material.

MEMBER RICCARDELLA: And then what about the cold worked versus annealed on the 316? Is there any evidence of one being better than the other?

MR. WILSON: I am not aware of any personally.

MEMBER RICCARDELLA: Cold worked would presumed to be higher strength?

MR. WILSON: Right, yes. Cold worked is. Again, as far as OE is concerned because you start out with different pre-loads, with one design versus another, it's again, you're fighting I'd say some design differences to make -- and it's making it difficult to make any material judgment between the two.

MEMBER RICCARDELLA: Between the three really.

MR. WILSON: What's that? Between the three, right, exactly right.

MEMBER RICCARDELLA: Thank you.

MR. POEHLER: The geometry of the type 347 bolts is with a shorter shank and sharper radius creates higher stresses. So it may be a function where the geometry that was used with the 347 bolts, but it's basically operating experience is showing they're more susceptible.

CHAIRMAN SUNSERI: Jeff, you said that the older Westinghouse plants had the 347 and the newer had the 316. Do we know what prompted that change? It must have come before the experience with the degradation?

MR. POEHLER: Yes. I'm not sure. I would have to defer to Westinghouse to answer that.

MR. WILSON: This is Bryan Wilson from Westinghouse. I can't answer directly why the change was made at the time. There was a change in the manufacturing processes that were intended to be used. I don't know -- the difference in the material follows that change in the manufacturing process very well. So it may have been an efficiency gain at the time, you know, given no other evidence of material differences.

CHAIRMAN SUNSERI: Okay. Thank you.

MEMBER RICCARDELLA: Is the cracking pretty much exclusively in the fillet radius or have we had any evidence of cracking in the thread regions?

MR. WILSON: There has been for some plants, yes, there's been indications either at the head-to-shank transition or the first thread.

MEMBER RICCARDELLA: Thank you.

MR. RUDELL: Bernie Rudell. It's overwhelmingly been at the head to shank, that's correct.

MEMBER RICCARDELLA: Thank you.

MR. POEHLER: So now I'm going to talk about some of the potential consequences of baffle-former bolt degradation. One of those is potential

movement or deflection of baffle plates. If you have large numbers of degraded bolts, that can allow detachment or deflection of the baffle bolts mainly during a LOCA or seismic event. And if that happens, the plates could impact on peripheral fuel assemblies and potentially cause fuel grid crush and localized fuel cladding damage.

Some plants also have control rods in peripheral locations which if the plate impact and the fuel assembly damage was severe enough, it could jeopardize capability to insert those peripheral rods. One mitigating factor is you have baffle-edge bolts which if they're intact they would help retain those plates and keep even with a lot of broken baffle-former bolts, it would help retain the plates from the moving. And we haven't seen any -- very little degradation of baffle-edge bolts I'll say.

Also, if you do get localized damage to peripheral fuel assemblies, you can perform a coolable geometry evaluation to show that core coolability would still be maintained.

Another consequence is baffle jetting which is basically flow leakage through the gaps between adjacent plates. You have baffle edge

bolts which basically help hold the corners of those plates together and that helps to prevent baffle jetting.

You can get -- but if you do get baffle jetting, basically the flow leakage out between the corners causes flow and just vibration of fuel pins resulting in localized -- which can result in localized fuel cladding damage. In some cases reaching the cladding. This can be detected by reactor coolant activity monitoring which can detect increases in flow and activity that can be indicative of fuel damage.

MEMBER KIRCHNER: Jeff, did I understand you correctly to say that you have seen little baffle edge bolt damage versus the other bolts?

MR. POEHLER: Yes. They have seen very little, but I will be touching on that later that one plant has found a few degraded edge bolts.

MEMBER KIRCHNER: Again, I'm searching for is there a technical reason why the edge bolts wouldn't fail at the same probability or statistically at the same rate as the other baffle bolts?

MR. POEHLER: I can't really speak to

that. There may be differences in stress.

MR. WILSON: I can help you out there, Jeff. There are differences in stress on those edge bolts. The edge bolts are generally not very highly loaded when all the baffle-former bolts are in plate. As you start getting degradation, significant degradation, you start to shift that load and create additional loads on those edge bolts.

MR. RUDELL: There is also less gamma heating on the edge bolts, right?

MR. POEHLER: They are influenced less by the gamma heating. I wouldn't say there's less gamma heat --

MEMBER SKILLMAN: But less dose?

MR. POEHLER: No, I wouldn't say that. Same dose. Another consequence is loose parts. Bolt heads and locking bars can become loose parts as the bolts completely fracture.

The clearances between the baffle plates and fuel assemblies are very small, probably in the order of that, which would tend to prevent -- the bolt heads really can't escape unless the reactor is de-fueled, but they can cause spreading of fuel assemblies because they're bearing right on

the fuel assemblies and they're caught in there.

MEMBER SKILLMAN: Jeffrey, has there been any experience where a peripheral fuel assembly has been blocked from removal because of the --

MR. POEHLER: There's parts wedged in there?

MEMBER SKILLMAN: No, because of the head of the bolt backing out and the fuel assembly not being able to slide freely out of that cell.

MR. POEHLER: Not that I'm aware of.

MR. WILSON: No, we have not had them experience that.

MR. POEHLER: But because these bolt heads are relatively small, if you had a few failed bolt heads, it's unlikely that the -- we don't think the loose part monitors would pick that up. If you had maybe a large number of loose heads, you might.

Baffle plates are unlikely to detatch during normal operation, but if they did, the potential for travel of the plate is limited by the type clearances and the large size of the plates.

Okay, now I'm going to go into the history of the operating experience or baffle bolt

degradation. So the early history, it was first identified in the late '80s in European plants, in the French 900 megawatts CPO plants, that should be a capital M, not a small M in front of the W.

Six plants found between about 1 percent and 11 percent of the bolts degraded and the French plants -- the Belgian plants --

MEMBER SKILLMAN: Wait a minute. Not so fast, not so fast. I understand that those CPO plants all are downflow plants.

MR. POEHLER: They were originally downflow, but they did convert to upflow in the early '90s.

MEMBER SKILLMAN: Hold on. The experience that you are pointing to is experienced during the time those were downflow plants. Is that accurate?

MR. POEHLER: Some of the inspections were during the '90s after they converted.

MEMBER SKILLMAN: I don't feel like I'm getting a good answer to my question. My understanding is only the CPO plants had a significant number of baffle-bolt failures.

MR. POEHLER: Right.

MEMBER SKILLMAN: The upflow plants did

not, the CPY plants.

MR. POEHLER: That's correct.

MS. MALIKOWSKI: That's correct. This is Heather Malikowski. We have a slide on the EDF experience with a little more detail.

MEMBER SKILLMAN: So as you go through the rest of these slides, I'd like to know what is upflow and what is downflow.

MR. POEHLER: We will be covering that, yes.

MEMBER SKILLMAN: Thank you.

MR. POEHLER: So in the Belgian plant the one three-loop Framatome 900-megawatt design is basically very similar to the CPO design. Performed five examinations between '91 and 2014. They found a total of 74 bolts degraded or uninterpretable.

Three other plants performed one ultrasonic examination each, finding just a handful of degraded bolts. So the mechanism for this degradation was attributed to irradiation-assisted stress corrosion cracking, or IASCC.

In 1998, the NRC issued Information Notice 98-11 to alert U.S. plant licensees to this issue. And then the U.S. industry kicked off a

program which included pilot inspections of baffle-former bolts at several plants.

So I am going to talk about the pilot plant inspections that were done in the U.S. There were two two-loop downflow plants. These were Westinghouse designed plants, the Type 347 bolts. And those were ultrasonically examined in the late '90s. Those plants found in the neighborhood of seven to ten percent of the bolts in each site, in each unit were degraded. So it came out to around maybe 50 some bolts per reactor. They replaced degraded bolts. One plant replaced a number of additional non-degraded bolts for additional margin.

One of the plants, they performed tensile testing of bolts that were removed that had indications and those tensile tests were good which indicated that it seemed like some of the UT results were false positives or over calling the indications.

And also two three-loop downflow plants or reactors with Type 316 bolts did an inspection, same time frame. They found no indications, but they preemptively replaced about 200 bolts per each unit.

And also, one B&W plant in 2005 found no indications.

MEMBER SKILLMAN: You identified that the two-loop l990 Westinghouse plants are downflow. The three-loop are downflow.

MR. POEHLER: Yes.

MEMBER SKILLMAN: Are all B&W plants upflow? Is that accurate?

MS. MALIKOWSKI: Yes, that is correct.

MEMBER SKILLMAN: Thank you.

MR. POEHLER: Now we are going to talk about MRP-227-A. Around the year 2000 to 2011, we had a bunch of plants applying for license renewal, PWR plants now we're talking about. And at the time, there was the industry reactor vessel internals Aging Management Program was not developed yet, was under development, so a lot of those plants made commitments to implement the industry program when it was issued. That program was MRP-227 rev. 0 was received by the NRC in 2009. We were reviewing it in the 2009 to 2011 time period. And we issued a safety evaluation on it in 2011. The approved or NRC-endorsed version of that topical report is MRP-227-A which is the inspection evaluation guidelines for PWR internals published

by EPRI.

So I'm going to talk about what MRP-227-A requires as far as inspection for baffle-former bolts. Basically, ultrasonic examination for the initial baseline examination. Westinghouse and CE have the same schedule and scope. It's 100 percent of the bolts and it's to be done between 25 and 35 effective full-power years. For B&W, it's slightly different. It's still 100 percent of the bolts, but the timing is no later than two refueling outages from the beginning of the license renewal period.

And then for both -- for all the different vendor-type of reactors, the follow-up inspections are going to be ten years, a maximum of ten years after the initial inspection. That's if you don't find -- now if you found significant degradation you might have to do it sooner, do the follow-up inspections sooner. And all PWRs have to do these inspections unless they don't have bolts.

So now in 2010, D.C. Cook Unit 2 found visual signs of failure in several -- a number of bolts. D.C. Cook is a four-loop Westinghouse downflow plant. Has Type 347 bolts. There are 832 total bolts and they saw 18 bolts that had visual

indications of failure. They ended up replacing a total of 52 bolts with Type 316 bolts. Most of those were on one large baffle plate. And a total of the 52 they replaced, 42 were cracked. That also includes the original 18 that they saw visually in that 42.

To establish extent of condition, they were basically from the three similar large baffle plates. Basically, there's four of the baffle plates are big plates and have a lot of bolts. On all of the similar plates, the licensee performed a VT-3 visual examination. Didn't see any visual evidence of degradation and they tensile tested one bolt from each plate and that came out fine. So they concluded there was no degradation. They didn't perform ultrasonic testing on any bolts. And they left two bolt locations vacant when they started up.

Westinghouse issued a technical bulletin about this operating experience to alert licensees.

MEMBER RICCARDELLA: Why were two bolts left vacant? And was there a reason for that?

MR. WILSON: This is Bryan Wilson from Westinghouse. Yes, upon completion of the

replacement campaign, there was initially no intention to replace the bolts near the edge because we felt that they were supported appropriately by the replaced bolts.

During the campaign and after it was completed, we found there were say bolts that had pushed the lock bars out as a result of basically torqueing up all the replacement bolts. So there was potentially some damage at those bolts that had caused the lock bars to pop out. So at that time, the tooling had already been removed from site and evaluations were done to show it's okay if we can remove them and leave them like that.

MEMBER RICCARDELLA: Okay, it wasn't for research purposes?

MR. WILSON: No, no, no. It was purely --

MEMBER RICCARDELLA: Okay.

MR. POEHLER: Now I am going to talk about in the 2011 to 2015 time period there were a number of plants performed their initial inspections as required by MRP-227-A. These were the Westinghouse two-loop, mostly Westinghouse two-loop designs and three-loop designs, also a few B&W designs. So the Westinghouse two-loop, as we

mentioned, they are Type 347 bolts. They did their inspections at around 30 to 40 EFPY. See, these are some of the oldest Westinghouse units. They found no more than about ten percent max of any of the units, the ten percent maximum of the bolts were degraded.

The Westinghouse three-loop plants were not quite as old, 30 to 32 EFPY. They also used Type 347 bolts. There were four reactors. Three of the units to the 100 percent inspections and found basically very few bolts, no more than eight per reactor and one of the units did a partial inspection, but didn't find any indications. They'll complete that at a subsequent outage.

Also, there were three B&W reactors that had the UT inspections and they found no more than -- the largest number of bolts failed per reactor or with indications I should say, was four. So really just a handful. None of the combustion engineering plants with bolts have performed their inspections to date.

MR. RUDELL: There is only one CE plant that has baffle-former bolts that's still operating.

MR. POEHLER: Yes, one of them is

shutting down or shut down.

MR. RUDELL: Shut down.

MR. POEHLER: But the other one has plans to do it in a few years, but they will meet the inspection time requirement.

MEMBER SKILLMAN: So Jeffrey, just for consistency, in the first group, the Westinghouse two-loops, those are downflow plants, correct?

MR. POEHLER: They are downflow -- originally downflow. Some of them may have converted at least one may have converted.

MEMBER SKILLMAN: Is there data that shows pre and post conversion for baffle-bolt failure?

MR. POEHLER: No.

MEMBER SKILLMAN: Thank you.

MR. RUDELL: That main data would be the EDF plants and we'll show you.

MR. POEHLER: Now I'm going to talk about Indian Point Unit 2. The 2016 refueling outage, they conducted the MRP 227-A inspection. The edge bolts were all acceptable. UT and visual examination found about 227 potentially degraded bolts out of 832 total. Some of those were found ultrasonically. Some were visually detected and

others, there were 14 inaccessible bolts and inaccessible bolts are considerably assumed to be degraded or failed.

So what did they do for corrective actions? They replaced 278 total bolts. That was all 227 of the potentially degraded. They also replaced 51 additional non-degraded bolts to provide additional margin. And they used Type 316 stainless steel. They completed an analysis to support return to service. That was inspected by NRC Region I inspectors and they sent a number of bolts to the laboratory for testing to support root cause analysis.

For the other unit at Indian Point, Indian Point 3, there was an operability evaluation performed which considered information from Indian Point 2 and Salem 1 which I'll talk about in a minute.

MEMBER BALLINGER: I have a question. We're getting a little bit loose with the definition of 316. Are they L or standard grade? They're cold work 316L grade?

MR. WILSON: No, cold work 316 standard.

MEMBER BALLINGER: Standard grade.

Okay.

MEMBER RICCARDELLA: Excuse me, what were the EFPYs for Indian Point 2 and 3?

MR. POEHLER: It was about 31 for Unit 2 and Indian Point 3 was about 27.

MS. MALIKOWSKI: It was just under 31 for Indian Point 2.

MEMBER RICCARDELLA: Okay, and 27 for 3.

Thank you.

MEMBER BALLINGER: Is there a reason for not choosing L? Because you can get the same strength for culvert no matter what. It's dual certified, right?

MR. WILSON: Yes. I can't comment on that, unfortunately.

MEMBER BALLINGER: Okay, and they're all standard grade, all the replacement 316s are standard grade?

MR. WILSON: That's correct. I'm sorry, for the Westinghouse replacement, yes. There are differences for other vendors.

MEMBER BALLINGER: But you're creating a whole separate database?

MR. WILSON: Right.

MR. POEHLER: At Indian Point 3, they rescheduled --- their baffle-former bolt examination had been scheduled for 2019. They moved it up to 2017.

Now I'm at Salem Unit 1. Also had a spring 2016 refueling outage. It's another four-loop Westinghouse downflow plant, with Type 347. They were conducting first visual examination, basically based on the Indian Point operating experience. Then they found about 11 bolts with visual indications of failure. Or actually, no, it was a total of about 30 bolts with visual indications failure. Eleven cracked at the head. Nine had lock bar. The lock bars were cracked and 19 of the bolts were protruding, bolt heads were protruding. So they decided to UT all the remaining bolts. They found 135 more bolts that were degraded based on UT and there 16 bolts that were unable to be UT'd. So those were also assumed to be degraded. So overall, there was about 190 bolts identified as potentially degraded and needing replacement.

There were significant clustering of bolts in several octants of the baffle-former assembly, so the bolts, the degraded bolts were

heavily concentrated in those areas and more sparse in the other areas.

MEMBER SKILLMAN: Jeffrey, what consideration was given to the meaning of that bullet? It seems to me that that's a very significant finding from all of the data coming out of Salem 1. So what is the conclusion of the Westinghouse team for the clustering?

MR. POEHLER: Yes, I'm going to talk a little about the clustering. It's definitely been seen at these four-loop plants and it was more severe at Salem. So I'm not sure we have an answer for why it was more severe at Salem than at Indian Point. I don't know if the Westinghouse team has an answer for that.

MEMBER SKILLMAN: Well, let me ask -- are the reactor coolant pump flows and reactor coolant pump discharge heads the same for the two plants?

MR. WILSON: They are very similar, the two plants. Not enough to conclude a major difference.

MEMBER SKILLMAN: Are the loop geometries and the pipe diameters the same?

MR. WILSON: I can't answer that here.

I'd have to confirm that one. I don't know off the top of my head.

MEMBER SKILLMAN: I'm obviously picking. Is there a standing wave or is there a resonance that comes from main passing frequency times four pumps, full power at the density that you're at full power?

MR. WILSON: Right.

MEMBER SKILLMAN: It just seems to me that there is information there that the very curious might try to mine.

MR. WILSON: Right. The one thing we do know is from looking at this, the flow into the baffle-former region has to go through a path that is a little bit more torturous than the norm. So it has to enter through a flow hole in the side of the barrel and then down through the formers. It doesn't -- usually what happens at that point is a lot of these say pressure variations and things get kind of washed out a bit by the time you make that path change.

So there hasn't been a strong correlation with say pump-induced related influences, but such as I think the disturbances you're mentioning. But that certainly is stuff

that we'll continue to look at as understanding the situation.

MEMBER RICCARDELLA: Does the fracture morphology show any signs of fatigue and is it conclusive, whether it be fatigue or stress corrosion?

MR. RUDELL: I can -- yes, there is fatigue in the bolts that we've looked at so far. Some have a significant amount of fatigue, maybe 50 to 60 percent of the failed fracture surfaces appear to be fatigued, and then others nearly all is intergranular, appears to be irradiated as stress corrosion cracking. And we're trying to put those pieces together. That's why we don't have a whole lot of information in the destructive examination work that's ongoing right now.

MEMBER RICCARDELLA: But it should also tell you if it's high cycle versus low cycle.

MR. RUDELL: Yes. We're trying to get that out of it, but you're right. It could be high cycle or low cycle fatigue, depending on what comes first.

MEMBER RICCARDELLA: But the fracture morphology should identify that, right?

MR. RUDELL: Yes.

MEMBER RICCARDELLA: I mean that would address your concerns about the vibration.

MEMBER SKILLMAN: What I'm really thinking is that the CPY plants, the French plants that are four-loop that are different than the CPOs that are four-loop downflow, the CPY plants are four-loop upflow. And this problem has basically ceased for the CPY upflow. So my hunch is that there's a standing wave and that there's a resonance cavity back in the lower internals of the Westinghouse design where the clustering would occur.

It's probably something envisioned, but it just strikes me that the downflow, upflow, and geometry of your baffles in the Westinghouse large plants invites a question that really gets to what Dr. Riccardella is talking about. If it's high-cycle fatigue and it's preponderance, then I would suspect that there's a standing wave and what you're really getting is tensile failure as a common -- as the consequence of fatigue.

MR. WILSON: Right. So for the upflow and downflow, there's a significant difference in the -- I'd say steady pressure, you know, that's behind that plate. So any kind of influences of

these say stress fluctuation or pressure fluctuations or anything would even be further magnified by that say steady difference. I mean it's basically an order of magnitude difference in pressure, in Delta P across that plate which is enormous.

MEMBER SKILLMAN: The real issue is the area because those plates are large plates, so even small Delta P magnifies to a very large force around those bolts.

MR. WILSON: Right.

MEMBER RICCARDELLA: Because then you would -- would you see -- that's obviously worse at the top than at the bottom.

MR. WILSON: Right.

MEMBER RICCARDELLA: Are we seeing a trend of more cracking near the top?

MR. WILSON: Yes. The patterns of failed bolts tend to be clustered in areas of the -- right, in the U.S., in areas of the highest pressure. So there's a clear trend there.

I'll touch on this clustering topic a little bit more in my presentation later. I know Jeff has some additional thoughts on this.

MR. POEHLER: Okay, Salem, what did

they do? Correct actions. They replaced 189 bolts, the Type 316. They also sent some bolts to the lab for analysis. They did a minimum bolting pattern analysis to determine or to confirm that their replacement scope was acceptable for one-cycle operation prior to reinspecting.

For the operating unit, Salem Unit 2, they performed an operability determination based on the extent of condition from Salem Unit 1. They also moved up the schedule UT examination of the bolts to next year from 2026.

MEMBER RICCARDELLA: Just to complete, what were the EFPYs for Salem, just two centimeters?

MR. POEHLER: It sounds a little less more in the 24 to 26 range.

MS. MALIKOWSKI: Salem 2 is in that range, and then Salem 1 is 28.

MEMBER RICCARDELLA: Salem 1 is 28 and Salem 2 is --

MS. MALIKOWSKI: Twenty-five.

MEMBER RICCARDELLA: Twenty-five.

MS. MALIKOWSKI: It will be 25 -- I believe in the spring.

MEMBER RICCARDELLA: Thank you.

CHAIRMAN SUNSERI: I would like to remind the speakers to make sure you speak loud enough so the transcriber can hear you.

MS. MALIKOWSKI: My apologies.

MR. POEHLER: Now D.C. Cook Unit 2 was performing UT inspection during the fall outage and this is the first plant to do the UT in accordance with some interim guidance from the MRP, EPRI MRP which we'll talk a little bit more about later.

They found a total of 179 potentially degraded bolts. There were also, as I mentioned, the two vacant bolt locations from 2010.

Some notable new elements at D.C. Cook where there were six of the replacement bolts installed in 2010 had indications. At least -- also, at least one of the vacant bolt hole locations correlated with some damage to a fuel assembly.

MEMBER SKILLMAN: Jeffrey, would you please explain that more?

MR. POEHLER: The damage to the fuel assembly?

MEMBER SKILLMAN: Yes.

MR. POEHLER: I mean it was basically damage to the fuel assembly that was suggested of

baffle jetting type issues going on.

MEMBER SKILLMAN: So jetting from an empty hole?

MR. POEHLER: Apparently.

MEMBER BALLINGER: Now the replacement bolts were 316 though, right?

MR. POEHLER: Correct.

MEMBER BALLINGER: So this is 316 bolts that are cracked?

MR. POEHLER: Right.

MEMBER SKILLMAN: So wait a minute. Damage fuel assembly jetting from an empty hole, so what did the plant do? Did they remove this fuel assembly and correlate basically a shine on a fuel assembly or a set of pins or a spacer grid with a location of the adjacent hole? Is that what that means?

MR. WILSON: Yes, they had correlated the location. Actually, one of the fuel rods had I guess been displaced inward and they correlated the location of that fuel rod with the spacing from -- or to that location of the hole.

MEMBER SKILLMAN: Of the hole.

MR. WILSON: Of the hole. Within like an inch or so of that location.

MEMBER SKILLMAN: Okay, thank you.

CHAIRMAN SUNSERI: So based on the experience with baffle jet fuel failures, why was that not anticipated? It seems like that that should have been anticipated as a potential consequence.

MR. WILSON: Right, at the time -- with the lack of UT available at the time of that inspection for that configuration of bolt, we did, I guess, as much of what we thought was an extent of condition at the time as we could, removing bolts in those locations, and then, I guess, expected that we had covered the range of failures of those bolts at the time.

And the conclusion was that if we had locked down the plate by putting these replacement bolts in, that the baffle plate would have been in contact with the former plate such that you wouldn't really have a path for flow to get out of that hole.

So the baffle jetting occurs when you have some flexibility that allows a joint to open, but we had expected that that joint would have been closed or mostly closed, at least to the point where you could accept the amount of bypass in

jetting that was occurring, could occur from that hole. What we come to find out later is that say to the left of that, those replacements could have potentially been an additional amount of failed bolts that were undiscovered by the path that was chosen for looking for an extent of condition. And if those bolts were already failed, there was a mechanism to kind of loosen that whole section of the plate and allow for flexibility that allowed flow to go through.

MR. RUDELL: To Bryan's point, those two holes were vacated in 2010, coming out of that outage. So there were prior cycles where there was not any damage to the fuel associated with those holes and there were other units that are operating with baffle-former bolts vacated and I know the one I'm familiar with does prescribed inspections. I suspect that this unit did, too, particularly looking at the fuel assembly in that area, and knowing that that has a vacated baffle-former bolt there.

And the inspections I'm familiar with have not seen any indications of any jetting associated with those vacant holes. So here we have in this past cycle, this past operating cycle

at this one plant discovered this issue with the jetting out of vacated hole.

MEMBER SKILLMAN: Did the primary chemistry, cesium and strontium, elevate, indicating a weaker --

MR. RUDELL: They indications of a failure. Yes.

MEMBER SKILLMAN: They did. Thank you. Thanks.

CHAIRMAN SUNSERI: Were these vacant holes near the top of the former plate or lower?

MR. POEHLER: I think the next slide. This is a diagram -- this is the diagram of the baffle former assembly at D.C. Cook showing the locations. This is showing the as-found condition in the 2016 outage. And the way this works is the green bolts were good bolts with no indications. Red bolts have indications. And I think the gray ones were the empty holes which are somewhere right about in that area. One of them is right here and the other one is over here.

So this is the large baffle plate, right here. This is where in 2010 they had replaced the 52 bolts or most of them. So you can see these sort of -- the ones with the square

symbol are the replacement bolts.

In this yellow box here, this is the six replacement bolts that were found degraded and this right here is where they found five degraded edge bolts as well, right here, on this seam here.

MR. RUDELL: So the sea of red.

MR. POEHLER: Yes, you note that there's quite a few degraded bolts around where the edge bolts were and these five edge bolts were all in a row right on this.

MEMBER KIRCHNER: How does the -- with your map here, just refresh my memory. How do the primary coolant loops line up with the orientation there? And in particular, where's the cold leg coming in? Is that immediately adjacent to the red field of degraded bolts?

CHAIRMAN SUNSERI: Before you answer that question, let me make a request here. I've got some feedback from the people listening on the public lines. Since this is kind of a -- NRC is leading this presentation, but we have other people speaking, please identify yourself so that the people on the line can understand who is talking. Thank you.

MR. WILSON: So this is Bryan Wilson of

Westinghouse. Yes, the cold leg is in that vicinity. It's not exactly adjacent to those failures, but --

MEMBER KIRCHNER: This is where you would see the maximum pressure differentials, especially with the downflow?

MR. WILSON: Not necessarily. Because of the way the flow kind of has to go around the annulus and then into the flow holes, it doesn't necessarily line up always with the cold leg.

MEMBER KIRCHNER: But this is a downflow design?

MR. WILSON: Yes.

MEMBER KIRCHNER: I would expect this to be the point of highest pressure.

MR. WILSON: That is the elevation of highest pressure.

MEMBER KIRCHNER: Then going back to Dick's earlier comments on vibration --

MEMBER SKILLMAN: Let me push a little further on Dr. Kirchner's question. Is there history of operating with less than four pumps on this plant? In other words, could they have run for any extended time with two pumps on that loop where that is the cold leg dominant location?

MR. WILSON: No, I don't think there's any experience with operating for an extended period of time with that, with less than four pumps.

MEMBER SKILLMAN: I don't know the tech specs for this plant, but I know that you are permitted for a certain time period less than four, but it's a short time period. But I'm just wondering if this is on a loop closest to where there might have been an extended time period of less than four-loop operation for the standing wave on this corner of the internals might have been greatest from the pumps that were operating in the loop closest to that baffle.

MR. WILSON: Right, we have actually looked at -- we were primarily looking at pump sequence, end of start up and things like that. Related to your question, we were not able to come to a definitive conclusion that you would have expected a correlation between this. It certainly looks like there's a correlation, but from a fluid hydraulics standpoint, it doesn't, you know, the math doesn't work out, I guess, if you will.

MEMBER SKILLMAN: It could be manufacturing, too.

MR. WILSON: Right.

MEMBER SKILLMAN: So I understand that. But I just wanted to build on Walt's question. Thank you.

MEMBER RICCARDELLA: Just to complete the story, the EFPYs for Cook 2 and also I don't see much about Cook 1.

MS. MALIKOWSKI: This is Heather Malikowski. This is 28 EFPY for Cook 2.

MEMBER RICCARDELLA: Twenty-eight in 2016?

MS. MALIKOWSKI: Correct.

MR. POEHLER: Okay, so the corrective actions for Cook 2 was -- they're still underway, but they plan to replace a minimum of 181 bolts with Type 316. And that includes all the degraded bolts, plus the missing vacant locations. And they're planning to replace additional bolts up to 201 bolts as time permits in their outage.

They're investigating the indications in their replacement bolts. I just want to note that there have been replacement bolts in service for about 10 to 15 years from some of the late '90s inspection activities and they have seen no indications in those replacement bolts, Type 316

bolts.

The replacement bolts are going to be sent for laboratory analysis. I think that's being done very soon.

They're also having a sensitivity analysis performed to explore the effects on replacement bolt stresses from some of the failed original bolts in that area that would support the root cause for the replacement bolt failures.

So as far as the baffle edge bolts, the licensee is still determining what they're going to do about corrective actions for those. It's possible they may not remove them.

Now I'm going to go into factors that influence baffle-former bolt degradation. The first one is neutron fluence. So austenitic stainless steel is normally very resistant to stress corrosion cracking in a PWR chemistry environment. However, when you get high fluence, you get grain boundary and chemistry changes occurring.

The neutron fluence threshold for IASCC is 2 times 10 to the 20 per neutron per square centimeter or 3 dpa. Baffle plates and bolts, some baffle plates -- areas of the baffle plates and

bolts are going to get up to 75 dpa over a 60-year life. However, the patterns of the bolt degradation we've seen it at Indian Point 2, Salem 1, and Cook 2, don't seem to correlate with the highest fluence locations in the core.

Also, I'd like to note that the two and three-loop Westinghouse plants had similar higher fluence levels to the four-loop plants, but fewer degraded bolts.

CHAIRMAN SUNSERI: We need to be a little careful about fluence threshold at 2 or 3 dpa. That also requires an environment. So the microstructure is susceptible after 2 or 3 dpa, but you still have to have the environment which is not a PWR environment, right? In a BWR 2 and 3 dpa, you're off to the races for this material.

MR. POEHLER: Yes, I mean it can happen, the PWR environment. But at a low dpa like 3 dpa, you're going to need a very high stress. It's kind of a sliding scale of stress versus dpa, so the higher the dpa, you can have it at lower stress levels.

MEMBER RICCARDELLA: Could you also expect there to be a threshold at the other end, too, at the high end where you damage -- the damage

somewhat saturates and beyond that, additional dpa don't matter?

MR. POEHLER: Yes, I mean the mechanical properties at some point are going to saturate as it further increases in tensile and yield strength. Grain boundary chemistry might as well.

MEMBER RICCARDELLA: What's that dpa level?

MR. POEHLER: I don't know if I have a precise answer.

MEMBER RICCARDELLA: It doesn't have to be precise.

MR. POEHLER: I'm thinking something around 15 maybe.

MEMBER RICCARDELLA: Okay. Thank you.

MR. POEHLER: Also, one other thing, you know, most of the plants have switched to a low leakage core design so that's going to reduce the flux to the bolts which could slow initiation of new IASCC cracks. Some of these may have initiated earlier in life.

Let's talk about stress. So baffle-former bolts have stresses from a variety of sources. You have bolt pre-load early in life.

That will relax due to radiation assistance stress relaxation in a fairly short time of operation.

At higher fluences, you get void swelling in the baffle plates and that can put different stresses on the bolts such as bending stresses as the plates grow vertically.

You also have differential pressure so in a downflow plant design it's greater than in an upflow design. And that's going to become more of a factor as your pre-loads relax.

You also have differing numbers of bolts per plate area. And a four-loop plant has a larger core, larger plate area, but it has --- a four-loop basically has the same or less bolts than a three-loop plant.

Also bolt geometry, the head-to-shank radius, different geometries were used for different bolt styles, especially the 347, as we mentioned, has the sharper head-to-shank radius than the 316 design.

Fatigue loads can also have an influence that's affected by the number of transients you've had over the life of the plant.

MEMBER RICCARDELLA: Jeffrey, there you're referring to low cycle fatigue as opposed to

vibration fatigue that Dick is alluding to.

MR. POEHLER: Right.

MEMBER SKILLMAN: One thing that you didn't show and you may have considered, you just didn't write it down, is manufacturing tones. When you look at the map of the bolts, one can conceivably think that while the internals were being constructed, was constructed over a series of days as the bolts were inserted, torqued, and strapped, it may have had different crews, different materials, bolts from different lots.

So my question is what consideration did you give to manufacturing issues?

MR. POEHLER: Obviously, if the pre-load varies, if you had a higher pre-load or pre-load was out of spec or at the high end that -- that could potentially be a factor early in life as far as stresses. But if pre-load is low, then it's going to relax faster, too. If you're on the low end, maybe you get fatigue occurring sooner because your pre-load relaxes. So that's -- those are factors that in the root cause for these plants they're going to need to look at if they have that history in the manufacturing.

MEMBER RICCARDELLA: Are these

lubricated when they're torqued?

MR. POEHLER: Yes, they generally are, yes.

MEMBER BALLINGER: These are machined bolts, not forged and then machined? Are they machined from bar or are they hot headed and then machined afterwards?

MR. WILSON: Yes.

MEMBER BALLINGER: Yes and yes and yes?

MR. WILSON: So this is Bryan Wilson from Westinghouse. It depends on which bolts you're referring to because there are differences between the 316 and the 347 and how they were manufactured. All of them were thread rolled.

MEMBER BALLINGER: So the threads are rolled threads?

MR. WILSON: Yes. But the heading was either done by machining like in a 316 or by hot heading for the 347.

MEMBER BALLINGER: Hot heading.

MR. WILSON: So that was -- when I said earlier about manufacturing differences between a 347 and 316, there was a transition where they went from hot heading to machining these bolts. And so yes, the 347 to start with the nominal shank of the

bar and then hot head to -- you know.

MEMBER BALLINGER: That explains the 347 actually.

MR. WILSON: Yes.

MEMBER BALLINGER: Okay. I get it.

MEMBER SKILLMAN: Will the review of the manufacturing records be conducted?

MS. MALIKOWSKI: This is Heather Malikowski, Exelon. Yes, they have reviewed what they can retrieve. And I believe Bryan can elaborate. I believe there's only a few heats of material that were supplied for the bolts, but we can't correlate them to where they were put in the plates. So we know there's only a few common heats, but there's nothing to help say we can say oh, it's just this heated material. Those records were looked for.

And I think as far as the manufacturing and assembly, I mean they found what they can, but there's not enough granularity to it to say you can say there's some sort of rework or something like that that may have also helped lead to a focusing of the failures.

MR. WILSON: Right, yes, there wasn't a smoking gun, if you will, you know, related to

manufacturing defects.

And as Heather pointed out, these were supplied in batches and installed as such. There was control of the parts to a plant as far as what heats and stuff went to a plant, but not specific location on an individual plate.

MEMBER SKILLMAN: Let the record show that some of these plants were manufactured and constructed before there was an Appendix B to 10 CFR 50. Now that doesn't mean that these individuals who assembled these plants were careless. They used the greatest caution and the best material and the best assembly practice that they knew. There might still be some lessons learned from digging in some of those old records is all I'm saying.

MR. WILSON: Understood.

MEMBER SKILLMAN: Thank you.

MR. POEHLER: Okay, now moving on, I'm going to explain the difference between downflow and upflow. So this diagram shows on the left, this is a downflow plant. And basically here's -- the difference is flow direction in the space between the core barrel here and the baffle plates. Here are your former plates. You have holes in the

former plates.

In a downflow, you have a hole up at the top of the core barrel, the OD of the core barrel where a flow comes in and flows down, a portion of the flow flows down to this space and then up out at the bottom and up through to join the rest of the core flow through the core.

In an upflow plant, these two flows are parallel. There's no hole up here, so between the baffle and the core barrel space, flow is also going the same direction. So the DP, differential pressure, is very little with these two. Pressures are very similar, whereas in the downflow, there's quite a bit of difference, especially in the top of the baffle-former assembly.

So this kind of shows if you did an upflow conversion, you would plug that hole there, but even with an original upflow plant, you just would never have a hole up there.

MEMBER SKILLMAN: What is involved in an upflow conversion? How many holes need to be plugged?

MR. WILSON: This is Bryan Wilson, Westinghouse. All of the core barrel holes are plugged.

MEMBER SKILLMAN: How many?

MR. WILSON: I'm not an upflow guy.

MEMBER SKILLMAN: In a fuel assembly plant, there are 168 holes in the bottom. I would imagine it's a number like that on the core barrel periphery.

MR. WILSON: No, it's on the order of like say 16 to 32.

MR. RUDELL: A couple dozen. It's going to vary between two, three, and four-loop and also the design for the holes that are now going to be added and upper former plate will vary. And the Delta Ps also vary after you do that design, depending on the design of the flow holes through the other plates that were there originally. And they vary. And the degree of reduction in Delta P that you get will vary depending on the design of the original plant and the flow holes in those baffle formers. So it gets complicated.

The work itself is all done, of course, under water and controlled, special tooling, special controls for foreign material and other things. And as well as all of the design analysis. Now your accident analysis need to be revisited. Many of them need to be revised with all of the new

core bypass values, even going back to your core physics specifications will be changed accordingly, so from an engineering standpoint, there's a lot of paperwork and design modifications, 50.59 reviews, and revisions to perhaps a lot of these accident analyses. So it's quite an undertaking to convert, but it is and can be done.

MEMBER RICCARDELLA: Is there a significant difference in power output in the fluid or not?

MR. RUDELL: I guess I wouldn't say significant with the higher core bypass flow. You will lose some of --

MEMBER RICCARDELLA: I mean a percent?

MR. RUDELL: About that.

MEMBER RICCARDELLA: Thank you.

MR. RUDELL: About that core bypass flow, I don't know what it comes out to in power.

MEMBER SKILLMAN: Thank you.

MR. POEHLER: Now I'm going to talk a little about the clustering effects. So the theory that clustering is -- you initially baffle-former bolts get random failures in random locations. You get a failure of one bolt, that leads to load being transferred to adjacent bolts and over several

operating cycles this can cause the adjacent bolts to the original bolt to crack which leads to clusters of failed bolts. Some people call that unzippering. That has been observed in French plants when they did successive examinations. And in the U.S., in the four-loop plants, the more severe clustering was seen at Salem Unit 1.

So to summarize, some of the factors -- sorry.

MEMBER KIRCHNER: Jeffrey, may I interrupt?

MR. POEHLER: Sure.

MEMBER KIRCHNER: Have you looked at the patterns of clustering from plant to plant? Do they correlate and do they correlate, I'm thinking with the -- let's see, you have D.C. Cook, this diagram here. Do the other plants show the same clustering for degradation?

MR. POEHLER: You know, for Indian Point, the clustering was less pronounced than that. D.C. Cook, if you look at -- if you consider where the 2010 failures were and that was pretty clustered in that one plate. But Salem was probably even a little more clustered than D.C. Cook.

MEMBER KIRCHNER: By clustering though, could you give us a little more precision where was it happening in the relative core location? At the top, one would expect it, right?

MR. POEHLER: Yes.

MEMBER KIRCHNER: And in the smaller plates?

MR. POEHLER: Yes, both smaller -- I mean yes, smaller and larger plates, but at Salem it was more over about three of the octants -- basically three eighths of the total baffle-former assembly had a lot of clustering. And it wasn't necessarily -- I don't know, maybe Westinghouse could comment. I don't -- I didn't think it was really more either at the top or bottom or center of the elevation of the plates.

I think in Cook, as we definitely saw that one cluster from 2010 was more toward the top of the plate.

MR. WILSON: Yes, this is Bryan Wilson from Westinghouse. I mean for Indian Point and Cook, there seemed to be more of a trend towards the top. I think the Salem was a little bit more widespread at the time whenever the indications were discovered. Whether it started in a certain

location or not, is yet to be determined I think, but it did have -- I'd say as the clustering tailed off, it did have the trend to be say more towards the top, not focused necessarily on the bottom.

MEMBER KIRCHNER: And to follow up, was it more in the region of the smaller plates here, this part of the core?

MR. WILSON: There was both. Well, as far as -- coordinate location, you know, it was I'd say clocked similarly to Cook where it was a large patch in the wide plate, the 12 bolt wide plate and that it moved. I'd say clustering was occurring to say one side of that wide plate into the say narrower plates.

MEMBER KIRCHNER: The narrower plates having a higher Delta P across them?

MR. WILSON: No. They have the same Delta P, but because of their width, the load per bolt is slightly different.

MEMBER BALLINGER: The pre-load, back to the pre-load, you say they were lubricated?

MR. WILSON: Yes.

MEMBER BALLINGER: And they're torqued. Do you know what the uncertainty is on the pre-load stress?

I've got some experience with bolt loads and sometimes that uncertainty can be enormous, even for the best of circumstances.

MR. WILSON: Right, yes, I mean that has all been studied many times in the variations, but I don't know at that time what the variations that they were measuring. I know a lot of testing has been done since then to look at those variations and account for them.

I can't comment specifically at that point in manufacture, in this history what the --

MEMBER BALLINGER: Typically, 50 percent sometimes.

MR. WILSON: Right.

MEMBER RICCARDELLA: I've seen data where if you're shooting for 70 percent, you're getting anywhere between 50 percent and 90 percent of yield.

MEMBER BALLINGER: And since IASCC is a nonlinear function of stress.

MR. WILSON: Yes. Certainly, add to the list of contributors, right?

MEMBER SKILLMAN: Let me ask to go back to slide 25 just for a second, please? My question is for Bernie.

Bernie, you used the word 50.59. I would think if a plant wanted to change from downflow to upflow, the 50.59 screening would very quickly push you into a license amendment request. Is that your understanding, too?

MR. RUDELL: I actually don't know whether all these upflow conversions were done under license amendment or not.

MEMBER SKILLMAN: Could you find that out? It would seem to me that if you were to decide in your going through the 50.59 which is really a screening, you would tumble one or two or several things, either a change in analysis, change in fuel temperature, or major change in plant design. And I would think number three would be the hook that would require a license amendment request. It's a curiosity question, but I concur with you. This is a very substantial undertaking and I'm not suggesting that anybody should do that because of the magnitude of the work that is involved. But I'm curious, there have been other examples in industry where the applicant has used 50.59 and has suffered as a consequence by not going to a full LAR, license amendment request. So

would you get back to me, please?

MR. RUDELL: Okay.

MEMBER SKILLMAN: Thank you.

CHAIRMAN SUNSERI: Jeffrey, we are slightly behind schedule here, so I'm not asking you to speed up, but just be mindful of that. I suspect with the discussion that we're making up some time on the second part of this presentation, but let's try to move the pace along a little bit. Thank you.

MR. POEHLER: Yes, so just to summarize some of the factors influencing degradation of baffle-former bolts, it involves a complex interaction of stress, fluence, and fault material and design. You have some other aging mechanisms, void swelling, and stress relaxation can both influence IASCC bolts. Right now, the industry is working on predictive models for baffle-former bolt degradation which would account for all of these different variables.

Also, the staff observation is that the highest susceptibility to degradation seems more related to stress differences rather than differences in fluence between the various plant designs.

I'm going to briefly talk about inspection replacement. I think we've talked a lot about it already. Inspection is either by ultrasonic or visual examination. Ultrasonic testing is right now only for detection of flaws. It's not demonstrated for sizing. Any bolt with a crack-like indication is called potentially defective.

Visual examination, VT-3, is not specified for baffle-former bolts. Some plants have performed voluntary VT-3s in response to operating experience. It is used for the baffle edge bolts as the primary inspection. VT-3 can find evidence of failed bolts such as displaced lock bars for treating or missing bolt heads. And some of the failed bolts have been detected by other visual inspections that were not VT-3 inspections such as those at Cook in 2010.

Replacement bolts, I think she's going to talk a little more about this, but a lot of bolts can be removed intact when you cut the lock bar. The shanks, they're broken off. They can sometimes be removed mechanically, but might require electro-discharging machining.

The replacement bolt design uses an

expandable locking cup, so you don't have to weld on the highly irradiated baffle plate material. And those bolts are 316 as we mentioned. They may have been improved geometry or reduced stresses at head-to-shank transition.

Next. Okay, so the next part of this presentation I'm going to talk a little bit about evaluation of baffle-former bolt degradation, how do we evaluate when they find conditions that are -- that don't meet the acceptance criteria.

I'm going to talk about acceptable bolt pattern analyses. So there's WCAP-15029. That's the NRC approved generic methodology for determining acceptable patterns of intact baffle-former bolts and that goes back to around 2000 when they were doing the original pilot inspections.

The methodology uses the MULTIFLEX computer code to determine accident loadings. Some of the acceptance criteria include bolt stresses, fuel grid impact loads, momentum flux which is a parameter related to baffle jetting, also fatigue, but high and low cycle fatigue and core bypass flow.

When they evaluate, NES found bolt degradation, any degraded is assumed to carry no

load, no it's completely discounted in the acceptable bolt pattern analysis. It does use the irradiated material properties for bolts which increases the strength of the bolts. And plants have used this methodology both to evaluate as-found conditions, you know, if you have a few bolts degraded or it's okay to start up without placing, and also for potential replacement patterns to make sure that the stresses and other criteria are okay.

In some cases, you might not meet the stress and fuel grid impact criteria and if you do exceed those fuel grid impact criteria, you might need to demonstrate a coolable geometry with some damage to peripheral fuel assemblies. The WCAP-15029 provides some guidance on how you do that.

CHAIRMAN SUNSERI: I'm not sure where to ask this question. This might be a good place to do it, but I noticed a lot of the evaluation of acceptability has been based around stress and structural concerns.

I had a question that I got from looking at the Westinghouse Nuclear Safety Advisory Letter. It deals with the emergency flow cooling flow. I don't have a good sense for how much flow can get past this baffle and maybe bypass part of

the core. It indicated it was nominal. It was dismissed as nominal, but I mean do we have more quantification of that and whether that could potentially be a problem in a severe unzippering if you will?

MR. WILSON: Yes, so we did look at -- for the safety evaluation that was done by Westinghouse. I'll talk a little bit about this later, but we considered a condition where the baffle plate was basically had no baffle-former bolts intact. So I would say that's a very severe condition. We also -- regarding edge bolts, we assumed that the edges of the bolt -- of the baffle plate were simply supported, such that the baffle plate could flex inward towards the fuel assembly.

In that condition, it was evaluated at about 4.5 percent or 1.4 percent bypass flow could occur. That was looked at in these emergency core cooling scenarios, LOCA and non-LOCA, and was determined to be say within the bounds of margin, it was available in those evaluations and the levels of conservatism that were already built into those evaluations.

So we don't expect, under that circumstance, to be a concern with the amount of --

CHAIRMAN SUNSERI: So it's within the bounds of the margin, but I mean --

MR. WILSON: Yes, easily within. It wasn't up against the edge. It was -- I'd say dismissed as negligible with respect to what's available in conservatism and margin.

CHAIRMAN SUNSERI: Okay, thank you.

MR. POEHLER: And I'll talk about reinspection intervals. So if you do find a large number of degraded bolts, there is another WCAP 17096 which staff recently approved. It's reactor internals acceptance criteria and methodology and data requirements. And this provides guidance for engineering evaluation of baffle-former bolt degradation. It actually covers all of the different PWR internals, but baffle-former bolts is one of them.

So there's a numerical margin which can be determined which consists of additional bolts over and above the number and the minimum bolting pattern. The minimum bolting pattern is that pattern that has the fewest number of bolts that would meet all the acceptance criteria and that also -- the location of the bolts is also a factor. Obviously, if they're severely clustered they may

not need certain stress criteria.

So what the WCAP 17096 says the number of degraded bolts you find is less than half the margin, you can continue -- you can start up and reinspect again in ten years without replacing those bolts. If the number of degraded bolts is greater than half the margin, you have to justify a different reinspection interval. So the plants with these outages in 2016 that had large numbers of degraded bolts, they would not have met the WCAP criteria. So those plants elected to replace essentially all, at least all the bolts that were degraded through historical structural margin. Industry is still developing models for failure rates of baffle-former bolts. So we do need to see reinspections of bolts at shorter intervals, at least now, to establish what are the failure rates of these bolts.

Okay, now I'm going to talk a little about the NRC response to this operating experience of baffle-former bolts. First thing is regional inspections. So the NRC staff performed targeted inspections at the three plants, Indian Point, Salem, and we're currently still in the inspection at D.C. Cook. Those inspections are focusing on

the quality and accuracy of the non-destructive evaluation. They're focused on the corrective actions including how you show the operating unit is still okay to continue operating. They focused on adequacy of their replacement bolt pattern including margin for additional failures during the next cycle.

There's also these inspections, at least for Indian Point or Salem are documented and publicly available inspection reports. And regional inspectors are engaging with other plants with regard to operability evaluations and plants with upcoming outages. And those are the plants that -- mainly the plants that are similar to the ones that have found the degradation.

MEMBER SKILLMAN: Jeffrey, how is the inspection protocol established for these inspections?

MR. POEHLER: How is the protocol established?

MEMBER SKILLMAN: Yes. Do you have an inspection procedure?

MR. POEHLER: They do.

MEMBER SKILLMAN: And how was that established?

MR. POEHLER: You mean specifically for baffle-former bolts?

MEMBER SKILLMAN: Yes. At the top of 33, you're saying staff performed targeted inspections. What's their basis for the inspection? What does the inspection prove?

MS. ROSS-LEE: I'm sorry, wasn't the initial inspection -- sorry, this is M.J. Ross-Lee.

I think at least the inspections were done as part of the MRP 227A. So the aging management said hey, you should look at these in the certain EFPY for the plant. And so Indian Point did that so -- that's what the inspections were a part of was part of our aging management program. So in this period of time you should look, do this inspection, and that's what they were doing at Indian Point.

MEMBER SKILLMAN: So if I repeat back, it was part of the AMP for that plant?

MS. ROSS-LEE: Is that a correct characterization of the MRP 227?

MR. POEHLER: Yes, what the licensee -- the physical inspections conducted by the licensee, yes, are part of the AMP. But then the activities of the NRC, I think you're asking about the

activities of our NRC inspectors?

MEMBER SKILLMAN: I am. I'm referring back to the first bullet on Slide 33.

MR. POEHLER: Basically, I don't think they have a specific inspection procedure just for baffle-former bolts. But the type of inspections they were doing were -- they called it a problem identification resolution sample, so it's, you know, targeted at this specific issue, but --

MS. ROSS-LEE: We might have -- I'm sorry, this is M.J. Ross-Lee.

There might be regional people listening. We might have to get back to you. Yes, so I answered the question on why the industry was doing them. It was upon discover of the number of degraded bolts which was greater than what was expected that led our inspectors to engage and what we're, I guess, referring to there as a targeted inspection. But if you need more specific detail, I'll probably have to try to reach out to one of the regions to get those detailed answers.

MEMBER SKILLMAN: What I was really exploring was the formality of this inspection, whether it's a standing inspection procedure, or if this is a special inspection protocol that has been

developed between the NRC and industry, based on the MRP, for these specific findings or whether this is just a Keystone Cop showing up with a magnifying glass and a sounding pin.

I have a hunch, it's very formal. It's very well constructed. When you say it can be reviewed, that suggests to me a level of inspection formality that's very, very serious. So my question is what is that? Is that an IEP? Is that a special inspection? Is it just a PI&R? How formal is this for the NRC inspectors because if they have to do this at plants other than Salem and D.C. Cook and Indian Point, perhaps there needs to be a rigid formality to this.

MR. POEHLER: I mean I don't think it's not a special inspection. It is a PI&R. But they would have a test or an inspection plan ahead of time.

MS. ROSS-LEE: We might -- I'm sorry, this is M.J. Ross-Lee again. To get the answer to your question, we have to get -- it's a formal inspection. It's documented. There are inspection reports that are issued on it. I don't have access to those right here, but at least for the two that are done and public, we can pull those up, find the

actual inspection procedure that was used. So we can get that information back for you. I just don't have it with me right here.

MEMBER SKILLMAN: Thank you.

MR. POEHLER: Another part of the NRC response was we performed what we call a LIC-504 evaluation. LIC-504 is an office instruction. The title of it is Risk-Informed Evaluation of Emerging Issues. It's basically a process we use to evaluate the safety significance of new issues and so we look at different options for addressing the issue. Basically, it boils down to do you need to -- is this a safety issue that warrants shutting the plant down immediately or not?

But under the four options we looked at for the baffle-former bolt issue were immediate shutdown and inspection, or inspection next refueling outage. This is inspection of the baffle-former bolts. Doing a generic communication to gather more information, or just maintaining the status quo which would just be let them keep on inspecting as per the guidance to the MRP 227 on that schedule.

When you do the LIC-504 process, you look at five criteria and compliance with existing

regulations, consistency with defense-in-depth philosophy, maintenance of adequate safety margins, demonstration of acceptable levels of risk, and implementation of defined performance measurement strategies.

So what were the results of this LIC-504 evaluation? Well, the risk met the two LIC-504 criteria are core damage frequency of less than 1 times 10 to the minus 3rd and large early release frequency less than 1 times 10 to the minus 4 per year. So if you're above those risk levels, then that would indicate you probably need to shut down immediately. If you're below them, you don't.

The risk levels for LOCA, they're driven by basically the low frequency of large and medium LOCAs results in low core damage frequency due to LOCA. Small break LOCAs are more frequent, but we don't think they have the potential to detach or deflect baffle plates such that it would cause significant fuel damage.

Also, there was a separate seismic risk assessment performed that used bounding seismic hazard curves in U.S. based on recent updated seismic hazard submittals. And the seismic assessment also assumed a 75 percent reduction in

load-carrying capacity for the baffle-former bolts which is much greater than we've seen at any plant where it's been limited to about 25 percent degradation.

MEMBER POWERS: I don't contest your conclusion here, but it's -- I'm not understanding exactly how you arrive at a CDF and a LERF analysis here. You seem to be dependent on fairly large LOCAs

to get an incremental risk. And so I'm wondering how you concluded that those LOCAs were worse than ordinary LOCAs, LOCAs without this problem? How did you do that?

MR. POEHLER: So we have C.J. Fong here from the Division of Risk Assessment. I think he wants to speak to this question.

MR. FONG: Sure. Thanks, Jeff.

Hi, Dr. Powers. This is C.J. Fong at NRR Division of Risk Assessment. I've got Steve Laur on the phone who performed the detail risk analysis. If we really want to get into the nuts and bolts, I'll ask that we patch him in, but I think I can at least take a shot at the initial question.

What Steve did was he made the very

conservative assumption that any initiating event with the capability of imposing additional load on the baffle-former bolts could deflect them such that core flow would be blocked.

Now Mr. Wilson at Westinghouse pointed out, we think that's probably not the case, but just as a first cut, we assigned a conditional core damage probability of one to any event that could deflect or significantly deform the baffle-former plates.

And so it turned out that it was large and medium LOCAs. And what Steve did was he looked at both your kind of traditional LOCA cost by long-term material degradation and also LOCA that was induced by seismic event. And as Jeff pointed out, we used the weight of seismic curves and fragilities and also Steve used the LOCA frequencies from NUREG-1829.

MEMBER POWERS: Thank you. That certainly illuminates what you did. It's certainly a conservative approach. I'm struggling mightily to understand how much of a public health and safety risk this particular issues covers it. And that kind of very conservative analysis is useful, I think, especially if we're going through your

process. It's not unreasonable, but it's not illuminating for me with a realistic risk to public health and safety is here.

I hope somebody can explore that a little for me.

MS. ROSS-LEE: This is M.J. Ross-Lee, NRR. So I think that based on the somewhat bounding, perhaps conservative analysis that was done, as well as other criteria that we looked at, for instance, when you look at the different options, we balance things like transience or burden if we were to make them shut down as well as well, if we don't do anything.

And I believe that considering the realistic risk to the public is what led us to pick the option that we did which is to have them do inspections at a more frequent interval or sooner, so everybody is committed to doing inspections at their next outage at the most susceptible plant.

So I think we've tempered what the actual numbers ran versus the other knowledge that we have to feel comfortable that the risk to the public is such that it is acceptable to do Option 2 which is waiting for the next outage. At that point in time, perhaps based on those results that

are gleaned, we would maybe have to relook at our LIC-504, our choice in options.

So to quantify the realistic risk if that's what you're looking for as far as a number, we haven't done that. I do believe that we have put a number of factors together to believe is one, we don't need to shut them down right now, but Option 4, we're not going to ignore this and just wait and see what happens. So we picked what we thought was the realistic option and that is to have them move up the most susceptible plants in the initial tier to move up their inspections all of which will be completed by I believe the end of 2017.

MEMBER POWERS: Well, I think I understand what you've done for now. What I am wondering is what do you do in the future? Is this a problem that we can say okay, problem resolved, licensees, your problem or is it a continuing risk to the public health and safety? I'm not getting an understanding of why it's a continuing risk to the public health and safety.

MR. FONG: Dr. Powers, I think the way I would respond to that is that we really have two separate risk-informed processes. LIC-504 is for

emerging conditions. Do we have an immediate safety issue where we need to have plant shutdown right away? We feel the analysis we performed, while conservative, was certainly sufficient to support that decision.

Down the road, if there's a different question, for example, should a licensee be allowed to just live with this for the life of the plant, that would be maybe a Reg. Guide 1.174 decision which has different acceptance guidelines and it's a different process.

So I think we used the right tool for this decision. I certainly understand what you're saying down the road, if there's a different issue or different question that we're trying to answer, we can enter a different process, like a license amendment using Reg. Guide 1.174.

MEMBER POWERS: Thank you. I understand exactly what you've done. My question is a little different.

MS. ROSS-LEE: Sorry, this is M.J. Ross-Lee. So I believe at this point in answer to your question have we reached a decision that we're just going to let industry do what they want with this, I would say the answer to that is no.

Again, this tool gave us an answer for now. It had a recommendation that they need to move up the inspections. We're going to look at those inspections as an agency. We'll look at what the results of those are. We're staying very plugged in with the root cause analysis. That will be looked at. At each of those steps along the way, we'll have to look at what our agency response is. We'll have to look at is the current aging management guidance in MRP-227A acceptable or do we need to, in fact, change the inspection frequency in there because every ten years isn't the right answer. So I certainly don't feel that at any point we would just make this an industry issue or problem. We'll continue to stay engaged and we'll look at the tools that we have to stay engaged and perhaps make changes as necessary, based on the results that we get.

We have a couple of data points now. We're aware of those. We'll get more data points as we get inspections coming forward over the next year. If our position on the safety significance of this would change, we would react at that time.

MEMBER POWERS: The problem is I really don't understand the safety significance here. I

don't think you've outlined it. You've done a very conservative analysis and it's probably one I would have done to satisfy your immediate need. But to go forward and say what beyond this, I think I'd need a little more definitive, a little more realistic kinds of risk analysis to tell me how much effort I'm going to expend on this particular issue from a regulatory perspective.

MS. ROSS-LEE: Noted. I know that at the next point which we would have information to inform any sort of analysis would be following perhaps some of the bolt testing, some of the material information which currently we don't have. And we would have additional information from follow-on inspections that could --

MEMBER POWERS: I'm not sure that's your most formidable problem in getting to a realistic risk assessment. I think your most formidable problem is defending the idea you've got LOCA codes that can handle realistic deflections of the baffle-former plate to say there's any incremental risk associated with that. I think that's the challenge you're going to face is that you're burdened because of the limitations of your existing analytical capabilities to making pretty

Draconian conservative assumptions here. And I'm suspect of those. I think you'd have to come in and put a pretty stout defense on what you've done in your accident analysis tools. I don't know how you do that. I don't think you have the data to sustain an argument that there's a substantial risk to the public health and safety from this.

MR. FONG: Dr. Powers, I think I would, in general, agree with that. Risk, of course, if frequency times consequence. I think we've got a handle on the frequency of the different initiating events. We recognize there's some uncertainty there.

As far as the consequence goes, we had to make a very conservative assumption that hey, if the LOCA happens the baffle plates somehow enter a geometry where they're blocking flow completely and the core goes to core damage. I don't think that's really 100 percent chance of that happening, but right now don't have the information to assign a more realistic value.

As M.J. pointed out, we're gathering data. There's destructive engineering going on of bolts and things like that. We might, in the future, have a better way to model the consequences

as part of the risk equation, but right now --

MEMBER POWERS: I am not at all criticizing what you've done here. I am saying that you can research the bolts until the cows come home. You're not going to answer the real question on what's the risk to public health and safety from this particular event. That's not where the real technical issue is going to be. It is precisely the approximation that you found yourself forced to make to do this that's going to be the real question.

MS. ROSS-LEE: We are way behind. If you could try to move through. I think we've probably touched on this about as much as what information we can provide and then I think go on to the last slide.

MR. POEHLER: Yes. So second to the last slide. So just to summarize the NRC's evaluation of the recent operating experience. Our preliminary conclusion is that it's the Westinghouse four-loop design downflow plants with Type 347 bolts that are more susceptible to baffle-former bolt degradation than other PWR designs.

There are seven plants in that group which are listed here. Also, EPRI is going to talk

about -- industry is going to talk about this a lot more, but the EPRI and Materials Reliability Program has issued interim guidance that calls for UT inspection of all baffle-former bolts at the next refueling outage for these plants, these seven plants which they call Tier 1A.

The NRC is also monitoring inspections and other actions at all these plants. We feel that the immediate safety concern for these plants is addressed by our LIC-504 evaluation.

Finally, future activities for NRC, we're following the root cause investigation at D.C. Cook 2, focusing on the cause of degradation in the replacement bolts and also the baffle-edge bolts. We're going to look at the LIC-504 to determine if it needs to be revised based on the new developments at D.C. Cook Unit 2.

We are going to continue to engage with the industry, especially on the root cause for the three plants. We're going to discuss with the industry if they need to make changes to the interim guidance.

We plan to develop an information notice. We are going to document our staff assessment of the MRP's interim guidance. And

finally, we're going to determine if changes to the overall MRP-227-A guidance for baffle-former bolt inspections are needed.

So that concludes the NRC's presentation. If there are any further questions?

MEMBER KIRCHNER: I have a question getting at root cause. Do you have maps of each of the plants and where these baffle-bolt failures occurred like the nice presentation you gave us was Cook?

MR. POEHLER: Yes, we do.

MEMBER KIRCHNER: Have you looked at these patterns and do you see any repeats in terms of where you're finding clusters of bolt failures?

MR. POEHLER: I mean I think not necessarily exactly the same patterns are seen in all the plants. I don't know that we've done a detailed analysis of that.

MS. ROSS-LEE: This is M.J. Ross-Lee. I would say at least at this point we haven't -- there's not an obvious correlation or similarity in those bolting patterns that we've been able to determine.

MEMBER KIRCHNER: Thank you.

CHAIRMAN SUNSERI: All right, at this

time, thank you for the presentation. Appreciate the information you provided.

At this time, we're going to take a 15-minute break. We will resume with the industry presentations at 20 'til based on that clock.

(Whereupon, the above-entitled matter went off the record at 10:25 a.m. and resumed at 10:40 a.m.)

CHAIRMAN SUNSERI: All right, we're going to reconvene the Metallurgy Subcommittee briefing here. We'll begin with the industry presentation and Bernie Rudell is going to introduce the team and start us off.

Bernie?

MR. RUDELL: Yes. Thank you very much. It's a pleasure to be here with the Advisory Committee to share our experience so far with regards to this issue in the pressurized water reactor industry, specifically with the baffle-former bolt issues.

We have Heather Malikowski here today. She's the chairman of the PWR Owners Group Material Subcommittee. And Bryan Wilson, a member of that committee as well, but he represents Westinghouse and also a lot of the analysis that Westinghouse

has done for both the technical bulletin back in 2012 and now additional analysis that's been done. And you'll hear about that in this presentation.

I also have on the line, I believe, Steve Fifitch. Are you there, Steve? He's there and if we need so, he can jump in. He's representing AREVA, also a member of the PWR Owners Group.

My name is Bernie Rudell. I'm the chairman of the Integration Committee of the EPRI Materials Reliability Program and we have the lead as an issue program under NEI-0308, the Materials Degradation Management Program for the PWR reactor vessel internals.

So we have representatives from both the Owners Group and MRP and some PWR suppliers here to present information.

Jeff did such an excellent job and I read through his draft slides the other day and I said there wasn't much left for us to present. So hopefully, we can go through our slides and just hit maybe some of the highlights where we heard some questions and elaborate on those and you can jump in with additional questions if they come to mind. And we can give you the information that we

know there or get back to you at a later time, for example, with the 50.59 evaluation question that we had earlier.

As I mentioned, baffle-former bolt, we'll call it BFB, examination is a subset of the PWR reactor vessel internals Aging Management Program and that's really under the NUREG-1801 GALL or General Aging Lessons Learned Program and the NEI-0308 Guidelines for Management of Material Issues and Inspections and Evaluation. That guidance is prescribed in, as Jeff pointed out, our MRP 227 and it's got the -A now because it also includes the NRC's safety evaluation which accepts that guidance on PWR internals and inspections.

This spring at two PWRs, a large percentage of baffle-former bolts failed examination that had typically been experienced before. Before we were seeing one to five, and maybe an occasional near ten percent failure of baffle-former bolts.

Fortunately, there's a lot of margin in the number of baffle-former bolts in these designs and that margin varies, dependent on the design and it depends on that particular station's LOCA analysis and so forth as well. And the leak before

break that has been adopted for that unit because it's typically the LOCA and seismic event that governs the minimum bolt pattern analysis which, as Jeff pointed out, is also performed by NRC safety evaluation accepted methodologies.

Although there's margin in the number of required bolts from what is there originally, these findings did trigger a substantial safety hazard look-see under 10 CFR 21. And it turned out that these findings do not constitute a substantial safety hazard status under 10 CFR 21. The large percentage did not meet the acceptance criteria though in our WCAP 17096 that Jeff also alluded to. And in these cases, the distribution of failures have caused further attention to the topic, particularly this clustering effect that we see.

In response to this experience, MRP invoked the protocol for potential generic material issue and that protocol is prescribed under NEI 03-08. Westinghouse has issued a Nuclear Safety Advisory Letter, NSAL. And AREVA has issued a Customer Service Notice. The PWR and Material Reliability Program formed a joint baffle-former bolt focus group and have issued NEI 03-08 interim guidance that was approved by the PWR Materials

Management Program which is our executive branch, if you will, under the NEI 03-08 issue program for PWRs.

We've got a lot of attention to this. The entire EPRI year emergent funds for emergent issues has been allocated to the baffle-former bolt work that's going on. We've redirected efforts from lower priority work that we were doing to the baffle-former bolt. There's a lot of research involved here going on and both where it fits under our appropriate jurisdiction in the PWR owners group area and the EPRI MRP area and then in some cases to the specific licensee and work they're having done to go through and address programmatically what they need to do as well.

So the guidance, as I mentioned, is approved. A lot of that guidance so far as matched up almost one for one with the NSAL, but the guidance carries the NEI 03-08 needed requirements there so that we would be informed as would the NRC if there was any deviation from that guidance. And for example, all of these Tier 1A plants have inspection scheduled now in their upcoming refueling outage.

Communications were had. We've had

several face-to-face meetings. In fact, we had a meeting with the focus group Monday this week in San Antonio and there were about 70 people that attended that meeting. So we're very interested in this subject and getting it behind us as an industry. And I think we're doing all of this in the spirit of the GALL and the fact that this is a living program. We're going to learn. We're going to look at what appears to be the material degradation areas that may pop up and adjust from the results that we see. And in this case, I believe we see a few plants, we hope limited to a particular subgroup because and that's what it appears to be. We'll show you evidence of that, that we can get through and then resume to an inspection and monitoring and replacement program that will not cause us to hit unacceptable conditions of the results in the future.

MEMBER POWERS: Is the apparent confinement to a subset of plants just an accident of time? That is, if I go out another 20 years, then I'll find a broadened subset?

MR. RUDELL: That's a good question. And our research is looking into that question. Right now, today, where we are in time it appears

to be a particular subset. But the information that we get from a lot of this evaluation that we're doing and we have a lot of very smart people working on it will tell us do we need to change our inspection on the other plants going forward? In fact, that's what we'll be looking for in the next set, perhaps, of interim guidance that will come out. But that's a very good question.

MEMBER RICCARDELLA: To be clear, inspections are planned under MRP 227 for all the plants?

MR. RUDELL: That's correct.

MEMBER RICCARDELLA: It's just the timing of them and whether you accelerate them.

MR. RUDELL: That's correct.

MEMBER RICCARDELLA: Thank you.

MEMBER KIRCHNER: Bernie, to frame the issue again, just you have only to date found this problem in the PWRs with downflow in the baffle region, is that a fair assessment?

MR. RUDELL: The Tier 1 Alpha plants are the four-loop design Westinghouse plants that have operated and continually -- and are continuing to operate in a downflow condition.

MEMBER KIRCHNER: Right, so those other

Westinghouse plants or other PWRS with upflow are not experiencing this problem based on outages and inspection?

MR. RUDELL: That's correct.

MEMBER KIRCHNER: Okay, so you describe the --

MR. RUDELL: Based on the inspections that have been performed.

MEMBER KIRCHNER: -- root cause as a metallurgy and materials issue. Is it not a fluid hydraulic design issue with the plants in Tier 1? And if you were to change out those plants to downflow in the baffle region, would you essentially solve this problem?

I think replacing bolts is less expensive than changing flow configuration for the plant, but I'm just -- getting at the root cause may not be materials because as Dana might have suggested will you see this with further aging in other plants or is it really a fluid hydraulic phenomenon problem that's inducing this?

MR. RUDELL: Well, can we hold off on answering that response because I think there's a lot of things at play here and what's leading and what's lagging may even vary from one plant, a

model, to another plant model as well.

Let me check -- so just in summary, we have MRP 227 is the leading guidance that we have now. We know that baffle-former bolts is one of the primary exams, the degradation mechanism under that, and the bases that back the creation of that guidance indicate that it irradiated assisted stress corrosion cracking and fatigue. Both of those degradation mechanisms are susceptible to occurring in baffle-former bolts.

We have expansion criteria that we believe some of these plants have triggered and they'll need to enter into the expansion examination. The examination for baffle-former bolts is 100 percent of the accessible baffle-former bolts. And there's a prescribed time line that we're changing with some of the interim guidance now.

The UT that we perform is generally capable of detecting large cracks on the order of 30 percent of the volume. And as Jeff mentioned, the acceptance criteria is the WCAP 17096-NP-A approved and the minimum bolt pattern WCAP.

So without further ado, I'm going to turn it over to Heather, and we'll go through our

presentations.

MS. MALIKOWSKI: Thanks Bernie. Thank you, Bernie. This is Heather Malikowski. I am the chair of the PWR Owners Group Materials Committee. You'll see a lot of parallels in our slides with what Mr. Poehler presented, so I will try and just highlight certain points on some of them, but otherwise, unless there are specific questions, there's something you want to go back to on those particular figures, I'm going to move relatively rapidly.

As we discussed, I think we're pretty much grounded on the configuration of the assembly and this is where the baffle-former bolts are located. And as we've discussed, the material differences are there, mostly more correlated also to the design of the bolts amongst the different NSSS designs and the number of loops per plant. We said the shank lengths do vary depending on the design. And Bryan Wilson will probably go through some more details on that, how it affected our analyses.

We discussed the difference, obviously, downflow and upflow configuration is a big part of this discussion and so we'll continue to, as I go

through, I'll just point out we do try and document where for the plant inspections that have happened, which plants are downflow versus upflow. We don't have all the timing, say for the ones that have converted in all cases whether their exams were prior to conversion or not. If there's interest in that, we can follow up.

For your reference, I think this is a good time line to show. On the top it just gives the operating experience where we started from back in 1989 with the French OE and coming through to the present with our more recent findings at Indian Point, Cook, and Salem.

Down at the bottom it shows some of the guidance that's been issued by the industry including the NRC. We actually did -- I just want to basically show that when the response to the EDF OE in the late '80s, we did respond and review the issue, looking for commonalities to the fleet, comparisons of many parameters. At the time, Westinghouse Owners Group put together comparisons of all the different two, three, and four-loop designs at the Westinghouse plants, trying to understand relative risks in all of that to help inform future guidance and recommendations for the

industry.

So as you'll see, we get up to the late '90s, that's when the pilot inspections that Jeff referenced earlier were performed by a voluntary selection of similar plants to the EDF plants to get some additional information on the impact to the U.S. fleet.

And you'll see that it looks like there's kind of a large gap there where nothing happened, but that is really not the case and I'll kind of explain that there was ongoing work during that time until between when we did inspections in the late '90s until the MRP 227 industry guidance came out.

MR. RUDELL: You can see this is almost one of the first issues that the MRP worked on. MRP-03 back in 1990 addressed the baffle-former bolt OE that the French saw. And that inspired the Owners Group to do a lot of inspections in 1990-2000 time frame that you'll see as well in our slides a little later.

CHAIRMAN SUNSERI: So it is a little curious that the French plants and the comparison to the U.S. plants, the similarities must be more than just the design. Were they the same fluence

and age of materials? So why were the French having the problem and U.S. plants weren't, I guess is my question.

MS. MALIKOWSKI: So moving up to our past operating experience, the slide here does have some summary of the EDF experience from the '80s. They are a three-loop essentially Westinghouse design. They do have definite design differences. I don't know if anyone wants to elaborate on them, but they are even within the CPO design as was mentioned earlier. There are differences amongst those design plants. So the plot here that EDF shows is their inspection history over the last several years, they use operating hours as their nomenclature on the bottom for time. So it's a little different than EFPY for us. So I don't have the exact EFPY numbers.

But it basically shows that these are all CP0 plants shown, but they definitely have a very small trend of finding any baffle-bolt failures on was it Bugey 4 and 5 versus the Fessenheim or the other Bugey units. What it shows is that even though they did convert to an upflow configuration, they do have -- they do continue to find bolt failures, but not significant numbers and

not -- I don't believe and the interest in the OE is not necessarily clustered per se, but they are continuing to get failures and it is generally focused towards the high fluence regimes, the areas down lower in the baffle region. So they're basically seeing a continuing trend of failures, but they're not seeing an accelerated trend because they are doing replacements as they go along and continuing a replacement in the inspection campaign. So now they basically are tracking their inspection requirements with an anticipated five bolts per year failure rate and really looking at more as an irradiated effect versus now we have the extreme, the stresses have been lowered.

As we kind of discussed earlier, they do have this 316 material, but obviously that was a difference, but I think we are saying that's not necessarily a big indicator that that's going to fail quicker, because obviously our domestic experience doesn't show that.

And so other than that, I think -- other than we know there's operational differences of load following versus our more base load operation, but it's not clear the magnitude of impact they have on their failures. That's

something that's being thought of as maybe a contributor to their seeing the accelerated failure, but I don't think we have a quantification of that.

MEMBER RICCARDELLA: Heather, when the French replace, are they replacing with 347 or 316?

MS. MALIKOWSKI: 316. And they have replaced some replacement bolts, but we did confirm that was not for -- because of failure. It was actually just to continue to maintain the minimum pattern and to continue to keep the overall age of the bolts at a low amount.

MEMBER RICCARDELLA: Thank you.

MR. RUDELL: These are 316 originally.

MEMBER RICCARDELLA: Oh.

MR. RUDELL: Yes, all the French are 316 original baffle-former bolts.

MEMBER RICCARDELLA: Thank you.

MEMBER SKILLMAN: Heather, are the Bugey 4 and 5, the blue and the yellow, the only plants that are the three-loop upflows on this image?

MS. MALIKOWSKI: I believe they're all converted to upflow. I do not know if 4 and 5 are original upflow or not. Do you know, Bryan?

MR. RUDELL: They are all converted and there is not a significant difference between Bugey 4 and Bugey 2, as I understand it, other than perhaps in the time of life that it got converted, but as far as I understand and as with a couple of presentations from EDF and their experience and the best information they brought with us, they were all converted basically in the same year time frame or within a few years and they don't have a big explanation as to why some of these are unaffected by this issue and some are.

Their rate going forward and their philosophy going forward has been approximately five bolts per year, their failure rate. And their inspections and replacement pattern have been to get to a point where they can go to an extended ten-year interval between inspection and replacement campaigns.

But basically, they've been using a straight five bolt per year for original bolts' failure rate, comparing it to their minimum bolt design analysis and the replacement bolts they also have a lower failure rate that they work with. And are even replacing replacement bolts in the future.

MEMBER RICCARDELLA: On the Bugey 2

curve, where approximately on that curve was the change from downflow to upflow?

MS. MALIKOWSKI: I knew you were going to ask. I can't say. Unfortunately, I'd have to follow up to find where they're at on the curve as far as the conversion time frame. I don't have that number.

MEMBER RICCARDELLA: Thank you.

MEMBER SKILLMAN: It seems like that's a key question during this whole presentation because it suggests that Bugey 4 and 5 are running almost flat and after 220,000 hours, just a couple of indications whereas the other plants continue.

MR. RUDELL: But they're going through large replacement campaigns in this time line also. So that's where you need to have that piece also.

MS. MALIKOWSKI: And it may be a factor of when they get their replacements done. Did they do it prior to large failures starting?

MEMBER BALLINGER: Again, these are all 316 bolts?

MR. RUDELL: Yes, correct. 316 and Monday, they said they're 316 cold work.

MEMBER RICCARDELLA: But when you replace, you start to clock over again.

MR. RUDELL: Yes.

MS. MALIKOWSKI: And as far as our domestic experience, does help give some information as far as kind of helping correlate that yes, as far as upflow and downflow. And I'll explain that on another slide here.

Just to summarize, as we said, back in the late '90s, early 2000, we did have a Joint Owners Group Program to do voluntary inspections in relation to the Bugey OE. And so the selective plants, they were Ginna, Point Beach, Farley 1 and 2. And as you note, from the results we note here, and these were done basically because we're trying to look for plants that were similar -- have the longest operating time, so they tend to be the two- and three-loop plants. And we're looking at similar plants to Bugey, so there being three-loop that we focused on those plants. We do not have a four-loop inspection, but those tend to be -- they were the younger plants and a different design. So at the time, these were the plants that we did inspect.

There were some UT indications found, but as we note here for Ginna we did actually do a metallurgical examination of 14 bolts after

removal. We did do a replacement at that station and there were no cracking found. So there at least is some -- I believe we have some over calls on our examination results which at least is conservative so we've kind of bounded our -- the concern there. But at the same time shows that we're not seeing, we weren't seeing huge amount of cracking in our fleet.

And you'll also note here, we did have proactive replacements also done at the Farley units even though they did not have any UT indications. Basically, because we saw very little indications from these examinations, that's where we were led to the conclusion of all right, this is not a current licensing period of concern for our fleet. We do need to develop an aging management guidance for license renewal and that's really where the industry transitioned to was all right, we need to develop that kind of guidance. So that's where from this point on, we did say all right, we need to develop inspection guides, but we weren't then recommending further plant inspections other than their normal in-vessel exams during refueling for ISI.

And then to show some of the more

recent inspection results now that we do have MRP 227 as our inspection requirement which does, as Jeff had mentioned require UT inspection of 100 percent in 25 through 35 EFPY for the Westinghouse NSSS.

The two-loop plants' examination results are on the left-hand column. The three-loop plants are on the right. You can see that most of them are downflow configuration. The Point Beach units are upflow converted. So we do have some second inspection results on the two-loop side and we have new inspections on the three-loop side to show there really are very low amounts of inspection indications being found and we are seeing at least some amount of correlation that at least to this point we're not seeing a significant degradation effect. You can also see we do have reference here where most of them are 347 stainless steel in these plants.

For the other NSSSs, B&W and also some international results, we're obviously talking about EDF, but there are other results from other utilities shown there. Also, similarly, very low on inspection findings, and we do have actually several multiple -- the third inspection has been

done, just completed in August at Ringhals Unit 3 and finding a very small number of indications.

And note, we kind of mentioned this briefly earlier, the B&W NSSS, their design is an original upflow configuration. There's also many slots and flow holes in the baffle plate, so there's a lot of reduction in differential pressure there. And as noted, there really was only one crack-like indication detected in the four units there, the three Oconee units, and actually Crystal River. It was mostly inaccessibility they called it, just saying because we couldn't inspect it, that was the only other degradation fact except for one bolt. So giving some a little credence to the upflow design having some benefit there. And also, as we note, the CE bolted design is also upflow as the original design as well. So that's why we've kind of seen -- the focus has been more to the Westinghouse NSSS.

So just the broader OE, what we've seen up until this spring, we basically noted that the international OE and our domestic OE has really reinforced that the failures are expected to be IASEC with a random distribution. We did not see any clustering or ready focused failures in one

location or the other. They were bounded by historic safety assessments we did the mid- to late '90s. So we had some confidence that we were within the bounds of analyses already performed. And we did actually have quite a bit of replacement campaigns executed in the utilities and internationally and domestically that has seemed to have been helping maintain those plants in going forward.

So at least the experience up to this point was we were seeing no major trends other than what we would expect as a standard degradation mechanism that would proceed randomly in the assembly.

MEMBER KIRCHNER: So from this slide would one conclude that downflow is the root cause for the larger failure rate in the plants that you're going to discuss next?

MS. MALIKOWSKI: I would say it's part of it, but we will discuss that.

MR. RUDELL: The oldest operating PWR in the U.S. is a downflow plant, a two-loop plant. And that was one of the plants that was inspected in 1999. It got 100 percent inspection. Now there was some inaccessible because of the lock and weld

rollover. One hundred percent inspection with a UT technique that was at that time the best we had.

We found about 60 bolts. It's a 700 and some odd bolt baffle-former design and they found about 60 bolts with UT indications. They went to a replacement campaign. They were able to replace 56 of those. During the 56 bolts' removal at that time, 5 broke and the others didn't. They took two of the broken bolts and 14 of the unbroken bolts that came out that had UT indications and sent them off to a hot cell for extensive destructive examination.

One of the broken bolt fracture surfaces was too destroyed by the removal operation to get any valuable information out. The other one showed 100 percent virtually intergranular, like irradiated assisted stress corrosion cracking fractography features.

The other 14, they were UT'd in the shop with the shear wave on the side of the bolt and they thought they saw indications in those bolts. The other 14 then were ultimately PT, fluorescent PT examination and they saw no indications in those bolts. Of the 14, 2 were cut up to do some tensile specimens and the other 12

were pulled to failure. No PT indications, pulled to failure, and they showed full strength expected for their irradiated strength and basically they were -- I'll just use the word flawless. So we from that found that the sample out of 14 bolts that came out whole, that had indications, all 14 were flawless.

Another plant at the time did a lot of pull tests and in the pull at that time when they were pulling, removing bolts, and they all came out showing full strength. So here we are with a downflow unit. I'm just answering your question that as the oldest plant in the United States and the last inspection, by the way, where we didn't do 100 percent inspection, but we inspected about 125 bolts. We had one bolt with an indication out of 100. So that's less than one percent. So it almost looks like we had -- although this slide said 7 percent or something like that, maybe really was 1 percent back in 1999 in that unit and in 2011, when we looked at it under the MRP 227 program, it looked like 1 percent failure at that downflow plant. So it's not downflow necessarily, I think.

MEMBER POWERS: Aside from the focus of

your presentation, does it appear that our inspection technique is susceptible to a high level of false positives?

MR. RUDELL: Well, we're learning more. Of the bolts, we have about 30 bolts. I'm just going to talk round numbers. We have about 30 bolts in the hot cell now out of Indian Point 2 and half a dozen or so out of Salem and we took six more of the bolts out of -- that were in the Ginna spent fuel pool that were removed in 2011 and they're in the hot cell also now.

We've only done of the bolts that were removed intact from Salem and Indian Point, we've only looked at four so far and it looks like those four are flawless. So that's four of the red bolts that appeared to note.

Now, on the other hand, I can't say that without saying this. There were some green bolts that were removed at one unit and they broke so we got some of those also in the lab and we'll look at those.

We know that UT is difficult. We know that -- you need to certainly assume a ten percent. Probably in most of our assumptions, we're assuming a 20 percent probability of a detection being

missed or a false call. So there is -- some of the effort needs to go into understanding what's the characteristic of these failures that are making NDE not as accurate as we would like it to be as well. And that's another element of our focus group.

MEMBER RICCARDELLA: Was there a significant improvement in the inspections from -- you know, the Ginna was like 1999, 2000 versus what we're doing now 15 years later?

MR. RUDELL: I'd have to say yes. We have now MRP-228 which governs some of the inspection demonstration and qualification. The UT on bolts, I understand, it's not an Appendix 8 PDI qualification, but it's a low rigor qualification one might say demonstrated. And we actually have made a change going forward to increase the rigor slightly of that.

MEMBER RICCARDELLA: But your initial hot cell data at the Indian Point post is not --

MR. RUDELL: But I'm sure we're going to find some of them failed, but I'm just saying we're only four so far.

MEMBER RICCARDELLA: Yes.

MS. MALIKOWSKI: And I think you saw

the figure, the picture that Jeff had in his earlier presentation. This particular configuration bolt was very difficult to get an examination, a quality examination done. It's not -- it's probably one of the most challenge of all of the configurations we have. So unfortunately, they do what they can to improve on the technique, but it does still come down to being it's a challenge to get a sound bounced back and forth and read reliably so. I think calling it a little conservatively has been the preference at least to make sure that we're not missing things.

MEMBER RICCARDELLA: I understand. Thank you.

CHAIRMAN SUNSERI: Based on the amount of time that we have left today, I'm going to ask you to prioritize your remarks now and make sure you tell us the most important things that you want us to hear. Okay?

MS. MALIKOWSKI: I understand.

MEMBER RICCARDELLA: Maybe your lecture should be to the committee and not to the presenters.

(Laughter.)

CHAIRMAN SUNSERI: You guys are doing

good.

MS. MALIKOWSKI: That's fine. I mean I think we've discussed the reason we already as far as to just more for your reference we do have the EFPY counts at the time of these inspections. This was the 2010 OE from Cook 2.

As relating to UT, they did not do a UT examination that time because they did not have a qualified method and because it was one of these challenging configurations. So they did what they could with the availability at the time.

But I think to just jump to the conclusions as we've been discussing, as you said, the recent OE has been focused on -- and you said not just downflow, but four-loop downflow as it was kind of alluded to earlier, the larger reactor design. They do have less bolts than say the three-loop design has over a thousand. It's a little over 800 for the four-loop. And then this two has over 700. So there are design differences that we see or design commonalities amongst the most recent OE that do tend to give us some reason to believe there's some commonalities to cause the clustering that we've been discussing.

And we will, as we said, the most

recent Cook 2 UT results and their failures observed, we are as and Bryan will say, we're following the industry to understand them. Although I would say as you have the map in front of you, it's not necessarily completely surprising with that sea of red surrounding those replacement bolts that cause the failures not to surprise them, so to speak, but we will learn more from the --

MEMBER RICCARDELLA: I assume some of those are going to be examined in the hot cell as well, right?

MS. MALIKOWSKI: They are definitely taking replacement bolts, failed replacement bolts for analysis. Yes, this is the question of the edge bolts, what they're going to do with them. But yes, we want to learn from that.

MEMBER KIRCHNER: Just based on the prior presentation, it looks like for Cook in that time period that -- I hate to put precision on this, but 6 bolts that had been replaced out of about 50, so about 10 percent of the bolts were cracked or indicated some problem 6 years later. You changed from 345 to 316. So it seems to me going back to my issue whether this is a design problem that is more fluid mechanics, stress

induced than it is a materials problem.

MR. RUDELL: And here's a theory --

CHAIRMAN SUNSERI: Please remember to introduce yourself.

MR. RUDELL: Bernie Rudell.

CHAIRMAN SUNSERI: Thank yous.

MR. RUDELL: Here's a theory that goes with the new bolt failures. It's basically -- I'll sum it up, it's collateral damage associated with the number of failed original bolts in the vicinity of that -- of those new bolts and the flexure of that plate. Now we probably will see a different mode of failure to those new bolts when we look at to prove our theory. And the same thing goes with the edge bolts we saw there. We believe that's collateral damage from that.

MEMBER KIRCHNER: Being formally from Los Alamos, I would recommend a different phraseology than collateral damage.

(Laughter.)

MEMBER RICCARDELLA: You know, I think a significant point is that in 2010 when those replacements were done, there was no ultrasonic examinations done. And so they replaced some bolts based on visual, but a lot of those other red spots

around them might have been there at that time.

MEMBER BALLINGER: The replacement bolts, did they have the same head design?

MR. RUDELL: Yes. Well, I would say current replacement. Not the same as the originals.

MEMBER BALLINGER: These are the modified, supposedly less smaller stress concentration?

MR. WILSON: Yes.

MS. MALIKOWSKI: Bryan has a picture of that.

MR. WILSON: Yes, I have a picture of that.

MS. MALIKOWSKI: So in order to get to Bryan's discussion, just to pictorially show since we don't show necessarily the EFPY for all the other plants, and he'll describe as we kind of already mentioned Tier 1 related to Westinghouse Nuclear Safety Advisory Letter.

The inspections that have been done so far domestically are kind of applied here and Bernie will say we are developing an OE database to help look and try and find any other trends amongst various parameters. And this is just showing,

based on the inspection timing what the number -- it says "assume degraded bolts" because they are basically taking anything they couldn't inspect and assuming it degraded just for the purposes here to conservatively call them. But as you can see, for at least the inspection guidance, other than our four-loop downflow Tier 1 plants, the inspection results do tend to show that we're finding -- I think we have an appropriate inspection regime for at least part of our population so while we are, as we'll discuss, investigating what we need to alter in our MRP guidance, there is at least some feeling that we have some reasonable guidance for part of the industry. So we're continuing to look at it and we'll move forward.

I'll turn this over to Bryan so he can continue into his discussion.

MR. WILSON: Hello, this is Bryan Wilson, Westinghouse. So I guess -- I'm going to have to hustle, so the point of my discussion here is really going to be to provide a little bit of explanation for some of the things I think Walter has been kind of alluding to is what are the mechanisms that are really leading off to causing this degradation. And then pair that with what's

the safety significance that we've evaluated as part of our safety evaluation and then how led into recommendations that were made through the NSAL to the industry.

So first is the factors influencing baffle-former bolt degradation. Jeff had covered a lot of these, so the high points really are that we've got bolts that are in a susceptible or an area or a region that's susceptible to IASCC. And that susceptibility is kind of common amongst various plants but what we're seeing in the OE is a differentiation from one plant to another. We're seeing much larger failures in one plant than another.

So the question is what's causing that differentiation? One theory is stress. Stress is the other contribution to propagating cracks. And there's a lot of different things that impact stress. I'd say some of the most important things are one, the stress relaxation. Stress relaxation is occurring in all of these baffle-former bolts or the majority of them.

MEMBER BALLINGER: That stress relaxation occurs most of it within five or so DPA?

MR. WILSON: Right.

MEMBER BALLINGER: So that's basically two years or thereabouts?

MR. WILSON: Right, so my point with that is that once you get rid of that, these external loads, externally-applied loads become important. If you were to have a tight joint, then we'd probably eliminate a lot of what we're talking about here. So that's one key point.

The other is bolt design differences. So head to shank transition radius, materials maybe not so much because of our lack of data to support that, but bolt length.

I think the question was asked earlier about what the bolt length, how that got segregated between the plants. A lot of the early plants did use 347 bolt with a shorter shank and so some, I'd say, not all of the two-loops. And all of the four-loop downflows and some of the other four-loops, there's not -- again, it's not across the board that all the four-loops use these shorter shanks. They started transitioning in later years to a longer shank. But I'd say the early plants were using these shorter shank bolts.

As you transition away, the bolt shanks were going from like an inch and a half long to two

to three inches long and stresses were coming way down as a result because a lot of the stresses in these bolts are a result of thermal expansion of the plates basically, relative temperature between the barrel and the plate itself, and baffle plate.

MEMBER RICCARDELLA: And swelling differences.

MR. WILSON: And swelling differences. Yes, there's a lot of other extraneous things, but I'd say nominal stresses on the bolts are a result of that.

The other factor influencing stress on these bolts is the load, the hydraulic load on the bolts. So this speaks to the downflow versus upflow in the plant design. So four-loops, three-loops, two-loops, the pressure across the plate varies say consistently with the amount of flow, total flow in the plant. So it's not an exact, for instance, a four-loop plant, 50 DPA versus three-loops 30 and two-loops 20, but it's that same kind of trend. So that's one contributor, right? So that naturally says four-loops generally have higher load.

MEMBER RICCARDELLA: One thing that surprised me is counter intuitive is that the

three-loop plants have more bolts than the four-loop plants?

MR. WILSON: That was exactly my next point. So if you look at the bolt distribution on a four-loop, versus a three-loop versus a two-loop, the order is that four-loop -- or two-loops actually have the least, the smallest plates, right? Followed by four-loops, followed by three-loops. Three-loops have considerably more bolts than a four-loop.

So in a bolt per plate load distribution, the four-loops generally see a significantly higher load than either the two or the three. So that also adds to the negative or depending on your perspective, deposited for identifying which ones are leading.

Other contributions are thermal, as I mentioned. There's a thermal gradient across the plate as well as a thermal gradient between the baffle plate and the core barrel which causes say more of a growth of the plate, relative growth difference. So those are -- keep that in mind, I guess.

Some other things that are affecting the failures or failure, say propagation is

clustering of bolts. So once one fails in a more highly susceptible plant that has say a high external load like a pressure differential, you have a better likelihood or a higher likelihood for load shed to the adjacent bolts. And then once you get that load shed, you're increasing your susceptibility to IASCC, you know, you're reducing the critical flaw size, right, for a bolt if you're looking at fracture mechanics base. So it all kind of starts to trend in one replacement.

So on the opposite of that is bolt replacement. Once you put bolt replacement, you're affecting the load distributions for all the different bolts that are in the system. So these all have to be considered. It's just that it's a very complex situation and a lot of things, different things going on.

But I think if you boil down what the deltas are between the plants and where we're seeing the high number of failures, I think what you're seeing is a trend towards the higher pressure, the plants with higher differential pressure which are downflow. And then four-loops which generally have a smaller number of bolts than say a three-loop or a two-loop. So those all seem

to trend in a direction that we're -- that is following the OE.

So what we're seeing from the OE is three different failure patterns, right? One is randomly distributed more IASEC is governing and you're getting kind of -- failures are occurring kind of in areas either you would expect because of the high fluence or they are say more well distributed and they can be representative from statistical evaluations like a Weibull distribution or something like that.

The next is dose related and this is more like what maybe EDF is seeing where the failures are say not necessarily cascading in nature, but they're more say focused on areas of high fluence, high load and maybe high amplitudes of like fatigue loads, for instance, if you're doing load follow.

And then you've got clustered which is kind of what we're seeing, I think, at Indian Point, Salem, Cook where you've got some failures that look like they're just spreading from a nucleus, right, and going out. And those are the ones I would say are the most concerning from a management -- a degradation management standpoint

because those are ones that you want to make sure you stay ahead of at the infantile stages so you don't let them expand.

So I can get into cluster --

CHAIRMAN SUNSERI: Let me interject here. I think based on the time, why don't you skip ahead to the consequences and evaluations of the baffle-bolt degradation because we kind of heard this story that you're about to tell and we might have more interest in some of the downstream topics. I'm kind of reading ahead in your presentation.

MR. WILSON: Excellent. That's fine. I appreciate the suggestion.

So as I alluded to previously, Westinghouse had conducted a safety evaluation basically of the degraded condition looking at an extreme condition, a condition at which was not -- did not have -- I said was well beyond the OE that was being experienced to look at what the potential of Part 21 reportability of this might be. And so for the condition that we evaluated, we looked at basically a quadrant of the baffle-former assembly with all the bolts, all the baffle-former bolts degraded. And the edge bolts were left. I would

say in some state, we didn't necessarily define at the time what degraded state those were in, but we acknowledged that there's some overlapping of plates that would create a simple supported edge condition and the edge bolts would be in some condition that they would also support that simply supported condition, whether they're all there or some random distribution of failures occurring there as well. So that was say the basis for the evaluation.

Things we looked at were impact on core bypass, control rod insertability as a result of plate deflections and impact with the fuel and fuel assembly grid crush and core coolability. So for the control rod insertability grid crush evaluations, we looked at a dynamic analysis where we basically took a loose plate or plate that only had simple supports.

Yes, Peter? Oh, I'm sorry. I thought you were raising your hand. Okay.

So we took a plate that was simply supported, imposed the pressure distributions of a LOCA on this plate and had models of the fuel assembly stacks, you know, basically rose and looked at how those fuel assemblies interacted with

one another and how they interacted with the plate to predict both what the peak deflections of the fuel assemblies were at control rod locations and also looked at what the grid deformation looked like.

In those evaluations, what we found was that the grid deformations were more I'd say cellular in nature, so in an individual grid cell unit you would have some sort of say small deformation, shifting, making it more like a parallelogram rather than a square. But the overall configuration geometry didn't change such that you would block flow. And the control spacings didn't say get tremendously closer, much closer to one another such that you would have concerns from departure from nucleic boiling and things like that. So that was in grid crush.

For control rod insertability we looked at maximum deflections of the control or of the fuel assembly after the event had occurred to see if we can get the rods in in a bowed configuration. And in that configuration, we also looked at thimble tube stresses or guide tube stresses I should say that -- to make sure those didn't exceed allowables for the fuel assembly

because maybe all bets are off after -- you know, if those fracture, right?

So those evaluations showed that the fuel did not go to an amount that would impact control rod insertability. We have test data, historical test data, where we looked at pulling fuel assemblies, you know, by a significant amount and inserting rods and found that for that deflection that was applied and I won't state it necessarily here, for that deflection that was applied the control rod insertability times were only impacted by .02 seconds and it was a considerable amount of deflection.

So what we found was all of these evaluations that we did assuming a loose baffle plate, the fuel assembly, say bowing or lateral deflection, didn't exceed those numbers that were say bound by the test data that we had. So we felt comfortable that in this extreme condition that the fuel assembly, even in an bowed state would be an acceptable level of bowed state and that the grids would remain in a condition that were acceptable for core cooling.

And so we looked then deeper into LOCA, non-LOCA impacts on core coolability and safe

shutdown margin and things like that. I would say all came out positive with ample margin on those. So the control rods would be able to fully insert at all locations. There would be some amount of grid deformation at peripheral assemblies as well as some inboard assemblies, but to a lesser degree and core coolability would largely remain unaffected.

MEMBER RICCARDELLA: In the analysis, the baffle plates actually impinge on the fuel rods?

MR. WILSON: Yes, they push in.

MEMBER RICCARDELLA: Push in enough so that your deflections are big enough.

MR. WILSON: Yes.

MEMBER RICCARDELLA: Okay.

MR. WILSON: The deflections are, I'd say for that kind of a plate configuration, it's long and narrow and thin, right? And it can push in a good bit.

MEMBER SKILLMAN: What did you use as the basis for the number of bolts that you did have retaining load?

MR. WILSON: So we had zero baffle-former bolts retaining load.

MR. RUDELL: In a full quadrant.

MR. WILSON: In a full quadrant.

MR. RUDELL: Not just an octant, but just one plate.

MEMBER SKILLMAN: Got it. Thanks.

MEMBER POWERS: From a hydraulic colleagues' part of my mind rigorous in their evaluation of such analyses, could you defend them before that?

MR. WILSON: Could I defend them before that?

MEMBER POWERS: Could you defend these calculations before -- from a hydraulics' community? That is, do you have enough experimental data to say that you adequately simulated?

MR. WILSON: I personally can't speak to that, but I believe that yes, we do have adequate data to support this.

CHAIRMAN SUNSERI: You had involvement by your thermal hydraulics course?

MR. WILSON: Yes.

CHAIRMAN SUNSERI: I imagine. Okay. So I really hate that --

MEMBER POWERS: What was the case? One

from a hydraulicist is a very pleasant fellow. Two is the problem.

(Laughter.)

CHAIRMAN SUNSERI: I hate to ask you to do this, but we do have another subcommittee meeting following this one, so we have to really stick to the schedule and I apologize for rushing you, so I'm going to give you five minutes to hit your main points because we're going to allow some public comment period here and I need to do that. We may need to have some more committee follow-up questions and that's going to take some time to get through. So I apologize for putting you in there, but we do have the slides, so we are able to read it. So conclude what your main point is here.

MR. WILSON: Right. So I'll skip past this, but a couple of other fuel-related things we did look at is baffle jetting and loose parts as well, but Jeff, I think, covered that rather well.

Now this led to communication to the industry. So the industry was informed of all of what's going on. All of our evaluations and I'd say a culmination of what I talked about before about likely cause of the issue, apparent cause of the issue and what's leading to more susceptibility

ranking of the plants.

We were able to say tier the plants to address plants at an early enough time frame or as soon as we can. The most susceptible ones were addressed first and then progressively provide recommendations as you decrease in susceptibility.

Let's see, so yes, so the intent was to promote early identification of failures, as I said. You want to find this as soon as possible to prevent any expansion of the clustering.

And then we did allow freedom in our NSAL recommendations for evaluating extent of condition. We recognize that we don't know everything at the beginning, so this is where MRP, I think, picks up in looking at this issue more holistically from the data that's being gathered so that kind of speaks to the bottom. Two points there that we're using lessons learned to further grow on this topic.

I trust that you guys have maybe all read the NSAL or have knowledge of the NSAL so I won't go through, but essentially the big topic is Tier 1 plants here that Tier 1A plants are doing UT inspections at the next refueling outage and they will all be completed by end of next year. So that

is the near term.

There's a Tier 1 Bravo and the differentiation here is really the head-to-shank radius and the material difference. It's, I'd say, prior experience. And so there's a noted, say potentially less susceptibility for this plant. So NSAL does say that those plants will need to do a visual inspection and the idea there is that all of the other plants, when they had significant degradation had been able to find this through a visual inspection. It's not an optimum inspection necessarily, but it will identify if you've got say large quantities of failures.

MEMBER RICCARDELLA: You've got seven in the first group, seven units in the first group.

MR. WILSON: Yes.

MEMBER RICCARDELLA: How many in the second group?

MR. WILSON: Two.

MEMBER RICCARDELLA: Oh, okay.

MR. WILSON: So then Tier 2, largely Tier 2 has been inspected. So our recognizing that, our guidance was really that those plants should go back, consider the OE that we have now and look at how that impacts what their inspection

results and how they reacted to the issue or reacted to their initial inspections. And then adjust accordingly. And MRP went further with the recommendations with this as well in NEI-308 space.

And then Tier 3 and Tier 4, there's acknowledgment that there's, especially for converted upflow plants as it could have been early damage as a result of being downflow, but that has maybe been say reduced or you know, say corrected in the conversion upflow. So there's a kind of a time limit. If you had been downflow for a long period of time, then you would need to maybe take action there.

MEMBER SKILLMAN: Bryan, how many Tier 3 plants are there, please?

MR. WILSON: Well, there's -- I don't have -- I think a large number of them are actually -- and there's really only one.

MS. ROSS-LEE: I think there's like three Tier 3 plants. There's more than three.

MR. RUDELL: It goes beyond four-loop. The first bullet there specifically, the four-loop set, but there's a lot of three-loops.

MEMBER SKILLMAN: That are converted upflow?

MR. RUDELL: Yes.

MEMBER SKILLMAN: Thank you.

MR. WILSON: So that's -- oh, wait, I don't want to skip my picture here. So here's the replacement bolt, right? So as we've discussed before, some of the -- the real design differences, the change in the material, but I'd say as far as improved susceptibility that one is -- I think the jury is still out on that one necessarily. The semi-parabolic head to shank transition fillet is really the key item there. And then the other changes or things you see on this bolt design are really related to install, ability to install the bolt and crimp it without welding to an irradiated baffle plate.

MR. RUDELL: And inspectability.

MR. WILSON: Right, and so the flat head here allows for ease of inspectability.

MEMBER RICCARDELLA: That looks like a relatively short shank. Is that prototypical?

MR. WILSON: Yes, that's standard for this short bolt. It's a really short shank.

MEMBER RICCARDELLA: Okay.

CHAIRMAN SUNSERI: All right, anything else?

MR. WILSON: Bernie, do you have anything to add?

MR. RUDELL: I think you summarized everything --

CHAIRMAN SUNSERI: Because we need to open up the phone lines here.

MR. RUDELL: I think I said everything that was in the other slides with regards to forming the focus group and working through that. That's in the rest of the slides.

CHAIRMAN SUNSERI: Okay, well, we appreciate your interaction with the committee. It's been really interesting.

So at this time, I'm going to ask for the phone line to be open and I hope that they are open. So if I could have somebody on the phone line at least speak something so we can confirm the phone line is open and once we confirm the phone line is open, we'll ask for comments. Is anybody out there?

MR. LEWIS: Marvin Lewis, a member of the public.

CHAIRMAN SUNSERI: Okay, great. So the phone lines are open. At this point, I will ask for any comments from the people on the line.

MR. LEWIS: Marvin Lewis.

CHAIRMAN SUNSERI: Yes, go ahead.

MR. LEWIS: Look, I'm not saying that the testing and the statistics cited are in any way wrong or screwed up or anything else.

In the late 1950s, we had a place called Shoshone in Simi Valley of Los Angeles County. And it has a very similar problem. A baffle plate broke loose. I don't know why. I can't remember that out of my head, but 60 years is a long time, 70 years, almost. Yes, over 60 years. It was a long time to remember back to that, but I seem to remember that just about everything I heard today was said before we had that problem at Shoshone and I'm saying hey, 60 years and you've got the same problem? Come on. Don't we ever progress? Thank you.

CHAIRMAN SUNSERI: Thank you. Anyone else? Anybody else on the phone line?

All right, let's close the phone line then. And we'll turn to the audience now.

MS. CURRAN: Good morning. I'm Diane Curran representing Riverkeeper. In the Indian Point license renewal proceeding, we have an admitted contention, several admitted contentions

in conjunction with the New York State Attorney General's Office that deal with issues related to aging equipment.

I really appreciate the presentations this morning. They've been very helpful, very informative. And I have a concern that I'd like to express for Riverkeeper about what appears to be a lack of a plan to include consideration of the root cause analyses for plants other than D.C. Cook. I am thinking of page 37 of your presentation, Mr. Poehler, where you mentioned that the NRC is going to be following up with the D.C. Cook root cause analysis and I would just like to confirm that I'm assuming that root cause analysis will be done -- well, one has been done for Indian Point and I don't know whether one has been done for Salem, but I would think that these would be very important studies that should be looked at together and integrated because it's clear that there's a lot of questions about what causes the degradation of these bolts and I wonder if I could confirm that with you and also see if these reports are all going to be submitted to the NRC.

CHAIRMAN SUNSERI: Unfortunately, this is not an opportunity to interact. It's just an

opportunity to make a comment. So we understand your concern has been expressed and is there anything else you would like to offer?

MS. CURRAN: Well, I was hoping since I seem to be the only person that is commenting that I could have a little interaction.

CHAIRMAN SUNSERI: We're obligated by the federal regulatory rules for this meeting, so I apologize for stifling that.

MS. CURRAN: Thank you.

CHAIRMAN SUNSERI: Any other members in the audience would like to make a comment?

All right, so let's go around the room here and hear from the ACRS members any further comments. And we'll start with Ron.

MEMBER BALLINGER: Thank you very much for the presentations. They're very informative and it brings everybody up to date on what's going on. So I thought it was a great job and thank you very much.

CHAIRMAN SUNSERI: Pete?

MEMBER RICCARDELLA: No comments other than to echo what Ron said.

CHAIRMAN SUNSERI: Dick?

MEMBER SKILLMAN: No further comment.

Thank you.

CHAIRMAN SUNSERI: Dana?

MEMBER POWERS: Well, of course the presentations were very nice on the specific issue. And if we look at the specific issue, it's moved along since the first findings in France which may to some people seem slow, but for those of us that worry about some screen blockage, it's been at a blindingly fast pace.

What I expressed concern about is moving forward beyond what's now planned. And for that to happen within the NRC, we need to have a nexus to the protection of public health and safety. Some very conservative calculations were done following the processes at the NRC and those are fine. I have no troubles with that. That's probably the only thing you can do.

The industry has indicated, however, they can do calculations that are substantially more rigorous I would say. Whether those are defensible and in front of my thermal hydraulic colleagues I don't know. They're rather picky. But it's that kind of analysis we're going to have to do to show that there is a nexus to the protection of public health and safety that

mandates continued NRC following of this issue beyond what's planned now.

As a parenthetical note, we continue to see that NDE techniques were not in a state of high reliability. This is certainly not an issue for the NRC to take on and may not even be an issue for the nuclear industry to take on, certainly show where the entire burden. I think this is a national issue that we need to recommend on a national basis as something to focus for the private sector, the academic sector and the government sector to take on.

CHAIRMAN SUNSERI: Thanks, Dana. John Stetkar?

MEMBER STETKAR: Nothing at all. Thank you.

CHAIRMAN SUNSERI: Walt.

MEMBER KIRCHNER: Thank you for the presentations. I can't pass up on Dana's earlier comments. I think I'm the only Thermal Hydraulics Subcommittee member here.

MEMBER POWERS: And proof that having one is a pleasant experience.

MEMBER KIRCHNER: Yes, exactly. Oh, two. So it did turn out pleasant. It might be

worth considering hearing more detail about the actual calculations that are done, but I'll leave that to the chair of the Thermal Hydraulics Subcommittee.

MEMBER POWERS: Oh, you guys are in trouble now.

CHAIRMAN SUNSERI: I would like to extend my appreciation to both the NRC and the industry representatives here for the informative presentation. Looks like there's still plenty of work ahead. I'm going to speak in advance for the subcommittee, but I would imagine that we'll be interested in the hot-cell work and examinations that are forthcoming and we'll likely be seeking a further update as more information becomes available in the future.

So thank you and at this point we will close this meeting.

(Whereupon, the above-entitled matter went off the record at 11:55 a.m.)

Recent Operating Experience with Baffle-Former Bolt Degradation

Jeffrey C. Poehler

Office of Nuclear Reactor Regulation
Division of Engineering

Advisory Committee on Reactor Safeguards
Meeting of the Subcommittee on Metallurgy & Reactor Fuels

November 16, 2016

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2. Design, Functions and Materials of PWR Internals, Baffle-Former Assembly, and Baffle-Former Bolts
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4. History of BFB Degradation
5. Factors Influencing Baffle-Former Bolt Degradation
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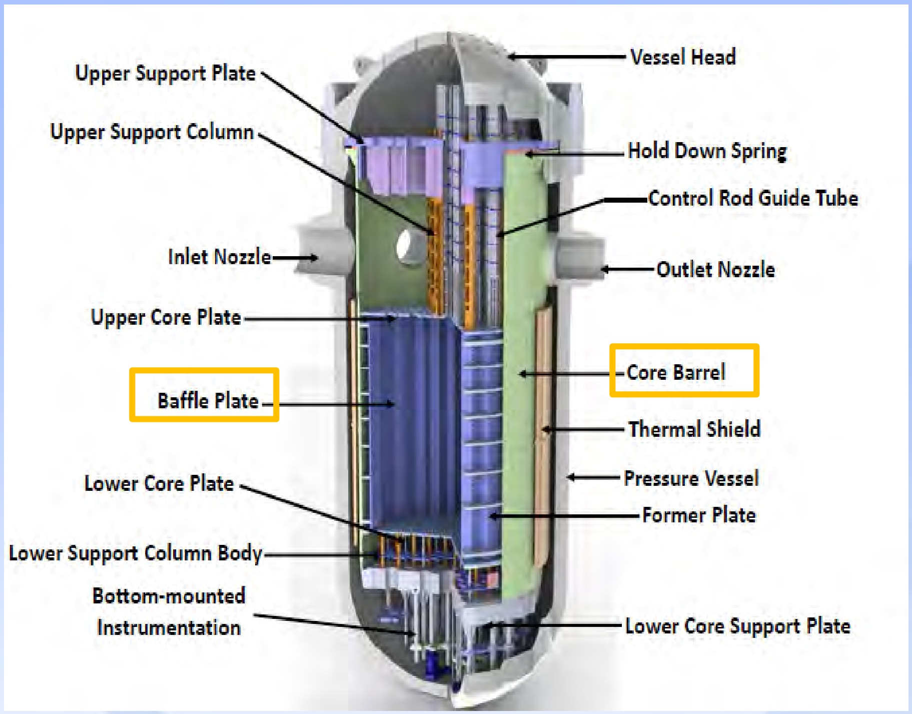
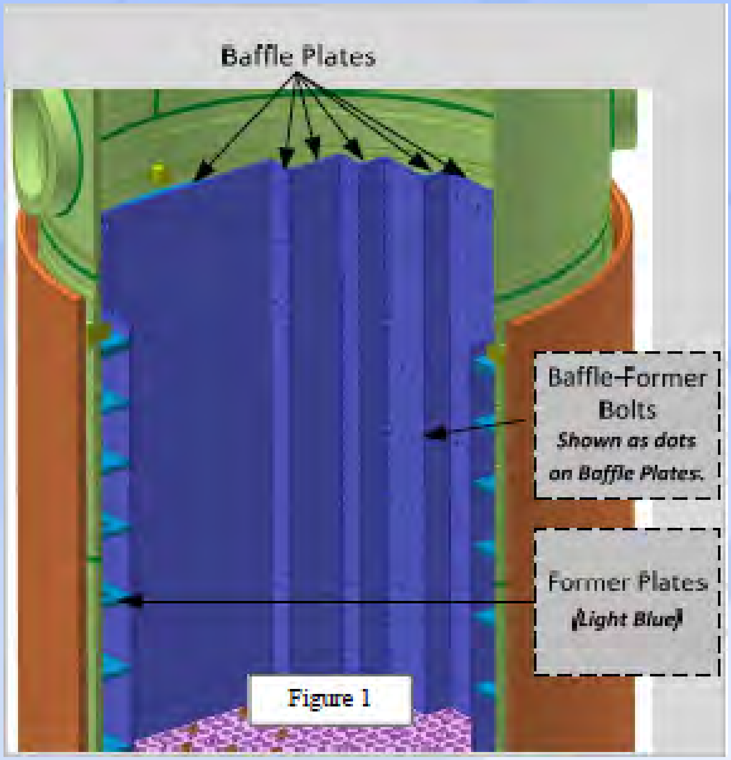
Design and Functions of the Baffle-former Assembly



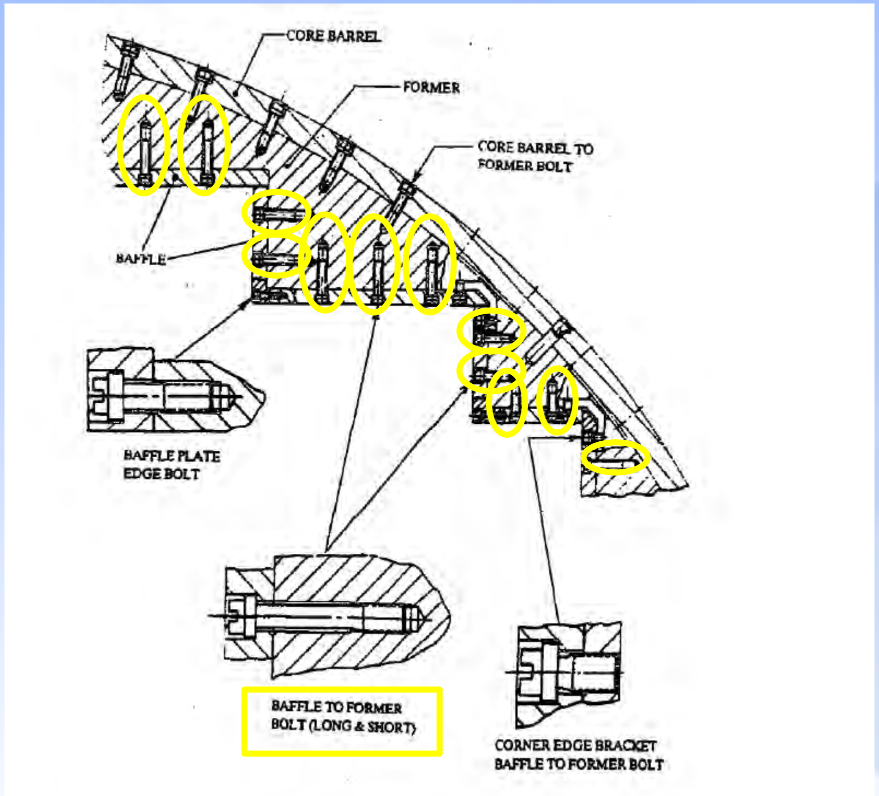


Function of baffle-former assembly is to direct coolant flow through the core. It also provides lateral support to the core during a seismic event or loss-of-coolant accident (LOCA).

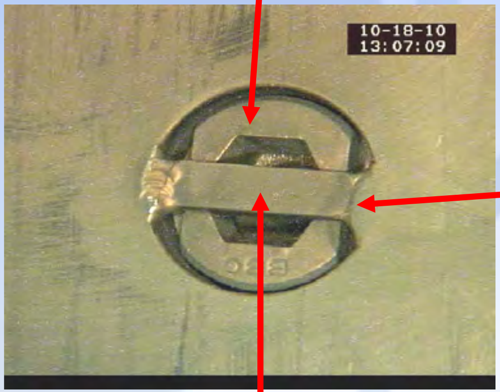
Design and Functions of the Baffle-former Assembly

Looking down into the core barrel of a Westinghouse-design PWR

Design and Functions of the Baffle-former Assembly - Design of Baffle-former bolts



- **Stainless steel bolts are 5/8” dia. x ~2” long and attach the baffle plates to the former plates to form the baffle-former assembly.**



Design and Functions of the Baffle-former Assembly - *Baffle-former Bolt Materials*

- **Westinghouse (W) plants:**
  - Type 347 stainless steel
    - Most older Westinghouse plants
    - Bolt design has sharper head-to-shank radius and shorter shank than in Type 316 cold-worked bolts.
  - Type 316 cold-worked stainless steel
    - Newer Westinghouse plants
    - All replacement bolts
- **Other NSSS designs**
  - B&W plants use Type 304
  - CE plants use annealed Type 316 (2 plants – others have welded core shroud).

Potential Consequences of Baffle-Former Bolt Degradation – Baffle Plate Movement

- Large numbers of degraded baffle-former bolts could allow detachment or deflection of baffle plates, particularly during a LOCA or seismic event.
- Plates could impact peripheral fuel assemblies, potentially causing grid crush and localized fuel cladding damage.
- In plants with control rods in peripheral locations, plate impact could jeopardize capability to insert these rods .
- Intact baffle-edge bolts would help mitigate plate detachment or deflection.
- For localized damage to peripheral fuel assemblies, a coolable geometry evaluation can be performed.

Potential Consequences of Baffle-Former Bolt Degradation – Baffle Jetting

- Baffle jetting is flow leakage through gaps between adjacent plates.
- Function of baffle-edge bolts is to ensure baffle plate integrity which prevents baffle jetting.
- Flow leakage causes flow-induced vibration of fuel pins resulting in localized fuel cladding damage, in some cases breaching cladding.
- Reactor coolant activity monitoring can detect increases in coolant activity that may be indicative of fuel damage.

Potential Consequences of Baffle-Former Bolt Degradation – Loose Parts

- Bolt heads and locking bars can become loose parts if bolts completely fracture.
- The clearances between the baffle plates and fuel assemblies are very small, which would tend to prevent bolt heads from escaping until the reactor is defueled.
- Likely result of loose bolt heads is fretting causing localized fuel cladding damage.
- Due to small size, it is unlikely that a few failed bolts would be detected by the loose part monitor.
- Baffle plates are unlikely to detach during normal operation, but if they did, potential for travel is limited by tight clearances and large size of plates.

History of Baffle-Former Bolt Degradation – Early History

- First identified in late 1980’s in European plants
- French 900mW CPO Plants
  - 6 plants found between 1% and 11% of bolts degraded
- Belgian plants
  - One 3-loop Framatome 900 mW design performed 5 examinations between 1991 and 2014 finding a total of 74 bolts degraded or uninterpretable.
  - Three other plants performed one UT examination each, finding a handful of degraded bolts.
- Mechanism for degradation is irradiation assisted stress corrosion cracking (IASCC).
- NRC issued Information Notice 98-11 to alert U.S. plant operators.
- US Industry initiated a program which included pilot inspections of baffle-former bolts at several plants.

History of Baffle-Former Bolt Degradation – US Pilot Plant Inspections

- Two 2-loop downflow plants with Type 347 bolts (1998-1999)
  - Plants found 7-10% of bolts degraded
  - Replaced degraded bolts, one plant replaced additional non-degraded bolts
  - Tensile testing of removed bolts performed at one unit, indicated number of defective bolts was less than indicated by UT results.
- Two 3-loop downflow plants with Type 316 bolts (1998-1999)
  - UT examined essentially all bolts – no indications
  - Pre-emptive replacement of >200 bolts each unit
- One B&W plant (2005)
  - No indications found

History of Baffle-Former Bolt Degradation – MRP-227-A

- **2000-2011, Most plants applying for license renewal made commitment to implement industry RVI program when it was issued.**
- **Industry program (MRP-227, Rev. 0) under review by staff 2009-2011.**
- **Industry program was approved by NRC staff in 2011 (MRP-227-A).**

History of Baffle-Former Bolt Degradation – MRP-227-A Inspection Requirements for Baffle-Former Bolts

- **Ultrasonic (UT) Examination**
- **Initial (baseline) Examination**
  - **Westinghouse and CE: 100% of bolts between 25-35 effective full power years**
  - **B&W: 100% of accessible bolts no later than two refueling outages from the beginning of the license renewal period**
- **Inspect every 10 years thereafter (or sooner if required by analysis of any observed degradation).**
- **All PWRs with baffle-former bolts must perform these inspections.**

History of Baffle-Former Bolt Degradation – D.C. Cook, Unit 2 (2010)

- **D.C. Cook, Unit 2 is a 4-loop downflow plant with Type 347 bolts (832 total baffle-former bolts).**
- **Eighteen bolts had visual signs of failure.**
- **Licensee replaced a total of 52 bolts with Type 316 SS, most on one large baffle plate, 42 found to be cracked.**
- **To establish extent of condition, on the three similar (large) baffle plates, licensee:**
  - **performed VT-3, no degradation**
  - **Tensile tested one bolt from each plate, no degradation**
- **No UT performed**
- **Two bolts locations left vacant .**
- **Westinghouse issued Technical Bulletin TB-12-5 to alert licensees.**

History of Baffle-Former Bolt Degradation – *Inspections under MRP-227-A (2011-2015)*

- **Westinghouse 2-loop (Type 347 bolts, 34 EFPY), five reactors**
  - Maximum defective bolts was 10.3%, 34 EFPY
  - One repeat inspection (plant inspected 1998), 15 additional degraded bolts, 2.7% of original bolts
  - One 2-loop plant that inspected in 1998 did a partial UT examination and replacement.
- **Westinghouse 3-loop (Type 347 bolts, 30-32 EFPY), four reactors**
  - UT examination of 100% of bolts at three units (1088 bolts each), number of potentially degraded bolts was 1, 2 and 8.
  - Partial UT examination of 305/1088 bolts at one unit, stopped due to equipment problems, no indications
- **B&W (Type 304 bolts, 30-32 EFPY) – Three reactors inspected, no more than 4 bolts with indications in each.**
- **CE – (Type 316 bolts, 27-28 EFPY) No inspections to date**

History of Baffle-Former Bolt Degradation – Indian Point, Unit 2

- **IP2 is a 4-loop, downflow plant with Type 347 bolts**
- **During Spring 2016 refueling outage, IP2 conducted MRP-227-A inspection per license renewal commitment.**
- **Visual examination of 1232 baffle-edge bolts, all acceptable.**
- **UT and visual examination of 832 baffle-former bolts**
  - **227 potentially degraded baffle-former bolts identified**
    - **182 ultrasonic testing failures**
    - **31 visually identified as protruding**
    - **14 inaccessible, conservatively assumed failed**

History of Baffle-Former Bolt Degradation – Indian Point, Unit 2 Corrective Actions

- Indian Point Unit 2 (IP2)
  - Replaced 278 baffle-former bolts (227 potentially degraded + 51 more to provide margin) with Type 316 SS
  - Completed analysis to support baffle-former assembly return to service - inspected by Region 1
  - Bolts sent to laboratory for testing to support root cause.
- Indian Point Unit 3 (IP3)
  - Operability evaluation of baffle-former assembly considering information from IP2 and Salem 1
  - Reschedule future baffle bolt examinations from 2019 to 2017

History of Baffle-Former Bolt Degradation – Salem Unit 1

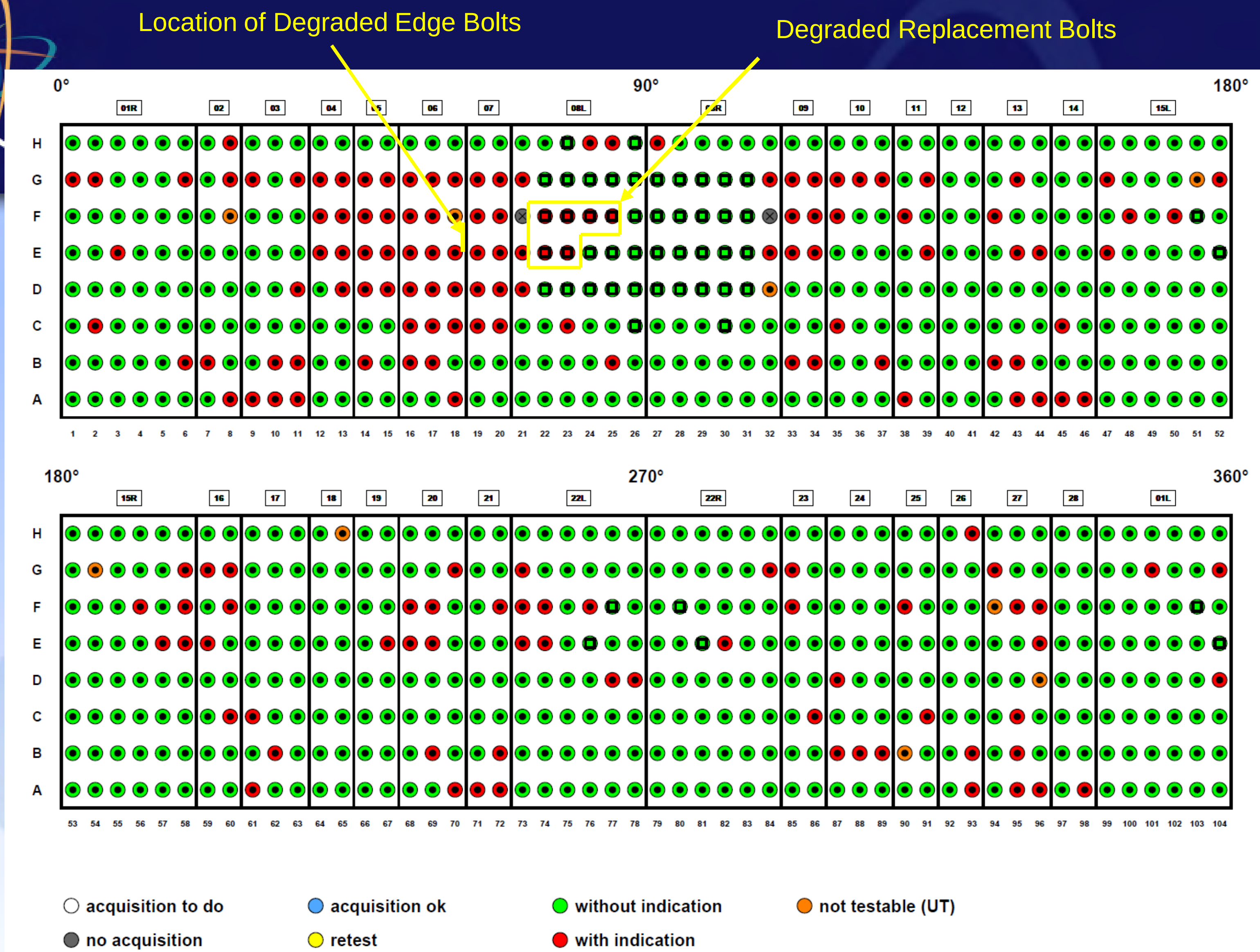
- Salem, Unit 1 is a 4-loop, downflow plant with Type 347 bolts.
- During Spring 2016 refueling outage, licensee was conducting augmented visual inspection of baffle-former bolts due to known degradation issues (832 total baffle-former bolts).
- Identified 11 bolts cracked at head, 9 had visually cracked lock bar welds, 19 bolts protruding from counterbore
- Follow up UT of remaining baffle-former bolts determined 135 bolts were potentially degraded, plus 16 unable to be tested.
- Overall, ~190 bolts identified as needing replacement
- Significant clustering of degraded bolts in several octants
- No baffle-edge bolt degradation observed

History of Baffle-Former Bolt Degradation –
Salem, Unit 1 Corrective Actions

- **Salem Unit 1**
  - **189 bolts replaced w/Type 316 SS**
  - **Analyzing selected bolts to confirm IASCC**
  - **Minimum bolting pattern analysis performed to determine replacement scope and justify operation for 1 cycle prior to re-inspection**
- **Salem Unit 2**
  - **Operability determination based on extent of condition from Unit 1**
  - **UT inspection of all bolts scheduled for spring 2017 (moved up from 2026)**

History of Baffle-Former Bolt Degradation – D.C. Cook, Unit 2 (2016)

- **During October 2016, D.C. Cook, Unit 2 performed UT examination of baffle-former bolts (832 total) on accelerated schedule in accordance with MRP interim guidance.**
- **Total of 179 potentially degraded bolts**
  - **170 with UT indications**
  - **9 untestable bolts**
- **2 vacant bolt locations from 2010**
- **Six (6) replacement bolts installed in 2010 had indications.**
- **At least one vacant bolt location correlated with a damaged fuel assembly.**
- **Visual examination of baffle-edge bolts found 5 degraded bolts.**



As-Found Condition of D.C. Cook, Unit 2 Baffle-Former Bolts
Fall, 2016 Refueling Outage

History of Baffle-Former Bolt Degradation – D.C. Cook, Unit 2 (2016) – Corrective Actions

- Replace minimum of 181 bolts w/type 316 SS - all potentially degraded bolts (179) plus missing bolts (2), plus additional bolts up to 201 total as time permits
- Indications in replacement bolts will be further investigated.
  - Previous plants that have re-inspected replacement bolts in service for 10-15 years found no indications.
  - Replacement bolts will be sent for laboratory analysis.
  - Performing sensitivity analysis to explore the effects on replacement bolt stress from failed original bolts in vicinity.
- Corrective actions for baffle-edge bolts to be determined.

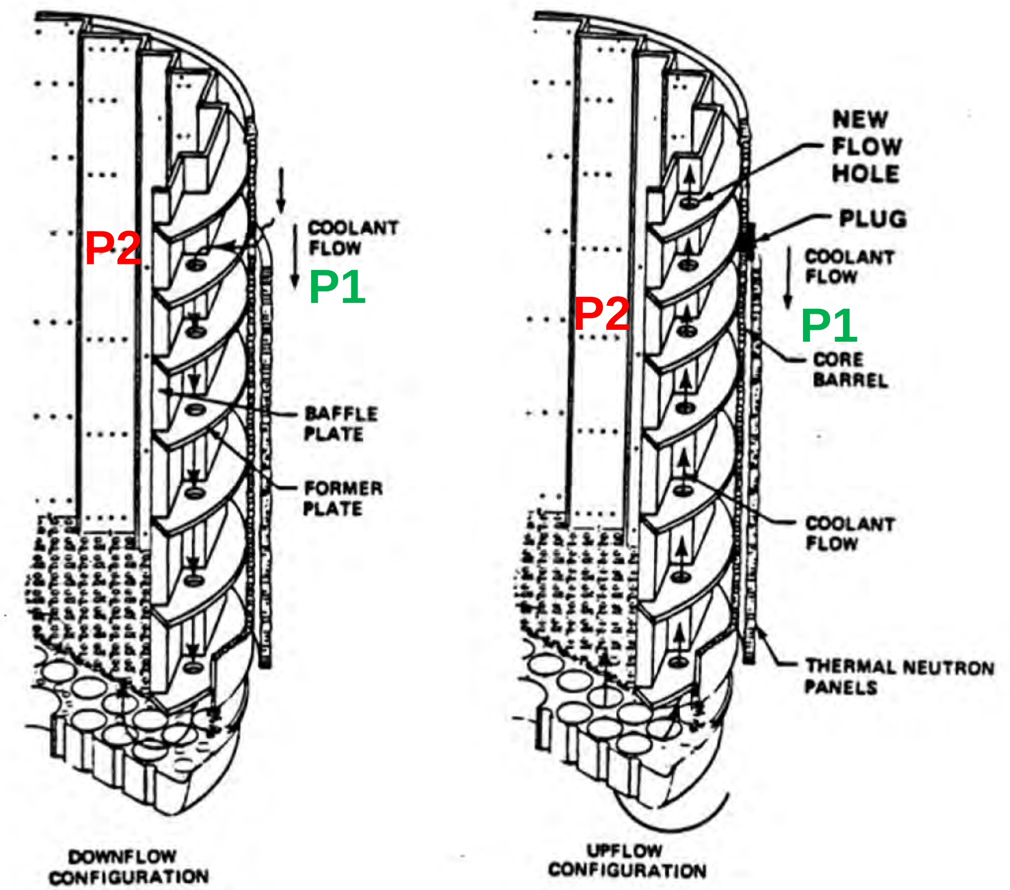
Factors Influencing Baffle-Former Bolt Degradation – Neutron Fluence

- Austenitic stainless steels are normally resistant to SCC in a PWR environment.
- With high fluence, grain boundaries changes occur.
  - Neutron fluence threshold for IASCC is ≥ $2x10^{21} n/cm^2$ (3 dpa) - Baffle plates and bolts receive up to 75 dpa in 60 years.
- Patterns of bolt degradation in IP2, Salem 1, and Cook 2 do not correlate with highest fluence locations in the core.
  - 2-loop and 3-loop plants also had similar or higher fluence levels but fewer degraded bolts.
- Switch to a low-leakage core design will reduce flux to bolts, may slow initiation of new IASCC cracks.

Factors Influencing Baffle-Former Bolt Degradation - Stress

- Stresses on baffle-former bolts are from a variety of sources:
  - Bolt preload stresses
  - Irradiation assisted stress relaxation
  - Void swelling of baffle plates
  - Differential pressure - greater in "downflow" than in "upflow"
  - Number of bolts per plate area
  - Bolt geometry –head-to-shank radius
- Fatigue loads may have an influence.
  - Affected by operating history – number of transients

Factors Influencing Baffle-Former Bolt Degradation - "Downflow" vs. "Upflow"



P1 > P2

Tries to force baffle plate inward = higher stress on bolts

P1 ≈ P2

Lower stress on bolts

Factors Influencing Baffle-Former Bolt Degradation – Stress (Clustering)

- **Baffle-former bolts initially crack randomly.**
- **Failure of a bolt leads to more load being carried by adjacent bolts.**
- **Over several operating cycles, this can cause the adjacent bolts to crack leading to clusters of failed bolts ("unzippering").**
- **Observed in French plants in successive examinations**
- **Severe clustering seen at Salem, Unit 1**

Factors Influencing Baffle-Former Bolt Degradation - Summary

- **Degradation of baffle-former bolts involves a complex interaction of stress, neutron fluence, and bolt material/design**
- **Other aging mechanisms, such as void swelling and irradiation assisted stress relaxation influence IASCC of bolts**
- **Industry is developing predictive models for baffle-former bolt degradation accounting for neutron fluence, stress, and material**
- **Higher susceptibility to degradation seems more related to stress differences rather than fluence**

Bolt Inspection and Replacement – Nondestructive Examination

- **Ultrasonic examination (UT)**
  - **Demonstrated for flaw detection only, not sizing**
  - **Any bolt with a detected cracklike indication is called defective.**
- **Visual examination (VT-3)**
  - **Not specified by MRP-227-A for baffle-former bolts, but some plants have performed voluntarily in response to OE**
  - **Specified for baffle-edge bolts**
  - **Can detect evidence of failed bolts such as displaced lock bars, protruding or missing bolt heads**
  - **Evidence of failed bolts has also been detected by non-VT-3 visual inspections.**

Bolt Inspection and Replacement – Replacement

- **Removal**
  - Many bolts can be removed intact once lock bar is cut.
  - Broken shanks can sometimes be removed mechanically but may require electro-discharge machining (EDM).
- **Replacement**
  - Replacement bolt design uses an expandable locking cup so no welding on highly irradiated baffle plate material is necessary.
  - Replacement bolts are cold-worked Type 316 stainless steel.
  - Replacement bolts have improved geometry to reduce stress at the head-to-shank transition.

Evaluation of Baffle-Former Bolt Degradation – *Acceptable Bolt Pattern Analyses*

- **WCAP-15029-P-A describes the NRC- approved generic methodology for determining acceptable patterns of intact baffle-former bolts.**
- **Uses the MULTIFLEX computer code to determine accident loadings**
- **Acceptance criteria include bolt stresses, fuel grid impact loads, momentum flux, fatigue and core bypass flow.**
- **When evaluating as-found bolt degradation, any degraded bolt is assumed to carry no load.**
- **Irradiated material properties are used for bolts.**
- **Plants use this methodology to evaluate as-found conditions and potential replacement patterns.**

Evaluation of Baffle-Former Bolt Degradation – Coolable Geometry

- **In some cases, as-found degradation may not meet stress and fuel grid impact criteria.**
- **If fuel grid impact criteria are exceeded, may need to demonstrate a coolable geometry with some damage to peripheral fuel assemblies**
- **WCAP-15029-P-A provides some guidance.**

Evaluation of Baffle-Former Bolt Degradation – *Reinspection Interval*

- WCAP-17096-NP-A, “Reactor Internals Acceptance Criteria Methodology and Data Requirements,” provides guidance for engineering evaluation of baffle-former bolt degradation.
  - Numerical margin consists of additional bolts over and above the number in the minimum bolting pattern.
  - If the number of degraded bolts is less than half the margin, may reinspect in ten years.
  - If the number of degraded bolts is greater than half the margin, a different reinspection interval must be justified.
- Plants with large numbers of degraded bolts would not have met the WCAP-17096-NP-A criteria so replaced all bolts to restore full structural margin.
- Industry is developing models for failure rates of baffle-former bolts – need reinspections of bolts at < 10 year interval to establish.

NRC Response – Regional Inspections

- The NRC staff performed targeted inspections at Indian Point and Salem and is performing a similar inspection at D.C. Cook.
- Inspections focused on:
  - NDE quality and accuracy (VT, UT)
  - Corrective actions, including evaluation of operating units
  - Adequacy of replacement bolt pattern, including margin for additional failures during next cycle
- Results of the NRC inspections are documented in publically available inspection reports.
- Regional inspectors engaging with other plants with regard to operability evaluations and plans for upcoming outages.

NRC Response – *LIC-504*

- **Evaluated four options,**
  1. **Immediate shutdown and inspection;**
  2. **Inspection next refueling outage;**
  3. **Generic communication;**
  4. **Maintain status quo**
- **Acceptable options must meet five criteria:**
  1. **Compliance with existing regulations;**
  2. **Consistency with the defense-in-depth philosophy;**
  3. **Maintenance of adequate safety margins;**
  4. **Demonstration of acceptable levels of risk;**
  5. **Implementation of defined performance measurement strategies**

NRC Response – LIC-504 - Results

- Risk met LIC-504 criteria of $CDF<1x10^{-3}$ and $LERF<1x10^{-4}$
  - Low frequency of large and medium LOCAs results in low CDF due to LOCA
  - Seismic risk assessment performed using bounding seismic hazard curve for U.S. based on recent updated seismic hazard submittals. Seismic assessment assumed 75% reduction in load capacity for baffle-former bolts, much greater than observed in any plant.
- Determined both Options 1 and 2 meet the five criteria of LIC-504: Options 3 and 4 have more risk uncertainty
- Option 1, immediate shutdown, places an unnecessary burden on licensees, thus Option 2 was recommended.
- Interim industry guidance effectively implements Option 2.

NRC Response – Operating Experience Summary

- Based on NRC staff review of operating experience, preliminary conclusion is that Westinghouse 4-loop design, downflow plants with Type 347 bolts are more susceptible to baffle-former bolt degradation than other PWR designs.
- Plants in this group are:
  - D.C. Cook, Units 1 and 2
  - Diablo Canyon, Unit 1
  - Indian Point, Units 2 and 3
  - Salem, Units 1 and 2
- EPRI MRP Interim Guidance calls for UT inspection of all baffle-former bolts at the next refueling outage for these plants (designated Tier 1a). The NRC is monitoring inspections and other actions at these plants.
- The immediate safety concern for these plants is addressed by the LIC-504 evaluation.

NRC Response – Future Activities

- Following root cause investigation at D.C. Cook 2, with focus on cause of degradation of replacement bolts and baffle-edge bolts
- Will determine if the LIC-504 requires revision based on new developments at D.C. Cook, Unit 2
- Continue to engage with industry focus group, especially on root cause from the three plants. Discuss with industry if changes to interim guidance are necessary
- Develop Information Notice
- Document assessment of MRP interim guidance
- Determine if changes to MRP-227-A guidance are needed

NRC – ACRS Metallurgy Subcommittee Briefing

Bernie Rudell, MRP Chair, Exelon

Heather Malikowski, PWROG MSC Chair, Exelon

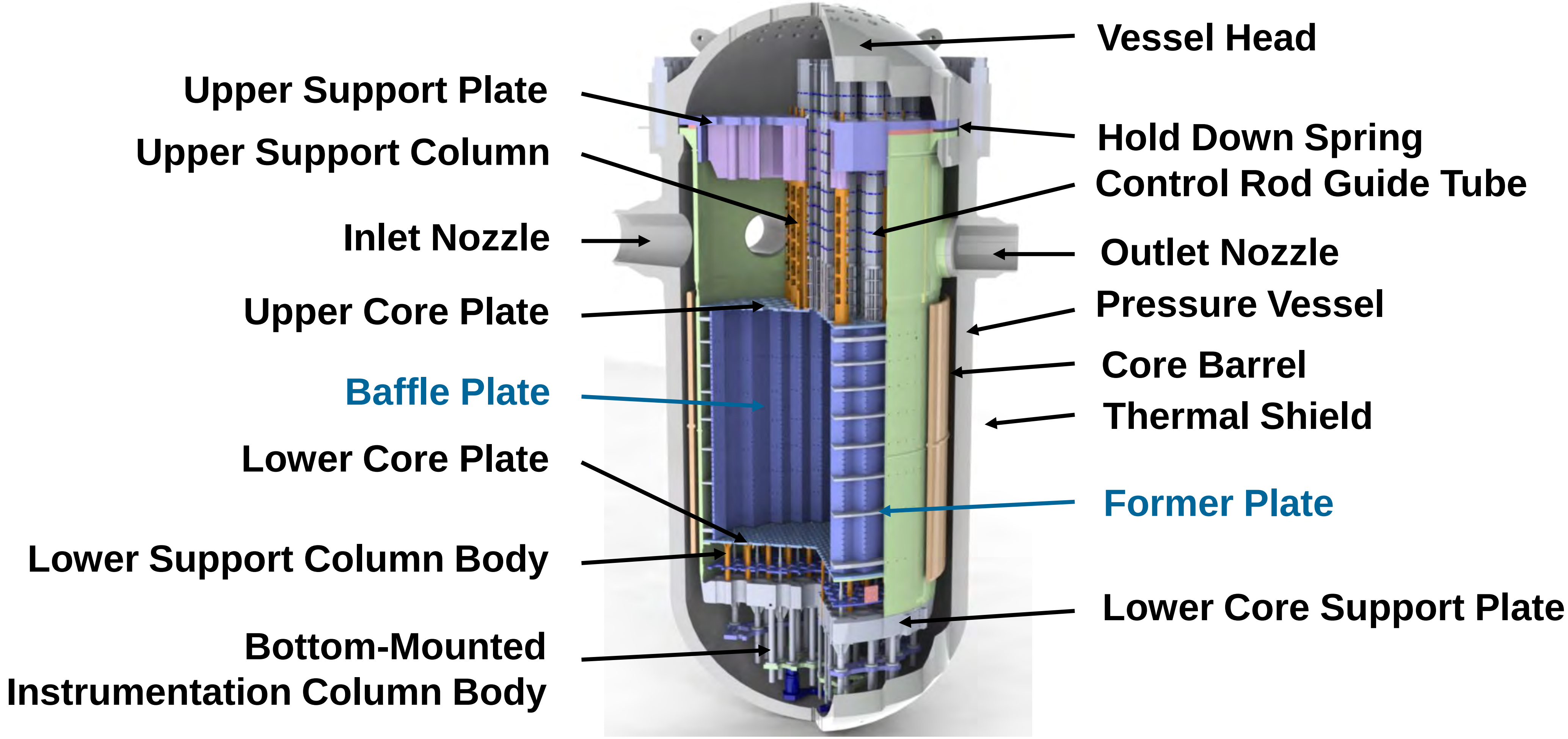
Tim Wells, BFB Focus Group Chair, Southern

Kyle Amberge, Project Manager, EPRI

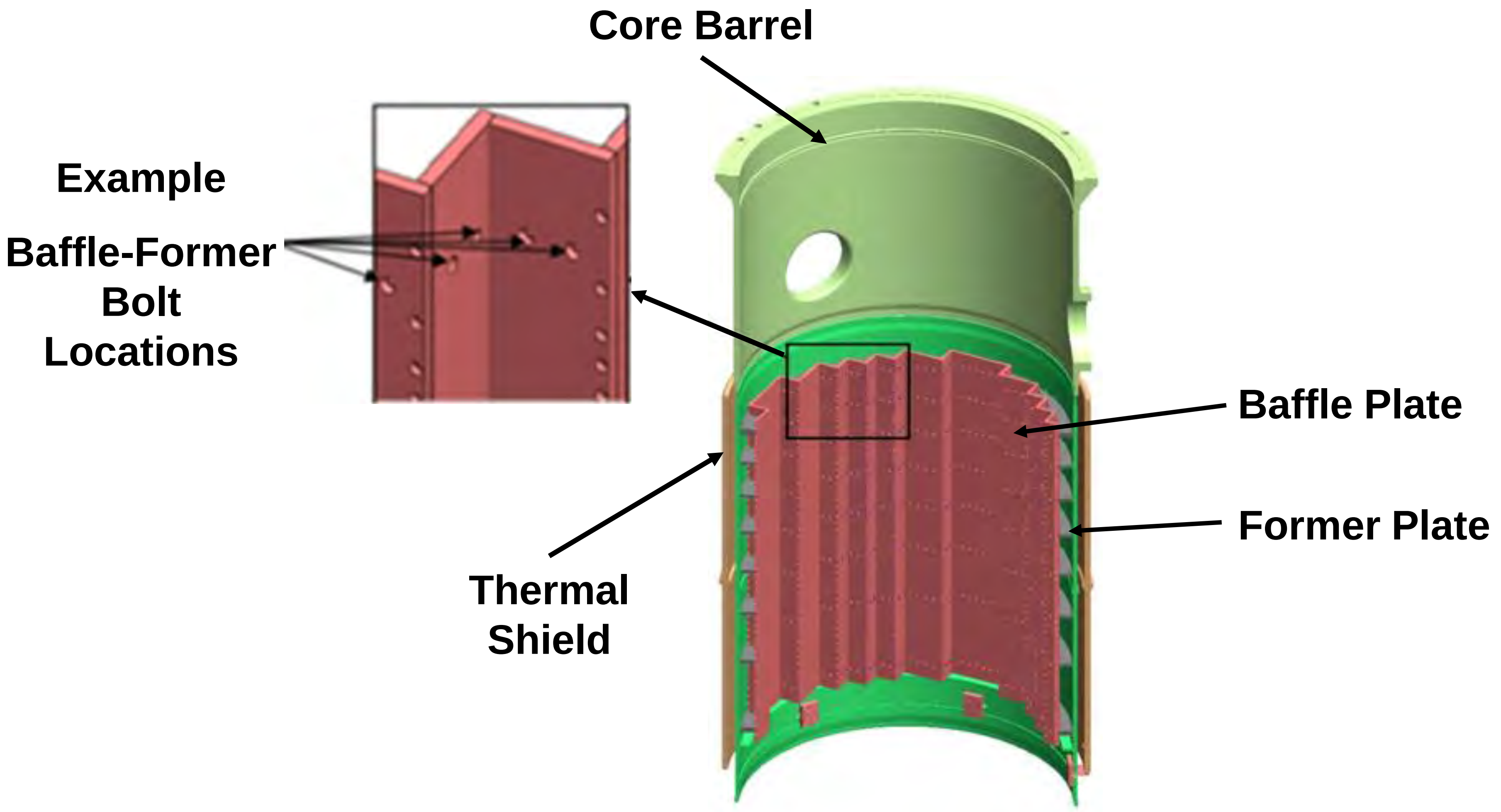
Bryan Wilson, Fellow Engineer, Westinghouse-PWROG

Rockville, MD, November 16, 2016

Westinghouse NSSS Internals



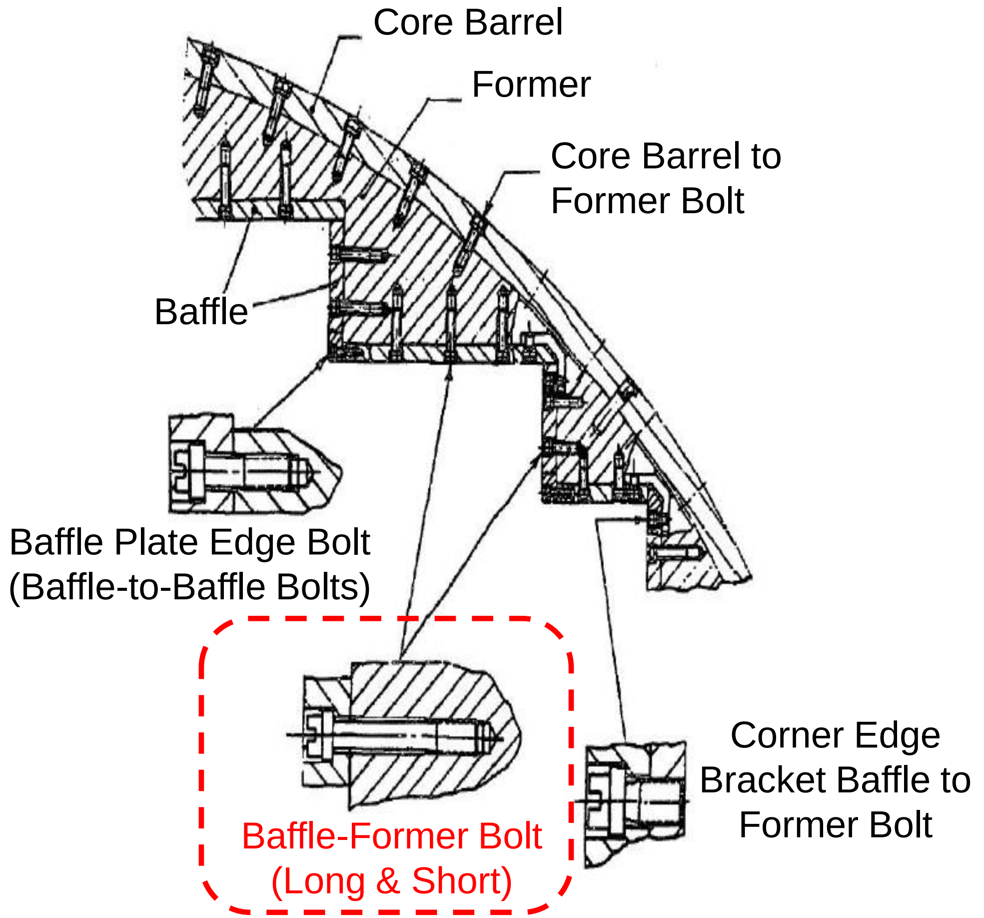
Baffle-Former Assembly



Source: ML15331A264

Baffle-Former Assembly Details

- Core barrel, baffle and former plates
  - Type 304 austenitic stainless steel material
- Baffle-Former Bolts (BFBs)
  - Attach the baffle plates to the former plates in the reactor lower internals assembly
  - Type 347 or Type 316 cold worked austenitic stainless steel material
  - Bolt head designs and shank lengths vary from plant-to-plant



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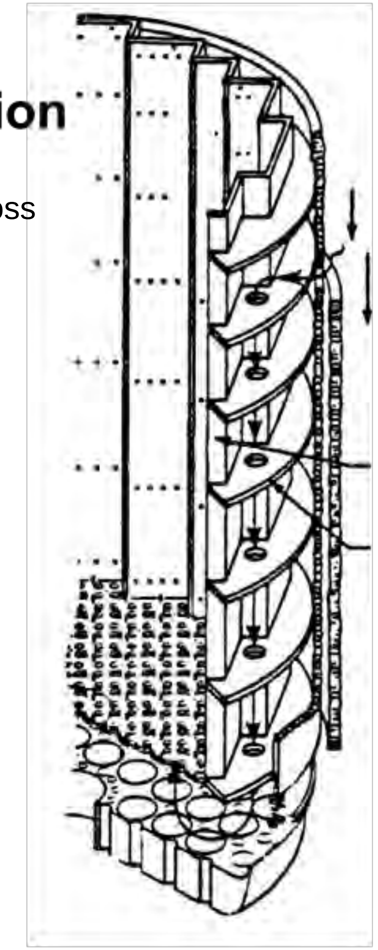
Coolant Flow Configurations

Downflow Configuration

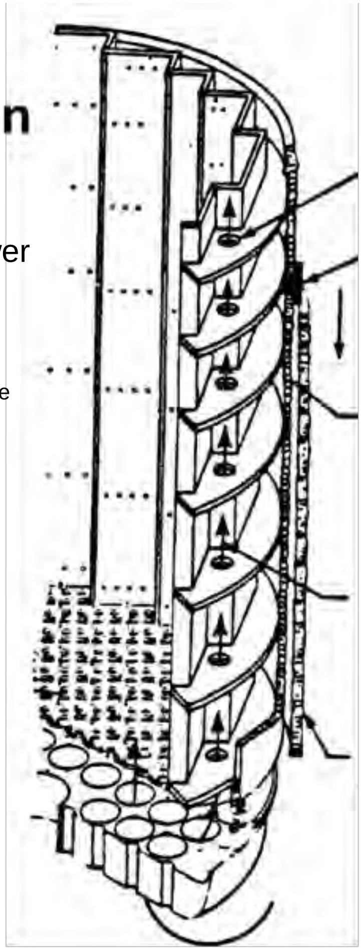
Large Differential Pressure (ΔP) Across the Baffle Plate -> greater BFB bolt loads



Upflow Configuration

Small Differential Pressure (ΔP) across the baffle plate -> lower BFB bolt load

*Figure shows modification made for Upflow Conversion



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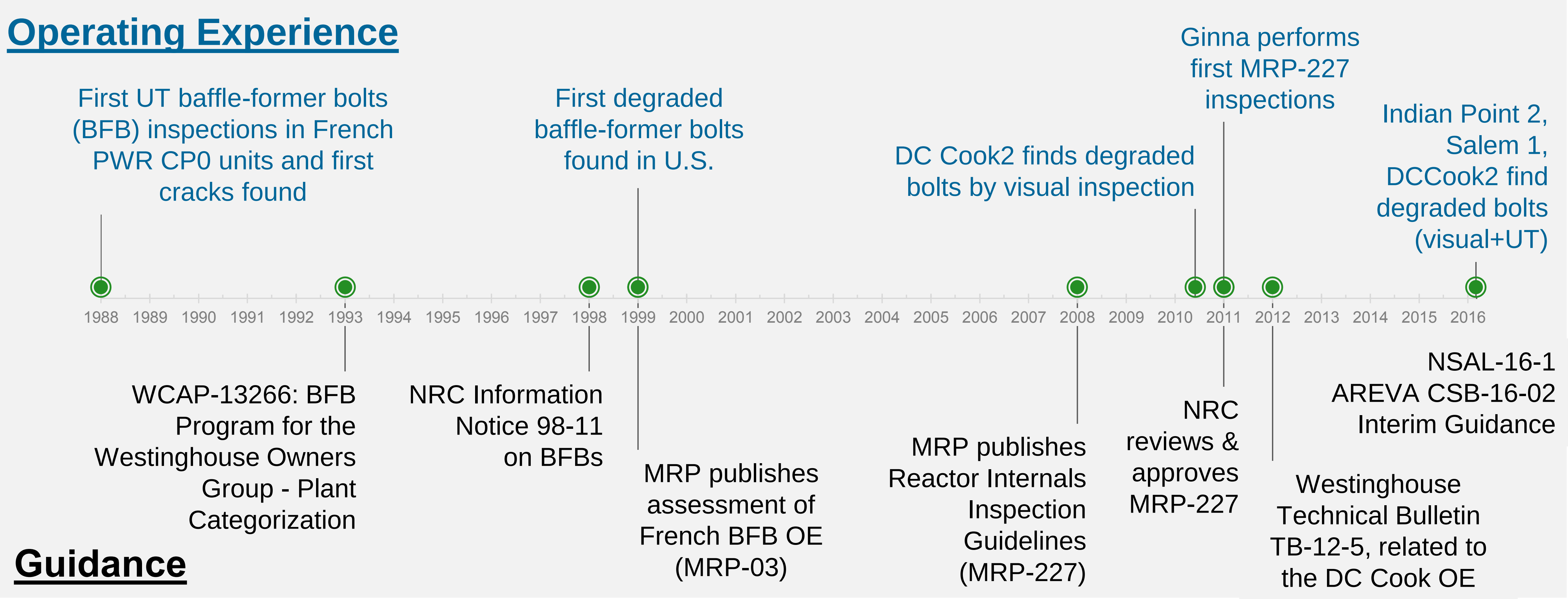
Operating Experience Overview



Per MRP-227-A, BFB UT exam is performed for WEC plants initially at 25-35EFPY and repeated every 10-years

Note: UT deployed as it becomes available and qualified for the various sites

Past Plant Operating Experience

EDF 1989-Present

Joint Owners Group Program 1998-2000

Westinghouse NSSS MRP-227-A Inspections

B&W NSSS and International Plant Results

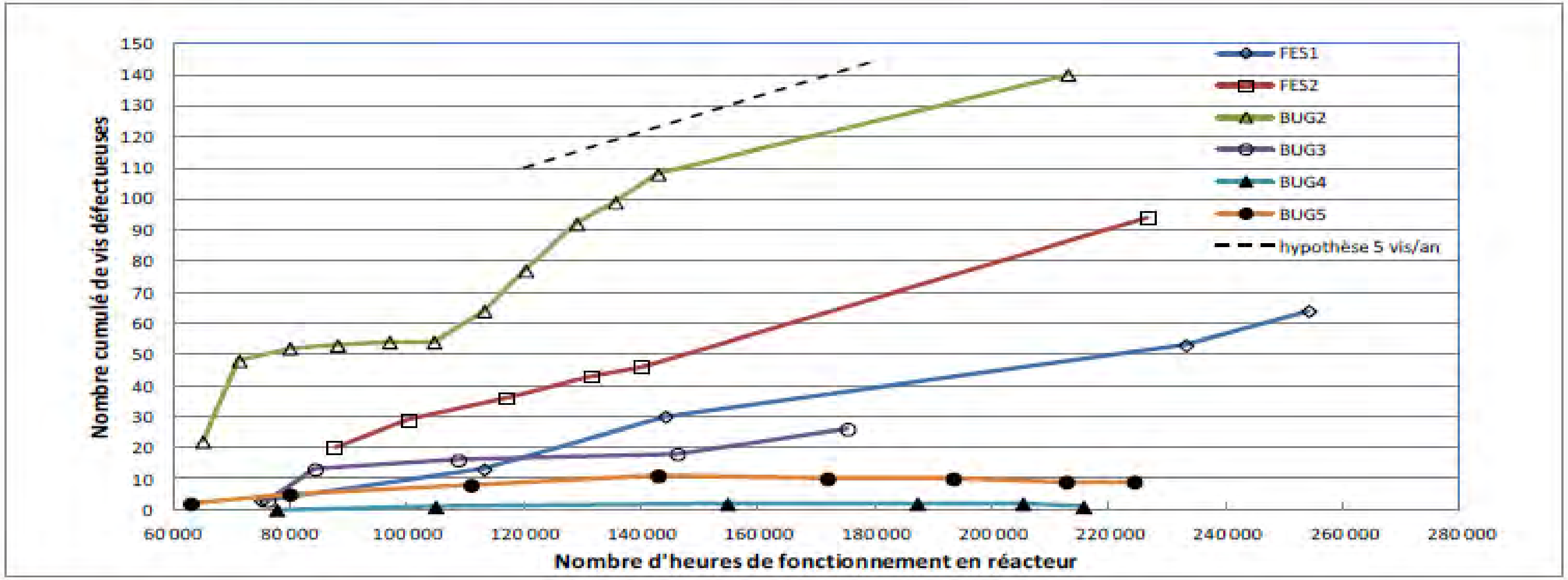
EDF Experience 1989-Present

- Baffle bolt failures reported - Limited to 'CP0' design
  - 3-loop (converted to upflow), with significant plant-to-plant variability (CPY design <5 indications over life of plant)
- EDF Periodic bolt replacement of failed original bolts, and periodic replacements included previously replaced bolts
  - Maintain sufficient number of "healthy bolts" to push next inspection to 10 years (based on observed failure rate of 5 bolts/year)

Overall BFB Timeline – CP0 Units

- X-axis: number of operating hours
- Y-axis: cumulative number of bolts found 'failed'/'unconclusive'
- -------- : evolution trend, assuming 5 failed bolts per year

BUG2-FES2-FES1-BUG3: 4/6 CP0 units are 'affected'
BUG5-BUG4: 2/6 CP0 units are 'unaffected'



2 – Industry Baffle Former Bolt Focus Group – May 19th 2016 – EDF Operating Experience

Joint Owners Group Program (15-22 EFPY)

- Sponsored Inspections of four plants (1998-2000)
  - Ginna: 2-loop, Downflow, Type 347SS
    - 9% UT Indications (Of these, 14 were sent for metallurgical examination. Results showed no indications of cracking, so this 9% likely contains a number of false calls)
    - Partial replacement program
  - Point Beach Unit 2: 2-loop, Upflow (converted) , Type 347SS
    - 8% UT Indications
    - Partial replacement program
  - Farley Unit 1: 3-loop, Upflow (converted), Type 316SS
    - No UT Indications
    - Proactive replacement of minimum pattern
  - Farley Unit 2: 3-loop, Converted Upflow (downflow at time of inspection), Type 316SS
    - No UT Indications
    - Proactive replacement of minimum pattern
- Inspection results and metallurgical exams of bolts removed during this program led to conclusion that BFB degradation was not a concern for the original plant operating period and that this could be addressed by an aging management program for license renewal (MRP-227-A).

Westinghouse NSSS MRP-227-A BFB Inspections

through Sept. 2016 (excluding IP2, SAL1, DCCook2)

- Ginna: 2-Loop, Downflow, Type 347SS
  - 2nd Inspection (2011)(partial inspection of 123 original bolts and 56 replacement bolts): one additional UT Indication (Partial Replacement of 25 bolts)
- Point Beach Unit 1: 2-Loop, Upflow (converted), Type 347SS
  - 1st Inspection (2013): No UT Indications
- Point Beach Unit 2: 2-Loop, Upflow (converted), Type 347SS
  - 2nd Inspection (2014): 2% Additional UT Indications
- Prairie Island Unit 1: 2-Loop, Downflow, Type 347SS
  - 1st Inspection (2014): 6% UT Indications
- Prairie Island Unit 2: 2-Loop, Downflow, Type 347SS
  - 1st Inspection (2013): 10% UT Indications
- Surry Unit 1: 3-Loop, Downflow, Type 347SS
  - 1st Inspection (2010): <1% UT Indications
- Surry Unit 2: 3-Loop, Downflow, Type 347SS
  - 1st Inspection (2011): <1% UT Indications
- Robinson: 3-Loop, Downflow, Type 347SS
  - 1st Inspection (2013): <1% UT Indications
- Turkey Point Unit 3: 3-Loop, Downflow, Type 347SS
  - 1st Inspection (2015 - partial inspection of 305 bolts): No UT Indications
- North Anna Unit 1, 3-loop, Downflow, Type 347SS
  - 1st Inspection (2016): <1% UT indications

B&W NSSS and International Plant Results

- Crystal River Unit 3, Type 304SS (2005)
  - No relevant UT indications - UT performed due to visual indication from baffle-to-baffle bolts Oconee Unit 1, Type 304SS (2012)
  - No relevant UT indications - Four BFBs uninspectable due to large welds on locking bars
- Oconee Unit 2, Type 304SS (2013)
  - No relevant UT indications - One BFB uninspectable due to UT probe not seating properly
- Oconee Unit 3, Type 304SS (2014)
  - One BFB identified with crack-like indications - One BFB uninspectable due to UT probe not seating properly
- ANO Unit 1, Type 304SS (2016)
  - UT exams currently underway as of 11/14/2016
- Doel 1: 2-Loop Downflow, Type 316SS
  - 1st Inspection: No relevant UT indications (1991)
  - 2nd Inspection (2005) and 3rd Inspection (2015): 2% UT Indications (replaced 9 bolts in 2015)
- Doel 2: 2-Loop Downflow, Type 316SS
  - 1st Inspection (2006) and 2nd Inspection (2015): <1% UT Indications (replaced 7 bolts in 2015)
- Krsko: 2-Loop, Downflow (prior to inspection), Type 316SS
  - 1st Inspection: <1% UT Indications (2013)
- Tihange 1: 3-Loop, Upflow (converted), Type 316SS
  - 960 of 1088 bolts inspected in each of the following inspections
  - 1st Inspection: 4% UT Indications (1995)
  - 2nd Inspection: 3% UT Indications (2002)
  - Most recent Inspection: No relevant UT Indications (5 bolts either not inspectable or not interpretable) (2014)
- Ringhals 3: 3-Loop Downflow, Type 316SS
  - 1st and 2nd Inspections: <1% UT Indications/uninspectable (2000 and 2007)
  - 3rd Inspection: <1% UT Indications/uninspectable (2016)

Observations from Broader OE

- Excluding the OE at Cook Unit 2, Indian Point Unit 2, and Salem Unit 1 (discussed later in the presentation), the following observations can be made based on inspection OE gathered to date from international and domestic plants:
  - Bolts with UT indications tend to be randomly distributed
  - Distributions are consistent with expectations of IASCC failures and fluence effects
  - Quantity and distribution of bolts with indications bounded by historical generic safety assessment generated in mid-1990s (documented in report WCAP-15328)
  - Industry response to replacement of bolts with indications has been positive

Recent Plant Operating Experience

DC Cook 2 – fall 2010

Indian Point 2 – spring 2016

Salem 1 – spring 2016

DC Cook 2 – fall 2016

DC Cook Unit 2 (2010 / 22 EFPY) (4-Loop Downflow)

- Fuel failure in peripheral assembly attributed to wear against broken bolt head
- Bolt heads and lock bars found on lower core plate
- Visual inspections revealed 18 degraded bolts on 270^{o} baffle plate in Rows 2-5
  - Additional bolts removed from plate with visual indications to define extent of localized degradation (approx. 40 bolts in single patch)
  - Additional test bolts removed from symmetrical locations to evaluate potential for degradation on other plates (all of these test bolts were found to be intact)
- No UT inspections performed in 2010 (at that time UT was not qualified or optimized for the Cook 2 bolt design)
- Degraded and test bolts replaced (total of 52 bolts and 2 open holes)
- Westinghouse issued Technical Bulletin TB-12-5
- 100% Visual VT-3 inspection conducted in 2012 with no additional indications

Indian Point Unit 2 (2016 / 31 EFPY) (4-Loop Downflow)

- Degraded bolts/lock bars noted visually prior to planned MRP-227 100% UT exams
- Markings on periphery of neighboring fuel assembly identified (no fuel failure).
- Inspections identified 227 BFB with visual or UT indications (includes 14 uninspectable)
- UT indications were clustered
  - Spanned various quadrants, mostly in former Rows D through G
  - Multiple groups of 10+ adjacent failures / At least one cluster of 50+ adjacent failures
- Observed failures exceed WCAP-17096-A engineering acceptance criteria
- Site-specific response
  - Performed Acceptable Bolting Pattern Analysis (ABPA)
  - Performed Replacement Bolting Pattern Analysis
  - Performed engineering evaluations supporting Unit 3 Extent of Condition Evaluation
  - Performed engineering evaluations supporting Unit 2 Assessment of Potential Safety Impacts
  - Performed baffle-former bolt removal and replacement
  - Quarantined select bolts for future testing

Salem Unit 1 (2016 / 28 EFPY) (4-Loop Downflow)

- Conducted visual exams every other refueling outage in response to DC Cook Unit 2 OE and TB-12-5; MRP-227 exams were not planned until 2017
- Degraded bolts/lock bars noted in visual exams followed by doing 100% UT exams
- Loose/protruding bolt heads resulted in fuel fretting and one fuel clad failure
- Inspections identified 182 BFB with visual degradation or UT indications (includes 18 uninspectable)
- UT indications were clustered
  - More concentrated (than Indian Point 2) to a few adjacent octants
  - Multiple groups of 10+ adjacent failures / At least one cluster of 50+ adjacent failures
- Observed failure pattern exceeds WCAP-17096-A engineering acceptance criteria
- Site-specific response
  - Performed Acceptable Bolting Pattern Analysis (ABPA)
  - Performed Replacement Bolting Pattern Analysis
  - Performed engineering evaluations supporting Unit 1 Justification for Past Operation
  - Performed engineering evaluations supporting Unit 2 Extent of Condition Evaluation
  - Performed baffle-former bolt removal and replacement
  - Quarantined select bolts for future testing

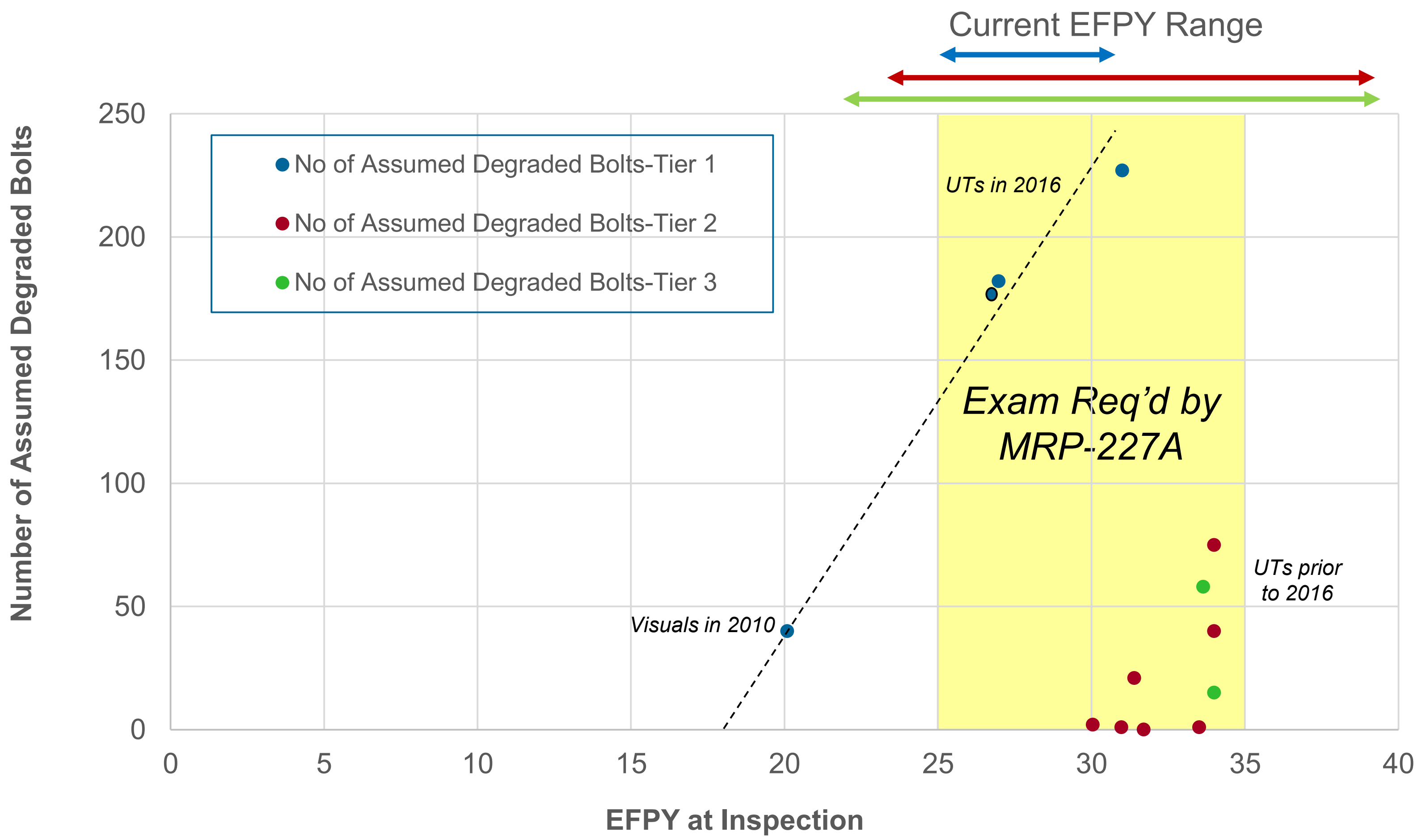
DC Cook Unit 2 (2016 / 28 EFPY) (4-Loop Downflow)

- Two (2) on-line fuel leaks identified during the last fuel cycle associated with two (2) empty bolt-holes from bolts that were not replaced in 2010 (suspected damaged by jetting through a vacant BFB hole)
- Inspections identified 179 BFB with visual degradation or UT indications (includes 9 that were not inspectable, and three (3) with visibly degraded lock-bar welds)
  - Includes 6 replacement BFBs from 2010 event that exhibit UT indications
  - Five (5) Baffle-Edge-Bolts on one seam appear visually failed
- UT indications were clustered
  - Spanned various quadrants
  - Multiple groups of 10+ adjacent failures / At least one cluster of 50+ adjacent failures
- Observed failure pattern exceeds WCAP-17096-A engineering acceptance criteria
- Site-specific response currently being implemented
  - Performed Acceptable Bolting Pattern Analysis (ABPA)
  - Performed Replacement Bolting Pattern Analysis
  - Performed engineering evaluations supporting Unit 2 Justification for Past Operation
  - Performed engineering evaluations supporting Unit 1 Extent of Condition Evaluation
  - Performing baffle-former bolt removal and replacement, expect to replace 200 bolts and plan to complete by 12/5/2016
  - Plan to quarantine select bolts for potential future testing

Conclusions from Recent OE

- These three plants share a common plant design configuration (4-loop downflow), bolt design, and bolt material
- Bolts with visual or UT indications tend to be clustered
- Distributions seem to indicate the presence of a mechanism causing adjacent bolts to become more susceptible to failure
- Assessing impact of new findings from DC Cook 2 exams:
  - Replacement 316 CW BFBs (6) from 2010 event with UT indications
  - Visually degraded edge bolts (5) on one panel, in the center of a large area/cluster of BFB failures

US Trends – BFB Focus Group Industry OE Database



Factors Influencing BFB Degradation

- Fluence, Stress, Material, and Time
  - Contribute to a condition conducive to IASCC crack initiation
  - Stress is influenced by plant design (loads), bolt design, stress relaxation, clustering (failure progression), and bolt replacement
- Plant design – number of loops, downflow/upflow
  - Impact the stresses that develop in the baffle-former bolts
- Bolt design
  - Bolts are either type 347 or cold-worked type 316 austenitic stainless steel material
  - While it is believed 316 has improved IASCC resistance based on our limited OE, insufficient direct comparative data exists at this point to make a definitive conclusion due to the introduction of additional variables (i.e. bolt design, plant operating parameters, etc.).
  - Type 347 bolts tend to have a sharper head to shank transition radius as compared to the type 316 designs
  - Type 347 bolts are generally shorter than the 316 bolts
  - Bolt length and head-to-shank transition radius (stress concentration) impact the average and peak stresses impacting IASCC and fatigue susceptibility and are believed to be more influential with respect to BFB degradation than material differences

Factors Influencing BFB Degradation (cont.)

- Stress relaxation
  - Occurs over a relatively short duration at high temperature and fluence
  - Reduces joint efficiency causing increased bolt loads
- Clustering of bolts
  - As failures occur, stresses redistribute into surrounding bolts
  - Increased stress can enhance IASCC susceptibility and fatigue / propagate existing cracks
- Bolt Replacement
  - Modifies how stress is distributed across bolts in the structure
  - May slow the process of degradation for nearby original bolts by tightening the structure and reducing load carried by original bolts

Explaining Failure Patterns

1. **Randomly Distributed Failures**
   - Hypothesis: IASCC failure rate governed by stress, temperature, dose, time
   - Key variables: Material, Plant Design, Bolt Design
   - Simplified empirical representations and comparisons between plants can be used in this case: e.g. Weibull Distribution
2. **Dose-Related**
   - Hypothesis: IASCC failure rate driven by temperature and stress from high dose rates which causes local acceleration
   - Key variables: Fluence, Gamma Heating, Irradiation Creep, Void Swelling
   - Good correlation with deterministic results from existing MRP aging model (IRRADSS Model): Predicted patterns similar to bolt failure experience in French CP0 plants
3. **Clustered**
   - Hypothesis: IASCC failure rate still affected by same parameters as random distribution but local stress increases around groups of adjacent failed bolts due to transfer of primary loads
     - Particular issue for downflow configuration plants due to high baffle plate ΔP
   - Current modeling efforts are underway with the goal of replicating and predicting this failure pattern

Why Clustered Failures?

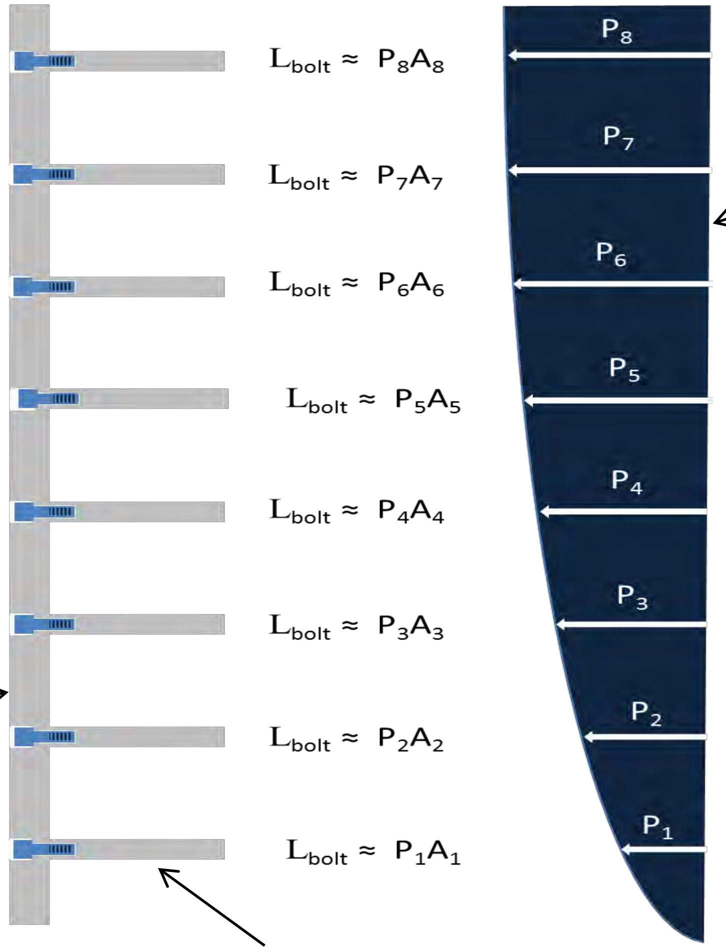
- **Answer 1**: Failures have a common cause that localizes the effects of IASCC or fatigue
  - Stress anomaly
  - Asymmetric fatigue
  - Local hot spot
  - Bolt source or installation sequence

- **Answer 2**: Failure propagates after initial random failures reach a critical level
  - Failures are random until adequate clusters of failed bolts form
  - Probability of failure in neighboring bolts increases with the increased load
  - Group of bolts "unzippers" as more and more adjacent bolts fail

- Possibly a combination of both of these answers

Vertical Stress Variation in Downflow Design

Bolt load (L_{bolt}) varies due to varying pressure differential across baffle plate



Differential Pressure across the baffle plate (P_i) varies with elevation

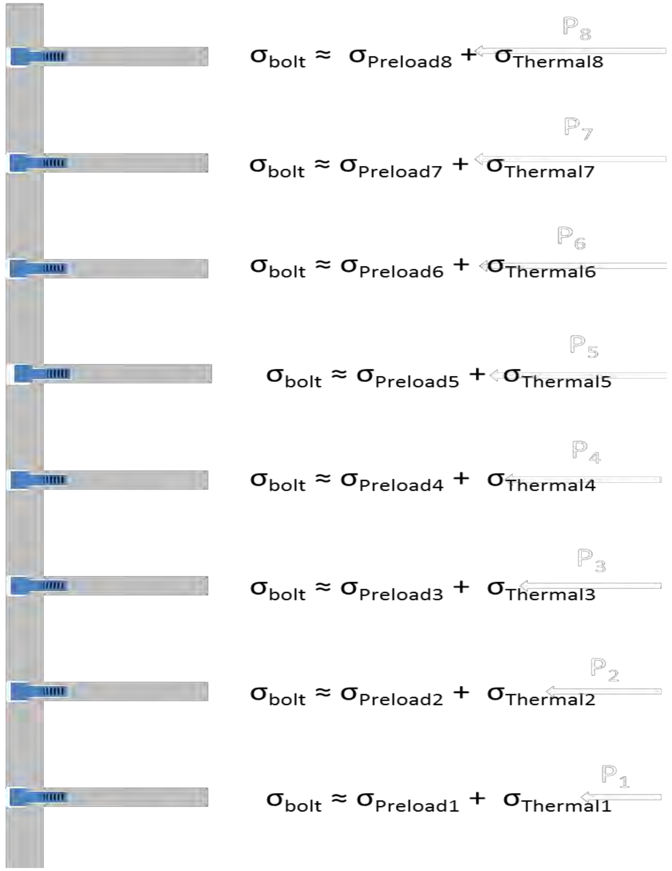
Baffle Plate

Former Plate

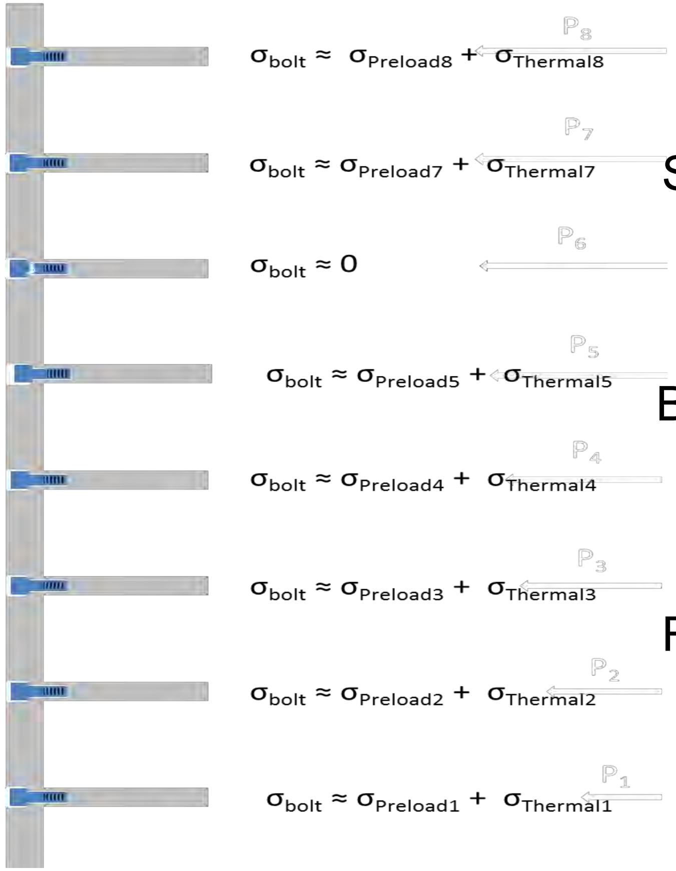
Westinghouse

PWROG PWR Owners Group

Stress Redistribution to Adjacent Bolts - Secondary



0 Failed Bolts



1 Failed Bolt

Secondary stresses predominant in early life (bolt preload)

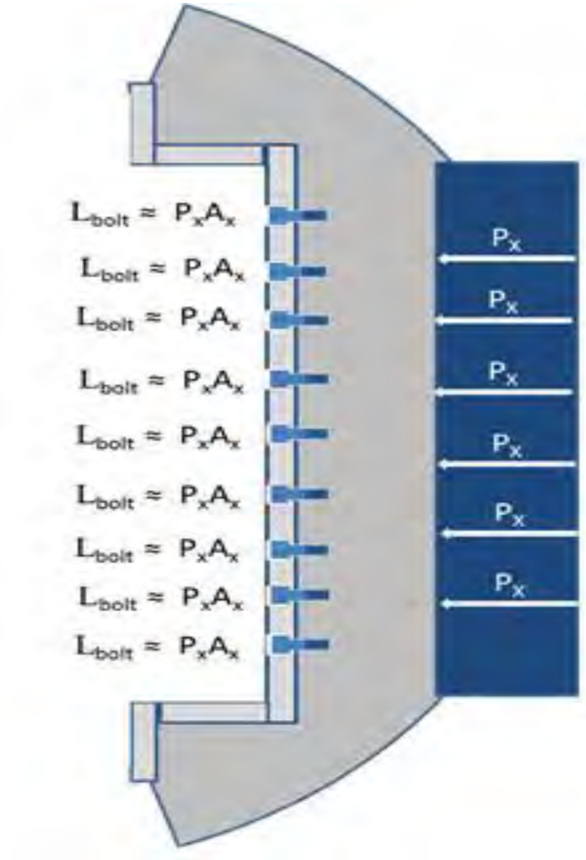
But do not redistribute with failure

Irradiation Induced Stress Relaxation diminishes secondary stresses over time

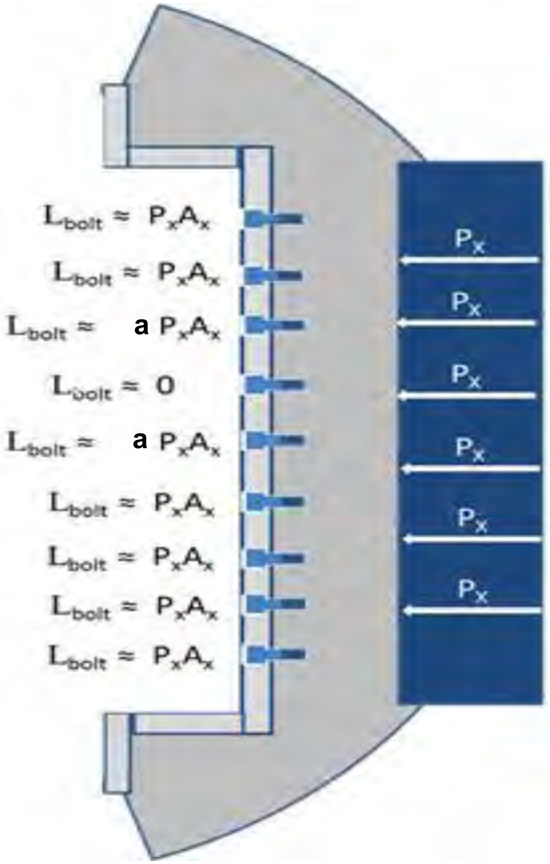
Westinghouse

PWROG PWR Owners Group

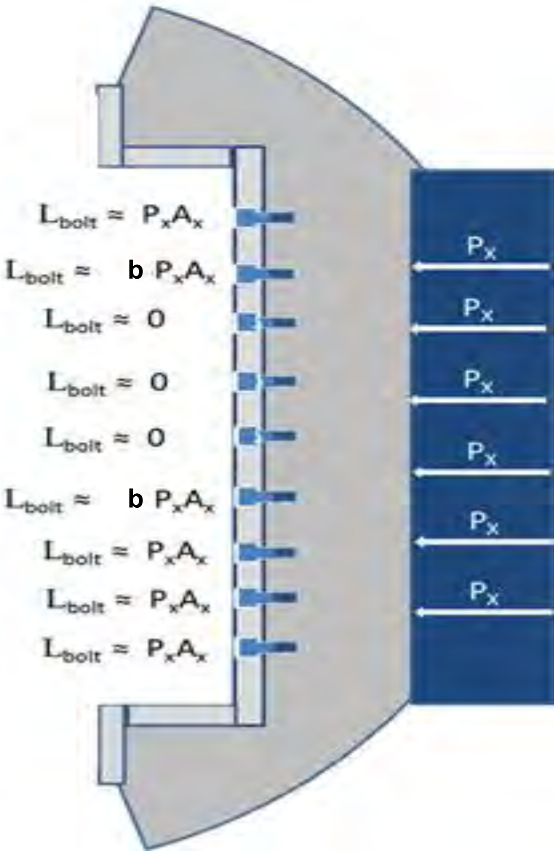
Stress Redistribution to Adjacent Bolts - Primary



0 Failed Bolts



1 Failed Bolt



3 Failed Bolts

Primary stresses dominate after irradiation stress relaxation occurs.

Stress redistributes as bolts fail.

First order approach shown in figure: b > a > 1

Consequences and Evaluation of BFB Degradation

- Degraded condition assumed for safety evaluations
  - It was assumed that all baffle-former bolts in a quadrant were degraded and a pinned constraint of baffle edges remained due to the presence of some edge bolts and/or overlapping plates
- Safety evaluations considered the following:
  - Impact on core bypass flow for non-LOCA and LOCA safety analysis
  - Control rod insertability
  - Fuel assembly grid crush and core coolability
- Conclusion was that plants would remain in a safe condition
  - Non-LOCA and LOCA safety analysis showed acceptable results when considering the increased bypass flow associated with this condition
  - Fuel assembly grid crush could occur in the peripheral fuel assemblies and to a much lesser degree in inboard fuel assemblies; however, this would result in a negligible impact on core coolability
  - Control rods would be able to fully insert to shut down the plant
  - Therefore, the ability to cool the core, maintain reactor shutdown, and remove decay heat in the long-term after a LOCA, would not be compromised by baffle-former bolt degradation

Consequences and Evaluation of BFB Degradation (cont'd.)

- Additional evaluations of BFB degradation included:
  - Baffle jetting
    - Loose baffle plates could result in opening up of baffle-to-baffle gaps or flow holes around broken bolts
    - Result would potentially be increased fuel rod vibration and wear
    - Monitoring of coolant activity will detect the presence of damaged fuel
  - Loose Parts
    - Heads of failed BFB generally remain trapped by the lock bar
    - Cracked lock bar welds and protruding bolt heads have been found in areas of large clusters of failed BFB
    - Loose bolt heads and lock bars can result in localized fretting of the fuel rod cladding
    - Monitoring of coolant activity will detect the presence of damaged fuel
    - Loose bolt heads will remain trapped by the adjacent fuel assembly
    - Lock bars can enter the reactor coolant systems but have been determined to have a negligible impact on safe operation

Industry Communication

- Tech Bulletin, TB-12-5, was issued in March 2012 after the Fall 2010 Cook Unit 2 visual findings of damaged baffle-former bolts
- Nuclear Safety Advisory Letter, NSAL-16-1, released on July 5
  - Determination of leading and affected plants was consistent between TB-12-5 and NSAL-16-1
  - Westinghouse 4-loop downflow plants are most susceptible
  - All Westinghouse designed NSSS plants with baffle-former bolts and CE designed plants with bolted core shrouds are potentially affected by this issue
  - The Westinghouse AP1000® plant design does not utilize baffle-former bolts and is not affected by this issue
- Affected plants broken down into 4 Tiers

Intent of NSAL Recommendations

- Current working theory is that the cause of the failures is correlated more closely to key design features (previously discussed) rather than plant-specific operation or conditions
- Based on this, a tiered approach was chosen to rank plants based on how closely key design features matched design features of the plants that have experienced the OE
- The tiered approach also used past inspection data to inform a relative ranking between tiers and sub-tiers
- The NSAL Recommendations are intended to:
  - Promote early identification of failures (ideally before significant clustering has occurred) to retain safety margin
  - Progressively evaluate the extent of condition (this also helps to prove/disprove the working theory)
  - Enable lessons learned from initial inspections to be applied in developing future actions for plants that are perceived to be less susceptible

NSAL Recommendations

- **Tier 1a (4-loop downflow plants with Type 347 bolt design)**:
  - Complete a UT volumetric inspection of the baffle-former bolts at the next scheduled refueling outage
  - In preparation for this inspection, the plant should consider developing an ABPA and be prepared to replace any baffle-former bolts with visible damage or UT indications
  - Additional mitigation strategies include upflow conversion and preemptive bolt replacements
- **Tier 1b (4-loop downflow plants with Type 316 bolt design)**:
  - Complete a VT3 (visual) inspection of the baffle-former bolts at the next scheduled refueling outage
  - If any visual indications are found, it is recommended that the plant completes a UT volumetric inspection of the baffle-former bolts
  - If no visual indications are found, it is recommended that the plant completes a UT volumetric inspection of the baffle-former bolts prior to the completion of the second refueling outage after the issuance of this NSAL

NSAL Recommendations (cont'd.)

- **Tier 2a, 2b, and 2c (2- and 3-loop downflow plants)**:
  - Plants that have previously completed UT inspections should review the inspection records to identify any indication of the onset of clustering before the next scheduled refueling outage
  - Clustering is defined as 3 or more adjacent bolts or a total number of failures in a single baffle plate > 40% of the total number of bolts on that baffle plate
  - Any indication of clustering should result in the consideration of an accelerated re-inspection schedule
- **Tier 3 (Converted upflow plants)**:
  - 4-loop plants that have operated in a downflow configuration for more than 20 years should evaluate the need to perform a UT volumetric inspection of baffle-former bolts on an accelerated schedule
  - All other plants should follow the General Recommendations for all Tiers (see next slide)
- **Tier 4 (Designed upflow plants)**:
  - Follow the General Recommendations for all Tiers (see next slide)

NSAL Recommendations (cont'd.)

- **General Recommendations for all Tiers**:
  - If visually damaged baffle-former bolts or lock bars are detected, it is recommended that the fuel assemblies that were adjacent to the baffle in the previous cycle, and are scheduled for use in the next cycle, be inspected for fretting wear on the face that was adjacent to the baffle
  - It is recommended that the plant continues to follow the current MRP-227 guidelines and implement any revisions to the MRP-227 recommendations

Acceptable Bolting Pattern Analysis (ABPA)

- Methodology based on PWROG report WCAP-15030-NP-A, "*Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distribution Under Faulted Load Conditions*," February 1999.
- Acceptance criteria for MRP-227-A inspection of baffle-former bolts per WCAP-17096-NP-A, Rev. 2, "*Reactor Internals Acceptance Criteria Methodology and Data Requirements*," August 2016.
- Bolting pattern is evaluated to satisfy normal, upset and faulted condition design qualification allowables from the ASME Section III and functional requirements.
- Bolting pattern is evaluated using 1/8th symmetric models of the baffle former assembly.
- Edge bolts are not credited in the demonstration for structural acceptance of the baffle-former assembly.
- Although the large percentage and clustering of BFB failures at the 4-loop downflow plants recently examined exceeded the acceptance criteria, there remained margin such that these findings did not trigger a substantial safety hazard status under 10CFR Part 21.

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Evaluate pattern against ASME Section III and functional requirements

Westinghouse Replacement Bolts

- Replacement bolt design improvements to reduce impact of factors influencing bolt degradation
  - Cold Worked Type 316
    - Past operating experience using this material for baffle-former bolts and guide tube support pins has generally been positive. However, insufficient direct comparative data exists at this point to make a definitive conclusion
  - Semi-Parabolic head-to-shank transition fillet radius to reduce stress concentration and increase flexibility
    - Improved fatigue resistance
    - Reduced susceptibility to IASCC initiation
- Additional design improvements made for installation efficiency and ease of examination

Westinghouse

BFB Focus Group Activities

Industry Response

- The Industry Baffle-Former Bolt Focus Group (BFB FG) was formed in May 2016 to support an integrated approach among industry organizations to address recent operating experience
  - AREVA
  - EPRI
  - PWROG
  - Utility Staff
  - Westinghouse
  - Others
- Six focus areas with key actions defined

Focus Area	Lead Organization
#1 – /Extent of Condition, Interim Guidance, Technical Interfacing with the NRC	MRP
#2 – Plant/Fleet Operating Experience Assessment	PWROG
#3 – Repair/Replacement	PWROG
#4 – Inspection/NDE	MRP
#5 – Irradiated Testing Support	MRP
#6 – Aging Management Assessment	MRP

Industry Response

Joint EPRI/PWROG BFB Focus Group established to determine how MRP-227 guidance will change as a result of recent OE

- *Focus Area #1*: Cause and Extent of Condition, Interim Guidance
  - Develop BFB OE database and verify data
  - Evaluate/trend BFB OE
  - Develop Interim Guidance
- *Focus Area #2*: Plant and Fleet Operating Experience Assessment (PA-MSC-1473) (PWROG Lead)
  - Westinghouse to complete the 10CFR21 evaluation
  - Westinghouse to complete an NSAL; AREVA to produce a similar document for B&W plants (Customer Service Bulletin)
- *Focus Area #3*: Repair and Replacement (PWROG Lead)
  - Work with vendors to develop a contingency plan for tooling and bolt inventory for the upcoming outage seasons (Fall 2016 and Spring 2017)

Industry Response

- *Focus Area #4*: Inspection / NDE
  - Review bolt inspection protocols to see if lessons learned suggest modifications
  - Understand UT Probability of Detection as related to UT-acceptable bolts
- *Focus Area #5*: Irradiated Testing Support
  - Establish an integrated testing plan to build upon the Indian Point 2 root cause evaluation/analysis and further advance IASCC susceptibility knowledge
  - Evaluate the need to include Salem 1 and Ginna BFBs into an integrated testing program
- *Focus Area #6*: Aging Management Assessment
  - Review previous aging management assessments and compare to current OE experiences
  - Evaluate prediction models like the Weibull distribution in MRP-03 (which is based on French data)
  - **Long term functionality of MRP-227-A**
  - Has merged with Focus Area #1

Near Term Industry BFB FG Actions Completed

- Supported presentation to NSIAC on 5/23/2016
  - Westinghouse Technical Bulletin TB-12-5 remains valid
- Provided Industry Alert Letter from the PMMP Chairman to PWR site VPs on 6/1/2016
  - Expect that NEI 03-08 Interim Guidance will require the 4-loop plants identified in the Westinghouse TB-12-5 bulletin to perform UT inspections of all the BFBs or replace an acceptable pattern of bolts at their next outage.
  - Consideration should also be given to proceeding with procurement of replacement bolts prior to issuance of interim guidance due to potentially long manufacturing lead times.
- Westinghouse NSAL 16-1 issued 07/05/16 and revised 08/01/16
- AREVA CSB issued 07/14/16

Near Term Industry BFB FG Actions Completed

- Issued NEI 03-08 "Needed" Interim Guidance regarding BFB inspections for Tier 1 plants (7/25/2016) and for Tier 2 plants (9/29/2016) as identified in Westinghouse NSAL 16-1
- Assessed Fall 2016 and Spring 2017 outage seasons for developing a contingency plan for tooling and BFB material needs
  - Fall 2016: 3 planned MRP-227 UT inspections (1 of 3 is a Tier 1a plant) and 1 VT-3 inspection (Tier 1b plant)
  - Spring 2017: 2 planned MRP-227 UT inspections (both Tier 1a plants), 1 planned UT inspection (non MRP-227 but a Tier 1a plant), and 1 VT-3 inspection (Tier 1b plant)
- Initiated Hot Cell Post Irradiation Examinations of Indian Point 2 BFBs
  - Microscopic examinations have begun and are currently underway

Planned BFB FG Activities through Mid-2017

- Finalize BFB OE database by adding international data and UT inspection results from 2016-2017 exams in the US
- Continue with Hot Cell PIE work for IP2 and SAL1
- Explore providing additional NEI 03-08 Interim Guidance for the remainder of U.S. PWR fleet later in Fall 2016 or early 2017
- Establish fundamental understanding of BFB failure mechanism(s) and develop potential changes to MRP-227 inspection guidance as needed
  - Re-inspection frequency for UT exams
  - Allowance for proactive BFB replacement to manage aging

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PWROG
PWR Owners Group

The Materials Committee is established to provide a forum for the identification and resolution of materials issues including their development, modification and implementation to enhance the safe, efficient operation of PWR plants.