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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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US-APWR SUBCOMMITTEE

OPEN SESSION

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THURSDAY

OCTOBER 18, 2012

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ROCKVILLE, MARYLAND

The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 8:30 a.m., John W. Stetkar, Chairman, presiding.

SUBCOMMITTEE MEMBERS:

JOHN W. STETKAR, Chairman

J. SAM ARMIJO, Member

SANJOY BANERJEE, Member*

DENNIS C. BLEY, Member

JOY REMPE, Member

MICHAEL T. RYAN, Member

WILLIAM J. SHACK, Member

NRC STAFF PRESENT:

GIRIJA SHUKLA, Designated Federal Official
JAMES GILMER, NRO
HOSSEIN HAMZEHEE, NRO
STEPHEN MONARQUE, NRO
RUTH REYES, NRO
JEFFREY SCHMIDT, NRO

ALSO PRESENT:

CARL BEYER, PNNL
JOHN CAREW, BNL
DAVID DIAMOND, BNL
HIROSHI FUJISHIRO, MHI
HIROSHI HAMAMOTO, MHI
MASAYA HOSHI, MHI
MASAYUKI KAUCHI, MHI
ATSUSHI KUMAKI, MHI
CHIKARA KURIMURA, MHI
LOUIS C. LANESE, MNES
KEVIN LYNN, MNES
YUTA MARUYAMA, MNES
NOZOMU MURAKAMI, MHI
MASATOSHI NAGAI, MNES
TAKAYUKI NAKANO, MHI
JUNTO OGAWA, MHI

KEITH PAULSON, MNES
ETSURO SAJI, MHI
RYAN SPRENGEL, MNES
REBECCA STEINMAN, MNES
TAKAYUKI SUEMURA, MHI
KURT WALTER, MNES
ROBERT A. WEINER, MHI
DOUGLAS WOOD, MHI
DON WOODLAN, Luminant Power

*Participating via telephone

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P R O C E E D I N G S

8:30 a.m.

CHAIRMAN STETKAR: The meeting will now come to order. This is a meeting of the United States Advanced Pressurized Water Reactor Subcommittee. I am John Stetkar, chairman of this subcommittee meeting.

ACRS members in attendance are Dennis Bley, Sam Armijo, Mike Ryan, Bill Shack and Joy Rempe. I believe that Dr. Sanjoy Banerjee will be joining us on the bridge line at some time during the meeting. I'm not sure if he's on right now, but I've been told he will be joining us. Mr. Girija Shukla of the ACRS staff is the designated federal official.

The Subcommittee will discuss Chapter 4, Reactor, of the Safety Evaluation with open items associated with the US-APWR Design Certification and the Comanche Peak Combined License Application.

The Subcommittee will also discuss topical report MUAP-07008-P, Mitsubishi fuel design criteria and methodology; Topical Report MUAP-07010-P, non-LOCA methodology, and the staff Safety Evaluation Reports associated with those topical reports.

You will hear presentations from Mitsubishi Heavy Industries, Luminant Generation Company and the NRC staff.

We have received no written comments or requests for time to make oral statements from members of the public regarding today's meeting.

The Subcommittee will gather information and analyze relevant issues and facts and formulate proposed positions and actions as appropriate for deliberation by the full committee.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the Federal Register.

Parts of this meeting may need to be closed to the public to protect information proprietary to MHI or other parties.

I'm asking the NRC staff and the applicant to identify the need for closing the meeting before we enter into such discussion.

Only people with the required clearance and need to know are present. A transcript of the meeting is being kept and will be made available as stated in the Federal Register Notice.

Therefore, we ask that participants in the meeting use the microphones located throughout the meeting room when they address the Subcommittee.

The participants should first identify

themselves and speak with sufficient clarity and volume so that they can be readily heard. A telephone bridge line has also been established for this meeting.

To preclude interruption of the meeting, the bridge line will be placed in a listen-in mode during the presentation and during the discussions.

Please silence your cell phones during the meeting. We will now proceed with the meeting. We're going to hear first from Luminant Generation regarding Chapter 4 for their combined License Application, and I'll ask Steve Monarque if you have any questions - any opening statements.

MR. MONARQUE: Thank you, Chairman.

My name is Stephen Monarque. I'm the lead project manager for the Comanche Peak COL review. I want to thank the ACRS Subcommittee members for giving us the opportunity to present the self-reported Safety Evaluation to the Subcommittee here today. Last month, at the full committee meeting, we made progress - a lot of progress on the project.

And having said that, I want to introduce my branch chief, Hossein Hamzehee.

MR. HAMZEHEE: I am glad to be back.

CHAIRMAN STETKAR: You have to come to the microphone and speak with sufficient clarity and volume to be readily heard.

MR. HAMZEHEE: I thank you and -

CHAIRMAN STETKAR: You should remember this.

(Laughter.)

MR. HAMZEHEE: After three months, I forgot all the rules and regulations.

CHAIRMAN STETKAR: Don't say that. This is a public meeting.

MR. HAMZEHEE: I don't have anything more to say. I just want to thank the ACRS members and I'll be here for most of the day today.

CHAIRMAN STETKAR: Thanks, Hossein.

MR. MONARQUE: And with that, I'll go ahead and turn it over to Luminant for their presentation before the staff presents theirs.

CHAIRMAN STETKAR: Great. Don.

MR. WOODLAN: Good morning.

I'm Don Woodlan. I'm the licensing manager for the NuBuild project for Luminant and we're here to talk about Chapter 4.

(Discussion off the record.)

MR. WOODLAN: For the chapters I had first choice on, I picked this one. As you know from having reviewed it, this is probably the most straightforward chapter in our FSAR in that we are completely adopting the standard plant design.

And as you read through it, it says we incorporate by reference the material from the standard plant design. Here is my slide.

We do not take any departures from the standard plant and we do not provide any supplemental information. And we have no contentions pending in front of the ASLB Board.

I can elaborate, but that's it.

(Laughter.)

CHAIRMAN STETKAR: Brevity is appreciated.

Any of the members have any questions for Luminant?

MEMBER ARMIJO: Don, what is the degree to which you actually review - that Luminant staff have actually reviewed Chapter 4?

MR. WOODLAN: Well, when you -

MEMBER ARMIJO: I can understand where you say, look, this is a competent supplier, they've been in business, they know their stuff, it's okay, or do you actually dig into it a little bit?

MR. WOODLAN: I should probably mention a little bit what the Luminant staff is. The actual

Luminant employees, we have about a half a dozen people working in Dallas that are managing this project.

We've had more. When we started out, we had a larger staff. And at that point, we did review the entire range of products for the standard plant, as well as the Comanche Peak production.

And our staff really is much larger than just the Luminant employees, though. We have one full-time contractor who works directly with me in licensing, as well as MNES is contracted to us to provide engineering support as we go through the review.

So, although we reviewed these in depth up front, we continue to follow the progress of the entire standard plant design.

And in particular, we focus on the RAIs and questions that come up during the review especially if they relate to long-term operation of the plant or operability. We probably bring more experience with respect to that than a lot of the engineers already have.

So, we try to review it from that perspective, look at tech specs especially very closely, because we will be operating to those, and how we have to develop our operating procedures and programs associated with the standard -

MEMBER ARMIJO: But primarily more from an operator's point of view than a designer's point of view.

MR. WOODLAN: Yes, unless there's an issue.

MEMBER ARMIJO: Yes.

MR. WOODLAN: And when the issues come up especially in the RAIs, we do get involved depending on how much we think we can offer.

MEMBER ARMIJO: Okay. Thank you.

MR. WOODLAN: Okay.

CHAIRMAN STETKAR: Anything else for Don?

(No response.)

CHAIRMAN STETKAR: Don, I really appreciate you coming up and giving us a presentation. As you know, this is a necessary part of our deliberations. It's important for us to have the opportunity to ask you the questions and you have the opportunity to field them.

And with that, thank you.

MR. WOODLAN: Pleasure to be here. Thank you.

CHAIRMAN STETKAR: I guess we'll now hear form the staff on Chapter 4 on the COLA.

(Pause in the proceedings.)

CHAIRMAN STETKAR: Ruth.

MS. REYES: I'm waiting for the presentation to be loaded.

CHAIRMAN STETKAR: Oh. I think you have control.

MEMBER SHACK: This is a low-budget operation. This appears to be a very sophisticated situation -

MS. REYES: But it's not.

MEMBER SHACK: It's not. It's a low-budget operation. It's empowering. You have control over everything.

(Discussion off the record.)

MS. REYES: Okay. So, I apologize. I didn't know that. Okay. So, well my name is Ruth Reyes. I am the Chapter 4 chapter PM for the design certification and also the COL application review. And I'll be presenting the Chapter 4, Reactor.

On this page, the lead project manager is Steve Monarque. Like I said, I'm the chapter PM. And the supporting technical staff, same reviewers as for the design certification. They are all here in the audience in case the ACRS staff have any questions.

In the next slide I listed all the

sections. And like Luminant said, all the sections are being incorporated by reference with no departures.

And the only open item on the Safety Evaluation is because the evaluation of the design certification has not been completed. So, we are tracking that as an open item. So, that's the only open item on the Safety Evaluation. And that's all I have.

Any questions?

CHAIRMAN STETKAR: And that's a standard open item in every chapter of the COLA.

MS. REYES: Correct.

CHAIRMAN STETKAR: Any members have any questions for the staff?

(No response.)

CHAIRMAN STETKAR: If not, thank you very much for the summary.

MR. MONARQUE: And no open items for COL, okay.

CHAIRMAN STETKAR: No open items.

MR. MONARQUE: No action items from ACRS members.

CHAIRMAN STETKAR: Try as we might.

MR. MONARQUE: Okay. Thank you.

CHAIRMAN STETKAR: This one went very, very well. Thank you.

And with that, we'll start the discussion of Chapter 4 from the design certification where I'm confident we'll actually have a more active discussion.

So, I ask MHI to come up and get your slides up for that.

(Pause in the proceedings.)

(Discussion off the record.)

CHAIRMAN STETKAR: All right. We are running uncharacteristically ahead of schedule. So, I don't see Jeff Ciocco over there. I don't know whether -

MR. HAMZEHEE: Jeff is out. Steve was acting for Jeff, and then I am Jeff's backup.

CHAIRMAN STETKAR: Okay. Yes, that's appropriate defense in depth. I don't know if NRC management wants to make any introductory remarks, Stephen?

MR. MONARQUE: No, I just want to thank again the Committee members for allowing this opportunity to present the DCD to the Subcommittee.

And do you have anything else, Hossein?

MR. HAMZEHEE: No.

MR. MONARQUE: And with that, I'll go ahead and turn it over to MHI.

CHAIRMAN STETKAR: Great. Thank you very much. So, we'll turn the meeting over to MIH, and I don't know who is leading the discussion.

MR. LANESE: I am. Good morning. My name is Lou Lanese. I am the lead chapter engineer for Chapter 4 for MNES. And we have our panel of experts here who will introduce themselves.

MR. NAGAI: My name is Masatoshi Nagai with MHI.

MR. NAKANO: Good morning. My name is Takayuki Nakano, MHI, fuel engineer.

MR. MURAKAMI: Good morning. My name is Nozomu Marakami. I am also a fuel engineer.

CHAIRMAN STETKAR: Thank you. And I'll ask you, please, make sure you speak up enough so that we pick you up. The transcript for the meeting comes directly from the microphones. So, we need to make sure we hear you fairly clearly. Thank you.

MR. LANESE: Okay. Well, thank you for having us here. We will be going through, obviously, Chapter 4 section by section and providing a summary of the information that's there and the open items for each section.

There are only 17 open items that are identified at this point and they all belong to Sections 4.2 and 4.4. So, we will address each one of those hopefully fairly briefly.

Section 4.1 of the DCD is a summary of the US-APWR fuel design. It is essentially very similar to other large PWR designs, except that it's at a higher power level and the active fuel length of 14 feet allows it to have a lower linear heat rate.

So, we have improved thermal margins and we achieve the larger power level with more rods and more active fuel length.

MEMBER ARMIJO: What are your peak powers? An average linear 4.6 is certainly very conservative, but, you know, what's the variation from average to peak?

MR. LANESE: The variation of average to peak power.

MEMBER ARMIJO: Yes, because an individual fuel rod, of course its performance is controlled by is peak power. So, what would the peak power be, your design peak power for a fuel rod in this core?

MR. NAGAI: Okay. Is the question about FQ or F delta H?

MEMBER ARMIJO: No, just kilowatts per foot

- peak kilowatts per foot for fuel element in this core.

MR. NAGAI: In US-APWR -

MEMBER ARMIJO: Please come up to the -

MR. KAUCHI: My name is Masayuki Kauchi. I'm responsible for nuclear engineering. I wasn't sure I heard the question, but I think your question is maybe relative peak for maybe the power design limit for the relative peaking is fuel rod power. It is 1.73.

So, if you can just multiply --

MEMBER ARMIJO: So, it's 1.73 times 4.6.

MR. LANESE: Right.

MR. KAUCHI: This will be fuel rod average load.

MEMBER ARMIJO: Okay.

MR. KAUCHI: If you would like to know the local peak power, then FQ is 2.6.

MEMBER ARMIJO: So, local peak power would be 2.6 times this average.

MR. KAUCHI: Yes.

MEMBER ARMIJO: So, basically nodal peak power was 2.6 times 4.6 -

MR. KAUCHI: Yes.

MEMBER ARMIJO: - as a design basis.

MR. KAUCHI: Yes.

MEMBER ARMIJO: Okay. I'll be busy multiplying here.

(Laughter.)

MEMBER ARMIJO: This may take a while. I was not good at math. Thank you. That's all I need to know right now.

CHAIRMAN STETKAR: I have a spreadsheet that's 11.96, or 12 for rounding.

MEMBER ARMIJO: See, that's what I was trying to get at. These peak powers are what controls some of the performance limits.

MR. LANESE: And aside from the, you know, the 14-foot active fuel length, the only unique feature is that the core has a stainless steel radial neutron reflector.

The other design features are all very similar again to existing US designs; fuel enrichment, the number of control rods, the type of burnable poison and the CRDMs are conventional materials.

MEMBER ARMIJO: Your gadolinia is higher than past practice. It's 10 -

MR. LANESE: 10 percent is the maximum.

MEMBER ARMIJO: It's 10 percent. Do you have experience, operational experience with that high gadolinia content in your Mitsubishi fuel?

MR. NAKANO: It's 10 weight percent in Japanese plant.

MR. MURAKAMI: Yes, this is Murakami speaking. We have experience to use our - such high gadolinia content. We have since 2004 in commercial reactor in Japan.

MEMBER ARMIJO: Okay. And performance has been good?

MR. MURAKAMI: There is no significant difference among the 10 percent or eight percent or six usually use gadolinia content. It's not so far away from us.

MEMBER ARMIJO: Okay. Thank you.

MR. LANESE: The next slide shows the calculation approach that are being used in support of the application, the two topical reports.

The FINE topical report 008 is going to be discussed this afternoon. The FINDS report on the seismic response is scheduled for discussion in January and we don't intend to be talking about the structural response today. This is, you know, core design.

And then there are - the PARAGON report provides the nuclear constants for the core, and the hot channel thermal hydraulics is done with VIPRE and described in a technical report.

MEMBER ARMIJO: Of these four topical reports, which are the ones that are already approved? The table says, approved or under review.

MR. LANESE: Yes, 008 and 034 are still being reviewed.

MEMBER ARMIJO: Okay.

MR. MURAKAMI: Yes.

MR. LANESE: And the technical reports of PARAGON and VIPRE, are they approved yet?

MR. NAGAI: My name is Masatoshi Naga, MHI. The technical reports for PARAGON/ANC is a part of Section 4.3. So, it has not been approved yet, but it's being used.

CHAIRMAN STETKAR: Are those - I'm trying to look up my references. Are they technical reports, or topical reports?

MR. LANESE: The last two are technical.

CHAIRMAN STETKAR: They're technical. So, they'll be folded in with the staff's review -

MR. LANESE: Correct.

CHAIRMAN STETKAR: - of the respective sections. The topical reports get a separate review.

MR. NAGAI: Just to clarify, for FINE and

FINDS we have -

CHAIRMAN STETKAR: Those are topical.

MR. NAGAI: And for VIPRE-01M, we have all the topical report.

CHAIRMAN STETKAR: VIPRE is a - that's what I thought. VIPRE is a topical report.

MR. NAGAI: But for PARAGON/ANC we have technical report.

CHAIRMAN STETKAR: So, that's folded in supporting information for the respective chapter. And VIPRE has not yet been approved? Maybe I'll look to the staff.

MS. REYES: The two topical reports, we are presenting one today. The other two topical reports we're presenting in January.

Right now the review is done. It has been concluded, but is in concurrence and we haven't issued the SE for those topical reports.

CHAIRMAN STETKAR: Great. Thank you.

MEMBER ARMIJO: So, which is the approved topical report?

CHAIRMAN STETKAR: There aren't any.

MEMBER ARMIJO: There aren't any?

CHAIRMAN STETKAR: No.

MEMBER ARMIJO: So, they're not yet

approved.

CHAIRMAN STETKAR: We'll be hearing in January -

MEMBER ARMIJO: Okay.

CHAIRMAN STETKAR: - the review of FINDS - let me get my - you have to excuse me because I can't see two things at once here. I'm old.

VIPRE and FINDS in January. And we'll be hearing about FINE this afternoon.

MEMBER ARMIJO: Okay. Very good.

MR. LANESE: There are no open items in Section 4.1, which is the summary description of the core.

Section 4.2 deals with fuel system design. The concept in the US-APWR fuel is to provide flexible core operation by having the large core and the high gadolinia content.

97 percent theoretical density in the fuel allows a higher fuel economy. And the zirc-4 grids are proving grids and provide high neutron economy.

Reliability for this fuel is improved with corrosion-resistant material. And the one unique feature on this - shown on this slide is that there are 11 grids for the 14-foot fuel assembly as opposed to 10 grids with the - with the assemblies that - 14-

foot assemblies you've reviewed previously.

CHAIRMAN STETKAR: You'll find that we're adept at interrupting you.

MR. LANESE: Oh.

CHAIRMAN STETKAR: So, if there's silence, the appropriate strategy is to charge through as fast as you can.

(Laughter.)

MEMBER ARMIJO: You didn't go quite fast enough.

CHAIRMAN STETKAR: See, you didn't go fast enough.

MEMBER ARMIJO: I have just one question, and you may get to it later or in the presentation. There is an open item related to the grid structural integrity and deformation of the grid under some seismic loads.

Are we going to discuss that later in your presentation?

MR. LANESE: There's an open item associated with that and I think the staff is probably going to discuss it as well, but we'll have a chance to talk about that, yes.

MEMBER ARMIJO: Okay. Well, I'll just withhold until you're ready.

MR. LANESE: Okay.

The next slide just highlights all of the things we've said previously. This is very similar to fuel that you have seen before. The big difference, I guess, is having an additional grid spacer.

We do have the 10 percent gadolinia content and the higher theoretical density for the fuel.

This section meets the requirements of the Standard Review Plan 4.2 in terms of content. We addressed the GDC-10 reactor design to show that we don't damage the fuel. 17 to provide an appropriate reactivity control from both rods and from ECCS.

And GDC 35, emergency core cooling, we demonstrate that we have abundant cooling so that there's no fuel damage, and 10 CFR 50.46.

The Standard Review Plan deals with system fuel damage, rod failure and fuel coolability. System damage means that the fuel is - would prevent all the mechanisms listed on that slide by properly designing the fuel so that during power operation none of those things will occur. And, again, these are the standard items that are addressed in the SRP.

In terms of fuel failure, these are not addressed directly in Chapter 14. But, again, the fuel design assures that during accidents and transients that you don't have fuel failure as a result of any of these mechanisms such as hydriding or cladding collapse or embrittlement, fuel rod ballooning, et cetera.

This slide shows the relationship of the various codes and topical reports and technical reports that are being used to support Chapter 14.

And, again, as we mentioned earlier in the discussion, our focus is on the fuel design itself, not on the structural design of the assemblies or the core.

MEMBER REMPE: I have a question.

MR. LANESE: Yes.

MEMBER REMPE: When I was looking through some of the documents, maybe this is obvious, but this is a topical report, the staff is reviewing it and the staff and the intended MHI is not to just have FINE approved for the design certification. It's for use in subsequent issues such as there's licensing issues you're going to apply FINE to address it in the future.

And if so, is the staff reviewing the QA? Because I know like in tomorrow's documents, they talk about the QA with MARVEL, but I didn't see that in the documentation in exchanges from the staff.

Maybe I missed it. There's a lot of information we were given for the FINE code.

MR. SPRENGEL: This is Ryan Sprengel, MNES. So, first I'll answer our question that, yes, they are topical reports and I think there is potential in the future for Mitsubishi to apply these codes to other fuel outside the US-APWR.

MR. SCHMIDT: This is Jeff Schmidt from the NRC. We're only approving FINE for the US-APWR fuel.

MEMBER REMPE: For design certification, or can Constellation use it for future licensing issues that come up?

MR. SCHMIDT: I'm not sure I understand the -

MR. SPRENGEL: Well, let me clarify. Right now Mitsubishi is only requesting it be approved for US-APWR.

MEMBER REMPE: For design certification -

MR. SPRENGEL: Correct.

MEMBER REMPE: - or for our future - for just whenever there's another issue?

MR. SPRENGEL: Well, I guess if Constellation built a US-APWR, they would use it. But if they do not have a US-APWR, then they would not use it right now.

MEMBER REMPE: Right. That's what I'm asking, yes.

MR. SCHMIDT: That's correct, if that's what you're asking is -

MEMBER REMPE: And so, you've gone through this -

MR. SCHMIDT: - only for the US-APWR.

MEMBER REMPE: - going to be QA controlled and all those kind of things are going to be part of the staff's review on how they might use it in the future that - there were a lot of issues with MARVEL, more than what I saw with FINE, where certain options were invoked, coefficients were changed to do this and that and the other.

And I didn't see that so much with FINE, but I just am wondering how good the pedigree and the control of it would be. And that's always part of the staff's review.

MR. SCHMIDT: Right, yes.

MEMBER REMPE: Okay.

MR. SCHMIDT: Again, it's just for the US-APWR for Constellation uses it for - on the US-APWR.

MEMBER REMPE: Right.

MR. SCHMIDT: They can't use it for

anything else.

MEMBER REMPE: Right. Again, I understand that. But I'm just thinking five years down the road it will be controlled carefully, is what I'm asking.

MR. SCHMIDT: They'll have to come back five years down the road.

MEMBER REMPE: Okay.

MR. SPRENGEL: And let me expand upon that a little bit. We have had inspections of our QA program. So, that has been looked at in the broad sense, and then there have been some specific targeted areas as well that have been looked into.

MEMBER REMPE: Okay. Thanks.

MR. SHUKLA: Yes, and just for the record it's Comanche Peak, not Constellation.

MEMBER REMPE: I'm sorry, did I say the wrong -

CHAIRMAN STETKAR: Luminant.

MEMBER REMPE: Luminant, I'm sorry.

MR. SPRENGEL: I thought Constellation was a good point because -

(Laughter.)

MEMBER REMPE: Sorry about that.

MR. SPRENGEL: Maybe you knew something we didn't. I don't know.

MEMBER REMPE: Okay.

CHAIRMAN STETKAR: Consider it as marketing, but not all that effective.

MEMBER SHACK: Maybe this is a good time to ask you a little bit about FINDS, although you say that we're not here to discuss that today. But I was just curious since it's fuel inelastic deformation under seismic condition, you obviously plan to have this thing deform plastically under the SSE, which is what it seems to do. It does buckle. It take it that's standard Japanese design practice.

What assures me that I have margin beyond the SSE for this shutdown? I normally think that most of our seismic design sort of takes you to elastic. And then when you go plastic, that sort of gives me extra margin. Now, I'm going plastic right at the SSE.

As I try to read 08007, it seems to me there's some sort of geometric limit. You talk about maximum credible deformation.

Is that the argument that although it deforms plastically, it's limited. And so, you still have margin beyond the SSE, or that's not something to be discussed?

MR. LANESE: I think until the analysis is reevaluated by MHI, that it's better to discuss that in January when they're sure what it is they want to present.

MEMBER SHACK: To do, okay.

MR. LANESE: Okay.

MR. SPRENGEL: That's a fair question and we'll take that on for our future discussions. In January, we should be discussing FINES being the methodology.

But as you've said, you know, we've communicated to the staff about new seismic inputs that are underway related to 3.7 and 3.8, and those will be incorporated into the results technical report that shows what happened.

So, I think we can have discussions in the future and we'll be aware of that for our preparation.

MEMBER ARMIJO: Now, is the same issue, does it exist in Japanese plants where you have deformation of the grids in an SSE?

MR. LANESE: In the Japanese plant, do we have plastic deformation under SSE condition?

MR. MURAKAMI: In general terms, or is - if very high seismic or acceleration, is assumed that such case is considered a case in Japan.

I don't have any specific -

MEMBER SHACK: But very high means SSE, or beyond SSE?

MR. MURAKAMI: Very high meaning beyond SSE.

MEMBER ARMIJO: Well, my question was limited just to SSE. In the Japanese plants, first of all, do you use the same grid subject to similar or identical loads, or not?

MR. SPRENGEL: Real quick, I have one note. This is Ryan Sprengel again.

When we're talking about the buckling that we've seen in our analysis, that is based on nonexistent sites.

So, when we look at site-specific analysis, the results are different. So, we have a very, you know, conservative spectrum of analysis and site conditions - or, sorry, site parameters that we're assuming in the DCD for the standard design. So, that is a factor in it.

MEMBER SHACK: But since the code is fuel inelastic deformation under seismic condition, it sort of sounds like you're designing for it to go plastic.

MR. SPRENGEL: Right, and then we've done further evaluations for what happens after that deformation. That is correct, right.

And we've looked at the effects and looked at how we can handle -

MEMBER SHACK: But you do design it to go plastic, is I think where Sam and I are coming from, yes.

MEMBER ARMIJO: That's what we're trying to find out.

MR. SPRENGEL: Okay.

MR. LANESE: And the original question was in the Japanese plants, do you predict plastic deformation of the fuel?

MEMBER ARMIJO: Exactly.

CHAIRMAN STETKAR: At the SSE.

MR. LANESE: At the SSE.

MR. NAKANO: Excuse me. This is Nakano speaking. That kind of information includes proprietary in Japan. So, we cannot answer the question at this time.

And also the member here, MHI member here --

MEMBER ARMIJO: Mr. Nakano, could you speak a little louder or with the microphone closer?

MR. NAKANO: Sorry.

MEMBER ARMIJO: We can barely hear you.

MR. NAKANO: The question - your question includes proprietary information in Japan. So, we cannot answer that here.

And also, MHI engineer here doesn't have such specific information. So, if you want an answer, yes, we can respond, but we should ask to information to Japan.

CHAIRMAN STETKAR: That would be fine.

MEMBER ARMIJO: So, we table it until January?

CHAIRMAN STETKAR: Yes. Either table it until January or - I don't know whether the January meeting will be closed.

MR. SHUKLA: There's no meeting in January though.

MR. NAKANO: We will have meeting. And regarding with seismic information and the grid information, we will have a meeting in next January.

CHAIRMAN STETKAR: Yes, let's discuss the schedule later off line. For the purposes since the schedule seems to be in question, I'll - what I'll do is I'll ask MHI to - if this is proprietary information, can you just use it as a takeaway and communicate with us separately?

We can keep the proprietary information proprietary that way.

MR. SHUKLA: Or if they have information, they can provide this afternoon or tomorrow morning. That will be both proprietary sessions.

CHAIRMAN STETKAR: Oh, okay.

MEMBER ARMIJO: This afternoon is proprietary?

MR. SHUKLA: Yes.

CHAIRMAN STETKAR: Oh, okay. If you have the information -

MEMBER ARMIJO: We can discuss this afternoon.

CHAIRMAN STETKAR: Or tomorrow.

MALE PARTICIPANT: They have to go back to Japan, I think.

MR. SPRENGEL: Okay. We'll look into that and follow up.

CHAIRMAN STETKAR: Look into it. I mean, if we can get it closed out either today or tomorrow morning, anyway, that would be more timely, anyway. Otherwise, we'll decide how else to communicate.

MEMBER ARMIJO: What I'm really trying to get at is this is - is this grid unique to the US-APWR fuel and very, let's say, optimized for economy, but somewhat fragile, or is this grid consistent with current Japanese practice that it's - or in use today in Japanese plants?

If it is in use today in Japanese plants, then they must be designed for some plastic deformation if they're doing - if it applies to the APWR fuel.

So, I'm just trying to find out - I'm looking for experience with this particular grid and this design practice.

MR. NAKANO: We are looking for to find that information.

MEMBER ARMIJO: Okay.

MS. REYES: This is Ruth Reyes. There's just one issue that I would like to clarify, because I think I believe the staff said that we can discuss this in January.

In January, what we are presenting is the topical report that discussed the seismic code. But in reality, this information is for the technical report and the staff won't get the technical report until later in 2013.

CHAIRMAN STETKAR: That's part of Chapter 3.

MS. REYES: Well, it's really Chapter 4 pertaining to the report, which is 0809, but we won't get the updated or the revision of that report until later in 2013.

So, in January won't be ready to discuss the issue. Again, what we are discussing, January is the topical report that discuss the code.

CHAIRMAN STETKAR: Understand that, okay. So, for clarification we won't really hear about the design applications until the next iteration on your review; is that - of Chapter 4; is that correct?

MR. SCHMIDT: This is Jeff Schmidt with the NRC.

Yes, it's really going to be under Technical Report 08007, which right now I think the preliminary date is July of 2013.

CHAIRMAN STETKAR: Okay. And the next time that we'll actually be presented with that information will be in the next phase of Chapter 4 from our purpose; is that correct?

MR. SCHMIDT: Yes, it would effectively, you know, we have that as an open item. You'll see it in our presentation.

And so, that would just maintain as an open item until we're successfully able to close it in sometime next year.

CHAIRMAN STETKAR: So, just for logistics here, that means that we won't have a formal information about that part of the design with respect to the US-APWR certified design until our next phase of the review.

That doesn't leave MHI off the hook for the question about what's done in Japan. So, I'd appreciate some feedback on that.

MR. SPRENGEL: I think we can address these items and we will have a revision to the report. And so, I think that will be made available on - and I think we were speaking to the present planning.

If there's some additional meeting that's needed, I mean, we can do that. That's something that we can look at after we provide the revision to the results.

CHAIRMAN STETKAR: Thank you.

MR. LANESE: So, the open items, both of them on this page are associated with the FINDS topical report and with the technical report that we have just discussed. And that does show January and July 2013.

And the other open item which has been closed and resolved by the staff was the question relating to the use of thousand-hour fretting data and how it was applied to the fuel.

MHI responded in July. And as I said, the staff has indicated to us that that's closed and resolved.

There was also a question associated - an open item associated with flow-induced vibration in the fuel.

MHI responded and indicated that the excitation for fatigue failure was several orders of magnitude difference than the actual excitation in the fuel, the amplitude in the fuel. And staff hasn't closed that item yet. I believe it's still under review.

There was also a question associated with the buckling loads on the fuel and whether it was worse at beginning or end of life.

That RAI was also responded to and no changes of the DCD. It was an expansion, an explanation of the methodology and staff is reviewing that response.

These next few items are related again to the topical reports - or technical reports, excuse me, on the structural response of the fuel. So, they will be dealt with in the future.

The next open item had to do with liftoff of the fuel under hydraulic forces and what was the limiting condition.

MHI responded indicating that the limiting condition for fuel liftoff was using the mechanical design flow during their cold startup condition. And that's also under review by the staff.

There was another question related to the hydraulic forces on the fuel. MHI responded explaining how they calculated those forces using beam theory and test data. And that response is under review by the staff.

The next open item had to do with hydrogen absorption and the control rod guide thimble and what the appropriate limit should be. MHI responded and that's a closed/resolved issue as indicated by the staff.

The next one is also a closed/resolved item related to the burnable absorber, the amount of gas released. It had to do with when the gas was released and how long that absorber might be - burnable absorber rod might be in the core. And MHI responded and that issue was resolved.

Open Item Number 12 was related to stresses on the control rod cladding. Additional clarifying information was provided and staff is reviewing that response.

There was a question about the effect of a tenth of a second delay in the control rod insertion time and if that was accounted for.

MHI responded and indicated that the actual rod time - insertion times to dash pod entry versus the assumptions in the safety analysis report were conservative.

The rods insert in about 1.6, 1.7 seconds. The safety analysis assumes three-second insertion time. So, there's a large margin which is - that tenth of a second additional delay would be incorporated in that margin.

Now, the last open item had to do with ensuring core coolability during a LOCA. Staff has issued that question and MHI is preparing a final response for that.

MEMBER ARMIJO: Okay. Could you go back to your Slide 21?

MR. LANESE: Which open item are we -

MEMBER ARMIJO: The hydrogen absorption limit in the guide thimble. Yes, that one. Now, that's closed and the staff is satisfied with the response, apparently.

So, can MHI tell us why, you know, what justified that limit? I'm just trying to understand why it was closed and -

MR. LANESE: The 800 ppm limit isn't the one that was eventually used. It was adjusted to --- ppm to be consistent with the limit on the fuel.

And then if you need additional -

(Discussion off the record.)

MR. LANESE: All right. The rest is proprietary. We will discuss that in -

MEMBER ARMIJO: Okay. So, let's just hold that question until we get into the afternoon session and maybe you can answer it then, but I'd like to understand exactly how it was closed and what was adjusted.

MR. LANESE: We have some additional slides.

MEMBER ARMIJO: Okay.

CHAIRMAN STETKAR: For your information, we don't always see all of the RAIs and responses, because we - if we did that, we'd never sleep.

So, some of the questions that you'll hear from us if you haven't been in these sessions before, may elaborate on information that you've already presented in an RAI response.

We just don't typically see those unless we specifically request them. If we ask the staff, they would send them to us.

MEMBER ARMIJO: We have a bunch of them.

CHAIRMAN STETKAR: We don't necessarily want to see all of those.

MEMBER ARMIJO: Just some of them that get our interest and we want to understand how it was closed.

CHAIRMAN STETKAR: Well, we can bring this up this afternoon -

MEMBER ARMIJO: Sure.

CHAIRMAN STETKAR: - in the closed session.

MR. LANES: Okay. And that would close out all the open items on 4.3. And we can go to Section - I'm sorry, we're starting 4.3. We just closed out 4.2.

(Discussion off the record.)

CHAIRMAN STETKAR: Any members have any other questions on 4.2?

MEMBER ARMIJO: Well, I have a number of questions on the 07008 report, MUAP-07008-P, which is the FINE.

CHAIRMAN STETKAR: Yes.

MEMBER ARMIJO: We're going to talk about that this afternoon?

CHAIRMAN STETKAR: That's this afternoon.

MEMBER ARMIJO: Okay. So, I can just hold off, yes.

CHAIRMAN STETKAR: That's this afternoon. Otherwise he'll just say it's all proprietary and we'll talk about it this afternoon anyways.

MEMBER ARMIJO: Okay, okay. We'll just wait.

(Pause in the proceedings.)

MR. LANESE: Do you want to introduce yourself?

MR. KAUCHI: Again, my name is Masayuki Kauchi and I'm responsible for nuclear designing. Thank you very much for taking time for us today.

CHAIRMAN STETKAR: And, again, I'd ask you to make sure you speak up enough so that we pick you up on the microphone so that the transcript is accurate. Thank you.

MR. LANESE: So, the intent of the nuclear design is to provide limits on the power distribution so that we don't damage fuel during the power operation, assure that we have reactivity coefficients that protect the fuel, and assure that we have control systems that can maintain appropriate reactivity and prevent any power oscillations during normal operation.

The low power density of these assemblies allows a 24-month operation. And the power distributions during normal operation are maintained with design limits. And those limits are used in the initial conditions for the Chapter 15 safety analysis. So, there is the 2.6, you know, factor we were talking about earlier.

The next slide just illustrates the fact that with the use of burnable poison, the moderator and temperature coefficients are negative throughout the entire operating cycle.

This illustrates the rod patterns for a quarter quadrant and we maintain shutdown margin 1.6 percent with most reactive rod stuck out.

In terms of power pertubations, the horizontal oscillations are stable throughout the fuel cycle. And the axial oscillations while they are divergent during cycle burnup, are slow enough that they can easily be controlled with rod motion similar to existing large APWRs. The frequencies are just a little different. Nothing dramatic.

And there are no open items in Section 4.3. I take it I should move on rather than -

MEMBER ARMIJO: You're on a roll.

MR. LANESE: Okay. We can go to Section

4.4, thermal hydraulic design. We're going to make another change here.

(Pause in the proceedings.)

MR. SUEMURA: Good morning. I am Takayuki Suemura, MHI. I am a thermal hydraulic engineer.

MR. HOSHI: Good morning. My name is Masay Hoshi, thermal hydraulic engineer right now in MNES.

CHAIRMAN STETKAR: Before we start this section of the discussion, one of our members, Dr. Banerjee, may have some questions in this area. I don't know whether he does or not.

I've received a note that he'll be calling in on our bridge line around ten o'clock. So, it's just a warning that we may go back and revisit some of this section once he comes online.

I don't know if he has any questions or not, but he's our thermal hydraulics expert and he had other commitments. He couldn't make it to the meeting today and will be calling in.

So, this is just a bit of a warning that we may have to retract over this part of the -

MR. LANESE: Would you prefer that we went to 4.5 and 4.6 and come back, or just continue on?

CHAIRMAN STETKAR: You know, that might be a good idea to keep the process moving. That's an excellent idea. Let's do that. That will buy us some time and -

MR. LANESE: Okay.

CHAIRMAN STETKAR: - we might take an early break then and give us a little bit of -

(Laughter.)

MR. KURIMURA: So, my name is Chikara Kurimura, mechanical engineer of MHI for 4.5 and 4.6.

MR. LANESE: Sections 4.5 and 4.6 of the DCD describe the materials that are used in the control rod drive mechanism and in the reactor internals, and then refer to other sections for any detailed evaluation.

The earlier sections in the DCD would refer to the technical and topical reports instead. So, that's just a little difference in the way they're structured.

But the materials in the control rod drives and in the internals are materials that have been used in existing PWRs for long periods of time and there's extensive operating experience with them.

The control rod drive system, the pressure boundary materials use either Type 316, Type 316-LN. The welding materials use austenitic steel and appropriate welding specifications.

The non-pressure boundary materials may use austenitic, martensitic, nickel-based alloys, and, in some cases, cobalt-based alloys.

Reactor internals were chosen obviously to prevent any existing problems with - in operating plants with corrosion. So they, again, use the strain-hardened materials, nickel alloys. It complies with the ASME code and with any of the ASME code cases spelled out and approved in Reg Guide 1.84.

And long-term reliability from irradiation is assured by complying with the guidelines developed in EPRI MRP 175 for screening for aging of components.

MEMBER SHACK: I did have a question about the neutron absorber that you essentially make up, I guess, from rings which are then held together with tie rods.

Where are the tie rods at? Are they through the rings? Are they on the outside of the rings?

MR. KURIMURA: Through the ring.

MEMBER SHACK: Through the rings. So, they're not visible. They're not open for visual inspection.

MR. KURIMURA: There are some holes around tie rod sort -

CHAIRMAN STETKAR: You have to be at the microphone, but you can use the mouse to point. It's a lot easier.

MR. KURIMURA: So, the tie rod goes through the hole. So -

MEMBER SHACK: Well, I guess let me tell you where I'm heading from this question, you know. In one of the RAIs you give the expected fluence at the inside of that absorber, which again you have in brackets. So, I'll assume I can't say it out loud.

But then you note that it's going to decrease by an order of magnitude by the time you get to the tie rods. But even if I take that bracketed number and I reduce it by an order of magnitude, then I look at 175. That doesn't give me a warm, fuzzy feeling.

So, I would like to be able to inspect these. But if they're inside the rings, then how am I going to inspect them?

MR. KURIMURA: Over the estimated effect. I can't say the value of that, because -

CHAIRMAN STETKAR: Just a moment, please.

(Discussion off the record.)

CHAIRMAN STETKAR: If it requires some discussion, you can discuss it and answer it after the break or this afternoon sometime.

MR. SPRENGEL: Because I think the basic question is just, can you inspect them?

MEMBER SHACK: Right, because I think they do need to be inspected.

MR. LANESE: Based on the fluence level.

MEMBER SHACK: Based on the fluence level that appears to be there.

MR. KURIMURA: So, the fluence level is discussed in RAI, but I can't say it now because it's proprietary information. And inspection is area of 3.9.5.

MR. LANESE: So, the inspection of those bolts is described in 3.9.5?

MR. KURIMURA: So, discussion of that -

MR. SPRENGEL: We'll take notes and can explain it to you later.

MEMBER SHACK: Okay.

MR. LANESE: All right. There were no open items identified in Section 4.5 of the DCD otherwise.

MEMBER ARMIJO: The bottom line is you're not using any unique materials that you haven't used before in the - in Japanese plants; is that correct?

MR. KURIMURA: We use approved material for it.

MEMBER ARMIJO: I had a question about the strain-hardened austenitic stainless steels which are for obviously high-strength applications.

MR. KURIMURA: For the fourth one in the reactor internals.

MEMBER ARMIJO: And you've used that at the same level of strain-hardening in the past?

MR. KURIMURA: Yes.

MEMBER ARMIJO: So, it's common use in Japanese plants?

MR. KURIMURA: Yes.

MEMBER ARMIJO: I just wanted to -

MR. SPRENGEL: I'll reiterate the point that we're not trying to apply any revolutionary materials at this point.

MEMBER ARMIJO: That's just a general question I -

MR. SPRENGEL: They've all been -

MEMBER ARMIJO: - often like to ask in case there's anything new and unique.

MR. SPRENGEL: I understand.

MEMBER ARMIJO: Okay.

MEMBER SHACK: And they have set up a peak stress on that that's consistent with the ASME code.

So, it's -

MEMBER ARMIJO: And is that the same stresses used on those bolts through the -

MEMBER SHACK: I can't find out anything about those tie rods. I don't know where they are. I don't know what they look like.

MEMBER ARMIJO: What material?

MEMBER SHACK: How they're loaded, they're a mystery to me. But the only thing I can guess is that they probably are susceptible to the radiation-assisted stress corrosion cracking from a fluence point of view.

MEMBER ARMIJO: Sure.

CHAIRMAN STETKAR: Would it help with saying how many drawings that they present and -

MEMBER ARMIJO: Yes.

MEMBER SHACK: Yes, I couldn't find a drawing anywhere that showed me these things.

CHAIRMAN STETKAR: Even if you could dig it up, you know, tomorrow.

MR. SPRENGEL: We'll follow up on that and see if we can provide something today.

CHAIRMAN STETKAR: Thanks.

MR. LANESE: Okay. Well, then the next section would be on functional design of the reactivity control systems.

Again, this is standard design that has been used in other operating PWRs. There are two forms of reactivity control. One is the rods, and the other is the chemical reactivity control injected by the ECCS.

The control rod drive mechanisms are described in the DCD. The control system is described in Section 7. And the CRDM cooling is described in Section 9.4. ECCS is in 6.3.

The only other point about this section is that the CVCS is not credited in any of the Chapter 15 safety analyses for reactivity control.

And there were no open items in Section 4.6. So, we're back to 4.4.

CHAIRMAN STETKAR: And I think, you know, just in the interest of efficiency, what I'll do is take an early break. Nobody ever complains about taking a break. Just so we allow Dr. Banerjee to come online and avoid possibly repetition of information.

So, what I'll do is let's take a break until ten o'clock. I'll be really generous about this. We're well ahead of schedule.

MEMBER ARMIJO: Thank you, Mr. Chairman.

CHAIRMAN STETKAR: So, we'll recess and -

MR. SHUKLA: Do we need to tell Banerjee about -

CHAIRMAN STETKAR: Yes, give him a call.

MEMBER BLEY: He sent an email. You've got it on your computer saying -

CHAIRMAN STETKAR: We can communicate with him. We're recessed until ten o'clock.

(Whereupon, the proceedings went off the record at 9:37 a.m. for a brief recess and went back on the record at 10:03 a.m.)

CHAIRMAN STETKAR: Okay, we're back in session. I believe that Dr. Banerjee will join us in five to ten minutes, but I thought - first, I thought I heard that you may have some clarifications on information that was - for information that was raised earlier.

MR. SPRENGEL: Right. A couple items just to do some immediate follow-up. Some of them we'll defer, but first item tied to the question of do we design for plastic deformation.

We're going to cover that - since a lot of that design work is actually proprietary, we're going to specifically address that in the closed session later today.

CHAIRMAN STETKAR: Okay.

MR. SPRENGEL: So, we'll speak to that item. And of course a lot of that detail will come when we actually cover the topical report later, and then when we submit the revision to the results.

CHAIRMAN STETKAR: Well, okay, the topical report is just on the methodology.

MR. SPRENGEL: Correct.

CHAIRMAN STETKAR: Okay.

MR. SPRENGEL: Correct.

CHAIRMAN STETKAR: All right.

MR. SPRENGEL: And then another question was related to the grid design and if it was a unique or different grid design. And the answer is no.

MEMBER ARMIJO: This is a standard grid design.

MR. SPRENGEL: There are, you know, very minute - I think you would be familiar with the minute changes that are always happening.

MEMBER ARMIJO: Tweaks, yes.

MR. SPRENGEL: But, no, it is not a different design. It is a standard design that Mitsubishi has.

MEMBER SHACK: But it's a longer assembly, right?

MR. SPRENGEL: The overall fuel - yes.

MEMBER SHACK: Yes.

MR. SPRENGEL: That's correct, right, but the grids themselves.

Um, let's see. And then there's another question related to comparisons to the Japanese plants and their seismic performance, basically. And it was indicated, but I just wanted to confirm that we're not able to speak to that.

So, if there is a specific request, we would need that documented and then we can pursue - there was a request related to the performance of the Japanese plants and how their fuel performs on their seismic conditions and we can't disclose that.

That's proprietary, the Japanese utilities. So, we won't be coming back to that.

MEMBER SHACK: Well, but you could answer, perhaps, what the design approach is.

MR. SPRENGEL: And we'll touch on that later, yes.

MEMBER SHACK: You'll touch on that later.

MR. SPRENGEL: Yes, agreed.

And then the last one was the recent discussion related to tie rods. And I think there is some discussion related to some proprietary numbers and we can discuss that later once it's closed, if needed.

But the direct question of is there ability to do visual inspection of the tie rods, the answer is no.

And then that's all of our follow ups right now. Again, we'll touch on some of this later in the closed session.

MEMBER SHACK: And there's no capability to do an ultrasonic down the tie rod?

MR. KURIMURA: The tie rods are replaceable.

(Discussion off the record.)

MEMBER REMPE: It's replaceable, is what you're saying?

MR. KURIMURA: The tie rods can be replaceable and we can attach a sensor on the top of the tie rod. Make it accessible.

MEMBER SHACK: So, you can put a sensor on it? It's designed to do that? You have that capability?

MR. KURIMURA: Yes, top of the tie rod is flat. So, we can attach that.

MR. SPRENGEL: The answer is yes, there can be a sensor put on it. I'm not sure of the details and I think we can take that as an action to follow up on.

I do want to point out that through discussions with the staff, we have added a diagram in 3.9.1 and I don't know if you want - it's in an RAI response. So, we can provide that to you so you can see the diagram, but it has been added to the DCD.

MEMBER SHACK: Or at least identify the - I think we probably have it.

MR. SPRENGEL: Oh, I agree, yes. So, that's something that we'll do before the meeting is over to get you the RAI number. And then you can see the diagram that was added showing, you know, the location and some detail of it.

We'll take the action, though, to follow up on not visual inspection, but, you know, some other evaluation that can be done.

MEMBER SHACK: Okay.

CHAIRMAN STETKAR: Yes, if you can get us the RAI number, that would at least -

MEMBER SHACK: And then we can discuss the fluence numbers then in the closed session this afternoon.

MR. SPRENGEL: Yes, that's correct. Yes.

MEMBER SHACK: Okay.

MEMBER ARMIJO: But, Bill, where the

connections would be at the top and the bottom for bolts and things like that, that's where the highest stressed would be.

MEMBER SHACK: Which is a tie rod, you know. It's going to be stressed the whole -

MEMBER ARMIJO: But it's a smooth bar.

MR. SPRENGEL: All right. I think we've addressed the inspection part of it. So, now we'll transition to discussing, you know, the conditions experienced in the closed session.

CHAIRMAN STETKAR: Yes.

MR. SPRENGEL: Okay.

CHAIRMAN STETKAR: Good plan.

Let's start the presentation on the thermal hydraulic design. Dr. Banerjee will join us when he joins us, and see how much we get through before he joins in.

MR. LANESE: Okay. So, we're on Section 4.4, thermal hydraulic design. And the intention of this slide is to present the idea that we meet our design basis by maintaining DNB with a 95 percent confidence level for normal and operating AOOs. We do that with a WRB-2 correlation and that we account for appropriate uncertainties in poor conditions in the CHF prediction of DNB.

The safety analysis limits are set accounting for penalties for rod bow and for any transition core geometry. And there is additional margin for some core operating flexibilities built into those numbers as well.

Fuel melting is prevented at a 95/95 percent level for normal and AOO operation and that analysis is performed in the FINE code.

Finally, those analyses are done choosing appropriate core flow rates. And analyses in Chapter 15 say, you know, demonstrate that we have adequate core cooling depending on the transient or accident that's involved.

MEMBER REMPE: Isn't one of the limits on the FINE code that the cladding temperature has to be below like 1100 C that the staff's put on it for using that code?

Isn't that what - I know we'll be - maybe I shouldn't be bringing that up now, but what I'm trying to say is how can you use the code with some of the limits the staff is putting on the code for predicting fuel melting?

So, is there a way to say that in the open such as -

CHAIRMAN STETKAR: We can rely on people to tell us that they can't answer.

MEMBER REMPE: Okay.

CHAIRMAN STETKAR: You can ask anything.

MEMBER REMPE: Okay. So, are the limits the staff has put on the code for when it can be applied going to interfere with using the code? Which is why I asked the question earlier today.

And maybe the staff should answer that question, but it's something that I was just kind of wondering when I was reading through.

MR. SPRENGEL: We'll discuss it later.

MEMBER REMPE: Okay.

CHAIRMAN STETKAR: The nice thing sitting where we are is we can ask the questions and -

MEMBER REMPE: I shouldn't have -

CHAIRMAN STETKAR: No, that's fine. If it's proprietary, they can tell us. If it's more appropriate for the staff to answer it, they can tell us.

MR. SPRENGEL: Okay. Just to get into any of that discussion it will go quickly to proprietary. So, we'll specifically address that in the FINE discussion when we get there.

MEMBER REMPE: Okay.

CHAIRMAN STETKAR: In the FINE?

MR. SPRENGEL: Yes.

CHAIRMAN STETKAR: Okay.

MR. SPRENGEL: Today.

CHAIRMAN STETKAR: Today. So, hold it until this afternoon.

MEMBER REMPE: Hopefully I won't forget.

CHAIRMAN STETKAR: Write notes.

(Laughter.)

MR. LANESE: Okay. So, in the thermal hydraulic analysis, again we addressed the uncertainties associated with the DNBR, the fuel and cladding temperatures and the hydraulics and we have conservative evaluations for all those limits.

RCS flow measurement is also addressed in Chapter 14. The power distribution is tested in Chapter 14. And component and fuel inspections are discussed in Section 4.2.

The related and required safety limits that assure - that account for these uncertainties are specified in the tech specs. So, we have appropriate operating limits.

The next slide gives a comparison of some of the key features of US-APWR fuel and typical 12 and 14-foot fuel in other large PWRs.

Again, you know, the thermal design flow is proportionately higher to support the higher core power. The linear heat rate and the DNBRs are all within - when taken as a conglomerate, are small differences from existing plants. And that's really the message from this slide. No dramatic changes.

Then the open items for this section, the first one had to do with the appropriateness of the rod bow penalty that was being used.

And MHI responded to that with no changes of the DCD, just an expansion on an explanation on how it was applied. And the staff has that under consideration.

The thermal design methodology review is still ongoing. And the final open item was one related to DCD RAIs in Section 14.2 on pre-op testing. And that's been closed and resolved by the staff.

CHAIRMAN STETKAR: Okay. I don't know whether Sanjoy is on the line yet. Sanjoy, if you are -

MR. SHUKLA: He has called in.

CHAIRMAN STETKAR: And we would hear that?

MR. SHUKLA: Yes.

CHAIRMAN STETKAR: Okay.

Let's do this: We'll go on to the staff and - as long as none of the other members have any questions for MHI on any of the topics in Chapter 4. Anyone?

(No response.)

CHAIRMAN STETKAR: Well, thank you very much for your presentation. I'll warn you, you may be back up if Dr. Banerjee has any specific questions that we can answer in open session.

If he has questions that pertain to the closed session, we'll elicit those questions from him sometime before this afternoon and make sure that they're brought up in closed session, because I don't think he'll be joining us this afternoon.

And with that, again, thank you very much. I apologize for a little bit of the disorganization here this morning. It's sometimes difficult shuffling around our specific subject matter experts, but I really appreciate your patience and thank you for a good presentation.

And with that, I guess we'll hear from the staff on Chapter 4. Ready for - we're ahead of schedule, I know. I see people shuffling.

Do you have everybody that you need?

MS. REYES: Right now we do.

MR. SCHMIDT: Yes, just give me about a minute.

CHAIRMAN STETKAR: Okay. We're on the record here.

MS. REYES: Okay. Just one thing since, like you say, we're ahead of time, we were expecting our contractors to call in. So, we have to let them know that -

CHAIRMAN STETKAR: On Chapter 4?

MS. REYES: Yes.

CHAIRMAN STETKAR: Okay.

MR. SCHMIDT: Yes.

MS. REYES: But they are in the west coast. So, it's kind of early for them. So, we're hoping that -

MEMBER ARMIJO: Oh, come on. They can wake up.

(Laughter.)

(Discussion off the record.)

(Pause in the proceedings.)

CHAIRMAN STETKAR: Bear with us another couple of minutes. There was a question about whether or not the bridge line is active. Apparently if there's no activity, it's so smart it hangs up.

(Discussion off the record.)

CHAIRMAN STETKAR: I believe we're ready. We'll just monitor that line. You might hear dial tones or something. I'm not quite sure what's going to happen, seems to be a theme for today, but we'll get started.

And I don't know if Ruth or Jeff or who's going to kick it off.

MS. REYES: I will start.

Once again, good morning. My name is Ruth Reyes and I am the chapter PM for Chapter 4. And the lead project manager is Jeff Ciocco. He was not able to be here today.

And the technical staff, I have two here with me, James Gilmer and Jeff Schmidt. And the rest of the staff, there are some in the audience here.

Right now the only sections that the staff has open items are on Sections 4.2 and 4.4, and those are the sections that we are going to be discussing here.

Having said that, I'll leave Jeff Schmidt to start his presentation.

MR. SCHMIDT: Okay. I'm Jeff Schmidt from NRO Reactor Systems. And with me is Jim Gilmer also from Reactor Systems. We were the primary reviewers of 4.2.

We had consultant help from PNNL from both a fuel system standpoint, you know, guide tubes, grids, and then also PNNL from a fuel rod performance standpoint, the FINE code we'll be talking about this afternoon.

(Telephone interruption.)

CHAIRMAN STETKAR: Bear with us for just a second here.

(Discussion off the record.)

CHAIRMAN STETKAR: Okay. Sanjoy, where we are now just to orient you, MHI has given their presentation on Chapter 4.

So, if you have any questions for them on the thermal hydraulic design of the fuel, what we'll do is table them for the moment.

MEMBER BANERJEE: Okay.

CHAIRMAN STETKAR: We're covering proprietary information on thermal hydraulic design this afternoon. So, if you have any questions regarding the topical report, I'll ask you to kind of communicate them to us before the afternoon session.

As I understand it, you will not be available this afternoon; is that correct?

MEMBER BANERJEE: Right.

CHAIRMAN STETKAR: Okay.

MEMBER BANERJEE: But I will be available, but it will be after probably four o'clock.

CHAIRMAN STETKAR: No, that's too late. We're way ahead of schedule. So, I just wanted to orient you. Where we are now is the staff is just starting their presentation on Chapter 4.

So, I'll let them continue. And then if you have specific questions for MHI, we will call them back up after the staff -

MEMBER BANERJEE: Okay, sure.

CHAIRMAN STETKAR: After the staff finishes.

MEMBER BANERJEE: Go ahead, John, and I'll follow along. And then if something comes up for MHI, I'll get back to you later on.

CHAIRMAN STETKAR: Okay, good. Thank you.

MEMBER BANERJEE: After the staff presentation.

CHAIRMAN STETKAR: Great. Thank you very much.

MEMBER BANERJEE: All right. Thank you.

CHAIRMAN STETKAR: Jeff, sorry for the interruption.

MR. SCHMIDT: No problem. I'm going to be discussing 4.2, and then Jim will be discussing 4.4. Since we have no open items for 4.3, I can handle any 4.3 questions you might have. So - or at least I hope so. I probably shouldn't say that so confidently, huh?

CHAIRMAN STETKAR: That's a bold statement.

MR. SCHMIDT: that was kind of a bold statement. I'd like to retract that - no, just kidding.

So, you know, I'm going to talk about 4.2, which is the fuel pellet, the fuel assembly, fuel cladding and control rods and any other in-reactor components.

So, we go to Slide 4. As Mitsubishi already pointed out, there's two topical reports that support the DCD 4.2. One is the FINE fuel rod performance code, which is 07008, and then there's the 07034 which is the FINDS. We usually call them E and D just to be clear.

And, you know, 07034 is really how the structure behaves under seismic events. It's basically the methodology.

And there's two technical reports that support Chapter 4.2. And then there's 07016 which is kind of the fuel system design. It looks at things like the actual, say, for a given core, what type of hydrogen limits you'll get for the clad, stress and strain on the guide tube material. It's kind of like the application of Topical Report 07008, the fuel rod performance code.

And then as we discussed this morning, there is MUAP-08007 which is the application of the D, or the FINDS code, to the actual fuel behavior whether you have, you know, plastic grid deformation, what the stresses are in the guide tubes during, you know, combined LOCA/seismic events. So, it's the application of FINDS, okay.

And, again, this is kind of a repeat of what you saw earlier from Mitsubishi as the open items. As Ruth mentioned this morning, MUAP-07034 which is the structural seismic and LOCA code, is still under review. It's with projects going through the process. Should shortly, I think, be going to OGC.

MUAP-08007 which is the application for the seismic methodology as we talked about this morning, that's being revised where the staff is expecting a revision sometime next year. And we'll talk more about that even in the staff presentation coming up here too.

One of our open items was grid to rod fretting. In the Technical Report 07016 there was a methodology by which MHI was going to predict fretting wear and clad reduction - fuel clad reduction.

The staff historically doesn't review and approve methodologies that try to predict that. We usually try to base it on the thousand hour grid to rod fretting test.

So, what the staff asked for was based on the experimental data that they had, the hydraulic test data, extrapolate out, you know, based on what type of fuel duty you would have, your clad thinning based on grid to rod fretting and whether you met the 10 percent criteria.

So, that was an open item. It has since been closed. They've done basically the experiment and then conservative extrapolation methodology and the staff has resolved that issue.

Just to make it clear, though, the staff is not reviewing or approving the methodology in 07016 which tries to deterministically predict clad fretting.

The following open item again was based on the hydraulic testing. There was some statements in the hydraulic testing which indicated that MHI believed that the fuel vibration and assembly vibration were acceptable, but there was no clear, in the staff's mind, what the acceptance criteria was for that acceptability.

So, the follow-on here is really, you know, what criteria are you using to evaluate acceptability. So, that's still an open item.

MEMBER BANERJEE: So, what did they propose as the acceptability criteria? This is Sanjoy.

MR. SCHMIDT: Yes, we haven't seen it yet. I haven't reviewed that. They probably responded to the RAI, but I haven't reviewed it yet, Dr. Banerjee.

MEMBER BANERJEE: Okay. Thank you.

CHAIRMAN STETKAR: Does MHI want to add anything?

(No response.)

MR. SCHMIDT: The other open item that was also covered by Mitsubishi was the BOL and EOL elastic limits and the FINDS impact model to determine, you know, under the various seismic and LOCA events how many grids will actually buckle.

So, it's your, what we call, the p crit design or p crit value which the grids will buckle at. That's still an ongoing review item.

The next one which is MUAP-09002 which is the seismic and accident load conditions for primary components and piping which is really like at Chapter 3, we're just saying that that's still open and, hence, we can't resolve all of the fuel issues until all of the other design features are incorporated or changes based on 09002.

So, it's kind of just a tracking item that we can't complete our design review until that's completed.

Next slide. Again, this is really just a tracking item to say what I just said there is that we can't really complete the application for the fuel which is the MUAP-08007 until the final soil profiles of MUAP-09002 are complete.

In other words, we have to know what the soil profiles look to do our final evaluation of fuel, of seismic and LOCA response.

The other open item is the AOO holddown spring evaluation. MHI also discussed that. Trying to determine if under AOOs, if the spring will plastically deform in such a manner that it would not perform its function of keeping it basically seated and on the pins. So, that's still under staff review.

Again, the next open item is related to that same thing is plastic deformation of the holddown spring during an RCP overspeed event. Those are still waiting for staff review.

Go to the next slide. The staff had a

guide tube hydrogen question that has since been closed and resolved.

Mitsubishi had originally asked for a 800 ppm hydrogen limit in the guide tube. The staff asked for additional data which supported that 800 ppm.

We've since gone to a number which is lower than that and the staff is satisfied with that lower number and probably should get into some of the actual numbers that were agreed upon in the proprietary discussion.

So, if you want the final numbers, I guess we can discuss those.

CHAIRMAN STETKAR: Yes.

MR. SCHMIDT: The next open item was burnable poison. The issue there was rod internal pressure due to B10 release. And through follow-on discussions, this has been resolved/closed.

Mitsubishi is taking a conservative analysis that all the boron - all the B10 basically is used up in its first cycle and that the rod internal pressure doesn't exceed its design criteria.

And then the last item was control rod cladding stresses and the material properties. That's still under staff review. There's some questions on the stress and strain values used for the control rod guide - or control rod cladding material.

As we talked about before and we're going to talk about more as we go through the presentation here is the staff has asked MHI to evaluate a tenth of a second drop in the Chapter 15 accidents due to plastic grid deformation. And we'll talk about that in the upcoming slides here.

There's been some testing that's done to show that there's about a 0.1-second delay in control rod drop time. We're just asking that it be evaluated in Chapter 15.

And the following or last open item is fuel rod coolability under LOCA-induced loads. If you look at some of the guidance, and we'll talk about that in a second here, 4.2 looks at, say, typically combined loads of LOCA plus SSEs.

What staff has asked is do we get any plastic grid deformation under just LOCA loads? LOCA loads only. And MHI is preparing a response for that.

So, I want to go back now to which is MUAP-08007 which is the application for the seismic code, and talk about some of the issues that the staff is dealing with in the review of that document. So, this relates to open item 4.2-2 which is MUAP-08007.

As MHI discussed this morning and you guys also discussed this morning that US-APWR does have - is predicting plastic deformation of the assembly grids under SSE-only loads. Not combined loads, but SSE-only loads. That's the first application that the staff is aware of that we're seeing deformation under SSE-only loads.

So, when we went to look for guidance, we ended up at GDC-2 which basically says, you know, you shall evaluate SSEs with appropriate combination of normal and accident conditions along with natural phenomenon.

So, the natural phenomenon here being the earthquake and the accident conditions being whatever accident conditions you might have, say, in the Chapter 15 accident analysis and LOCA.

So, really the staff is trying to determine how to evaluate what appropriate combination of effects of normal and accident conditions are with natural phenomenon.

Assembly components are evaluated under SSE, LOCA and the combined loads. As you can tell, we had an RAI just on LOCA loads alone. We've had - SRP guides you to looking at both SSE and LOCA loads. So, we're looking at obviously all those loads both in combination and in singular.

SRP acknowledges the possibility of plastic grid deformation under Appendix A. If you do have plastic grid deformation, you look at things like control rod insertability as a criteria, your LOCA requirements, your 5046 LOCA requirements, and then it doesn't really provide a lot of additional guidance for the staff to review.

It's kind of unclear of how when you have plastic deformation under SSE-only loads do you evaluate, say, the remainder of the Chapter 15 accidents.

Next slide. SRP does provide a little more guidance. It does say consequences of grid deformation are small and gross deformation of many assemblies would be needed to interfere with control rod insertion.

It also goes on to say that, you know, you obviously have to be able to demonstrate control rod insertability under combined loads which would be SSE plus LOCA loads.

Currently, licensed plants do not predict grid deformation in rodded locations. So, historically plants haven't had to concern themselves with this potential, say, second part of the SRP which says you check control rod insertability. There are plants out there that do deform grids, but they're not in control rod locations.

MHI will have as of right now predicting at least currently deformation in control applications. So, that makes it somewhat different for the staff's review.

MHI has performed, though, in around 1986 there was a bunch of tests done for deformed grids and looked at control rod insertability.

So, there is pretty extensive test data from 1986 that does look at things like plastic grid deformation and control rod insertability.

The NRC has reviewed that report and found that the deformations that MHI is predicting, that the control rods will insert. There is a slight delay time. And that slight delay time was the 0.1 seconds I talked about earlier.

MEMBER SHACK: I mean, my confidence in analysis is very high when we're in the linear space.

MR. SCHMIDT: Right.

MEMBER SHACK: When I'm starting to compute deformations in plastic especially with buckling, I get a whole lot more -

MR. SCHMIDT: Right.

MEMBER SHACK: - uncertain. And you're

confident that the deformed grids in these tests were, in fact, some sort of limiting values that -

MR. SCHMIDT: They were severely deformed. There were multiple, I think - and MHI may be able to help - five grids that were deformed axially.

So, it was a very extensive and limiting test from a deformation standpoint, yes. And having only a 0.1-second delay with those deformations is a pretty small delay when you normally have like a four-second drop time delay.

So, yes, the staff has spent a fair amount of time reviewing that report and, you know, we're pretty confident that that covers the bounding case here for control rod insertability.

MEMBER BANERJEE: Was there a lot of plastic deformation? Could it be calculated? Like Bill, I always worry about, you know, calculations done in the plastic regime.

MR. SCHMIDT: Right. Yes, I mean, FINDS has the capability of predicting the plastic deformation, but we're not really - while we're using that as information, we're using it more of a go, no-go for deformation.

You know, I don't want to go too far into this subject because it's being reevaluated by MHI.

So, what I'm saying right now is kind of the current state of affairs, but that may change when we get the revision to 08007.

But, yes, FINDS does theoretically have the capability of predicting amount of deformation.

MEMBER BANERJEE: Okay, thanks.

MEMBER ARMIJO: These are all Zircaloy grids, right?

MR. SCHMIDT: That's correct. They're all Zircaloy.

MEMBER ARMIJO: So -

MR. SCHMIDT: Well, I shouldn't say - the top and bottom are Inconel, but the middle ones with the deformation, the maximum deformation are Zircaloy.

MEMBER ARMIJO: So, in service they're going to oxidize, they're going to pick up some hydrogen.

MR. SCHMIDT: Yes.

MEMBER ARMIJO: They're going to get some radiation and hardening.

MR. SCHMIDT: Hardening, right.

MEMBER ARMIJO: And so, you get into buckling with something that's potentially embrittled.

MR. SCHMIDT: Right.

MEMBER ARMIJO: I'm sure those are all the things you worry about, but -

MR. SCHMIDT: Yes.

MEMBER ARMIJO: Were the tests done on grids that had in some way simulated that level of service, or were they just took some Zirconium and squeezed it and said, okay, it's buckled enough, we can test that?

MR. SPRENGEL: I'd request that maybe we defer some of this discussion. I mean, we're getting into the methodology and the testing that was done.

So, we can answer this and have some discussion on this topic today. A lot of it will be covered in future discussions, but I think we can go ahead and discuss some of those, but I'd request -

MEMBER ARMIJO: Sure.

MR. SPRENGEL: - that we move it to the closed session.

MEMBER ARMIJO: We can wait.

CHAIRMAN STETKAR: That's fine.

I apologize. I didn't look far enough ahead. I'll let you finish this and then I was going to ask something else that's peripherally related.

MR. SCHMIDT: Should I go to the next slide?

CHAIRMAN STETKAR: This topic, yes.

MR. SCHMIDT: Yes, okay. So, we're on Slide 12 now. Again, I kind of talked about already appropriate combination of effects of normal and accident conditions on the natural phenomenon.

Current plants I also talked about do deform under SSE plus LOCA loads and non-limiting core locations typically on the core periphery or the assemblies in the baffle.

Deformation under SSE-only loads and limiting core locations, as I mentioned, brings up additional staff questions on how you evaluate, say, just even full-power operation under a seismic event, as well as the Chapter 15 accidents now that you have plastically-deformed grids potentially in limiting core locations.

So, those are all -

MEMBER BANERJEE: Were there any experiments done on DNBR with the slightly deformed grids or deformed grids?

MR. SCHMIDT: None that I'm aware of.

MEMBER BANERJEE: Okay. So, how does that affect the evaluations of deformation?

MR. SCHMIDT: Well, that's part of the staff's review of this 08007. We haven't come to a conclusion on that, Dr. Banerjee.

MEMBER BANERJEE: Okay. Thank you.

MR. SCHMIDT: Okay. On the next slide, basically MHI as they talked this morning, is currently revising 08007.

At the time of this presentation when this was put together, we anticipated a July 2013 date trying to address, I think, many of the questions that we've all raised here today.

MEMBER BANERJEE: Are they going to do some experiments, or is this just going to be sort of an evaluation? I haven't been following this area of the effect of deformation on the DNB and LOCA.

Is it purely so the analysis is being put forward?

MR. SPRENGEL: This is a revision to incorporate a new response spectra that's being developed out of our 3738 review.

MEMBER BANERJEE: Right.

MR. SPRENGEL: So, the methodology is set and the methodology has already been applied once in a previous results report, but now we're going to update the results using a new response input.

MEMBER BANERJEE: That's the effect of the earthquake, right?

MR. SPRENGEL: Yes.

MEMBER BANERJEE: On deformation. So, then the effect of the deformation on DNBR and LOCA, how is that evaluation being done?

I haven't followed it, unfortunately. So, maybe it's all there for me to look at somewhere.

MR. SCHMIDT: This is Jeff Schmidt from the NRC.

The staff does not know what MHI plans on presenting going forward.

MR. SPRENGEL: Right. That's an ongoing discussion tied to that open item. And as it's noted there that we do have a public meeting that we're trying to set up this coming November.

MEMBER BANERJEE: When is that? In November -

MR. SCHMIDT: Yes.

MEMBER BANERJEE: - is the meeting?

MR. SCHMIDT: Yes.

MEMBER BANERJEE: Okay. Thank you. That answered my question. Go ahead.

MR. SCHMIDT: Yes, I mean, as I think we're all, you know, discussing here that core-wide plastic deformation introduces significant technical, regulatory and scheduling uncertainty in the review.

And as Ryan just alluded to, we have a

meeting coming forward in November to try to find out what, you know, potential path forward MHI might be proposing.

MEMBER BANERJEE: Okay. Thanks.

MR. SCHMIDT: Yes.

So, as far as conditions and limitations from a 4.2 standpoint, and this really comes from the FINE review, but the staff has imposed s surveillance program on oxide measurements both - if it's a 24-month cycle, both after the first cycle and the second cycle to measure oxide thickness.

Because the power histories that were presented in FINE weren't necessarily - I should say the US-APWR rod power histories could be different and more limiting than the database that was in the FINE topical report.

So, we've instituted a measurement campaign to make sure that the model in FINE, the oxide model in FINE is indeed conservative and that the number that we can talk about in the proprietary session is not violated.

So, that's the one condition and limitation we put in 4.2 on fuel rod performance, really.

CHAIRMAN STETKAR: Before we go to 4.4, I wanted to ask a question that's peripherally related to this topic of plastic deformation, but is also related to PRA-type analyses.

This is not the PRA section and I realize we're looking at design basis accident analyses here, but as part of the risk assessment to this design I quite honestly don't recall whether MHI is proposing a seismic margins assessment, or a complete probabilistic assessment of seismic risk.

Regardless of their approach, they will have to perform an evaluation of margins at both the SSE acceleration and beyond the SSE acceleration.

So, I was curious whether the staff is aware of any work in that area which might help to improve your confidence even within the design basis part of the analysis, or whether MHI had actually done any of those analyses to give us some information about the margins that might be available.

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

CHAIRMAN STETKAR: And I realize it's kind of at the interface between -

MR. SCHMIDT: Right.

CHAIRMAN STETKAR: - design analyses and PRA, but I think what we're all hearing is there's concerns at the SSE. I think given that there would be concerns beyond the SSE -

MR. SCHMIDT: Yes.

CHAIRMAN STETKAR: - which ought to be evaluated as part of the risk assessment process. So, I see several people getting up and down. I don't know whether -

MR. SPRENGEL: I mean, I can speak to the seismic margin analysis. I know the staff have brought up before and we're aware of the interest in it.

It's also tied to our seismic work that's ongoing.

CHAIRMAN STETKAR: Yes.

MR. SPRENGEL: And we plan to have a submittal for that around summer of next year. So, I can get a firm date for you on that, but that will be coming in.

So, that may not be part of our Chapter 19 discussion coming up, but there is a commitment to provide that information.

CHAIRMAN STETKAR: Yes, because I think we're up on Chapter 19 earlier than that.

MR. SPRENGEL: Yes. Yes, that's correct.

CHAIRMAN STETKAR: Okay. All right. So,

it's just part of the ongoing story. I don't know if the staff had anything to add.

(No response.)

CHAIRMAN STETKAR: Okay. Thanks. All right. Thank you.

MS. REYES: Okay. And that concludes our presentation for 4.2. We don't have any open items on 4.3. So, we don't have any presentation or a slide on that section.

But if the staff has any questions related to that section, we can address the questions now. If not, we will continue with 4.4.

(No response.)

MS. REYES: Okay.

MR. GILMER: Okay. The key part of DCD Section 4.4 is the reference to the thermal design methodology topical report which basically is the subchannel code VIPRE-01M which is MHI's version of the standard VIPRE code.

It does have some enhanced model selection features and the key modification is the APWR-specific critical heat flux correlation.

The details of the code are discussed in the topical report MUAP-07009. And for that topical report, the staff safety evaluation is nearing completion. So, it should be available in the near future.

CHAIRMAN STETKAR: Yes, we're currently scheduled, I think, to review that in January.

MR. GILMER: You'll hear all the details -

CHAIRMAN STETKAR: We'll be doing that.

MR. GILMER: - at that time.

MEMBER BANERJEE: Sorry I missed that. When was it scheduled for?

CHAIRMAN STETKAR: January 15th.

MEMBER BANERJEE: Sorry.

CHAIRMAN STETKAR: January 15, 1-5.

MEMBER BANERJEE: Okay. Thank you.

CHAIRMAN STETKAR: 2013.

MR. GILMER: The well-worn Westinghouse WRB-1 and 2 correlations were actually shown by tests to be appropriate for the APWR fuel. Appropriate and conservative with the grids that APWR have.

MEMBER ARMIJO: You mean US-APWR?

MR. GILMER: US-APWR, yes. I'm sorry.

MEMBER ARMIJO: Just wanted to make sure.

MR. GILMER: Yes, it is.

The actual DNB tests, if memory serves me, were done about two years ago at the KATHY facility in Karlstein, Germany. And several members of the staff did witness those tests.

Next slide. The details of the test report is in MUAP-11010 which is available in SharePoint and ADAMS.

The staff evaluation of the test is covered under the thermal design topical - MUAP-07009 which is coming out very soon.

Uncertainties in power, inlet temperature, pressure flow, peaking factors, code uncertainties and correlation of the uncertainties were evaluated by the staff-approved Westinghouse Revised Thermal Design Procedure. So, staff found that it was also appropriate for the US-APWR fuel.

MEMBER BANERJEE: Were the tests in Karlstein full length, or was it slightly reduced length? Can that facility take the full length?

MR. GILMER: They were full length. 14 foot.

MEMBER BANERJEE: Okay. Because normally I didn't realize the Karlstein facility can take 14 feet.

MR. GILMER: I believe they had to do some physical modification, but they did accommodate it.

MEMBER BANERJEE: Okay. Fine. So, this was full length prototypical in every way, right, with the mixing bins and everything there.

MR. GILMER: Yes.

MEMBER BANERJEE: Okay. Thanks. I haven't looked at these tests, but interesting. Go ahead.

MR. GILMER: Okay. Now, Section 14.2 of the DCD describes the traditional startup testing that would be performed for core performance and staff found it acceptable.

There were some issues - open issues in 14.2 that have been now closed related to the instrumentation and procedures.

The GSI-191 LOCA-generated debris issues are being addressed in Section 6.3 of the staff's safety evaluation which I think will be discussed in -

MS. REYES: I believe it's March.

MR. GILMER: - March.

MEMBER BANERJEE: Okay.

MR. GILMER: The only open items in Section 4.4, the first one was related to less than one percent rod bow penalty.

The staff had a question that it was actually lower than we've traditionally seen with Westinghouse plants.

That response had been received and still under review by the staff, but we believe that it is acceptable preliminarily.

Another open item was the - basically the tie-in to the approval for the topical report from the thermal design methodology. That is about to be closed with respect to Section 4.4.

And there was an open item on Section 4.5 which is on the - related to testing that was tied to the open items in Chapter 14 that has now been resolved.

MEMBER ARMIJO: Could you expand a little bit on why you have this rod bow penalty on this?

MR. GILMER: Why?

MEMBER ARMIJO: Yes.

MR. GILMER: Well, it was one of several uncertainties traditionally that's accounted for by an assumed penalty and -

MEMBER ARMIJO: But it's not related to the fact that the assembly is longer or grid spacing is different or there's nothing unique, it's just sort of a traditional penalty?

MR. GILMER: It's a traditional penalty. We did ask the question about whether there are any length-related differences in the bowing. So, there was an RAI on that that we were satisfied was acceptable.

So, any other questions on this section?

MEMBER BANERJEE: What uncertainties - I just want to go back with regard to the power distribution with the 14-foot core. Are there uncertainties in the power distribution that are greater than with 12-foot cores or is that - are there reactor physics calculations that power distribution calculation was verified and validated?

MR. GILMER: Do you want to take that?

MR. SCHMIDT: Yes, I'll take that. This is Jeff Schmidt.

Yes, as far as like the physics code and the ANC code is well-validated for 12 or 14-foot fuel. It's really insensitive to that.

The only difference would be, you know, how many axial nodes you might put in your physics model, but it's well-established that ANC can handle 14-foot fuel.

MEMBER BANERJEE: Okay. So, from the uncertainty point of view, there is no increase compared to 12-foot cores.

MR. SCHMIDT: No, there's no additional increase.

MEMBER BANERJEE: Okay. Thank you.

And the flow measurement uncertainties are just whatever they are for any other plant. Are they - how are they - what sort of uncertainties are related to the flow measurement?

MR. SCHMIDT: I'm sorry, what type of measurements?

MEMBER BANERJEE: Flow measurements.

MR. SCHMIDT: Flow measurements?

MEMBER BANERJEE: Yes.

MR. SCHMIDT: Flow measurements.

MR. GILMER: There were some instrument uncertainties. The traditional ones that are accounted for in this type of methodology.

MEMBER BANERJEE: But there is no additional factor involved here other than the traditional way to do this.

MR. SCHMIDT: No.

MR. GILMER: No, it follows the same existing Westinghouse Revised Thermal Design Procedure. Yes, there would be no reason for additional flow uncertainties there.

MEMBER BANERJEE: Okay.

Do they actually make measurement in the primary circuit or on the elbows and things? How do they actually - do they have any such measurements that they do?

MR. SCHMIDT: Well, they do have the measurements for, you know, hydraulic. They did hydraulically test an assembly, the fuel assembly itself. So, they have a good idea of pressure drop across the assembly for their grid designs.

As far as the plant testing, that kind of what Jim was referring to is Section 14.2, which is the core flow and pressure drop of the kind of the as-built US-APWR.

MEMBER BANERJEE: Yes, I'm just trying to remember whether there are some pressure measurements made around the loop to validate the flows.

Do they have any such measurements that are being made? Pressure losses or -

MR. GILMER: There's pressure tests done in 14.2 for pre-operational tests.

MEMBER BANERJEE: Okay. Anyway, I need to read up on this. Thanks. Carry on, please, yes.

MR. GILMER: Well, we're pretty much done unless there are further questions.

CHAIRMAN STETKAR: I was going to say do any of the members have any other questions for the staff?

(No response.)

CHAIRMAN STETKAR: Sanjoy, do you have

anything else specifically for MHI regarding the fuels?

MEMBER BANERJEE: I don't at the moment, John, but that doesn't mean I will not have at the end of, you know, I missed the MHI presentation. So, I'm going to go over it and is it okay if I ask any questions just for clarification later on?

CHAIRMAN STETKAR: Well -

MEMBER BANERJEE: Is your crew there going to vanish?

CHAIRMAN STETKAR: Well, stay on the line. What I'd like to do is thank the staff if there are no more questions for the staff.

And because this will be the end of our open session for today, what I'd like to do is see if there are any questions or statements from members of the public who might be here.

I don't believe that we have anybody on the bridge line that we need to open things up for. So, if there's anyone from the public or any other questions or comments?

(No response.)

CHAIRMAN STETKAR: Hearing none -

MS. REYES: I was going to add, but it looks like the staff doesn't have questions, we don't have any open items in the remaining of the sections, but the staff, the technical staff is here so in case of the ACRS staff has questions on 4.5 and 4.6 which we didn't discuss, but the staff is here.

CHAIRMAN STETKAR: I understand. And I'm assuming that if anyone did have a question, they would have raised it by now.

Now, again I'd like to thank the staff. Now, we face a bit of a scheduling question. The question is we were - according to the original agenda, we were scheduled to break for lunch at 11:30 and reconvene at 12:30 to pick up the afternoon session.

The material that we have left to discuss today is the proprietary topical report and I'll now ask both MHI and the staff for input about do you want to start that discussion now and go to about noon and break for lunch at 12:00, or should we break for lunch now and reconvene at 12:30 according to the original agenda? Long lunch.

And the reason I ask this is I don't know whether you have other resources that are planning their schedules around our agenda, people who might be calling in or staff members who might be planning their time.

So, I need feedback from you. So, I'm proposing either we start now and go to 12:00, or we break now and reconvene at 12:30. And it's a resource issue on the -

MR. SPRENGEL: We have our resource here and we're ready. So, I defer to the staff.

CHAIRMAN STETKAR: Yes, we'll ask about the staff.

(Discussion off the record.)

MEMBER BANERJEE: So, John, are we going into closed session with MHI?

CHAIRMAN STETKAR: Yes, we are. The reason I wanted to do this, Sanjoy, is if we can if it's feasible to maximize your input in realtime.

MEMBER BANERJEE: Right.

CHAIRMAN STETKAR: Because, you know, you're only available until noon, right?

MEMBER BANERJEE: Right.

CHAIRMAN STETKAR: Okay.

MEMBER BANERJEE: I can maybe stay until 12:15, but I really need to leave.

CHAIRMAN STETKAR: Yes. We'll see where we get to a convenient break point in the presentations around that time. So, we'll do that. For scheduling, we'll do that.

(Discussion off the record.)

CHAIRMAN STETKAR: While we're getting set up here, we'll have MHI come up first. We're getting the handouts.

What I'd also like to do is we're going to go into closed session. So, I want to make sure that the staff and MHI confirm that only people are here who are authorized to hear this presentation.

So, if you'd check the audience and make sure that there aren't any people who you don't want here, I would appreciate that.

(Discussion off the record.)

CHAIRMAN STETKAR: And what I'll do is I'll close the session now so that we can discuss the proprietary material.

(Whereupon, the meeting went off the record at 11:05 a.m. to begin the closed session portion of the meeting.)

Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards US-APWR Subcommittee: Open Session
Docket Number:	(n/a)
Location:	Rockville, Maryland
Date:	Friday, October 19, 2012

Work Order No.: NRC-1967 Pages 1-3

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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

+ + + + +

US-APWR SUBCOMMITTEE

OPEN SESSION

+ + + + +

FRIDAY

OCTOBER 19, 2012

+ + + + +

ROCKVILLE, MARYLAND

+ + + + +

The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 8:30 a.m., John W. Stetkar, Chairman, presiding.

SUBCOMMITTEE MEMBERS:

JOHN W. STETKAR, Chairman

SANJOY BANERJEE, Member*

DENNIS C. BLEY, Member

JOY REMPE, Member

WILLIAM J. SHACK, Member

NRC STAFF PRESENT:

GIRIJA SHUKLA, Designated Federal Official

JEFF SCHMIDT, NRO

MICHAEL TAKACS, NRO

ALSO PRESENT:

DAVID CARAHER, ISL

HIROSHI FUJISHIRO, MHI

KEVIN LYNN, MHI

YUTA MARUYAMA, MNES

JUNTO OGAWA, MHI

RYAN SPRENGEL, MNES

REBECCA STEINMAN, MNES

DOUG WOOD, MHI

*Participating via telephone

P R O C E E D I N G S

8:31 a.m.

CHAIR STETKAR: The meeting will now come to order. This is the second day of the meeting of the United States Advanced Pressurized Water Reactor Subcommittee. I'm John Stetkar, Chairman of the Subcommittee, for the record. Members in attendance are Dennis Bley, Bill Shack, Joy Rempe, and Sanjoy Banerjee has joined us via the bridge line.

Today we will be discussing topical report MUAP-07010-P, non-LOCA methodology. The material that will be presented today is proprietary to Mitsubishi Heavy Industries, and, therefore, this meeting will be closed to protect the proprietary nature of that material. Before we close the meeting, are there any comments that any of the members want to make on the public record or anyone else? If not, we'll close th meeting, and I'll ask MHI and the staff to confirm that there is no one in this room who should not be. We'll have the bridge line closed, except for Dr. Banerjee.

MEMBER BANERJEE: I'm on.

CHAIR STETKAR: Thank you.

(Whereupon, the foregoing matter was concluded at 8:33 a.m.)

LUMINANT GENERATION COMPANY

Comanche Peak Nuclear Power Plant, Units 3 and 4

ACRS US-APWR Subcommittee

FSAR Chapter 4 – Reactor

October 18, 2012

Comanche Peak 3&4 FSAR Chapter 4 Summary

- Ch 4 of US-APWR DCD is incorporated by reference in the R-COLA with no departures or supplements
- No contentions pending before ASLB

Presentation to the ACRS Subcommittee

Comanche Peak Nuclear Power Plant, Units 3 and 4
COL Application Review

Safety Evaluation Report with Open Items

CHAPTER 4: Reactor

October 18, 2012

Staff Support

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- Lead Project Manager - Stephen Monarque
- Project Manager – Ruth Reyes
- Supporting Technical Staff:
  - Jeffrey Schmidt
  - Fred Forsaty
  - Jim Gilmer
  - Robert Davis
  - Joel Jenkins
  - John Budzynski

Sections Reviewed

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

- 4.1 Summary Description
- 4.2 Fuel System Design
- 4.3 Nuclear Design
- 4.4 Thermal-Hydraulic Design
- 4.5 Reactor Materials
- 4.6 Functional Design Of Reactivity Control System

- These sections are incorporated by reference with no departures or supplements

Presentation to ACRS Subcommittee

Chapter 4: Reactor

October 18, 2012

Mitsubishi Heavy Industries, Ltd.

MHI Presenters

Lead Presenter:

Lou Lanese

Technical Experts:

Nozomu Murakami

Takayuki Nakano

Masayuki Kauchi

Takayuki Suemura

Chikara Kurimura

Introduction

Chapter 4 Reactor

- Section 4.1 – Summary Description
- Section 4.2 – Fuel System Design
- Section 4.3 – Nuclear Design
- Section 4.4 – Thermal-Hydraulic Design
- Section 4.5 – Reactor Materials
- Section 4.6 – Functional Design of Reactivity Control Systems

Open Items for Chapter 4

- There are no Open Items for Sections 4.1, 4.3, 4.5 and 4.6.
- There are 17 Open Items in total for Sections 4.2 and 4.4 (discussed in later slides)

➢ US-APWR Core and Fuel Design Summary

Features	*US-APWR*	*Typical 4 Loop Plant in U.S.*
Core thermal power (MWt)	4,451	3,565
Fuel rod lattice	17 x 17 (264 fuel rods)	17 x 17 (264 fuel rods)
Number of fuel assemblies	257	193
Number of RCCAs	69	53
Active fuel length (ft)	14	12
Average linear heat rate (kW/ft)	4.6	5.7

- Large thermal power by the large sized core
- Large thermal margins due to lower average linear heat rate

US-APWR reactor core design features

- Some unique designs to the US-APWR are:
  - ✓ 257 fuel assemblies with 14 foot active fuel length
  - ✓ Stainless steel radial neutron reflector

- Most design features are essentially the same as U.S. operating PWRs:
  - ✓ 17x17 fuel assembly with 264 fuel rods, 24 control rod guide thimbles, and an instrument thimble in the center
  - ✓ UO_2 fuel with less than 5% enrichment of U-235
  - ✓ Burnable Absorbers:
    - – Gadolinia (integrated in fuel)
    - – Borosilicate glass (discrete)
  - ✓ Control rod drive system (CRDS) structural materials, reactor internals and core support materials

Section 4.1 - Summary Description

- Calculational Codes
  - Approved or under review by the NRC:

Category	*Code*	*Description*	*Technical or Topical Report Number*
Fuel System (4.2)	FINE	Fuel rod integrity evaluation	MUAP-07008
Fuel System (4.2)	FINDS	Fuel assembly horizontal vibration evaluation due to a seismic event	MUAP-07034
Nuclear (4.3)	PARAGON/ANC	Nuclear constant generation and core simulations	MUAP-07019
T/H (4.4)	VIPRE-01M	Core thermal-hydraulic analyses including DNBR evaluation	MUAP-07009

Major RAIs for Section 4.1

- Open Items
  - There are no Open Items for Section 4.1.

Section 4.2 – Fuel System Design

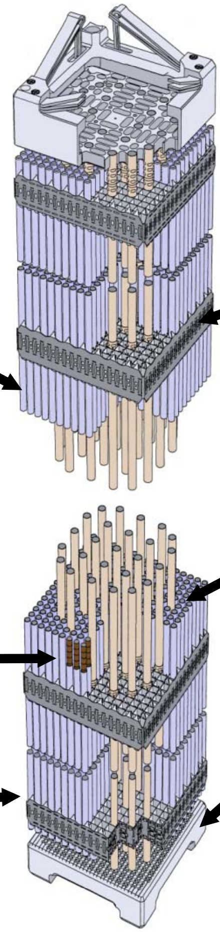
US-APWR Fuel Design Features

Flexible Core Operation

- 14ft Active Length
- 17x17 Rods Array
- High Gadolinia Content Pellet (up to 10wt%)

Enhanced Fuel Economy

- High Density Pellet (97%T.D.)
- Zircaloy-4 Grid (for Neutron Economy)



High Reliability

- Grid Fretting Resistant Design (11 Grids & improved Grid Spring Design)
- Corrosion Resistant Cladding Material (ZIRLO™)
- Anti-debris Bottom Nozzle with Built-in Filter

Fuel Design Specification

Fuel Assemblies	US-APWR	Typical Fuel Design in the U.S.
Fuel Rods Array in Fuel Assembly	17 x 17	←
Number of Fuel Rods per Fuel Assembly	264	←
Number of Control Rod Guide Thimbles	24	←
Number of in-core Instrumentation guide tube	1	←
Number of Spacer Grids	11	8 or 10
Fuel Rods		
Outside Diameter	0.374 in.	←
Cladding Thickness	0.022 in.	←
Active Fuel Length	14ft (4200mm)	12ft or 14ft
Enrichment	Max. 5 wt%	←
Gadolinia Content	Up to 10 wt%	Up to 8wt%
Pellet Density	97 % T.D.	95.5%T.D.
Materials		
Cladding	ZIRLO™	←

Design Requirements and Guidance

- US-APWR design meets the following requirements and guidance
  - 10 CFR Part 50
    - ✓ GDC 10: Reactor design
    - ✓ GDC 27: Combined reactivity control systems capability
    - ✓ GDC 35: Emergency core cooling
  - 10 CFR Part 50.46
  - Regulatory Guide 1.206 (1.70) Section 4.2
  - Standard Review Plan Section 4.2 (NUREG-0800)
    - ✓ Fuel System Damage
    - ✓ Fuel Rod Failure
    - ✓ Fuel Coolability

- For Fuel System Damage:
  Analyses to determine if Fuel Damage Phenomena occur
  - Cladding Stress
  - Cladding Strain
  - Stress and Loading Limit for other than Cladding
  - Cladding Fatigue
  - Fretting Wear
  - Cladding Oxidation and Hydriding (and Crud Buildup)
  - Dimensional Changes
  - Rod Internal Pressure
  - Assembly Liftoff

Design Criteria, Methodology and Evaluation

- For Fuel Failure:
  Potential Fuel Rod Failure Modes
  - Hydriding
  - Cladding Collapse
  - Overheating of Cladding
  - Overheating of Fuel Pellets (Melting)
  - Excessive Fuel Enthalpy
  - Pellet/Cladding Interaction (PCI)
  - Bursting
  - Fuel Rod Mechanical Fracturing
- For Fuel Coolability:
  Design Considerations Affecting Fuel Coolability
  - Cladding Embrittlement
  - Violent Expulsion of Fuel
  - Generalized Cladding Melting
  - Fuel Rod Ballooning
  - Structural Deformation from External Forces

Relation between DCD and Fuel Topical/Technical Reports

US-APWR

DCD \ Fuel system	NO, AOOs, Shipping &Handling			Seismic+LOCA
	Fuel Rod	FA Structural Parts ICCC (RCC, BA, PS,SS)		Rod, FA & ICCC
4.2.1 Design Basis Criteria	MUAP-07008		MUAP-07016	MUAP-07008
4.2.2 Description and Design Drawings	MUAP-07018			
4.2.3 Evaluation				
Methodology	MUAP-07008		MUAP-07016	MUAP-07008
Calculational Code	FINE MUAP-07008	――――――	――――――	FINDS MUAP-07034
Evaluation Description and Results	MUAP-07016			MUAP-08007
	――――――	MUAP-11017	――――――	

Topical Reports	MUAP-07008(R2)	Mitsubishi Fuel Design Criteria and Methodology	2010/7
	MUAP-07034(R3)	FINDS: Mitsubishi Fuel Assemblies Seismic Analysis Code	2010/7
Technical Reports	MUAP-07016(R3)	US-APWR Fuel System Design Evaluation	2010/8
	MUAP-07018(R0)	US-APWR Fuel System Design Parameters List	2007/12
	MUAP-08007(R2)	Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads	2010/12
	MUAP-11017(R0)	Hydraulic Test of the Full Scale US-APWR Fuel Assembly	2011/5

Major RAIs for Section 4.2

➢ Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4.2-1	N/A	N/A	Completion of MUAP-07034	(1) Safety Evaluation : 12/15/12 (2) ACRS Sub. Meeting : 01/15/13 (3) ACRS Full. Meeting : TBD
4.2-2	N/A	N/A	Staff's evaluation of MUAP-08007-P, "Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads."	MUAP-08007 will be revised in July 2013 due to revision of input wave for FA seismic response analysis. The wave change results from revision of the seismic analysis model for US-APWR plant.

Major RAIs for Section 4.2

➤ Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4.2-3	929-6380	04.02-55	Using the 1000 hour fretting data gathered as part of the full-scale hydraulic testing of the US-APWR fuel assembly (MUAP-11017), provide a conservative extrapolation of the cladding wear for a conservative end of fuel assembly lifetime. Provide the basis for the extrapolation method used and explain how the extrapolated results compare to the acceptance criteria.	➤ MHI provided additional details in the response submitted on July 10, 2012. ➤ No impact on the DCD Closed-Resolved

Major RAIs for Section 4.2

➢ Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4.2-4	953-6437	04.02-64	The staff requested the basis for the applicant's conclusion that acceptable flow induced vibration of the fuel rod and fuel assembly will occur in the US-APWR fuel assembly over the entire range of expected operating conditions, including the acceptance criteria and source reference.	➢ MHI provided additional details in the response submitted on September 27, 2012. ➢ No impact on the DCD (Contents) MHI does not set a uniform and quantitative criterion of the vibration amplitude because the amplitude can not be discussed separately from the fuel cladding fretting wear. Since measured vibration amplitude in the hydraulic test agreed with the calculated value, it was regarded that the flow induced vibration amplitude of US-APWR fuel assembly was acceptable. In terms of fatigue failure of the grid spacer spring, since the limiting amplitude for fatigue failure of the grid spacer spring is 2 orders of magnitude larger than the amplitude at the grid spacer spring, which is provided by the hydraulic test, it is judged that there is no possibility of fatigue failure.

Major RAIs for Section 4.2

➢ Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4.2-5	948-6383	04.02-60	For BOL and EOL conditions, provide the grid spacer's elastic limit values used in the FINDS impact model and the 95th percentile buckling load values used to determine the number of buckled grids.	➢ MHI provided additional details in the response submitted on August 10, 2012. ➢ No impact on the DCD (Contents) The elastic limit value used in the FINDS impact model and in determining the number of buckled grids is obtained from the 95-percentile critical buckling force.

Major RAIs for Section 4.2

- Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4.2-6	N/A	N/A	Staff's review of MUAP-09002-P, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping."	The revised analysis result will be submitted in MUAP-08007(R3).
4.2-7	948-6383	04.02-62	The staff requested a revised structural analysis in MUAP-08007 which utilized the appropriate soil profiles.	The revised MUAP-08007 will be submitted in July 2013.

Major RAIs for Section 4.2

- Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4.2-8	929-6380	04.02-54	MUAP-07016-P provides data which supports the conclusion that assembly liftoff does not occur during normal operation and AOOs, but does not describe the limiting AOO or if the cold startup condition bounds the limiting AOO. Provide a description of the AOOs evaluated, the limiting AOO and details of how the limiting AOO was determined.	➢ MHI provided additional details in the response submitted on July 10, 2012. ➢ No impact on the DCD (Contents) The limiting condition is the Mechanical Design Flow (MDF). Furthermore, the cold start-up condition in the assembly lift off evaluation is conservative from the view point of hydraulic lift force.

Major RAIs for Section 4.2

➢ Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4.2-9	929-6380	04.02-56	Provide an explanation of how MUAP-011017, "Hydraulic Test of the Full Scale US-APWR Fuel Assembly" demonstrates that no additional holddown spring plastic deformation occurs under pump overspeed conditions.	➢ MHI provided additional details in the response submitted on July 10, 2012. ➢ No impact on the DCD (Contents) Plastic deformation of the holddown spring will not occur during lift off during a hot pump-over-speed condition based on the force balance calculation using load-deflection curve of the holddown spring obtained with classical beam theory, a FEM analysis and a mechanical test.

Major RAIs for Section 4.2

➢ Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4.2-10	928-6378	04.02-52	MHI's hydrogen absorption limit for a control rod guide thimble is 800 ppm. Justify why the hydrogen limit of greater than 400 ppm is acceptable.	➢ MHI provided additional details in the response submitted on June 19, 2012. ➢ No impact on the DCD Closed-Resolved

Major RAIs for Section 4.2

➤ Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4.2-11	929-6380	04.02-53	MHI makes an assumption that the burnable absorber (BA) released gas reaches its maximum value within one cycle of operation. From the information provided, the staff could not reach that conclusion as removal burnable absorbers are not prohibited from insertion into multiple duty cycles. Provided the limiting BA internal pressure and whether internal or external pressure is limiting from a cladding strain perspective.	➤ MHI provided additional details in the response submitted on July 10, 2012. ➤ No impact on the DCD Closed-Resolved

Major RAIs for Section 4.2

- Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4-2-12	948-6383	04.02-61	Staff requested detailed calculations of the acceptance limit stresses including the assumed material condition and Sy, Su, and Sm at the operating temperature for control rod cladding.	➢ MHI provided additional details in the response submitted on August 10, 2012. ➢ No impact on the DCD

Major RAIs for Section 4.2

➤ Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4.2-13	929-6380	04.02-57	Evaluate the impact of up to 0.1 second delay in control rod insertion on the scram worth versus position or time curve used in the Ch.15 analyses.	➤ MHI provided additional details in the response submitted on July 10, 2012. ➤ No impact on the DCD (Contents) The margin for the insertion time used in the safety analysis is larger than 1 second. The control rod insertion time delay does not impact the safety analysis.

Major RAIs for Section 4.2

- Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
4.2-14	953-6437	04.02-65	The staff requests analysis that ensures coolability during worst case LOCA loading.	➢ Response is under preparation.

Section 4.3 – Nuclear Design

Section 4.3 – Nuclear Design

- Design bases of the US-APWR complies with SRP 4.3:
  - Power distributions are maintained within the design limits throughout the cycle
  - Reactivity coefficients are maintained negative during power operation including Hot Zero Power (HZP)
  - Control systems are capable of providing enough shutdown margin and of controlling power distribution oscillations

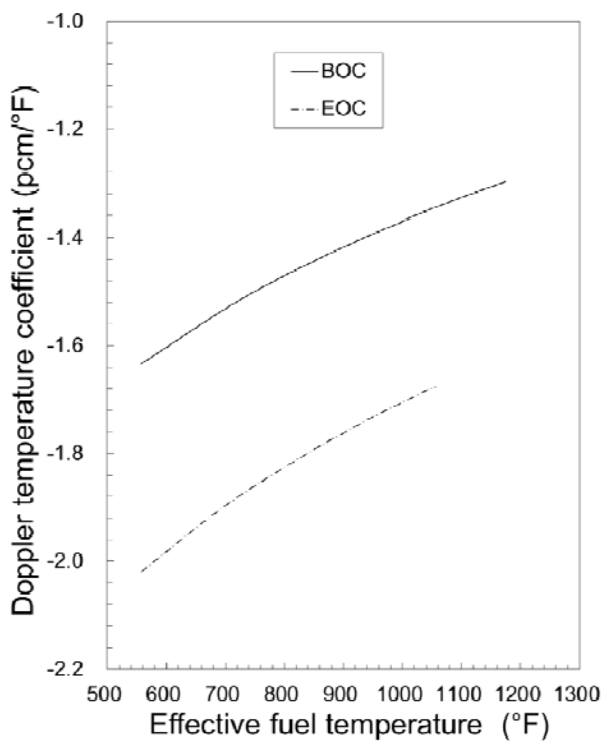
➢ Core Power Distributions

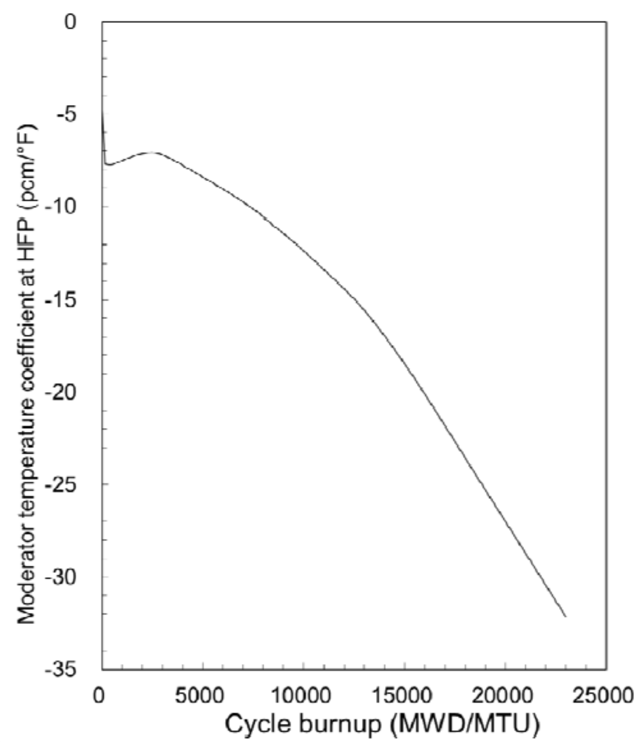
- The low power density allows flexible fuel management and operations
  - ✓ Up to 24 months operation (23,000 MWD/MTU) is available with two-batch reload under the constraints of U-235 enrichment less than 5 wt%

- The power distributions during Normal Operation are maintained within the design limits throughout the cycle:
  - ✓ Nuclear enthalpy rise hot channel factor $F_{\Delta H}^{N}$ is less than 1.73
  - ✓ Nuclear heat flux hot channel factor F_Q is less than 2.6
  - ✓ Maximum fuel rod burnup is less than 62,000 MWD/MTU

- The power distribution conditions during Normal Operation are used as the initiation of safety analyses discussed in Ch.15

➢ Reactivity Coefficients
- Fuel Doppler Temperature Coefficient
  - ✓ Negative throughout the cycle
- Moderator Temperature Coefficient
  - ✓ The coefficient is dependent on the soluble boron concentration
  - ✓ Burnable absorbers achieve a negative coefficient at BOC





Section 4.3 – Nuclear Design

- Reactivity Control System
  - Consists of 69 RCCAs
  - 24 absorber rods per assembly
  - Provides enough shutdown margin throughout the cycle (greater than 1.60 %Δρ)
  - Rod insertion limit is a function of core power

	J	H	G	F	E	D	C	B	A
9	A		D		SA		D		
10								SB	
11	D		SC		B		C		
12								SD	
13	SA		B		C		A		
14						D			
15	D		C		A		B		
16		SB		SD					
17									

		Number of RCCAs
A	Control group, Bank A	9
B	Control group, Bank B	12
C	Control group, Bank C	12
D	Control group, Bank D	12
SA	Shutdown group, Bank A	4
SB	Shutdown group, Bank B	8
SC	Shutdown group, Bank C	4
SD	Shutdown group, Bank D	8
	Total	69

Section 4.3 – Nuclear Design

- Stability
  - Total core power perturbation
    - ✓ The core is inherently stable to total power because of negative power reactivity coefficient throughout the cycle
  - Power distribution perturbation
    - ✓ Xenon-induced power distribution oscillations may occur; however they are readily detected by ex-core detectors
    - ✓ Horizontal oscillation
      - – The core is stable throughout the entire cycle
    - ✓ Axial oscillation
      - – The core is stable at BOC
      - – Although the oscillation becomes divergent with cycle burnup, the control rod system easily controls the xenon transient
      - – The stability characteristic is NOT unique to US-APWR and is similar to 12 ft PWRs

Open Items

- There are no Open Items for Section 4.3.

Section 4.4 – Thermal Hydraulic Design

Design bases are assured as follows:

- DNB does not occur at 95/95 basis for normal operation and AOOs.
  - DNBR evaluation is performed by VIPRE-01M code and WRB-2 correlation (MUAP-07009: under NRC review)
  - DNBR design limits are provided based on the approved revised thermal design procedure (RTDP), including the uncertainties of core conditions and CHF prediction by WRB-2. The limits are to be confirmed with plant specific instrumentation uncertainties.
  - Safety Analysis Limit (SAL) of DNBR is determined after accounting for DNBR penalties due to rod bow and transition core geometry, and reserving more core operational flexibilities.
- Fuel melting does not occur at 95/95 basis for normal operation and AOOs.
  - Fuel temperature analysis is performed with FINE code (MUAP-07008)
- Core flow rate is determined to ensure an adequate core cooling

Section 4.4 – Thermal Hydraulic Design

- Uncertainties associated with DNBR, fuel and cladding temperatures, and core hydraulics are provided to obtain conservative evaluations.

- Testing and verification are planned:
  - RCS flow measurement (Chapter 14)
  - Power Distribution Measurement (Chapter 14)
  - Component and Fuel Inspections (Section 4.2)

- Related safety limits, safety system setting, and limiting conditions for operation are provided in the Technical Specifications: Sections 2.0, 3.4, 3.5 and 4.2.

US-APWR

Thermal-Hydraulic Comparison between US-APWR and Other Designs

- Comparable thermal margins are reserved to the typical plant in US.

Design Parameters	US-APWR	Typical 12-ft 4loop	Typical 14-ft 4loop
Core thermal output (MWt)	4,451	3,565	3,853
System pressure (psia)	2,250	2,250	2,250
Thermal design flow (10^6lbm/hr)	168.2	139.4	145.0
Core average coolant mass velocity (10^6lbm/hr-ft^2)	2.25	2.41	2.59
R/V average coolant temperature (F)	583.8	588.4	582.3-593.8
Core average linear heat rate (kW/ft)	4.65	5.69	5.20
Minimum DNBR at nominal condition (Thm)	1.98	2.33	2.11
Minimum DNBR during AOOs (Thm)	>1.33	>1.23	>1.24
Maximum peak linear heat rate during AOOs (kW/ft)	<21.9	<22.4	<22.0
Maximum fuel centerline temperature during AOOs (F)	<4,620	<4,700	<4,700
Core pressure drop (psi)	32.1	25.8	39.78

Major RAIs for Section 4.4

- Open Items

Open Item No.	RAI No.	Question	RAI Topic / NRC Concern	RAI Response / DCD Impact
OI 4.4-1	952-6333	04.04-42	Appropriateness of δ_{bow} calculation	(1) RAI response was submitted on 9/7, 2012 (2) No impact on DCD
OI 4.4-2	-	-	Completion of MUAP-07009 “Thermal Design Methodology” review	(1) Safety Evaluation : 12/15/12 (2) ACRS Sub. Meeting : 01/15/13 (3) ACRS Full. Meeting : TBD
OI 4.4-3	845-6116		Section 4.4.5 is pending acceptable resolution of DCD Section 14.2 RAIs.	Closed-Resolved

Section 4.5 – Reactor Materials

Section 4.5 – Reactor Materials

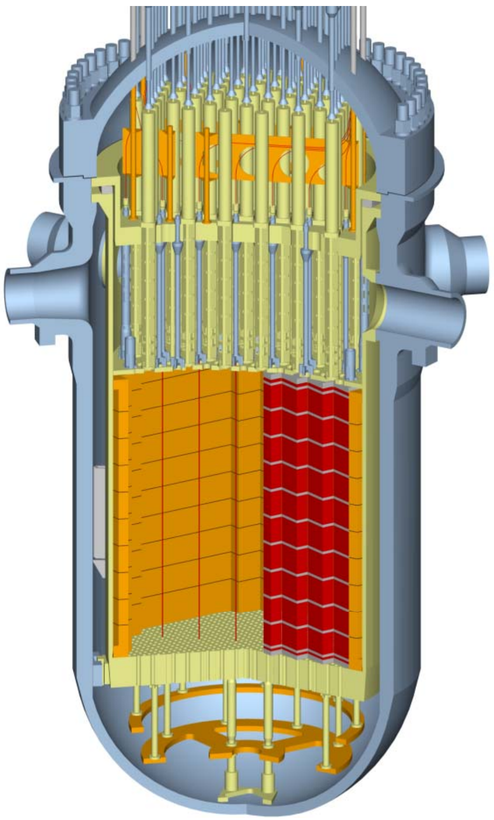
- All materials used in the US-APWR have been used successfully in both the United Sates and Japanese PWRs.
  - 4.5.1 Control Rod Drive System (CRDS) Structure Material
  - 4.5.2 Reactor Internals and Core Support Structure Material

Section 4.5 – Reactor Materials

- 4.5.1 Control Rod Drive System Structure Material
  - All pressure boundary materials comply with the requirements of ASME code Section III
  - The pressure housing material is type 316 or type 316LN austenitic stainless steel and the welding materials used for joining the austenitic stainless steels meet the requirements of the welding material specification SFA-5.9 (ER316L and EC316L)
  - Non-pressure boundary materials in contact with the reactor coolant water are austenitic stainless steel, martensitic stainless steel, nickel based alloys, and cobalt based alloys

Section 4.5 – Reactor Materials

- 4.5.2 Reactor Internals and Core Support Materials
  - Corrosive-resistant materials such as stainless for reactor internals
  - Strain hardened austenitic stainless steel for threaded structural fasteners and bolts
  - Nickel Alloy such as Alloy 690 is selected for its high and proven resistance against stress corrosion cracking
  - Comply with the ASME, Section II supplemented by ASME Code Cases, as approved in RG 1.84
  - Confirmed long term reliability for irradiation based on EPRI MRP175



- Selected materials have been widely used in many operating plants.

- Open Items
  - There are no Open Items for Section 4.5.

Section 4.6: Functional Design of Reactivity Control Systems

Section 4.6: Functional Design of Reactivity Control Systems

- The reactivity control systems for the US-APWR include:
  - The mechanical reactivity control of the control rods
  - The chemical reactivity control of the emergency core cooling system (ECCS).
- Control Rod Drive System (CRDS)
  - Control Rod Drive Mechanism (CRDM) (Subsection 3.9.4)
  - The CRDS is part of the reactor trip system (Section 7.2)
  - Cooling System for the CRDS (Subsection 9.4.6)
- Emergency Core Cooling System (ECCS)
  - Safety Injection System (SIS) of the ECCS (Section 6.3)
- No credit for the reactivity control capabilities of the chemical and volume control system (CVCS) is taken for anticipated operational occurrences and postulated accidents described in Chapter 15.

Major RAIs for Section 4.6

- Open Items
  - There are no Open Items for Section 4.6.

Acronyms

AO	:Axial Offset
AOO	:Anticipated Operation Occurrence
BOC	:Beginning of Cycle
DCD	:Design Control Document
DNB	:Departure from Nucleate Boiling
DNBR	:Departure from Nuclear Boiling Ratio
EOC	:End of Cycle
FQ	:Heat Flux Hot Channel Factor
FΔHN	:Nuclear Enthalpy Rise Hot Channel Factor
HFP	:Hot Full Power
HZP	:Hot Zero Power
NO	:Normal Operation

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Protecting People and the Environment

Presentation to the ACRS Subcommittee

US-APWR Design Certification Application Review

Safety Evaluation with Open Items

Chapter 4: Reactor

October 18, 2012

Staff Review Team

- Lead Project Manager – Jeff Ciocco
- Project Manager – Ruth Reyes
- Supporting Technical Staff:
  - Jeffrey Schmidt
  - Fred Forsaty
  - Jim Gilmer
  - Robert Davis
  - Joel Jenkins
  - John Budzynski

Major Review Areas

- Review of fuel system thermal, mechanical, and materials behavior for normal operation, transient and accident conditions
- The fuel system consists of
  - Fuel pellet
  - Fuel rod/cladding
  - Fuel structure
  - Control rods
  - Other incore components

Chapter 4.2 Supporting Documents

- Topical Reports
  - MUAP-07008, “Mitsubishi Fuel Design Criteria and Methodology”, fuel rod performance code FINE
  - MUAP-07034, “FINDS: Mitsubishi PWR Fuel Assemblies Seismic Analysis Code”, fuel structure behavior for seismic accelerations
- Technical Reports
  - MUAP-07016, “US-APWR Fuel System Design Evaluation”, evaluation of fuel rod, assembly structure and other incore components for non-seismic events
  - MUAP-08007, ″Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads”, fuel assembly structure/ components for seismic events

Description of Open Items

- **Open Item 4.2-1, MUAP-07034, “US-APWR Fuel System Structural Response to Seismic and LOCA Loads“**
  - Staff Safety Evaluation in concurrence
- **Open Item 4.2-2, MUAP-08007, “Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA Loads"**
  - Discussed in later slides
- **Open Item 4.2-3, RAI 929-6380, Question 4.2-55. Grid to rod fretting wear**
  - Closed-resolved
- **Open Item 4.2-4, RAI 953-6437, Question 4.2-64. Acceptability of fuel and assembly flow induced vibration**
  - Staff reviewing MHI response

Description of Open Items

- **Open Item 4.2-5, RAI 948-6383, Question 4.2-60. BOL and EOL elastic limit loads used in the FINDS impact model, and buckling load values used to count the number of buckled grids**
  - Staff reviewing RAI response
- **Open Item 4.2-6, The staffs review of MUAP-09002-P, "Summary of Seismic and Accident Load Conditions for Primary Components and Piping"**
  - Tracking item which will be closed when the final safety evaluation for Topical Report MUAP-09002-P is issued

Description of Open Items

- **Open Item 4.2-7, RAI 948-6383, Question 4.2-62. The staff requested a revised structural analysis in MUAP-08007-P which utilized the appropriate soil profiles**
- **Open Item 4.2-8, RAI 929-6380, Question 4.2-54. AOO Holddown Spring Evaluation**
  - Staff reviewing RAI response
- **Open Item 4.2-9, RAI 929-6380, Question 4.2-56. Plastic Holddown Spring Deformation on RCP Overspeed**
  - Staff reviewing RAI response

Description of Open Items

- **Open Item 4.2-10, RAI 928-6378, Question 4.2-52. Guide Tube Hydrogen Limit**
  - Closed-resolved
- **Open Item 4.2-11, RAI 929-6380, Question 4.2-53. Burnable Absorber B^{10} Release and Internal Pressure**
  - Closed-resolved
- **Open Item 4.2-12, RAI 929-6380, Question 4.2-61. Control Rod Cladding Stresses and assumed Material Properties**
  - Staff reviewing RAI response

Description of Open Items

- **Open Item 4.2-13, RAI 928-6380, Question 4.2-57. Evaluate Control Rod Drop Time Delay of 0.1 secs on Chapter 15 Accidents**
  - Staff reviewing RAI response
- **Open Item 4.2-14, RAI 953-6437. Evaluation of Fuel Coolability under Worst LOCA-only Fuel Induced Loadings**
  - Waiting for RAI response

Open Item 4.2-2 , Fuel System Structural Response to Seismic and LOCA Loads

- US-APWR predicts plastic fuel grid deformation throughout the core under SSE only loads
  - First licensing application with SSE only deformation
- GDC 2, Design bases for protection against natural phenomena

  "The design basis for these SSCs shall … reflect appropriate combination of the effects of normal and accident conditions with the effects of the natural phenomena"
- Assembly components evaluated against SSE, LOCA, and SSE+LOCA loads
- SRP 4.2 acknowledges the possibility of plastic grid deformation
  - Control rod insertion
  - 50.46 LOCA requirements
  - Unclear guidance to evaluate effects of grid plastic deformation

Control Rod Insertion

- SRP 4.2 states,
  - Consequences of grid deformation are small and gross deformation of many assemblies would be needed to interfere with control rod insertion
  - Ability to insert control rods must be demonstrated under combined loads
- Currently licensed plants do not predict grid deformation in rodded core locations
- MHI performed full scale control rod insertability tests with deformed grids
  - Impact on control rod delay time was sufficiently small
  - NRC staff reviewed rod drop tests and found them acceptable

Accident Analysis

- SRP 4.2 provides unclear guidance to evaluate GDC 2 statement “appropriate combination of the effects of normal and accident conditions”
- Current plants may deform under SSE+LOCA forces in non-limiting core locations
- Deformation under SSE only forces, in limiting core locations, introduces DNBR and LOCA evaluation concerns

Review Status

- MHI currently revising MUAP-08007, "Evaluation Results of US-APWR Fuel System Structural Response to Seismic and LOCA," scheduled completion is July 2013
  - Revised soil profiles and basemat design might alter predicted SSE applied loads
- Core wide plastic grid deformation introduces significant technical, regulatory, and scheduling uncertainty.
- Public meeting scheduled for November 2012 on path forward

Conditions and Limitations

Surveillance program to provide oxide measurements on interior and peripheral rods on the maximum power and burnup assemblies from the first and second cycle of US-APWR (assuming a 24 month fuel cycle) to confirm the FINE corrosion model

US-APWR Chapter Section 4.4 Thermal and Hydraulic Design

- VIPRE-01M code is used for analyses

  - MHI version includes enhanced model selection and fuel-specific critical heat flux correlation

  - Code is described in Topical Report MUAP-07009-P

  - Staff SER for VIPRE-01M completed as separate document referenced by DCD FSER

  - Westinghouse WRB-1 and WRB-2 correlations were shown by tests to be applicable to APWR fuel (with mixing vane grids)

- DNB tests were performed at KATHY facility (Karlstein, Germany) and were witnessed by staff. Evaluation is documented in FSER for MUAP-07009-P

US-APWR Chapter Section 4.4 Thermal and Hydraulic Design

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- Staff evaluation of test report (MUAP-11010-P, “DNB Tests for US-APWR Fuel”) is provided in the Final Safety Evaluation Report for the VIPRE-01M code
- Uncertainties in power, inlet temperature, pressure, flow, peaking factors, code, and DNB correlation are evaluated by approved WEC Revised Thermal Design Procedure (RTDP)
- Core flow and pressure drop testing will be performed during initial startup (described in Section 14.2)
- LOCA-generated debris blockage is addressed in FSER Section 6.3

Description of Open Items

- 4.4-1: < 1% rod bow penalty – RAI response under evaluation by staff
- 4.4-2: Final staff acceptance of MUAP-07009-P is needed to complete the Section 4.4 review (resolved/closed)
- 4.4-3: Section 4.4.5 is pending acceptable resolution of Section 14.2 RAIs (resolved/closed)

Backup Slide

Technical Topics of Interest

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Fluence Calculation Methodology

- The applicant employs a calculational methodology for determining the pressure vessel >1-MeV neutron fluence that generally conforms to the guidance of the NRC RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."
- Staff's review concluded that the methodology is acceptable for determining the US-APWR pressure vessel fluence subject to the following:
  - When the actual as-built and operating fluence data becomes available bias and uncertainties shall be evaluated, and updated if non-conservative when compared to those in MUAP-09018, latest revision
  - A Chapter 5 COL item addresses the commitment to evaluate fluence bias and uncertainties documented in MUAP-09018 as US-APWR data becomes available